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COMPONENT FAILURES at PRESSURIZED WATER REACTORS

FINAL REPORT

Volume I

Prepared by
COMBUSTION ENGINEERING, INC.
Nuclear Power Systems
Windsor, Connecticut 06095

**CE POWER
SYSTEMS**

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Printed in the United States of America

Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

Price: Printed Copy \$16.00; Microfiche \$3.00

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PRESSURIZED WATER REACTORS

Final Report

Volume I

M. F. Reisinger
Program Manager

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Prepared for

U. S. Department of Energy
Light Water Reactor Safety Technology Management Center
Sandia National Laboratories
Albuquerque, New Mexico 87185

Sponsored by

U. S. Department of Energy
Division of Nuclear Power Development
Washington, D. C. 20545

Work performed under Sandia Contract No. 13-6442

Printed: October 1980

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ABSTRACT

Combustion Engineering, Inc. conducted a study of system and component failures at all U.S. commercial PWRs which caused reactor scrams and forced outages. The objectives were to identify those systems having major impact on safety and availability (i.e. to identify those systems and components whose failures have historically caused the greatest number of challenges to the reactor protective systems and which have resulted in greatest loss of electric generation time). These problems were identified for engineering solutions and recommendations made for areas and programs where research and development should be concentrated.

The program was conducted in three major phases: Data Analysis, Engineering Evaluation, Cost Benefit Analysis.

The Data Analysis phase analyzed the historical data available to identify and rank the major systems and components leading to challenges to the reactor protective system and lost availability (generation time). The Engineering Evaluation phase 1) reviewed those problems identified in the data analysis to verify or determine "root causes", 2) evaluated the problems for solutions which could reduce or eliminate the problems and 3) recommended R&D programs, where needed, to reduce or eliminate the problems. The Cost Benefit Analysis phase evaluated the savings which would result from the reduction of lost generation time caused by challenges to the protective systems.

This report has been published in two volumes. Volume II consists of Appendices 1 through 4.

ACKNOWLEDGEMENT

Combustion Engineering, Inc. would like to acknowledge the assistance and cooperation of all who directly or indirectly made this study possible. The United States Department of Energy and its program office at Sandia Laboratories are commended for taking a leading role in facing today's energy problems by sponsoring this program.

The Electric Power Research Institute is commended for sponsoring the Limiting Factors Analysis and thanked for the permission to use the results of these studies prior to publication.

Numerous persons at C-E contributed significantly to this effort. They are D. J. Finnicum, Principal Investigator for Data Analysis; T. E. Fitzsimmons, Principal Investigator for Fluid and Mechanical Components; J. K. McNally and M. F. Strollo, Principal Investigators for Cost Benefit Analysis; F. J. Safryn, Principal Investigator for Instrumentation Control and Electrical components; N. K. Woodward, Principal Investigator for Systems Integration and E. W. Hewitt, Development Programs Manager. To these and many other unnamed persons, I wish to express my appreciation.

M. F. Reisinger
Technical Program Manager

EXECUTIVE SUMMARY

Background

Sandia Laboratories, acting for the Department of Energy, contracted two programs to study causes of shutdowns at light water reactors located within the United States. The first program was on Boiling Water Reactors and the second was for Pressurized Water Reactors (PWR). The studies were part of the Department of Energy programs in the area of Light Water Reactor Safety. The subject of this report is the program to study causes of shutdowns at Pressurized Water Reactors.

The objectives of this program were to determine the major causes of shutdowns at PWRs and evaluate those problems which occurred frequently or had a significant impact on plant safety or availability for ways by which the problem can be reduced or eliminated. An additional goal included qualitative evaluation of the cost benefit incentive for implementing the potential solutions.

The base data for this study was the computerized Combustion Engineering, Inc. (C-E) PWR Operating History Data Base. It contains a listing of all PWR shutdowns and forced outages from the inception of commercial PWR operation to the end of 1978. A combined total of 192.4 plant years of commercial PWR experience from forty plants is represented in this data base. The data base is commonly referred to as the Shutdown Data Base and was drawn from publicly available information sources.

Scope and Methodology

Shutdowns at commercial nuclear power plants occur for two major reasons, refueling and component failures. This study concentrated on the latter. Causes of manual and automatic scrams and forced outages were determined and categorized. Next, the problems were ranked according to importance (frequency, impact). Then those problems which occurred frequently or had a significant impact on safety or availability were evaluated for ways by which the problem could be reduced or eliminated. Additionally, a qualitative assessment of cost benefit of solutions was derived.

For this study, the impact on safety is defined as the frequency of challenges to the Reactor Protection System (RPS) and post-trip heat removal systems. The impact on availability was the measure of plant unavailability due to the component failures. Shutdowns as well as scrams were considered as the operator may shutdown in anticipation of a scram condition.

The study was conducted in basically three phases; Data Analysis, Engineering Evaluation and Cost Benefit Analysis. The Data Analysis objectives were to identify and rank the causes of plant shutdowns and was conducted by individuals proficient in data analysis techniques. Engineering Evaluation scope included verification of the causes of shutdowns and evaluation for potential solutions. These evaluations were conducted by engineers with expertise in the area of concern within the specialization of three major engineering divisions within C-E: Fluid and Mechanical Components; Instrumentation, Control and Electrical Components; and Systems Integration. The analysis of Cost Benefit was assigned to a group for special projects. The objective was to assess the present worth of cost benefits resulting from increased availability over the remaining life of the plant. Knowing the economic present worth of a potential solution aids in evaluating the alternative safety improvements for implementation.

Conclusions

The average number of PWR scrams (challenges to the Reactor Protection Systems) per plant year is 9.3 scrams per plant year. Of this, 7.2 scrams per plant year are attributed to systems and component malfunction or failure. The average annual unavailability for PWRs is 26.3 percent of which half is due to systems and component failures. The other half is refueling and other miscellaneous causes.

The systems which caused the greatest number of shutdowns at PWRs were determined to be:

- o Feedwater and Condensate System
- o Turbine Generator System

- o On-Site Power System
- o Reactor Protection System
- o Control Rod System
- o Main Steam System
- o Reactor Coolant System

With the exception of the Reactor Protection System, the above systems also had the greatest impact on plant availability. This indicates a strong relationship between safety and availability. Please refer to Figure 1 for the relative impacts of each system. The availability impact from the Reactor Vessel and Internals was attributed to a few singular events of large duration. Further evaluation was not conducted of these events.

Of the forty systems which were reported to have caused plant outages, these seven systems accounted for over eighty percent of the challenges to the RPS and plant unavailability. Emphasis on improvement should therefore be concentrated on these seven systems for both a tangible safety benefit (reduction in challenges to the RPS) and a tangible cost benefit (increased annual plant availability expected from a solution.)

The results of the engineering evaluation are as follows:

- o The Feedwater and Condensate System ranked highest in impacting plant safety and fourth in unavailability contribution. Challenges to the RPS from this system occurred almost twice as often as from any other system. In a limited operator opinion poll conducted, this system ranked number one in concern for plant safety and availability. Failure to control steam generator level, feedwater pump trip, control valve malfunction and problems, feedwater pump problems and feedwater heater leaks were the more recurrent categories, the majority of which resulted in automatic scrams. Automatic scrams from this system occurred on the average of 1.83 times per plant year. For forty PWR plants in the United States, this projects to an average of seventy-three reactor trips per year due to Feedwater and Condensate System failures. The recent Three Mile Island accident also started in this system.

Failure categories were determined to be numerous as were the underlying causes for the failures. Three major root causes of this system's poor performance were determined to be as follows.

1. The Non-Optimization of Feed Train Configuration

From a preliminary review, there appears to be some relationship between the number and type of events experienced and the configuration of the plant's feedwater and condensate system. A minimum of uniformity was found in component redundancy, capacity, motive power and speed control systems. Control and equipment protective setpoints appeared related to the protection of individual components, with very little consideration allowed for the effect these setpoints have on overall plant operation. The backfitting of systems and components (e.g. condensate filters) to the condensate and feedwater system appears to degrade system performance and reliability. Additionally, the reliability of control systems and power supplies are not evaluated for emergency conditions unless the system is designated as required for the safe shutdown of the reactor (e.g. auxiliary feedwater system). Finally, the absence of redundancy encourages power reduction and degraded plant performance when the maintenance and testing of components is performed during plant operation. Design integration of feed train components to systems requirement can be significantly improved.

2. The Non-Optimization of Low Power Feedwater Control

Current system designs emphasize efficiency at full power operation resulting in large components and capacities, control system sensitivity for large components and capacities, control system sensitivity for large components and fluid flowrates, and fixed protective system setpoints. This results in extremely difficult manual operations and numerous challenges to the reactor protective system during operation at low power.

3. Non-Optimum Hydraulic Resiliency

PWR feed and condensate system design appears to lack resiliency to small flow and pressure instabilities. Anticipatory pump trips assume the worst condition has occurred. For example, the feedwater pump low suction trip anticipates a total loss of makeup water but may actuate during a temporary low suction pressure condition. Large pressure transients can be common in the "solid" liquid phase regions of the system. This results in the unnecessary loss of feedwater flow usually followed by a reactor trip on steam generator low water level.

The basic recommendation resulting from the evaluation of component failures is that PWR feedwater and condensate be the subject of more detailed system level optimization studies focused on improving existing and future system designs, availability and reliability, and therefore improving overall plant safety. The feedwater and condensate system is important in maintaining the normal heat sink for the reactor. More emphasis should be on the maintenance of this system's function. Integration with the NSSS as well as equipment integration with the system should be improved.

- o The Turbine Generator System ranked second with respect to scrams and thus impact upon plant safety, and first in contributing to plant unavailability. The more frequent causes of plant scrams were attributed to problems with the Electro-Hydraulic Control (EHC) System and turbine valves. Major unavailability contributors were turbine blade failure, bearing failures and other corrective maintenance on the turbine and generator. The greatest potential for reducing challenges to the RPS from this system is in improved EHC system and valve design for more redundancy or reliability. A reduction in turbine valve testing should also be evaluated. Improved turbine blade performance is believed to be obtainable through improved Moisture Separator Reheater design and control

for better performance and turbine load equalization. Improved diagnostic monitoring would aid in improving performance and minimizing damage to the turbine. A detailed program is recommended for this system to evaluate system reliability.

- o The major categories of failures causing scrams in the On-Site Power System were the loss of vital or control instrument bus, the loss of a DC/AC inverter and the loss of the Reactor Coolant Pump RCP Bus. Loss of the instrument non-vital bus was the principal cause. It is recommended that control instrument buses be provided with uninterruptable DC/AC inverters and that two battery plants consider the four battery system. Further evaluation of using four separate RCP busses is also needed and recommended.
- o The Reactor Protection System was ranked fourth in causing reactor scrams. Major categories causing reactor trips were:

- Spurious trips during testing
- Spurious trips in one-out-of-three mode
- Noise (RMF/EMI)

Over seventy percent of the ninety-five occurrences had one or a combination of the above descriptions in the event descriptions. These scrams are considered unnecessary as they were not the result of component failures. The RPS systems are triply or quadruply redundant, usually in a two-out-of-three or a two-out-of-four logic requiring two channels to trip for a scram to occur. Further, the RPS components are of the highest quality and highest sensitivity of all channels in the plant. Considering the resulting plant transient and the sensitivity of the RPS channels, the requirement for one-out-of-three or even one-out-of-two logic may be unwise and should be reevaluated. Two-out-of-two and two-out-of-three logic intuitively has less probability of causing an inadvertent trip requiring the actuation of post-trip heat removal systems. Allowance for a channel bypass during testing would help to reduce this problem.

- o Major problems in the Control Rod System were rod drops, Control Rod Drive Mechanism (CRDM) motor failures and CRDM seal housing leaks. These problems appear to have been resolved. Impact from this system is expected to decrease.
- o Frequent Main Steam System related outages were due to Steam Generator (SG) Tube leaks, Main Steam Isolation Valve (MSIV) malfunction, and component support problems. Comprehensive programs are underway in resolving SG tube problems. The MSIV problems have been associated with the back-to-back MSIV design for which corrective action has been taken. Component supports should be improved by replacing springs and seals with improved materials of newer designs along with improved installation techniques.
- o Reactor Coolant System recurrent events were recorded due to Reactor Coolant Pump seal failures, Pressurizer spray control valve leakage and other miscellaneous valve leakage. A Commonwealth Edison study cited the need for improved operating, installation and maintenance procedures for prevention of seal failures. Improved zero leakage valves for pressurizer spray valve application are recommended for development as well as relocation of the spray valves to a better environmental area for easier access and maintenance. A general valve study is recommended to assess the need and potential for a good zero leakage valve for general purpose nuclear application.

The major objective of the cost benefit phase of this program was to assess the present worth of additional availability expected due to reduced challenges to the RPS from reduced component failure impact on the plant. Computer techniques were used to generate the graphs necessary to allow this benefit assessment. Factors included in the assessment were:

National Average Mix of Electrical Generation
Fuel Heat Rates
Escalated Fuel Costs

Escalated Replacement Power Costs
Levelized Annual Fixed Charge Rate
Weighted Cost of Capital

The resultant curves; Figures 4.1, 4.2 and 4.3 of the text; were generated for a 1000 MWe plant. The total present worth of the cost savings of one day of increased availability at the average plant and a forty year average remaining lifetime is shown to be 32 million dollars. The curves further show that seventy-five percent of the savings is due to capital costs and twenty-five percent to replacement power costs. Fuel costs have significantly changed since the escalated values were assigned. The cost savings can therefore be assumed to be greater than 32 million dollars. Other costs could not be evaluated or estimated in this program, however, they would only serve to increase the potential benefit above that identified.

An example of a significant benefit would be the incorporation of a time delay relay in the feedwater pump NPSH protective trip circuitry. The cost of implementation is not expected to exceed \$1000. However, the potential benefit of 0.064% (5.6 hours or .23 days/year), can be shown to be worth approximately 10 million dollars for a plant with an expected remaining life-time of 40 years. This represents a significant net cost benefit which would be larger for that fraction of plants which were affected most by these spurious pump trips. Extension of the computer techniques allowed uncertainty to be included in a cost benefit program for the computer evaluation of individual component failures and solutions. The computer model is described in the text.

Summary

For this study, safety impact was defined to be the frequency of challenges to the RPS and which required operation of post-trip safety grade heat removal systems. The process (control grade) systems or components were determined to cause the most challenges to the RPS.

The major conclusion from this study is as follows:

IMPROVED RELIABILITY AND AVAILABILITY OF PLANT PROCESS SYSTEMS AND COMPONENTS CAN RESULT IN IMPROVED SAFETY THROUGH A REDUCTION IN CHALLENGES TO THE REACTOR PROTECTION SYSTEMS AND POST-TRIP HEAT REMOVAL SYSTEMS. THE IMPROVED PERFORMANCE CAN ALSO AID IN THE MITIGATION OF EVENTS.

More emphasis should be placed on "elimination of the initiating event" in preventing nuclear plant incidents and improving industry performance. It is suggested that the emphasis may have been overly placed on "assume a worst case event and design a backup safety system to contain that event".

Process system improvements which reduce challenges to the RPS can be shown to provide a significant economic incentive for implementation. Reduced replacement power costs and capital expenditures are the major areas of cost savings.

To promote the idea that safety, availability and reliability are intimately related and should be emphasized in all phases of nuclear power, the triangle on the cover of this report is suggested as a reminder toward providing excellence in the nuclear industry.

MAJOR PWR SYSTEMS CHALLENGES
TO THE REACTOR PROTECTION SYSTEM

MAJOR PWR SYSTEM CONTRIBUTION
TO PLANT UNAVAILABILITY

FIGURE 1

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1.0 INTRODUCTION

1.1 BACKGROUND

Whenever a nuclear power plant shutdown occurs, regardless of the nature or severity of the shutdown, the general public becomes concerned. This concern is due to fears about safety and possible impact on utility bills. Because of this and because the availability of nuclear power plants can have a significant impact on the "energy problem" through a reduction in oil consumption, a strong incentive exists for improving nuclear plant safety and performance by reducing unscheduled shutdowns.

Sandia Laboratories, acting for the Department of Energy, initiated two programs to study causes of shutdowns at commercial Light Water Reactor plants in the United States. One study was for Boiling Water Reactors (BWRs) and the other for Pressurized Water Reactors (PWRs). Combustion Engineering, Inc. (C-E) was selected to conduct the PWR study.

Several types of shutdowns occur at nuclear plants, of which the two most prevalent causes are 1) refueling and 2) system and component failures. The first is planned; the second is unplanned. This study was concerned with the latter, and is titled "PWR Component Failure Study".

This study was conducted in a seven month period starting in April, 1979 and was completed in October, 1979.

1.2 STUDY OBJECTIVES

Commercial PWR power plant experience in the United States had accumulated to 192.4 plant years as of December 31, 1978. Much can be learned from that experience, both for improvements for operating plants and for the design of new plants.

The objectives of this study were 1) to identify the various system and component failures causing automatic and manual shutdowns and forced outages, 2) to rank the shutdown causes according to impact (frequency, duration);

and 3) to evaluate those problems which occur frequently, or have a unusually significant impact on safety or availability, for improvements which either reduce or eliminate the problem.

Nuclear plant safety has many definitions and few people agree on any measure of impact upon safety. For this study, impact on safety was defined as the frequency of challenges to the Reactor Protective Systems (RPS), because such challenges require the RPS and post-trip heat removal systems to perform their safety functions. Scrams were chosen as the subjective measure of safety impact because scrams require standby systems to function for plant safety. As a outage type identifier, the scram listing was a convenient "tool" which was available in the data base to sort out system failures that can possibly lead to safety problems, the resolution of which will improve safety by reducing the number of scrams, reducing challenges to the RPS and post-trip heat removal systems.

It is recognized that some safety concerns may not result in challenges to the RPS. Also, although they may result in challenges to the RPS, all component failures are not safety concerns nor should all component failures be treated equally from a safety viewpoint. However, those failures that lead to challenges to the RPS, which occur frequently, present a safety concern because the RPS or post-trip heat removal systems might fail to perform their safety functions, possibly resulting in a more severe event.

Except for failures in the RPS which cause scrams, scrams are the result of failures in process systems or components, which cause plant conditions to proceed toward and approach unsafe conditions. The dependence is on the RPS and post-trip systems to mitigate conditions, preventing a more severe event.

The motivation for this study is that improvements which reduce challenges to the RPS reduce the dependence on the RPS to assure safety. In addition to providing a safety benefit, an economic benefit is provided. This is achieved thru increased nuclear plant availability.

The impact on availability is measured in percent unavailability defined as the fraction of the 8760 hours in a year that the plant is unavailable for full power production.

1.3 IMPLEMENTATION

The scope of this project was divided into six phases with some overlap between phases.

Phase 1 - Data Update

A program of this magnitude could be conducted in a seven month period only because of the availability of a data base previously developed by C-E, which suited the project objectives. This data base is the "PWR C-E Operating History Data Base". This data base contains a listing of all reported shutdowns and power reductions at all U.S. commercial PWRs (not just plants with C-E supplied Nuclear Steam Supply Systems) from July 1962 thru December 1978. The last 6 months of data was added during this phase.

Phase 2 - Data Analysis

The purpose of the data analysis phase was to identify and rank the system and component failures leading to challenges to the protective system and causing plant unavailability. The data reduction was done by system and then by component within each system, in order to better delineate impact on the plant.

Phase 3 - Data Verification

The data verification phase was not limited to a fixed time span within the program. Verification occurred during all phases but mainly during the engineering evaluation phase. Additional information and detail were continuously sought throughout the program to further define the problems identified in the data analysis phase above.

Phase 4 - Engineering Evaluation

The scope of the engineering evaluation was to review the data and verify the problems, search for root causes, select the most important problems, and recommend solutions which would reduce or eliminate the problems. In addition, this phase included program recommendations in those areas where further investigation or R&D is warranted. This portion of the study was performed by three different C-E engineering groups, each experienced in the problems they evaluated. The three engineering divisions were:

- o Fluid and Mechanical Equipment
- o Instrumentation Control and Electrical Equipment
- o Systems Integration

The scopes of the first two areas are self explanatory. The systems integration group examined significant systems interaction problems.

Phase 5 - Cost Benefit

The goal of the cost benefit phase was two fold. The first was to evaluate the life cycle benefits obtainable from improved plant availability. Secondly a method was developed which would address the uncertainties involved in cost benefit analysis as applied to the engineering improvements for nuclear plants.

Phase 6 - Final Technical Report

In this phase, the results of each individual task was prepared by the investigation engineers and integrated into a final technical summary detailing the evaluations and results of each phase of the project.

1.4 REPORT ORGANIZATION

Section 2, Data Analysis, discusses the information sources and data bases used for this study and presents the results of the data base analyses.

Section 3, Engineering Evaluation discusses the results of the engineering team findings and data verification. Discussed here are the potential root causes of the problems and the potential solutions to these problems. It also contains recommendations for future investigation where warranted.

In Section 4, Cost-Benefit, financial considerations involved in implementing engineering solutions are discussed. Included is a methodology for handling the uncertainty which applies to most cost benefit discussions.

Section 5, Summary and Conclusions, reports this study's findings.

Section 6, Recommendations, reports on the team's recommendations for potential improvements and for future analysis and development work.

The Appendices present supporting information compiled in this study which is too detailed and did not lend itself to presentation in the main body of this report. Appendices are organized by appropriate section. Appendix 1 provides a glossary of acronyms and definitions. Appendices for Section 2 provide systems descriptions and reduced data which lead to the data analysis conclusions. Appendices for Section 3 include information supporting engineering findings and the Appendices for Section 4 include descriptions of computer codes and assumed values used for the cost benefit analysis.

Section 2.0

DATA ANALYSIS AND VERIFICATION

2.1 DATA ANALYSIS INTRODUCTION

The purpose of the data analysis phase of this program was to identify causes of PWR scrams and forced outages and rank those causes according to their impact on plant safety and availability. The primary data source used was the C-E PWR Operating History Data Base. This data base provided a comprehensive record of PWR shutdowns and power reductions at commercial PWRs located in the United States. Additional sources of information were used to supplement this data base.

The general procedure used in the data analysis was to:

1. identify impact by system
2. rank systems by impact
3. select systems with greater impacts
4. identify problems within selected systems
5. provide engineering divisions with resulting problem lists for the engineering evaluation phase.

Besides the impact based on the total accumulated PWR experience, information about how the accumulated experience affected problems was desired. For this reason, time frame analysis was accomplished on the system level on both a calendar year basis and a commercial year basis. Calendar year analysis indicates industry experience while commercial year analysis provides indicators of plant experience.

Section 2.2 discusses the information sources and data bases for this study. A description of the data sources and their usefulness for this program is included.

Section 2.3 describes the methodology used in this program to reduce the compiled data to meaningful results.

Section 2.4 provides the results of the data analysis phase of this study. Identification of problems based on total accumulated experience as well as calendar and commercial year results are provided.

Section 2.5 discusses the objectives and results of the data verification effort done as part of this data analysis phase.

Section 2.6 provides recommendations for data base improvements based on the limitations of the information used in this study and additional trends analysis of the problem level. Additional analysis should determine which problems still exist and which problems have increasing impact.

The appendices provide greater detail and substantiate the conclusions reached in this section. Appendices are referenced in each subsection and are provided for those who desire more information.

2.2 INFORMATION SOURCES AND DATA BASES

2.2.1 Information Sources

2.2.1.1 General

The information used for this study was limited primarily to information available in the public domain. This information was supplemented, when possible, with information obtained from three unpublished final reports for the EPRI Limiting Factors Analysis studies for PWRs, (references 1, 2 & 3), and with information obtained directly from operating PWRs. The primary public domain information sources are:

- a) The NRC's "Operating Units Status Reports" (The Grey Book)⁽⁴⁾
- b) Licensee Event Reports⁽⁵⁾
- c) Utility semiannual or annual operating reports
- d) The Nuclear Plant Reliability Data System (NPRDS) Failure Reports⁽⁶⁾

Figure 2.2-1 shows the general relationships between the public domain information sources and their relationship to the total body of information available at a nuclear power plant (this figure is not to scale). The total body of information available at a nuclear power plant encompasses all phases of plant operation. Information pertaining to systems and components is a part of this larger total. As shown in Figure 2.2-1, no

public domain information source covers the total body of available information or the total available body of information pertaining to systems and components. This is because each public domain information source was developed to fulfill specific needs and therefore each uses only a selected subsets of the total body of information. The following paragraphs briefly describe each of the information sources.

2.2.1.2 NRC Operating Units Status Reports

The NRC's "Operating Units Status Reports" are monthly reports that provide operating data for all commercial nuclear power plants in the United States. The primary objective of these reports is to provide information for use in evaluating the availability and electric power producing capability of nuclear power plants. The first issue of this report was published in June, 1974. Prior to this time, plant availability information had to be extracted from the utility semiannual or annual operating reports. The information that these reports contain includes:

1. The number of hours each plant was operating during the reporting month.
2. The total thermal and electrical power produced by each plant during the month.
3. Brief descriptions of all shutdowns and significant power reductions that occurred during the month.

These reports also contain a list of Licensee Event Reports (LERs) written during the month, and some information on NRC inspection findings. These reports are brief and consequently do not contain detailed information on the root causes of problems that caused the shutdowns, and very little information on other maintenance that took place during an outage.

2.2.1.3 Licensee Event Reports

Licensee Event Reports (LERs) are mandatory reports to the NRC covering violations of a Utility's Operating License or Technical Specifications. The primary objective of the LERs is to collect information on unscheduled

incidents or events that might be of significance from the standpoint of public safety or health. These reports are used to identify "abnormal occurrences", that is, "an unscheduled incident or event the commission determines is significant from the standpoint of public health or safety". The LER reporting requirements are delineated in Title 10 of the Code of Federal Regulations (10 CFR), and in the Utilities' Operating Licenses and Technical Specifications. Basically, an LER contains the following information:

1. Identification of when and where the event occurred.
2. A verbal description of the event and its consequences.
3. A verbal description of the event causes and corrective actions.
4. Identification of system and component that caused the event.
5. Impact on the plant in terms of changes to plant status (i.e. shutdowns), radiation releases, equipment damage and/or personnel injury.

Reference 2 contains a more detailed description of the information contained in a LER and the LER reporting procedures and instructions.

2.2.1.4 Utility Semiannual and Annual Reports

The required utility semiannual/annual reports were reports which described the operation of a particular plant for a specific time period. The requirement for a semiannual/annual report was discontinued after 1977. The primary objective of these reports was to provide an overview of all phases of plant operation. These reports contain the following types of information:

1. Plant operating time and power production data.
2. Brief descriptions of shutdowns and power reductions.
3. Summaries of LERs submitted during the period.
4. Brief descriptions of safety related maintenances.
5. Brief descriptions of plant design changes.
6. Description of inspection and test procedures and significant results.
7. The results of environmental monitoring and tests.
8. Personnel training and changes.

2.2.1.5 Nuclear Plant Reliability Data System

The Nuclear Plant Reliability Data System is a voluntary industry-wide data system whose primary objective is to collect reliability data on systems and components important to nuclear safety (safety Class 1, 2, and 1E Systems and components). The NPRDS System was started in June, 1974. The NPRDS primarily covers safety system components and components in the Nuclear Steam Supply Systems (NSSS), also called the primary systems. The NPRDS contains engineering descriptions of the systems and components and information on system and component failures. The system and component engineering descriptions contain the following information:

1. Component/system identification number
2. Component type
3. System/component operating environment (internal and external)
4. Component description
5. Component Manufacturer
6. System/component operating time expressed as a fraction of plant operating time.

The system and component failure reports contain the following information:

1. Plant, system and component identification
2. A verbal description of the failure
3. A verbal and coded description of the failure mode and cause
4. A verbal and coded description of the effect of the failure
5. A verbal and coded description of the corrective action
6. Identification of failure detection method

2.2.1.6 Information Source Limitations

The information sources described above are considered the best available, and have their uses and limitations for the purposes of this study. One of the limitations, as mentioned previously, is that the scope of each source is limited by its objectives. The utility semiannual and annual report has the broadest scope, while the NPRDS has a very limited scope. LERs

and the NRC Grey Book fall somewhere between the two extremes in terms of scope. The effect of these scope limitations on this study is that no single information source could provide all the information needed for the determination of the root causes of problems.

A second limitation of the information sources is their information format, that is, how the information is organized and presented. The information in the semiannual and annual reports is presented in text, with supporting material in tables and figures. LERs and NPRDS reports both have verbal descriptions, but they also have information coded in a prescribed format using special codes. The NRC Grey Book information also has a prescribed format, but makes less use of codes than for LERs and NPRDS reports. A good mix of coded data and verbal description for information makes it much easier to extract and use the most significant facts for problem identification.

A third limitation is the level of detail available from a given information source. NPRDS reports and LERs both have detailed information while the NRC Grey Book information is not very detailed. The level of detail in the semiannual and annual reports varies, with details often buried in text. The level of detail available made it difficult to determine exact root causes for problems and necessitated direct contact with the power plants.

Another, and very significant, limitation of the information sources is information consistency. All four information sources have general reporting requirements and procedures that leave room for judgement and interpretation. The information contained in these information sources is provided by personnel at the various "nuclear" utilities, and their interpretations of what is required and how it should be presented vary a great deal. Reporting differences can obscure the impact of given problems if some utilities report them and others don't, or if problems are reported such that the same problem is classified in different ways. For this study, engineering judgement and direct utility contacts were used to mitigate this limitation.

2.2.2 Data Bases

To make information more accessible and more usable, it is encoded and compiled in a data base. The consistency of information is also improved by entering it in a data base because of the structured format requirements. The data in a data base can be logically processed and summarized, either manually or by computer, to obtain meaningful statistics on the information contained therein.

Data bases are generally designed for a special purpose or purposes. Therefore, data bases can have limitations on their usefulness, especially when used for purposes other than those for which they were designed. This must be remembered when using any data base or formatted information source.

The primary data base used in this study was C-E's "PWR Operating History Data Base", which is commonly referred to as the Shutdown Data Base. The objective of this data base is to collect information on the causes of shutdowns and power reductions that have occurred at all commercial PWRs in the United States. The data is primarily used to maintain a chronological operating history for all PWRs, and to evaluate the impact of various systems and components on PWR availability. This data base was designed to have one eighty character record for each event, where an event is defined as a power reduction, a shutdown, or a continuation of a previous outage because of a different problem. Each record contains the following data:

1. The plant at which the event occurred.
2. The date the event occurred.
3. How the plant was shutdown.
4. The event duration in hours (for shutdowns only).
5. The system most directly responsible for the problem.
6. The component most directly responsible for the event (type only i.e.; valve, pump, etc.).
7. A brief description of the event.

The data base data was primarily extracted from the NRC Grey Book and the semiannual and annual reports. The data base covers all reported shutdowns and most power reductions that have occurred during commercial operation at all PWRs.

The software that manipulates this data base can sort the data on any one of the first six data fields except for event duration. Two primary reports generated by the data base software are the Chronological History Report and the System Impact Report. The first report provides a chronological listing of shutdowns and power reductions for each PWR, and calculates the plant availability on a yearly and overall basis. This report also provides a summary table (see Appendix 2B) that lists average plant availability by calendar year and PWR vendor. The second report provides a listing of events attributable to each system. Also produced is a summary table (see Appendix 2B) which provides the following information for each system:

1. Number of shutdowns attributed to the system.
2. Number of power reductions attributed to the system.
3. The number of plants affected.
4. Total shutdown hours.
5. Total period hours.
6. Minimum, maximum and average outage duration.
7. Average unavailability contribution for the system.

The primary limitation of this data base, as used for this study, is the sparse amount of detail on the "root causes" for the individual events. This limitation is primarily attributable to the limited detail of the original information sources, and to the fact that the data base design limits the amount of detail that can be included. A second limitation of this data base is that manual data processing is required to identify and evaluate events attributable to specific components or problems. This limitation is a result of the original design objectives of the C-E data base. Other limitations of this data base for use in this study include: a) scrams that occur while a plant is being synchronized to the grid are not reported, b) only event initiating causes are included unless an outage

is reported to be extended for a specific reason, and c) systems interaction problems are not reported. These limitations are attributable to the limitations of the information sources used.

The second data base used for this report was the LER data base. This data base contains a direct transfer of all information on all LERs submitted to the NRC. The primary objective of this data base is to simplify sorting the LERs into various groups for evaluation of potential abnormal occurrences. For this study, C-E used data base reports generated by the NRC software. The report used was a summary of events listed by plant, system and component. The primary limitations of this data base are a) LERs are not written for all shutdowns and power reductions and b) LERs are written for a very large number of non-shutdown events. Because of these limitations, both attributable to the design objectives for the LERs, a large amount of manual data manipulation and search was required to correlate LERs to shutdowns in C-E's Shutdown Data Base.

The third and fourth data bases used for this study were the EPRI Limiting Factors Analysis Study (LFAS) data bases for the Maine Yankee and Turkey Point 3 Power Plants.^(5,6) The EPRI LFAS Draft Final Report for Oconee⁽⁷⁾ was made available through EPRI late in the study, but the data was not entered in a computerized data base. Therefore, this data was not used as extensively as were the other PWR LFAS data bases. These data bases were used primarily to obtain additional data on events in the shutdown data base and to verify the general results of the study. These data bases are not publicly available at this time, but will be published by EPRI and will be available through the National Technical Information Service (NTIS).

The objective of the EPRI Limiting Factors Analysis studies was to identify problems that are limiting the availability and productivity of high availability nuclear power plants that had completed at least one refueling cycle. The EPRI LFAS data base contains information collected directly at three specific PWRs, one for each NSSS vendor, over a limited time period. These data bases contain detailed information on problems that had caused, or could have caused, shutdowns or power reductions.

The two major limitations of the LFAS data bases, for use in this study, are: a) only three PWRs are covered, and b) only a limited time frame is covered for each of the three PWRs. These limitations are a result of the objectives of the EPRI Limiting Factors Analysis studies. A third, less important, limitation was that manual data manipulation is required to correlate the event data in the sorted reports to the full event descriptions in the chronological history reports. One additional factor does have to be kept in mind when using these data bases. This factor is that the objective of the EPRI Limiting Factors Analysis studies was to evaluate the performance of high availability plants, that is, plants that have not had a large number of problems.

The fifth data base considered for this study was the NPRDS failure report data base. This data base contains a direct transfer of all the information contained on each NPRDS failure report form (see Section 2.2.1). The primary use for this data base in this study was to provide additional detail for events in the shutdown data base. This data base proved to be of limited use because of the following problems:

1. The NPRDS has a very limited scope, and does not cover portions of a nuclear plant which are major contributors to shutdowns and power reductions.
2. The NPRDS is concerned primarily with component reliability, so shutdown data is included as an "after thought", and is very sketchy.
3. The NPRDS failure reports that are publicly available do not identify the plants at which the failures occurred, making it extremely difficult to correlate NPRDS data with other data.

2.2.3 Summary Comment on Information Sources and Data Bases

Information sources and data bases described above were developed to collect data for specific objectives. By the nature of their existence, they can be assumed to be successful in meeting their individual objectives. For

this study's purposes, however, each of the information sources and data bases have limitations. This is because their designed objectives did not totally coincide with the objectives of this study.

Subtle factors in format and reporting procedures were also found to be limiting. Some data was listed by occurrence date while some data bases listed the events by reporting date. This made it extremely difficult to cross check event reports for more detail and information.

In general the data bases were useful in providing gross information for statistical work in identifying major problem areas. The amount of detail necessary to identify root causes was missing from the data bases. This is understandable considering the complexity of systems and the different plant and system configurations which exist at U.S. PWRs.

The information sources which came closest to providing the detail necessary for root cause determination were the Limiting Factors Analysis reports. These analysis reports were the result of concentrated evaluation efforts at only three plants for limited time spans. The fact that these reports were the results of concentrated efforts at these plants is indicative that root cause determination may only result from detailed discussions with plant personnel or evaluations and experimentation at the plant sites.

2.3 DATA ANALYSIS METHODOLOGY

2.3.1 System and Component Failure Identification

As described in Section 2.1, the general analysis steps followed in this study were:

1. Identify system problems (system or component failures) that have resulted in challenges to the protection system and have caused plant shutdowns.

2. Determine, as closely as possible, the root cause for the problems.
3. Determine means by which the problems can be eliminated or by which their impact can be reduced.

The data analysis phase of this study covered step 1 above, and a portion of step 2 to the extent that general problem causes were identified. Exact determination of problem root causes, and the determination of means to eliminate these problems (step 3 above) constitute the engineering evaluation phase of this study. Figure 2.3-1 shows the general flow of the data analysis. The following paragraphs describe the data analysis methodology in more detail.

The first step in the data analysis was to determine the impact of the various systems in terms of protection system challenges and plant unavailability contributions. Appendix 2A contains a general description of the scope of each system reported to have caused a plant shutdown. To do the first step, a shutdown data base System Impact Report was generated for the time period, January 1, 1961 through December 31, 1978 (this time period covers all PWR commercial operating experience to December 31, 1978). The System Impact Report Summary Tables (see Appendix 2B) explicitly provide the average plant unavailability contribution for each system which has caused a Plant Shutdown. Protection system challenges (scrams) for each system were obtained by counting the number of automatic and manual scrams attributed to each system. Scram counts were then divided by the total number of plant operating years (192.4 plant years) to obtain an average annual scram frequency for each system.

Once the system impacts were determined, the systems were ranked in decreasing order of importance in terms of both scram frequency (protection system challenges) and unavailability contributions. Systems ranking high in terms of both safety impacts and unavailability were selected for additional analysis. The systems chosen accounted for over 80% of the reported impact.

The next step was to identify the major problems within each of the systems selected for detailed analysis, and to determine their impact on the plant

in terms of protection system challenges and average plant unavailability contributions. This particular step required a great deal of manual data processing due to the judgement decisions involved in defining categories for problems and assigning problems to categories. Considering the limited detail of data bases and information sources, this was a formidable task in some cases, especially when components could not be identified. This phase of the analysis was done on a system-by-system basis in the following manner:

Step 1. Specific problems were determined from the event descriptions in the system event listing portion of the system impact report.

Step 2. For each specific problem the following data was determined:

- a. Average number of shutdowns per plant year
- b. Average number of scrams (automatic and manual) per plant year
- c. Average shutdown duration (hours) attributed to the problem
- d. Average annual (plant year) unavailability.

Step 3. Problems with an impact of greater than 0.025% unavailability were ranked in descending order of importance in terms of impact on safety and unavailability contribution.

Major problem profiles resulted from the last step. These profiles were lists of problems and their impact contribution on safety (scrams) and availability for each of the selected systems. Other important information which could enhance engineering evaluations such as shutdowns and average outage duration were also included in the profiles.

Results of this data analysis, Major Problem Profiles for the major systems, were the input for the next phase Engineering Evaluation, see Section 3.0.

2.3.2 Trends Analysis

Comparative trends analyses were performed at the plant and system level. Analyses were done on calendar year and commercial year basis, where commercial year is defined as the twelve month period between successive anniversaries of commercial operation for a given nuclear power plant. The objective of the comparative trends analysis was to look for plant and system level trends that might indicate trends, such as an increase (evolutionary improvements) or decrease in reliability. The Trends Analysis Process is summarized on Figure 2.3.2.

Calendar year analysis is the conventional approach, and is fairly easy to perform. The objective was to identify potential industry learning trends. A drawback to calendar year analysis is that the plant populations in each year have some significant differences. Differences include:

- a. Plant ages
- b. Plant operating experiences (related to age and previous plants)
- c. Generations of plant designs
- d. Plant sizes
- e. Plant vendor differences
- f. Refueling cycles

The time period selected for the calendar year analysis was the eleven year period between January 1, 1968 and December 31, 1978. (Prior to 1968, only one PWR was in commercial operation.) The first step in this analysis was to determine the plant population in each calendar year. This involved a simple counting procedure. The second step was to determine the average PWR plant availability in each calendar year. This data was obtained directly from the shutdown data base Chronological History Report Summary Table. The third step was to determine the average plant unavailability contribution attributable to each system in each calendar year. This was done by generating a shutdown data base Systems Impact Report for each calendar year, and extracting the system unavailability contributions from the summary tables. Data extracted from these summary tables was also used to calculate the unavailability contributions on the plant level for the six major unavailability cause categories:

1. Equipment problems (systems contributions) including:
 - a. Failure
 - b. Testing
 - c. Maintenance
2. Refueling and maintenance outage
3. Regulatory or administrative requirements
4. Personnel Error
5. Event external to the plant
6. Other (outages for which cause was not identified).

Commercial year analysis is felt to be a more beneficial approach than calendar year trend analysis. The objective is to identify potential problem trends such as decreases in reliability because of component wear-out. Plant differences that can affect commercial year trends analysis include:

- a. Generation of plant design
- b. Plant size
- c. Plant vendor
- d. Refueling cycle

The commercial year systems analysis followed the same steps as the calendar year analysis, but was more difficult. Software is not designed for commercial year analysis and thus more manual processing was involved.

The time frames selected for the commercial year analysis was the first eleven commercial years (only one PWR has more than eleven commercial years of operation). In calendar time, this analysis covers operating data from 1961 through 1978. The first step in this analysis was to determine the population of plants that had completed each commercial year. This again was a counting process with the ground rule that for the Nth commercial year ($N = 1, 2, 3, \dots, 11$). Only plants that had completed the Nth commercial year were counted in that year. To determine the average plant availability and the system and major cause category unavailability contributions a series of Commercial Year System Impact Reports were generated for each plant. Summary tables for each of these reports were then merged into a single summary table for each commercial year.

With the calendar year and commercial year system data reduction completed, the results were tabulated and plotted for comparison at the plant, major cause category and system level.

2.3.3 Calendar Year Problem Profiles

Calendar year data reductions were done for the problems identified in the system and component failure analysis phase of this study. As in the system and component failure analysis phase, this reduction was done on a system by system basis, working with the system problem lists which had been developed. For each system problem, the following data was extracted from the Calendar Year Systems Impact Report event listing for each calendar year and for the accumulated total impact:

- a. Average annual unavailability
- b. Number of shutdowns attributed to the problem
- c. Number of scrams attributed to the problem
- d. Number of power reductions attributed to the problem
- e. Average shutdown duration (hours) attributed to the problem
- f. Fraction of commercially operating plants affected.

This data was used to calculate the unavailability contribution, the shutdown frequency, the scram frequency and the power reduction frequency for each calendar year for each system problem. The data resulting from this analysis are contained in the calendar year Problem Profile Tables in Appendix 2E. Evaluation of these data tables can lead to many conclusions. Because of the many factors involved, the construction of graphs to show trends was not felt to be profitable. Also, due to the extensive listings, evaluations of each table were not accomplished. Conclusions obtainable from these tables include:

- a. If the impact was the result of one year experience averaged over total PWR experience or average experience.
- b. What fraction of plants are affected.
- c. Industry experience effects.

A commercial year problem profile would require much more manual data reduction to complete as the data base software is not setup to accomplish this task. Time and budget considerations prohibited accomplishing such a task at this time.

It is felt though that commercial year problem analysis would be much more valuable than the calendar year analysis. Commercial year analysis would allow assessment of the following: when in a plants life the problems are likely to occur, whether the average frequency of occurrence is a reasonable assumption for safety analysis, where emphasis on problems and operator training early in plant life should be concentrated, and plant/operator learning trends.

2.4 DATA ANALYSIS RESULTS

The primary objective of this analysis was to identify the major systems and components contributing to challenges to the RPS (scrams) and plant unavailability. A secondary objective was to determine how the impact varied as experience was gained. Results of the problem identification phase is contained in Section 2.4.1. Section 2.4.2 dicusses the results of the trends analysis phase.

2.4.1 System and Component Failure Identification Results

This analysis covers 192.4 plant years of operating experience for forty PWRs that entered commercial operation in the United States between July 1, 1961 and December 31, 1978. Cummulative average plant availability for these PWRs is 73.7%. This puts the cummulative average plant unavailability at 26.3%. Fifty-two percent (52%) of the plant unavailability is due to system and component failures, thirty-nine percent (39%) is due to refueling outages, and the other nine percent (9%) is due to personnel error, external causes, and regulatory or administrative restrictions. The average scram frequency for all PWRs is 9.3 scrams per plant year. The scram frequency for system and component malfunctions is 7.2 scrams per plant year, or eighty percent of the overall scram frequency.

Table 2.4-1 lists the plant unavailability contributions and scram frequencies for all systems that have caused a PWR shutdown. Due to the association and interaction of some of the systems listed, some of the systems were logically combined. This also would limit confusion due to system definitions as well as systems boundary definitions. Combinations of systems were as follows:

- a. Reactor Coolant System =
Reactor Coolant System plus Reactor Coolant Pumps
- b. Feedwater and Condensate System =
Main Feedwater System plus Condenser System
- c. On-Site Power System =
On-Site Power System plus Electrical Distribution System
- d. Reactor Vessel and Core =
Reactor Vessel Internals and Reactor Core & Fuel

From this listing, systems were ranked and selected for further study according to their importance on safety (scrams) and availability. Tables 2.4-2 and 2.4-3 provide a resultant listings of the systems contributing the majority of impact on safety and availability.

As indicated in table 2.4-2, the system which had the major (most significant) impact on safety was the Main Feedwater and Condensate System. The Turbine Generator System was counted as causing the next significant amount of challenges to the RPS and On-site Power systems ranked third. Ranked fourth was the RPS system and completing the list are the Control Rod Drive System, Main Steam System and the Reactor Coolant System.

The list was discontinued with the Reactor Coolant System because the next entry in the list would have been Non-Nuclear Instrumentation which is not a system per se. It is a "catch all" category for instrumentation failures which could not be associated with their applicable systems from information available. Eventual determination of associated systems would have spread the impact to other systems.

The systems listed accounted for approximately eighty-eight percent (88%) of the challenges to the RPS and eighty-seven percent (87%) of the plant shutdowns reported. Shutdown data was included to show the relative portion of scrams causing systems related shutdowns and to indicate the potential for additional scrams that may have resulted, had not the operators detected and reacted to potential problems.

Approximately fifty seven percent (57%) of all PWR systems related shutdowns were found to be scrams. This percentage is in the high seventies for the Main Feedwater and Condensate and the On-site Power System. Excluding the RPS, other systems listed are comparable to the average or lower.

The above percentage data also suggests that a significant portion of the systems related unavailability is attributable to scrams. This relationship was not confirmed. Systems determined to have the greatest impact on unavailability are listed in Table 2.4-3. Six of the seven systems listed are also listed as major systems causing scrams (table 2.4-2) which also indicates this strong relationship between safety and availability. Systems causing a large number of scrams also caused a large number of normal shutdowns and thus had large unavailability impacts.

Two systems on these lists did not follow the above relationship and thus were not listed in both tables. Many internally caused, unnecessary scrams resulted from the RPS whereas very few other shutdowns were attributed to the failure of the RPS. In addition the average shutdown duration ranked very low. Thus the RPS did not rank high as an unavailability contributor.

The other system, actually components, was the Reactor Vessel Internals and Core. Very few scrams and shutdowns were attributed to these components but understandably, several of the problems related to these components resulted in the longer outages. These components thus ranked high in impact on plant availability.

Evaluating the contributors to unavailability, Table 2.4-3 indicates that the systems with the major impact are the steam systems. The Turbine/Generator and Main Steam systems rank first and second respectively, well above other

systems. Ranked third is the Reactor Coolant System which is significantly above the Main Feedwater and Condensate Systems and the Reactor Vessel Internals and Core. The On-site Power System and Control Rod Drive Systems also caused significant amounts of unavailability.

A general observation of tables 2.4-2 and 2.4-3 is that systems with the larger components are attributable for large amounts of unavailability whereas failures of small components cause the greater number of scrams. This can be verified in the problem profiles of the systems with major impact which was the next step in this data analysis phase.

The next step was to identify problems within the major system. Major systems are defined as those systems listed in both tables 2.4-2 and 2.4-3. Problem profile lists for these systems, ranked by safety impact (scrams), are contained in Tables 2.4-4 through 2.4-9. Profile lists from which the above were compiled, ranked by availability impact, are located in Appendix 2B.

Identification of problems for tables 2.4-4 through 2.4-9 were limited to those problems having an unavailability impact of greater than 0.025 percent. This was done to keep the scope to manageable proportions. The lists give a profile of each problem, scram occurrence per plant year, shutdown (scrams plus normal shutdowns) occurrence per plant year, average shutdown duration in hours and the contribution to average annual unavailability from the problem.

The major purpose for these profiles is to provide the engineering principal investigator a list of the major problems for each of the major systems. These lists along with known problems were to be the basis for the selection of problems for the engineering evaluation phase. Discussion of these lists here is limited to the problems causing major impact on safety and availability.

Table 2.4-4 identifies the problem profiles of problems related to the Main Feedwater and Condensate System. The first four to five problems appear to be systems interaction problems, since specific component failures were not

identified as causing the scrams and only general problem areas were identified. Note the one to one relationship between scrams and shutdowns; almost every plant shutdown was the result of a reactor scram for these items. The largest availability impact for this system resulted from condenser tube leaks and control valves.

Problem profiles for the turbine generator system are provided in Table 2.4-5. Major challenges to the RPS from this system appear to be in the Electro-Hydraulic Control system and Turbine Valves. The major unavailability contributor in the turbine generator system is due to problems with turbine blades and other turbine mechanical components.

Table 2.4-6 contains the problem profile for the On-site Power System. Major challenges to the RPS result from the Non-Vital or Instrument Bus failures. Besides Power Cable fires, Main Transformer failures are the largest contributors to unavailability from the On-site Power System.

Control Element Assembly (CEA) Drops were determined to be the largest cause of scrams from the Control Rod Drive System. See Table 2.4-7. Causes for these drops were not reported in the information sources for a majority of cases. Control Rod repatch and change out along with Control Rod Drive housing seal leaks were the major contributors to unavailability.

For the Main Steam System, Table 2.4-8 identifies the major problem profiles. Main Steam Isolation Valves caused the greatest number of scrams and Steam Generator tube leaks accounted for the longer outages.

Challenges to the RPS from the Reactor Coolant System, Table 2.4-9, were determined to be mostly from reactor coolant pump seals and pressurizer spray valve problems. Reactor Coolant pump seals also had the greater impact on availability for the Reactor Coolant System.

The above discusses briefly the major problems determined to have significant safety impacts due to challenges to the RPS. Also indicated is the historic availability impact resulting from these problems. If the past is assumed to be an indication of the future, then the information provided can be

used as a quantitative measure of the potential benefit to be gained if solutions to these problems are found. Evaluation of the problems for solutions was left for the engineering principal investigators. This data analysis task provided the necessary identification of areas where the engineering evaluation should be concentrated for the greatest benefit.

2.4.2 Trend Analysis Results

Calendar year and commercial year availability trend analyses were performed to identify potential plant or system performance trends and to ascertain whether there is any indirect evidence of system and component aging (wear-out) problems. The time frames selected for the calendar year analysis and commercial year analysis are described in Section 2.3.2. It should be noted that although plotted together there is no direct relation between commercial and calendar year results. Data provided for a specific calendar year summarizes the performance of all PWRs in one specific calendar year. Data provided for Nth ($N = 1, 2, \dots, 11$) commercial year summarizes the performance of all PWRs which have completed N total years of operation where the Nth commercial year of operation for a given PWR can fall anywhere between July 1, 1961 and December 31, 1978 in calendar time.

Figure 2.4-1 shows the plant population in each calendar year, and Figure 2.4-2 shows the plant population in each commercial year. These figures illustrate two important facts based on industry growth:

- a. For the calendar year analysis, the early calendar years (1968, 1969) have a small plant population
- b. For the commercial year analysis, the later commercial years (10th, 11th) have a small plant population.

These facts should be remembered when comparing trend plots because when population sizes are small, singular events can have significant effects on averages when evaluating impact on a per plant year basis.

Figure 2.4-3 shows average annual plant availability for the last eleven years. (Note: Tabulated data for all trends plots is contained in Appendices 2C, 2D and 2E). On a calendar year basis, plant availability has varied rather erratically with no consistent upward or downward trend over the eleven year period. On the other hand, on a commercial year basis, average plant availability has shown a definite and consistent upward trend over the eleven commercial years (Figure 2.4-4). This indicates, at least tentatively, that plant performance improves with experience, that is, there is a learning, a burn-in, or improvement processes taking place.

Figures 2.4-5 and 2.4-6 show the relative unavailability contributions for the six major unavailability cause categories:

1. Equipment Problems (SYS)
 - Failures
 - Testing
 - Maintenance
2. Refueling and Maintenance Outages (RFL)
3. Regulatory or Administrative Requirements (REG)
4. Personnel Error (PER)
5. Events External to the Plant (XTN)
 - Electrical storms, etc.
6. Other (OTH)
 - Outages for which cause not identified.

The two cause categories that have the largest unavailability contributions are equipment problems and refueling and maintenance outages. These two cause categories exhibit similar trends for both the calendar year and the commercial year analyses. These trends are: The unavailability contributions attributed to equipment problems are decreasing while the unavailability contributions attributed to refueling and maintenance outages are increasing. One possible explanation for these trends is that plants are doing more preventive and corrective maintenance during the refueling and maintenance outages so the equipment performs better.

To determine whether individual system performance is improving, availability trends analyses were performed for ten systems and subsystems with significant unavailability contributions:

1. The Turbine/Generator System
2. The Main Steam System
3. The Reactor Coolant Pumps
4. The Control Rod System
5. The Main Feedwater System
6. The On-Site Power Systems
7. The Condenser Subsystem
8. The Electrical Distribution System
9. The Engineered Safety Features System
10. The Chemical and Volume Control System.

Comparative trends plots were made for the five systems with a total unavailability contribution of greater than 0.5%. These graphs are discussed in the following paragraphs. The supporting data for the graphs are contained in Appendices 2C and 2D.

Figure 2.4-7 shows the Turbine/Generator System unavailability contribution trends for both calendar and commercial year time frames. The behavior exhibited by the Turbine/Generator System is erratic in both analysis time frames, although the commercial year trends curve does have a pronounced "W" shape. A review of the system problem data for the Turbine/Generator System indicates that the unavailability contribution spike in 1974 is attributable to multiple turbine blade failures, as are the spikes in the first and sixth commercial years. The unavailability contribution spike in the eleventh commercial is attributable to a generator winding failure at one plant. A small plant population accounts for the magnitude of this spike.

Figure 2.4-8 shows the Main Steam System unavailability contribution trends for both the calendar and commercial year time frames. The calendar year trend curve shows an upward trend in unavailability contributions between 1968 and 1974, and somewhat of a downward trend from 1974 through 1978.

The commercial year trend curve indicates a downward trend in unavailability contributions from the fourth through the eleventh commercial years. The major Main Steam System problem reported is steam generator tube leaks. This problem still exists, and it is suspected that the plants are doing more steam generator inspection and maintenance during refueling outages, thereby, decreasing the impact.

Figure 2.4-9 shows the Reactor Coolant Pump unavailability contribution trends for both the calendar year and commercial year time frames. The calendar year trend curve indicates a rather erratic upward trend while the commercial year trend curve indicates a downward trend with increasing experience. The major Reactor Coolant Pump problem is seal failures. Discussions with plant operators indicate that Reactor Coolant Pump seals have experienced early life failures because of contaminants left in the Primary Coolant System after construction. As the contaminants are removed and the plant operators learn the peculiarities of their own system, the seal performance does improve.

Figure 2.4-10 presents the calendar year and commercial year unavailability contribution trends for the remainder of the Reactor Coolant System. The Reactor Coolant System unavailability contributions exhibit slight (and erratic) downward trends for both the calendar year and commercial year analyses. A qualitative review of the Reactor Coolant System problems did not reveal any particular trends that could account for the system trends. In general, the unavailability contributions for the individual problems was rather erratic.

Figure 2.4-11 presents the calendar year and commercial year unavailability contribution trends for the Main Feedwater System. The calendar year trend curve shows a slight, but very erratic, downward trend, and the commercial year trend curve shows a distinct, consistent downward trend. The Main Feedwater System is the primary contributor to protection system challenges, mostly consisting of several short duration shutdowns resulting from scrams. Automatic scrams on low steam generator level while bringing a plant to power in manual feedwater control accounted for a significant portion of the Feedwater System unavailability contributions in the first and second

commercial years. Reasons for this decrease could not be determined. Operator learning curve could account for this improvement, getting the plant back on line faster, thus decreasing the availability impact. Better manual control, low power automatic control or improved operator training on this system earlier in plant life may improve this problem.

Figure 2.4-12 presents the calendar year and commercial year unavailability contribution trends for the Control Rod System. Neither trend curve shows any particular trends with the exception of a spike in 1972, and a similar spike in the seventh commercial year. The 1972 spike is attributable to two singular problems; a complete control rod exchange at one plant, and control rod drive mechanism housing seal leaks at another plant. The seventh commercial year spike is also attributable to the housing seal leak problem as is the smaller spike in the first commercial year.

2.4.3 Calendar and Commercial Year Problem Profiles

A calendar year data reduction was performed for the system problems identified in the system and component problem identification phase of this study. These problem profile data tables are contained in Appendix 2E. These tables are provided for information only on problem trends and how industry experience may have affected these problems. Appendix 2F contains the problem profiles by plant commercial year of operation. The profiles provide additional information on plant and operations experience.

2.5 DATA VERIFICATION

The data verification phase of this study had three major objectives:

1. Verify the general conclusions of this study.
2. Obtain additional detailed information on specific events in the data base.
3. Verify the accuracy of the primary data source (C-E's "PWR Operating History Data Base").

Four data verification methods were used. These methods were:

1. Comparison and correlation of the "PWR Operating History Data Base" with other data bases and data sources.
2. Comparison of general study results with other published studies (primarily the EPRI Limiting Factors Analysis Study reports).
3. Plant visits to operating PWRs.
4. Telephone conversations with plant personnel at operating PWRs.

Data verification methods 1, 3 and 4 were used to meet data verification objectives 2 and 3, and data verification methods 2, 3, and 4 were used to meet data verification objective 1.

In general, the data verification showed the "PWR Operating History Data Base" to be complete and fairly accurate. Changes were made to approximately one percent of the data base records based on additional information obtained from the operating PWRs contacted. The Licensee Event Reports provided additional information on the causes of some shutdowns, and the EPRI Limiting Factors Analysis study data bases provided additional information on the causes of shutdowns and power reductions at the plants covered by these studies. These two data sources supplemented the "PWR Operating History Data Base" and did not reveal any required corrections or additions.

Discussions with plant personnel via plant visits or telephone conversations tended to verify the general results of this study in terms of the systems which are major contributors to protection system challenges and plant unavailability. These discussions also verified a number of study results regarding specific component and/or system problems, and they highlighted a number of plant specific problems. This information is covered in more detail in Section 3.0 Engineering Evaluation.

Comparison with the results in the final reports for the EPRI Limiting Factors Analysis studies also tends to confirm the general results of this study. One difference between the results of this study and the results of the Limiting Factors Analysis studies is the total unavailability impact attributable to problems such as reactor coolant pump seal maintenance and

turbine/generator maintenance. This type of maintenance often takes place during refueling or other extended outages. When this happens, these problems are not uniquely identified in the "PWR Operating History Data Base". This difference does not affect the study results regarding protection system challenges and has only a minor impact on the results regarding unavailability impact (i.e. economic impact).

2.6 RECOMMENDATIONS FOR DATA BASE IMPROVEMENTS AND ADDITIONAL TREND ANALYSIS

2.6.1 Information Source and Data Base Improvement

While this study could not have been done without the currently available information sources and data bases, a number of limitations were identified during the analysis and data verification phases of this study. The limitations in turn limited the identification and evaluation of problems that affect nuclear power plant safety, reliability and availability.

Nuclear power plants collect a large amount of information related to all phases of nuclear plant operation. The primary objectives for collecting this information are:

- a. Provide information needed to keep the plant operating.
- b. Fulfill regulatory requirements.

In general, the information collecting activities at a nuclear power plant are specific to that plants needs and much of the information is not accessible in an easy manner. Because of regulatory requirements, some of the information collected at a nuclear power plant is publicly available through various sources. (See Section 2.2.1.) However, the publicly available information sources have several limitations:

1. No single public information collection agency requires a complete report of all plant events. Thus no single public information source covers the available information or the information needed for this type of analysis.

2. Inconsistent reporting.
3. Insufficient detail.
4. Data sources overlap in scope but correlation is difficult.
5. In some information sources the information is presented in such a manner that it is difficult to use. No consistent format or recording requirements.
6. All information sources have specific objectives that limit the usability of the information collected for other purposes.

Information collected in the various information sources has been entered into data bases for ease of manipulation. These data bases are also subject to limitations:

1. The data bases are subject to the limitations of the information sources from which data is obtained.
2. The design objectives of the data bases and their software limit the manner in which the data can be manipulated.

Based on the observed information source and data base limitations, we recommend a program be undertaken to improve the quantity and quality of plant operating data available to the industry and to enhance the efficiency of the information collection activities at nuclear power plants while decreasing the reporting burden on the utilities. This program should include the following phases:

1. Identify plant and industry information/data needs.
 - a. Type of information.
 - b. Level of detail.
2. Identify the type and amount of information currently collected at nuclear power plants.
3. Identify the methods used to collect, store and disseminate information at a nuclear power plant.
4. Identify the redundant reporting requirements for existing public information sources.

5. Identify data analysis manipulation and reporting requirements
 - a. For plant
 - b. For the industry
 - c. For regulatory agencies.
6. Design an Integrated Information System consisting of two related segments.
 - a. A plant information collection and management system.
 - b. A public information system.
7. Recommend other methods to improve the efficiency of information collection and dissemination.

2.6.2 Additional Data and Trends Analysis

This study has evaluated the impact of component failures and problems within major systems on the safety of nuclear power plants. This study concentrated on safety impact measured in challenges to the protection system (scrams). Limited trend analyses were also performed to identify potential wear-out or improvement trends. This study has produced significant and useful results, but additional data analysis should be performed. The objectives of this additional data analysis would be:

- a. Evaluate the impact of all systems on plant safety.
- b. Evaluate maintenance data to determine relationship to system performance.
- c. Identify potential new problems.
- d. Provide a set of base line information for the evaluation of future performance.
- e. Provide a set of statistical methods for predicting future performance.

The study should consist of three tasks as follows:

1. Evaluate system and component performance for all systems using shutdown and maintenance data, and identify specific problems and their overall impact on plant operation.

2. Perform calendar year and commercial year trend analyses on all systems and problems. These trend analyses should use more rigorous statistical methods, and should take into consideration differences such as:
 - a. NSSS Vendors
 - b. Architect-Engineers
 - c. Plant size
 - d. Plant design vintage
 - e. Plant age
3. Develop a set of statistical procedures for predicting system and plant performance, and test the methods using current data.

The results of this study should provide a valuable tool for identifying potential new challenges and for evaluating the impact of design changes resulting from other programs.

REFERENCES FOR SECTION 2

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3. Limiting Factors Analysis of a High Availability Nuclear Plants Study Performed at Turkey Point Unit 3, EPRI RP894-4 Final Report in publication.
4. NUREG 0020; "Operating Units Status Report"; U.S. Nuclear Regulatory Commission; published monthly since July, 1974.
5. NUREG 0161; "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File"; U.S. Nuclear Regulatory Commission; July, 1977.
6. "Reporting Procedures Manual for the Nuclear Plant Reliability Data System"; Southwest Research Institute for the ANSI N18.20 Committee; Revision 6, September, 1979.
7. NPRDS-Q02; "Quarterly System and Component Failure Listing"; Southwest Research Institute for the ANSI N18.20 Committee; Published quarterly since July, 1974.

ABBREVIATIONS FOR SECTION 2

ANSI - American National Standards Institute
C-E - Combustion Engineering, Inc.
PWR - Pressurized Water Reactor
NRC - Nuclear Regulatory Commission
NPRDS - Nuclear Plant Reliability Data System
NSSS - Nuclear Steam Supply System
BOP - Balance of Plant

Table 2.4-1
PWR SYSTEM IMPACT LIST
JULY 1961 TO DECEMBER 1978

Sheet 1 of 2

<u>System</u>	<u>Unavailability Contribution Per Plant Year (Percent)</u>	<u>Shutdowns Per Plant Year</u>	<u>Scrams Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>
Turbine/Generator	3.764	3.01	1.50	109.7
Main Steam	2.667	1.07	0.37	219.5
Reactor Coolant Pumps	1.225	0.54	0.15	200.6
Control Rod	0.772	0.86	0.42	78.4
Main Feedwater	0.762	2.99	2.47	22.4
Reactor Coolant	0.614	0.68	0.14	79.1
Reactor Vessel Internals	0.610	0.03	0	1716.1
Reactor Core/Fuel	0.584	0.05	0.02	985.5
On-Site Power	0.500	0.72	0.63	60.7
Condenser	0.439	0.35	0.09	110.5
Electrical Distribution	0.347	0.24	0.13	124.4
Engineer Safety Features	0.142	0.13	0.03	95.6
Chemical and Volume Control	0.142	0.12	0.04	99.6
Nuclear Instrumentation	0.087	0.20	0.09	37.7
Reactor Protection	0.078	0.49	0.48	13.8
Auxiliary Feedwater	0.075	0.03	0.02	211.4
Circulating Water	0.074	0.41	0.08	46.3
Seismic Restraints	0.064	0.03	0	178.7
Reactor Vessel	0.057	0.04	0.01	120.2
Shutdown Cooling	0.055	0.04	0	116.4
Emergency Diesel Generator	0.055	0.04	0.01	131.8
Non-Nuclear Instrumentation	0.046	0.27	0.22	15.3
Containment	0.038	0.04	0.02	92.0
Blowdown System	0.034	0.04	0.01	71.1
Containment Isolation	0.031	0.04	0.01	64.4

Table 2.4-1 (Continued)
PWR SYSTEM IMPACT LIST
JULY 1961 TO DECEMBER 1978

Sheet 2 of 2

<u>System</u>	<u>Unavailability Contribution Per Plant Year (Percent)</u>	<u>Shutdowns Per Plant Year</u>	<u>Scrams Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>
Compressed Air	0.030	0.10	0.09	26.6
Rod Position Indicating	0.021	0.04	0.03	43.8
Reactor Regulating	0.020	0.12	0.11	14.4
Boric Acid Injection	0.014	0.03	0.01	38.8
Sampling	0.011	0.01	0	94.4
Component Cooling Water	0.005	0.03	0.02	16.1
Cooling Tower	0.003	0.02	0.01	17.7
Area Monitoring	0.003	0.01	0	48.9
Containment Ventilation	0.002	0.01	0	18.3
Demineralized Water	0.002	0.01	0	34.7
Plant Process Computer	0.001	0.01	0.01	17.8
Loose Parts Monitoring	0.001	0.01	0	10.6
Containment Air Purification	0.001	0.01	0	10.6
Turbine Bypass	FO.001	0.01	0	6.8
Communication	FO.001	0.01	0.01	1.2

Table 2.4-2
MAJOR PWR SYSTEM FAILURES RESULTING
IN PLANT SCRAMS
(7/61 THROUGH 12/78)

<u>System</u>	<u>Scrams Per Plant Year</u>	<u>Shutdowns Per Plant Year</u>
1. Main Feedwater and Condensate System ⁽¹⁾	2.56	3.34
2. Turbine Generator System	1.50	3.01
3. On-Site Power Systems ⁽²⁾	0.76	0.96
4. Reactor Protective System	0.48	0.49
5. Control Rod Drive System	0.42	0.86
6. Main Steam System	0.37	1.07
7. Reactor Coolant System ⁽³⁾	0.29	1.22
Other	<u>0.86</u>	<u>1.68</u>
TOTALS	7.24	12.63

(1) Main Feedwater + Condenser Systems

(2) On-Site Power + Electrical Distribution Systems

(3) Reactor Coolant + Reactor Coolant Pump Systems

Table 2.4-3
PWR PLANT UNAVAILABILITY CONTRIBUTIONS
(7/61 THROUGH 12/78)

<u>System</u>	<u>Average Unavailability Per Plant Year</u>
1. Turbine/Generator System	3.76%
2. Main Steam System	2.67%
3. Reactor Coolant System ⁽¹⁾	1.84%
4. Main Feedwater and Condensate System ⁽²⁾	1.20%
5. Reactor Vessel, Internals and Core ⁽³⁾	1.20%
6. On-Site Power Systems ⁽⁴⁾	0.85%
7. Control Rod Drive System	0.77%
Others	<u>1.09%</u>
TOTAL	13.38%

(1) Reactor Coolant System + Reactor Coolant Pump

(2) Main Feedwater + Condenser System

(3) Reactor Vessel Internals + Reactor Core/Fuel

(4) On-Site Power + Electrical Distribution System

Table 2.4-4

MAJOR PROBLEM PROFILE

MAIN FEEDWATER & CONDENSATE SYSTEM

<u>Problem</u>	<u>Number of Scrams Per Plant Year</u>	<u>Number of Shutdowns Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>	<u>Average Annual Unavailability Contribution</u>
Steam Generator Level Trips - Power Level Not Given	0.61	0.61	7.9	0.055%
Steam Generator Level Trips - During Startup	0.55	0.55	4.2	0.026%
FWP Trips - Cause Not Specified	0.24	0.26	21.6	0.064%
Feedwater Control System	0.17	0.17	35.5	0.069%
Feedwater Control Valves - Problems & Maintenance	0.11	0.18	58.8	0.119%
Feedwater Pump Speed Controller	0.06	0.08	89.4	0.079%
Misc. Feedwater System Problems and Maintenance	0.04	0.06	63.6	0.045%
Condenser Tube Leaks	0.02	0.18	191.2	0.385%
Feedwater System Piping	0.02	0.05	134.1	0.080%
Feedwater Heater Leaks	0.01	0.10	35.6	0.040%
TOTALS	1.83	2.24		0.962%

Table 2.4-5

MAJOR PROBLEM PROFILE

TURBINE/GENERATOR SYSTEM

<u>Problem</u>	<u>Number of Scrams Per Plant Year</u>	<u>Number of Shutdowns Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>	<u>Average Annual Unavailability Contribution</u>
EHC System-Misc. Problems	0.21	0.28	12.7	0.041%
EHC Auto-Stop Oil/Fluid Systems Problems	0.17	0.30	21.2	0.072%
Turbine Control/Stop Valve Malfunctions or Maintenance	0.08	0.23	78.7	0.210%
Turbine/Generator Mechanical Problems/Maintenance	0.06	0.21	276.6	0.672%
Turbine/Generator Inspections and Tests	0.06	0.23	49.9	0.130%
Generator Exciter Malfunctions	0.05	0.09	67.5	0.048%
Turbine Blade Failures	0.04	0.10	1343.3	1.592%
Turbine Bearing Problems	0.02	0.06	381.0	0.271%
Turbine/Generator Balancing	0.02	0.12	28.3	0.039%
Moisture Separator/Reheater Tube Leaks	0.02	0.07	41.3	0.034%
Generator Stator Winding Problems	0.01	0.02	961.4	0.171%
Misc. Moisture Separator/Reheater Maintenance	0.01	0.09	112.0	0.113%
Turbine Piping/Casing Steam Leaks	0.01	0.15	41.9	0.072%
TOTALS	0.78	2.01		3.529%

Table 2.4-6

MAJOR PROBLEM PROFILE

ONSITE POWER SYSTEMS

<u>Problem</u>	<u>Number of Scrams Per Plant Year</u>	<u>Number of Shutdowns Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>	<u>Average Annual Unavailability Contribution</u>
Non Vital/Instrument Bus Faults	0.14	0.17	47.7	0.091%
Circuit Breaker/Switch Faults	0.09	0.10	25.5	0.029%
Switchyard Relay/Breaker Faults	0.05	0.06	72.6	0.052%
Main Transformer Failures	0.03	0.07	347.7	0.268%
Power Cable Fires	0.01	0.01	2313.9	0.274%
	<hr/>	<hr/>		<hr/>
TOTALS	0.37	0.43		0.755%

Table 2.4-7

MAJOR PROBLEM PROFILE

CONTROL ROD DRIVE SYSTEM

<u>Problem</u>	<u>Number of Scrams Per Plant Year</u>	<u>Number of Shutdowns Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>	<u>Average Annual Unavailability Contribution</u>
CEA Drops - Cause Not Given	0.09	0.14	18.8	0.030%
CEA Drops - Control Rod Drive Mechanism Failures	0.04	0.13	59.8	0.089%
Control Rod Drive Housing Seal Leaks	-	0.11	164.5	0.205%
Control Rod Drive Mechanism Faults - No CEA Drop	0.01	0.05	80.2	0.043%
CEA's Inoperable	0.01	0.04	73.00	0.030%
Control Rod Repatch/Changeout	-	0.02	164.5	0.269%
	<hr/>	<hr/>		<hr/>
TOTALS	0.15	0.49		0.666%

Table 2.4-8

MAJOR PROBLEM PROFILE

MAIN STEAM SYSTEM

<u>Problem</u>	<u>Number of Scrams Per Plant Year</u>	<u>Number of Shutdowns Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>	<u>Average Annual Unavailability Contribution</u>
MSIV Closures - Cause Not Given	0.10	0.14	37.6	0.060%
Misc. Problems and Maintenance	0.08	0.14	66.8	0.103%
Steam Generator Tube Leaks	0.04	0.40	385.3	1.734%
MSIV Solenoid Valve Failures	0.04	0.04	73.9	0.031%
Steam Generator Inspection, Testing and Maintenance	0.01	0.10	456.2	0.514%
Steam Line Drain Line Leaks	0.01	0.04	78.5	0.033%
	<hr/>	<hr/>		<hr/>
TOTALS	0.28	0.88		2.549%

Table 2.4-9
MAJOR PROBLEM PROFILE
REACTOR COOLANT SYSTEM

<u>Problem</u>	<u>Number of Scrams Per Plant Year</u>	<u>Number of Shutdowns Per Plant Year</u>	<u>Average Shutdown Duration (Hours)</u>	<u>Average Annual Unavailability Contribution</u>
Reactor Coolant Pump Seal Failures	0.04	0.18	302.8	0.628%
Pressurizer Spray Valve Problems	0.03	0.17	56.0	0.110%
RCP Seal Injection System Problems	0.02	0.04	119.6	0.057%
Unidentified Reactor Coolant Pump Malfunctions	0.02	0.03	437.9	0.129%
Loop Isolation Valve Problems	0.02	0.03	168.5	0.040%
Leakage from Misc. Reactor Coolant System Valves	0.01	0.15	78.5	0.130%
Reactor Coolant Pump Lube Oil System Problems	0.01	0.14	56.0	0.110%
Pressurizer Manway Gasket Leaks	0.01	0.05	74.8	0.040%
Misc. Reactor Coolant System Maintenance	0.01	0.03	138.3	0.041%
Reactor Coolant Pump Motor Faults	0.01	0.02	289.2	0.069%
TOTALS	0.19	0.92		1.672%

Note: Not to scale

Information Source Relationships

Figure 2.2-1

DATA ANALYSIS
PROBLEM IDENTIFICATION PROCESS

Figure 2.3-1

TRENDS ANALYSIS PROCESS

*SDB = Shutdown Data Base
 **CHIRST = Chronological History Report Summary Table
 ***SIRST = System Impact Report Summary Tables

FIGURE 2 3 2

Figure 2.4-1
Number of Pressurized Water Reactors
In Commercial Operation
by Calendar Year

*Three Mile Island 2 not included

Figure 2.4-2
Number of Pressurized Water Reactors
in Commercial Operation
by Commercial Year

Figure 2.4-3
PRESSURIZED WATER REACTOR
PLANT AVAILABILITY
by
Calendar Year

Figure 2.4-4
PRESSURIZED WATER REACTOR
PLANT AVAILABILITY
by

Commercial Year

$M = 1.167$

Figure 2.4-5
Unavailability Contribution Trends
by Major Cause Category

Figure 2.4-6
Major Unavailability Contributors
by
Commercial Year

Figure 2.4-7
Turbine/Generator System
Unavailability Contribution Trends
by
Calendar and Commercial Year

Figure 2.4-8
Unavailability Contribution Trends
for
The Main Steam System
by Calendar and Commercial Year

Figure 2.4-9
Unavailability Contribution Trends
for
The Reactor Coolant Pumps
by Calendar and Commercial Year

Calendar Year
Commercial Year

Figure 2.4-10
Unavailability Contribution Trends
for
The Reactor Coolant System
by Calendar and Commercial Year

Figure 2.4-11
Unavailability Contribution Trends
for
The Main Feedwater System
by Calendar and Commercial Year

Figure 2.4-12
Unavailability Contribution Trends
for
The Control Rod System
by Calendar and Commercial Year

Section 3.0

ENGINEERING EVALUATION

3.0 ENGINEERING EVALUATION

The purpose of the engineering evaluation was to review the data analysis, select and evaluate those problems which occur frequently or have significant impact on safety or availability for ways by which the problem can be reduced or eliminated. Recommendations for future R&D efforts were also to be included.

Due to the limitations of the data base, before any meaningful evaluation could occur, more information and a verification of the problem was necessary. Usually the reports from which the data was obtained only contained the problem and possibly symptoms of the problem, the effect and not the cause. It was therefore necessary to track backwards in the cause and effect chain for root causes and ways by which the effects could be minimized.

This analysis overlapped to some extent the data analysis in that the engineering investigators had to be familiar with the data descriptions as well as the sources from which the data was derived. The problems then were usually further broken down into sub-categories or cause(s). An example is the Feedwater Pump and causes for its failures, i.e., bearings, impeller, housing, hydraulic controls and so on.

In the beginning of this study, for efficiency it was decided to divide the engineering evaluation into three major areas appropriate to the C-E organizational structure. This would allow participation by those individuals most capable of performing the individual problem tasks involved. The division of scope was as follows:

- Fluid and Mechanical Components (FMC)
- Instrumentation, Control and Electrical (ICE)
- Systems Integration (SI)

Fluid and Mechanical components addresses mechanical and fluid system equipment; pumps valves, turbines, filters and heat exchangers. Instrumentation Control and Electrical addresses electrical and electronic problems and Systems Integration looked at the overall picture including the integration

of Fluid and Mechanical Systems with Instrumentation Control and Electrical equipment. This allows consideration of systems interaction problems as well as a review of systems design changes having the potential to decrease component failure problems.

The organization of this section of the report is arranged per the three categories mentioned above. Section 3.1 addresses the fluid and mechanical component evaluation; Section 3.2 discusses the instrumentation control and electrical efforts; and Section 3.3 discusses the areas of major impact upon systems interaction and integration.

Section 3.1

ENGINEERING EVALUATION
of
Fluid and Mechanical Components

3.1 Engineering Evaluation of Fluid and Mechanical Components

3.1.1 Introduction

The objective of the Fluid and Mechanical Components (FMC) portion of this study was to evaluate major component failures leading to challenges to the RPS and plant unavailability for solutions which eliminate or reduce the problem. In such an evaluation, an ideal solution would be the elimination of the cause. This is not always possible as elimination of the cause may be cost prohibitive. Therefore, the option available is to reduce the resulting effects from the cause(s).

The data base was limited in the amount of root cause information as the allotted field for component failure description was limited to approximately fifty characters. Moreover, information sources tend to report the catastrophic failure of equipment and not the underlying root causes. Effects were reported rather than the cause(s) in most cases.

This lack of root cause information provided a significant challenge to the investigators. Determining root causes require a knowledge of equipment and systems in order to deduce probable root causes which resulted in the effects reported. The experience of the evaluators was also important since equipment would not be available for visual inspection and experimentation. The investigators were limited to the information obtainable, either in documentation or through discussions.

The investigators assisting in this FMC portion of the program were experienced in the design, specification and procurement of NSSS fluid and mechanical components. Their experience was of significant value in determining root causes from the effects reported. This was a major milestone to be overcome in working toward potential solutions to the problems reported.

Nuclear power plants are very complex as well as the equipment within the plants. Different systems configurations exist as well as a variety of equipment and vendors. Standard plants do not exist; each plant has its own systems characteristics. Operating procedures can also vary, thus

similar problems at different plants may have obscure root causes, and thus a solution for one plant may not apply to the others. The complexity of plants and the operating differences can effect the results if information is available at one plant only. This should be kept in mind when considering the problems as reported.

The rest of this section is organized as follows: Section 3.1.2 addresses the scope and methodology of this FMC portion of the study. Section 3.1.3 provides a discussion of the problems and causes and 3.1.4 contains the FMC summary and conclusions.

3.1.2 Scope and Methodology

This FMC portion of the study concentrated on major fluid and mechanical equipment problems. This evaluation included problems with pumps, valves, seals, heat exchangers and other hydraulic components. For those problems found to be caused by instrumentation and electrical parts, evaluations were left for review by I&CE engineers and are reported in Section 3.2. Non-mechanical failures were left to the systems integration group and are reported in Section 3.3.

Preceding the performance of engineering evaluations, the shutdown data base was independently reviewed by the engineering evaluators for several purposes. One reason for this independent review was to allow the design engineering group to become familiar with the data and establish some confidence in the data reduction results. More importantly, in determining root causes of problems, the design engineers would have to seek reference to the information sources from which the data was derived to obtain additional detail. The above preliminary step helped to guide the engineering investigators directly to where more detailed information of the specific equipment failure was located. This independent review also allowed a data reduction on FMC problems alone, I&CE and SI problems excluded. This data reduction was in general agreement with the data analysis results reported in Section 2.0, giving additional credence that the major problems needing evaluation were indeed established.

With the results from the data reduction above, the detailed evaluation for root cause(s) could proceed. Sources of information varied from the information sources reported in Section 2.0 to the discussion of events with personnel located at the nuclear plant involved. In many cases, additional information requirements required this direct contact with plant personnel. The information available in documented information sources helped establish rapport and probe memories to obtain the additional information needed. Once root causes were determined, the evaluation for solutions to eliminate or reduce the root cause or minimize the effects could proceed.

The phases of the mechanical equipment study program developed from the initial examination of the shutdown data base, through a detailed researching of each event, progressing to identification of the root cause of the equipment problem. Within the limitations of the study, recommendations are made for evaluation of equipment design changes, procedural changes or further study.

Specialist specifying engineers were assigned the tasks of researching each event relating to equipment within their experience. This allowed them to bring to bear their knowledge of equipment design resulting from their experience with other equipment designers, manufacturers and operators and their installation, maintenance and troubleshooting experience. This experience enabled the engineer to predict a sequence of events leading to a probable root cause of many of the events that were researched; and to provide recommendations for improvement of the component and system performance.

The scope of the fluid and mechanical component study was limited after completion of the independent data analyses to the following systems:

- o Main Feedwater and Condensate System
- o Turbine Generator System
- o Reactor Coolant System
- o Main Steam System
- o Control Rod Drive System

These systems are also the logical FMC systems resulting from the data analysis phase as reported in section 2.0. The On-site Power and Reactor Protection systems were not appropriate to FMC review.

Limiting the scope to the above systems allowed a more detailed review of the major problems. This evaluation is discussed in the next section.

3.1.3.0 Engineering Evaluation

3.1.3.1 FMC Independent Data Analysis

The results of the FMC data analysis are consistent with the results obtained in the data analysis reported in Section 2. Tables 3.1-1, 3.1-2 and 3.1-3 present the results arranged by the important factors of concern; outage events, unavailability and the systems involved.

Table 3.1-1 lists the FMC components causing outage events along with the calculated average occurrence frequency and average unavailability contribution. This table shows the relative impact on safety, challenge frequency to the RRS, and the relative impact on unavailability. In terms of improvement capacity, the frequency of occurrence can be qualitatively related to the capacity for improved safety whereas the Average Annual Unavailability can be related to capacity for plant availability improvement. The first column is the relative incentive for improved safety and the second is the relative economic incentive.

Table 3.1-1 is ranked by the incentives for safety improvement. Table 3.1-2 presents the same information, however the ranking of components is by the magnitude of the unavailability contributions, the economic improvement incentive.

As can be seen in the tables, three problems; main condensers, turbine valve testing and feedwater filter cleaning; caused a significant number of power reductions. These were included because the significant occurrence frequency was obvious in the data base. The number of power reduction occurrences is felt to be indicative of a potential design weakness. The

weakness can be considered a design problem for reserve capacity, a sub-component reliability problem, or both. Moreover, the recurrent problems have the greater probability of occurring in conjunction with other problems, potentially with compounding effects.

The results are the summation of all occurrences at all plants averaged over the summation of all commercial years of operation at those plants. Therefore, some events can be listed which may be the result of the events at one or a few plants averaged over all plants. The incentives at some plants for improvement can thus be significantly higher. This portion of the study concentrated on major PWR problems as determined by the average impact on all plants. In general the problems are the results of the annual average experience at a majority of the plants.

Table 3.1-3 was constructed to show the consistency with the data analysis results as discussed in 2.0. As can be seen or table 3.1-3, all the problems listed are a part of the systems listed in section 2.0 as causing the majority of plant scrams and impact on plant unavailability. Also notable is the order of impact. This impact order change is due to the exclusion of I&CE and SI problems. Obviously, the On-site Power System and Reactor Protection System are excluded as they are not FMC systems.

The first objective of the FMC evaluation was to determine the major FMC component failure problems where further evaluation for improvements would be most beneficial. Table 3.1-3 establishes the major FMC systems and problems. The problems listed in this table became the basis for further evaluation. Detailed discussion of further analysis is contained in the following section.

3.1.3.2 Engineering Evaluation Results

The FMC systems determined to be the major contributors to challenges to the RPS and plant unavailability are as follows:

- o Feedwater and Condensate System
- o Turbine Generator System
- o Main Steam System
- o Reactor Coolant System
- o Control Rod Drive System

The results of the engineering evaluation of these systems and the major component contributors to outage event occurrences and duration of outages is discussed in this section. The discussion includes a description of the problem(s), the extent of the problems, and potential solutions and recommendations for further study when justified.

A. Feedwater and Condensate System

The major FMC components in the feedwater and condensate systems (see Table 3.1-3) contributing to plant scrams, outages, power reductions and unavailability are:

Feedwater Pumps
Feedwater Valves
Main Condenser
Feedwater Heaters
Feedwater Filters

A.1 Feedwater Pumps

Two basic types of feedwater pumps are used at nuclear power plants, turbine driven and motor driven. Each has its own characteristics. The motor driven pump is constant speed with a constant discharge pressure whereas the turbine driven feedwater pump is variable speed with a variable discharge pressure. The variable speed turbine pump has more controls and is considered less reliable than the electric driven pump. However, the electric driven pump has its drawbacks also in that a higher discharge pressure requires the feedwater control valves to work at higher differential pressures, especially at low power levels. It is thus important that feedwater valves be characterized to the type of pump.

One hundred forty-four (144) feedwater pump related outages at thirty-one plants were investigated using the references of the Data Analysis phase, telephone communication with stations and visits to stations. The following is a discussion of the reported causes:

a) Outages due to oil system problems

Fifty-nine outages (41% of 144) relating to turbine feedwater pump oil system(s) were found in over 1/3 of the stations. These outages can be further defined as follows:

1) Hydraulic control oil system

Outages of this type accounted for forty-three of the fifty-nine oil system outages. Included in this category are outages listed in the shutdown data base as oil pump failure, dirty oil, governor, control, steam flow/feed flow mismatch, and low steam generator level trip.

Outages occurred when contaminants (paint, rust, dirt, etc.) entered the control system and blocked small control orifices or blocked the seats of ball check valves causing improper closure. These blockages cause the speed controller to overspeed due to increased control pressure when control is transferred from the governor valve positioner to the governor. The outages are thought to be caused by the present design in which the hydraulic control oil and lubrication oil use the same reservoir.

2) Oil Line Leaks

Outages of this type accounted for ten of fifty-nine oil system outages and were caused by wear and aging of connecting hoses.

3) Miscellaneous Turbine Driven Feedwater Pump Oil Problems

Outages of this type accounted for six of the fifty-nine oil system outages and include leaking gaskets and valves.

These outages are caused by wear and aging during normal operation.

b) Outages due to Bearing Problems

Fourteen outages (10%) of this type were prevalent in 8 (20%) of the stations. These outages can be further defined as follows:

1) Repair or Replacement

Outages stated to be due to repair or replacement accounted for 7 of the 14 outages. No specific reasons for the repairs or replacement were reported. Bearing wear and vibration are the usual causes.

2) Outages due to Vibration

Outages of this type accounted for 7 of the 14 outages. The causes stated were improper clearance, wear, misalignment, unbalance and vibration.

Vibration, and repair or replacement of bearings usually attributable to bearing wear. The cause of bearing wear or degradation is normally a lack of sufficient lubrication or particles in the lube oil or bearing. Either of the above will cause accelerated wear of bearings.

c) Outages Due to Steam Leaks

Thirteen outages (8%) of this type were reported. The cause is attributed to the wear of gaskets due to normal expansion and contraction of joints.

d) Outages Due to the Pump Driver

Eleven outages (8%) of this type were reported. The outages were further defined as follows:

1) Outages due to the turbine driver

Outages due to the turbine driver accounted for seven of the eleven outages of this type. The cause was damage to the blades resulting in vibration. No specific information on the blade damage could be supplied by the stations involved.

2) Outages Due to the Motor Driver

Outages due to the motor driver accounted for four of the eleven outages of this type. The cause was due to electrical problems; no specific reasons could be supplied by the stations involved.

e) Outages Due to Impeller Failure

Six outages (4%) of this type were reported. No specific causes could be given for 4 of the outages by the stations contacted at the time. The remaining 2 outages were caused by foreign objects lodged in the impeller. These foreign objects were pieces of equipment from failed parts of upstream equipment.

f) Outages Due to Casing Failure

Four outages (3%) of this type were reported. These outages occurred in two "sister" plants only. The casing failures were due to porous welds that contained slag inclusions and sand and gas cavities in the castings. The cause of these failures was attributed to poor castings and loose specification.

g) Outages Due to Miscellaneous Reasons

Nine outages (6%) of this type were reported. These outages are singular in occurrence and include faulty valves, cracked couplings, shaft failure, etc.

h) Outages Without Identifiable Causes

Twenty-eight outages (19%) of this type were reported as "trip". No further information could be obtained from the documents used to verify the data, and response from the stations contacted was insufficient for cause determination.

Conclusions and Recommendations

Feedwater pump problems were found to result mostly from the hydraulic oil systems, steam leaks, and other pump drive components of turbine drive feedwater pumps. Bearing failures were next with the rest of the failures being random or non generic in nature. A significant number (29) could not be attributed to any specific causes.

The problems with the hydraulic oil system have as a root cause the inherent design, a combined hydraulic and lubrication oil system. Force-fed lubrication systems are characteristic of the current design used for turbine driven feedwater pumps. Bearing problems are also the result of oil problems, either insufficient or dirty oil. The remaining outages had no single common or identifiable root cause. Most of these outages were singular in nature and have no apparent recurrent pattern that is generic. The combined number of these events however is considered significant which may only be reduced by systems level design changes.

Feedwater pump problems are varied and have varied causes. Generally all occurrences resulted in scrams. For functional reliability, the preferred solution would be increased feedwater pump capacity through the addition of an additional pump or pumps. Additional study is needed and recommended to determine the feasibility of mixing pump types, e.g., two turbine and one

electric pump. From a reliability viewpoint using a more reliable electric driven spare feedwater pump with the necessary pump and valve control changes could potentially reduce the number of challenges to the RPS from feedwater pump failures.

For feedwater pump trips due to upstream problems, deaerating feed tanks could reduce the number of feedpump protective trips due to the loss of net positive suction head to one or both pumps. This additional feedwater capacity would increase the steam generator dryout time on the loss of a condensate or booster pump and allow sufficient time to survive this type of transient.

Other recommendations to reduce outages include:

1. Modify the hydraulic control oil system separating the lubrication from hydraulic oil.
2. Increase the diameter of hydraulic orifices.
3. Consider changing to pneumatic control.
4. Increase inspection frequency of oil line hoses and or replace faulty hoses with stainless steel or better tubing.
5. Establish a vibration or accoustic monitoring program for earlier detection of bearing or blade problems.

A.2 Feedwater Valves

The Feedwater Control Valves have a very important function in maintaining a heat sink for the reactor core by maintaining water level in the steam generators. Failure or malfunction of these valves results in automatic scrams and challenges to the RPS and the backup after trip heat removal systems. Many Steam Generator Level trips at low powers and at other power levels were reported but with unreported causes. Failure to control these valves is considered the predominant of causes.

The valve configuration typically is a parallel combination of a Main Control Valve and a bypass valve controlling flow to a steam generator from a common feedpump discharging header. The main valve is controlled automatically from the Feedwater Control System or manually from the control board and other locations. The bypass valve is typically controlled manually from the control board and/or local control stations. For startup and low power operation, the bypass valve is used whereas the main control valve can only be used effectively at the higher power levels. The greater the number of steam generators, the greater the amount of attention an operator will have to give to level control at low power and during startup.

As stated in the previous discussion on feedpumps, the control valve trim should be matched to the type of feedpump. In addition, the valve stroke has a definite impact on level control and stability. The valve stroke should be as long as possible (approximately 90%) over the operating range of 0 to 100% power. The longer stroke allows better control, i.e., more stability, at lower power levels.

Steam Generator water shrink and swell have an destabilizing effect on steam generator level control, especially at low powers where it is important not to overfeed the steam generator. The level initially decreases, shrinks, with a delayed level increase resulting after the subcooled water is heated to a new equilibrium temperature. Overfeeding causes a level overshoot and instability which can get out of hand occasionally.

The design philosophy used for most feedwater control systems was that little time will be spent at low powers on base loaded plants and thus low power automatic control is of little need. The instability at low power requires additional operator attention and distracts the operator from other duties. Automatic control at low power would thus free the operator for better start up and low power operation.

A integral part of feedwater valve operation is the feedwater control system, the system where the valve control signal is derived. This control system, commonly referred to as three-mode control, in that the inputs are

Steam Generator level, Steam Flow and Feed Flow. The electronic control schemes used vary as the number of process equipment vendors, NSSS vendors and architect engineering firms vary. Moreover, many advances have occurred in control systems design over the past ten to fifteen years. With the design and construction lead times of plants and the fact that new plants are based on older plants, these advances were too late for incorporation in the base design. Advanced control system schemes would have required expensive engineering development time and thus the base designs were not changed. Moreover, emphasis has been put heavily on safety systems to the detriment of process systems. With the cost of replacement power increasing, modified control system designs would add to system efficiency as well as reliability establishing a much larger cost benefit. The control system design resulting from any additional development should be based on more accurate steam generator water inventory control.

The accuracy of process inputs to the control system are as important as the level control system scheme. The control system sensors used are all differential pressure transmitter type, indirect measuring devices, calibrated for an assumed fixed set of dependent parameters. Assumed are temperature, pressure and density, which all vary as a function of power. Additionally, dependent on the sensing methodology, outputs can be affected by sensor characteristics. With the advances in instrumentation reliability and technology, compensation of these sensors for more accurate process measurement, and therefore better control, can be accomplished.

Feedwater Valve problems were found to have caused 0.35 outage events per plant year and 0.16 percent annual plant unavailability at over fifty percent of the plants. Due to the number of valves, the many different specific type applications, the characteristic system factors, and the time and budget which could be allotted to a specific valve component study, it was decided to defer a valve mechanical component study and address the larger impacting control components of the feedwater valve.

Conclusions and Recommendations

Further feedwater control system development should be conducted to advance steam generator level controls to the state-of-the-art. The goals should be greater efficiency and reliability. Consideration should be given to providing redundant feedwater valves in parallel with existing valves for system reliability upon failure of mechanical valve components. Use of common support systems should also be minimized with redundant power or uninterruptible power sources provided. The study should also include valve application with different types of pump drivers as well as component failures.

A.3 Condenser Tube Leaks

At most US PWR generating stations, the main condenser is of the deaerating type. Condenser functions include deaerating as well as condensing. Condenser tube leaks resulted in thirty six (36) shutdowns, four (4) scrams and one hundred seventy-eight (178) power reductions. Condenser tube leaks can be very detrimental to other feedwater and condensate systems components and can be especially detrimental to steam generator tube integrity.

The C-E Data Base Main Condenser events were checked against the following: NRC Operating Units Status Reports, EPRI Limiting Factors Analysis for Maine Yankee, and EPRI Limiting Factors Analysis for Turkey Point 3 (see references for section 2). Events were checked for the years 1977 and 1978. Except for a few events, the information in the C-E Data Base was confirmed as accurate. However, the information given in the reports was insufficient to draw any root cause reasons for tube leakage.

Nuclear Power Experience (NPE) (reference 1) gave lengthy descriptions of condenser tube leaks and provided some possible causes and fixes. These causes are tube vibration, jet impingement, corrosion/erosion and impingement plate failure. Vibration has been handled by "staking" the tubes or lacing them together. Jet impingement has been dealt with by the replacement of impingement plates; impingement plate failure has been reduced by insuring better workmanship and adding stronger supports. Corrosion/erosion has

been dealt with by changing materials mainly from admiralty to Cu-Ni combinations, stainless steel, or titanium. Additionally tube cleaning systems have been added. While some of the above fixes worked out rather well, they are not the final resolution of the problem. Condenser tube leaks continue to occur and cause more down time each year.

Root causes of tube leakage are workmanship (these are non-code low quality class items), tube-to-tubesheet leaks, and transient related failures. It must also be noted that operating philosophy of condenser tube leaks differ. Some plants will continue to operate for a lengthy period of time after a tube leak has been detected and prior to searching out the leak and plugging it. Other plants upon the detection of a leak, immediately search out the leak and plug it. The latter is preferred since tube leakage causes steam generator out of chemistry limits and aggravates further tube leakage, including possible problems with other feedwater system components.

Many solids and organic material are transported from the cooling water source into the condenser water box and tubes. This results in tube blockage and corrosion due to inadequate tube water flow. This in turn results in tube leakage due to corrosion and thermal shocking, especially when the turbine is opened up for maintenance. These corrosion products eventually end up in the condenser hotwell and are transported into the feedwater cycle unless adequately cleaned prior to startup. Feedwater cycle problems and steam generator problems are in part attributable to this crud transport.

Cycle air removal is presently accomplished in most PWRs by the condenser. Unless properly operated, this function is degraded during the colder periods of the year. Also, environmental limits on allowable cooling water temperature rise across the condenser preclude efficient deaerator section performance.

Many valves, drains, expansion joints, etc. permit air inleakage to the vacuum portions of the condenser and condensate system. This affects both feedwater purity and cooling water supply.

Conclusions and Recommendations

Turbine condenser system problems include both tube leaks and auxiliary system problems which lead to tube leaks and power reductions. Tube leaks are of major importance because of their detrimental effects on steam generator tube integrity and their effect on the design and operation of the feedwater and condensate systems. The scope of recommendations in this section includes most considerations except condenser tube and tube sheet integrity which is the subject of an EPRI sponsored program.

a. Cooling Water System (Heat Sink)

A program is needed to improve the present methods of removing entrained solids before reaching the condenser water box and tubes. This includes better trash removal equipment and filters as necessary. An improved method of reducing organic material effects on tube blockage is needed. The later effort should be coordinated with current EPRI sponsored programs in this area.

A program to investigate corrosion product source, control and removal is needed to protect feedwater system components and steam generators. This should be coordinated with feedwater and condensate system investigations, especially the deaerator function during post shutdown cleanup operation.

b. Deaeration

An investigation of current condenser designs for deaerator performance should be conducted in coordination with the feedwater and condensate system recommendations for satisfactory deaeration of feedwater before high temperature feedwater and steam generator use. This needs to be evaluated with environmental regulations and operating practice for cooling water flow during colder periods of the year. The investigation should include evaluation of the use of a separate deaerating feed tank downstream of the condensate and booster pumps.

c. Air In-leakage

A comprehensive investigation should be made of leakage paths and air removal in the vacuum portions of the condenser, condensate, and cooling water systems. The condensate portion should be coordinated with the feedwater and condensate system investigation. The cooling water portion needs coordination with the circulation water investigation to improve heat transfer conditions.

A.4 Feedwater Heaters

Feedwater heaters are shell and tube type heat exchangers used to heat the feedwater in order to allow a smaller temperature differential across the steam generator tubes. Leaky feedwater heater tubes were attributed to 0.19 outage events per plant year and 0.04 percent average annual plant unavailability for the 192.4 years of PWR plant experience. These occurrences were reported at thirteen of forty plants.

Feedwater heater tube leaks occur either at the tube-to-tubesheet joints or due to failure of the tube itself, similar to main condenser problems. Degradation of joints may occur due to corrosion, erosion, tube vibration, thermal shock, chemistry changes or improper operation (see below).

1. Tube-to-Tubesheet Joint Leakage

Feedwater heater tubes are strength rolled into grooved tubesheet holes. The tubes are expanded to contact the tubesheet hole and the expansion force is further applied to seal the joint and to attain local deformation into the grooves for mechanical joint strength. Many variables affect the integrity of the joint. The tubes and holes must be kept clean and free from foreign materials especially expander lubricant. The fabrication procedures must consider the cumulative effects of tolerances and variations in material properties. The process controls must address variations in expander torques and speeds, tool alignment and acceptance standards. Workmanship must be of consistent high quality with care being taken not to fail materials during fabrication.

2. Tube Vibration

Vibration of tubes causes deterioration of tube-to-tubesheet joints, mechanical abrasion of tubes rubbing against baffle and support plates, and tube fatigue. Vibration may be due to various design considerations (high velocities, inadequate tube supports, pressure fluctuations, localized high velocity steam jets, etc.), unanticipated transients or improper operation such as exceeding design parameters. Resonant vibration is extremely detrimental to tube failure.

3. Thermal Shock

Tubes and tube-to-tubesheet joints can be overstressed if subjected to large or frequent thermal shocks. Also, differential expansion of the tube bundle can result in stress induced failures particularly if tubes have high residual stress levels from manufacturing operations.

4. Corrosion and Erosion

Contributing factors include tube entrance effects and local high velocities from manufacturing defects or deformations within the tubes, the presence of foreign materials and improper operation resulting from chemistry changes. High residual stress areas are more prone to attack.

5. Materials

In addition to material defects, material types must be selected which are compatible with the chemistry and environment the units will experience.

Conclusions and Recommendations

Resolution development for tube leaks must be handled on a case-by-case basis depending on the contributing factor or combination of factors involved. Tube failures within the same unit may not be due to identical causes and

detailed investigation into each tube failure is required to formulate a sound engineering solution. Obviously this is infeasible in most instances.

Proper planning can go a long way towards avoiding or decreasing the severity of tube leaks. The heat exchanger designer should be as well informed as possible of the required performance, transient environmental conditions and expected chemistry in order to select compatible materials and to design around potential problem areas. Adequate industry standards are the best assurance of competence in the fabrication of heat transfer equipment. Such standards would be the best approach to satisfactory performance through a consolidation of industry design, maintenance, and operational experience.

Design requirements should address impingement protection, maximum tube support spans, tube-to-tubesheet and tube-to-baffle clearances and baffle types. Purchase specification should require analysis for tube vibration and transient conditions. Additional requirements should be imposed on the designer and manufacturer to preclude some of the potential contributing factors.

Fabrication specifications should include requirements for tube-to-tubesheet joints to be seal welded and air leak tested, requiring seamless tubes, for stress relief of tubing, for restricting tubing heat treating to open annealing vice bright annealing for stainless steel and for approval and witness testing of tube-to-tubesheet expansion. And finally operator and maintenance personnel training and procedures should include instruction on the maintenance startup and operation of feedwater heater such that sudden thermal transients and crud can be avoided.

A.5 Feedwater Filters

Feedwater filters accounted for only six shutdowns of which three were automatic scrams. However ninety-eight power reductions were recorded which were attributed to cleaning these filters. All the above events were reported at nine of the forty PWR plants.

These power reductions are considered significant as an indicator of a systems design problem. The number of events indicates a need for additional evaluation of feedwater problems and the need for additional filter capacity.

The number of power reductions and other events indicate abnormal feedwater level and a lack of feedwater filter design capacity at the nine plants involved. Additionally, feedwater filter problems could have explained some of the feedwater pump trip problems on loss of net positive suction head.

Conclusion and Recommendation

Further evaluation of feedwater booster pump capacity for expected filter and crud buildup levels should be conducted as well as the cause for abnormal feedwater crud levels.

B. Turbine Generator System

The most significant FMC problems identified in the Turbine Generator system were those related to the turbine, the moisture separator and reheater, and the turbine control, stop and intercept valves. Each of these components is addressed separately in the following paragraphs.

B.1 Turbine

The 183 turbine generator events listed in this C-E PWR Shutdown Data Base were verified through review of information sources used in generating the Shutdown Data Base. Additional information was obtained through direct contact with the utilities and contacts with the turbine generator manufacturers. The major FMC problems relating to outages were:

- o blade failures
- o bearing failures
- o steam leaks
- o imbalance and vibration

Each of the problems is discussed below in order of impact.

a. Blade Failures

Turbine blade problems accounted for 1.6% average unavailability contribution per plant year, 0.04 scrams per plant year and 0.10 shutdowns per plant year at fifteen of forty plants. The average shutdown duration from blade problems was 1343.3 hours (56 days).

It was found that, although blade problems have been experienced in both low and high pressure turbines, most failures occur in the low pressure turbine. The root causes of these blade failures were: erosion due to moisture produced in the turbine, inadequate Moisture Separator Reheater (MSR) performance, reliability feedwater system corrosion products which are carried over, corrosion from improper feedwater chemistry control, underfrequency operation and overstress. Turbine blade erosion is a function of water droplet size, steam quality, steam density, rotating blade speed, and inherent turbine and blade design features. If the variables can be controlled to an acceptable level, blade erosion can be limited. Mineral deposits on the blades can be limited by feedwater chemistry control and functional MSRs. Excessive turbine and main steam valve testing can also cause temporary increases in the moisture content in steam and accelerate blade erosion.

Improvements which can reduce or eliminate low pressure turbine blade failures are:

- o Maintain feedwater chemistry within specification and provide for 100% demineralization of the condensate when required.
- o Improve MSR performance and reliability.
- o Limit low frequency operation.

- o Provide temperature, pressure and vibration monitoring of turbine and MSR for load equalization among LP turbines and turbine stages.
- o Limit turbine and main steam valve testing.

Many industry programs are presently underway to evaluate turbine blade problems. These include:

EPRI	RP637	Detection of Water Induction in Steam Turbines
EPRI	RP735	Condensation in Steam Turbines
EPRI	RP912	Corrosion Fatigue of Steam Turbine Blading Material in Operating Environments
EPRI	RP1185	Evaluation of Turbine Blade Root Designs to Minimize Fatigue Failures
EPRI	RP1264	Titanium L-1 Blade Retrofit at Kincaid Station
*EPRI	RP1408	Low-Pressure Turbine Coatings Resistant to Steam Borne Corrodents
*EPRI	RP1409	Corrosion Detector for Steam and Combustion Turbines
*EPRI	RP1068	Deposition of Salts from Steam
*EPRI	RP1124	Turbine Chemical Monitoring (Once thru steam generator design only)
EPRI	RP1398	Cracking of Shrunk-on Discs in Low Pressure Steam Turbines

*Chemistry and materials related.

These programs are involved with blade problems associated with moisture damage, chemical attack and stress problems. High pressure and low pressure turbine problems as related to saturated and superheated LWR turbine designs should be investigated in detail. The above listed industry programs should be supplemented with programs to:

- 1) improve moisture separator reheater performance (see Section B.2)
- 2) determine the effects of standard startup, shutdown, and transient procedures on moisture generation in the turbine and on moisture separator reheater performance
- 3) determine the effects of turbine valve testing on blade damage
- 4) determine the best MSR system configuration and control system for the moisture separator reheaters to obtain better low pressure turbine load equalization
- 5) determine adequate turbine and moisture separator reheater diagnostic monitoring needs for improved MSR performance, turbine load equalization, performance, and corrective action.

Turbine maintenance is time consuming as indicated by the average outage duration. Good planning, procedures, special tools, as well as timely instruction on the procedures, possibly through video tapes, could significantly shorten turbine related outages.

b. Bearing Failures

Turbine bearing failures, which occurred an average of 0.06 times per plant year at eleven plants, ranked next in total outage. The report causes of most bearing failures are:

- o Dirty oil
- o Blocked oil passages
- o Emulsification
- o Improper cooling of lube oil
- o Bearing vibration

These problems can be minimized by increasing the frequency of of the oil system to identify the onset of dirt or emulsification build up, and by routine inspection of temperature and pressure gauges in the oil system. Long term solutions include the use of larger orifices in the lubrication system and improved lube oil filtration. Early detection through vibration and accoustic monitoring techniques can enhance the liklihood of corrective action prior to bearing failure.

c. Steam Leaks

Turbine piping and casing steam leaks were reported at nineteen of forty plants causing an average outage frequency of 0.15 percent per plant year. Steam leak root causes include the following: erosion, impingement, uneven heating, and improper joint and gasket sealing. The frequency of steam leaks can be decreased by proper maintenance and by observing proper heat up rates. Minor steam leaks (early stages of large leaks) can be temporarily repaired using a sealant or patch and minimize the number of non-welded joints.

d. Vibration and Imbalance

Turbine generator balancing was reported to have caused nineteen shutdowns and four scrams at ten of forty PWR plants. A common cause was reported as vibration. Turbine vibration and imbalance are usually indications of other problems. Causes of turbine vibration and imbalance are: shaft and rotor distortion, uneven blade erosion, mineral deposits on blades, improper clearance tolerances, misalignment, internal mechanical looseness, foundation and case distortion, internal rubbing, uneven bearing wear and a bowed rotor.

These causes of turbine vibration are second order. Degradation causes imbalance, which leads to vibrations. Sometimes these secondary problems are hard to determine without dismantling and visually inspecting the turbine internals. Vibration and acoustic monitoring programs could enhance the capability of detecting premature secondary problems and their locations. This would enhance early investigation for root causes and corrective action prior to an unscheduled turbine shutdown.

Temperature and pressure diagnostic monitoring of the steam systems and turbine could also aid in root cause determination and corrective action prior to mature problems. Proximity detectors could also monitor turbine alignment. Diagnostic monitoring of the turbine and steam systems could further be used to monitor the progression or regression of problems once corrective action is taken and to predict when and what turbine maintenance is needed.

B.2 Moisture Separator Reheater

All light water reactor saturated steam turbine systems are designed with Moisture Separator Reheaters (MSR) between the high pressure turbine and the low pressure turbines. Their purpose is to extract moisture from the steam exiting from the high pressure turbine and then reheat the steam to minimize condensation and moisture generation in the low pressure turbine stages. The result is an increase in the low pressure turbine performance, thermodynamic efficiency, and decreased low pressure turbine blade damage.

The C-E Shutdown Data Base shows fifty-one turbine shutdowns of which nineteen events were scrams due to MSR problems. These problems were reported at fifty percent of the forty PWR plants.

Forty-two of these fifty-one events were researched in this study. These events were verified through data analysis and direct contact with the utilities. Over seventy percent of the outages were caused by tube leakage. The remaining outages were caused by problems with baffle plates, demisters, separators, handholes and high water level trips.

The root causes of tube leakage are similar to other heat exchanger problems, thermal fatigue, vibration, corrosion and erosions. Factors which contribute to thermal fatigue, vibration erosion and corrosion are:

- o crud deposition on interior and exterior reheater tube surfaces as well as separator surfaces
- o inadequate tube bundle drainage
- o condensation in tubes
- o condensate subcooling of tubes
- o temperature gradients between tubes and tubesheet
- o non-uniform steam flow
- o control and intercept valve testing
- o vibration overstress
- o use of materials not compatible with the service
- o feedwater chemistry control problems

Because of a limited amount of large saturated steam turbine experience, there is a lack of prior experience with these devices in this application. Their performance record has been poor. Turbine valve testing requirements and inadequate drain and control system design interfaces have aggravated their performance resulting in thermal shocks, vibration and fatigue.

Operation and maintenance improvements as well as the design of systems interfacing with the MSR need to be investigated and implemented to improve the performance of existing installations and limit thermal shocks and fatigue.

It is recommended that further detailed study at operating power plants be conducted to reduce tube leaks and increase MSR performance, thus reducing turbine blade problems. It is believed the more immediate benefits can be achieved from the following evaluation areas of interfacing systems:

- o evaluate shell and tube side drains and level control for adequacy and improvements

- o evaluate improved reheat temperature monitoring and controls to limit thermal shocks and condensation subcooling in tubes
- o evaluate operation, maintenance and testing practices for improvements to limit thermal shock
- o evaluate outlet temperature controls for improved turbine loading and warmup.

Longer term evaluation and development areas are as follows:

- o investigate and characterize moisture carryover from separator to reheater section
- o investigate and recommend improvements in steam inlet header design
- o investigate and characterize outlet steam temperature and moisture distribution
- o investigate and recommend improvements to reduce steam bypass flow around separators and reheater tube bundles
- o investigate and characterize tube failures and tube to tubesheet joint failures
- o investigate and characterize stress levels where failures are occurring
- o investigate and recommend improved tube and shell materials to reduce tube failure and shell corrosion
- o investigate and recommend improved MSR configuration to improve reheat performance

- o investigate and recommend improved designs for backfit to operating plants
- o investigate and recommend an optimum MSR design to reduce space requirements and improve thermal performance
- o investigate and recommend optimum interface criteria between MSR and feedwater cycle including steam supply source
- o investigate desirability of higher ASME vessel code application to improve integrity

B.3 Turbine Control, Stop and Intercept Valves

The critical requirements of turbine valves to interrupt large steam flows in a fail safe manner puts severe requirements on these valves and their associated control system. Malfunction of these valves will cause turbine overspeed with potentially large economic consequences, and reactor trips upon spurious closure.

The shutdown data base contains a significant number of failures in the Turbine Control System, which result in stop and intercept valves closing, causing plant scrams, shutdowns, and power reductions. A summary list of turbine valve occurrence categories is provided below with the number of events and the fraction of plants affected.

	<u>EVENT OCCURRENCES*</u>			<u>F**</u>
	<u>SCRAMS</u>	<u>SHUTDOWNS</u>	<u>POWER REDUCTIONS</u>	
Miscellaneous Electro Hydraulic Control (EHC) Problems	42	56	5	25/40
EHC Auto Stop Oil and Fluid System Malfunction	34	60	4	25/40
Turbine Protective System Malfunction	26	32	0	21/40
Turbine Control Stop Valve Malfunction or Maintenance	16	46	10	21/40
Spurious Turbine Control Valve Closure	14	18	4	9/40
Spurious Turbine Control Valve Opening	6	6	1	5/40
Turbine Valve Tests	0	0	108	19/40
TOTALS	<u>138</u>	<u>218</u>	<u>132</u>	

*Numbers of PWR occurrences to December 1978

**Fraction of PWR plants affected

The above data indicates that the valve control system and turbine protective system problems result in the greater number of scrams and outages. Approximately 70 percent resulted in automatic scrams. Moreover, the data indicates a large disparity of plant operating or reporting procedures - nineteen of forty plants reported reduced power during turbine valve testing. This point was not pursued in this study.

Data for 47 potentially EHC related outages in 20 stations was verified using the EPRI Limiting Factors Analysis for Maine Yankee and Turkey Point, Nuclear Plant Reliability Data System, License Event Reports, Utility Annual Operating Reports, Operating Experience Status Reports, Experience Records for PWR's, visits to stations, and communications with stations. Results of this sample investigation follow.

a) Outages Due to Valve Malfunctions

Twenty-three outages of this type were investigated. The problems include valves not opening, valves not closing, valves sticking, valve failures, and spurious valve closures. Although more specific information could not be obtained, problems with valve controllers and general operation are suspected based on valve experience.

b) Outages Due to Repair or Replacement

Sixteen outages of this type were investigated. Outages were due to leaks and include some stem and disc replacements. Causes can be attributed to valve wear, some beyond repair.

c) Outages Due to Valve Modifications and Miscellaneous Reasons

Eight outages of this type were investigated. Two outages were due to modifications required by the manufacturer. The six remaining outages were due to auxiliary equipment such as servomotor and relay failure.

From the analysis of forty-seven outages due to turbine valves, thirty-nine outages (83%) can be related to operation and maintenance activities.

Valve control system problems were also found to be a major cause of the 23 outages due to malfunctions in the control, stop and intercept valves, not "direct" valve problems. All of these control system failures, which result in spurious valve closure, result in direct turbine and reactor trip. When investigated from a control system viewpoint, thirty-three (33) trips were caused by EHC failures, twenty-nine (29) trips occurred during testing (also EHC related), and ten (10) trips caused by governor intercept valve control problems.

Conclusion and Recommendations

It can be concluded that system design is the area most needing attention in the turbine control, stop and intercept valve category of challenges to the RPS. Better systems design for reliable control during expected plant

operations, and testing requirements, is sorely needed. Design for base load conditions alone places a severe restriction on other testing and operating requirements, especially the operations getting to and from base loaded conditions: startup, warmup, cooldown, and shutdowns. The present designs are extrapolative of older fossil turbine designs and do not reflect the additional requirements for technical specification required testing and the different characteristics of the nuclear steam supply conditions for PWR turbines.

Meanwhile, operating, testing and maintenance procedures should be reviewed for improvements as well as the required testing frequency. Control system and protective system design should be evaluated for short term improvements for the testing requirements. Valve design for frequent testing should be investigated.

General recommendations for improved operation and design development for a reduction in turbine valve related challenges are as follows:

- o An evaluation of operating, maintenance and testing procedures should be made to determine what existing procedures should be improved.
- o An evaluation of turbine valve design for frequent testing should be conducted to determine the impact of turbine testing frequency and to evaluate potential improvement.
- o The types of valves used by turbine manufacturers should be examined for valve application compatibility; valve design, materials quality control and design test frequency.
- o The testing requirements of these valves should be evaluated based on the type of valve control system used and the failure frequency.

- o A comprehensive design review should be conducted to improve the control circuits and hydraulic control systems for these valves. This needs to be coordinated with valve testing for operability and safety actuation as well as interfacing systems and components supported.

C. Control Rod System

Three major categories of FMC related problems resulted from the review of the C-E Shutdown Data Base. The categories are:

- o CRDM housing seal failures
- o CRDM motor stator failures
- o rod drops with unidentifiable causes.

Sixty-seven outage events resulted from the above categories of failures and contributed 0.37% to average annual unavailability. Only about twelve percent of the above outages were the result of automatic scrams. Significant numbers of power reductions also resulted from these categories of events. Further, seventy-six percent of the plant unavailability reported was experienced at only five of the forty plants.

C.1 CRDM Housing Seal Failures

Control Rod Drive Mechanism (CRDM) housing seal failures were responsible for twenty-nine (43 percent) of the CRS outage events. All events were experienced at one plant.

The main cause of these failures was boric acid attack on a nickel bond which was utilized to interface the aluminum oxide seal face to the stainless steel seat housing. Manufacturing methods and installation procedures also contributed to the seal failures. The seals were redesigned and the nickel bond eliminated. Since installation of modified seals in December 1978, no additional failures have occurred. The problem is presently considered resolved.

C.2 CRDM Motor Stator Failures

CRDM motor stator failures were responsible for sixteen (24 percent) of the CRS outage events. These events were reported at five plants where roller nut type mechanisms were used.

Many factors can contribute to motor stator failures. Some of these include, motor overheating, insulation degradation, and manufacturing techniques. Motor overheating can be caused by restricted coolant flow passages, insufficient air flow around CRDMs, shorted windings, and excessive stator current.

Individual CRDM failures are considered to result from any one or a combination of the above. Considering the possible failure modes, the number of CRDMs at the five plants involved, and the relatively small number of failures of this type, further evaluation in this program was not expected to produce a common cause where resolution would provide beneficial results. Continued replacement of failed stators appears to be the best alternative to expensive troubleshooting program at this time.

C.3 Rod Drops with Undetermined Causes

Twenty-two rod drops were reported at seventeen plants for which information on causes could not be obtained. This represents approximately 20% of the total number of rod drops. Over fifty percent of rod drops were related to faults in the rod drive control system. Rod drops with undetermined causes are believed to be caused by spurious instrumentation problems or noise. More detailed discussions at many plants would be required to investigate rod drop events further. Such an investigation is expected to lead to faults common to the rod drop events for which causes were determined, i.e., approximately seventy percent rod control system related and the remainder to miscellaneous CRDM problems. The fact that most rod drop events occur while the rod is being exercised reinforces this expectation. Further investigation in this program was not felt to be justified.

Conclusion

The number of failures in the Control Rod system appears to be related to the number of components in the system. Considering that the number of control rods per plant is fifty to one hundred; the number of failures is relatively small. The seal failures at the one plant is considered resolved and further evaluation of other generic failure modes does not appear prudent based on this investigation.

D. Main Steam System

The major FMC problems in the Main Steam system were determined to be Steam Generator tube leaks, malfunction and closure of Main Steam Isolation Valves (MSIV) and component supports. Besides causing a challenge to the RPS or to the plant operator to action, a Steam Generator tube leak is a break in the integrity of the reactor coolant pressure boundary, a cause of additional safety concern. The primary safety function of the MSIV is to isolate a ruptured steam generator upon a steam generator tube leak. Malfunction and closing of a MSIV will result in an unplanned partial loss of the heat sink for the nuclear reactor.

The results of the FMC investigation of these FMC components is discussed below.

D.1 Steam Generator Tube Leaks

Steam Generator tube leaks were reported to have caused approximately 80 shutdowns at 19 of 40 PWR plants. The majority of plant unavailability in the period 1972-1978 attributed to steam generators (other than that due to routine inspection and maintenance) has been related to the heat transfer tube bundle and its supports. In general, only leakage of tubes directly causes plant shutdown. Other unavailability attributed to steam generators relates to extension of inspection and maintenance shutdowns or more frequent shutdowns for inspection. Measures to prevent corrosion, such as feed system modifications, or expedient repair of condenser leaks, may cause shutdown or reductions in power. The incidence of tube leaks is minor in

comparison to the total number of degraded tubes taken out of service (plugged) during in-service inspection shutdowns. Tube bundle problems which have been reported are categorized below according to the type of secondary chemistry control in effect at the time.

1. Coordinated Phosphate Chemistry

- (a) Tubewall thinning due to fretting wear against upper antivibration spacer strips.
- (b) Intergranular stress corrosion cracking of tubes in sludge piles above tubesheet.
- (c) Intergranular attack of tubes in one plant.
- (d) Tubewall thinning ("wasteage"). This has not been reported in any plants which have operated solely with volatile chemistry.

2. Volatile Chemistry

Reported tube bundle problems in U-tube steam generators appear to be related to denting. Damage related to denting have varied with differences in steam generator design but has included, at one or more plants:

- (a) Distortion of support plates including "hourglassing" of central flow slots, ligament cracking and breaking away of "islands" of support plate material.
- (b) Stress corrosion cracking (leakage), originating at the inside diameter, at the top of the bends in the central rows of tubing, and in severe dents.
- (c) Tube denting (no leaks) in tubesheet crevices in sludge piles.

- (d) Cracking (leakage) in peripheral tubes a few inches above the tubesheet.
- (e) Dent-like eddy current signals at some "egg crate" supports and just above the tubesheet.

A small number of plants, eight in all, in operation by the end of 1978, have Once Thru Steam Generators (OTSG) (non-recirculating) generating superheated steam. These have straight tubes expanded into upper and lower tubesheets, and one row of tubes along a radius to the center of the generator has been left out to form an inspection lane. The tube bundle has full diameter plate tube supports with broached holes for the tubes. The broached holes have three equally spaced grooves in them for flow passages. All plants have full flow condensate polishers in their feed systems and utilize volatile boiler water chemistry. OTSG tube bundle defects have been different from that for U-tube steam generators, as noted:

- (a) Through-wall transgranular fatigue related circumferential cracking and fractures (leakage) in tubes between upper tubesheet and the highest support plate, with most leaks occurring in inspection lane tubes.
- (b) Serpentine depression grooves in affiliation with micro-cracks in tubes near upper tubesheet. Suspected to be under-deposit corrosion.
- (c) Erosion grooves, fretting wear and preferential surface pitting in tubes at upper support plate.
- (d) Impingement type surface damage on tubes at the next lower support plate. Thought to be due to corrosion particle impingement.

Conclusions and Recommendations

The following conclusions and recommendations apply to UTSG except as noted.

Tube wall thinning involving phosphates appears to have been eliminated by changing to volatile chemistry. Vibration assisted tube cracking in OTSG's has been reduced by operational changes including a revised method of turbine stop valve testing, auxiliary feedwater system operation, rerouting of heater drains formerly bypassing the condensate demineralizer, and revised steam generator lay up procedures. Changes in operations and equipment changes have also reduced the rate of progression of denting.

To solve the steam generator problems with operating plants, including OTSG's, extensive development programs are being performed by the Electric Power Research Institute, PWR Steam Generator Owners Group, NSSS Vendors and other organizations such as Battelle Columbus and Nuclear Waste and Water Technology. In addition, the NRC is sponsoring an intensive post-mortem examination of one of the Surry 2 steam generators which has been replaced. Development programs also are directed towards the steam generators not yet in service and to potential future steam generator improvements.

The development programs referred to in the above paragraph are so comprehensive that it is difficult to identify further needs for which potential practical solutions can be offered. A method of detecting the onset of tube/support crevice blockage (the precursor of denting) while the plant is in operation would be invaluable since chemical cleaning can be performed most effectively before denting occurs. Detection methods presently under development could only be used during shutdown, which normally occurs at yearly intervals (18 months for the potentially critical first fuel cycle). Possibly some type of noise monitor mounted external to the steam generator could be employed but this has not been seriously studied.

As pointed out, much of the plant unavailability attributed to steam generators involves the extension of planned inspection and maintenance shutdown periods. Hence, unavailability can be reduced by improvement in inspection and repair equipment and procedures as well as by steam generator modifications. In particular, during refueling shutdowns, the use of isolation valves or

barrier dams in the primary coolant piping could permit steam generator inspection and repair operations to proceed in parallel with refueling operations, saving several days, or in some cases weeks of downtime.

D.2 Main Steam Isolation Valves

The component that has historically caused the highest number of shutdown outages per plant year in the main steam system is the main steam isolation valve. These valves, according to the shutdown data base, are responsible for 0.12% unavailability for 0.27 outage events per plant year and for 0.17 scrams per plant year. Verification of the events was performed as follows:

- o Twelve events were verified from information in Nuclear Power Experience (Reference 1).
- o Forty-one events were verified by direct contact with cognizant operating plant personnel.

MSIV's are used primarily to isolate the steam generators from the turbine in the event of a fault in the main steam piping. They can also be used to isolate the turbine for maintenance while the plant remains at hot standby or to isolate a steam generator tube leak.

Early MSIV designs utilized back-to-back check valves, the valve closest to the steam generator being a stop-check valve held open with a pneumatic piston. This design had been successfully used before in fossil boiler systems as a turbine stop valve. The usual material of construction is carbon steel with valve sizes ranging up to forty inches. In more recent plant designs due to the critical aspect of the closing requirement, a stored energy system is used for valve closure. Current MSIV designs utilize various body styles and motive power sources. These include:

- o Hydraulic and gas actuated gate and globe valves
- o Air and spring actuated globe valves

- o Hydraulic and spring actuated ball valves
- o Main steam actuated gate and globe valves
- o Motor operated globe valves

Various operability test programs have been conducted to assure the capability to meet design specification operational requirements. This testing has included environment, seismic, pipe end loads, and mass flow interruption parameters. Inservice testing (dip tests) are performed as often as once a week to assure readiness and inservice capability to operate. When the valve receives a trip signal, it must close within three to five seconds. The inservice tests are usually performed at full power or only slightly reduced power during which the valves are dipped into the flow path approximately five to ten percent.

Typically, MSIV's receive an automatic closure signal based on low steam generator pressure or high containment pressure. They may also be operated by the plant operator from the control room or by hand station at the valve installation in the auxiliary building.

Overall, verification covered 80 percent of the significant events contained in the data base. The unavailability calculated ranks MSIV's at eight out of the fifteen highest component contributors to plant unavailability. Nearly 65 percent of the significant MSIV outage events referenced in the data base resulted as automatic scrams. Auto scrams are caused by inadvertent closures which lead to steam generator high pressure trips. Auto scrams account for only one-half of the overall MSIV related unavailability hours, thus the data supports the premise that inadvertent closures rather than repairs are the most important aspect of MSIV unavailability to address. Specific details of the occurrences reviewed are contained in appendix 3.1-A. The MSIV outage events can be grouped into five categories:

- o Inadvertent closures
- o Actuator failures
- o Materials failures

- o Pilot solenoid problems
- o Manufacturing errors

Conclusions and Recommendations

The C-E PWR Shutdown Histories Data shows that 73 percent of the MSI^V problem events involved back-to-back check valve designs which were supplied by two manufacturers. On new plants where gate and globe type valves are being utilized, the available data did not support further investigation.

Material failures have been principally in the rockshaft and key areas. These material failures (surface cracks, pitting and erosion of the rockshafts) have been corrected by material substitutions and better material heat treatment. One vendor has changed from a key joint design to a splined design.

The safety aspects of design and operation are being addressed by the vendors and utilities to meet the NRC guillotine break criteria. Older valves have been reanalyzed for higher loads and upgraded where necessary. Operability test programs are continuing on a case-by-case basis.

Inadvertent closures in back-to-back designs have been primarily attributable to air failures, disc's hanging into flowstream, weak actuator piston operators, and solenoid failures. These problems have been corrected by the following methods, which should be implemented by other plants where they are not yet implemented. Air volume tanks or accumulators have been installed to insure air supply when plant air is interrupted. Improved solenoid coils and redesigned solenoids have been backfitted. Air control systems were improved with more redundancy and greater reliability. Disc up-travel stops have been reworked to raise the disc higher out of the steam flow. Larger piston actuators have been installed to provide stability against flutter and to hold the disc against dipping.

The back-to-back design is no longer offered for MSIV service. A single valve, having bidirectional shut-off capability, such as a gate or globe, is cheaper and more reliable than the back-to-back designs.

In summary, MSIV failures appear to have been addressed and corrective action taken. Further evaluation should be conducted to further improve valve and control system reliability to eliminate all spurious MSIV closures.

D.3 Component Supports

The component supports, which contributed to 12 outage related failures and to a ranking in the top ten FMC components causing plant unavailability, were hydraulic shock suppressors, more commonly referred to as snubbers. Problems associated with mechanical snubbers have not generally been the cause of plant shutdown.

Information sources were examined which documented snubber failures leading to forced outages. The C-E shutdown data base, licensing event reports, and utility annual operating reports were utilized to quantify the incidents and verify the causes and effects of the failure.

The basic cause for the incidents is the loss of hydraulic fluid which leads to a loss of function of safety related snubbers. The loss of hydraulic fluid can be directly attributed to seal failures or potential seal failure on 100% of the recorded incidents. The causes of the seal failures are:

1. Broken accumulator springs which have caused seal surface scoring and subsequent leak paths (46.6% of incidents).
2. Seal material incompatibility (26.6%) which lead to adverse reactions to silicon hydraulic fluids leading to seal degradation and leakage.
3. Improper installation of snubbers during which seals were damaged (26.8%) leading to the remaining seal leakage problems.

Conclusions and Recommendations

Replacing older snubber seal material compounds with new materials of significantly longer life and replacing faulty springs with a newer design is being implemented. The installation problems are unfortunately continuing. It may be solved by better inspections after installation, strict adherence to manufacturer installation procedures and more frequent utilization of manufacturer's service personnel to supervise installation and periodic maintenance.

Specifications should define requirements to insure seal material compatability and life. Functional testing should be developed to simulate normal operating conditions over plant lifetime as well as seismic loading conditions. Installation and maintenance procedures for snubbers should be reviewed and evaluated to avoid mis-assembly and improper operation.

E. REACTOR COOLANT SYSTEM

Major FMC problems in the Reactor Coolant System that have made a significant contribution to the 1.67% average annual plant unavailability and 0.19 scrams per plant year are the Reactor Coolant Pump (RCP), the pressurizer spray valves and other miscellaneous valves. Most RCP outage contributions were due primarily to seal failures. Pressurizer spray valve related outages were caused by packing leakage.

When combined into a general category of miscellaneous valve leakage, this item becomes comparable to other categories. A general valve study would appear appropriate, however, as part of this study it would be ineffective in producing tangible results. Further study is recommended in specific follow-on programs. Results of the investigation are provided below.

E.1 Reactor Coolant Pump

The major recurrent FMC problems with the RCP were related to seal failures. Seal failures were reported to have caused thirty-six outages, of which six resulted in scrams, at seventeen out of the forty plants. Further, these

occurrences accounted for an average plant unavailability of greater than 0.5%, mostly due to maintenance time required. Outages were noted to occur frequently after very short periods of test operation or in-service operation.

The RCP seals along with the RCP seal injection and bleed off system serve to establish a reactor coolant pressure boundary and collect normal seal bleedoff at the penetration of the RCP shaft to the Reactor Coolant System (RCS). Several stages are used to reduce pressure from the normal operating RCS pressure of 2200 to 2300 psia to atmospheric pressure.

The nature of dynamic mechanical seals is such that they are sensitive to thermal gradients, solid particles in the seal fluid and to any small degree of misalignment. Inspection of seal assemblies that were replaced in the field, as a result of excessive leakage, has been identified cracked or deformed seal faces or damaged elastomer seals. Additionally, seal failures have occurred after short test operation or inservice operation after pump seal maintenance or plant shutdowns.

A report prepared by Commonwealth Edison Company and presented at the EPRI/DOE Workshops on Availability Engineering and Productivity Improvement contained a detailed methodology for "identifying and correcting root causes of outage problems". The report is titled "Reactor Coolant Pump Seal Investivation" (reference 2) and contains action items resulting from the investigation. The resulting major action items were stated as follows:

- "1. Revising Station procedures to eliminate seal failures caused by depressurization of the reactor coolant system at cold shutdown conditions with the pressurizer solid.
2. Reducing seal maintenance and inspection outage time associated with draining, filling, and venting reactor coolant system loops.
3. Decreasing frequency of scheduled seal inspections.

4. Expediting procurement and installation of ultrasonic flowmeters for measuring seal leakoff flow and reducing startup delays.
5. Reducing introduction of crud into the seal faces during reactor coolant system level changes.
6. Reviewing selected seal design features and inspection procedures and criteria with Westinghouse to improve Station procedures."

First hand maintenance experience suggests that more attention during the design phase should be given to maintenance support activities as well as the maintenance and installation procedures. Platforms to allow easier maintenance would also be beneficial as well as other work space considerations.

Conclusions and Recommendations

From the above discussion, it can be seen that preventing seal failures can best be accomplished through improved system operating procedures and inspection, installation, and maintenance procedures. Further, on the spot preparatory training on the procedures, disassembly, reassembly or installation techniques would help ensure proper alignment of RCP seals. This training could be through lectures, video tape techniques and rehearsal on a spare seal.

Additionally, consideration should be given to providing permanent maintenance platforms and equipment to aid in seal maintenance and repair. It is believed that this also would aid in the proper installation and alignment of seals. In turn, seal failures would be reduced as well as the impact on plant availability.

The causes of seal failure at other plants should be compared with the experience of Commonwealth Edison Company. In particular, the operating procedures should be reviewed as well as seal design, installation and maintenance for comparison to these features of the Commonwealth Edison Study.

E.2 Pressurizer Spray Control Valves

Pressurizer spray control valves operate to spray the steam bubble in the pressurizer in order to maintain RCS operating pressure. A certain amount of spray is desirable at all times to limit thermal shocking of the pressurizer spray nozzle. In parallel with the continuous spray valve, a modulating valve is used for the additional amounts of spray that may be required.

The C-E PWR shutdown histories data base indicates that Pressurizer Spray Control Valves have contributed approximately 0.11 percent to PWR unavailability through December, 1978. Through that date, there were an average of 0.14 outage events per plant year. This ranks the pressurizer spray valves at twelve out of the top fifteen highest fluid and mechanical component contributors to average annual unavailability. Approximately fifty-eight percent of the events recorded were investigated in this program.

Although air operated globe valves have been the most popular for this service, other types are in use. Valve types evaluated during this study were:

1. conventional packed pneumatic operated globe valves
2. pneumatic operated globe valves with bellows stem seal and backup packing
3. pneumatic operated ball valves
4. solenoid operated valves

Outage events were grouped into three categories:

1. Packing/bellows failures - 45% of the total outage time
2. Repairs - 30.4% of the total outage time

3. Excessive seat leakage - 15% of the total outage time

Packing and bellows failures cause leakage from the reactor coolant pressure boundary whereas excessive seat leakage requires additional pressurizer heater load to compensate for decreasing pressure caused by excessive spray.

A majority of the events investigated were the result of packing leaks and bellows failure with few problems due to seat leakage. Summary discussions with plants on these valve problems are contained in 3.1-B.

Conclusions and Recommendations

Solenoid type spray control valves are the best zero leakage valve in this application. Solenoids have some basic problem areas however. Solenoids burn out, stick, and have limited life. Some Vendor qualification has been done at 100,000 cycles, 600°F and 2500 psi in borated water. Effective application would depend on an individual plant's spray flow demand cycle. Operation is typically on-off, however a new modulating solenoid has been developed and is being qualified. It is intended that this valve will be used as a by-pass of the larger on-off (normally open) solenoid spray valve.

Valves with bellows stem seals can eliminate packing leakage problems. However, bellows seal life is cycle dependent. Valve vendors appear unwilling to warrantee more than 5,000 to 10,000 cycles. Bellows assemblies are delicate and require care in handling and assembly. Additionally manufacturing lead time is often extensive. A preventative maintenance change out program based on cycle life may be excessive.

Ball valves provide good flow control characteristics and are less likely to wear out packing due to their rotary motion. However, maintenance actions may be more difficult to perform compared to similar maintenance on globe valves.

Conventional packed globe valve packing leakage is best minimized by utilization of various Grafoil type packing combinations. Stuffing box designs could be improved to facilitate repacking. The main complaints are that there is not enough room between the stuffing box and the actuator yoke to easily do the work.

Access to the spray valves for maintenance personnel observation and packing adjustment was difficult. In some cases, it is easier to disassemble the valve and repack it at another location rather than work in the typically hot and restrictive areas where the valves are often located. Hot ambient conditions are also detrimental to the lifetime of electrical actuation components and solenoids as pointed out previously.

Relocation of these valves is recommended as well as the continued development and qualification of better solenoids and other types of valves for this application. Redundant valves with double isolation and a balanced inservice program would allow extended operation on failure of one of these valves.

E.3 Miscellaneous RCS Valve Leakage

As a separate category of events miscellaneous RCS valve leakage can be considered equivalent to other categories of events. The most common valve application reported was the reseating and leakage of pressurizer safety reliefs. However, these valves were a small portion of the problem. A general valve study was not felt efficient for this program.

In general further development of a better zero leakage valve for general application is desirable and would be beneficial for new plants and replacement of valves at other plants if replacement becomes necessary.

3.1.4 Fluid and Mechanical Component Summary Conclusions

The general conclusion of this portion of this program is that the greatest reduction in challenges to the RPS can come about through the improved systems design of process systems for better redundancy, capacity and

reliability. The process systems most needing improvements are the Feedwater and Condensate System and the Turbine Generator System. These two systems ranked highest in the number of challenges to the RPS and also rated high in causing plant unavailability.

FMC systems evaluated in this portion of the study were as follows:

- o Feedwater and Condensate System
- o Turbine Generator System
- o Control Rod System
- o Main Steam System
- o Reactor Coolant System

Feedwater and Condensate System

FMC components evaluated from the Main Feedwater and Condensate system included feedwater pumps, feedwater valves, condenser, feedwater heaters and feedwater filters. Feedwater pump and valve failures caused the greatest number of challenges to the RPS.

Feedwater pump problems were numerous and varied as well as the root causes for those problems. The addition of a installed standby pump was felt to be the solution with the greatest potential for a reduction in challenges to the RPS and maintenance of an adequate heat sink. The major problem area was found to be in the hydraulic and lubrication oil systems. Additional recommendations for pump reliability are also given.

The major problems with the feedwater valves were related to the control scheme and sensor input technology. Use of advanced control technology developed during the last fifteen years can improve steam generator level control with proper application. Additionally, advanced control systems can ease the burden on the operator. The use of redundant control systems and valves is also recommended.

Turbine Generator System

Major FMC Turbine Generator system component problems resulted from a variety of root causes. Turbine blade failures were related to moisture separator reheater operation, turbine load balance or equalization, and turbine valve testing and malfunction. MSR problems in turn were also caused by turbine valve testing plus operating procedures and typical heat exchanger type tube problems. Turbine valve control problems were the result of the electro-hydraulic control (EHC) system problems.

Recommendations for improvement for this system range from improved operating and heatup procedures, improved EHC system design for redundancy and reliability, improved MSR design, and improved diagnostic monitoring to allow load equalization, efficiency, and reduced turbine damage.

Control Rod System

Problems within this system appear to be well in hand. The CRDM seal failures at one plant seem to be resolved. Motor stator failures were isolated to a few plants and occurred infrequently compared to the numbers of these CRDMS. The frequency of rod drops is expected to decrease due to improvements already implemented and evolutionary improvement in the control systems design for newer plants. Overall, the impact from this system is expected to decrease.

Main Steam System

Major FMC problems within the Main Steam system were steam generator tube leaks, malfunction of main steam isolation valves and component support seal failure. Industry programs appear adequate in addressing steam generator tube leak problems. Further programs were not recommended. To reduce the outage duration, improved equipment and procedures as well as training would be beneficial.

Inadvertant closure of MSIVs were determined to be related to the back-to-back valve designs and attributed to the support components and systems failure. Improved valve actuator and control system design is needed to further improve MSIV reliability and availability. Snubber failures were traced to broken seals and faulty springs. Installation and maintenance procedures as well as better specifications and qualification techniques are needed to improve the components.

Reactor Coolant System

Major component problems in the Reactor Coolant System were determined to be Reactor Coolant Pump seals, Pressurizer spray valves and other miscellaneous valve leakage. Improved operating procedures and precautionary measures are recommended to prevent RCP seal failures. Improved training in the installation and maintenance area also appears to be beneficial.

Pressurizer spray control valve stem leakage was a source of problems in the RCS System. Valve maintenance was generally found to be difficult due to the location of these valves and also contributed to failures. Easier access to these valves is recommended as well the specification of zero leakage valves. Solenoid operated valves were considered to have the greatest potential for zero leakage.

Other miscellaneous valve leakage was also noted. This indicates a need for further development of a good zero leakage general purpose valve for nuclear service to further reduce coolant leakage.

General

As evidenced by the fraction of plants affected by different failures, some systems and equipment at some plants have better performance records than similar designs at other plants. Additionally, operation and maintenance at some plants is better than at other plants. A better way can almost

always be found to improve performance from different viewpoints, i.e. safety, availability, and reliability. Much can be learned by studying the characteristics at one plant to improve performance at others.

Determining the areas of greatest impact needing solutions allows concentration for the greatest potential improvement. Design for system reliability and availability as well as the design to facilitate operator and maintenance personnel is important. Process systems (vs safety systems) components caused the greatest number of challenges to the RPS. Improving the process systems has much potential for reducing challenges to the RPS and toward reducing the need for backup safety systems. Improved process reliability and performance can also help to mitigate events. Because improved process systems can reduce plant shutdowns and the need for more costly replacement power, a direct cost benefit can be obtained through increased plant availability.

Table 3.1-1
Summary of Major FMC Failures
(Ranked by Outage Events)

<u>Component</u>	<u>Outage Events⁽¹⁾</u>		<u>Average Annual⁽²⁾</u>
	<u>Per Plant Year</u>		<u>Unavailability</u>
1. Turbines	1.35		2.48
2. Steam Generators	0.48		2.25
3. Feedwater Pumps	0.44		0.08
4. Reactor Coolant Pumps	0.36		1.01
5. Control Rod Drive Mechanisms	0.35		0.37
6. Feedwater Valves	0.35		0.16
7. Main Steam Isolation Valves	0.27		0.12
8. Main Condensers	0.25	1.16 ⁽³⁾	0.35
9. Turbine Control, Stop and Intercept Valves	0.22	0.54 ⁽⁴⁾	0.12
10. Feedwater Heaters	0.19		0.04
11. Moisture Separator Reheaters	0.18		0.16
12. Pressurizer and Main Steam Safety & Power Operated Relief Valves	0.15		0.11
13. Pressurizer Spray Valve	0.14		0.11
14. Component Supports	0.06		0.12
15. Feedwater Filters	0.56 ⁽⁵⁾		

(1) Based on outage events per 192.4 plant years, power reductions excluded.

(2) Based on outage hours per the 1,686,980 possible operating hours in 192.4 plant years, power reductions excluded.

(3) Reported power reductions - 224 events.

(4) Reported power reductions - 106 events.

(5) Reported power reductions - 108 events.

Table 3.1-2
Summary of Major Component Failures
(Ranked by Contribution to Unavailability)

<u>Component</u>	<u>Percent Unavailability⁽¹⁾</u>	<u>Outage Events⁽²⁾ Per Plant Year</u>
1. Turbines	2.48	1.34
2. Steam Generators	2.25	0.48
3. Reactor Coolant Pumps	1.01	0.36
4. Control Rod Drive Mechanisms	0.37	0.35
5. Main Condensers	0.35	0.25 1.16 ⁽³⁾
6. Feedwater Valves	0.16	0.35
7. Moisture Separator Reheater	0.16	0.18
8. Main Steam Isolation Valves	0.12	0.27
9. Turbine Control, Stop, & Intercept Valves	0.12	0.22 0.54 ⁽⁴⁾
10. Supports	0.12	0.06
11. Pressurizer & Main Steam Safety & Power-Operated Relief Valves	0.11	0.15
12. Pressurizer Spray	0.11	0.14
13. Feedwater Pumps	0.08	0.44
14. Feedwater Heat Exchanger	0.04	0.19
15. Filters	NA	0.56 ⁽⁵⁾

(1) Based on outage hours/1,686,980 hours, power reductions excluded.

(2) Based on outage events/192.4 plant years, power reductions excluded.

(3) Power Reductions - 224 events.

(4) Power Reductions - 106 events.

(5) Power Reductions - 108 events.

Table 3.1-3
Summary of Fluid & Mechanical Component Failure
(Arranged by System)

<u>System</u>	<u>Component</u>	<u>Outage Events⁽¹⁾ Per Plant Year</u>	
Turbine Generator System	Turbine	1.35	
	Moisture Separator Reheater	0.22	
	Turbine Control, Stop and Intercept Valves	<u>0.18</u>	.54 ⁽²⁾
	System Total	1.75	
Main Feedwater & Condensate System	Feedwater Pumps	0.44	
	Feedwater Valves	0.35	
	Main Condenser	0.25	1.16 ⁽³⁾
	Feedwater Heaters	0.19	
	Feedwater Filters	<u> </u>	0.56 ⁽⁴⁾
	System Total	1.23	
Main Steam System	Steam Generator Tube Leaks	0.48	
	Main Steam Isolation Valves	0.27	
	Component Supports	<u>0.15</u>	
	System Total	.90	
Reactor Coolant System	Reactor Coolant Pumps	0.36	
	Safety and Power Operated Reliefs	0.15	
	Pressurizer Spray Valves	<u>0.14</u>	
	System Total	0.85	
Control Rod Drive System	Control Rod Drive Mechanism	<u>0.35</u>	
	System Total	0.35	

(1) Based on all outage events in 192.4 plant years of experience.

(2) Power reduction significant - 106.

(3) Power reduction significant - 224.

(4) Power reductions significant - 108.

REFERENCES FOR SECTION 3.1

1. Nuclear Power Experience, Volumes PWR 1-2. Encino, California: Nuclear Power Experience, Inc., 1972 with Monthly Updates.
2. EPRI/DOE Workshop on Availability Engineering and Productivity Improvement, March 13-16, 1979, Chicago, Illinois "Reactor Coolant Pump Seal Investigation".

SECTION 3.2

ENGINEERING EVALUATION
of
Instrumentation, Control and Electrical
Component Failures

3.2 ENGINEERING EVALUATION OF INSTRUMENTATION, CONTROL AND ELECTRICAL COMPONENT

3.2.1 Introduction

Concurrent with the FMC engineering evaluation, an evaluation of Instrumentation, Control and Electrical (ICE) component failures was conducted. The objective of this study was to identify the top contributors to challenges to the RPS and to plant unavailability. Like the FMC evaluation, the evaluation was conducted by individuals experienced in the specialization, the design, specification and procurement of ICE equipment. Their experience gave an added dimension to this program in that normal design issues for plant improvement could also be incorporated in this study.

Unlike FMC components, typical ICE repairs and equipment design changes, especially logic changes, can be accomplished in a shorter period of time. A long outage is not usually necessary to incorporate a design change which improves the performance of the plant. For this reason, plants are more capable and willing to incorporate an ICE design change. ICE solutions are typically viewed as the "easy solution" for this reason and can often win over other more reliable alternatives for improvement. The engineers conducting this portion of the study did not expect lingering ICE problems to exist due to this fact.

It was thought that a review of the data may be beneficial if conducted from a common component point of view due to the large number of common ICE types of components, i.e., relays, capacitors, fuses, controllers, etc. The system level review is felt to be more appropriate as the application of the components and the consequence of failure are more readily understood. Review by generic component application creates uncertainty and confusion from the causes and effects of failure. A review was conducted on the component level as well as a system level review.

This section records the results of the above reviews based on ICE component failures. A discussion is provided as well as recommendations for potential improvement and areas for consideration in the development of evolutionary improvements.

The rest of this section's contents are as follows: Section 3.2.2 discusses the scope and methodology of the ICE portion of this study; Section 3.2.3 contains the discussion of the results of the failure evaluation; Section 3.2.4 reviews the data verification and Section 3.2.5 contains the summary and conclusions of this ICE portion of the study.

3.2.2 Scope and Methodology

The scope of this portion of the PWR component failure study included the review of those events recorded in the C-E PWR Shutdown Data Base which are determined to be related to ICE component failures. The first approach taken in identifying the ICE component failures which cause plant outages was to review the data base by component. A manual review and data reduction was required to further sort and categorize a failure to a subsystem or component level. The second approach was to review the data base by system in which a failure occurred and then by component.

The nature of the information contained in the data base required a rather loose definition of component to be used. For purposes here the definition of "component" relates to the system, subsystem, component, part, personnel, and physical effects, i.e., storms, etc. All of the preceeding can contribute to a failure of a function required to maintain the plant in a safe condition and can result in a challenge to the RPS.

For the first approach to determining ICE components failure events, the entire C-E Shutdown Data Base of events (years 1962 to 1978) was reviewed for ICE related occurrences. An occurrence is defined as any event recorded which caused an automatic or manual scram, controlled shutdown or power reduction. The components were next sorted by common category of "component" and then sorted within these categories for common subcategories. An

example is "sensors" which subcategories would be level sensors, flow sensors, pressure sensors, etc. Little evaluation of these occurrences could be accomplished on a generic basis as the categories related to the component function alone and rarely to common systems applications where a common resolution is felt to be beneficial.

The second approach was to review a portion of the data base on the systems level by ICE related occurrences and then to subsort by system component. In many instances due to experience and knowledge of plants, knowing the system and subcomponent enabled the evaluator to pinpoint the application and typical physical location of components which aided the evaluation.

The time frame used for the second approach was twenty-six months, from October, 1976 to December, 1978. This was done to allow a more comprehensive review of individual events and because ICE problems were not believed to be lingering problems. Additionally it was felt that if a problem still existed it will have shown up to be a problem during this time frame studied.

For both the component approach and the systems approach, further evaluation was conducted only for those top categories resulting from the review. The engineering evaluation searched for predominant or probable root causes and suggests potential alternatives for resolution of the problems along with recommended programs for evolutionary development where justified for greater functional reliability.

As part of the engineering evaluation, site visits and a review of the Limiting Factors Analysis was conducted to verify the data base and obtain additional information on the problems and corrective action. In addition, during site visits, the plant's personnel were requested to rank systems relative to their concern for plant safety and availability as well as the application of a Disturbance Analysis System concept for improving operator response to incidents. This is reported in section 3.2.4.

3.2.3 Engineering Evaluation of ICE Components

3.2.3.1 ICE Component Review and Evaluation

The first approach in identifying major ICE problems causing challenges to the protective system and plant unavailability was to conduct a review of data on a component basis. The results of this review are presented in Table 3.2-1 which ranks the resultant categories by the total number of occurrences. The top seven categories were determined to be (in descending order):

- o Control Systems
- o Personnel related
- o Spurious
- o Sensor Channels
- o Relays
- o Motors
- o Power Supplies

The significance of the above can be interpreted to indicate where more emphasis needs to be put in the design of systems. Emphasis, therefore, should be put on the top four categories, control system performance, personnel training and the man-machine interface, spurious recurrent events and improved sensor channels.

Design and troubleshooting experience would tend to agree with the above. The control systems should consider various load conditions as well as base load design. Further, control systems and other instrumentation systems should be designed for more effective interface with the operator. More effective displays are now capable of being developed. Automatic testing could also reduce this personnel problem.

The spurious category indicates an over-sensitivity to noise or an overly conservative calculation of setpoints for control action. Relaxing some of the vigorous calculations for potential error should be considered if

justifiable, e.g., on four channel redundant systems. Additionally, characterization of required dynamic response of sensors should be considered. Research on the effects of adverse environments to identify characteristics of sensors at the adverse environmental conditions is needed as well as susceptibility to noise.

The total number of ICE occurrences for all plants was found to be 1392. Based on 192.4 cumulative plant years, this results in an approximate average of 7.2 occurrences per plant year for the 40 PWR's considered.

A summary of the results of an evaluation of the top seven components is provided below. A detailed listing of all categories of table 3.2-1 is provided in Appendix 3.2-A.

1. Control Systems

Of 282 control system occurrences counted, approximately 235 events pertained to general and unspecified automatic operation. About 60% of these were identified as occurrences or malfunctions of the Feedwater Control System. From experience, the most probable causes are the problem was failure to control during a plant transient, that the system was in the manual mode of operation, or that the particular component failure was not reported.

About 30% of the 235 events were Turbine Control System related in the areas of turbine trip due to low auto stop oil pressure, unidentified Electro-hydraulic Control malfunction or turbine governor and stop valve problems. About 10% of the 235 events were attributed to unexplained CEA drops due to the Control rod system malfunctions. The rest were basically considered in a miscellaneous category.

2. Personnel

Two hundred and eighty recorded occurrences in the C-E shutdown data base were attached to Personnel. Personnel testing and maintenance errors account for 40%. Manual feedwater control during plant startup

or shutdown contributes about 37%. The remaining 23% is attributed to miscellaneous operating errors associated with various systems and equipment.

3. Spurious

Spurious (sudden random occurrences without specific determined causes) events were reported one hundred sixty-nine times in the 192.4 years of plant experience.

Approximately 78% of the spurious control actions or signals or Electro-magnetic Interference or Radio Frequency Interference (EMI/RFI) are attributed evenly among the turbine generator Electro-Hydraulic Controls (EHC), flow, nuclear instrumentation and temperature channels, some sensitive RPS circuits.

Many of these are sensor inputs to the RPS, where a coincidence of two signals is required to trip the reactor. Some of these sensors are overly sensitive to noise due to the response required for stated worst case analysis. Additionally, often a too rigorous analysis is required for error analysis of some channels which result in an overly restrictive operating band for some parameters. When testing is performed on the RPS, the tested channel is sometimes required to be put in a trip condition which increases the probability that noise occurring in one of the other channels will cause an unnecessary trip.

Approximately 16% are induced by storms and lightning effecting turbine trips due to electrical distribution disturbance or instrumentation effects due to EMI/RFI.

4. Sensor Channels

One hundred eight sensor occurrences were related to sensor channels and reported as malfunctions.

Sensor channel malfunctions were found to be distributed as follows:

- 21% - Nuclear Instrumentation
- 16% - Flow Instrumentation
- 14% - Level Instrumentation
- 11% - RTD Instrumentation
- 10% - Pressure Instrumentation
- 28% - Miscellaneous

To put the above into perspective, it is estimated there is a total of 900 sensor channels in a typical PWR of which 30% may lead to a forced outage if they failed. For 40 PWR's this results in a potential population of about 10,800 sensor channels. The one hundred eight occurrences is an average of 1% failure of the potential sensors which is small and may be difficult to improve. From experience, sensor failures have been related more to the application than generic issues. The process conditions, operator procedures, and the environmental conditions that exist for the application are the main factors which impact sensor failure. From a generic viewpoint, better techniques for environmental qualification are needed as well as work on the development of sensor response characteristics for adverse environments. Characterization of sensor response for adverse environmental conditions can lead to improved incident monitoring and eventually lead to the evolutionary development of self compensating sensors.

5. Relays

Seventy-two occurrences were reported to be the result of relay failures. Protective relays for Electric Distribution Systems, Turbine Generator and Main Circuit Breakers contributed about 47%. All others appeared to be in miscellaneous systems or application.

6. Motors

Sixty-two occurrences were reported of which approximately 59% of the failures were attributed to CRDM winding or stator problems. About

10% involved the Reactor Coolant Pump winding or stator. This problem is addressed in the discussion on the control rod system in the systems portion of this section.

7. Power Supplies

Fifty-six power supply failures were counted. Loss of control grade power to instrumentation and control circuits accounted for 38% of these occurrences, internal CRDM logic or control power supplies about 27%, and internal Turbine Generator logic/control power supplies about 12%.

The remaining Instrumentation Controls and Electrical (items 8 through 27 of table 3.2-1) are not considered significant since the number of occurrences is relatively small and random in nature.

Ranking by outage duration could not be performed due to the inability to sort to the component level required. The observation was made that most of the ICE outages were in general of short duration when compared to other fluid and mechanical system outages. On the average, it is estimated the outage duration due to a typical ICE component failure is 10 to 20 hours. Assuming 15 hours each and an average of seven ICE outages per plant year, this results in 105 hours average expected down time per plant year due to ICE related causes. Although this appears to be significant as a total, it is due to a large spectrum of equipment failures and thus a smaller cost benefit per improvement. Improved safety through improvements in the process systems and reduced challenges to the RPS (i.e., accident prevention) is the best benefit from generic improvements to ICE equipment.

It was found that this component level method of review is limited when trying to track and evaluate the root causes, impact and potential solutions to ICE problems. Also there appeared to be only a few cases where the identical ICE component failure was found in different plant systems or plants, which would allow development of generic solutions. Therefore, a

second approach was used, the grouping of types of problems by system. The next section addresses the component failures found using this method.

3.2.3.2 ICE Component Review and Evaluation by System

The C-E PWR Shutdown Data Base was reviewed to identify those ICE related outages by system. These outages were then grouped and ranked on a plant system basis. Table 3.2-2 is the results of this effort. A comparison of total outages per plant year in Tables 3.2-1 and 3.2-2 indicate that approximately 80% of the ICE component problems are contained in the top seven systems of Table 3.2-2. Of these seven systems, the first three systems (Feedwater, Turbine Generator and On-site Power) contribute about 70% of the ICE occurrences. The result of the component problem identification grouping within the top seven systems is provided in Table 3.2-4.

The following engineering evaluation discussions address the major problems of each of the seven top ranked systems. Each discussion contains a discussion of the predominant root cause(s), sequence of events leading to the plant outage and any appropriate safety or availability concerns. Additionally, a discussion of potential solutions and recommendations for further development is provided.

A. Feedwater & Condensate System

The main Feedwater and Condensate System was the major cause of ICE related outages during the twenty-six month period of this review. Table 3.2-3 shows that the occurrences result from a multitude of components. The two most significant recurrent categories are shown to be manual control of steam generator level during startup and feed pump trips. A discussion of these events is presented below.

A.1 Loss of Steam Generator Level Control

Manual control of the feedwater system during transients, especially plant startups and shutdowns below 20% of full power, appears to be the predominate problem. The degree of difficulty is due to the number of variables controlled,

the complex characteristic steam generator response to power level changes and the required vigilance by the operator. The operator's difficulty in coping with this transient usually is reflected in a steam flow/feed flow mismatch resulting in a reactor and/or turbine trip.

One potential solution is to improve the automatic feedwater control system to control steam generator level below 20%. Two of the NSSS vendors currently offer such a feature. One method of implementation consists of selected electronic controllers and level compensation networks to automate the operation of the feedwater bypass control valve. The system utilizes the existing steam generator level signal, compensated to improve stability, to generate a flow demand signal which is sent to the feedwater bypass valve controller. See Figure 3.2-1. The overall system consists of an independent control loop for each steam generator. It is considered to be retrofitable on operating plants. Incorporation of this potential solution is expected to significantly reduce the number of challenges to the Reactor Protective System at low power levels.

Another potential solution is the application of Disturbance Analysis System techniques as an operator aide. Reference "On-line Power Plant Alarm and Disturbance Analysis System" EPRI NP-613, Feb. 1978 for a discussion of these techniques. (Reference 1).

A.2 Loss of Main Feedwater Pump(s)

The predominate cause of feedwater pump problems appears to be low pump suction pressure trips. The root causes found were feed pump suction flow transients caused by restarting a pump, strainer cleaning or other condensate and feedwater system malfunctions.

A typical example is putting a motor driven feed pump back on-line while at rated power on the operating pump. Under these conditions, the feedwater control valves are fully open when the pump is restarted. This permits full flow on the pump discharge with a resulting low flow transient condition

at the pump's suction header and, should the Feedwater Control system be in automatic, a rapid closure of the Feedwater Regulator Valves due to a large steam flow-feed flow mismatch.

Tripping the pump(s) on low suction pressure also causes a loss of feedwater transient which can result in an automatic turbine runback for plants so equipped along with a challenge to the RPS.

The following is a summary of potential solutions and recommendations:

- (1) The plants operational procedures could be made more specific to insure abnormally high transient flow conditions are avoided during feed pump startups by requiring power reductions, throttling of discharge valves and opening of the pumps' recirculation valve.
- (2) An alternative solution is to install a time delay in the feed pump's suction pressure equipment protective circuitry, to enable it to run through brief transients (see Appendix 3.2-B Section II.D.5). This alternative appears to be remedying one particular plant's problem. Other plants should determine an allowable time delay without posing a hazard to their type of pump.
- (3) The constant speed motor driven pumps could be changed to variable speed motor drive, variable coupling or steam turbine driver to prevent sudden changes in flow conditions when starting or stopping a pump.
- (4) Booster pumps with associated automatic starting control circuitry as a function of feed pump suction/flow/power level conditions could be implemented to help maintain the flow requirements during transient conditions.
- (5) Automatic turbine generator runback control circuits could be incorporated for the case where one feed pump is tripped off-line and steam generation

capability is possible with the remaining feed pump. This would rapidly reduce steam flow and close the FW control valves, allowing feed pump suction pressure recovery.

B. Turbine Generator System

Major ICE problems in the Turbine Generator system were determined to be EHC malfunctions, loss of generator excitation and other generator problems (see Table 3.2-3). The above problems are discussed below.

B.1 Electro-Hydraulic Control (EHC) Malfunctions

The most frequent EHC problems occurred in two categories; (1) Turbine trips while performing tests or maintenance and (2) turbine trips due to control system malfunctions whose root causes were not readily identifiable.

A variety of root causes during testing contribute to turbine or reactor trips. The only area that stands out is testing of the turbine intercept and control valves. Testing may cause fluctuations in the turbine hydraulic control system, inadvertant closure of valves or a spurious turbine trip on low EHC pressure. The remaining occurrences during testing appear to be incipient random failures coincident with the testing.

A significant portion of the turbine trips are spurious in nature without a root cause being identified. These types of events are attributed to a host of potential causes: low EHC pressure transients, momentary losses of DC control power, equipment protective settings which drift low, false overspeed trips, sticky relays, EMI/RFI electrical noise, dirty EHC oil, etc.

From the review and investigations performed under this study, there does not appear to be any easy ICE solutions to the potential causes from a generic systems viewpoint. The apparent random nature of the EHC occurrences causes each plant to solve its particular problems. Typical examples are found in Appendices 3.2-B & 3.2-C data bases.

In Table 3.2A-1 of Appendix 3.2-A, the 78-8-10 automatic shutdown was caused during the testing of turbine valves by switching from sequential valve opening to a single valve opening mode of operation, which caused a swing in megawatts resulting in a high pressurizer pressure reactor trip. The EHC computer controlled logic for the valves was revised to sequence them as a function of power level in order to smooth the accompanying transient. As noted in Appendix 3.2-B, Section B.2 identifies a case where a momentary loss of 125V dc control power was spuriously occurring. The plant's corrective action was to incorporate a short time delay in this turbine trip parameter to act as a filter for these momentary transients.

Further effort should be expended to investigate (1) the use of more redundancy in the EHC design, (2) techniques to improve testing methods and procedures (including automatic testing), (3) the use of advanced monitoring techniques (such as the Disturbance Analysis System) to identify and inform the operator of EHC system disturbances which may cause the turbine to trip and (4) the reason(s) one turbine vendor's equipment appears to have a significant advantage (.25 vs .43 outages per plant year) over that of his competitor with respect to turbine control malfunctions resulting in challenges to the protective system.

B.2 Loss of Generator Excitation

The data reviewed identified the possible root causes to be either (a) loss of field due to field relay fuses or power supply failures or (b) unidentified malfunctions of the regulator. About 60% of the regulator problems are classified as unidentified malfunctions.

The nature of the occurrences are spurious malfunctions with unknown root causes or events/failures which immediately result in the turbine tripping on loss of field. Further investigation is needed to (a) identify the unknown root causes and (b) determine if the reliability of the field relays and associated power supplies could be improved.

B.3 Generator Problems

Most causes of generator outages are bearings, stator/rotor, hydrogen cooler, exciter or circuit breaker problems. Of these, problems with stator and rotor grounds are the most significant because of the length of outage time required to find and correct the ground. Undetected circulating currents via grounded bearings has also caused excessive bearing wear resulting in substantial damage to the rotating equipment.

Better methods of detecting and monitoring of generator grounds and circulating currents should prevent unwarranted bearing damage. Generator design, materials and maintenance methods should also be reviewed to reduce the occurrence of grounds.

C. On-Site Power System

Major problems in the On-Site Power System were determined to be grouped in three areas: Loss of vital or control instrument bus, DC/AC inverter problems, and RCP bus loss. A discussion of these problems is provided below.

C.1 Loss of Vital or Control Instrument Bus Loss

Loss of Vital Busses (both 480 Vac ESF and 120 Vac) have occurred resulting in unwarranted plant trips and full ESF system actuation. Most of the plant occurrences reviewed were due to instrument bus loss. Root causes were not identified in the data base. Therefore, the most probable causes are reasoned to be: momentary transients during automatic transfer, equipment energization/de-energization, personnel error, ground/short faults, individual feeder circuit breaker malfunctions, and multiple events (combination of these causes).

Loss of a 120 Vac Control Instrument Bus has also caused the control systems (FWS, RRS, etc.) to fail in a mode requiring the RPS or ESF systems to be actuated. These occurrences are probably attributable to the same type of

root causes assumed to occur on the Vital Busses. One difference believed to be significant, but unconfirmed, is that an uninterruptible inverter power supply is not generally provided to back up the Control Instrumentation Bus on loss of the normal power supply.

Figures 3.2.2 and 3.2.3 depict two typical On-Site Power distribution schemes. The two battery design (Figure 3.2-2) is utilized in most presently operating plants. The four battery design (Figure 3.2-3) is predominate in plants presently going on line or in various stages of design or construction. The four battery On-site Power System is expected to provide some increased resistance to this type of occurrence and warrants consideration for back fitting on operating plants.

The addition of uninterruptible power supplies (dc/ac inverters and battery with charger) to the Control Instrument Bus should add to the reliability of the control systems and reduce the number of occurrences resulting in challenges to the RPS.

C.2 DC/AC Inverter Problems

Based on this program's review, a predominate number of converter fault root causes were classified as unknown. The review also showed the remaining root causes to be random and/or multiple events, without any noticable trend for any particular component.

Unfortunately the plants having an operational status which permitted visits to discuss the data base, did not exhibit inverter problems. Based on engineering judgement the three most probable causes are random component failures, electrical surges (frequency, voltage, or current) or personnel error.

The potential for inadvertent plant trips resulting from inverter failures is greater for the control system (FWS, RRS, etc.) than the safety systems. This is due to the control grade system usually relying upon a single source of power.

The Vital Instrument Bus Inverter system design is usually four fold redundant and the safety system logic requires a coincidence of two channel trips to actuate. It is therefore assumed that plant trips due to loss of Vital Instrument Bus Inverters must have been caused by multiple events such as a combination of EMI noise, personnel error and/or incipient random failure.

From a generic systems viewpoint, improvements in the degree of redundancy for the Control Instrument Bus should be considered. The design should be commensurate with the degree of redundancy improvements in both the On-site Power Distribution and the control system designs themselves. Plants without Control Instrument Bus dc/ac Inverters should add them and those that do have them should consider redundant supplies.

New plants and those that experience high personnel turnover rates should review and improve operating and maintenance procedures for invertors to insure that proper calibration and setting of inverter equipment protective and transfer circuitry are accomplished and verified.

C.3 Reactor Coolant Pump Bus Loss

As shown on Figure 3.2-3, Bus 1 and 2 typically power the reactor coolant pumps from the Main Generator via Unit Auxiliary Transformers. Electrical disturbances (frequency, voltage or current transients) on the bus may cause it to attempt switching to the Start-Up Transformer or trip the Reactor Coolant Pump (RCP) breakers. These disturbances are usually caused by events such as tripping the turbine generator off-line, periodic testing of the Standby Generator, instabilities on the utilities grid and personnel error during bus/load transfers.

The loss of the bus trips two RCP's, usually resulting in a reactor trip on low reactor coolant system flow or some calculated function, (i.e., power to flow ratio) dependent on the Reactor Protection System design.

A solution which could permit continued plant operation at reduced power is to consider the use of separate busses for each RCP. This would reduce the possibility of tripping two pumps simultaneously due to personnel error

during maintenance and testing of the busses, transformers, circuit breakers, etc. The effort would entail a review of the NSSS design and safety analysis. This is necessary to determine if continued plant operation is permissible at reduced power for the loss of a single pump and concurrent reactor and steam generator energy imbalances or flows. These analyses should also consider NSSS designs utilizing 2, 3 or 4 steam generator designs.

Further effort is required to identify potential solutions for the remaining types of bus transients. The effort should include the determination of the types of electrical transients the RCPs can sustain, and the improvement in RCS flow monitoring so flow transient conditions do not result in an unnecessary challenge to the reactor protective system.

D. Control Rod System

Two kinds of ICE related failures in the Control Rod System were predominant. The first was CRDM motor failures and the second was CRDM power supplies (see Table 3.2.3). Results of the investigation of these problems are reported below.

D.1 Control Rod Drive Mechanism Failures

Control Rod Drive Mechanism (CRDM) coil failures were noted to be prevalent in one NSSS vendor's design. The problem may also have been aggravated by CRDM seal housing leakage.

The problem is currently being resolved. Based on discussions with site personnel the affected NSSS vendor has improved coils and recommends maintenance procedures that require an N₂ purge during plant cold shutdown conditions. See Section 3.1.3.2.C.

D.2 CRDM Power Supply Problems

The types of CRDM Power Supply failures include tripping of CRDM motor generator sets, failures of internal control logic power supplies, and failures of individual CRDM power supplies.

Most of the problems appear to be random, with the root cause being difficult to identify. An exception is the internal control logic power supplies for one NSSS vendors design. One utility has replaced all the factory installed supplies with ones believed to be more reliable.

Recommendations for the addition of redundant auctioneered power supplies have been made to operating plants. For later plants, the NSSS vendor has modified control system design. Further effort is not warranted in this area.

E. Reactor Coolant Pump

A significant number of RCP bearing lube oil level problems were reported. This was thought to be due to sensor problems which had been experienced in the past. The problem with the detector was that it was mounted on the RCP and subject to high vibration. In addition, the lube oil tank was a large volume shallow tank. The span of the instrument was therefore very small, and therefore vibration effects on the sensor were significant, causing many alarms. The solution recommended was to remove the detector for remote mounting. It was found in this study that the cause of the incidents reported was oil leaks.

During the time interval included in this study there were only a few other ICE related RCP component failures leading to challenges to the protective system. These failures were considered random in nature. Therefore, further failure evaluation effort in this area was not warranted.

F. Reactor Coolant System

The major ICE problem in the Reactor Coolant System was determined to be process instrumentation of which the reactor coolant flow sensors occur most often (see Table 3.2-3). A discussion of the problem is included below.

F.1 Process Instrumentation

Reactor coolant flow transmitters appear to be the most troublesome sensor channels with respect to causing plant outages. The data base indicates the causes to be spurious flow dips or spikes. These are believed to be caused by actual flow disturbances associated with reactor coolant pump pulsations and the highly turbulent flow conditions being monitored. These occurrences manifest themselves as either: (1) plant trips caused by spurious small transients (see Appendix 3.2C, Section B.4) or (2) plant trips while performing periodic tests or maintenance with one of the redundant safety channels already in the tripped condition. Other root causes appear to be random in nature such as a power supply failure, loose parts, etc. The root causes have not been identified in a significant number of the cases and are assumed to be the same as those described above.

An improvement in the accuracy of flow sensor instrumentation should be attempted by either upgrading the existing methods or developing alternative methods of monitoring system flow (such as RCP speed or ultrasonic techniques). Improvement in sensor stability during environmental fluctuations (primarily temperature, vibration, radiation and electrical noise) should be pursued to reduce the error margin which is factored into the safety system setpoint analysis. This would permit plant operation closer to full power, without being subjected to an abnormal amount of unwarranted spurious trips. It should be noted that one NSSS vendor has switched to utilizing RCP speed sensors for newer plant flow signal input to the reactor protective system. It is also offered as a back fit option for operating plants.

G. Main Steam System

During the data period studied, a relatively few main steam related plant outages were attributed to ICE causes. As shown in Table 3.2-3, the Main Steam Isolation Valves (MSIV) and a few sensor malfunctions were identified.

As noted earlier in this report in section 3.1, the MSIV problem is being resolved by the valve manufacturers or by individual utility modifications. The limited number of sensor malfunctions resulting in plant shutdowns does not warrant further effort specifically for these sensors.

3.2.4 PWR Shutdown Data Base Verification

The data base verification effort included reviews of associated Licensee Event Reports (LER) and Utility Annual Reports where available, site visits to operating plants and a review of the "Limiting Factors Analysis" of the Maine Yankee Atomic, Oconee and Turkey Point Power Plants reports performed for EPRI. (References 2, 3, & 4)

3.2.4.1 Verification by Site Visit

Three site visits were made to operating plants to verify the data base input and results found to date. One site was visited for each of the three major NSSS PWR vendor designs. The three sites contained a combined total of six operating units. The age of plants ranged from 2-1/2 to 6 years and capacity from 600 to 800 MWe. The plants reflected a representative sample of the types of occurrences found in the C-E PWR Shutdown Data Base.

The specific objectives of the visits were to confirm and discuss with the operating staff (plant engineers, operators and maintenance personnel) from an ICE viewpoint, the following plant specific items:

1. For the systems selected, review in detail each of the events in the data base to:
 - a. Verify that the event and subsequent entry into the data base was properly interpreted,
 - b. Determine the root cause of the disturbance. This involved discussions and review of maintenance history.

2. Query the operating staff to determine if they agree with the order of ranking of plant system disturbances by frequency and impact on plant safety or availability.
3. Determine if there are categories of non-reportable occurrences in the systems chosen.

Each of the trip reports are documented in appendices 3.2-B, 3.2-C and 3.2-D. Each contains (1) a listing of the occurrences reviewed with corrections or clarifying information, (2) an opinion survey ranking of the system occurrences for the plant as viewed by it's operating staff and (3) comments and conclusions resulting from discussions with the site personnel.

The following summarizes the findings and observations resulting from the plant visits:

- a. The visits, based on reviewing the plant documentation and discussions with site personnel, in general confirmed the Data Base contents and suspicions.
- b. The systems chosen for review in section 3.2.3.2 were found to be in general agreement with the survey of site personnel. Site personnel were queried to determine their concurrence on the order of ranking for their plant systems by frequency of disturbances, plant safety impact, plant availability impact and the need for improved operator guidance systems such as the Disturbance Analysis System (DAS). A benefit of choosing this survey method is that operating staff are unlikely to identify unimportant problems.

Each site person was asked to rank the twelve systems listed in Table 3.2-4, in order of importance to (1) safety, (2) availability and (3) application of a system such as the DAS. The data taken was reduced by weighting the rankings on a 12 thru 1 scale in order of decreasing

importance. The weighted values were then summed for each system. The resultant top rankings for each category are presented in Table 3.2-4.

The survey indicates, as expected, that the Main Feedwater and Condensate system is of the highest concern and is the operator's most likely candidate for improvement. From a safety standpoint, it was the only system which concurred with the data base findings. This may be due to a difference between the definition of safety used in this study and that used by operators.

The Turbine Generator System was ranked second in impact on availability. This supports the shutdown data base findings.

The Chemical and Volume Control System is of sufficient operator concern to rank third as a major contributor to plant outages. This did not agree with the shutdown data base findings. Further investigation is recommended.

The rankings of the remaining systems do not exhibit any clear trend. It is interesting to note that the category "Personnel" was of sufficient concern that it was ranked seventh and eighth respectively for impact on plant safety and availability out of a possible thirty-six items.

- c. Review and discussions did not reveal any new class of events or unreportable occurrences leading to plant outages.
- d. The information input to the Shutdown Data Base needs improvement if it is to be better utilized as an engineering tool to evaluate systems or components. The data is not detailed enough to pinpoint root causes and the sequence of events leading to plant shutdown.
- e. A wide range of maintenance philosophies among the utilities exists. There appears to be a need to improve the average plant's test and maintenance methods, procedures and scheduling. Improvements are necessary to reduce the likelihood of personnel error and forced

outages due to equipment failures during the mature years of plant operation. Methods should be implemented to allow assessment of gradual degradation of equipment performance such that trends can be identified for problem resolution during scheduled outages.

- f. Information feedback to the equipment designer will also enable him to make adjustments and allowances in system design for the next generation of equipment.

3.2.4.2 Verification by Limiting Factor Analysis Review

A. Review of Maine Yankee Atomic Power Plant Limiting Factors Analysis

The Maine Yankee Atomic Power Plant Limiting Factors Analysis (Reference 2) is the result of the study of two and one-half years of mature plant operation. This study has been reviewed for ICE related occurrences contributing to plant outages. The Limiting Factor study included the first few months of this plant's world record for continuous operation.

The Maine Yankee Limiting Factors Analysis study generally supports the Shutdown Data Base findings in that:

- a. The Feedwater system is a major source of lost power resulting from failures, power control at low power, and fouling of flow venturis.
- b. The Turbine Generator System is another major problem area, especially in regard to failures in the EHC system.

Maine Yankee Atomic Power Plant is a very good example of the relationship between the attitude of the management and personnel and the plant availability and efficiency. In almost every case where a failure resulted in an outage the plant staff made design changes to prevent recurrence. An example is replacing types of power supplies. Other design changes were also made to

prevent the possibility of failures from causing recurrent outages. This is certainly a major contributor to this unit's record-setting performance.

The study emphasized areas in which significant improvements in overall plant performance can be made. These are listed as follows:

- 1) The improvement of EMI/RFI noise reduction techniques.
- 2) Improvement in the accuracy of flow measuring technique.
- 3) Automation of Refueling and Inservice Inspection equipment.
- 4) Low power automatic steam generator level control.

B. Review of Oconee Limiting Factor Analysis

The Oconee Nuclear Station Unit-1 was selected as the reference plant in the "Limiting Factor Analysis of High Availability Nuclear Plants" EPRI RP894-3 (Reference 4). This study was reviewed for ICE related occurrences contributing to plant outages.

In summary, the report did not identify the ICE equipment as a significant factor in this plant's unavailability. This was based on the relatively few occurrences when considering the large number of ICE components in a plant and their low mean time to repair. In general the report did not uncover any new instrumentation and control problems. It also tended to support the C-E Shutdown Data Base review findings and recommendations.

The study confirmed that feedwater pump trips were not the result of a component failure problem, but occurred as a result of fluid system instabilities. It also identified a need for improved feedwater heater level sensing and control.

Within the turbine generator system, excitor bearing problems were noted as well as EHC problems in the lube oil system and concurrent with valve testing.

CRDM seal failures were noted along with CRDM stator failures. Reactor Coolant Pump lube oil level control problems were also noted which the vendor is resolving.

C. Review of Turkey Point 3 Limiting Factor Analysis

Turkey Point Unit 3 was the reference plant in the "Limiting Factor Analysis of High Availability Nuclear Plants" EPRI RP894-4 (Reference 5). This study was reviewed for ICE related occurrences contributing to plant outages.

In summary the study supports the types of occurrences found in the C-E PWR Shutdown Data Base. It also agrees with this study's conclusions in general and notes that Root Causes are very hard to properly identify in current industry data bases, without contact with site personnel.

The following summarizes the review of the study results on a system basis.

A. Feedwater and Condensate System

- o Low power feedwater control improvements were recommended.
- o The Turbine Runback circuit was modified to prevent unnecessary runbacks due to loss of a feedpump.
- o The MSR/heater drain tank level controls should be reviewed to prevent unwarranted level oscillations.
- o Miscellaneous main feedwater and bypass control valve malfunctions have occurred.

B. Turbine Generator System

- o The turbine hydraulic controls exhibited erratic operation due to oil contamination.

- o The EHC test system was modified to allow on-line repairs to solenoid test valves.

C. Miscellaneous

- o Miscellaneous pressure, differential pressure process and neutron flux detector problems were noted (i.e. drift/inaccuracy), which due to system redundancy did not affect plant operation.
- o Specific types of relays that were found defective had to be replaced in the RPS/ESF systems.

3.2.5 ICE Summary, Conclusions and Recommendations

Component Review

The review of the C-E Shutdown Data Base on a "component" basis indicates that ICE failures caused approximately 7.2 scrams, controlled shutdowns or power reductions per plant year. The top four common categories were determined to be Control Systems, Personnel, Spurious and Sensor Channels. The remaining categories appear to be random considering the large quantity of ICE components found in a typical plant.

Most of two hundred eighty two Control System problems appeared in the Feedwater, Turbine Generator and Control Rod systems. The root causes for a majority of these occurrences were unable to be identified from the data base content.

The review identified two hundred eighty Personnel related occurrences of which 37% were attributed to manual feedwater control problems and the rest to testing or maintenance errors. Improved man-machine interface could help to reduce these numbers.

One hundred sixty nine events, labeled spurious, were mainly in the turbine generator, flow, nuclear and temperature instrumentation areas. This spurious category is believed to be the result of an over-sensitivity of sensors and channels to noise.

Sensor Channels contributed one hundred eighty occurrences with nuclear and flow instrumentation making up approximately 40% of the occurrence. It is estimated that the average total number of sensor failures is on the order of 1% of the potential population and that this may be difficult to reduce due to the large population. Areas needing further work are the defining characteristics of sensors under extreme environmental conditions and the required sensitivity of sensors.

System Review

The top seven systems based on impact to plant availability and safety were reviewed for the time period of October 1, 1976 through December 31, 1978. The top three systems ranked systems accounted for approximately 70% of the ICE related occurrences.

Table 3.2-5 summarizes the results of the review by system. For each of the seven systems it identifies:

- a) the most likely potential cause(s),
- b) the average event (occurrence) frequency,
- c) average duration per event, and
- d) potential solutions or recommendations based on this phase of the review.

In summary, the Feedwater and Condensate System root causes requiring further effort are loss of main feed pumps and steam generator level control at low power level. The Turbine Generator System test and maintenance methods along with the susceptibility to spurious EHC trips should be evaluated and improved.

The On-Site Power System loss of the 120 Vac instrument bus and loss of the Reactor Coolant Pump bus occurrences should be evaluated for design changes to improve their availability. The remaining four system problems do not appear to be significant ICE problems or are currently being resolved.

Review of Limiting Factors Studies

The Maine Yankee Atomic Power Plant, Oconee and Turkey Point-3 Limiting Factor Studies (LFS) were reviewed in detail for ICE related occurrences contributing to outages. In general, all LFS were found to contain evidence which supports the C-E Shutdown Data Base review findings.

The review of the Limiting Factor Studies, in general, suggests that a correlation exists between the degree of professionalism and attitudes of the utility staff and the plants overall high availability record. This conclusion is supported by one utility's apparent willingness to implement design and hardware modifications and by observations noted at other plants.

A. General Observations/Recommendations

The C-E Shutdown Data Base content appears satisfactory for its original intended purpose of tracking plant availability as a function of plant systems. The data input could use improvement for engineering evaluation purposes to identify and resolve causes for challenges to the protective systems or reduced plant availability. The shutdown data base provided the necessary information to start an effective engineering evaluation by allowing a hierarchy of problems to be established. The data record is not detailed enough to pinpoint root causes and the sequence of events leading to the plant shutdown. A failed part description would be helpful. It is also difficult to correlate with other data bases such as LERs and NPRDS. The present best method to verify and obtain detail is by examination of site outage documentation and discussion with site personnel. Further, equipment investigation and experimentation are needed in some cases.

If evolutionary improvements are to be developed the need for system performance information needs to be backfed to the designer in an efficient manner. This could be accomplished by either better data base input requirements or an on-site NSSS representative who would fulfill the service requirements and ensure that the proper design information is backfed to design engineers.

B. Man-Machine Interface

It is obvious from the data base review discussed in section 3.2.3.1 that the man-machine interface needs improvement if significant advances in plant safety and availability are to be accomplished. The recorded plant operating history indicates personnel, operational, maintenance and testing errors rank high in contributing challenges to the RPS and to plant unavailability. Recent history (TMI-2) also highlights the potential for degradation of plant safety due to operator errors. The above concerns dictate the need for an in-depth evaluation of human factors in PWRs. In addition, The ways to make operators and other plant personnel more effective should be determined by analyzing the operating history to identify trends and develop a clearer picture of circumstances contributing to personnel errors. Improved training techniques would be helpful. For example, timely preparatory video tapes on planned operating modes or evolutions would help, especially the precautionary considerations. A better understanding of all operators involved would result.

C. High Frequency Occurrences in PWR's

The general review of the C-E PWR Shutdown Data Base and its verification effort alludes to the suggestion that nuclear plants are fairly well designed to handle single failures. The plant's systems are either generally designed for single component/system malfunctions or the operator can easily cope with them. This leaves multiple failures as the next most important contributor to plant outages and challenges to the plants protective systems.

It is likely that the systems and components exhibiting high single occurrence rates also have a greater than average chance of failing simultaneously when the services of a previously failed system is not available. It also probable that the consequence of concurrent events will be more detrimental to the plant in terms of the severity of the event and the potential for damage.

It is recommended that a study be performed to identify high frequency singular PWR system or component failures which are the most likely to occur concurrently during the life of the plant. These should then be evaluated to determine which combination of those occurrences should be accommodated to improve overall safety and availability.

The study should include:

- a. Reviewing existing Shutdown Data Base, LERs and NPRDS to identify high frequency events where concurrent or the same initiating root cause exist.
- b. Perform engineering evaluations and statistical analyses to identify those events which are most likely to occur simultaneously or are likely to exist when a second failure occurs where the other function is not available for mitigation.
- c. Ranking those multiple occurrences based on overall severity or impact on plant operation.
- d. Development of additional recommended PWR Design Basis Guidelines from an assessment of the concurrent failures and their impact on the industry's Design Basis Events currently considered in the Safety Analysis.
- e. Determine when in the plants life the events are most likely to occur.

D. Advanced Control System Concepts

The desire to improve plant control system concepts, rather than protection systems, becomes apparent when one considers the number of times the plant's protective systems are actuated by control system or component malfunctions. Significant benefits in both safety and plant availability can be achieved by maintaining process system operation. The following two concepts are examples of the types of advanced control concepts which may be implemented:

(1) Control Override System

A control override system would monitor controlled parameters and cause required corrective control action or take-over control when a parameter passes a predefined limit.

The effort to develop the system should include:

- a. A review of the data base to determine the most significant events which result in plant trips. The feedwater system is the primary system for consideration.
- b. The development of conceptual designs for a system which is effective in reducing the number of plant trips.
- c. An assessment of the benefit in safety and availability which would result from implementing the conceptual design.
- d. The detailed design development and demonstration testing at one PWR power plant.

(2) Adaptive-Control Supervisory System

Control system and protective system setpoints can be overly conservative and restrictive dependent on the mode of operation. An Adaptive-Control Supervisory System modifies the control

system and/or protection system setpoints, or control procedures so that plant response is optimized relative to safety margins, efficiency and availability.

The effort to develop such a system should include:

- a. Development of analytical techniques for parameter estimations on a generic basis.
- b. Establishment of efficient plant system models for on-line application.
- c. Specification of computer software and hardware requirements.
- d. Demonstration of the effectiveness of these control methods by computer simulation experiments.
- e. Testing of a prototype system at an operating PWR.

E. Improved Process Instrumentation

Based on information obtained from the shutdown data base, it is estimated that 35% of the 169 identified spurious trips were due to sensor instrumentation channels. Another 108 occurrences were identified as sensor channel problems. A significant portion of these are due to erroneous flow, differential pressure, pressure and nuclear instrumentation signals.

It is recommended that a detailed investigation of process instrumentation sensor channels be conducted in order to improve sensor accuracies and the capability of sensors to withstand environmental fluctuations. Further, a study of parameter sensitivity and dynamic detector requirements could help to reduce noises related scrams. The goal is to eliminate unwarranted spurious challenges to the protective system. An added benefit could allow operation with reduced error margins and thus greater power output. Also sensors could be used for extended periods under adverse conditions if characteristic detector response were known for the adverse conditions.

The methods to meet these goals should include:

- a) A survey of a spectrum of existing sensor environments for vibration RFI/EMI noise, temperature, pressure and humidity.
- b) The development of baseline plant conditions from the environmental surveys. This defined set of environmental parameters may then be utilized as a common base for generic qualification and for comparison of sensor performance data for the set of specified conditions.
- c) The analysis of process parameters to determine minimum and maximum required sensitivity and dynamic response to minimize shutdowns due to over-sensitive equipment.
- d) Characterization of sensor response due to adverse environmental conditions.

REFERENCES FOR SECTION 3.2

- 1) On-Line Power Plant Alarm and Disturbance Analysis System, EPRI NP-613, Project 891 Interim Report, February 1978.
- 2) Limiting Factors Analysis of a High Availability Nuclear Power Plant Study Performed at Maine Yankee Atomic Power Plant EPRI RP894-2, in publication.
- 3) Limiting Factors Analysis of High Availability Nuclear Plants Study Performed at Oconee Nuclear Station - Unit 1 (Oconee 1) EPRI RP894-3, in publication.
- 4) Limiting Factors Analysis of High Availability Nuclear Plants Study Performed at Turkey Point Unit 3, EPRI RP894-4, in publication.

TABLE 3.2-1
Ranking of Plant Occurrences
 By
ICE Components

<u>Rank</u>	<u>Component</u>	<u>No. Occurrences</u>
1.	Control Systems	282
2.	Personnel Related	280
3.	Spurious	169
4.	Sensor Channels	108
5.	Relays	72
6.	Motor	62
7.	Power Supply	56
8.	Bus	49
9.	Transformer	39
10.	Valve Operator	36
11.	Breaker	35
12.	Fuse	34
13.	Inverter	28
14.	CEA Position	18
15.	Generator	17
16.	Voltage Regulator	16
17.	Connections	15
18.	Instrumentation (Misc.)	15
19.	Capacitor	14
20.	Switches	13
21.	Cables/Wiring	8
22.	Bistable	7
23.	Fans	7
24.	Printed Circuit Card	5
25.	Battery	4
26.	Diodes	2
27.	Computer	<u>1</u>
TOTAL		1,392

TABLE 3.2-2
RANKING OF PLANT OCCURRANCES BY SYSTEM

Time Frame: 26 Months (10-1-76 to 12-31-78) representing
80.5 plant years of PWR experience

<u>Subsystem Causing Outage</u>	<u>Number of Outages</u>	<u>Number of Outages Due to ICE Related Equipment</u>	<u>ICE Related Outages per Plant year</u>
Feedwater & Condensate System	218	182	2.25
Turbine Generator System	269	112	1.34
On-Site Power System	55	57	.70
Control Rod System	62	48	.60
Reactor Coolant Pump	74	40	.49
Reactor Coolant System	64	24	.30
Main Steam System	<u>78</u>	<u>20</u>	<u>.24</u>
Totals	820	483	5.92

TABLE 3.2-3
CATEGORIZATION OF ICE SYSTEM AND COMPONENT OCCURRENCES

Time Frame: 26 Months (10-1-76 to 12-31-78) representing
80.5 PWR plant years at experience

<u>System</u>	<u>Number of ICE Occurrences</u>	<u>Average Occurrences Per Plant Year</u>
<u>FEEDWATER AND CONDENSATE SYSTEM</u>	<u>182</u>	<u>2.25</u>
o Personnel Related		
Manual control during start-up	56	.70
Miscellaneous error	14	.17
o Main Feed Pumps		
Unexplained Pump trips	37	.46
Speed Control	7	.09
Miscellaneous Instruments	5	.06
o Feedwater Regulator Valves		
Unidentified manfunctions	11	.14
Solenoid	6	.07
Miscellaneous Instruments	3	.04
o Control System		
Plant transient or testing	5	.06
Miscellaneous I&CE	7	.09
o Heater Drain Pump		
Unidentified manfunctions	9	.11
Motor	2	.02
o Miscellaneous Valves	10	.12
o Miscellaneous Sensors	6	.07
o Condensate Pumps	4	.05

TABLE 3.2-3 (Continued)

<u>System</u>	<u>Number of ICE Occurrences</u>	<u>Avg. Occurrences Per Plant Year</u>
<u>TURBINE GENERATOR</u>	<u>112</u>	<u>1.34</u>
o EHC Control System	33	.39
o Trips During Testing	29	.34
o Voltage Regulator	21	.26
o Generator	19	.23
o Governor or Intercept Valves	10	.11
<u>ON-SITE POWER SYSTEM</u>	<u>57</u>	<u>.70</u>
o Inverter	17	.20
o Vital Bus	8	.10
o RCP Bus	11	.13
o Personnel Related	7	.09
o Instr. Bus	4	.05
o Battery Bus	5	.06
o Miscellaneous	5	.06
<u>CONTROL ROD SYSTEM</u>	<u>48</u>	<u>.60</u>
o CRDM Motor	18	.22
o Power Supply	13	.16
o Wiring/Cabling/Terminations	8	.10
o Fuse	7	.09
o Relay	1	.01
o Head Area Cooling	1	.01
<u>MAIN STEAM SYSTEM</u>	<u>20</u>	<u>.24</u>
o MSIV Control CKT	5	.06
o MSIV Solenoid	4	.05
o Unexplained MSIV Closure	3	.04
o Operator Close MSIV	2	.02
o Sensors	5	.06
o MSIV Motor	1	.01

TABLE 3.2-3 (Continued)

<u>System</u>	<u>Number of ICE Occurrences</u>	<u>Avg. Occurrences Per Plant Year</u>
<u>REACTOR COOLANT SYSTEM</u>	<u>24</u>	<u>.30</u>
o Loop Flow Transmitter	12	.15
o Pressurizer Pressure Transmitter	4	.05
o Loose Parts Monitor	3	.04
o Pressurizer Spray Valve	3	.04
o Pressurizer Level Transmitter	1	.01
o RTD	1	.01
<u>REACTOR COOLANT PUMP</u>	<u>40</u>	<u>.49</u>
o Low Lubrication Oil Level	27	.34
o Seal Sensing Line	5	.06
o Motor Surge Capacitor	2	.02
o Miscellaneous ICE	3	.04
o Motor Failure	1	.01
o Unexplained RCP Trip	1	.01
o Personnel Related	1	.01

TABLE 3.2-4
SITE SYSTEM RANKING SUMMATION

<u>Major Plant System/Component</u>	<u>NUMBER OF OCCURRENCES</u>		
	<u>Impacting Plant Safety</u>	<u>Impacting Plant Availability</u>	<u>DAS Application</u>
Auxiliary Feedwater System	2		4
Area Radiation Monitoring	8		
Boric Acid Injection System			
Chemical and Volume Control System	3	3	2
Compressed Air System	5	7	
Component Cooling Water System			
Condensor System		5	8
Containment Isolation System			
Containment System			
Communications System			
Reactor Core/Fuel/Vessel			
Control Rod System		4	
Circulating and Saltwater System			
Cooling Tower System			
Demineralized Water System			
Emergency Diesel Generators			
Electrical Distribution System (Grid)			
Emergency Core Cooling System	4		
Main Feedwater System	1	1	1
Main Steam System			
Process Instrumentation			6
Nuclear Instrumentation			
Natural Phenom/External Causes			
On-Site Power System			
Personnel	7	8	
Plant Computer			
Reactor Coolant Pumps			

TABLE 3.2-4 (Continued)

<u>Major Plant System/Component</u>	<u>NUMBER OF OCCURRENCES</u>		
	<u>Impacting Plant Safety</u>	<u>Impacting Plant Availability</u>	<u>DAS Application</u>
Reactor Coolent System		6	5
Reactor Protection System	6		7
Reactor Regulating System			
CEA Position Indicating System			
Sampling System			
Shutdown Cooling System			
Turbine Generator System		2	3
Waste Management System			

The top 8 systems ranked by system (1 thru 8) in decreasing order of importance for each of the three columns, based on survey of 21 site operating personnel.

TABLE 3.2-5
I&CE ENGINEERING EVALUATION SUMMARY

<u>System</u>	<u>Potential Cause</u>	<u>Events Per Plant Year</u>	<u>Duration(hrs) Per Event</u>	<u>Potential Solution/Recommendation</u>
1. Feedwater and Con-	Loss main feed pumps	.61	8.2	Improve procedures and operator training. Delay low flow (NPSH) pump trip, selective tripping. Change constant speed pumps to turbine, Provide booster pumps, Incorporate auto turbine runback.
	Loss SG level control	.70	5.8	Extended control range (F 15% Power). Backup control system Advanced Control System Apply Disturbance Analysis System techniques.
2. Turb Gen. System	Trips during test/maint.	.34	7.6	Further effort required to:
	Spurious/random EHC malfunctions.	.39	14.4	Determine unidentified root causes, Investigate use of redundant EHC, Improve test methods/procedures, Apply Disturbance Analysis System techniques.
	Loss field/voltage regulation.	.26	15.6	Further effort required to: Determine unidentified root causes. Improve reliability of voltage regulator field relays and power supplies.
	Generator bearings, stator, H ₂ cooler, ground problems.	.23	103	Improved methods of ground detection. Improved maintenance procedures. Improved H ₂ leak prevention/detection.

TABLE 3.2-5 (Continued)

	<u>System</u>	<u>Potential Cause</u>	<u>Events Per Plant Yr.</u>	<u>Duration(hrs) Per Event</u>	<u>Potential Solution/Recommendation</u>
3.2-43	3. On-Site power sys.	Loss vital/control instrument bus	.15	21.4	Two battery plants should consider going to four battery plant design.
		Loss DC/AC inverter	.20	17.1	Control instrument bus should be provided with uninterruptable DC/AC inverter and battery.
					Operating & maintenance procedures should be reviewed to ensure proper inverter operation.
		Loss reactor coolant pump bus.	.13	15.6	Evaluate use of four separate busses.
4.	Control rod sys.	CEDM coil failures	.22	49.0	Presently being resolved by applicable vendors design modifications.
		CEDM power supplies	.16	9.5	Presently being resolved by applicable vendors design changes.
5.	Reactor coolant pump	Random miscellaneous	.15	39.4	Random and few ICE occurrences does not warrant further ICE efforts.
6.	Reactor coolant system	RCS flow malfunction	.15	15.1	Evaluate using RCP speed sensor or other methods of flow measurement.
7.	Main steam sys.	MSIV control CKT. malfunction.	.06	7.0	Random problems resolved on individual plant basis, further ICE effort not warranted.
		MSIV solenoid malfunction	.05	14.0

Figure 1
 AUTOMATIC FEEDWATER CONTROL SYSTEM BELOW 20% POWER
 (TYPICAL CHANNEL)

Figure 1
 AUTOMATIC FEEDWATER CONTROL SYSTEM BELOW 20% POWER
 (TYPICAL CHANNEL)

Figure 3.2-2
TYPICAL TWO BATTERY ON-SITE POWER SYSTEM

BUS	EQUIPMENT	BUS	EQUIPMENT
1	REACTOR COOLANT PUMP (2)	10	PRESSURIZER HEATERS (BACK UP, 1 BANK, 150 KW) BATTERY CHARGER MOTOR CONTROL CENTER CHARGING PUMP
2	REACTOR COOLANT PUMP (2)	11	BORIC ACID PUMP BATTERY CHARGER
4	LOW PRESSURE SAFETY INJECTION PUMP HIGH PRESSURE SAFETY INJECTION PUMP	12	PRESSURIZER HEATERS (BACKUP AND PROPORTIONAL) MOTOR CONTROL CENTER CEDM MG SETS
5	LOW PRESSURE SAFETY INJECTION PUMP HIGH PRESSURE SAFETY INJECTION PUMP	13	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS
7	PRESSURIZER HEATERS (BACKUP AND PROPORTIONAL) MOTOR CONTROL CENTER CEDM MG SET	14	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS
8	BORIC ACID PUMP BATTERY CHARGER CHARGING PUMP	15	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS
9	PRESSURIZER HEATERS (BACK UP, 1 BANK, 150 KW) BATTERY CHARGER MOTOR CONTROL CENTER CHARGING PUMP	16	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS

Figure 3.2-3
TYPICAL FOUR BATTERY ON-SITE POWER SYSTEM
3.2-47/3.2-48

SECTION 3.3

SYSTEMS INTEGRATION EVALUATION

3.3 SYSTEMS INTEGRATION EVALUATION

3.3.1 Introduction

The Systems Integration portion of this study overviewed all reactor plant, steam plant, and auxiliary support systems. The Component Failure Study was expanded to include a system level evaluation because: (1) the root cause of a component failure in many cases lies within the system design and not within the failed component and, (2) in many cases, the resolution of a system problem will also provide relief to component failures (e.g. fluid induced vibrations may cause valve failures; reduced vibrations improve component performance).

Assumptions, evaluations, and recommendations pertain only to the system as a whole, including all its individual components: pumps, valves, piping, power supplies, control and protective circuits and functions. Individual component failures were considered in the scope of the system study only if they contributed to a scram or forced outage due to a non-mechanical failure (e.g. the failure of a pump because of a low suction pressure trip actuated by a loss of flow to the pump as opposed to the mechanical failure of a pump bearing or seal).

The turbine generator and main feedwater and condensate systems were considered for further evaluation because of their respective impact on plant reliability, availability, and therefore plant safety. The turbine generator system was the major contributor to plant unavailability while ranking second to the main feedwater and condensate system with respect to the number of scrams and forced power reductions per plant year.

Although the C-E Data Base related a large number of turbine generator system problems to mechanical failures, the operation of the turbine steam system directly effects steam generator water level, and therefore the operation of the main feedwater and condensate system. Additionally, both

systems interface within the main condenser where their interaction significantly effects the stability, operability, and reliability of the condensate and feedwater system, and therefore the safety of the reactor plant.

The main feedwater and condensate system, when compared with all systems, most frequently challenged the reactor protective system and caused the largest number of forced power reductions. In addition, initial scoping of the C-E Data Base showed a large number of reactor scrams resulting from steam generator high or low water level trips, with the recorded cause being unknown or attributable to non-mechanical component failures.

Because the budget and time restrictions of this study precluded the in-depth evaluation of system level root causes for all systems, it was felt that the main feedwater and condensate system most warranted further analysis due to the magnitude of this system's effects on both plant safety and performance. However, the scope of future studies should consider the turbine generator system analysis with consideration given to an evaluation of system deaeration performance, air-inleakage into the vacuum portions of the condenser and condensate systems, and corrosion product source detection, removal, and control to decrease the amount of corrosion products which, when transported into the feedwater cycle, cause feedwater system component and steam generator problems.

The evaluation of the main feedwater and condensate system enabled the major failures to be attributed to several suspected system level root causes. Because of the component orientation of the overall study and the available data, engineering verification for the system level root causes was not as readily ascertained as for component root causes. Therefore, corroboration was sought from previously completed studies and through operator interviews. These sources provided excellent independent verification of the root causes and enabled specific recommendations to be made. In general, these recommendations indicate that a comprehensive system level optimization study of PWR feedwater and condensate systems should be

made, focusing on improving existing and future system design. It is expected that such an evaluation would greatly increase plant reliability and availability, and therefore plant safety.

3.3.2 Scope and Methodology

The objective of the Systems Integration Study was to determine system level root causes for those events that resulted in scrams, plant unavailability, and power reductions.

In order to pursue a systems level study based on component oriented data, it was necessary to first define the scope of the effort. This study was primarily concerned with the evaluation of system related events; that is, events that were reported as being caused by non-mechanical component failures, events reported with an undefined cause, and events caused by the actuation of process parameter monitoring equipment (e.g. steam generator high and low water level trip setpoints). In addition, certain mechanical component failures were considered only with regard to their impact on power reductions, since system design might limit the magnitude or duration of the reduction.

The primary source of data was the C-E PWR Shutdown Data Base. The Data Base listed for each event: the type of shutdown, the outage duration, the reported cause, and a brief description of related information. Using the Data Base, all PWR process systems were ranked with respect to those occurring events that impacted plant safety or performance. Thus, each system was ranked first with respect to the number of times it challenged the reactor protective system (i.e. reactor scrams), as an indication of the effect on plant safety. Then each system was ranked according to its contribution to plant unavailability and its contribution to the number of power reductions experienced per plant year, as an indication of the effect on performance.

Based on the rankings described above, further evaluations of system level root causes were then limited to the one or two systems ranked high in the three categories.

The existing Data Base format presented limitations with respect to a systems evaluation; the reported data was component oriented and there was no accountability for the relative MWh(e) lost as a result of experienced power reductions. Further, the systems evaluation was limited because of the diversity in design of plant secondary systems, which precluded the timely comparison of all designs.

These limitations made the engineering verification of system level root causes difficult. Therefore, once several suggested root causes were determined, outside verification was sought from the Limiting Factors Analyses (reference 1, 2 & 3), operator interviews, and other previously completed studies (e.g. Reference 4). These outside sources confirmed the suggested root causes and helped provide the bases for the specific recommendations discussed at the end of this section.

3.3.3 Evaluation

During the initial investigation of available data, both primary and secondary systems were categorized with respect to the number of challenges to the reactor protective system originating from a given system, that system's contribution to overall plant unavailability, and the number of power reductions attributable to a specific system.

The main feedwater and condensate system was selected for further evaluation because it ranked first in both the number of scrams and power reduction per plant year and ranked fifth as a contributor to overall plant unavailability, as shown on Tables 3.3-1 through 3.3-3. With 2.56 and 2.97 scrams and power reductions per plant year, respectively, the main feedwater and condensate system was considered to have the largest impact of any process system on plant safety and performance.

Additionally, it should be noted that the relatively low unavailability ranking of the condensate and feedwater system was not a true indication of

that system's total impact on plant performance. Availability considers only shutdown hours while a large number of power reductions may cumulatively account for significant lost power generation.

Main feedwater and condensate system failures resulted in twice the number of scrams reported when compared to all other PWR systems (Table 3.3-1). More importantly, from a systems viewpoint, Table 3.3-4 shows that 61 percent of the scrams resulting in shutdowns are attributed to steam generator level trips and various problems with pump operation and feedwater flow control, not to the mechanical failure of individual components. Almost 50 percent of the steam generator level trips occurred during plant startup (J15% power), when feedwater flow is manually controlled by the operator, with the remainder occurring at high power levels with feedwater flow in automatic control.

The common non-mechanical feedwater pump failure mode was the low suction pressure trip. Further investigations of these events and discussions with plant operators indicated that a significant contributor to both pump and steam generator level trips was the temporary low feedwater flow and low suction pressure condition resulting from dynamic and static fluid instabilities in the feedwater and condensate system. Plant operators also complained of the difficulty of manually controlling feedwater flow at low power levels.

The suggested major cause of low power feedwater control problems is that current feed and condensate system designs emphasize efficiency at high power; therefore, the size and capacity of components are specified for flow-rates as high as 21,000 gpm per steam generator while low power (e.g. 5%) requirements may only require 1000 to 2000 gpm per steam generator. Hence, the inability of large components to effectively control small fluid flows. It appears that the design philosophy and operational interaction of the condensate, feedwater, and steam generating subsystems alone may be the root cause of many plant trips.

With 2.97 power reductions per plant year (Table 3.3-3), the main feedwater and condensate system was the initiator of more power reductions than the summation of the remaining listed systems. Despite the limited information available regarding lost generating capacity resulting from power reductions, it is felt that their contribution to decreased plant performance is very significant. For example, Reference (3) indicated that for one plant 1,217,343 Mwh(e) of lost productivity was attributable to power reduction and short term maintenance shutdowns over a two year period.

When evaluating the reported feed and condensate system events that resulted in power reductions, it should be noted that the system related steam generator level and feedwater pump trips contributed to 61 percent of the scrams but only accounted for 9 percent of the system power reductions (Table 3.3-4). Table 3.3-5 shows that with the inclusion of data related to condensate and feedwater system mechanical component failures, the percentage of reported power reductions increases to 65 percent while the percentage of plant scrams only increases by 15 percent, from 61 to 76 percent.

These mechanical failures, which are related to singular components (i.e. pumps, pump seals, lube oil systems, speed controllers, etc.), suggest that improved system design (e.g. more component redundancy) may decrease the magnitude or duration of experienced power reductions. For example, a plant with two 50 percent capacity main feedwater pumps will experience a 50 to 60 percent power reduction if one pump is stopped, whereas, a plant with three 50 percent pumps could remain at 100 percent power.

Redundancy is only one facet of a system's configuration. Table 3.3-6 contains the results of a limited survey of several operating PWR plants revealing a great diversity in feed and condensate system configurations and control schemes. Unlike primary system designs, secondary process system designs are not designed for as high a reliability and thus, do not reflect uniform redundancy and diversity.

There appears to be no systematic approach to control system and equipment protective setpoints. In many cases, the current configurations are such

that the loss of one component will cause the chain reaction loss of associated components (Domino effect), usually resulting in a plant trip. Component and system configuration appears to be directly related to both the number of power reductions and challenges to the reactor protective system a PWR may experience.

3.3.4 Suggested Root Causes

Based upon the previous evaluation, it is suggested that three major system root causes are the primary basis for those events originating in the condensate and feedwater system that result in plant scrams and power reductions.

The Non-Optimization of Feed Train Configuration

Based on scoping system design comparisons, design integration of feed train components to systems requirement is insufficient. There appears to be some relationship between the number and type of events experienced and the configuration of the plant's feedwater and condensate system. A minimum of uniformity was found in component redundancy, capacity, motive power and speed control systems. Control and equipment protective setpoints appeared related to the protection of individual components, with very little consideration allowed for the effect these setpoints have on overall plant operation. The backfitting of systems and components (e.g. condensate filters) to the condensate and feedwater system appears to degrade system performance and reliability. Additionally, the reliability of control systems and power supplies are not evaluated for emergency conditions unless the system is required for the safe shutdown of the reactor (e.g. auxiliary feedwater system). Finally, the absence of redundancy encourages power reduction and degraded plant performance when the maintenance and testing of components is performed during plant operation.

The Non-Optimization of Low Power Feedwater Control

Current system designs emphasize efficiency at full power operation resulting in large components and capacities, control system sensitivity for large

components and fluid flowrates, and fixed protective system setpoints. This results in extremely difficult manual operations and numerous challenges to the reactor protective system during operation at low power.

Non-Optimum Hydraulic Resiliency

PWR feed and condensate system design appears to lack resiliency to small flow and pressure instabilities. Anticipatory pump trips assume the worst condition has occurred. For example, the feedwater pump low suction trip anticipates a total loss of flow but may actuate during a temporary low suction pressure condition. This results in the unnecessary loss of feedwater flow usually followed by a reactor trip on steam generator low water level.

3.3.5 Verification

Component failure verification was undertaken through direct plant contact and a review of plant operating and maintenance reports. Because industry data is component oriented with respect to event causes, system level causes were difficult to verify. Therefore, the corroboration to support the suggested root causes was sought from operator interviews, the EPRI Limiting Factors Analyses, References (1, 2 & 3), the nuclear related section of the EPRI Feed Pump Outage Survey, Reference (4), and the U.S. NRC "Staff report on the Generic Assessment of Feedwater transients in Pressurized Water Reactors designed by the Babcock & Wilcox Company" NUREG-0560, Reference (5). The following discussions related to these sources are supportive of our findings.

Operator interviews conducted at two plants pointed to the main feedwater and condensate system as the system of greatest operator concern with respect to plant safety and unavailability. The operators stressed the difficulty in manually controlling feedwater flow at low power levels and the general sensitivity of low power feedwater system operation.

At one plant, a time delay feature was added to the feedwater pump low suction pressure trip to improve the pump's ride-through capability during small pressure transients in order to prevent a loss of feed flow and an unnecessary reactor trip. Although installed only within the last year, this mechanism appears to be decreasing the number of steam generator low water level trips experienced.

The Limiting Factors Analyses recommended a feedwater and condensate system evaluation because of the number of scrams initiated within this system. They recommended the optimization of system configurations and recognized that present configurations contribute to power reductions. It was noted that one plant under observation installed a third condensate pump to decrease the number of power reductions associated with pump maintenance and repair.

These studies recognized that the conservative sizing of valves contributes to a valve's inability to control small flows and in many cases leads to valve failure. They also emphasized the importance of considering system induced fluid vibrations in initial design analyses to prevent recurring component failures. The analyses also recommended the improvement of low power feedwater system operation and stress that system and component interrelationships are not easily assessed within the component approach.

The EPRI Feed Pump Outage Survey (Reference 4), although Fossil oriented, addressed both BWR and PWR plant main feedwater pumps. A result of this survey was the conclusion that feedwater system designs emphasizing full power operation produced unfavorable performance at part load operation, causing hydraulic instabilities and fluid vibrations that were related to component failures. Finally, this study concluded that unnecessary steam generator water level trips result from the improper balance of pump, control valve, and control system in feedwater and condensate system design.

NUREG-0560 (Reference 5) addresses the diversity of design of various feedwater subsystems with particular attention to condensate demineralizers. This study noted that demineralizer bypass flow control can lead to loss of

feedwater events. Also, because of the diversity in feedwater system design, this report recommended that a study of "main feedwater system design, operating procedures, and service procedures could lead to a means to reduce the frequency of loss of feedwater events".

3.3.6 Recommendations

3.3.6.1 General

The basic recommendation resulting from the evaluation of system level failures is that PWR feedwater and condensate and turbine generator systems be the subjects of detailed system level optimization studies focused on improving existing and future system designs, availability and reliability, and therefore improving overall plant safety. As previously discussed in Section 3.3.1, the interactions of both systems warrant that future turbine generator system level analyses should consider, as a minimum, the evaluation of system deaeration performance, air-inleakage into the vacuum portions of the condenser and condensate systems, and corrosion product source detection, control, and removal to decrease the amount of corrosion products which, when transported into the feedwater cycle, cause feedwater system component and steam generator problems.

Specific recommendations regarding areas for the main feedwater and condensate system level investigation are detailed below. Specific Turbine/Generator system recommendations are addressed in section 3.1.3.

3.3.6.2 Specific

Feedwater and Condensate System Configuration

More reliable and efficient feedtrain configurations are needed to improve the performance and availability of PWR plants. Existing system designs should be surveyed and event failure profiles established for the various configurations. Unique configurations that are susceptible to specific

events should be identified. Design comparisons among the various U.S. configurations and with foreign systems, such as the representative design of Figure 3.3-1, should be made with respect to design philosophies, system operation, maintenance, and testing. Based on the above, optimization guidelines should be developed for application to both existing and future NSSS designs. These guidelines should address:

- o redundancy of major components
- o sizing of major components
- o choices of motive power
- o power supply and control system reliability
- o reliability degradation due to backfits
- o operator-machine interfaces (operation and maintenance).

Low Power Feedwater Operation

Low power feedwater control capabilities should be improved in order to reduce the high number of steam generator low level trips and the corresponding challenges to the Reactor Protective System. Fixed control/protective system setpoints based on full power operation should be evaluated regarding the use of variable (or reduced) setpoints during low power operation. An analyses of steam generator level setpoint margins over the full range of power operation would identify reductions in steam generator inventory requirements at low powers and allow expansion of the level control band. Additionally, possible design changes aimed at improved manual feedwater control and/or automatic feedwater control at low power should be evaluated. Figure 3.3-1 illustrates one philosophy with respect to low power feedwater control through the use of a small capacity feedwater control valve, condensate and feedwater pumps.

In many cases, the addition of wide range level indication coupled with the above mentioned setpoint reductions could provide significant improvements in manual feed control. Improvements in automatic feedwater control systems to expand the ranges of reliable response over the full range of power operations should also be evaluated.

System Hydraulic Resiliency

An evaluation should be made of possible methods for increasing the feedtrain's capability to accept transitory hydraulic instabilities and variations without challenging the reactor protective system. Computer codes which combine best estimate secondary system modeling with existing NSSS performance models should be employed in examining Feedwater and Condensate System designs. For example, the application of surge/deaerating feedwater tanks and main feed pump selective or time delayed trips should be examined. Figure 3.3-1 again illustrates the foreign philosophy with respect to the use of the deaerator storage tank as a source of NPSH for the three 50% capacity feedwater pumps.

The interfacing of component level equipment protective setpoints (e.g. condensate and feed pump trips) and NSSS reactor trip setpoints should be optimized to decrease the extent of overall plant outages resulting from individual component problems (Domino effect). Additionally, potential improvements in system design performance over the full range of power operations should be evaluated in order to improve load following during forced power reductions.

REFERENCES FOR SECTION 3.3

- 1) Limiting Factors Analysis of a High Availability Nuclear Power Plant Study Performed at Maine Yankee Atomic Power Plant EPRI RP894-2.
- 2) Limiting Factors Analysis of High Availability Nuclear Plants Study performed at Oconee Nuclear Station - Unit 1 (Oconee 1) EPRI RP894-3.
- 3) Limiting Factors Analysis of High Availability Nuclear Plants Study performed at Turkey Point Unit 3, EPRI RP894-4.
- 4) Survey of Feed Pump Outages, EPRI-FP-754, April, 1978.
- 5) NUREG-0560, Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Design by the Babcock & Wilcox Company, May, 1979.

TABLE 3.3-1
ALL PWR SYSTEM FAILURES RESULTING
IN PLANT SCRAMS
OUTAGE DATA (7/61 - 12/78)

	<u>SCRAMS/PLANT YEAR*</u>
1. MAIN FEEDWATER AND CONDENSATE SYSTEM	2.56
2. TURBINE GENERATOR SYSTEM	1.50
3. ON SITE POWER SYSTEM	0.76
4. REACTOR PROTECTION SYSTEM	0.48
5. CONTROL ROD DRIVE SYSTEM	0.42
6. MAIN STEAM SYSTEM	0.37
7. REACTOR COOLANT SYSTEM	0.29

TOTAL PWR PLANT YEARS IN DATA BASE = 192.4

TABLE 3.3-2
 MAJOR PWR PLANT UNAVAILABILITY CONTRIBUTORS
 PLANT OUTAGE DATA
 (7/61 - 12/78)

<u>SYSTEM</u>	<u>AVERAGE UNAVAILABILITY*</u>
1. TURBINE GENERATOR SYSTEM	3.76%
2. MAIN STEAM SYSTEM	2.67%
3. REACTOR COOLANT SYSTEM	1.84%
4. MAIN FEEDWATER AND CONDENSATE	1.20%
5. REACTOR VESSEL INTERNALS AND CORE	1.20%
6. ON SITE POWER SYSTEM	0.85%
7. CONTROL ROD DRIVE SYSTEM	0.77%

*AVERAGE CONTRIBUTION TO UNAVAILABILITY PER PLANT YEAR.

TABLE 3.3-3
PWR SYSTEM FAILURES
RESULTING IN POWER REDUCTIONS
OUTAGE DATA (7/61 - 12/78)

	<u>POWER REDUCTIONS/ PLANT YEAR*</u>
1. MAIN FEEDWATER AND CONDENSATE SYSTEM	2.97
2. TURBINE GENERATOR SYSTEM	1.03
3. CONTROL ROD DRIVEN SYSTEM	0.18
4. REACTOR COOLANT SYSTEM	0.14
5. MAIN STEAM SYSTEM	0.12
6. ON SITE POWER SYSTEM	0.10

*TOTAL PWR PLANT YEARS IN DATA BASE = 192.4

TABLE 3.3-4
 MAIN FEEDWATER AND CONDENSATE
 MAJOR PROBLEM PROFILES - SYSTEM RELATED
 PLANT OUTAGE DATA (1/61 - 12/78)

	AVERAGE ANNUAL UNAVAILABILITY <u>CONTRIBUTION</u>	SCRAMS PER PLANT <u>YEAR</u>	POWER REDUCTION <u>PER PLANT YEAR</u>
FEEDWATER CONTROL			
FWCS (MANUAL/AUTOMATIC)	0.07%	0.17	0.01
SG LEVEL TRIPS (Unreported Power Level)	0.06%	0.61	0.00
(During Startup)	0.03%	0.55	0.00
FEEDWATER PUMP			
PUMP TRIPS (Most to Low Suction Pressure)	0.06%	0.24	0.26
TOTALS	0.22%	1.57	0.27
	=====	=====	=====
SYSTEM TOTAL (FROM TABLES 3.3-1, 2 & 3)	1.20%	2.56	2.97
PERCENT OF SYSTEM TOTAL	18.3%	61.3%	9.1%

TABLE 3.3-5
MAIN FEEDWATER AND CONDENSATE
MAJOR PROBLEM PROFILES - CONFIGURATION RELATED

	<u>AVERAGE ANNUAL UNAVAILABILITY CONTRIBUTION</u>	<u>SCRAMS PER PLANT YEAR</u>	<u>POWER REDUCTION PER PLANT YEAR</u>
FEEDWATER CONTROL			
FWCS (MANUAL/AUTOMATIC)	0.07%	0.17	0.01
SG LEVEL TRIPS (UNREPORTED POWER LEVEL)	0.06%	0.61	0.00
(DURING STARTUP)	0.03%	0.55	0.00
FEEDWATER PUMP			
PUMP TRIPS (LOW SUCTION PRESSURE)	0.06%	0.24	0.26
SPEED CONTROLLER	0.08%	0.06	0.08
MISCELLANEOUS PROBLEMS/MAINTENANCE	0.02%	0.05	0.08
PUMP MECHANICAL PROBLEMS/SEALS	0.01%	0.02	0.19
PUMP LUBE OIL SYSTEM PROBLEMS	0.01%	0.04	0.07
PUMP SUCTION STRAINER PROBLEMS	0.01%	0.02	0.49
CONDENSATE/CONDENSATE BOOSTER PUMP	0.01%	0.05	0.09
HEATER DRAIN PUMP	0.01%	0.05	0.18
HEATER DRAIN TANK SYSTEM	0.02%	0.06	0.05
FEEDWATER HEATER LEAKS	0.04%	0.01	0.09
MAIN CONDENSOR			
TESTS/MAINTENANCE	0.01%	0.01	0.16
CLEANING/BACKWASHING	0.00%	0.00	0.17
TOTALS	<u>0.45%</u>	<u>1.94</u>	<u>1.92</u>
SYSTEM TOTAL (FROM TABLES 3.3-1, 2 & 3)	1.20%	2.56%	2.97%
PERCENT OF SYSTEM TOTAL	47.5%	75.8%	64.6%

TABLE 3.3-6

EXAMPLE: CONFIGURATION DIVERSITIES

	<u>PLANT A</u>	<u>PLANT B</u>	<u>PLANT C</u>	<u>PLANT D</u>
DATE OF OPERATION:	6/74	12/71	8/73	7/73
MFW PUMPS (NO./CAPACITY EA.):	3/50%	2/50%	2/50%	2/50%
DRIVER:	MOTOR	STEAM	STEAM	STEAM
COND. PUMPS (NO./CAPACITY EA.):	3/50%	2/50%	3/33%	3/50%
COND. BSTR. PUMPS (NO./CAPACITY EA):	0	0	0	3/50%
STEAM GEN. (NO./TYPE):	2/U-Tube	2/U-Tube	4/U-Tube	2/OTSG
STEAM GEN. LEVEL TRIPS:	HI/LOW	HI/LOW	HI/LOW	HI ONLY

Figure 3.3-1

REPRESENTATIVE FOREIGN CONFIGURATION

3.3-20

SECTION 4

COST-BENEFIT ANALYSIS

4.0 COST BENEFIT ANALYSIS

INTRODUCTION

The "availability" of a base loaded power plant is an important measure of its economic and engineering success. Annual availability is measured by the percentage of the 8760 hours available in the year that the plant is capable of producing electricity. Effectiveness of a plant to continuously produce its design power is measured by the plants capacity factor, defined as the ratio of energy generated during a period to the energy which could have been generated if operating at maximum capacity throughout the year.

Nuclear fuel costs are lower than most alternative fuels. Any interruption in nuclear generation will generally result in higher production costs for the utility. Under certain conditions nuclear plants may be used for cycling (Reference 1) but typically they are operated in the base loaded mode.

There are three major incentives to increase nuclear plant availability and capacity factor.

1. The fewer shutdowns that are associated with increased availability improve nuclear safety by imposing fewer challenges to the reactor protective systems and after trip heat removal systems.
2. With increased capacity factor, the same load demand can be met with fewer plants, thereby decreasing capital expenditures for new plants.
3. As fewer incidents requiring shutdown occur to a base load plant, less high cost replacement power will be required from other sources, especially oil.

In order to rationally evaluate programs designed to increase nuclear plant capacity factor, it is necessary to quantify the costs and benefits of such improvements. The costs of improvements typically include engineering design, hardware, and installation costs. The benefits of averting a

nuclear plant shutdown include money not spent for replacement power during plant shutdown, money not spent repairing failed equipment causing the shutdown, and money not spent due to a reduced need to have additional generation capacity available. The major factors affecting the amount of money saved are differential fuel costs and the frequency and duration of outages.

For an improvement program to be cost beneficial, the benefits must exceed the costs. The term used to describe the tradeoff of costs and benefits is the net benefit. A positive net benefit indicates that the program will be economically beneficial and a negative net benefit suggests a economic loss will be incurred if the proposed program is implemented.

The objective of this cost benefit analysis was twofold:

1. To provide benefit sensitivity curves which would enable a user to estimate the dollar value of increased availability compounded over the remaining life of a nuclear plant.
2. To develop a rigorous methodology for the quantification of net benefit where uncertainties exist in major inputs to the analysis.

The cost benefit portion of this project was separated from the remainder to permit a concentrated and in-depth analysis of the cost benefit factors involved. Utilizing computer techniques, certain cost and benefit factors could be quantified allowing more precise assessment of these factors. In addition, utilization of the computer allows the uncertainty in individual factors to be collectively analyzed.

A method to quantify the benefits of increased availability was developed. This method, implemented on the computer, was then used to generate cost sensitivity curves showing the cost savings from a decrease in annual outage time due to availability improvements.

The use of single valued estimates is impractical where large uncertainties are considered in fuel cost, capital, escalation, outage frequency and

outage duration. Further refinements utilizing improved analytical techniques permit a range of values and associated uncertainties to be used for each factor. In turn, the resultant computer output is a calculated range of values reflecting the combined uncertainties of all input factors and represents the probability range of cost benefit from a proposed modification.

Section 4.1 is a description of the benefit factors and the sensitivity curves. Net benefit considerations, uncertainty and model development are described in Section 4.2 and the cost benefit summary and conclusions are provided in Section 4.3.

The Appendices give more detailed descriptions of the cost benefit models and the assumptions used.

4.1 BENEFIT OF INCREASED CAPACITY FACTOR

This section describes the methodology used to quantify the benefit from increased capacity factor resulting from shutdown avoidance. Sensitivity curves that relate annual shutdown hours to dollar benefit are also developed.

In order to quantify the benefits of increased capacity factor, dollar values are assigned to the savings from three sources:

COST OF REPLACEMENT POWER SAVED
FUTURE CAPITAL EXPENDITURES SAVED
OUTAGE REPAIR COSTS SAVED

A model was developed to handle the data required to quantify the benefit of improved availability. The BENEFIT calculational model sums the replacement power costs, the capital avoidance costs, and the direct outage costs over the plant lifetime. It then determines the present worth of the future benefit savings in 1979 dollars. This can be shown by the equation below:

$$\text{Benefit Present Worth} = \sum_{n=1}^{n=RPL} \frac{1}{(1 + DR)^n} (AB)$$

where:

RPL = the remaining plant life (years)

DR = the discount rate, also defined as the weighted cost of capital

AB = ANNUAL BENEFIT = the sum of the yearly cost savings due to improvements made = Replacement Power Costs + Capital Avoidance Costs + Direct Outage Costs. (dollars per kilowatt)

Since this equation deals with the present worth, in 1979 dollars of future savings, inflation is excluded in the discount rate. Only the escalation rate is used when summing the benefits. The remainder of this section discusses the components of the annual benefit calculation. A more detailed explanation of the benefit equation can be found in Appendix 4-A. Values assumed for use in the benefit equation are given in Appendix 4-B with their sources.

4.1.1 Replacement Power Costs

One of the major benefits of improving capacity factor is the reduction of the use of replacement power. When a nuclear plant is shutdown or forced to operate at low power level, it is necessary to replace the lost power by generating power from other sources or by buying it from an adjacent utility.

For each type of power generation there are characteristic fuel costs and operation and maintenance (O&M) costs (both in \$/kwh). These O&M costs are often called variable costs and are proportional to the amount of electricity produced. Nuclear units are characterized by lower fuel costs than fossil units. Therefore a large additional expense is incurred during the shutdown

of a nuclear unit due to the differences in fuel costs between the base loaded nuclear unit and the replacement power fuel costs. Improving the capacity factor will reduce this expense.

The timing of equipment failures requiring plant shutdown or power reduction affect the cost of replacement power. Regardless of the length of advance notice prior to shutdown, the lost generation needs to be replaced almost immediately on most power grids. The replacement generation will always be more expensive since the less expensive units are already be in full power operation. Furthermore, quick startup units such as combustion turbines can provide power but only at increased cost due to higher fuel costs (\$/Btu) and less efficient utilization of that fuel (Btu/kwh). As less expensive generating units can be brought up to power, the cost of replacement power will decrease but still remain above the generating cost of the nuclear unit. Further details can be found in References 2 and 3.

For this study the timing related effects are treated in a way that assumes that replacement power is generated by combustion turbines during the first six hours following the nuclear plant shutdown. This assumption can be refined by considering the system load as a function of daily, weekly and seasonal cycles plus the maximum rate at which generation replacement can be brought to power. References 4 and 5 detail one such refined approach.

Utilizing the timing techniques above adds considerable complexity to the model. Furthermore, the sensitivity of costs to the length of advance notice is likely to be highly dependent on the generation mix and the unit capabilities within a utility control area. This effect substantially increases the power cost during the first six hours of an outage. This may be an unrealistic assumption for utilities which have available pumped hydro storage capacity or significant oil fired steam turbine units on spinning reserve.

Other factors which support the retention of the six hour penalty method are:

1. The future generation mix will become increasingly dependent upon coal fired units which are difficult to load rapidly (oil replacement).
2. Reserve margins have been decreasing, making it more likely that low cost generating units will not be available using any fuel.
3. Any outage will increase the stresses on plant equipment thus decreasing its life. Rapid loading of peaking units reduces their life. These effects are not accounted for in this analysis.
4. Few nuclear plant shutdowns will last less than 8 hours.
5. Future load management efforts to level power demand may reduce the significance of daily and weekly cycles (not considered).
6. Costs for longer duration outages may be underestimated as more generation may be required from combustion turbines during peak load periods than is reflected by the installed generation mix (not considered).

The replacement power cost portion of the benefit model (Appendix 4-A) can be simplified by the relationship:

$$\begin{aligned} \text{Replacement Power Cost} &= \text{short term replacement power costs} \\ &\quad + \text{long term replacement power costs} \end{aligned}$$

Assumptions which apply to the replacement power cost relationship include:

1. Initial short term replacement power will be provided by high cost quick startup units. These are represented by combustion turbines, the highest fuel cost units.
2. The initial short term penalty is ended after 6 hours into the incident since other sources can be brought on line within that time.

3. Other units brought on line after the first 6 hours will be assumed to be the projected 1980 national generation mix (less the nuclear portion). The generation mix (or generation mix ratio, GMR) is the ratio of each of the different generation types to the total which make up the generation network. The national average GMR's were used in this analysis (See Appendix 4-B).
4. The generation mix ratios are assumed to remain constant at their 1980 values. As no estimate for the future installed generation mix is available beyond a 10 year horizon, the complexity of varying the generation mix with time was not considered to be worthwhile.
5. The power costs are broken into fuel costs plus operating and maintenance (variable) costs. Fuel cost is cost per million BTU of the fuel considered multiplied by the average heat rate for the each type of generation considered. Heat rates are a measure of thermal efficiency and are measured in BTUs per kilowatt hour (Fuel Cost = $\$/\text{BTU} \times \text{BTU}/\text{kwh} = \$/\text{kwh}$).
6. Operating and Maintenance costs for the affected unit are assumed to continue at the average annual rate while the unit is shutdown.

4.1.2 Avoidance of Future Capital Expenditures

Capacity factor improvement reduces the amount of capital expenditures required to provide adequate reserve margin. Capital expenditures are normally considered fixed costs and thus are not relevant to cost-benefit relationships. However the time span used in this study (plant remaining lifetime) causes these costs to become relevant. The capital expenditures represented in this model are future costs which are avoided by capacity factor improvement.

Future capital expenditures are important because they comprise a very large portion of the total cost of power production. These costs are time

sensitive since they have historically increased faster than the general inflation rate. The magnitude of these costs depend upon the regulatory environment, which causes the rate of increase to be highly uncertain. Values used for this study are in Appendix 4-B.

Assumptions used for future capital expenditures are:

1. Capital cost for a 1000 MWe nuclear unit were used as a basis to estimate future capital expenditures per KW.
2. Capital cost rate of change (exclusive of inflation) is assumed to be fixed throughout the forecast period.
3. Excluded from the capital costs are the following: main transformer, switchgear and transmission facilities; owner's costs, including consultants, site selection, etc.; fees and permits; state and local taxes; spare parts; and contingency. Many of these items should be included in existing plant modification, but were not included in this study. The benefit would be greater if these factors were included, since they are part of new plant construction costs.
4. Interest during construction (i.e. cost of money) is included.
5. An annual charge is calculated using the levelized annual fixed charge rate concept. This includes interest and dividend charges (Reference 6).

4.1.3 Avoidance of Direct Outage Costs

The direct outage costs are those costs incurred to repair the equipment causing the incidents which lead to the outage or power reduction. This includes labor and materials for the repair or correction of the problem. For the majority of the incidents investigated, these costs could be considered significant but not large enough to affect the results. The replacement

power and avoidance of future capital expenditure costs are much larger. Provisions for direct outage costs were included for the instances where large or costly component repairs are involved.

Assumptions made for Direct Outage Costs are:

- o Cost of repair parts remains constant
- o Cost of labor remains constant.

4.1.4 Sensitivity Curves and Conclusions

Sensitivity curves were made to demonstrate the following relationships:.

- o Cost of downtime vs remaining plant life for 3 types of replacement power. (Figure 4-1).
- o Cost of downtime vs. days/year of downtime for 4 different remaining plant lifetimes. (Figure 4-2).
- o Cost of downtime vs. remaining plant lifetime for capital cost and replacement power cost components. (Figure 4-3).

The following assumptions apply to the above curves.

- o Best estimate data values were used (See Appendix 4-B).
- o A single 1000 MWe PWR was the assumed base plant.
- o The first six hours of replacement power is assumed to be generated by gas and distillate combustion turbines.
- o The national average generation mix was used in calculating the replacement power costs during the remaining eighteen hours of the outage.
- o Repair costs were not included in the calculation of the downtime costs.

These curves enable the user to compare the benefit of increased availability to the costs of implementing a proposed solution. Figures 4-1, 4-2, and 4-3 show the benefit of avoiding one day of shutdown per year over the lifetime of a 1000 MWe PWR. This value is approximately 32 million dollars for 40 years.

Figure 4-1 shows the economic effects of different replacement power generation types for a one day outage per year. The upper curve is based upon all replacement power being generated by gas or distillate combustion turbines. The center curve uses the weighted average of replacement power used in the benefit computation algorithm; during the first six hours combustion turbines are used for replacement power and the remaining eighteen hours use the national generation mix (see Appendix 4-B). The bottom curve is for base and intermediate peaking units used for replacement power generation. These include oil fired steam turbine, coal fired steam turbine, gas fired steam turbine and nuclear fueled steam turbine plants. For outages longer than the 24 hours, assumed in this curve, the weighted average line would approach the base and intermediate peaking line. This is because the contribution due to the peaking units used during the first six hours would become proportionately smaller during longer outages.

Figure 4-2 demonstrates the downtime cost as a function of outage duration up to 100 days per year. The four curves are for various remaining plant lifetimes. The shallow slope of the curves for short outages is due to the high replacement power costs during the first six hours of the outage. It is important to note that this curve is for a single 24 hour outage, not a sum of the downtime over the course of a year. The cost is the sum of each year's contribution over the remaining life of the plant.

Figure 4-3 shows the contribution of capital avoidance costs and replacement power costs to the total cost over the remaining plant lifetime. Note that (A) the penalty replacement power cost (for 6 hours generation) is almost equal to (B) the non-penalty replacement power costs (18 hours generation), and that the contribution (D) due to capital cost avoidance is larger than that due to (C) the replacement power costs. Further, the replacement power costs are approximately 75% of the total costs.

A computer program called BENEFIT was used to calculate the cost savings from avoiding one day of down time per year over the total remaining lifetimes of the 40 operating PWR's in the United States. These savings are summarized

in table 4-1. Cost savings are in 1979 dollars for this table and the rest of the analysis. This is done to include escalation of costs and to avoid predicting the inflation rate over the next 40 years.

4.2 NET BENEFIT APPLICATION METHODOLOGY WITH UNCERTAINTY

In assessing improvements for power plants a quantitative cost benefit analysis is desirable. When a decision is to be made between several different options for improving plant availability, each option will have an implementation cost and an availability benefit which can be estimated. However, in most cases the costs and benefits of proposed decisions cannot be estimated with certainty. This is particularly true for improvements which have not completed the development through demonstration cycle. Future fuel costs have a high uncertainty. An analysis which considers a 40 year period should have a probable range as well as an expected value. The objective is to develop a specific methodology to aid the decision process and to consider the uncertainty involved.

The method should also use the uncertainties involved with component failure rates and cost estimates. While a parametric study is adequate for observing trends when one parameter is varied while the others remain constant, this method is inadequate when many parameters contain uncertainties. An approach should be used that will include the added dimension of uncertainty. Probability distributions, describing the likelihood of a parameter being at a certain value within a prescribed range, should be set. Several input parameters can be "varied" simultaneously and the result will be a range of possible values. The results obtained will quantitatively indicate the likelihood of the resultant at intervals within a range of possible values.

A computer program developed by Combustion Engineering enables up to 16 probability distributions to be combined. This program is called SIGMA. A more detailed description of the SIGMA code is in Appendix 4-C.

Applying SIGMA to a cost benefit equation will result in a curve which will illustrate the value, with its associated uncertainty of the cost benefit for engineering changes. The value of the net benefit is defined as:

$$\text{NET BENEFIT} = \text{BG} - \text{TAC}$$

where:

BG = Benefit Gained (due to increased availability)

TAC = Total Annualized Cost (of the engineering changes)

The resultant curve (the NET BENEFIT CURVE) is a probability density that shows the likelihood of the net benefit over a range of values. The NET BENEFIT CURVE is a graph of dollars vs. probability. With zero uncertainty the curve will be a vertical line equal to 1.0 at the net benefit value. Since uncertainties do exist, the net benefit curve is a distribution of values which is highest at the expected value.

4.2.1 Net Benefit and Uncertainty Methodology

Four computer routines are used to manipulate the net benefit data. Two were written for this project. BENEFIT contains the algorithms to calculate the annualized benefit of increased availability due to a proposed engineering change (Section 4.1). IMCOST sums the costs associated with the implementation of the proposed solution (See Section 4.2.2).

Two existing C-E programs are also used. SIGMA combines the probability distributions of several independent variables. The CEGGS (Combustion Engineering Generalized Graphics System) routine then fits a curve to the points in the net benefit data and plots the graph on a drum plotter. BENEFIT, IMCOST, and SIGMA were combined to run as a single code called CENBAC (Combustion Engineering Net Benefit Analysis Code).

Capital cost limits, component and labor availability, and other factors can affect the decision making process. These factors could not be considered in a quantitative manner.

4.2.2 Implementation Costs

Evaluation of solutions needs to consider implementation costs. In most cases these costs must be estimated. Components of the implementation cost

are engineering hardware and installation costs. Each of these costs are variable and a range of estimates should be made for each.

It is assumed that implementation either does not require plant shutdown or is accomplished during a scheduled refueling outage. Thus no replacement power need be purchased during the period of alteration to the plant. For cases involving major plant modifications requiring several months shutdown, the model can be modified to include this effect.

Engineering Costs

Two types of engineering costs are considered in the analysis. The ONE TIME ENGINEERING COST is that cost associated with the development of a detailed engineering solution by the NSSS vendor, AE, or other engineering staff. This cost is divided by the number of plants implementing the solution, in order to distribute the cost among several plants. Engineering costs are assumed to be independent of plant size.

The PER PLANT ENGINEERING COST is the cost of the specific engineering and planning needed to apply the solution to a particular plant. Very few of today's nuclear plants are exact duplicates and therefore each plant will require some engineering specific to that plant.

Hardware Costs

Hardware costs are on a per plant basis.

Installation Costs

The installation cost consists of the direct labor and supervisory costs required for the installation and testing of the modification. Provisions are made for additional annualized operation and maintenance costs, if required. An example of this is the maintenance of diagnostic instrumentation for a steam turbine.

Cost Model

The IMCOST model sums the engineering, hardware, and installation cost estimates and then multiplies this sum by the levelized annual fixed charge rate to produce the annualized equivalent cost. The annual costs, operation and maintenance, are then added. This sum is the total annualized cost.

4.2.3 Net Benefit Calculation Example

As an example, the problem of steam generator level control failure during low power operation was assumed to have caused an average of 0.7 outages per year, each averaging 5.8 hours before the plant returns on line. An engineering solution was identified. The installation of automatic feedwater control below 15% power was assumed to reduce the outage frequency by one half. Seven plants are assumed to be sharing the one time engineering cost for this problem. The range of cost estimates for implementing the low power level automatic feedwater control system is summarized below.

	Minimum	Best Estimate	Maximum
One Time Engineering Costs	\$50,000.00	\$85,000.00	\$150,000.00
Per Plant Engineering Costs	\$10,000.00	\$25,000.00	\$ 50,000.00
Per Plant Hardware Costs	\$10,000.00	\$24,000.00	\$ 40,000.00

Using the fuel cost curves in Appendix 4D, the CENBAC program calculates the net benefit to be greater than \$2,600,000 with a 91.2% confidence. The most likely value of the net benefit is \$3,200,000.00.

It is important to note that the net benefit values are a result of a probabilistic simulation of future events. The results of such analysis are not exactly repeatable, but are within definable bounds. This results from the Monte Carlo random sampling techniques used in the CENBAC program. Repeated calculations should yield quite similar results.

4.3 CONCLUSIONS AND SUMMARY

Some PWR component failures will result in the reactor protection systems being challenged and plant shutdowns. The electricity production loss during this period must be replaced from other sources. These sources are in most cases less economical than base load nuclear plants.

The first cost benefit objective, to generate sensitivity curves is satisfied by Figures 4-1, 4-2 and 4-3. These curves show that the prevention of one day shutdown per plant year over the 40 year lifetime of a 1000 MWe PWR can save 32 million dollars. The total benefits of a one day availability increase for the forty operating PWR's in the U.S. today is shown to be $1,185 \times 10^6$ dollars (see Table 4-1).

It was not possible to include in these curves, the global impact of decreased oil consumption, sulfur dioxide and other pollution, coal mining costs, or additional facilities. These considerations are factors which increase the total impact of nuclear plant outage reduction.

The second program objective, to develop a methodology to include uncertainties in the net benefit calculations is satisfied by the CENBAC computer program. CENBAC (Combustion Engineering Net Benefit Analysis Code) is a probabilistic cost-benefit model that uses data with significant uncertainties. CENBAC uses variables whose values are assigned probability distributions and calculates the benefits of increased availability. It then subtracts the costs of development and implementation. Using the probability distribution of the input variables, CENBAC produces a probability density curve of the net benefit for the proposed solution. This curve shows the probability of achieving a positive or negative benefit from a suggested solution.

The methods developed in this study can be used to evaluate the benefits of improved availability. Better economic performance and improved safety will result from the proper choice of proposed engineering solutions.

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- 10) Turbomachinery International, "Utility Cost Study Part 1 Gas Turbine Utility Plants", July 1978.

Table 4-1
COST OF 1 DAY SHUTDOWN PER YEAR
OVER REMAINING PLANT LIFE

PLANT TYPE (# OF PLANTS)	AVERAGE CAPACITY (MWe)	AVERAGE REMAINING LIFETIME (YEARS)	TOTAL PLANT CAPACITY (MWe)	TOTAL BENEFIT (\$ x 10 ⁶)
C-E (7)	749	35.2	5,243	\$ 206
WESTING- HOUSE (25)	751	35.4	18,775	\$ 739
B&W (8)	872	36.3	6,978	\$ 240
<hr/>				
U.S. PWRs (40)	775	35.5	30,996	\$1,185

*TOTAL DOLLAR BENEFIT OF ADDING 1 DAY PER YEAR AVAILABILITY TO EACH PLANT FOR THE REMAINING LIFETIME OF THE 40 OPERATING PLANTS IN THE U.S. (1980 DOLLARS)

FIGURE 4-1

TOTAL⁽¹⁾ COST OF 1 DAY/CALENDAR YEAR OUTAGE ($\$ \times 10^6$) vs REMAINING PLANT LIFE (YEARS)
(SINGLE 1000 MWe PWR "EXPECTED" FUEL AND REPLACEMENT PLANT ESCALATION)

(1) Includes Capital, replacement power

(2) 6 hrs peaking units, 18 hrs normal mix

FIGURE 4-2

RECURRING OUTAGE COST (\$ x 10⁶) vs OUTAGE DURATION (DAYS/YR)

COSTS FOR SINGLE 1000 MWe PWR, EXPECTED FUEL AND REPLACEMENT
PLANT ESCALATION

FIGURE 4-3

SHUTDOWN COST BREAKDOWN ($\$ \times 10^6$) vs REMAINING PLANT LIFE (YEARS)

COST FOR SINGLE 1000 MWe PWR, 1 DAY SHUTDOWN PER YEAR, NO REPAIR COSTS

SECTION 5

SUMMARY AND CONCLUSIONS

5.0 SUMMARY AND CONCLUSIONS

The primary objectives of the component failure study were 1) to determine causes of challenges to the RPS and post-trip heat removal systems, and 2) evaluate the major problems for ways by which they can be reduced or eliminated. Secondary objectives were to identify the lost plant availability impact of those challenges and evaluate the cost benefit from reducing that impact.

The major causes of the 7.3 scrams per plant year and 13.2 percent unavailability per plant year were the result of problems in the following seven systems:

- o Feedwater and Condensate
- o Turbine Generator System
- o On-Site Power System
- o Reactor Protection System
- o Control Rod System
- o Main Steam System
- o Reactor Coolant System

The major causes in the Feedwater and Condensate System and the Turbine Generator System were multiple and diverse. It is recommended that these systems be the subject of comprehensive systems optimization studies to reduce the vulnerability to loss of system function upon loss of single components or as the result of system interactions.

Recommended On-Site Power System improvements include providing control and instrumentation buses with uninterruptable power sources. This may involve going to four batteries per plant if not already provided.

The majority of scrams, seventy percent, caused by the RPS were determined to be spurious and unnecessary. An evaluation of the use of 2-out-of-3 or 2-out-of-2 logic rather than 1-out-of-3 or 1-out-of-2 logic should be conducted to eliminate the unnecessary challenges to the post trip heat removal systems.

Control Rod System problems have occurred at only a few plants and are being resolved. Main Steam System steam generator tube leaks are the subject of major industry programs. Further work should be conducted to identify improved Reactor Coolant Pump Seal installation procedures. Pressurizer Spray Valves should also be further evaluated for improvements in design as well as maintenance.

In the Cost Benefit portions of this study, the present worth of improved average annual unavailability for plants was shown to be significant. From the present worth curves constructed, a one day increase in average plant availability for a 1000 MWe PWR is shown to be 32 million dollars over the 40 year lifetime of the plant.

The principal conclusion of this program is that most challenges to the RPS come from process or control grade components. Therefore, improvement of the reliability and availability of process systems can result in a significant safety benefit through a reduction in challenges to the reactor protection system and post-trip heat removal systems. In addition, improved process system performance can also aid in the mitigation of transient events.

SECTION 6

RECOMMENDATIONS

6.0 RECOMMENDATIONS

The scope of this study was limited to the study of events contained in the C-E PWR Shutdown Data Base. The depth of research in any one area was intentionally restricted. The study reviewed in depth those component categories identified as being major contributors to shutdowns and plant unavailability. Investigators were limited to the information available or easily obtainable. Following the cause and effect chain leading to the failure of a component can often be successful on a plant specific basis, while on a generic basis, as was attempted by this program, differences in plant configurations can be misleading. More comprehensive programs, which integrate the efforts of plant operators, component designers and NSSS designers, could be a more effective approach to the identification of root causes for component failures.

The programs recommended are of three types: well defined programs capable of solution by development programs, problems capable of solution by evolutionary improvements, and problems relating to data gathering analysis and reporting. The following are the programmatic recommendations by this study.

6.1 PWR FEEDWATER SYSTEM EVALUATION

A detailed investigation of U.S. PWR condensate and feedwater system designs for the purpose of identifying methods of improving the interaction of the feedwater train with the Nuclear Steam System (NSSS) should be conducted. Such an evaluation would increase the overall plant reliability and availability and reduce the challenges to the reactor protective system, thereby increasing plant safety.

The overall investigation of PWR condensate and feedwater systems should be divided into four subtasks as described below.

1. Condensate and Feedwater System Configurations:

Existing system designs should be surveyed and event failure profiles established for the various configurations. Unique configurations

that are susceptible to specific events should be identified. Design comparisons among the various U.S. configurations and with foreign systems should be made with respect to design philosophies, system operation, maintenance, and testing. Based on the above, optimization guidelines should be developed for application to both existing and future NSSS designs. These guidelines should address:

- o redundancy of major components
- o sizing of major components
- o choices of motive power
- o power supply and control system reliability
- o reliability degradation due to backfits
- o operator-machine interfaces (operation and maintenance).

2. Low Power Feedwater Operation:

Fixed control and protective system setpoints based upon full power operation needs should be evaluated for the possible use of less restrictive setpoints during low power operation. An analysis of steam generator level setpoint margins over the full range of power operation should be performed to identify reductions in steam generator inventory requirements at low powers to allow expansion of the level control band. Additionally, design changes aimed at improved manual feedwater control and/or automatic feedwater control at low power should be evaluated. Foreign design approaches to low power feedwater control through the use of a smaller capacity feedwater control valve and feedwater and condensate pumps should also be evaluated. Improvements should be investigated in automatic feedwater control systems to expand the ranges of reliable response over the full range of power operation.

3. System Hydraulic Resiliency:

An evaluation should be made of possible methods for increasing the feed train's capability to accept transitory hydraulic instabilities and variations without challenging the reactor protective system.

Computer codes which combine best estimate secondary system modeling with existing NSSS performance models should be employed in examining Condensate and Feedwater System designs. For example, the application of surge/deaerating feedwater tanks and main feed pump selective or time delayed trips should be examined. Optimizing the interfacing of component level equipment protective setpoints (e.g., condensate and feed pump trips) and NSSS Reactor trip setpoints should be examined in order to reduce plant outages resulting from individual component problems (domino effect). Additionally, potential improvements in system design performance over the full range of power operations should be evaluated in order to improve load following during forced power reductions.

4. Feedwater Train Support Considerations:

Optimization of the various aspects of system and plant design that support the operation of the feedwater train should be examined. These include:

- o Feedwater Control Systems
- o Chemistry Cooling
- o System Maintenance and Testing
- o Susceptability to Operator Error
- o Power Supply Reliability

6.2 TURBINE GENERATOR SYSTEM RELIABILITY AND OPTIMIZATION

The Turbine Generator System ranked second to contributing to scrams and shutdowns at PWRs. Areas needing additional study are as follows:

a) Turbine Valve (Control)

Valve control problems were found to be the cause of most of the 23 outages due to malfunctions in the control, stop and intercept valves and the majority of the 53 MSIV failures investigated. Problems with these control systems led to spurious valve closure and result in

turbine and reactor trip. When investigated from a control system viewpoint, 33 trips were caused by EHC (Electro-Hydraulic Control) failures, 29 trips which occurred during testing and 10 trips were caused by governor intercept control problems. In addition 20 trips caused by MSIV control failures resulting in reactor trips. EHC failures also caused an average of 0.73 trips per plant year with durations of 7.6 to 14.4 hours.

A comprehensive effort needs to be undertaken to improve the control and protection control circuits for the above valves. This needs to be coordinated with the valve testing requirements for operability for safety reasons. The present designs are extrapolations of older fossil turbine designs which do not adequately address the necessity for Technical Specification required testing and the different nature of the steam supply conditions for PWR turbines. The resulting control/protection system design should be similar to that of the reactor protection system and should have comparable reliability.

b) Turbine Stop, Control Valves and MSIVs (Mechanical)

Malfunctions of these valves will cause turbine overspeed with potentially large economic consequences and inadvertent reactor trips upon spurious closure. The main steam isolation valves are included because of their similar characteristics and effect upon the nuclear steam supply system and turbine.

For those valves, the following mechanical considerations require further study:

- o valve design
- o quality control
- o improved materials
- o improved maintenance and operation procedures

The testing requirements for these valves also need to be investigated, with respect to the type of valve and control system used and their historic failure frequencies. Spurious closure during test has been a major contributor to spurious turbine-reactor trips.

c) Turbine Auxiliary Systems

The major areas investigated in this study for turbine auxiliary system failures involved lubricating oil systems and vibration. Of the 183 events investigated for turbine generator failures, bearing failures ranked second only to blade failures. Vibration is directly related to bearing failure in that it accelerates bearing wear.

A comprehensive investigation of lubrication and control oil system problems should include sources of contaminants, blocked passages and orifices, emulsification and proper cooling. The influence of fabrication, shipping, and erection practices on control oil system contamination should be investigated.

The current study reached the conclusion that 90 percent of all turbine generator problems are either directly or indirectly caused by vibration. This suggests a need to improve current vibration detection systems in order to be able to detect incipient failure of turbine components before significant damage occurs.

d) Generator

Failures investigated during this program which involved the main plant generator included loss of field/voltage regulation, damage to generator bearings, stator failures, hydrogen cooler failures, and ground problems. Generator bearing and stator problems resulted in an average of 0.23 shutdowns per plant year with an average duration of 103 hours.

Further efforts should be initiated to determine root failure causes, improve reliability of voltage regulator field relays and power supplies, improved methods of ground detection, improve maintenance procedures, and improve hydrogen leak prevention and detection instrumentation.

e) High Pressure and Low Pressure Turbines

High pressure and low pressure turbine problems as related to saturated steam LWR turbine design should be investigated in detail. These involve mainly blade problems associated with moisture damage, chemical attack and stress problems. Inadequate moisture separator reheater performance also affects LP turbine blade failure through excess moisture formation in the LP turbine.

Moisture Separator Reheater

All LWR Steam Turbines are equipped with Moisture Separator Reheaters (MSR) between the HP and LP turbines to improve LP turbine performance and increase thermodynamic efficiency. Because of a lack of prior experience with these devices in this application, they have universally been poor performers for the industry. Their poor design and performance has been aggravated by turbine valve testing requirements and inadequate drain and control system design during installation.

Operation and maintenance improvements need to be investigated and implemented to improve the performance of existing MSRs. The inadequate performance of this component and its related service systems has a direct impact on LP turbine blade failures. These program recommendations are the result of an investigation of 42 MSR failure events. No current programs are known to be underway for this component.

This program should involve short, medium and long term goals. These are outlined below.

Short Term

- o Investigate and recommend changes for shell and tube side drains and level control.
- o Investigate and recommend changes in reheat temperature control.
- o Investigate and recommend changes in LP turbine stop and intercept valve testing to reduce thermal transient damage to MSR.
- o Investigate and recommend changes to MSR operation and maintenance practices.
- o Investigate and recommend changes to improve MSR safety valve performance.

Medium Term

- o Investigate and characterize moisture carryover from separator to reheater section.
- o Investigate and recommend improvements in steam inlet header design.
- o Investigate and characterize outlet steam temperature distribution.
- o Investigate and recommend improvements to reduce steam bypass flow around separators and reheat tube bundles.
- o Investigate and characterize tube failures and tube to tube sheet joint failures.
- o Investigate and characterize stress levels where failures are occurring.

Long Term

- o Investigate and recommend improved tube and shell materials to reduce tube failures and shell corrosion.
- o Investigate and recommend improved MSR configuration to improve reheat performance.
- o Investigate and recommend improved designs for backfit to existing turbines.

- o Investigate and recommend an optimum LSR design to reduce size and space requirements and thermal performance for new plants.
- o Investigate and recommend optimum interface criteria between MSR and feedwater cycle including steam supply source.
- o Investigate desirability of higher vessel code application to improve integrity.

Turbine Condenser System

Turbine condenser system problems include both tube leaks and auxiliary system problems which lead to tube leaks or power reductions. Tube leaks are of major importance to the PWR industry because of their detrimental effects on steam generator tube integrity and their effect on the design and operation of the feedwater and condensate systems. The scope of recommendations in this section will include all considerations except condenser tube and tube sheet integrity which is the subject to an EPRI sponsored program.

The recommendations presented cover cooling water system designs to minimize tube fouling and plugging, condenser deaerating performance, condenser air in-leakage problems, cooling water fill effectiveness, corrosion control and crud removal, and operation and maintenance inadequacies.

Cooling Water Contamination

Many solids and organic materials are transported from the cooling water source into the condenser water box and tubes. This results in tube blockage and corrosion due to inadequate tube water flow. This in turn results in tube leakage due to corrosion and performance loss due to fouling.

A program to improve the present methods of removing entrained solids before they reach the condenser water box and tubes is needed. This should include better trash removal equipment and filters as necessary.

An improved method of reducing the effects of organic material on tube blockage is needed and should be coordinated with current EPRI sponsored programs in this area.

Condenser Deaerating Performance

Cycle air removal is presently accomplished in most PWRs by the condenser. Unless properly operated, this function is degraded during the colder periods of the year. Also, environmental limits on allowable cooling water temperature rise across the condenser preclude efficient deaerator section performance.

An investigation of current condenser designs for deaerator performance should be conducted in coordination with the feedwater and condensate system recommendations for satisfactory deaeration of feedwater before high temperature feedwater and steam generator use. This should be coordinated with environmental regulations and operating practice for cooling water flow during colder conditions.

Condenser Air-Inleakage

Many valves, drains, expansion joints, etc. permit air in-leakage to the vacuum portions of the condenser and condensate system. This affects both feedwater purity and cooling water supply.

A comprehensive investigation should be made of leakage paths and air removal in the vacuum portions of the condenser, condensate, and cooling water systems. The condensate study should be coordinated with the feedwater and condensate system investigation. The cooling water portion needs coordination with the circulation water investigation to assure that all condenser tubes are receiving cooling water and that tube blockages are not occurring.

Corrosion and Crud Control

There are large surface areas in the turbine shell and condenser internals which are subject to corrosion and thermal shocking, especially when the turbine is opened up for maintenance. These corrosion products eventually end up in the condenser hotwell and are transported into the feedwater cycle unless adequately cleaned prior to startup. Feedwater cycle problems and steam generator problems are in part attributable to this crud transport.

A program to investigate corrosion product source, control and removal is needed to protect feedwater system components and steam generators. This needs to be coordinated with feedwater and condensate system investigations, especially the deaerator function during post shutdown cleanup operation.

Improved Operation and Maintenance

Areas for improved operation and maintenance are as follows:

- o Improved O & M of the condenser and associated systems.
- o Improved condenser tube leakage location and plugging techniques.
- o Improved operating procedures to improve air removal by condenser during cold weather operation.
- o Improved shutdown and startup procedures to minimize crud formation and adequate removal prior to startup including chemical treatment.
- o Improved O & M procedures to reduce air inleakage from valves, pipes, flanges, etc. in the subatmospheric sections of the condenser.

6.3 VIBRATION MONITORING

Results obtained during this PWR Component Failure study indicate that vibration is both a cause and the result of many equipment problems. Vibration monitoring has the potential to detect imminent equipment failure or malfunction. Vibration monitoring techniques properly utilized can

predict problems before outages result and can improve installation and alignment of critical rotating equipment. An extensive vibration monitoring investigation at PWR power plants should be conducted in order to identify potential degradation of equipment due to vibration and degradation caused vibration. This study should include a detailed review of vibration monitoring techniques and their potential for application to PWR plant equipment. The study should cover equipment vendors and operating plant experience at both fossil and nuclear. Pumps, valves, turbines and piping applications should be investigated. Procedures and techniques should be evaluated for recommended improvements.

6.4 ADAPTIVE CONTROL SUPERVISORY SYSTEMS PROGRAM

Unintentional challenges to the protection system adversely affect plant unavailability and result in an unnecessary decrease in safety margins. Many such challenges can be avoided by appropriate feedback of plant characteristics to control systems. The plant protective system is challenged approximately eight times per plant year. A control override system will reduce the risk of safety system failure by reducing the need for safety system action. Plant availability is increased by taking backup control actions which avoid the need for plant trip by maintaining the system on-line. Such control actions are less severe than a plant trip and the risk of human error is reduced by avoiding the need for rapid operator action. The objective for this program should be to develop and demonstrate the benefits of an adaptive-control supervisory system which automatically adjusts plant control and/or protection system setpoints or control operation such that plant response is optimized relative to margins of safety, efficiency, and availability.

This program should also address development of analytical techniques for parameter estimation on a generic basis, establishment of efficient system models for on-line application, specification of computer software and hardware requirements, demonstration of computer software and hardware

requirements, and demonstration via computer experiments of effectiveness of methods. The testing of a prototype system on an operating PWR should also be included.

6.5 IMPROVED PROCESS INSTRUMENTATION INVESTIGATION

This study should be to conduct a detailed investigation of process instrumentation sensor channels for the purposes of improving sensor accuracies and their contribution to withstand environmental fluctuations. The goal is to eliminate unwarranted spurious challenges to the protective system. Operation with reduced error margins and thus greater power output would be An added benefit. Process sensor outputs should also be characterized under adverse conditions.

The scope of such a study should include:

- a) Identification of generic sensor environments for vibration RFI/EMI noise, and temperature, and development of environmental requirements;
- b) Development of baseline plant conditions for environmental surveys;
- c) Development of Specifications and Requirement for a Testing & Qualification Chamber/Facility followed by construction of demonstration project;
- d) Investigation Level Detection Methods for the reactor pressure vessel, the steam generator and the pressurizer;
- e) Evaluation alternate methods for process measurements in order to improve accuracies and reduce error margins;
- f) Participation by process equipment vendors and national test laboratories.

6.6 REACTOR COOLANT PUMP SEAL INSTALLATION AND MAINTENANCE

The objective is to recommend procedures for RCP seal maintenance, installation and startup at all plants. Specific design data should be accumulated with

respect to seal design, installation and maintenance. Interviews with seal manufacturers, pump manufacturers and operators should be conducted. Experience at plants should be transferred between plants.

6.7 PWR VALVE APPLICATION AND DEVELOPMENT STUDY

Degradation and malfunction of valves caused approximately 1.13 shutdowns per plant year during the first 192.4 plant years of operation. Further, valve degradation has the potential of compounding other problems. The limited funds available for development, testing and qualification can delay addressing the problems. The objective should be to survey PWR valve applications including actuators and operators, and investigate and sponsor valve vendor development for unique nuclear applications including valve demonstration of qualification and testing.

Four specific applications for study are outlined below:

1. Pressurizer and Main Steam Safety Valves

- A. Problem: Problems with valve reseating
- B. Cause: failure of mechanical elements
- C. Solution: improve existing valve design features and evaluate new designs
- D. Recommended programs:
 - a. investigate vendor development programs and encourage work in specific areas, i.e., valve seat materials, operation, etc.
 - b. Sponsor the evaluation and qualification testing, of new valve design features, e.g., better valve seats.

- c. Investigation of power operated relief valves for use with steam or water.

2. Pressurizer Spray Valve

- A. Problem: packing leaks
- B. Cause: inaccessibility for maintenance
- C. Solution: relocate valves in more accessible area
- D. Recommended programs:
 - a. Determine the reasons for the location of these valves and identify possible alternatives.
 - b. Investigate and specify zero leakage valves.

3. Main Steam Isolation Valve

- A. Problem: inadvertent actuation resulting in valve followed by reactor scram
- B. Cause: faulty subcomponents such as solenoid valves
- C. Solution: qualify all parts of valve system and dependent systems
- D. Recommended programs:
 - a. Evaluate valve and actuation system as a total package, specify design features and eliminate any unnecessary component parts.

- b. Propose and evaluate a complete qualification program for this valve system.
- c. Identify needed reliability and redundancy of subcomponents.

4. Turbine Control, Stop and Intercept Valves

- A. Problem: valve malfunction, repairs and replacement.
- B. Cause: poor maintenance, lack of exercising, leakage, manufacturers changes.
- C. Solution: improve maintenance programs, reduce requirements for maintenance by improved design, improve valve exercise scheduling, improve valve testing control system.
- D. Recommended programs:
 - a. Evaluate valve selection process for these applications, and recommend specific criteria. This program could be expanded to cover all safety-related valves.
 - b. Evaluate and recommend improvements to maintenance and testing procedures.

6.8 PWR COMPONENT FAILURE TRENDS STUDY

In this PWR component failure study, a trend analysis was completed for each of the major systems impacting plant safety or availability. Analysis of additional systems and their problem profiles should be performed and improved prediction techniques should be developed. This study should identify those systems and problems which are worsening, improving, or staying the same.

The objective of such a study should be to develop baseline data and a valid prediction technique for PWR component failure trends. Predictive techniques should then be developed and tested with a current prediction of PWR component, system and overall plant performance.

6.9 STUDY OF HIGH FREQUENCY FAILURES IN PWR's

This program's review of the C-E PWR Shutdown Data Base and its verification effort suggests that nuclear plants are fairly well designed to cope with single process system failures. The plant's systems are either adequately designed to handle single component/system malfunctions or the operator can satisfactorily correct the problems when they occur. This leaves multiple process system failures including operator error as the next most important factor contributing to plant outages and challenges to the plant protective systems.

It is postulated that the systems and components exhibiting high single failure rates also have the greatest chance of failing simultaneously, thereby initiating serious accidents. It is believed that the TMI-2 sequence might have been predicted from such a study, along with other possibly severe events.

The study should identify high frequency single PWR system or component failures which are likely to occur concurrently or while some essential system or component is out of service for maintenance. These failures will then be evaluated to determine which combinations should be accommodated by design changes to improve a nuclear plant's overall safety and availability.

The following should be included:

1. Review existing data bases, LER's, and NPRDS to identify high frequency events and to verify that concurrent failures are likely to occur.

2. Perform engineering evaluations and statistical analyses to identify those events and failures which are most likely to occur concurrently or during the repair periods of some other failed component or system.
3. Rank concurrent occurrences based on overall impact on plant operation.
4. Develop additional recommended PWR Design Basis Guidelines from an assessment of the concurrent failures and their impact on the industry's Design Basis Events currently considered in the plant's Safety Analysis.

6.10 OPERATOR/PERSONNEL ERROR STUDY

Recorded plant operating histories indicates that operator/maintenance errors rank equally with control systems in contributing to plant unavailability. Recent history also highlights the potential for degradation of plant safety due to personnel errors. The combination of both safety and productivity concerns dictates the need for an in-depth evaluation of human factors. Ways in which operators and other plant personnel can be made more effective such be determined in order to reduce the probability of human error. Operating history should be analyzed for both safety and productivity losses incurred due to personnel errors. This subset of operating data should then be broken down parametrically in order to identify significant trends and provide a more clear picture of circumstances contribution to personal errors.

6.11 CONCEPTUAL STUDY OF INFORMATION SYSTEM NEEDS FOR AN IMPROVED INTEGRATED NUCLEAR PLANT INFORMATION AND REPORT SYSTEM

Every day much information is generated and processed, sometimes redundantly, at nuclear power plants for use in both internal and external reports. Since the filed information is recorded manually, it is not easily accessible for future use. A computerized plant information system could significantly increase the potential for improvement in plant operations, maintenance, testing, reliability and preventive safety measures.

The objective of this program should be to identify nuclear plant information needs and uses, both internal and external, and to assess the feasibility of computerized integration for both internal and external purposes. A further objective is to identify redundant information processing and to reduce the reporting burden of nuclear plant operators.

The scope of this study should be a survey of both external and internal plant reports. Analysis of end uses and redundancy, and assessment of the feasibility of consolidating and facilitating data processing by use of computerized data collection and distribution methods should be investigated.

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