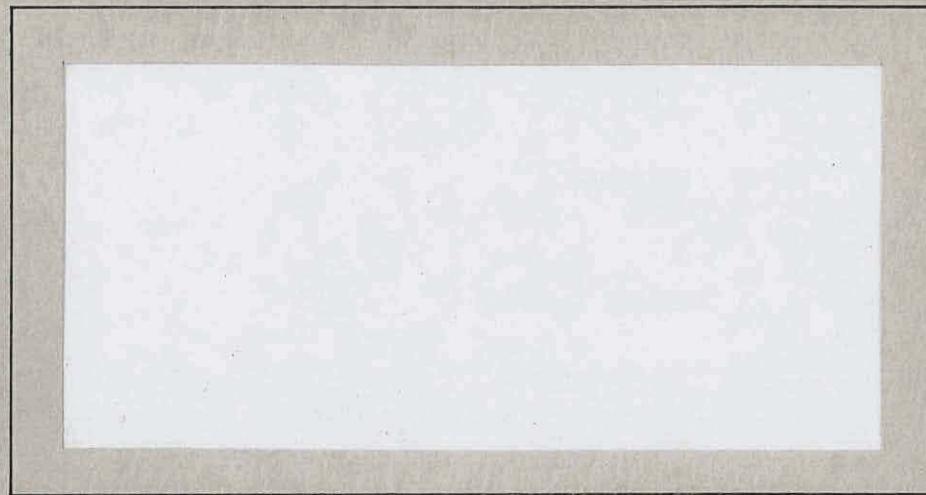


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COMPONENT FAILURES THAT
LEAD TO REACTOR SCRAMS

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COMPONENT FAILURES THAT
LEAD TO REACTOR SCRAMS

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ABSTRACT

This report summarizes the operating experience scram data compiled from 35 operating U.S. light water reactors (LWRs) to identify the principal components/systems related to reactor scrams. The data base utilized to identify the scram causes is developed from an EPRI-utility sponsored survey conducted by SAI coupled with recent data from the USNRC Gray Books.

The reactor population considered in this evaluation is limited to 23 PWRs and 12 BWRs because of the limited scope of the program. The population includes all the U.S. NSSS vendors. It is judged that this population accurately characterizes the component-related scrams in LWRs over the first 10 years of plant operation.

Based upon this LWR population, the scram transient data is categorized according to principal component and contributing causes. The principal components are ranked according to frequency for both BWRs and PWRs. In addition, an investigation is conducted into the time dependence of the overall scram incidence rate and the scram incidence rate for individual components.

Having established the ranking of the principal components/systems leading to scrams, several potential fixes are presented which could reduce the incidence of LWR scrams. Because the data does not include a detailed analysis of the root cause of each scram, the suggested design fixes are general in nature. Only a detailed root cause investigation of each incident could uncover the basic contributing factors to each scram and potential methods of preventing these causes.

1.0 INTRODUCTION

1.1 Objectives

The Reactor Safety Study⁽¹⁾ (WASH-1400) as well as subsequent studies^(2,3,4) have shown that a substantial portion of the public risk from the operation of nuclear power plants is associated with the potential inability to cool the core following plant transients that require a rapid reactor shutdown. Therefore a decrease in the number of these transients would reduce the total calculated public risk. WASH-1400 also shows that with the possible exception of low frequency common mode failure initiators, the magnitude of possible consequences (radioactive releases) is not extremely sensitive to the initiating transient. In other words, the contribution of each type of plant transient to overall public risk is directly proportional to its frequency of occurrence. Hence, a large potential for public risk reduction lies in minimizing those plant transients that most frequently require reactor shutdown.

A number of reports^(5,6,7,8) have identified component failures, miscalibration, instrument drift, improper repair, or human error related to a specific component as the principal causes of plant trip. Generally speaking, each of the reports has focused in detail on a particular class of initiators for reactor shutdown.

This study is conducted in two parts. First, the present report will address those transients involving scram. These scram transients are referred to as anticipated transients since they are considered in plant design; however, they require an immediate and full plant shutdown scram. A second report deals with transients involving "controlled" shutdown, which also requires removal of decay heat. The objectives of this first comprehensive study are to:

- (1) identify and rank, according to frequency and plant age, the historical causes of reactor scrams, particularly those related to component failures, human error, or faulty procedures, and

-
- (2) suggest modifications in hardware or procedures which will reduce the number of component-related plant trips.

1.2 Outline of this Study

Two alternate methods for calculating the frequency of transients that lead to a reactor trip are:

- (1) Develop a detailed plant model and calculate the frequency of transients using analytical techniques such as fault tree analysis or other methods from reliability theory.
- (2) Combine historical data with engineering judgement to calculate the frequency of transients.

The second approach has been used in the current analysis since it is generally accepted that operating experience data is the "best" available prediction of future operation.

Several potential sources of data for this study were evaluated. Based on this evaluation, it was decided that a careful analysis utilizing a representative sample of operating plants was more likely to yield meaningful results than a less thorough study that attempted to include all nuclear units. The data base for this approach was constructed by combining information compiled for an EPRI-SAI ATWS study⁽⁷⁾ with NRC Gray Book⁽¹⁰⁾ listings. This approach will identify, on a generic basis, the systems or components that are most frequently responsible for reactor scrams. A portion of this study was then devoted to investigating whether variations in scram causes exist between units of similar designs; or whether the principal causes of scram are plant specific.

1.3 Report Organization

A brief description of each potential data source is presented in Section 2. Results of the analysis to identify the generic causes of plant trips are shown in Section 3. In addition, trends in component or system

failure rates with plant design and plant age are also given in Section 3. Section 4 examines several Babcock and Wilcox plants for variations in the causes of their transients. Component as well as procedural improvements are proposed in Section 5.

2.0 DATA SOURCES

Data needed to determine the root cause of plant transients that eventually lead to reactor scram may be obtained from several sources.

These sources are:

- ATWS-SAI/EPRI Nuclear Plant Transient Event Data Base⁽⁷⁾
- Nuclear Regulatory Commission (NRC)-Licensee Event Reports (LER)
- Oak Ridge-Nuclear Safety Information Center (NSIC)
- "Nuclear Power Reliability Data System" (NPRDS)
- Nuclear Regulatory Commission "Operating Units Status Reports" (OUSR) Gray Books-NUREG-0020⁽¹⁰⁾
- Edison Electric Institute^(EEI), "Equipment Availability Data System"⁽¹¹⁾
- FEED-SAI Nuclear Plant Availability Data Base⁽⁹⁾
- Vendors, utilities, and industry sources
- Review of existing technical literature

The first part of this section briefly describes the major attributes of each of these data sources.

2.1 Evaluation of the Data Sources

Since none of these potential data sources were designed specifically to pinpoint the component causes of plant transients, it is necessary to evaluate each source for its applicability to the present project. The criteria of evaluation are:

- a) Completeness: This refers to the relative fraction of events contained in the data base compared with the total number of events which actually occurred at each plant in the data base for each year included.

- b) Accuracy: In some cases the data recorded in data files are incorrectly labeled or categorized and therefore bias the results unless each entry can be verified.
- c) Level of Detail: The level of detail reported in each data entry varies considerably from plant to plant and source to source. The significant features of the data we are interested in are the plant, the data, the component involved if applicable, and the failure mode. It may also prove important to define the power level from which the transient occurred.

2.2 Descriptions of Available Data Sources

2.2.1 SAI-EPRI ATWS Nuclear Plant Transient Event Data Base

Selected utilities were asked by EPRI to provide data concerning transients experienced at their plants. Based on the initial response, an expanded, continuing data collection effort was initiated with cooperating utilities. For each experienced transient the following information was requested:

1. Date of the scram
2. A brief description of the scram sequence including the component failure type and failure mode
3. The plant mode and power level at the time of the scram
4. The reactor status following the scram
5. The type of scram

The data were collected directly from the utilities on forms such as shown in Table 2.1. Direct communication was established with each plant to clarify the understanding of data items when necessary. These

data were used to classify and establish the frequency of broad categories of transients.

Table 2.1.
EXAMPLE OF BASIC DATA IN EPRI TRANSIENT DATA BASE

PLANT				RATED POWER	3293 MW(t)	EFFECTIVE IN-SERVICE DATE	8/28/74
DATE	PRIOR TO SCRAM		REACTOR STATUS AFTER SCRAM	OUTAGE TIME (HR)	TYPE OF SCRAM	BRIEF DESCRIPTION OF SCRAM SEQUENCE-INCLUDE THE ACTUAL INITIATOR OF THE SEQUENCE	
	MODE	POWER					
#22 1/17/75	Startup	66	Hot Standby	10 hours	Rod Sequence B	Scrammed manually when changing to rod sequence "B".	
#23 1/19/75	Run	1054	Shutdown	96 hours	IRM Hi-Hi	Relief valve opening (PCV1-4) causing emergency shutdown.	
#24 1/24/75	Run	1317	Hot Standby	9 hours	Hi-Hi Flux	The reactor scrammed on high flux when pressure spiked during a period when work was being done on the EHC.	
#25 1/28/75	Run	2173	Hot Standby	10 hours	Turbine trip	The reactor scrammed due to a turbine trip which was caused by a false high water level signal from a faulty circuit card in the moisture separator trip logic.	
#26 2/3/75	Run	2964	Hot Standby	10 hours	Reactor low water level	Condensate boost pumps tripped on low suction pressure caused by a problem with the condensate demineralizer.	
#27 2/11/75	Run	3161	Shutdown	96 hours	STI-27	Tripped for full load rejection scram.	
#28 3/22/75	Run	3293	Shutdown		Manual	Shutdown due to fire in cable trays.	

These data are compiled to support an EPRI-SAI study on the frequency of anticipated transients in nuclear reactors. They have been described by the NRC in NUREG-0460⁽¹²⁾ as

"the most extensive data on plant transients available to the staff (providing) the best basis for estimating the frequency of anticipated transients in nuclear power plants"

As with any of the applicable data sources, the reactor population as it exists today can only support a limited set of conclusions. Thirty-five of the nearly 70 operating light water reactors are presently in the EPRI-SAI data base. These include twenty-three (23) pressurized water

reactors and twelve (12) boiling water reactors. Tables 2.2 and 2.3 give the following data for each reactor.

- (1) name,
- (2) nuclear steam supply system vendor,
- (3) turbine - generator vendor,
- (4) utility operating the plant,
- (5) thermal output of the plant, and
- (6) initial date of commercial operation.

It is readily evident from these tables that the reactors in this data base encompass a variety of designs and ages. The data are concentrated in plants between three and ten years old. Only 1/7 of the plants are more than ten years old. Therefore, the population is heavily biased toward young plants. There is also a strong negative correlation between years of operating experience and plant size. The oldest plants are also the smallest plants. These trends are illustrated in Figure 2.1.

Since most of the current generation of plants have been custom designed to utility specifications, it is possible that not all of the operating experience from these plants will be directly applicable to the larger, more standardized designs currently under construction. This suggests that caution must be applied in extrapolating the results of any analysis to other plant populations or extending operating experience beyond the initial ten years of plant operation.

2.2.2 Licensee Event Reports (LER)

The NRC has computerized some of the abnormal occurrence reports (AOR) and the licensee event reports (LER) so that they can be sorted by predetermined classifications. Since neither SCRAM nor SHUTDOWN is one of the key words, all the LER data must be searched by hand to isolate the plant trips. Furthermore, the completeness of the data is suspect because plants

Table 2.2. PWRs in the EPRI-SAI Scram Data Base

Unit Name	Reactor Vendor	Turbine Vendor	Utility	Power (MWT)	Date of Commercial Operation
Arkansas 1	B&W	W	Arkansas Power & Light	2584	12/74
Calvert Cliffs 1	CE	GE	Baltimore Gas & Electric	2570	5/75
Ft. Calhoun	CE	GE	Omaha Public Power District	1500	9/73
R.E. Ginna	B&W	W	Rochester Gas & Electric	1520	3/70
Haddam Neck	W	W	Connecticut Yankee	1825	1/68
Indian Pt. 1	B&W	W	Consolidated Edison	615	10/62
Indian Pt. 2	W	W	Consolidated Edison	2758	8/73
Kewaunee	W	W	Wisconsin Public Service Corp.	1721	6/74
Maine Yankee	CE	W	Maine Yankee	2440	12/72
Millstone 2	CE	GE	Northeast Utilities	2560	12/75
Oconee 1	B&W	GE	Duke Power Co.	2568	12/73
Oconee 2	B&W	GE	Duke Power Co.	2568	9/74
Oconee 3	B&W	GE	Duke Power Co.	2568	12/74
Palisades	CE	W	Consumer Power Co.	2472	1/72
Pt. Beach 1	W	W	Wisconsin Electric Power Co.	1518	12/70
Pt. Beach 2	W	W	Wisconsin Michigan Power Co.	1518	8/72
H.B. Robinson	W	W	Carolina Power & Light Co.	2300	3/71
San Onofre	W	W	Southern California Edison Co.	1347	1/68
Surry 1	W	W	Virginia Electric Power Co.	2441	12/73
Surry 2	W	W	Virginia Electric Power Co.	2441	5/73
Three Mile Island 1	B&W	GE	Metropolitan Edison Co.	2535	9/74
Trojan	W	GE	Portland General Electric	3423	5/76
Yankee Rowe	W	W	Yankee Atomic Electric Co.	600	1/61

Table 2.3. BWRs in the EPRI-SAI Scram Data Base

Unit Name	Reactor Vendor	Turbine Vendor	Utility	Power (MWT)	Date of Commercial Operation
Browns Ferry 1	GE	GE	Tennessee Valley Authority	3293	11/72
Browns Ferry 2	GE	GE	Tennessee Valley Authority	3293	10/73
Brunswick 2	GE	GE	Carolina Power & Light Co.	2436	11/75
Cooper Station	GE	W	Nebraska Public Power District	2381	7/74
Duane Arnold	GE	GE	Iowa Electric Light & Power Co.	1543	2/75
Hatch 1	GE	GE	Georgia Power Co.	2436	1/76
Humboldt Bay 3	GE	GE	Pacific Gas & Electric Co.	210	8/63
Millstone 1	GE	GE	Northeast Utilities	2011	12/70
Monticello	GE	GE	Northern States Power Co.	1670	2/71
Nine Mile Pt.	GE	GE	Niagra Mohawk Power Co.	1850	10/69
Oyster Creek	GE	GE	Jersey Central Power & Light	1930	12/69
Vermont Yankee	GE	GE	Vermont Yankee Nuclear Power	1593	11/72

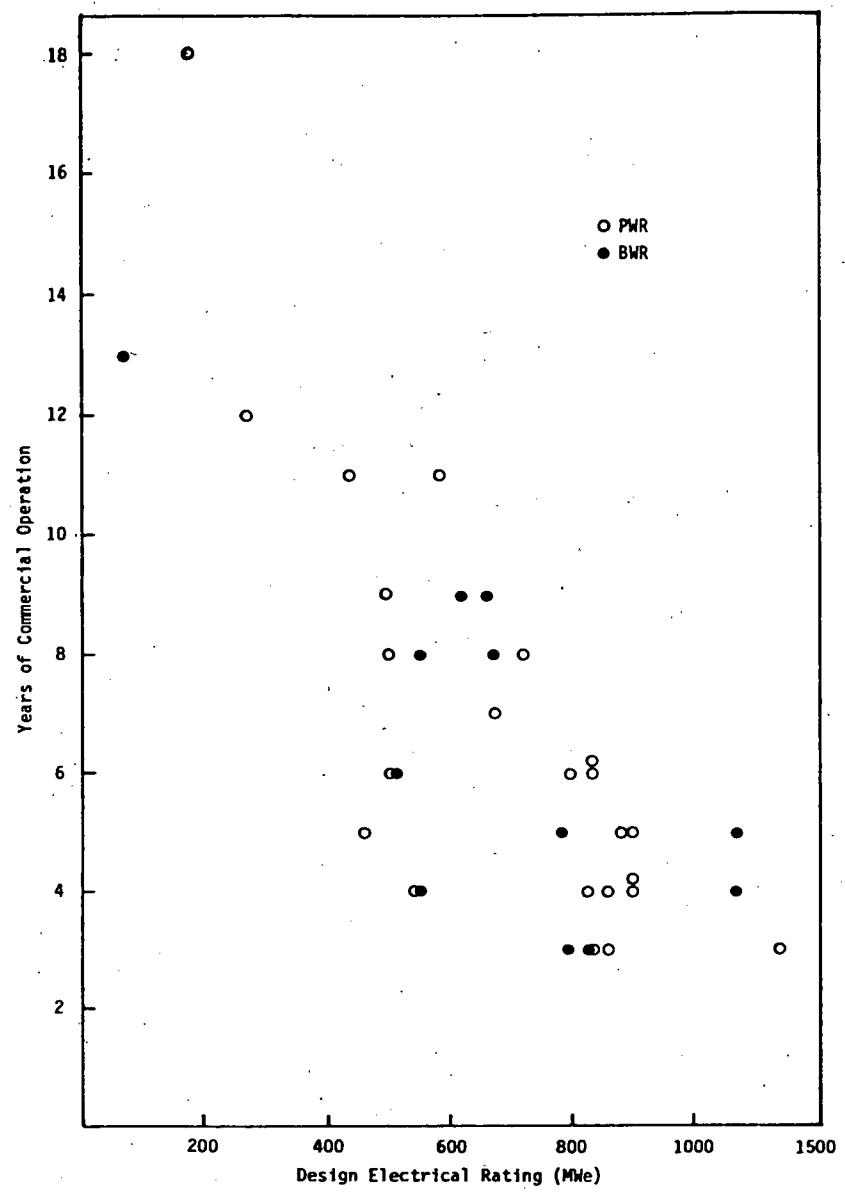


Figure 2.1. Comparison of Years of Operating Experience as a Function of Plant Design Electrical Rating

were not required to report all scrams until 1978. Therefore, LERs can only provide background information for events which are known from other sources. The LER data file, by itself, cannot be relied on to provide an adequate assessment of causes of plant trips. The complete computerized LER file was searched during this study, the results were compiled, and the events cross checked to determine if additional information beyond that discussed in Section 2.2.1 could be obtained. There was less than .1% additional information which was added by this review process.

2.2.3 Oak Ridge-Nuclear Safety Information Center (NSIC)

The data available from NSIC is based upon Licensee Event Reports (LER), Abnormal Occurrence Reports, and other available data from utilities and the NRC. Therefore, the level of detail, accuracy, and completeness is similar to that discussed above for LER. It is interesting to note that while some of the data are redundant, there is a portion of the data which is unique to each file. In addition, the NSIC data contain nearly 10,000 events which can be sorted by "SCRAM (REAL)" and "SCRAM (SPURIOUS)" for PWR and BWR populations, and by data. This sorting makes the job of comparison with other data sources much easier. However, these key words were not added to the system until 1976.

Because of the similarity between LER information and the NSIC file, and because of the lack of ability to search on "SCRAMS" prior to 1976, the NSIC file was not used extensively.

2.2.4 Nuclear Power Reliability Data System (NPRDS)

The data available in NPRDS are only for safety-related equipment. Therefore the plant transients due to non-safety related equipment are not necessarily included. In addition, since NPRDS did not become operational until 1974-75 and as yet only a few plants are reporting operational events and failures into the system, the amount of useful data is quite limited.

It is important to note that there has been no attempt to codify the operating data prior to each plant's initial reporting date. Therefore, virtually all data are from 1975 and later and only for the few plants which are participating.

If a large percentage of plants participate in this system and accumulate a significant number of years of operating experience, the system will become quite useful in the safety evaluation of nuclear power plants.

2.2.5 Operating Units Status Reports (OUSR) Gray Books - NUREG-0020

The "Gray Books" were first published in May 1974. They contain information on both safety-related incidents and events affecting plant availability. In general, the Gray Books are meant to contain a nearly complete compilation of the scrams which have occurred at each plant and the data of occurrence. However, SAI's experience with this source is that the accuracy of the data may be marginal due to the lack of verification of the event descriptions. In addition, there is a small percentage of the events (~10%) which are incorrectly reported or not reported at all. Also, the level of detail is severely lacking in a large portion of the reported scrams so that the related component and the failure mode would be impossible to identify. The completeness of this source is adequate, subject to the above problems, since the highest frequency scram incidents are reported; however, only data from 1974 through the present are available. On the whole, the Gray Books offer a source of data to augment other sources, especially since 1978 when the format was slightly changed, and the level of detail and accuracy appear to have improved significantly. In fact, this may be the only source of very recent data since other data files have a backlog which will not be factored into the data system when the search of the file is made.

2.2.6 Edison Electric Institute

The EEI data system covers steam, internal combustion, and hydro-electric plants. It contains information about availability, capacity factor, and other reliability measurements. It also contains descriptive information on units and major components and reports of unit outages. A recent EPRI study⁽¹³⁾ draws the following conclusions about the EEI data base:

- (1) It has a large number of omissions and inaccuracies.
- (2) The cause codes for outage do not allow for enough detail to accurately determine the cause of scram.
- (3) Comparisons of EEI data with the NRC Gray Books do not show much overlap.

2.3 Assessment of Plant Transient Data Base

The principal conclusions from our evaluation of these data sources are:

- (1) Statistically significant results may not be possible because of limited data.
- (2) The majority of the data are representative of the early years of the present product lines.
- (3) Due to differences in reporting practices, there is a considerable variation between reports from different plants.
- (4) Differences in the objectives, time of initiation, and the reporting format lead to differences in the content between data sources.

Even though several hundred reactor-years of operation have been recorded, the fact that approximately ten scrams occur per reactor-year implies that only a few thousand scrams have occurred. Since these scrams are spread over a diverse set of systems and reactor designs, the number of

scrams attributed to a particular component is generally low. Therefore, with the exception of the most frequent transients it is difficult to isolate, with high confidence, the component or system and the root cause ultimately responsible for the scram. In addition, the evolution of reactor designs has been such that the present product lines have only been in existence for a few years. Since these plants are also the most numerous, the majority of the available data is applicable to the infant mortality portion of the failure rate curve for the present generation of plants.

Another major problem lies in the variability of data from plant to plant. In some cases, significant differences in reported performance between similar plants can be traced to design or operating differences. In many instances, the variation in reported performance is simply due to misinterpretations or errors on the part of the reporting organization. It is a noteworthy fact that some organizations are consistently more diligent than others in compiling and disseminating information of high quality.

To alleviate some of the problems discussed above, it is judged that a careful analysis utilizing a limited but representative sample of plants is more likely to yield meaningful results than a less thorough study incorporating the entire population of operating units. Since the EPRI-SAI ATWS data base was developed specifically to investigate reactor scrams and is complete for the plants considered, it was selected as the basic source of data for the study. Supplementary data are being taken from the NRC Gray Books. In fact, the majority of the data for 1977-1978 are provided by the Gray Books.

3.0 CAUSES OF REACTOR SCRAMS

3.1 Analysis of the Data

The present study of the selected data involves the following four tasks:

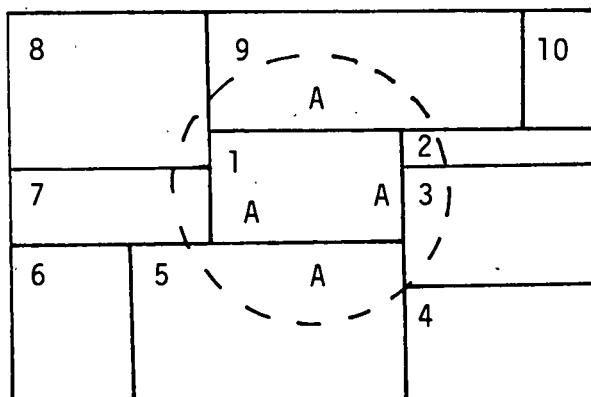
- (1) Identify the root cause of each scram.
- (2) Attribute the root cause of each scram to one class of components, plant system, or human error.
- (3) Prioritize the components leading to scram by frequency.
- (4) Analyze the number of scrams within each system as a function of plant age and calendar year.

The remainder of this subsection discusses how each of these items was handled.

The first task consists of identifying the root cause of each scram by carefully examining the text that describes the plant conditions leading to the scram. In some cases the descriptions were ambiguous and engineering judgment was utilized to infer a reason for the scram. In the development of this analysis and reduction of available data, all the scrams related to a component were defined as being attributed to that component, whether it was a component failure, malfunction or spurious operation.

Whenever possible, contributors (to the root cause) such as human error or faulty maintenance have also been identified. For example, in some cases human error is the root cause, while in other cases human error may contribute to the cause. In order to further clarify the distinction between a root cause and a contributor, consider the Venn diagram in Figure 3.1. Each rectangular region corresponds to one cause for scram. In this figure, ten causes for scram have been used for illustration. Let us suppose that Region 1 represents human error, while the other nine regions correspond to various components or systems. The area of each of these ten regions is

proportional to the relative frequency with which the corresponding system or component causes the scram. The area, A, enclosed by the dashed lines represents a contributor to scram. In this illustration the region A represents human error as a contributing cause to scram. Hence area A includes all of region 1 because when human error is a basic cause of scram, it must also be a contributor. As shown in the figure, area A also overlaps several component areas. Again the amount of overlapped area is proportional to the frequency of human error as a contributing cause to each of the component-related scrams.



*Region 1 represents human error as the root cause to the scram.
Area A represents human error as a contributor to the scram.

Figure 3.1 Human Error as a Root Cause
and a Contributor to Scram

After the primary cause of each scram has been identified, the causes are grouped into classes such as a particular plant system, a class of components, or human error. The definition of these classes posed a difficult problem. If a very detailed classification scheme is used, the number of scrams attributed to each class will likely be quite small, and the statistical fluctuations may dominate. On the other hand, definitions which encompass a large variety of components will tend to lose valuable details of the information. Although the classification scheme used in this study is somewhat arbitrary, it does attempt to strike a compromise between these two extremes.

The first classification recognizes the fact that design differences between boiling water reactors (BWRs) and pressurized water reactors (PWRs) may lead to distinctly different causes for reactor scram. Thus the classification scheme for the BWR differs from that of the PWR. However, within the BWR and PWR class, detailed design differences such as size or plant safety requirements are ignored. In other words, the major results are intended to be representative of a generic PWR or BWR. Variation in the cause of scrams within a given class of reactors was assessed by examining the bounding problem of nearly identical units (Babcock and Wilcox PWR) installed at different sites (see Section 4).

In addition to determining the relative contribution of each system to the total number of scrams, it is also useful to know the time-dependence of the relative contributions of each system. Analysis based on plant age as well as calendar year have been performed. The results of these analyses can be expressed symbolically by the function $f_i(t, T_0)$. The subscript i indicates the i^{th} system, while the variable T_0 represents the date when the plant began commercial operation. Thus $f_i(t, T_0)$ is the average scram rate attributed to the i^{th} -system. In reality, large variations may exist between the f_i of different plants due to plant-specific items such as siting, maintenance schedules, operator training, and specific component vendors and/or grades of components used.

Figure 3 shows as a function of time a hypothetical $f_i(t, T_0)$ function. The "bathtub" shape of this curve is typical of component failure rates as a function of age ($t - T_0$). Early in the lifetime of a system or component, there

is a large number of failures due to faulty operations or manufacturing defects such as poor electrical insulation, weak parts, bad assembly, and poor fits. During the middle period of equipment operation, an equilibrium stage is reached in which fewer failures take place because the "bugs" have been worked out of the system and few parts are wearing out. In general, they seem to occur when the environmental stresses exceed the design strengths of the part or equipment, or when preventive maintenance activities adversely affect their operation. Since it is difficult to predict the environmental-stress amplitudes or the part strengths as deterministic functions of time, these middle-life failures are often called random failures. As the item reaches old age, things begin to deteriorate, and many failures occur. This failure region is quite naturally called the wearout region.

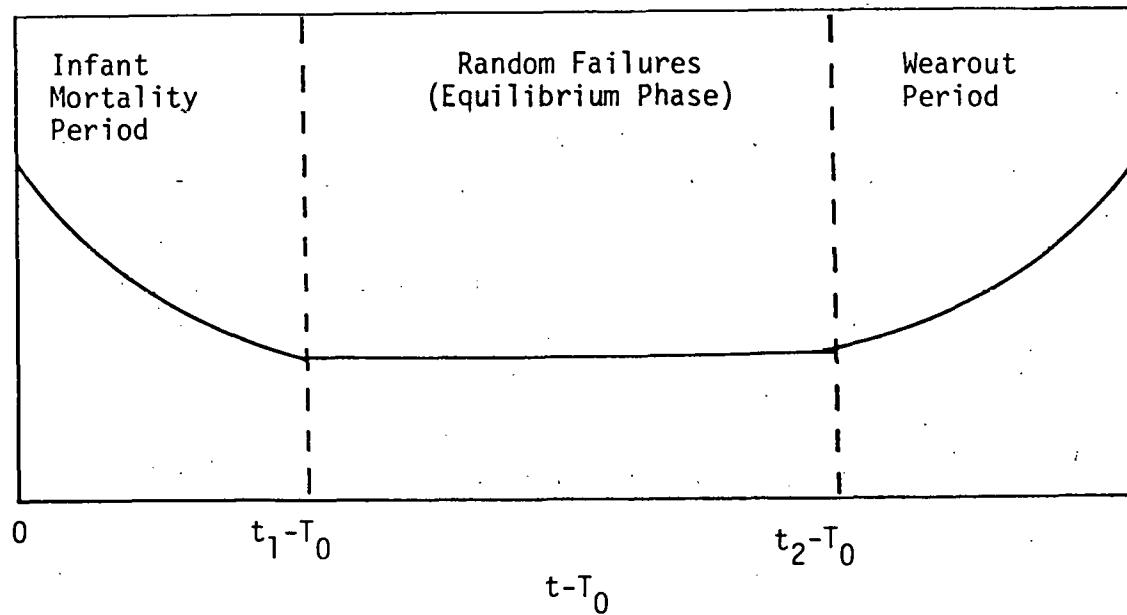


Figure 3.1 Typical Failure Rate Curve

When only one plant is involved, it makes no difference whether the data are plotted as a function of plant age or calendar time, because one is simply a linear translation of the other by an amount T_0 . On the other hand, if several plants with different T_0 are involved, it is generally more meaningful to normalize the data as a function of plant age rather than calendar time to display any apparent effects consistently.

3.2 Summary of Scram Population

The data for this report are taken from a population of thirty-five LWRs, all of which differ appreciably in size, design, and age. Appendix A provides a graphical display of the number of scrams per operating year as a function of plant age, with the frequency adjusted to reflect plant availability. The total number of scrams considered in this analysis is 1918, distributed among the thirty-five LWRs. Because of the wide diversity in the plants, it is important to apply the data carefully, recognizing that they represent a limited sample of custom-designed plants which have been treated as a homogeneous quantity. Therefore, while we have chosen to call our population "homogeneous" by neglecting the effects of size and detailed design features.

The analysis performed here is based upon data from the initial seven years of plant operation (less than one-fifth of the projected plant life.)

In order to provide a perspective on this population of reactors and the scrams which occur, Figure 3.2 graphically displays all the data of Appendix A for the PWRs, BWRs, and all LWRs averaged together. The histogram in Figure 3.2 was constructed by taking the total number of scrams at each reactor age and dividing by the total years those reactors were available to scram. For the i^{th} plant during the j^{th} year of commercial operation, let s_{ij} and a_{ij} be the number of scrams and the availability respectively. Then

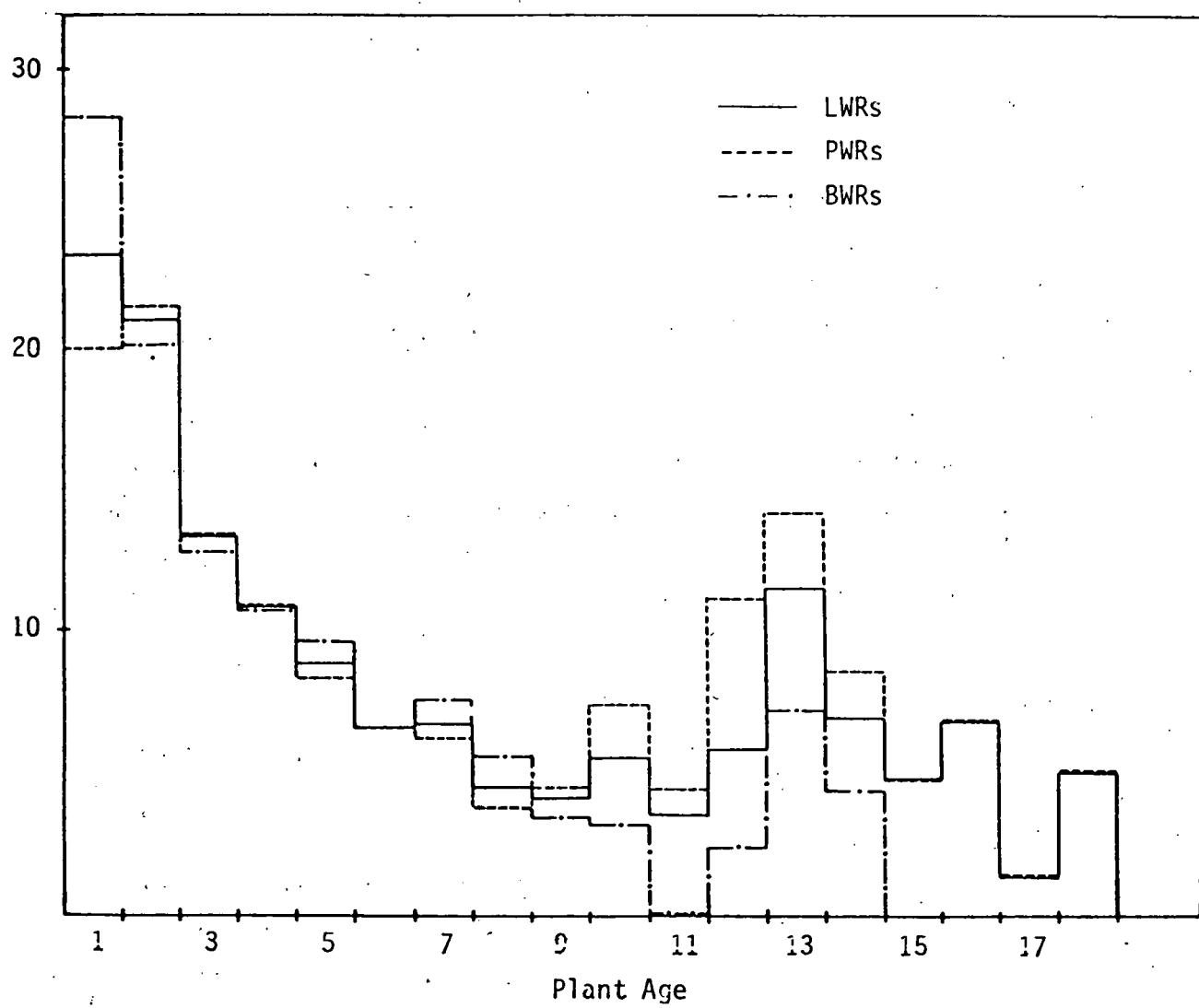


Figure 3.2 Summary of LWR Scram Frequency
(Scrams per Available Reactor Year)
As a Function of Plant Age

(Note: Frequency is calculated based upon the time the plant was available to operate, not upon the total calendar year.)

$$s_j = \frac{\sum_{i=1}^N s_{ij}}{\sum_{i=1}^N a_{ij}} \quad (3.1)$$

is the height of the j^{th} bar. This variable normalizes the number of scrams to the amount of time during which a scram was possible. This eliminates fluctuations in the number of scrams due to long downtimes such as occurred at Browns Ferry 1 and 2, San Onofre, Indian Point 1, and Humboldt Bay. The histograms in Appendix A are obtained by plotting

$$\bar{s}_{ij} = \frac{s_{ij}}{a_{ij}} \quad j=1 \dots N_i \quad (3.2)$$

where N_i is the number of years the i^{th} plant has been in commercial operation, and the symbols s_{ij} and a_{ij} have the same meaning as in Equation (3.1).

One way to summarize the data collected on plant transients for an overview is to calculate the frequency of plant trips per reactor year. This type of comparison will yield an estimate of the number of demands per year imposed on a "typical" plant for decay heat removal operation. It is appropriate to identify a value for scram frequency for all plant years represented and, in addition, for all plant years minus the initial two years. This is an attempt to estimate the expected plant trip frequency typical of a "mature" plant. The plant trip data are displayed in Table 3.1.

Table 3.1 Comparison of Scram Frequency Based Upon the EPRI Data File for Scrams from All Power Levels

(Note: These frequency estimates do not consider the plant availability during the year.)

	Frequency (Trips/Year)		
	PWR	BWR	LWR
Total No. of Plant Trips per Year (Includes all Years)	8.3	8.6	8.4
No. of Plant Trips per Year in a Mature Plant (includes all years past the initial two years)	5.6	5.6	5.6

For a reactor which is beyond its first refueling, the difference in scram frequency, when one considers only the calendar time for which the plant is available to operate, is as follows:

Case A (Use total calendar year)	Case B (Use available calendar time - ~ 72% available)
5.6	7.8

Before proceeding to the presentation of these data, the reader is reminded that any conclusions drawn from these data must recognize that they are based on operating experience and that historical trends may not be suitable for predicting future scram rates. Items such as

- 1) basic design differences
- 2) increase (or decreases) in the frequency of surveillance or maintenance on safety systems
- 3) improvements in operator or maintenance procedures

will have significant impact on the scram rates.

It is judged that the areas suggested in this report where reductions in the number of scrams can be achieved are applicable to the current population of plants and to those to be constructed in the near future. However, the data are not necessarily applicable to plant operation beyond ten years (i.e., from ten to forty years of plant operation).

3.3 Ranking of Principal Causes of Scrams

The principal objective of this study is the identification of the component causes of scrams coupled with suggestions of potential fixes which could be applied to reduce the frequency of the major contributors. The purpose of this section is to present the scram data from the 35 LWRs and rank the principal causes of scrams by related component. As discussed in Section 3.1, the analysis is divided generically into BWRs and PWRs. It must be noted that the component failure cause is uncertain for many scrams, as the root cause of scram is unknown or ambiguously described. Therefore, there is some potential variation in the rankings in Table 3.1 if better reporting were available. The number of component failures is based on evaluations of each of the individual reports⁽⁷⁾ from the utilities associated with each of the plants or from the NRC Gray Books.⁽¹⁰⁾

3.3.1 Ranking of the Principal Causes of Scrams in PWRs

The scram data for PWRs is summarized in Table 3.2 by system and by component within each system. Each scram is presented in the calendar year in which it occurred. This table of scram data provides the best one-page synopsis of the PWR causes of scram. Based upon this table and some additional information, the causes of PWR scram can be ranked to demonstrate where the major problem areas may be. Figure 3.3 is one such method of ranking these causes. Figure 3.3 is a histogram indicating the specific components that have been directly involved in causing PWR scrams. However, there are other ways of displaying the data that may add another perspective to the causes of PWR scrams.

In Figure 3.3, each of the scrams is attributed (as nearly as can be) to the specific component most closely involved in causing the reactor scram. If, on the other hand, all generic components are lumped together (e.g., pumps, valves, instrumentation and control), one finds a slightly different ordering of components as shown in Table 3.3. Here, all plant control and instrumentation has been placed in a single category, and this category becomes the dominant contributor to reactor scrams. In addition, there is a large fraction of scrams caused by disturbances in the plant electrical system. As the category is treated here, the electrical problems can be either outside the plant (off-site power) or problems on major electrical buses or local component electrical problems (e.g., breakers).

Previous efforts have not identified the top two items as major contributors to plant outages. Therefore, these problems can be characterized in general as short-duration outages but of a frequent nature and leading to reactor scrams.

Notably absent from Table 3.3 are those scrams related directly or indirectly to human error or faulty procedures (e.g., test or maintenance). One of the principal findings of this analysis is that while a specific component is involved in virtually all of the scrams, there is a large percentage of the reported scrams which are in some way attributable to

Table 3.2

CAUSES OF SCRAMS IN PRESSURIZED WATER REACTORS BY SYSTEMS AND COMPONENTS

SYSTEM AND COMPONENT	CALENDAR YEAR																		
	78	77	76	75	74	73	72	71	70	69	68	67	66	65	64	63	62	61	TO'
I. TURBINE-GENERATOR SET																			
Turbine Overspeed	2	4	11	5	3	4	3	6		2	3		1	1	1				4
Turbine Valve	4	10	2	4	3	3	1	1	1	1									3
Turbine Mechanical	2	2	6	6	3	4	1	1	1										2
Generator																			4
Other	5	4	8	6	6	8	2	2	4										4
TOTAL	13	20	28	21	15	19	7	9	5	3	3	0	0	1	4	1	0	0	14
II. FEEDWATER/CONDENSATE SYSTEM																			
Valves	5	5	7	7	6	1	2	1		1									3
Pumps	7	8	9	18	14	5	4	1	1		1	1		1	1				7
Condenser	1	3	1	6	3	1	1	1											1
Other																			1
TOTAL	13	19	17	31	25	6	7	3	2	2	0	1	1	0	2	5	0	0	13
III. CONTROL AND INSTRUMENTATION SYSTEM																			
Reactor	11	7	15	10	14	3	5	7	3	5	6			1	7	10	13		11
Feedwater	9	8	8	12	8	3	3	3	3	1	1								5
Turbine	2	7	9	13	2	4	3	3	1	2	1								4
Other			1	4	4	10	1	1	2										3
TOTAL	22	22	33	39	28	20	9	14	6	7	8	2	1	7	10	16	2	1	24
IV. ELECTRICAL POWER																			
AC Power*	3	7	8	11	7	4	3	3	1	(4)	(5)	(3)	(3)	(2)	(1)	(2)			(13)
Offsite Power, Switchyard, Station Loss																			
Partial Breaker	9	10	8	10	12	3	7	4	1										
DC Power	2	5	6	10	7	6	2	6	1										5
Diesel Generator	1	1	1	1															1
TOTAL	15	22	23	32	26	13	12	13	3	4	5	3	5	2	2	6	0	0	18
V. PRESSURIZER	1	0	0	1	7	5	0	0	0	0	0	0	0	0	1	0	0	0	15
VI. MAIN STEAM ISOLATION VALVE																			
Solenoid Operated Valve	3				2	1	1												7
Air-supply fails					2	7													9
Other	4				6	4	6	2	2		1								25
TOTAL	7	0	0	10	12	7	2	2	0	1	0	0	0	0	0	0	0	0	41
VI. STEAM GENERATOR CONTROL																			
Hi-Level Trip	3	7	16	11	7	6	4	4											59
Low-Level Trip	1	2	10	5	3														21
Steam/Feed Mismatch	6	5	15	6	10	9	11	9	1										73
TOTAL	10	12	33	27	22	18	15	13	1	0	0	1	0	0	0	1	0	0	153
VII. REACTOR COOLANT PUMP	4	5	4	4	2	8	2	1	1	0	0	0	0	0	0	0	4	0	35
VIII. REACTOR	0	1	1	5	6	1	0	1	0	0	0	0	1	0	0	0	0	0	16
IX. CONTROL ROD DRIVE	3	5	14	16	3	2	2	7	1	4	4	0	3	10	4	8	0	0	86
X. HUMAN ERROR																			
Operator		6	4		2	3	2	1	2										25
Maintenance	1	1	5		3	3	1												13
TOTAL	1	7	9	0	2	6	2	2	2	0	0	0	1	0	3	0	0	3	38
XI. TEST	0	1	6	6	2	2	0	2	0	1	0	0	0	0	0	0	0	0	20
XII. OTHER																			
Pumps	2	3	5	4	1	3													20
Valves	4	4	4	10	5		1												29
Pipes/Vessel	1	1	1	2	3	3	4	2	5										9
Turbine Trip																			20
Manual Trip	2	7	3	8	6	10	6	6	2	1	7	0	1	1	6	4	1	65	
TOTAL	9	15	15	25	17	21	9	7	0	1	8	0	1	1	8	5	0	1	143
XIII. OVERALL TOTALS	98	129	183	217	167	128	67	74	27	23	28	8	12	21	34	46	2	5	1263

*From 1963-1969, AC power as a cause of scram was not further broken down.

Figure 3.3. Distribution of Top Twenty Scram-Related Components in PWRs by System and Component

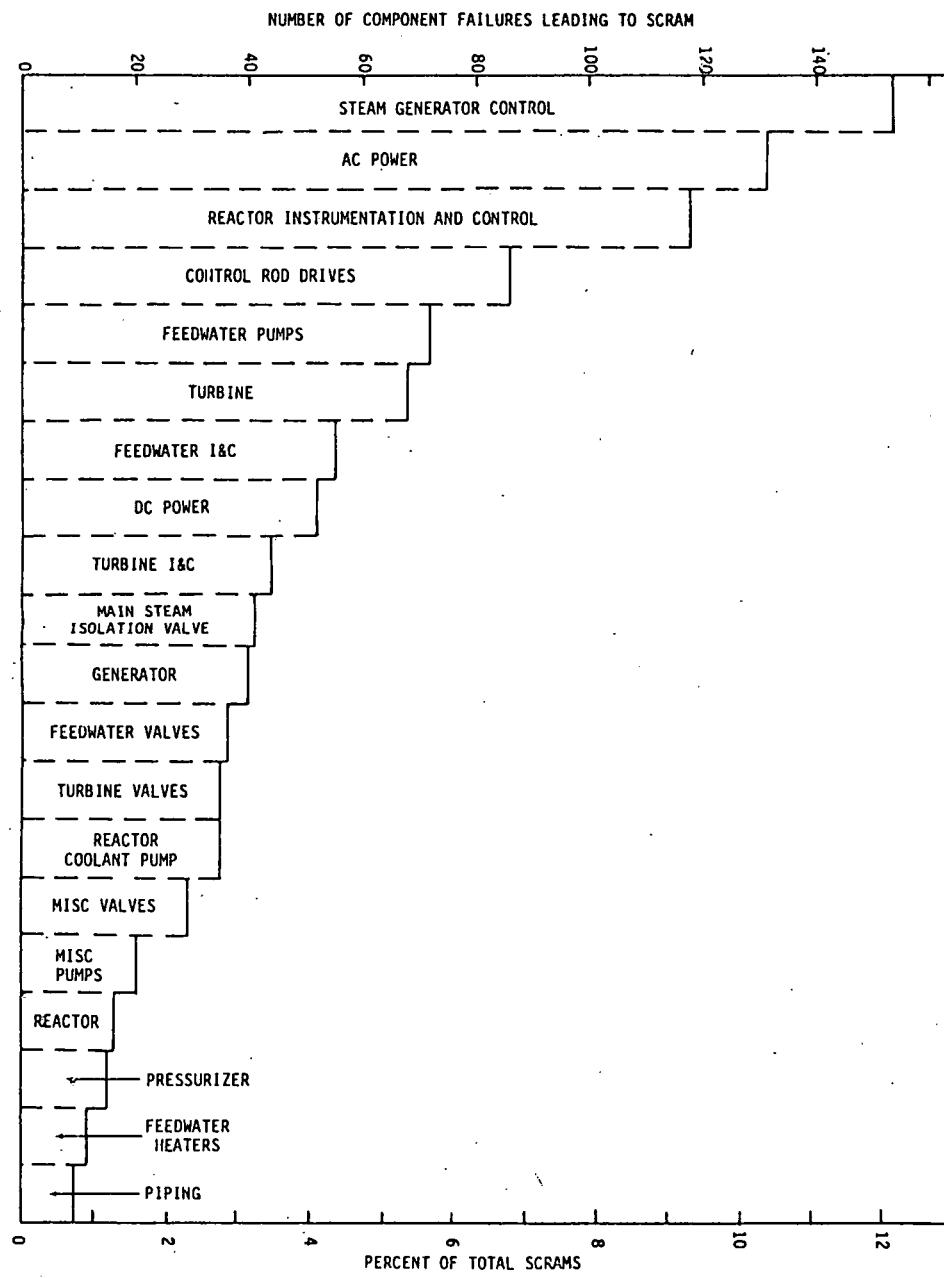


Table 3.3 Summary of Generic Component Causes for PWR Scrams

Rank	Component Type	# Failures	% of PWR Scrams
1	Control & Instr.	400	32
2	Electrical	186	15
3	Valves	140	11
4	Pumps	126	10
5	CRD	86	7
6	Turbine	68	5
7	Generator	40	3

a human error or faulty procedure. For PWRs, the number of scrams that can be related to human error or faulty procedures is approximately 13%. This places human error as one of the top three contributors to scrams. In addition, it is felt that even this estimate may be understated, since some utilities are not explicit in their identification of scrams related to operator error. While the category of human error or faulty procedure pervades all components, there appears to be a strong relationship between I&C and human errors that lead to scram. Therefore, the principal component classification in which the human errors show up as a contributing cause is in the I&C-related area.

3.3.2 Ranking of the Principal Causes of Scrams in BWRs

The format for the BWR data is similar to that presented for PWRs in Section 3.3.1. The BWR scram data is summarized in Table 3.4 by system, and by component within each system. The data are presented by calendar year. Table 3.4 provides the best one-page summary of the BWR scram data and their causes. Knowing this information, the causes of BWR scram can be ranked to highlight the principal problem areas. Figure 3.4 is a graphical display ranking the major causes of BWR scrams.

Table 3.4
CAUSES OF SCRAMS IN BOILING WATER REACTORS BY SYSTEMS AND COMPONENTS

SYSTEM AND COMPONENT	CALENDAR YEAR																	
	78	77	76	75	74	73	72	71	70	69	68	67	66	65	64	63	62	
TURBINE-GENERATOR SET																		
Turbine Overspeed	1	1	0	0	1	1	0	1	2									7
Throttle Valves																		
Control	6	1	5	7	1		3	1	2									16
Bypass	2		2	4		1												8
Stop			2	2	2	2	3											13
Other	4	8	2	2	2	2	3	12	1	1								17
Turbine Mechanical	2	4	3	2	2	2	3											17
Moisture Separator Drain Tank	3	4	4	7	5	2	2	12	3	1								17
Generator																		31
Miscellaneous			2	4	1			3	1	1								12
TOTAL	10	17	18	18	25	11	5	17	13	2	1	1	0	0	0	0	0	138
FEEDWATER/CONDENSATE																		
Valves	2	1		2	4	1	1	2										13
Pumps																		
Feedwater	4	4	3	2	4	5	1	1	1									25
Condensate Booster				2	8	2	2	1	4	1	1							12
Condenser	1	1	8	2	4	2	1											25
Drywell - leakage	4	3	1	1	1	1												8
Condenser Valves																		3
Miscellaneous			2		1	1		1										5
TOTAL	11	11	12	9	22	11	3	8	2	1	0	0	0	0	0	0	1	91
HEAT REMOVAL SYSTEM																		
Recirculation Pump	12	6	8	1	5	1	1	2										36
Main Steam Isolation Valve																		
Spurious Closure	5	7	3	3	5	4	2	5	5									39
Solenoid Valve Fails		1	1	1														3
Air System Fails								1										1
Relief Valve	2	4	2	3	5	5												16
Miscellaneous																		6
TOTAL	19	18	14	8	20	5	3	8	6	0	0	0	0	0	0	0	0	101
ELECTRICAL POWER																		
AC Power																		
Offsite Power, Switchyard, Station Loss	2	2	4		5	5		2		5	1							27
Partial	6	1	2		4	1	2	3	1									20
DC Power								1										1
TOTAL	8	3	6	0	9	6	3	5	1	0	5	1	0	1	0	0	0	48
CONTROL AND INSTRUMENTATION																		
Reactor	8	7	6	7	7	2	4	4	3						1			49
Feedwater	7	2		4	4	3	2	1	2									25
Turbine	4	10	5	5	4	1	4											34
Miscellaneous	4	2	4	2	2	1	1	2	3									21
TOTAL	23	21	15	18	15	8	11	6	7	3	0	0	1	0	1	0	0	129
REACTOR LEVEL	0	2	0	2	1	2	1	1	1	0	0	0	0	0	0	0	0	10
REACTOR INTERNALS	1	3	4	1	3	0	0	0	0	0	0	1	0	0	2	2	2	17
CONTROL ROD DRIVES	1	2	3	1	1	3	3	0	0	0	0	0	0	0	0	2	0	16
HUMAN ERROR																		
Operator	1	6	8	1	1			1										17
Maintenance	2	4	7	2	2	1		1	1	1								21
TOTAL	3	10	15	2	3	1	0	2	1	1	0	0	0	0	0	0	0	38
OTHER																		
Pump	1	3	3	1	1	1				1								10
Piping/Vessel	2		1															4
Refueling	3	3	2	5	2													16
Air Compressor	3	1	3	5	1													8
Miscellaneous	2	10	9	5	1			1	1									29
TOTAL	11	17	18	11	4	1	0	1	2	2	0	0	0	2	0	0	0	67
OVERALL TOTAL	87	104	105	70	103	48	29	48	33	8	6	3	1	2	5	3	655	

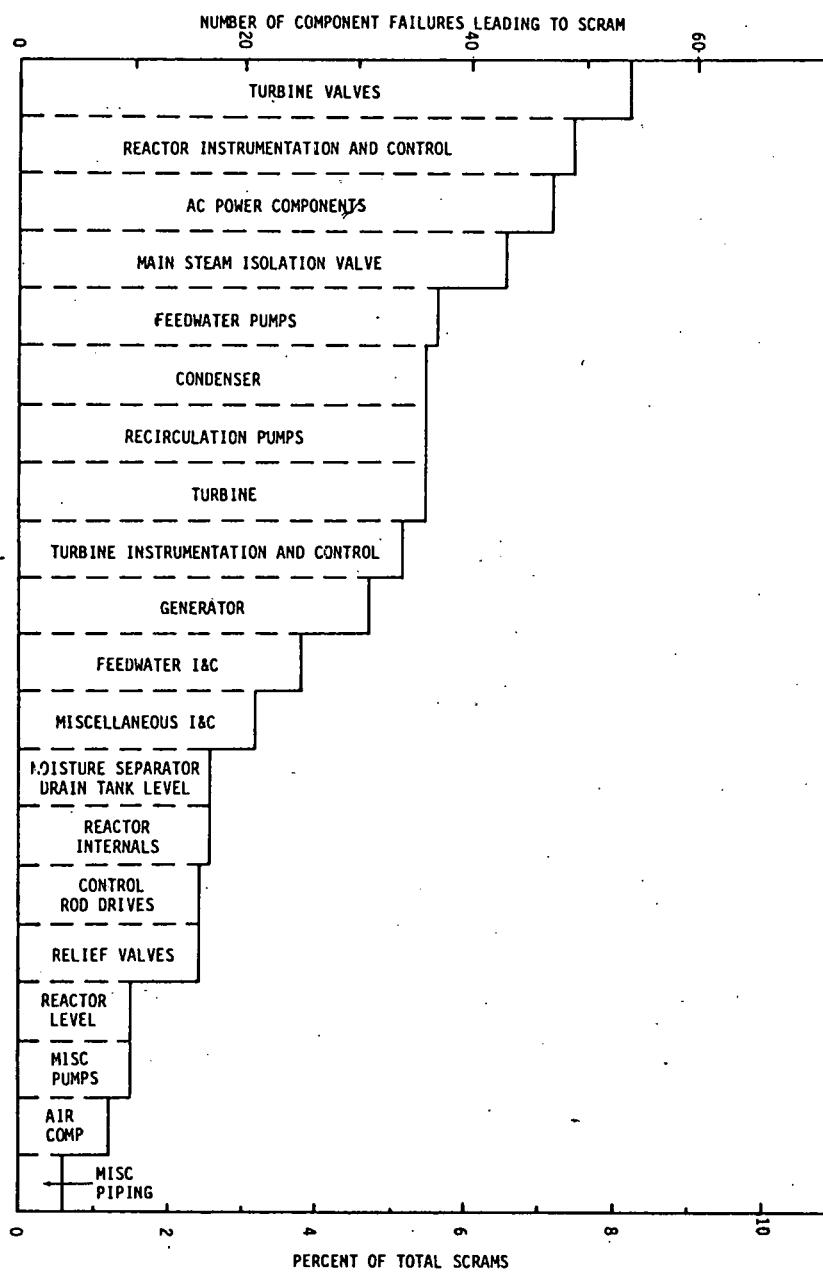


Figure 3.4. Distribution of Top Twenty Scram-Related Components in BWRs by System and Component

As in the case of PWRs, instrumentation and control is also the most frequent contributor to scram in BWRs. However, its relative contribution is considerably greater (32 vs 20 percent). If all generic components are lumped together (e.g., pumps, valves, instrumentation and control), a slightly different ranking is obtained. The fraction of valve- and pump-related scrams is somewhat greater in BWRs than in PWRs. On the other hand, the percentage of the electrical failures leading to scram is less for BWRs than PWRs.

Table 3.5 Rankings of Principal BWR Scrams by Generic Component Type

Rank	Component Type	# Failures	% of BWR Scrams
1	Control & Instr.	129	20
2	Valves	126	19
3	Pumps	83	13
4	Turbine	53	8
5	Electrical	48	7
6	Generator	31	5
7	CRD	16	2

3.3.3 Results and Conclusions

The average number of scrams per year in a mature LWR is approximately five. PWRs and BWRs have nearly the same overall average frequency of scrams. The major component failure causes of scram can be attributed to control and instrumentation failures (~20-32%). Other significant component causes include electrical component failures, valve failures, pump failures, AC electrical problems, control rod drive problems, turbine and generator failures.

The following items summarize the scram data presented in the section:

- 1) Control and instrumentation in general is the major cause of scram in both BWRs and PWRs. The failure modes and causes associated with these failures have not been reported for the most part.
- 2) Turbine valve problems are a major cause of scram in BWRs. Many of these scrams take place during testing of the valves.
- 3) Reactor trip due to reactor instrument failures is a major cause of scram in both PWRs and BWRs.
- 4) AC power component failures are a major cause of scram in both PWRs and BWRs. These components range from switchyard failures and loss of off-site power to open breakers and individual motor control center failures.
- 5) MSIV closures are a major cause of BWR scrams and a significant cause of PWR scrams. Solenoid valve failures and air supply leakages are leading causes of MSIV closures.
- 6) CRD problems are a major cause of PWR scrams and a significant cause of BWR scrams. Seal leakage and control rod drops are the leading failure modes of CRD problems.
- 7) Feedwater pump failures are a major cause of both BWR and PWR scrams. Both feedwater and condensate booster pumps are likely to fail, causing scram in BWRs. For the most part, failure modes and causes of feedwater pump failure were not reported.
- 8) Steam generator control failure at low power level is a major cause of scram in PWRs. The cause of low or high steam generator trip has generally not been reported.

5

- 9) Human error has not been reported to be a major cause of scram in both BWRs and PWRs. However, careful review of the scram data, detailed investigation, and some engineering judgment indicate it to be a major cause. (see Section 3.4).
- 10) Reporting of scrams has been incomplete and generally does not lend itself to the precise determination of the root cause of scrams.

3.4 Trend of Scrams as a Function of Plant Age

The overall trend of scrams as a function of plant age (see Figure 3.2) is a useful piece of information. However, these may be unique variations of the scram rate for individual plants or for different systems within the plant. Appendix A gives a graphical summary of the scram rate by plant for each PWR and BWR in our sample population. The following discussion is aimed at determining whether breakdowns by system or component will indicate differences in trends; that is,

- a) Are there early wearout phenomena occurring in some systems?
- b) Is there a learning curve associated with certain systems?
- c) Is there a constant scram rate associated with each system?

Before proceeding, one note of caution involved in describing events as a function of plant age is, for example, the potential for identifying a component failure as occurring in the fifteenth year of plant operation, when in fact the component had been replaced in the fourteenth year. In this example, the component failure actually occurs during the first year of component operation (i.e., "infant mortality"). This is indeed a hazard in this approach, since the data are inadequate to identify such replacements. However, the net result would be to overestimate the random failure rate of a component and to underestimate the infant mortality. In addition, since the available data effectively cover only about eight years of plant operation, there are

relatively few instances of major replacements that cause subsequent problems unless it is a case of a chronic problem of a certain plant. The purpose of this type of analysis is to uncover any striking trends that might be attributed to:

- a) Systematic trends in all plants for all components
- b) Training of personnel: this may be crucial to the cause of "infant mortality" if inexperience is a root cause of these early plant component failures.

3.4.1 Trends in Scrams for PWRs

The total scram rate determined for an "average" PWR as a function of plant age is given in Figure 3.6. Note that the plant availability has not been factored into this figure. The intent of this figure is to present the average trend of the PWR population sampled in this report. This overall trend then serves as a baseline to compare the trends that are developed for individual system or component types.

Figure 3.7 shows the specific trends of scrams related to major systems as a function of plant age. The following discussion points out the nature of these trends and the assumptions used to construct the figures.

- Steam Generator Level Control: One of the difficult tasks involved in operating a PWR is the proper control of the steam generator level to avoid the following:

- a) Low water levels which may lead to excessive tube dryout and unacceptable stresses on the tubes
- b) High water levels which may lead to moisture carryover into the turbine

Therefore a balance must be maintained among:

- 1) Power generated in the core
- 2) Steam flow out of the steam generator to the turbine
- 3) Feedwater flow into the steam generator

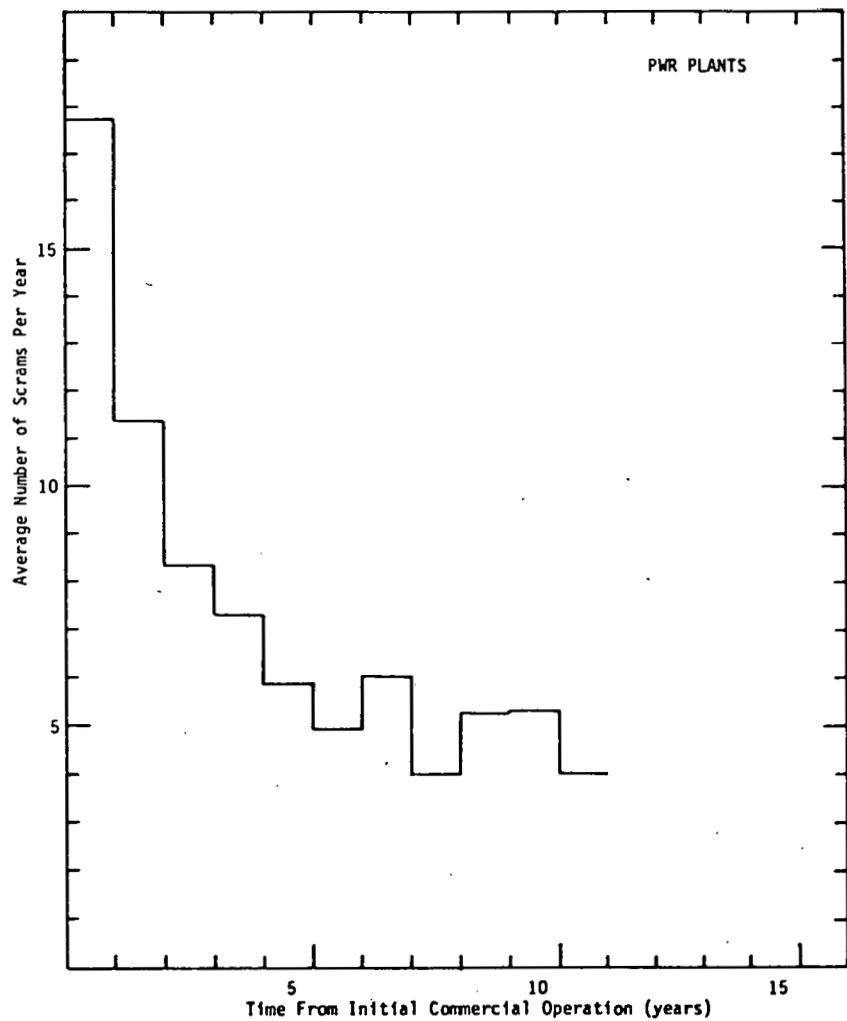


Figure 3.6. Average Frequency of Scram Events for PWR Plants as a Function of Time From Initial Commercial Operation

SUMMARY FOR PWR PLANTS

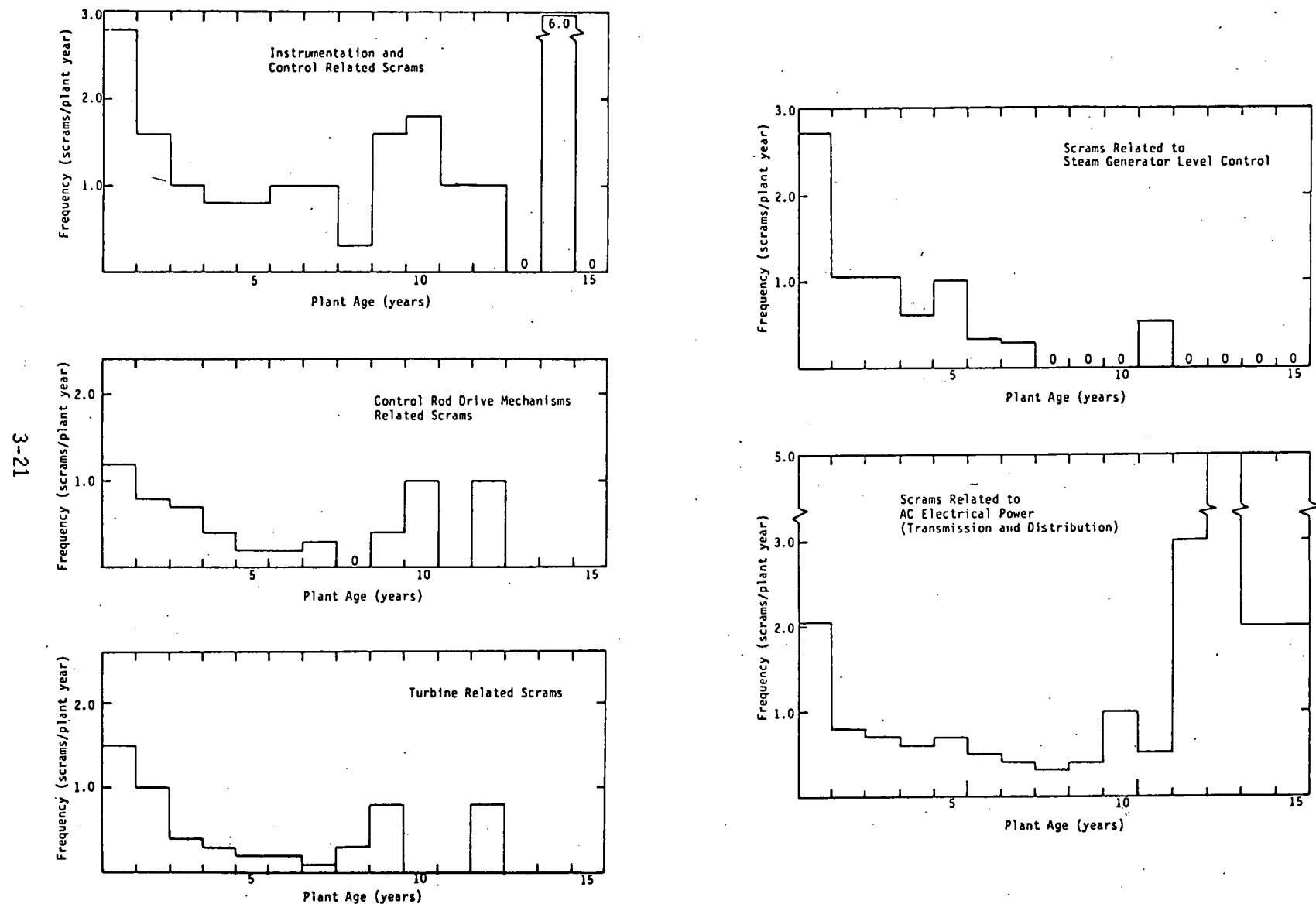


Figure 3.7. Trend of Scrams Related to Components in Specific Systems as a Function of Plant Age for PWRs

SUMMARY FOR PWR PLANTS

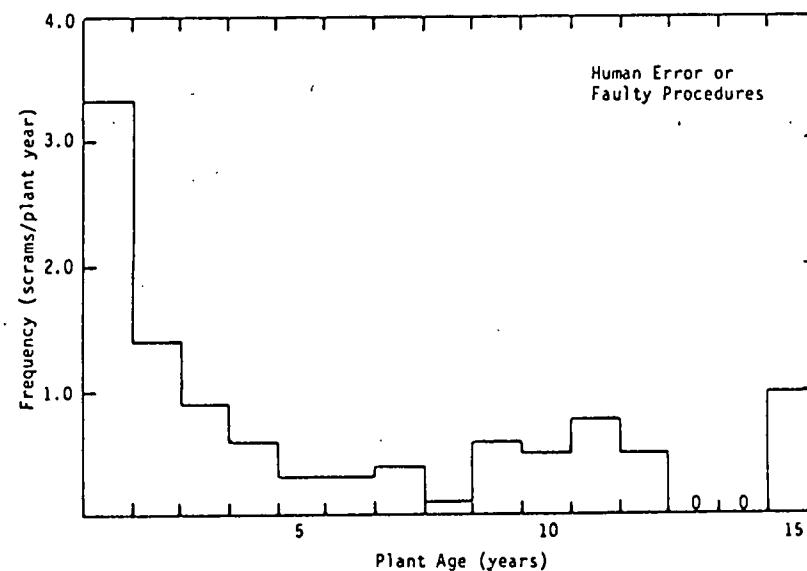
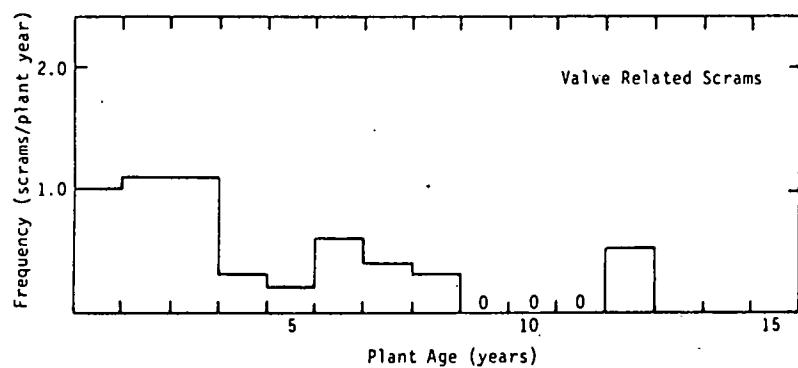
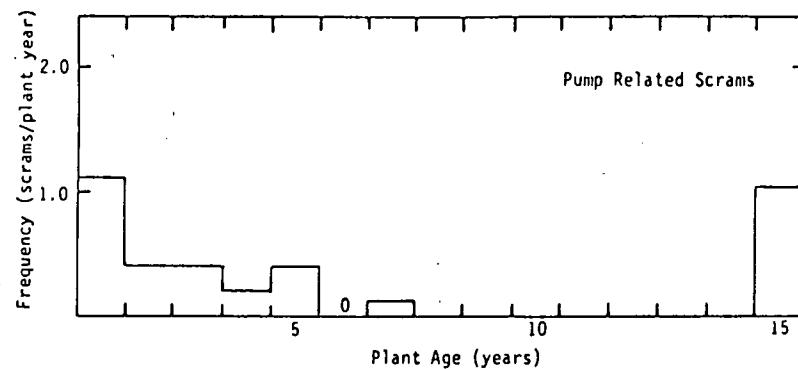


Figure 3.7 (Cont'd)

Operating experience indicates that whether automatic control systems are used or whether operator control of steam generator level is employed, that maintaining level is extremely sensitive to perturbations in the system. In Figure 3.7, we have lumped all perturbations together into one category to determine if there is a learning curve associated with steam generator level control. Indeed, it appears that after approximately three years of commercial operation, the control of steam generator level within specifications has been mastered, and a significant reduction in the average incidence of steam generator level control has been accomplished. This problem therefore appears to be one that is encountered early in plant operation and not one of a long-term nature.

A detailed investigation into the scrams caused by steam generator level control uncovered the fact that a significant fraction of the scrams resulted from loss of proper level during low-power operation* with the level in manual control. This indicates that human error or faulty procedures may be to blame.

- AC Electric Power (Transmission and Distribution): Electrical power is essential for plant operation. Operating experience indicates that even small disturbances in power can result in reactor scram. This category has included in it both:
 - a) Loss of offsite power cases which lead to a scram
 - b) Disturbances on the plant AC buses which cause loss of one or more components required for plant operation

Figure 3.7 shows that again the initial year of commercial operation has a relatively high incidence of scrams related to this system. Subsequent to the initial years, it appears that there is a relatively constant value, indicating that

* Low-power operation can be during startup, a controlled shutdown, or testing.

electrical power-related incidents are a persistent problem potentially throughout plant life. The limited data in the years 12 through 15 indicate a drastic increase in the scram rate related to electrical power; however, because of the small plant sample in these years, this effect should be used only to indicate an area which may require future monitoring to establish clearly if this is a potential long-term problem for the "average" plant. The nature of the problem with AC power supply varies from plant to plant, but the following are items that appear most often:

- 1) Loss of offsite power or significant disturbance on the grid
- 2) Human error in switching AC buses or in maintaining an electrical bus
- 3) Electrical breaker failure

- Instrumentation and Control (I&C): Virtually every major piece of equipment required for plant operation has both monitoring instrumentation and control circuitry. The scram protection for nuclear plants is designed such that loss of indication on instrumentation on critical equipment will lead to a scram. Given the ubiquitous nature of instrumentation and control, it is not surprising to see in Figure 3.7 that problems with I&C are a major contributor to scrams. As we have noted before, the initial year of operation has the largest number of scrams related to I&C problems. However, it is quite important to note that the "constant" level of I&C-related scrams reached after the second year of operation is significantly higher than the scram rates associated with other "systems".
- Control Rod Drive Mechanisms: PWR control rod drive mechanisms as used in this analysis include the mechanical portions of the mechanisms plus the logic and power circuitry required for their operation. There is a large

number of CRDMs used in a typical PWR. Although the reliability of each CRDM is quite high, the large population makes the probability of some failure relatively high. Since the CRDMs are fail-safe, the principal mode of failure is to drop a control element assembly (CEA) into the core. There are some instances of stuck CEAs or CEAs out of line; however, these are much less frequent. The trend shown in Figure 3.7 indicates a steady reduction in the number of CRDM-related scrams over the first eight years of plant operation; however, in the years to follow, there are indications that the frequency of failures may be increasing. However, because of the exceedingly small data sample, these indications can only serve as a signal to closely monitor CRDM performance as the plants progress in age.

- Turbine-Related Problems: Previous efforts⁽⁶⁾ in analyzing nuclear power plant data indicate that turbines can be a major cause of plant outages; however, these are generally planned manual shutdowns. In this report, the emphasis is on those component failures leading to a scram. Figure 3.7 shows that turbine-related problems are one principal source of scrams early in plant life. Most of these scrams are due to turbine control problems (i.e., electrohydraulic control system failures, turbine stop valve malfunctions). A small number are related to turbine bearings, lubrication, etc.
- Pumps: A generic summary of pumps for PWRs is also presented. Surprisingly, the incidence of scrams related directly to pumps is relatively small.
- Valves: A similar PWR generic summary for valves also indicates that the contribution to scrams from valve malfunctions is noteworthy but not an overwhelming contributor. It appears that valves are a lingering problem. Sandia⁽⁷⁾ has found that a majority of the reportable incidents from nuclear power plants as contained in the NSIC⁽⁸⁾ data base

are related to valves. However, in the case of scrams, valves are important, but not the chief contributor.

Human Error: An attempt has been made in this analysis to identify those scram events which are related to a human error or faulty procedure*. Caution must be exercised in these cases, since some utilities are not explicit in their reporting of human interaction with components which may have led to scrams. Therefore, it is judged that the scrams attributed to human error may be understated in their analysis; however, it is felt that the trend would remain approximately the same. Figure 3.7 gives the trend of scrams related to instances of human error or faulty procedures. Note that the initial two years of commercial operation uncover a substantial number of such errors. Once the management and operating crew learn the plant, there is a marked reduction in the incidence of human errors causing scrams; however, it appears that there is always some constant level of scram incidence caused by human error throughout the plant life (at least the fifteen years of data included in this analysis).

6

3.4.2 Trends in Scrams for BWRs

The scram rate determined for an "average" BWR as a function of plant age is given in Figure 3.8. This will serve as the baseline comparison for the trends of individual systems plotted in Figure 3.9. As in the case for PWRs, it is also useful to consider the variation in scram rate for different systems for the "average" BWR. Figure 3.9 shows the trend of scrams related to major systems as a function of plant age. The following discussion defines the assumptions used in this comparison and the nature of the dominant trends.

*As discussed in Section 3.1, all scrams that human error or faulty procedures identified as a contributing cause are included in this assessment of the trend.

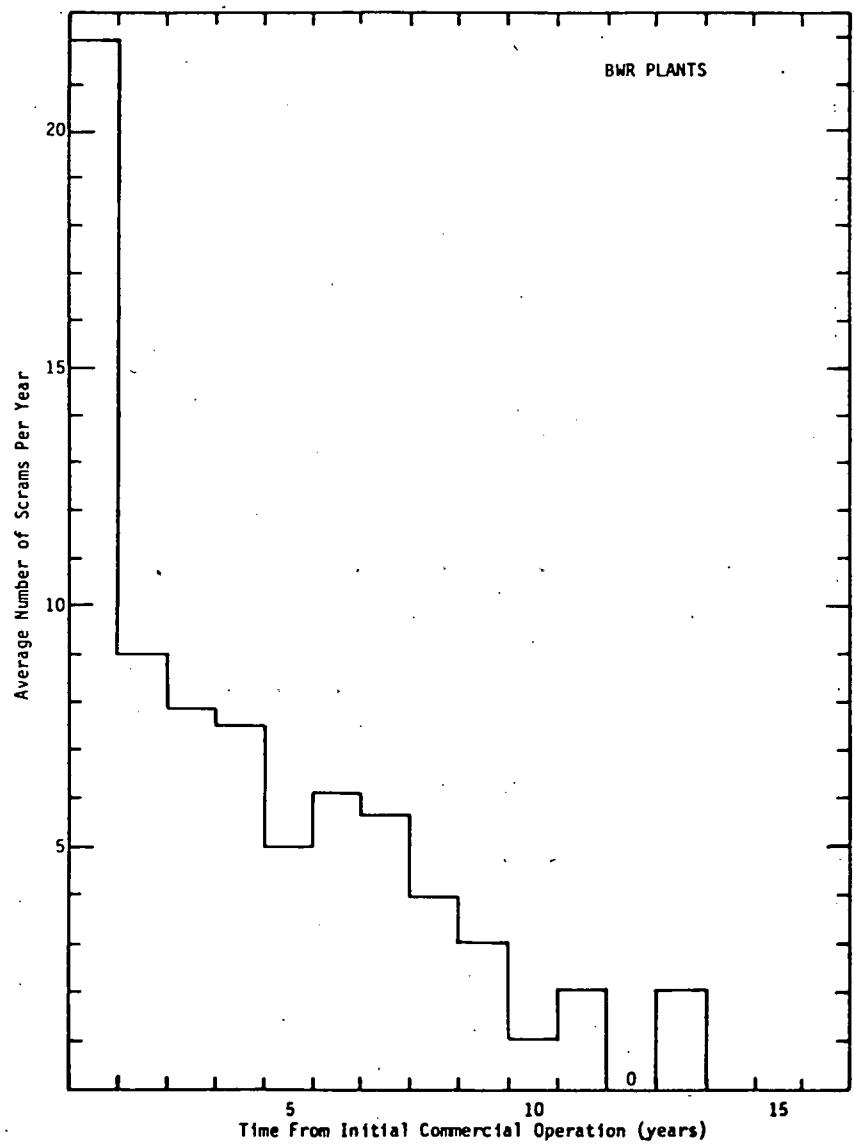


Figure 3.8. Average Frequency of Scram Events for BWR Plants as a Function of Time From Initial Commercial Operation

BWR SUMMARY

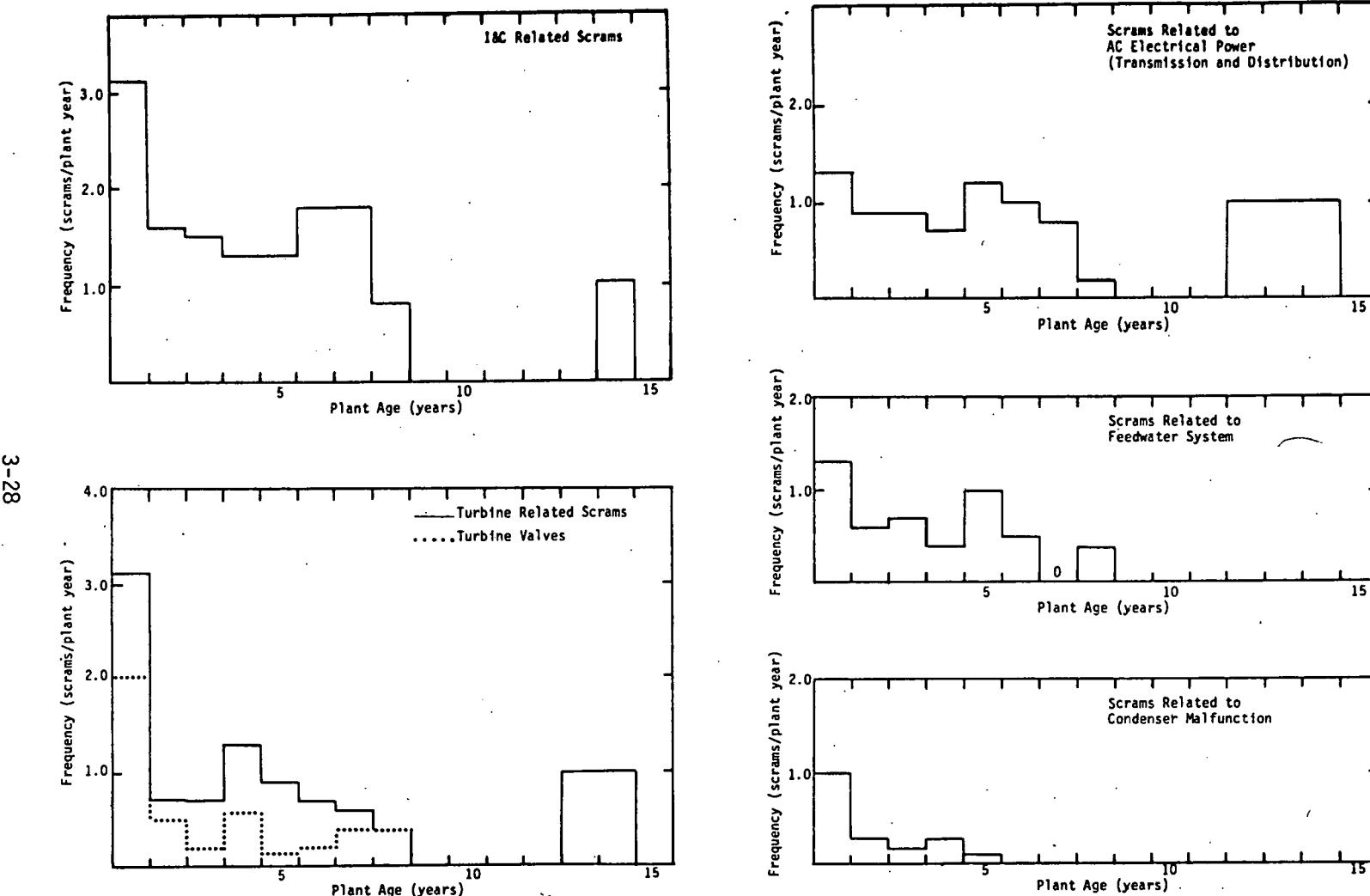
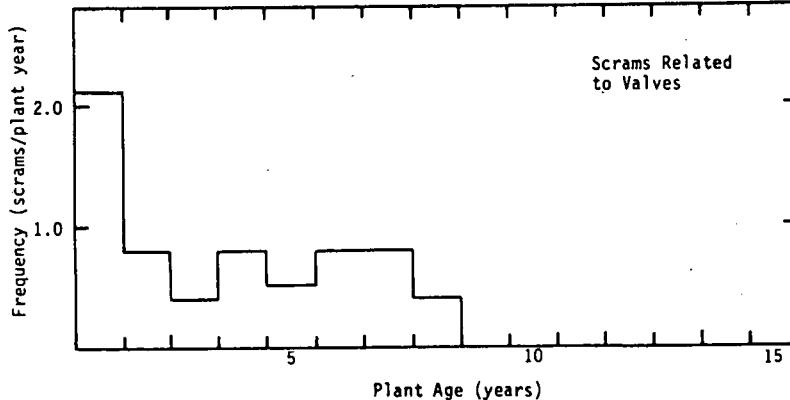
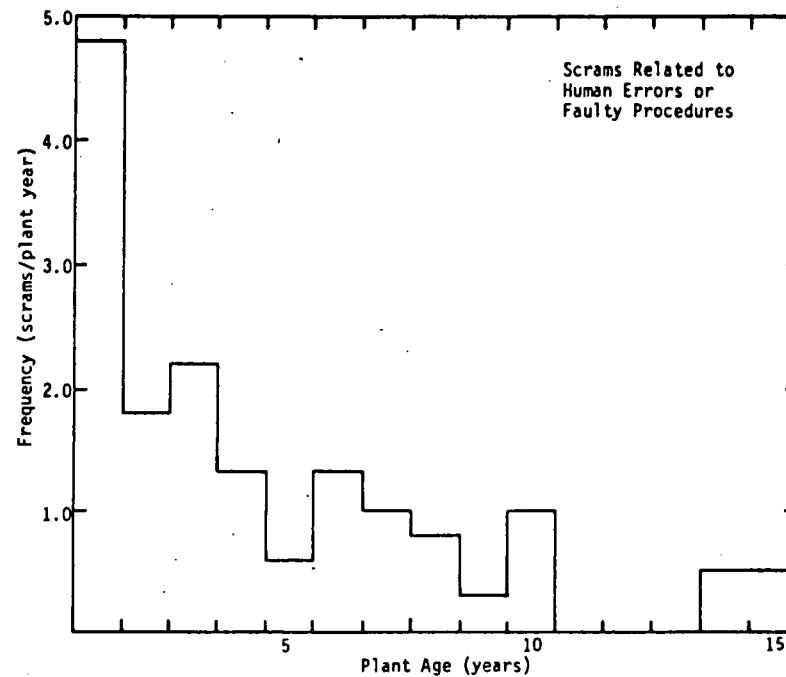
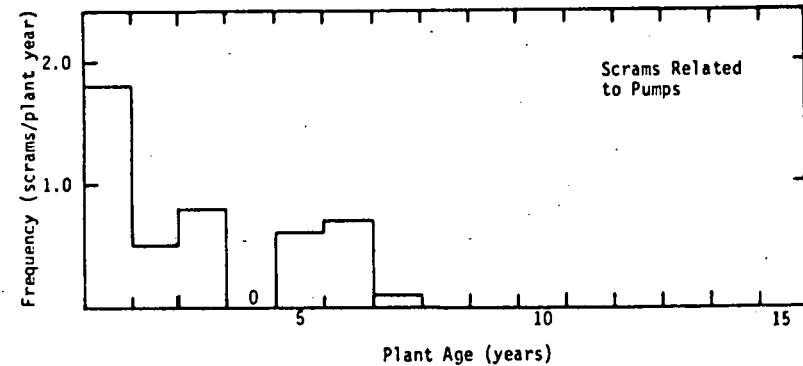


Figure 3.9. Trend of Scrams Related to Components in Specific System as a Function of Plant Age for BWRs

BWR SUMMARY



- Instrumentation and Control (I&C): As in the case of PWRs, I&C-related scrams are the major contributor for the population considered in this evaluation. But the contribution of I&C to BWR scrams is only about one-half of that determined for PWRs. (If one were to determine a constant equilibrium scram rate associated with I&C, the BWR rate would be higher than the PWR rate [1.4 for BWRs versus 1.0 for PWRs.]) It must be carefully noted that there is virtually no data for BWRs beyond eight years, and therefore we should concentrate on the data for the initial eight years of operation. Because the plant sample is relatively small, one expects a certain amount of fluctuation in the data. After the first year of commercial operation, I&C is responsible for approximately 1.4 scrams/plant year. This represents the highest rate of scrams for any system after the first year of operation. (Note that all I&C have been included in this assessment, while in Table 3.7 the reactor I&C has been separated out from the Balance of Plant contribution.)
- Turbine: It is interesting to note that previous work⁽⁶⁾ has established that Westinghouse turbines have historically a much higher accumulated outage time than GE turbines*. However, in the case of scrams, BWRs (GE turbines) have actually been associated with a higher frequency of scrams than PWRs. The BWR scram rate (assume zero slope) is approximately .9 scrams per plant year versus .3 to .4 for PWRs. Therefore, there is a higher frequency of short-duration outages caused by failures associated with the turbine, generally the turbine control system, in BWRs versus PWRs. However, the problems that lead to scram are basically control problems, whereas

*There is an approximate correlation between PWRs and Westinghouse turbines; and there is a one-to-one correlation between BWRs and GE turbines.

the longer-duration outages⁽⁶⁾ in turbines generally have been related to failures of the turbine blading. This indicates that BWR turbine control may be more sensitive than PWR control.

- AC Electrical Power: As noted in the PWR comparisons, persistent problems exist with the electrical distribution system both inside and outside the plant. The in-plant problems involve failures of breakers, buses, cable connectors, and faults in the generator. The extra-plant problems that are encountered are due to a wide variety of problems, including:
 - a) Weather
 - b) Disturbances on the grid
 - c) Transformer failures

There is virtually a constant level of scrams per plant year of approximately one per plant year based upon the 12 plant BWR data base experience. The PWR experience indicates a value of approximately .7 per plant year but similar in trend to the BWRs because of its constancy and persistence throughout plant life.

- Feedwater System: In BWR operation, a very important system (as in a PWR) is the feedwater system (for PWRs see Steam Generator Level Control). There is a wide diversity of failures within this system which can lead to a scram. The principal mode of failure is a failure of the feedwater control or instrumentation. However, approximately 50% of the scram failures are shared by the feedwater regulating valve malfunction or feedwater pump problems. Following the initial year of commercial operation, it appears that a constant failure rate of approximately .7 scrams per year is found for the "average" BWR.

- Condenser: A relatively small contributor to the overall scram incidence is caused by the condenser. Included in this plot are failures to maintain adequate vacuum due to
 - a) Failure of circulating water system
 - b) Failure of steam jet air ejectors
- Human Errors or Faulty Procedures: The reported incidence of scrams related* to human errors or faulty procedures in the BWR sample population indicates a very high scram rate during the initial year following commercial operation. The scram rate is higher than that found for PWRs (4.8 scrams per plant year for BWRs versus 3.2 scrams per plant year for PWRs). The 50%-higher scram rate may be due to an inherent difficulty in operating a BWR plant. However, a detailed review of the data suggests that a large fraction of the human errors is due to carelessness of maintenance personnel, coupled with poor design layout of crucial reactor instrumentation and control systems. The proper layout and mounting of the instrumentation cabinets would substantially reduce scrams caused by carelessness of maintenance personnel.
- Pumps and Valves: A generic grouping of all pumps and valves involved in plant scrams is also given. The most notable portions of these curves are:
 - a) The relatively high incidence of scrams induced during the first year of commercial operation
 - b) The constant levels attained in the second through eighth years of plant operation

*As discussed in Section 3.1, all scrams which have human error or faulty procedures identified as a contributing cause are included in this assessment of the trend.

4.0 PLANT SPECIFIC PROBLEMS LEADING TO SCRAM

This chapter discusses plant specific problems leading to scram. First, several plants of similar design (B&W) were lumped together and compared with a more general population. Next, a group of "sister" B&W plants were compared to the total B&W population. Finally, a review of some individual plants was made to determine whether large deviations from the "average" plant were occurring in specific plants.

4.1 Evaluation of Plants of Similar Design

Up to this point in this report, the focus of our investigation has been on generic PWR and BWR plants. There has been no other attempt at identifying plant specific problems. The question then arises as to whether any specific areas of potential concern can be identified if a set of plants of similar design are broken out from the overall population. The group of plants chosen here for review are the Babcock and Wilcox PWR plants with ratings of 2772 Mwt. These plants are the following:

Plant	Date of Initial Criticality	Thermal Design Rating (Mwt)
a) Three Mile Island 2	3/78	2772
b) Davis Besse	9/77	2772
c) Rancho Seco	9/74	2772

As the "control population", consider the other B&W plants which are operating. We shall use the other B&W plants to provide a baseline of comparison with the above plants. These baseline plants are:

Plant	Date of Initial Criticality	Thermal Design Rating (Mwt)
a) Oconee 1	4/73	2568
b) Oconee 2	11/73	2568
c) Oconee 3	9/74	2568
d) Three Mile Island 1	6/74	2535
e) Arkansas 1	8/74	2584
f) Crystal River 3	1/77	2452

The data presented here covers the time from initial criticality through 1978. In some cases the time between initial plant criticality and commercial power operation can be nearly one year. For all plants there is some valuable data which can be obtained which will be indicative of future plant operation. Note that the assessment in Chapter 3 indicates that the initial year of commercial operation has a significantly larger number of scrams than subsequent years. This also holds true for the time between initial criticality and the beginning of commercial operation. For the determination of generic plant risk, this initial time of testing may not be of general interest. However, on a plant specific basis, the initial plant operation prior to commercial operation can provide valuable information on an individual plant's operation and may provide an early warning system as to the possible causes of future scrams or component failures.

Figure 4.1 is a graphical summary of the total number of scrams at the operating B&W plants. The applicable data is provided in Appendix B. In general there are no deviations from the conclusions one would reach based upon the generic "average" plant analysis performed in Chapter 3. There is a characteristically large number of scrams early in plant life, and a rapid fall off to an approximately constant level. (The Oconee plants are the exception to this rule. There is no large peak during intial plant startup.) Therefore, on a whole plant basis, there are no indications

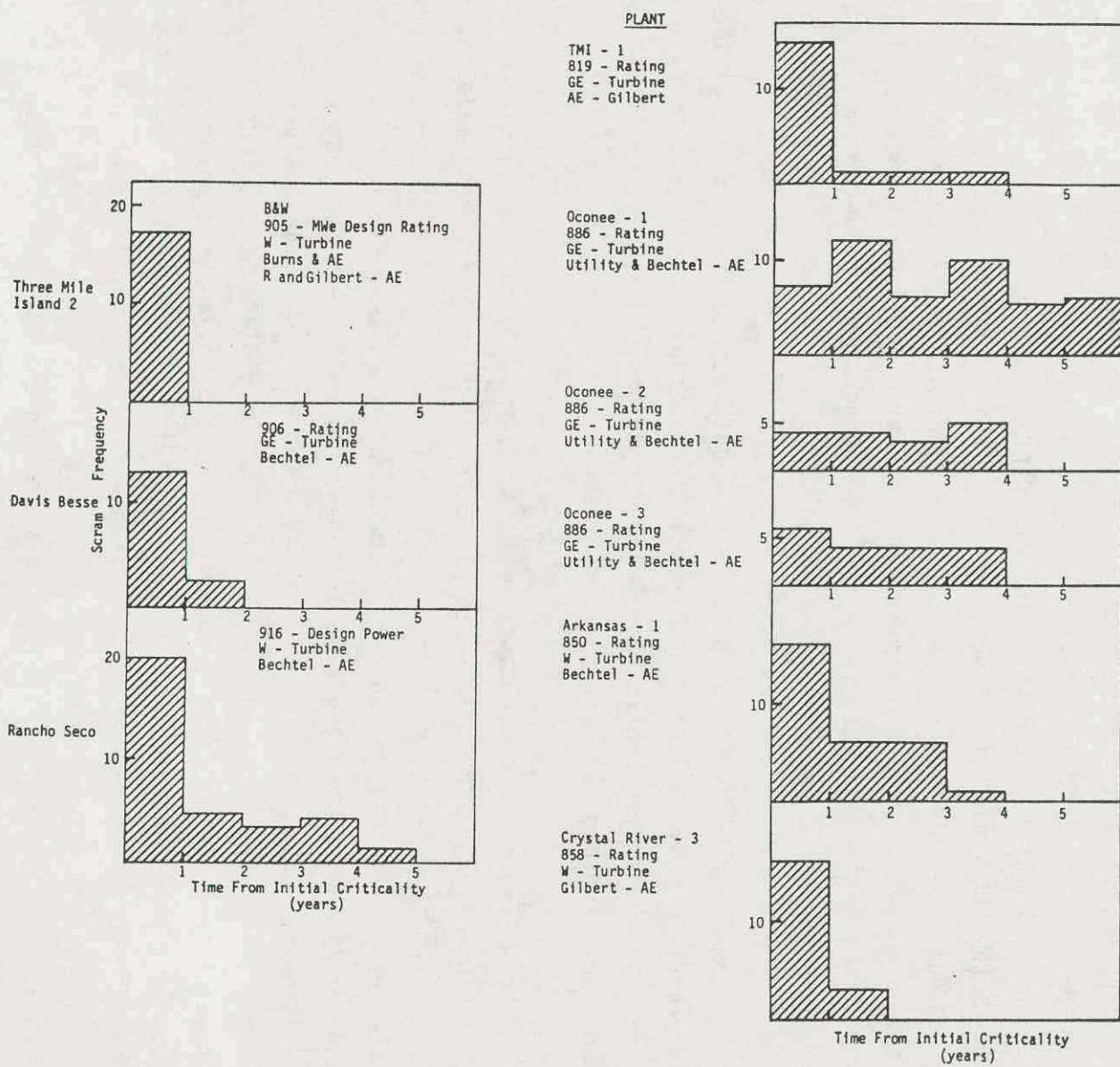


Figure 4.1. Total Number of Scrams at the Operating B&W Reactors as a Function of Plant Age From Initial Criticality

of any unusual problems at any one site. However, let us carry the investigation to another level of detail.

Figures 4.2 and 4.3 are summaries of the scrams related to two particular systems for the population of B&W plants. This type of investigation is the initial stage of a methodology to identify if chronic problems exist at one site. Note that it is very important that the investigation not be carried out by each reactor in a "vacuum". That is, each reactor plant must know what the average or anticipated number of scrams related to a given system or component is. Otherwise there is no benchmark to determine if the plant performance is unusually good or bad.

From Figure 4.2, it can be seen that both Oconee 1, the first of the B&W plants to come on line, and Crystal River 3 experienced greater than average problems with Control Rod Drive Mechanisms. However, the other plants have avoided the same problem area.

From Figure 4.3 a much more dramatic item is found. By comparing the latest B&W design (2772 Mwt) versus the other earlier design for feedwater related scrams, a rather striking result surfaces. The six plants which make up the "older" B&W design indicate no particular problem associated with the feedwater system. There are some initial problems in year one, but nothing surprising for TMI-1, the Oconee plants, Arkansas 1, or Crystal River 3.* However, two of the three plants of the 2772 Mwt vintage have a much different profile of feedwater related problems. TMI-2 and Davis Besse both have extremely high incidences of scrams related directly to the feedwater system. This rather crude review of the plant operating history points out that in these two plants a particular system has been the cause of a chronic problem and will probably continue to lead to scrams unless there is a design fix. From the limited data available, it appears that the specific problem

* Note that there are some increased scrams in the fourth and fifth years. This may correspond to control system changes, technical specification changes, or be due only to statistical fluctuations.

CRDM RELATED SCRAMS

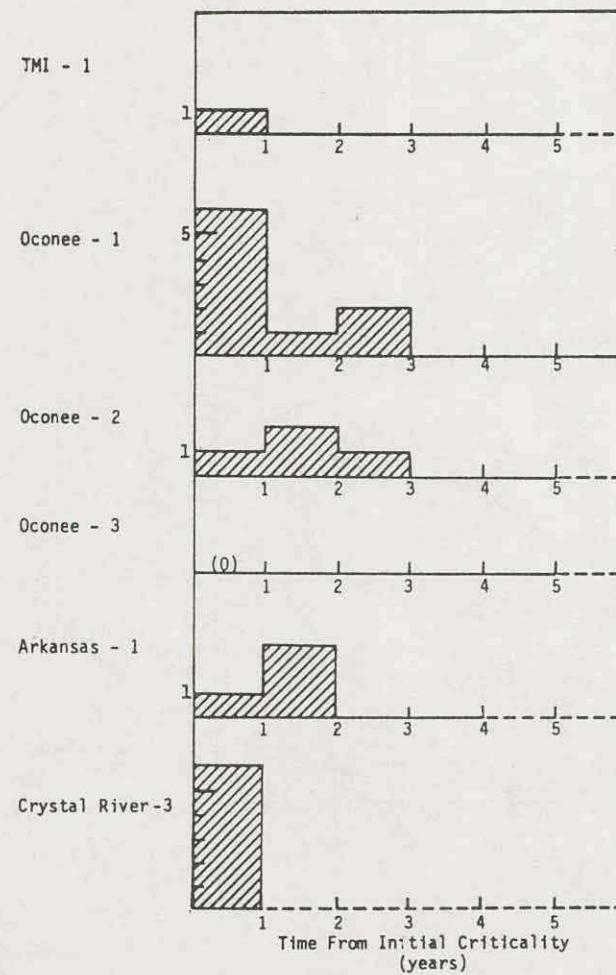
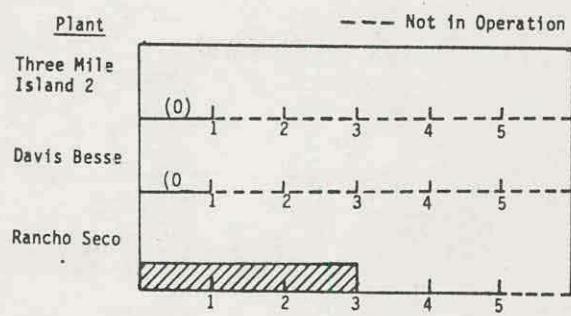


Figure 4.2. Comparison of the Number of Scrams Related to Control Rod Drive Problems Versus the Time of Operation for B&W Plants

FEEDWATER RELATED SCRAMS

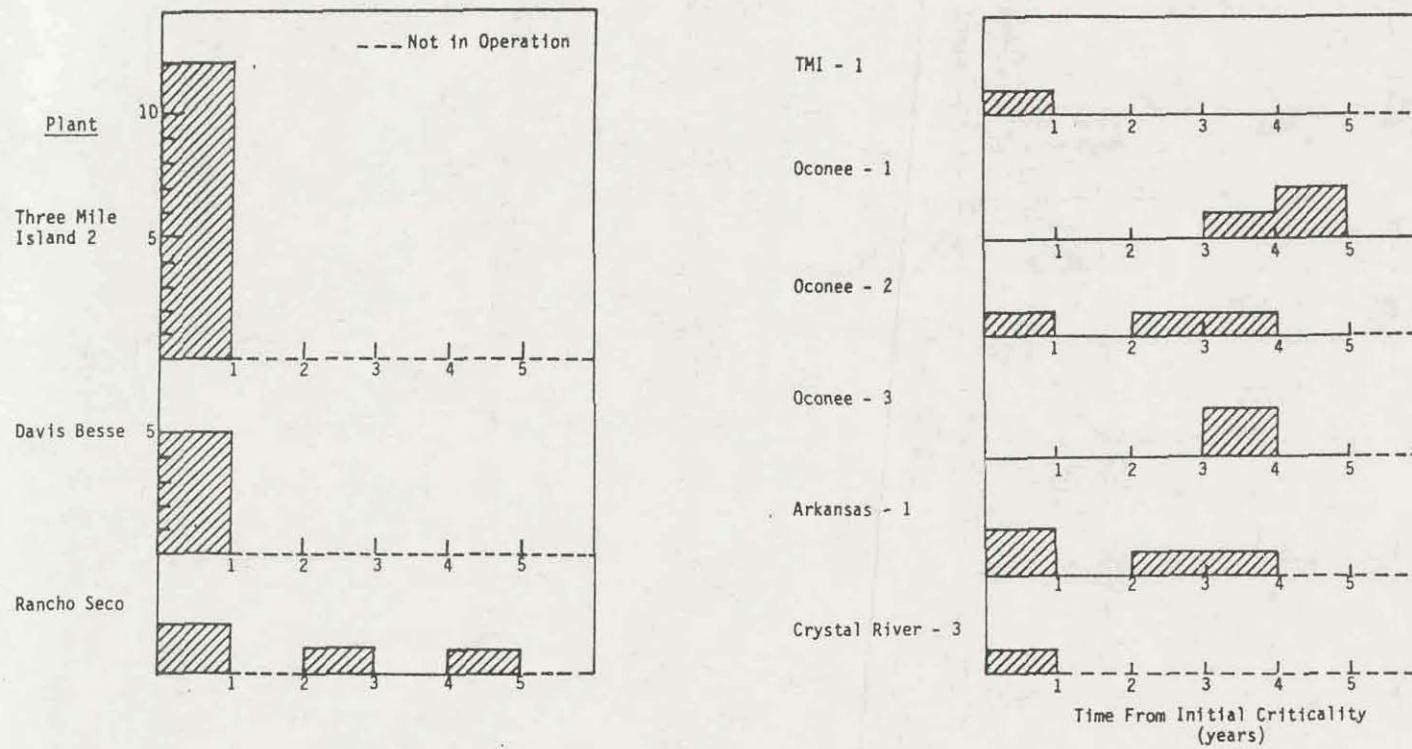


Figure 4.3. Comparison of the Number of Feedwater Related Scrams by Plant Versus the Time of Operation for B&W Plants

associated with these plants lies in the Integrated Control System (ICS) which monitors a wide variety of plant parameters and attempts to maintain proper feedwater flow to the once through steam generators (OTSG). Note that Rancho Seco (which is of the same design except it came on line much sooner than TMI-2 or Davis Besse) has not experienced a similar rash of feedwater related problems. This could be due to the fact that it does not have the "advanced" ICS installed. Further discussion of the ICS is provided in Chapter 5, "Design Fixes". Appendix B summarizes the data for these plants.

Some additional information gleaned from the operating experience data for B&W plants indicates the following:

- (1) If all the incidents of scram which are related to human intervention* through testing, maintenance, or operator error are compared with the total, these represent approximately 27% of all scrams at B&W plants, or a frequency of 1.67 per reactor year. This is approximately double the human related scrams determined in Chapter 3 for all PWR's, which represents approximately 13% of all PWR scrams based upon the data evaluated.
- (2) Turbine related incidents account for 17% of all scram incidents at B&W plants or an average frequency of one (1) scram per reactor year. If the first year is excluded, the frequency in B&W plants is approximately .5 per reactor year compared with .4 per reactor year for all PWR's.

*Note that these incidents may be included elsewhere when they are related to a specific component.

- (3) Instrumentation and Control related incidents are a major cause of scram in B&W plants as they are in other PWRs. B&W plant experience indicates that nearly 25% of all scrams are related to an I&C component. This compares with approximately 32% of all scrams in the PWR data evaluation in Chapter 3.
- (4) As seen in Chapter 3, AC power related problems represent a persistent source of scrams in both PWRs and BWRs. The B&W plant data indicates that approximately 5% of all the reported scrams are related to AC power transmission or distribution. This compares with approximately 15% of all reported scrams in our PWR population evaluation in Chapter 3.

It should be noted that this report has focused on those events which have led to scrams; however, there is additional information which can be gleaned from a review of critical reactor shutdowns caused by equipment problems. One notable area where this has been identified is in B&W plants where there has been a high incidence of pressurizer relief valve leakage requiring plant shutdown. These events, while they do not lead directly to a reactor scram, do have appreciable safety significance because they can have a large effect on the course of an accident once it is initiated by other component failures. In other words, operating experience can be applied to determine the principal causes of transient initiators, and it can be used to identify important issues in the accident scenarios which are postulated for LWR accidents. These topics will be discussed more fully in Volume 2 of this study which deals with components leading to shutdowns.

4.2 Evaluation of Single Plants with Persistent Problems

A look at specific plants indicates that many times there may be one or more troublesome problems which continue to plague a given plant. The apparent reason for the continuing problem appears to be that despite being

identified, it is never completely corrected, but rather a repair is performed which allows the plant to continue to operate while the root cause of the problem is never adequately addressed. Two such examples are cited here. However, at many of the plants there are characteristic problems which persist for an extended time until the root cause is finally eliminated.

First, at Millstone 1 (BWR), there occurred a series of turbine trip scrams related to level on the moisture separator drain tank. The scram frequency was 2.13 per reactor year. There were very few similar occurrences at other facilities, indicating that this problem was plant specific due to an anomaly occurring at Millstone 1. This is the case of a problem in the secondary plant, outside containment, leading to a chronic problem and causing a significant number of scrams (approximately 22% of all scrams at Millstone 1).

A second plant specific problem which can be noted as leading to a significant number of scrams at one plant is the difficulty with the D.C. instrumentation power bus at H.B. Robinson. As can be noted from Appendix A, the H.B. Robinson plant has an unusually high number of scrams per plant year. Approximately 10% of these are due directly to problems with the instrumentation bus; this is a frequency of nearly 1.4 per year. This fraction is significantly higher than those attributed to the "average plant"*. Therefore a major benefit from a plant specific analysis is that it would point up occurrences which are far outside of what the "state-of-the-art" seems to dictate, and therefore areas which could be significantly improved or upgraded.

* Note that Indian Point 1 also experienced a large percentage of instrumentation related scrams.

5.0 DESIGN FIXES

The preceding sections have identified and ranked, according to frequency, plant transients that lead to reactor scram. In many cases, it may be possible to introduce either physical or administrative modifications that can significantly reduce the frequency or consequences of these plant transient. Several such design fixes are suggested in this section. Both generic (BWR versus PWR) as well as plant-specific transients are addressed.

5.1 Introduction

The results of the data evaluation on the root cause contributors to reactor scrams presented in Chapters 3 and 4 indicate that a large variety of systems contribute to reactor scrams. Within each of these systems there are a large number of component types which may be the cause of these scrams. In addition, there may be a variety of types of components and/or vendors supplying them. In other words, there is a large number of components of diverse type whose failure or malfunction can lead to scram. There are not just a few major component contributors to the scram frequency, but a large number of components, each representing less than a few percent contribution.

While recognizing that no component is a dominant contributor to plant scrams, this section focuses on possible design fixes or other methods of reducing the number of scrams of those components which lead to the most frequent scrams. The most specific fixes will be for those for which a detailed root cause analysis exists; however, there are very few instances where a detailed failure analysis was performed by the utility to determine the contributors to the scram. From our analysis of the time dependence of the scrams, it is found that there are some classes of scrams which may appear to have a small overall frequency but which will actually tend to dominate the scram frequency in the later years of operation. The electrical power system appears as a dominant contributor to the cause of scrams after the first few years, even though the level of scrams remains nearly constant.

Even though the complete elimination of all plant transients is the continuing goal of both vendors and utilities, it is nevertheless quite unlikely that this will ever be accomplished. A more realistic objective would be to reduce the frequency of transients as much as possible and to design measures to mitigate the consequences of those transients that will inevitably occur. For example, human errors and random failures can be reduced by better training and quality assurance, but they will not be totally eliminated.

5.2 Instrumentation and Control

There are four aspects of the I&C related scrams which need to be delineated.

- (1) First, there are the component failures.
- (2) Secondly, there is poor design of the method of packaging the I&C components and their interfaces.
- (3) Thirdly, there is a high frequency of instrument drift problems leading to scram due to improper setpoint.
- (4) Finally, there is the human interaction with the I&C system during test and maintenance operations.

Each of these will be discussed separately:

- a. Component Failures: Each I&C component in a nuclear plant should be of high quality and chosen using low failure rate as one of the criteria. Increases in the redundancy of non-safety released I&C should be a principal area of investigation. The cost of the I&C component is probably negligible relative to its potential impact upon plant safety and availability.
- b. Packaging of I&C Equipment: Many of the instances of I&C failures involve actuation of scram signals due to bumping of cabinets by maintenance workers, condensation of moisture on electronic equipment, dirt or dust in relays, or high temperatures in the logic components. If the adverse environmental condition which may exist inside a nuclear plant are duly considered in the design of the I&C cabinets, then a number of these incidents could be prevented. In addition, the placement of the I&C cabinets could be important. A controlled access room which also has precise specifications on the environmental conditions would reduce a substantial fraction of the I&C failures by limiting the number of personnel who have access, and providing precise environmental conditions for the equipment.

- c. Setpoint Drift: Reactor scram due to instrument drift can be reduced by a more careful monitoring of the setpoints. It is suggested that the setpoints on components whose failure may lead to scram be continuously and automatically monitored. These readings can be fed into a computer system which compares them to the design values. Any noticeable drift from the desired value would then be annunciated to the operator. This system can be installed with a relatively small cost.
- d. Human Interaction During Test and Maintenance: While testing, maintaining or calibrating I&C equipment, there are a number of instances where spurious signals on the remaining channels lead to plant scrams. Two methods of changing the I&C system may lead to a reduction in these types of plant scrams:
 - (1) First, isolate each of the channels from the other scram channels. This isolation should be both physical and electrical so that: (a) jarring the one channel will not cause a trip signal from another channel; plus, (b) isolation of electrical signals so that disturbances on the tested channel do not affect the other scram channels.
 - (2) Secondly, a change in the design theory of the scram channels to require additional channels to actuate to cause a scram. This may be a 3-out-of-6 logic for PWRs. This would allow taking one channel out of service and still providing 2-out-of-5 logic. This accomplishes the result of reducing the number of scrams which may be caused by perturbations on a single channel which again have a high frequency of occurrence during test and maintenance.

5.3 AC Electrical Power

This failure classification is a major contributor to the frequency of scrams in a mature plant. The types of failures included are loss of off-site power, AC power bus failures, and electrical disturbances in the AC transmission and distribution system. We shall treat each item separately.

- a. Loss of Offsite Power: It seems prudent to have at least two transmission lines from offsite sources to supply to the plant if needed. Ideally these would be from two separated grids (e.g., at San Onofre there are connections to both San Diego Gas and Electric and to Southern California Edison grids). In addition, since weather (e.g., snow, sleet, lightning, hurricane) may cause a significant fraction of the loss of offsite power initiators, a method of reducing the influence

of the environment on the transmission lines and on the in-plant electrical systems would be beneficial. Typical examples of these fixes would be:

- (1) Putting transmission lines underground
- (2) Provide more effective methods of directing lightning away from electrical equipment (e.g., transformers, transmission lines)
- (3) Reduce power during severe weather conditions (e.g., hurricane, tornado) to minimize the impact of scram if it occurs

b. Electrical Disturbances on the Plant AC Distribution System:

A more stable power supply system should be provided which will damp fluctuations before they can cause scram on the reactor.

c. Breaker Failure: The use of more expensive breakers with lower failure rates could reduce the incidents of such failures. Alternatively, a redesign of the circuitry which would provide breakers in parallel. This circuit arrangement would require failure of both breakers before disabling the AC power system. The tradeoff, of course, is that both breakers must then operate correctly to protect plant equipment. For nonsafety-related equipment, this would result in component damage but may not have any plant safety implication.

Several examples of loss of AC power are included to provide background information.

Cooper 9/27/75	Scrammed from a turbine trip caused by reactor high water level. Trip followed a loss of a critical 4160 V bus which supplies the FWCS. The bus was lost due to a faulty relay.
Nine Mile Point 11/17/73	With one 115 kV line marked up, a relay on the other 115 kV line was bumped causing a loss of all offsite power.
Brunswick 2 5/30/75	The site suffered a loss of offsite power due to failure of a 230 kV breaker. The plant scrammed due to MSIV closure caused by low condenser vacuum.
Oyster Creek 12/29/72	Closure of main stop valves caused scram following loss of generator field due to opening of 24 kV potential transformer cabinet.
Hatch 1 2/14/75	Air was lost to the inboard MSIVs due to inadvertent opening of circuit breakers.

5.4 Steam Generator Level Control

In PWRs, instrumentation and control related to steam generator level is a leading cause of reactor trip. A large fraction (12%) is due to steam feed/flow feed mismatch in the steam generators during startup, shutdown or low power operation. Two possible fixes are:

- (1) Develop or improve the control system so that the steam generators can be better controlled at low steaming rates, and
- (2) Upgrade operator training so that the occurrence of flow mismatches or abnormal fluid levels during manual operation is minimized.

In the discussion of plant specific problems, one additional area which was uncovered as a potential area of investigation was the apparent sensitivity of the integrated control system (ICS) on Babcock and Wilcox plants. This section discusses the complexity involved in such an automatic system. The purpose of presenting a detailed discussion of this system is to highlight the complexity involved in a system of this nature. This complexity, when combined with a secondary system such as in B&W PWRs, may lead to unreliable operation. It may be prudent to reduce the complexity of the system, add redundancy to the sensors and control circuits, and overdesign the reactor components to allow larger tolerance bands for acceptable operation.

The integrated control system (ICS) provides the proper coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam generator, feedwater system, and turbine. When any single portion of the plant is at an operating limit or a control section is on manual, the integrated control system design uses the limited

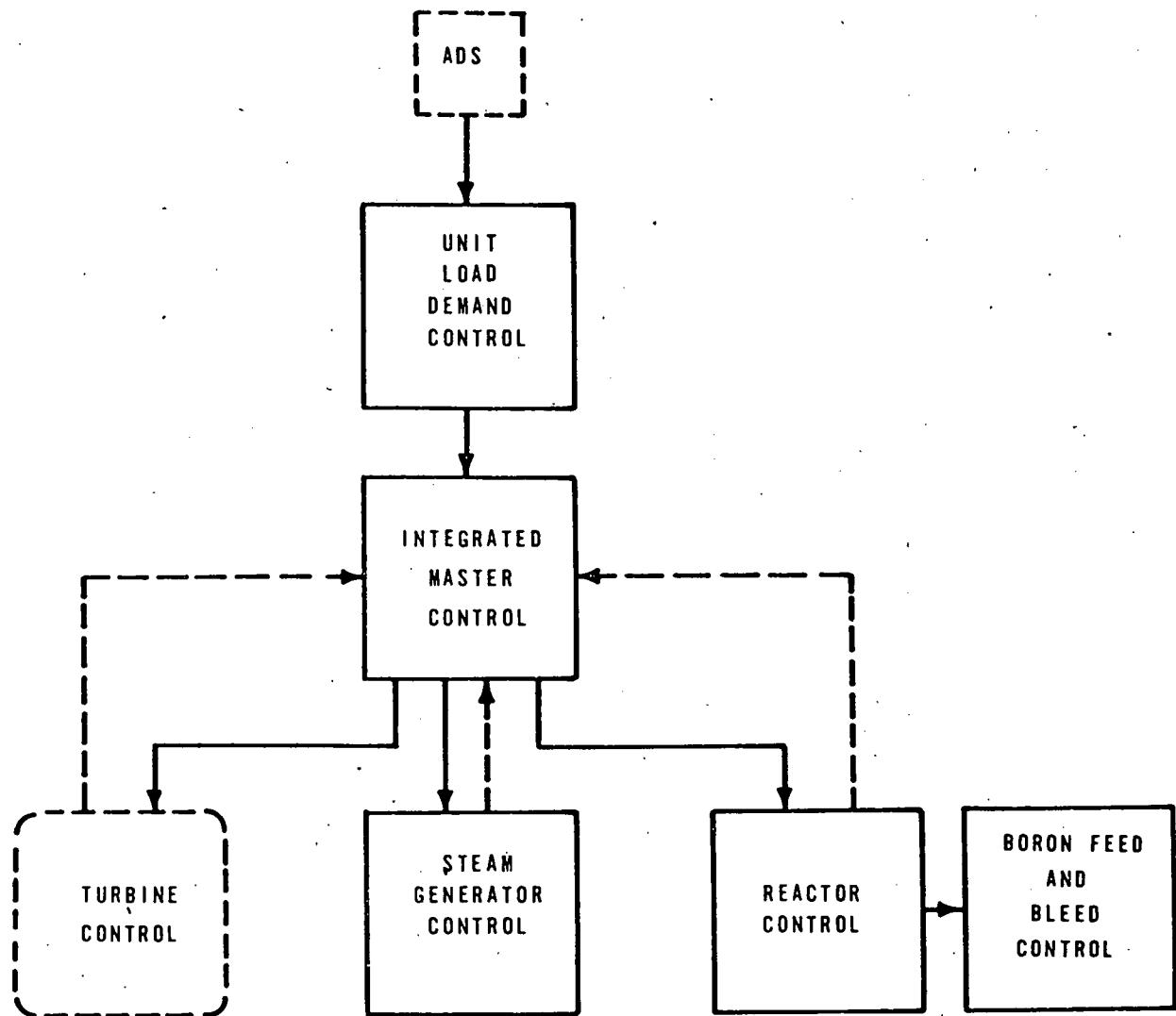
or manual section as a load reference. The integrated control system maintains constant average reactor coolant temperature between 15 and 100% rated power and constant steam pressure at all loads. Optimum unit performance is maintained by limiting steam pressure variations; by limiting the imbalance between the steam generator, turbine, and the reactor; and by limiting the total unit load demand upon loss of capability of the steam generator feed system, the reactor, or the turbine generator. The control system provides limiting actions to ensure proper relationships between the generated load, turbine valves, feedwater flow, and reactor power. The normal response of the reactor coolant system and the feedwater system to increasing and decreasing power transients is limited by the integrated control system as indicated in Figures 5.1 and 5.2.

The basic function of the ICS is matching megawatt generation to unit load demand. This is accomplished by coordinating the steam flow to the turbine with the rate of steam generation. To perform this function efficiently, the following basic reactor/steam generator requirements are satisfied.

- a. The ratios of feedwater flow and heat input to the steam generator are balanced as required to obtain the desired steam conditions.
- b. Heat input and feedwater flow are controlled:
 - (1) To compensate for changes in fluid and energy inventory requirements at each load
 - (2) To compensate for temporary deviations in feedwater temperature resulting from load change, feedwater heating system upsets, or final steam pressure changes

The integrated master has been designed to receive the megawatt demand signal from the unit load demand subsystem and convert this signal into a demand for the feedwater, turbine, and reactor control.

Control of the steam generator is based on matching feedwater flow to the feedwater demand produced in the integrated master control. However, loss of feedwater for a few seconds can result in boiling dry the steam generators and reactor scram. The basic control actions for parallel steam generator operation are:

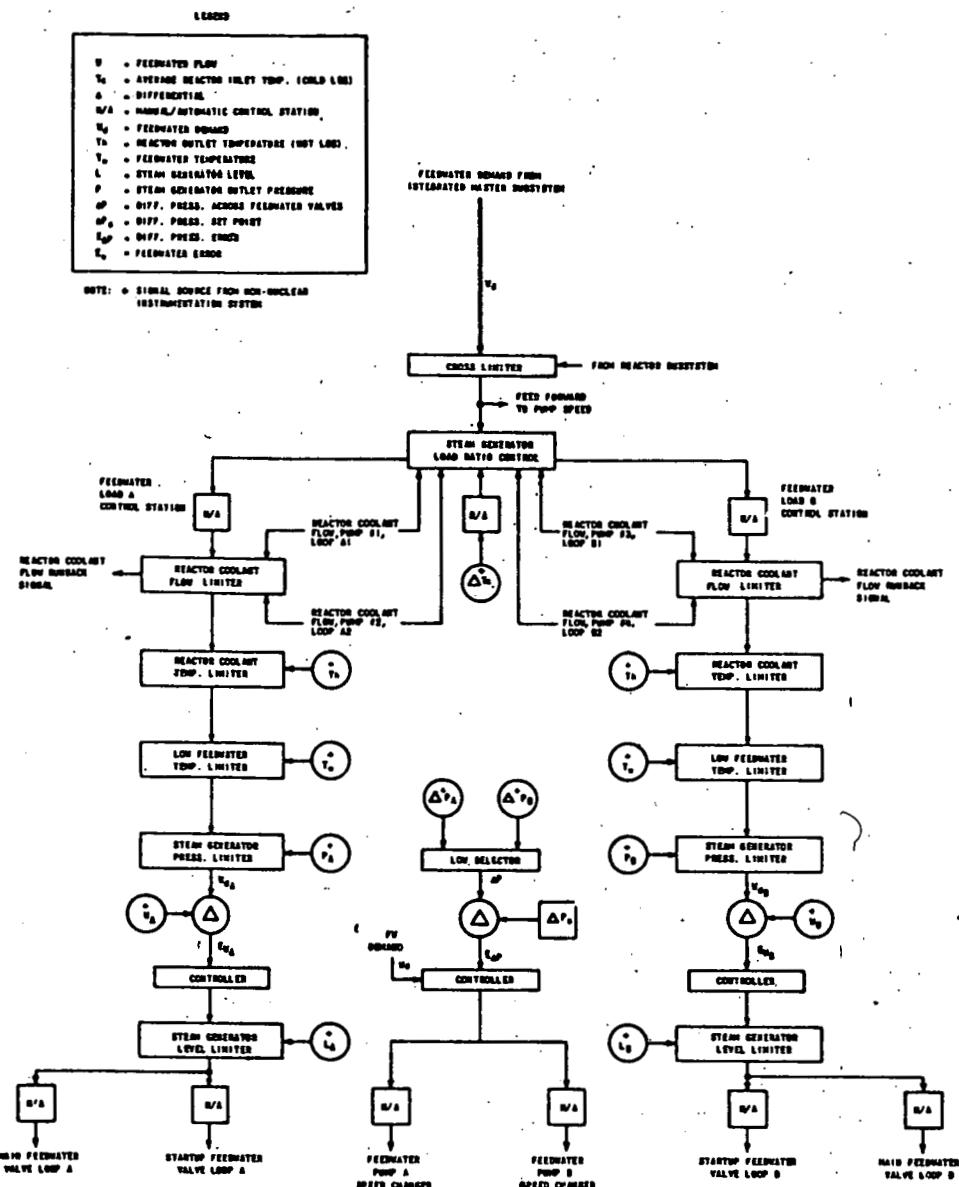


NOTES:

1. TURBINE CONTROL IS FURNISHED WITH THE TURBINE EQUIPMENT
2. ADS IS AUTOMATIC DISPATCH SYSTEM

Bellefonte Nuclear Plant
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INTEGRATED CONTROL SYSTEM



**Bellefonte Nuclear Plant
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**STEAM GENERATOR
CONTROL ICS**

- a. Megawatt demand converted to feedwater demand
- b. Total feedwater flow demand split into feedwater flow demand for each steam generator
- c. Feedwater demand compared to feedwater flow for each steam generator. The resultant error signals position the feedwater flow controls to match feedwater flow to feedwater demand for each steam generator.

The conclusion from a review of the data is that most steam generator level control systems are significantly less than 100% reliable, and the more sophisticated and complex these systems, the more difficulties appear.

5.5 MSIV Malfunction

Closure of Main Steam Isolation Valves (MSIVs) in BWRs have caused a relatively high percentage of the reactor scrams. Several of these were caused by loss of instrument air to the solenoid operated valves (pilot valves). As an example, Quad Cities 1 had two such failures in 1973: one in which the pilot valve had an air leak and the other involving a broken air line. Air leaks have been traced to foreign material in the air. These small particles cause scoring of the valve seals resulting in leakage. Failure of air lines have been found to be due to corrosion. Filtration and adequate maintenance of instrument air usually will protect against foreign material. The change to pilot valves with improved seal design would generally accommodate some foreign particles; however, in the case of some of these solenoid-operated valves, slivers of metal from manufacturing were found to have caused leakage. Adequate inspection prior to installation would catch these potential leakers.

Another cause of MSIV-related outages in BWRs is jamming of the pilot valves during surveillance testing. For example, on Hatch 1 (2/14/75) the test shop was performing a test that called for 10% closure of each MSIV. One MSIV had closed and would not open subsequently. Two other MSIVs drifted closed causing a scram. The cause of the first MSIV failure to open was a jammed solenoid operated valve (the drift was not explained). In another instance, a forced outage occurred at Millstone 1 (11/15/74) when during a test of the MSIVs, one failed to close. Foreign material was found on the air slide valve (solenoid operated valve). The underlying causes of these outages are essentially the same as those associated with instrument air leakages: foreign materials. The long-term corrective action should include: cleaning up the air, changing the pilot valves (to those with viton seals), and more careful inspection of these pilot valves prior to installation.

5.6 Human Errors or Faulty Procedures

As discussed in Section 3 and 4, human errors and faulty procedures together account for a significant fraction of the reported scrams. Some examples of these human errors are included here for reference:

<u>Plant</u>	<u>Date</u>	<u>Description of Event Leading to Scram</u>
Yankee Rowe	3/77	Preventive maintenance work on vital bus motor generator set resulted in low voltage on bus and false scram signal on pressure low level.
San Onofre	5/77	Trip channel had not been reset prior to testing another channel during routine testing of power range nuclear instrumentation.
Oconee 1	12/77	Feedwater valve was closed due to personnel error.
Oconee 1	5/77	Turbine shaft oil pump was tripped by technician error.
Calvert Cliffs 1	6/77	While performing maintenance on #12 steam generator feed pumps, a drop in oil pressure caused a pump trip.
Brunswick 2	5/77	Mechanic error while cleaning EHC oil strainers caused low EHC oil pressure and reactor scram.
Duane Arnold	11/77	While valving in nitrogen supply to one MSIV caused low nitrogen pressure to another MSIV which began drifting shut causing a scram.
Hatch 1	10/77	Personnel failed to follow procedure correctly.
Hatch 1	5/77	Error in manufacturer's manual led to scram.
Monticello	6/78	Mode switch inadvertently placed in shutdown.
Point Beach 1	1/78	"B" loop MSIV was not completely latched after a test.
Calvert Cliffs	11/78	Operational error while bypassing the condensate filter system caused a low steam generator level.
Haddam Neck	9/76	No. 3 reactor coolant pump shutdown due to operator error.
Brunswick 2	2/76	During wiring change on distribution panel, cable disconnected from a vital breaker by mistake.

Brunswick 2	8/76	While cleaning the circulating water intake pumps, a plug blew out of the strainer causing a loss of lube water.
Hatch	8/76	While placing the reactor protection system bus on Alternate "A" in preparation for de-energizing 600 V bus 1C, operator placed transfer switch in wrong position.

The above errors are typical examples of the types of problems which arise in the interface between man and machine. Some observations were consistently made by the utilities.⁽¹⁴⁾ Among these were:

1. The rapid turnover of personnel, brought about in part by expansion in the fossils area and staff transfer from the fossil to nuclear areas, has resulted in understaffing fossil plants with less experienced personnel.
2. Turnover has resulted in the need for extensive training programs which are very costly. Employees show less interest now than in the past in voluntary on-the-job training of the basic skills. New employees are frequently completely inexperienced (recent high school graduates) and must be fully trained in minimum time.
3. The dedication of the newer employees to their jobs is believed much less than the previous generation of workers.
4. The shortage of experienced, skilled personnel results in excessive overtime for the experienced employees, with accompanying fatigue and, perhaps, an increase in the likelihood of error.
5. The operator's tasks and responsibilities have become more complex. The operator is frequently overloaded, even to the extent that he has too much to monitor.
6. In general, the hardware system is suffering from inadequate routine and post-maintenance inspections. Abnormalities are neither recognized at an early stage nor are they reported.
7. Vendors are not providing adequate operation and maintenance documentation with the equipment they deliver.
8. The competence of contract maintenance crews is highly variable.
9. The austere financial atmosphere requires extensive documentation of any problem and the proposed solution before funds can be expended. It has not been possible to document personnel errors sufficiently to date to justify major efforts to reduce them.

10. The possibility of a more effective personnel selection system was often mentioned, but because of the complications brought about by union agreements, government requirements, lack of highly justifiable and reliable selection tools, and the difficulty of obtaining dedicated funds for such a program, most utilities are pessimistic regarding its successful implementation.
11. Among most of the larger utilities, there is a growing recognition of the need to consider personnel effects on reliability. However, there appears to be little understanding of the required methodologies to identify and solve their personnel problems.

As noted in Section 4, there is approximately an exponential learning curve associated with these types of errors, characterized by the drop in human error rate leading to scrams over the initial two years of commercial operation. It should be evident from these "learning curve" trends that the problem can be substantially reduced if the operators receive additional preparation and training prior to operation of a nuclear power plant. Figure 5.3 is a summary of a typical operator training schedule. It is apparent from the training summary that only supervisors and engineers receive concentrated preparation or instruction in the plant operation. The operators, electricians, and mechanics -- the people who actually perform the operation, testing and maintenance -- receive less than 5 months of maintenance-only training, and this during the busiest time for the plant, just prior to an initial fuel load. In contrast to this, the U.S. Navy, which trains operators for nuclear power submarines, requires all personnel to attend 6 months of reactory theory school and 6 months of training on a prototype reactor. Then there is, in addition, a trial period on the submarine where the operator must qualify for operation of the submarine. Thorough operator training is essential to the minimization of plant transients.

The benefits to be gained from better operator preparation are not only in the reduction in transient initiators, but also in the improvement in operator response to high stress conditions. It is well recognized that operator error can be quite high under conditions of stress. However, a well-trained and informed operator will react in a more positive and knowledgeable fashion than an operator who does not thoroughly understand the system operation.

POSITION	MONTHS PRIOR TO INITIAL FUEL LOADING												PRE-OPERATION EVALUATION	REVIEW BY TIME FUEL LOADING	MONTHS FOLLOWING													
	PREPARATORY TRAINING						PRE-SITE TRAINING			ON-SITE WORK TRAINING																		
26	25	24	23	22	21	20	19	18	17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1	1	2	3
PLANT SUPERINTENDENT																												
ASSISTANT SUPERINTENDENT																												
TECHNICAL SUPERVISOR																												
NUCLEAR ENGINEER																												
TECHNICAL ASSISTANTS																												
HEALTH PHYSICIST																												
RADIATION MONITORS																												
CHEMICAL ENGINEER																												
LABORATORY TECHNICIANS																												
ELECTRICAL ENGINEER																												
INSTRUMENT ENGINEER																												
OPERATIONS SUPERVISOR																												
SHIFT SUPERVISORS																												
SENIOR CONTROL OPERATORS																												
CONTROL OPERATORS																												
AUXILIARY OPERATORS																												
EQUIPMENT ATTENDANTS																												
MAINTENANCE SUPERVISOR																												
MECHANICAL FOREMAN																												
MECHANICS																												
ASSISTANTS																												
WELDER																												
ELECTRICAL FOREMAN																												
ELECTRICIANS																												
ASSISTANTS																												
INSTRUMENT FOREMAN																												
TECHNICIANS																												
ASSISTANTS																												

AEC REVIEW OF TRAINING PROGRAM

1. BASIC NUCLEAR COURSE
2. BASIC RADIATION PROTECTION
3. FAMILIARIZATION - OPERATING REACTOR
4. TECHNOLOGY COURSE
5. OPERATOR TRAINING
6. FUEL MANAGEMENT COURSE
7. RADIATION PROTECTION TRAINING
8. RADIOCHEMISTRY TRAINING
9. CONTROL & INSTRUMENTATION TRAINING
10. MAINTENANCE TRAINING - ONSITE DURING CONSTRUCTION, ERECTION AND INSTALLATION OF EQUIPMENT
11. ON-THE-JOB
12. PREPARATION OF MAINTENANCE MANUALS AND OPERATING PROCEDURES
13. RADIATION PROTECTION INDOCTRINATIONS BY STATION HEALTH PHYSICIST
14. PREOPERATIONAL TESTING OF EQUIPMENT
15. CHECKOUT OF OPERATING PROCEDURES
16. INITIAL FUEL LOADING & STARTUP TESTING

TYPICAL TRAINING SCHEDULE

Figure 5.3

It is recommended that operators and maintenance personnel all undergo a training program, including basic reactor system theory. This program would be best administered by a central industry school. In addition, actual on-the-job training or simulator training is essential.

5.7 Reduction in Frequency of Recurring Transients

Examination of the scram data base shows that in many plants the same component is responsible for several scrams which occur within a short period of time. As an example, consider Table 5.1 which presents the causes and dates for selected scrams at the Palisades plant. Table 5.2 shows the dates for shutdowns at Browns Ferry 1 and Browns Ferry 2 due to oil leaks in the electro-hydraulic system. Although the guilty component may be different, a pattern of several scrams occurring in a short time due to the same cause appears at many other units. Thus, it is suggested that following a scram, a thorough investigation for the root cause be undertaken and the problem corrected before the plant is brought back to full power.

Other plants with unique problems include H. B. Robinson. For example, the instrument bus at H. B. Robinson has led to a significantly higher scram incidence than any other plant in our sample. This indicates that the solution to this problem has a high benefit for H. B. Robinson, but may not be of interest to other plants. Similarly, the available data for TMI-2 and Davis Besse indicate an unusually high rate of problems with feed-water operation and control. Again, a significant reduction in the individual plant risk may be obtained by focusing efforts on the improvement of the ICS for these plant types.

5.8 Common Cause Problems

In this subsection, consider two scenarios for events, which, while rare*, do occur and can have profound effects on plant safety. The first event is a loss of DC power; the second deals with a fire.

*Since these events are rare, they do not show up in the analyses included here which focuses primarily on high frequency events.

Table 5.1

EXAMPLES OF SCRAMS AT THE PALISADES PLANT
WHICH OCCUR FOR THE SAME REASON

(1) Low Steam Generator Level Trip Due to Unstable Feedwater Control System			
1/11/1972	1/12/1972	7/6/1972	
(2) CRDM Seal Leakage			
3/15/1972	3/31/1972	6/26/1972	8/31/1972
10/23/1972	6/30/1975	7/29/1975	8/12/1975
8/30/1975	9/6/1975	11/26/1976	7/31/1978
(3) Feedwater Pump Trip			
5/10/1976	1/17/1977	3/25/1977	3/27/1977
8/7/1978	10/17/1978	12/16/1978	
(4) Low Steam Generator Water Level			
5/11/1978	5/20/1978	6/7/1978	6/8/1978
6/13/1978			6/13/1978

NOTE: This is a summary of persistent problems occurring at one plant. There are bars connecting events which cause a reactor scram within 20 days.

Table 5.2

DATES OF SCRAM DUE TO ELECTRO-HYDRAULIC
CONTROL PROBLEMS IN BROWNS FERRY 1 AND 2

Browns Ferry 1		Browns Ferry 2	
6/2/74	Oil Leak	9/18/74	Controls
9/30/76	Pressure Regulator Startup Test	9/14/76	Pressure Regulator Startup Test
10/13/76	Malfunction	12/10/76	Oil Leak
1/20/77	Oil Leak	6/11/77	Oil Leak
1/27/77	Malfunction		
2/12/77	Oil Leak		
2/25/77	Oil Leak		
6/6/77	Malfunction		

NOTE: This is a summary of persistent problems occurring at one plant.
There are bars connecting events which cause a reactor scram within
20 days.

1. Loss of DC Power: While a reactor was operating at 85% power (615 MWe), a 50 hp DC emergency oil pump, which is powered by one of two sets of battery banks was started for a routine weekly, 2-hour test run. The pump was inadvertently not stopped as planned and 4.4 hours later, the station battery was so depleted that the reactor tripped on low DC voltage to the trip coils of the trip breakers. Turbine trip occurred, and the stop valves closed, but the emergency breakers (#1 and 2) could not close because of low DC voltage. Breakers #3 and 4 closed as designed. Diesel Generator A started, but the E-1 emergency bus could not close either because of low DC power. The loss of startup transformer buses 1 and 2 caused loss of the AC turbine oil pump. The emergency DC oil pump was already inoperable because of the depleted battery. As the turbine slowed down, the shaft oil pump became ineffective; thus, lubrication failed. The turbine stopped in 17 minutes with bearing seizure instead of the usual 1.25 hours. The plant computer failed and there were many erroneous visual indications. One charging pump was started immediately to re-establish seal and charging-water flow. The failure of instrument power actuated the safety injection system. About 3 minutes after the turbine stopped, the battery bus tie between Battery A and B was closed manually, thus supplying both buses and allowing the breaker between the diesel generator and the E-1 bus to close. The operator proceeded to close the startup transformer breakers and restore normal operation, but the turbine bearings had seized.

a. Perspective on the Loss of DC Power Event

The only damage as a result of this event was to the turbine bearings, with resulting plant outage for repair. Although the initiating event that caused loss of DC bus was not anticipated, the loss of a DC bus had been anticipated in the plant design since necessary equipment would go into failsafe configurations in the eventuality of its occurrence. The initiator for the event was human error, but correct operator performance was important in mitigating the event. If the DC battery tie could have been closed a few minutes sooner, even the damage to the turbine bearings might have been avoided.

To assure that this event could not recur: (1) operating procedures and training were changed to insure against longer-than-planned tests; (2) low voltage alarms were installed on the DC bus. Notifications were sent throughout the industry.

b. Conclusions to the Loss of DC Power Event

(1) The initiator for this DC power failure was human error combined with a design deficiency, but correct operator performance contributed greatly to mitigating the event.

- (2) Steps were taken to prevent the recurrence at the original plant and throughout the industry.
- (3) A common cause failure analysis of the DC system would have uncovered this event before it occurred.
- (4) Because of redundancy, only a partial failure occurred as a result of the failure of this DC bus.
- (5) The failure was detected and corrected by plant personnel.

2. Fire: A fire occurred at a three-unit nuclear power plant during construction of the third unit and during operation of Units 1 and 2. The ignition source was a candle used to test for air leaks in the electrical cable penetrations between the cable spreading room and the reactor buildings. The candle flame was drawn into the penetration and ignited the polyurethane foam cable-sealant. This fire fed by the draft caused by the lower pressure in the reactor building ignited the wiring insulation, which burned horizontally and vertically for several hours. It burned cables in all ten trays within the penetration, damaging 2000 cables. Because of the fire, normally used shutdown cooling components for Unit 1 were inoperable for several hours. Other installed equipment was used to provide the shutdown cooling of Unit 1; no significant problems were encountered with the shutdown cooling of Unit 2.

a. Perspectives on the Fire

One important system that functioned during the fire was the reactor protection system for both reactors. The reactors were shut down soon after the fire became known. Even if this were not the case and had the fire progressed to the shutdown control wires, this would have caused reactor shutdown because of the failsafe design of the reactor protection system.

In Unit 1, problems arose providing shutdown heat removal following the fire, because the sources of power for the equipment normally used were not available, so other installed equipment was used. Unit 2 had no such difficulties. If this installed equipment had failed in removing the residual heat, at least three other systems (Control Rod Drive pump on Unit 2, Backup Control Rod Drive pump, Standby Liquid Control pumps, and RCIC with auxilliary plant boiler steam supply) were in readiness as backup and other means could have been utilized if the functioning and planned backups had failed.

The time for the accident to develop was not instantaneous; it took place over many hours, during which time-emergency

procedures and contingency planning were instituted. The time span was significant in controlling and ameliorating the accident.

b. Conclusions Derived from the Study of This Fire as a Common Cause Failure

- (1) As a result of this fire and as a result of changes in regulations, procedures and general awareness, the probability that an accident of this type will recur is low.
- (2) The accident sequence was not instantaneous but extended over sufficient time to mobilize emergency resources more than adequate to mitigate the situation.
- (3) The failsafe design of the reactor protection system would have shutdown the reactors even if control of the scram system had been lost.
- (4) While a human error initiated the accident, human performance and the defense-in-depth design were instrumental in mitigating the accident.

The design fix associated with these types of events is one requiring an extensive plant specific analysis of the as-built plant to ensure that there are methods of precluding single events leading to a cascade of failures which could lead to core melt. Common cause failures are a potential concern in the industry if there is no overview analysis method used by the utility or designer to cross-check his plant to ensure there is not a coupling of failures together.

5.9 Flow-Induced Vibration/Loose Parts Monitoring

There are, in addition to those scram causes which show up in the operating experience, some problems which may arise in the future and lead to additional, spurious plant transients. Currently, NRC is considering implementation of instrumentation associated with a loose parts monitoring system. This would be a system (possibly a set of accelerometers) attached to a component (e.g., steam generator) which would detect deviations from a baseline trace. From these deviations it would be inferred that there may be loose parts in the primary system. There are two different approaches to this problem:

1. The loose parts monitoring system itself must be made more reliable so that signals from the system are meaningful and do not result in spurious plant scrams or shutdowns. This will require being able to discriminate against other background noise. This appears to be an area requiring extensive laboratory and hardware development.
2. Secondly, the primary and secondary loops of nuclear power plants are subjected to very high flow rates. These high flow rates may have adverse effects on core components, instrumentation, or other items inside these circuits. A method of calculating the onset of flow-induced vibration and a workable set of criteria to preclude flow-induced vibration are important aspects of the problem which requires increased attention to avoid future problems like those encountered in BWR local power range monitors and B&W reactor internals.

5.10 Recommended Actions Which Can be Applied Generally by All Utilities

The following actions are recommended for reactor operators or groups of operators of similar plants:

1. Package I&C equipment in a manner to:
 - (a) Preclude accidental jarring of the equipment
 - (b) Minimize the impact of adverse environmental conditions.
2. Establish greater redundancy of scram circuitry.
3. Continuously monitor setpoints of all instrumentation with a minicomputer and print out a warning of any deviation.
4. Upgrade all electrical breakers.
5. Improve lightning arrestors.
6. Simplify feedwater control systems.
7. Perform a root cause analysis on each scram which occurs at each plant to determine the major and significant causes of scram and to definitively class the human element regarding scram.
8. Provide more operating training for off-normal conditions with perhaps extra personnel during startup and shutdown. These two fixes reduce the number of scrams due to steam generator level trips and feedwater flow-to-steam flow mismatch.
9. Limit on-line testing to less than that currently scheduled. Fewer on-line tests would generally mean fewer scrams. Alternatively, add sufficient redundancy in instrumentation to maintain 2-of-3 or 2-of-4 logic during testing.

10. Maintenance errors, miscalibrations, and errors during testing can be minimized by extensive checking of procedures and work performed by management as well as other workers. This may require practicing the procedure on a simulator, mockup, or classroom run-through.
11. Reduce the incidence of repeated scrams from single root cause. Repeated scrams have indicated several problems that each need corrective action: improper recording of scrams, insufficient examination of scram causes, improper maintenance and repair, improper procedures and management failure.

The corrective action to preclude these repeated scrams should include:

- Standardized recording procedures and proper utilization of a recognized standard scram reporting sheet.
- Standardized scram investigation to assure root cause determination such that proper corrective action can be employed.
- Raise standards for maintenance of equipment. Particularly, have management in a new plant inform all maintenance personnel of the importance of the care that is necessary to assure against spurious scram due to bumping equipment. Several scrams of this nature have occurred and they usually happen in plant infancy. Moreover, proper management procedures concerning immediate repair and restoration of equipment is essential. At no time should a reactor be without sufficient management to assure proper corrective action.
- Assuring proper procedures must start with an investigation of plant management with effective action to eliminate repeated failures.

12. Establish better criteria for management performance. A failure modes and effects analysis (FMEA) of the management would be a useful tool in determining management failure causes and proposing effective corrective action. Rather than focusing on success or failure of one person whenever a series of outages has created a low plant availability, the FMEA could selectively find the management problems most likely to have correlation with the outages. However, in order for the FMEA to be useful, the investigation must include the follow-up implementation of necessary corrective actions.

In the case of repeated failures: generally, the root cause has not been found; management has not followed up on investigation until after several scrams; the

operator writing up the report has been on his own with a rubber stamp backup concerning review; corrective action has been minimal and, in some cases, totally disregarded. To eliminate these failures: implement proper management procedures to assure root cause determination immediately, followup reporting must include active management participation at the operation, maintenance, and overall plant level.

6.0 CONCLUSIONS

The Reactor Safety Study⁽¹⁾ (WASH-1400) and subsequent studies^(2,3,4) have shown that a substantial portion of the public risk from the operation of nuclear power plants is associated with the potential inability to cool the core following plant transients that require rapid reactor shutdown. Therefore a decrease in the number of these transients would reduce the total calculated public risk.

This report evaluates a large sample of the accumulated reactor operating experience to determine the components which lead to reactor scrams. A sample population of 35 light water reactors (LWRs) is used to characterize the general population over a period of 15 years. A simple ranking scheme is used to display those components which are the most frequent contributors to reactor scrams for PWRs and BWRs (see Figures 6.1 and 6.2).

In addition to the ranking scheme used to highlight the principal component causes of scram, an evaluation of the time dependence of these component contributors is also presented. Figure 6.3 gives the results of the variation of scrams in PWRs as a function of plant age.

The principal conclusions which can be reached from the detailed analysis of the scram data are:

- (1) The chief component type leading to reactor scrams can be identified as instrumentation and control. (Note: all instrumentation and control items have been summed together.) However, in the I&C category there is a strong correlation between human error, faulty maintenance procedures, and testing errors leading to scrams.

Figure 6.1. Distribution of Top Twenty Scram-Related Components in PWRs

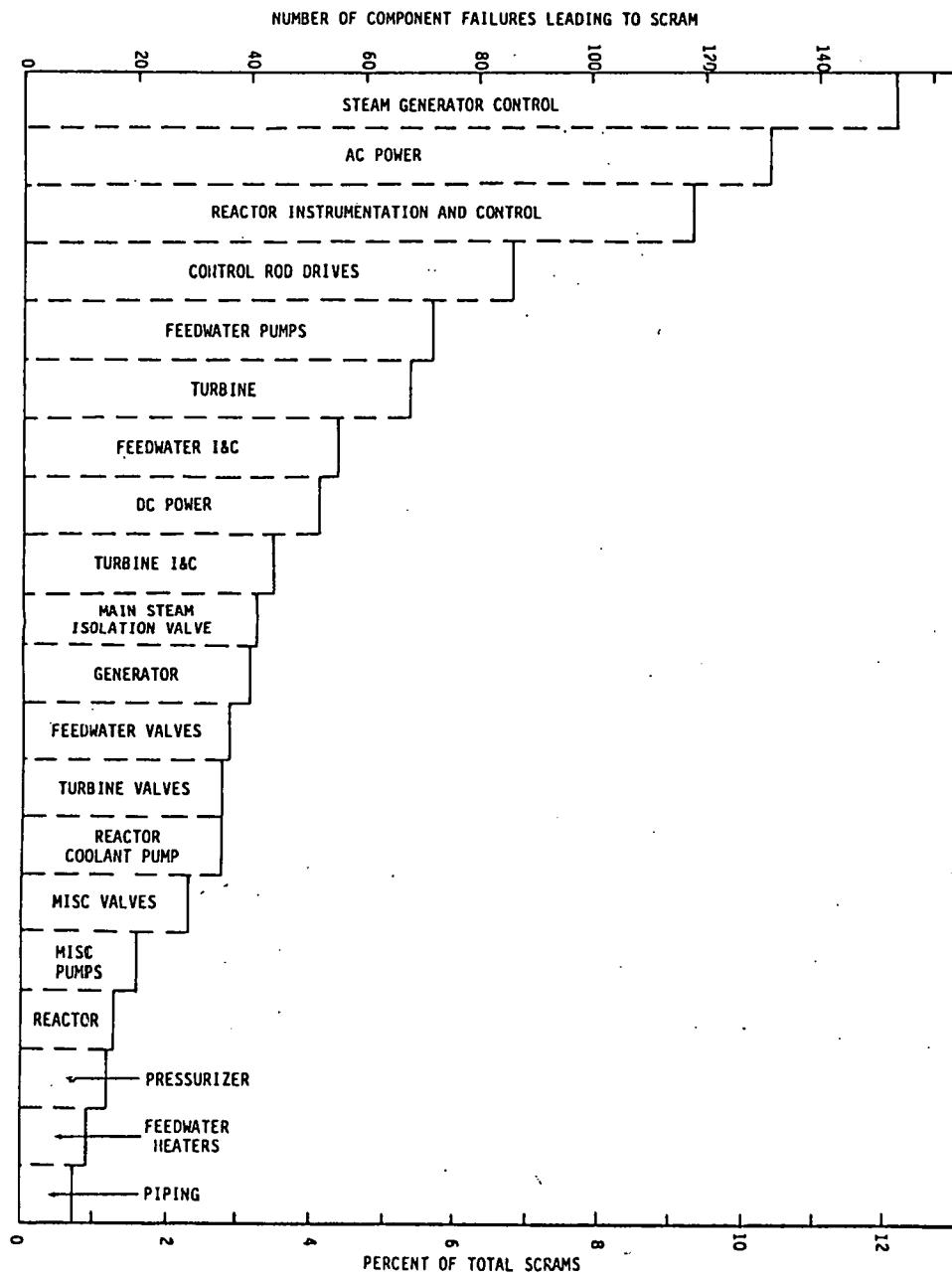
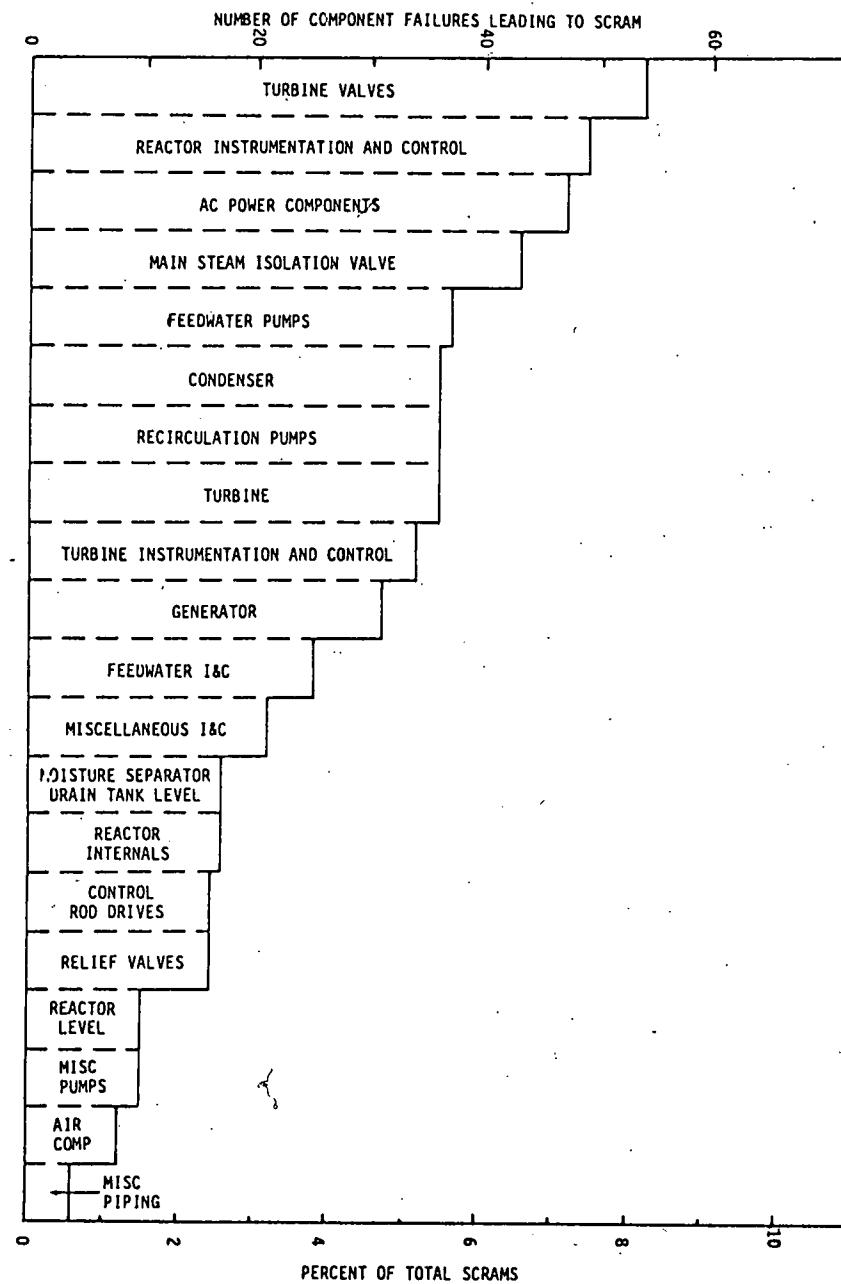


Figure 6.2. Distribution of Top Twenty Scram-Related Components in BWRs



SUMMARY FOR PWR PLANTS

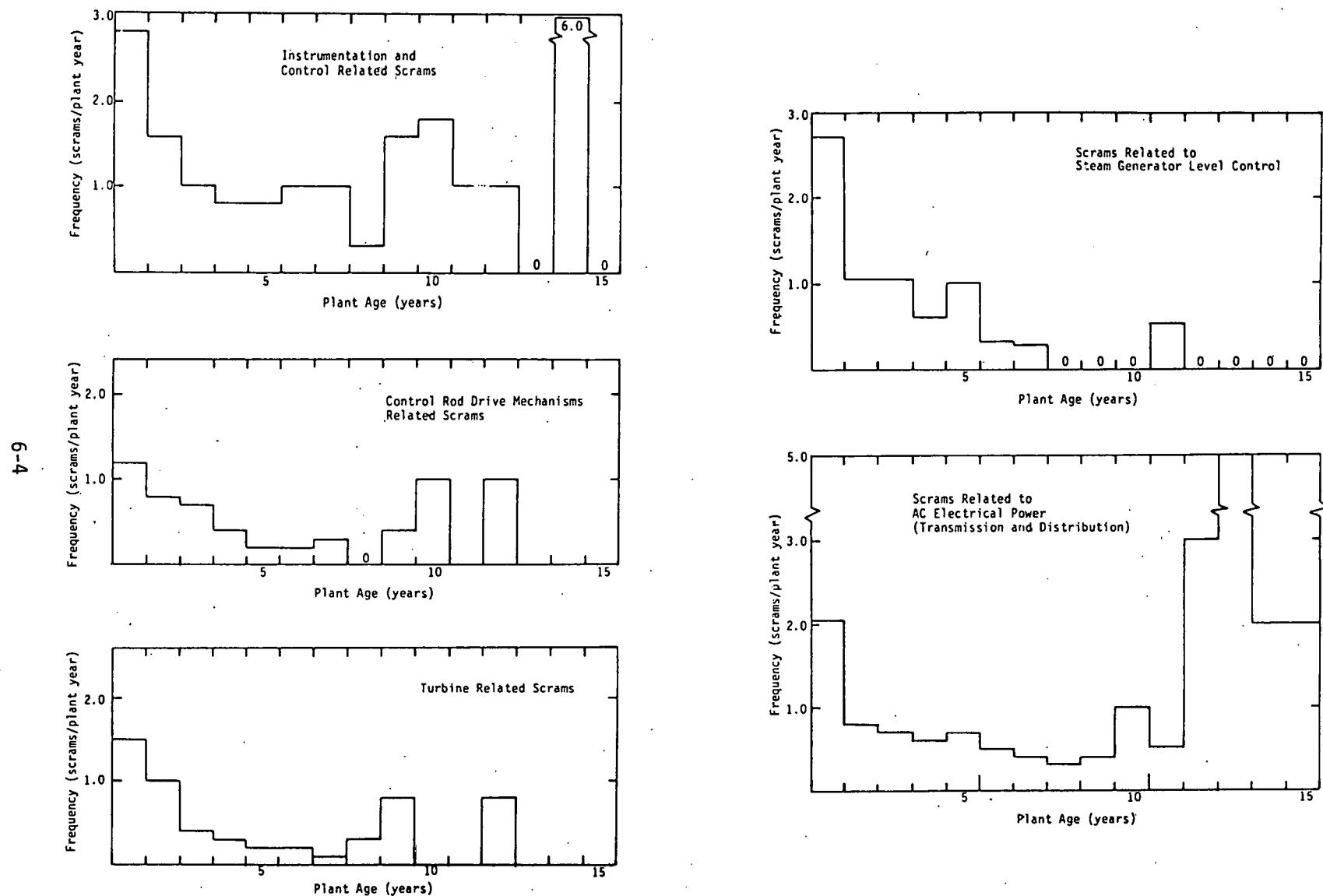


Figure 6.3. Trend of Scrams Related to Components in Specific Systems as a Function of Plant Age for PWRs

SUMMARY FOR PWR PLANTS

G-9

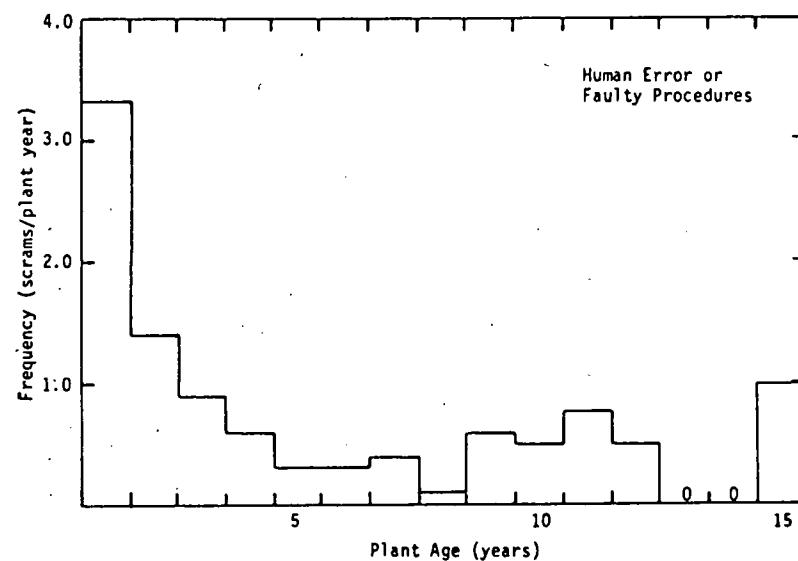
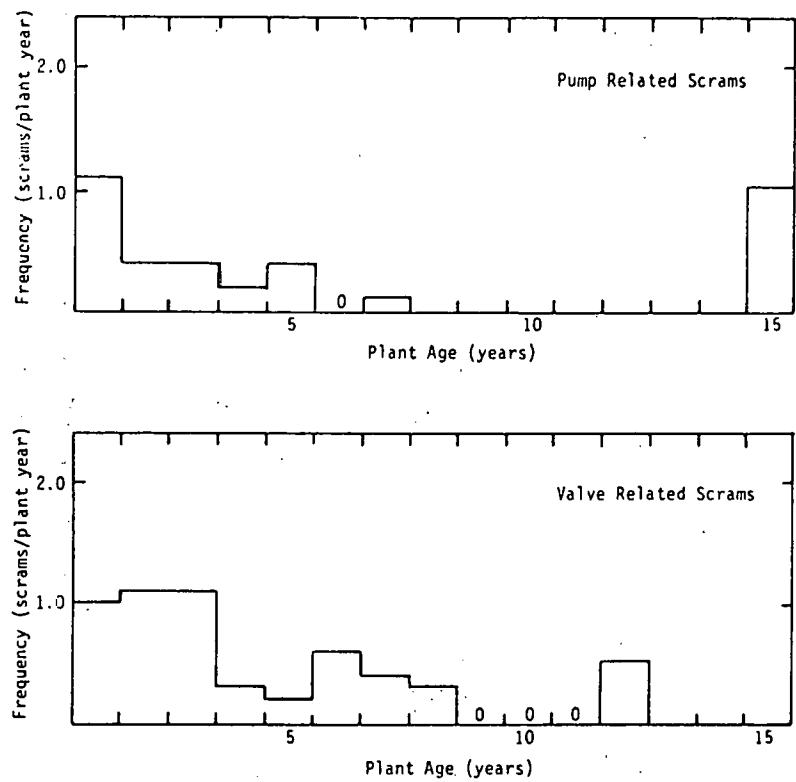


Figure 6.3 (Cont'd)

- (2) Electrical transmission and distribution related incidents have a nearly equal effect on the frequency of scrams in nuclear plants investigated over their entire life, not just during initial startup. These components rank second to I&C as the leading cause of scrams in PWRs (~15%), while they are fifth in BWRs (~7%).
- (3) Turbine control incidents which are related to scrams represent a significant fraction of the scrams in BWRs (~ 6 %). In PWRs there are fewer turbine related scrams than in BWRs.
- (4) The number of components in a nuclear power plant required for proper plant operation is so large (i.e., high parts count) that while the reliability of each component can be quite high, the aggregate reliability of the "system" of combined components may be significantly lower. Simply stated:

The overall system unreliability is a combination of unreliabilities from a wide and diverse set of components.

- (5) There is a characteristic overall trend of scram transient frequency as a function of plant age. The trend fits an exponential function very closely. A similar phenomena is also exhibited if individual component related scrams are plotted as a function of age. This exponential function indicates that the initial two years of plant operation are atypical from the points of view which make this period of time a high risk period:

- (a) There are a large number of scram occurrences during the initial reactor operation.

(b) The operating personnel may not be completely familiar with the subtle operating peculiarities of the plant, and may in addition be unfamiliar with procedures to cope with unusual accident scenarios which may develop.

(6) Plant specific data compared with "average" plant data is extremely useful in highlighting particular problem areas in a given plant where substantial reductions in scram frequency could be obtained if proper design or procedure changes were implemented. This conclusion is emphasized by a detailed example for B&W reactors and their integrated control system.

(7) Because of the high incidence of scrams related to human errors or faulty procedures, it appears fruitful for each plant to focus increased attention on the following areas:

- (a) Personnel training including:
 - Simulator training
 - Care in handling sensitive electronic equipment
 - Instruction on keeping the control room informed of all maintenance
- (b) Procedural review
- (c) Administration
- (d) Maintenance methods.

A caution which must be mentioned in the analysis of the scram data is the question of extrapolation to end of life trends. Since we are dealing with a very young population of plants (the current data can only support conclusions

over the initial 7 to 10 years of plant operation), extrapolation of the data beyond 10 years is nearly impossible. Therefore great care must be exercised in the application of the data to end of life operation or long-term operation of nuclear power plants. Another potential application of the data is to future plants. Again the population sample we have used is one of custom designed plants of an older vintage than that typical of future designed plants and, therefore again, caution must be exercised so that potentially new problems are not overlooked.

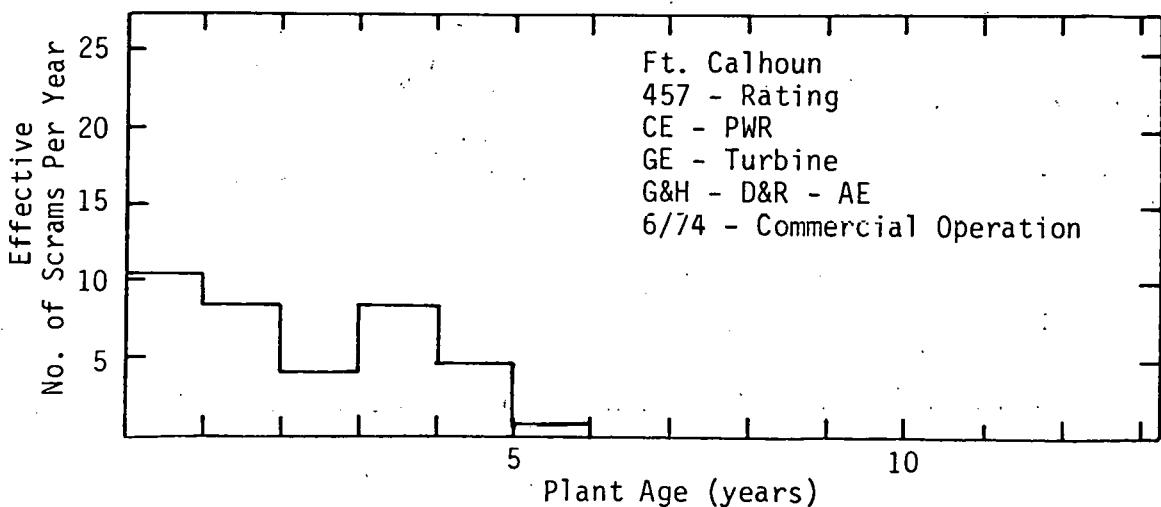
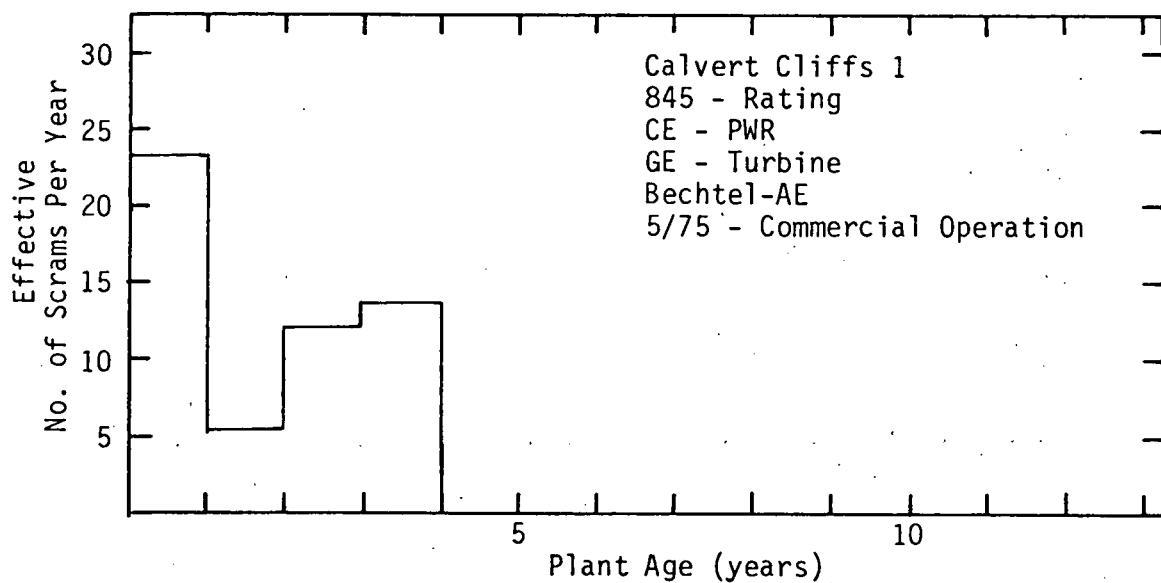
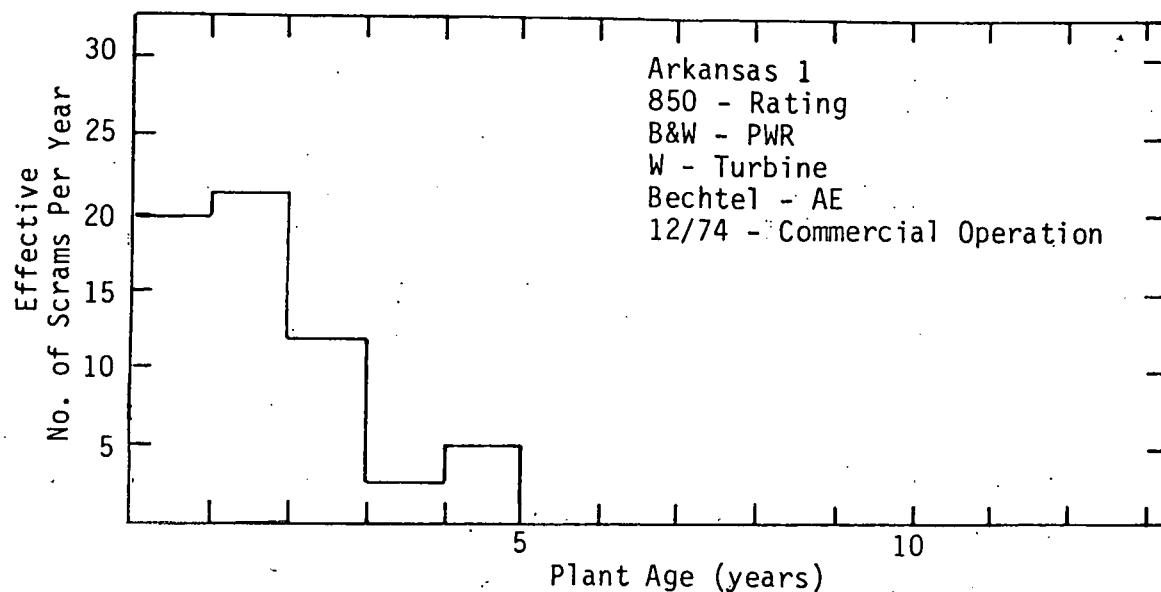
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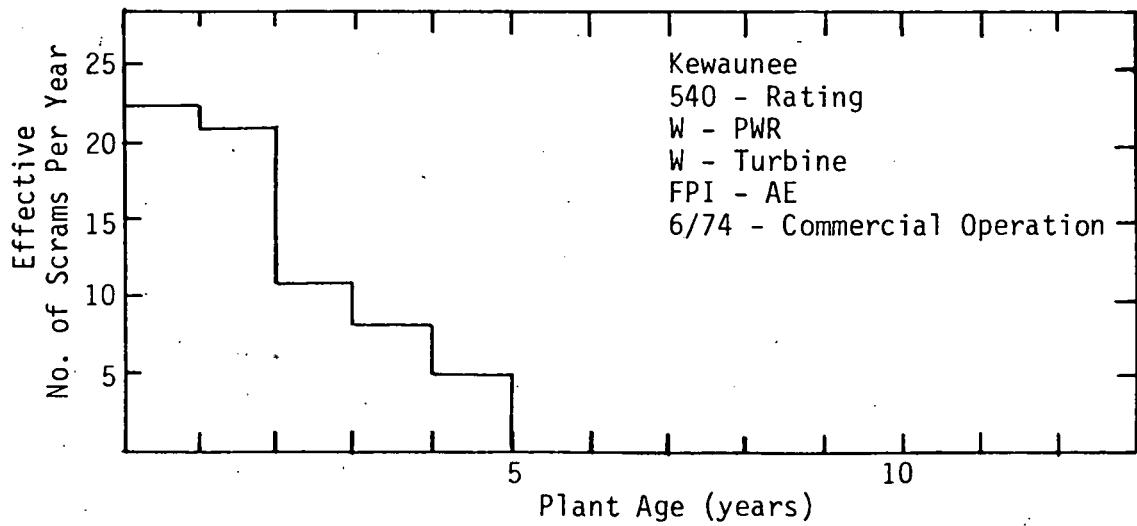
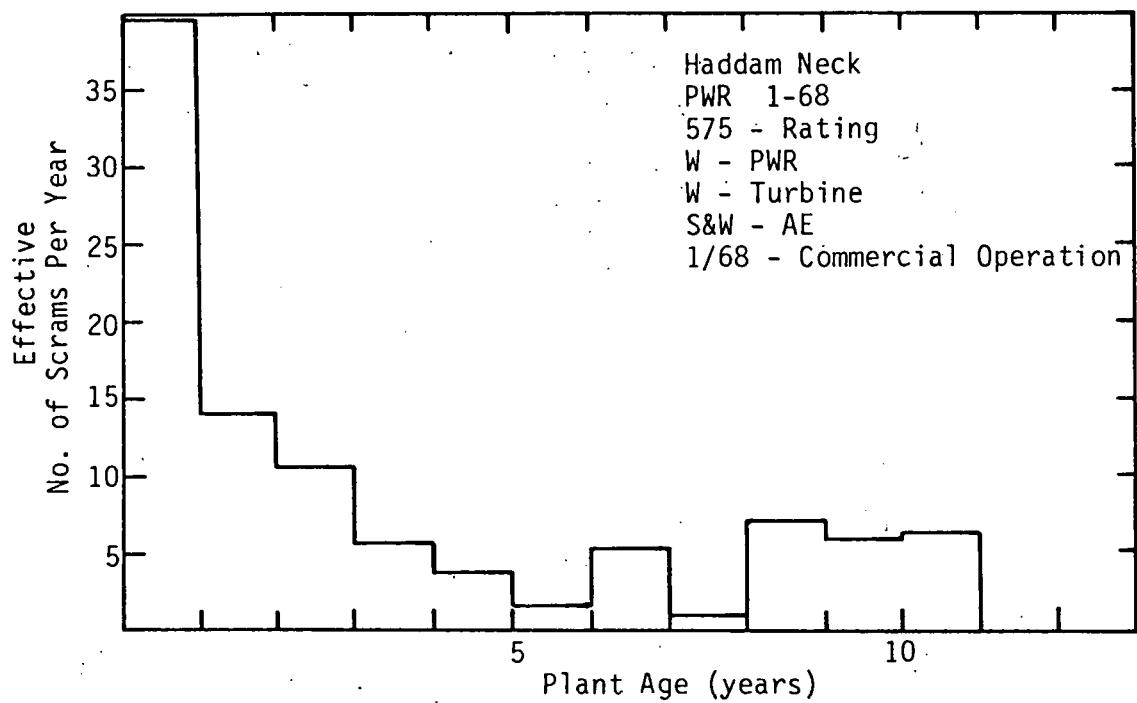
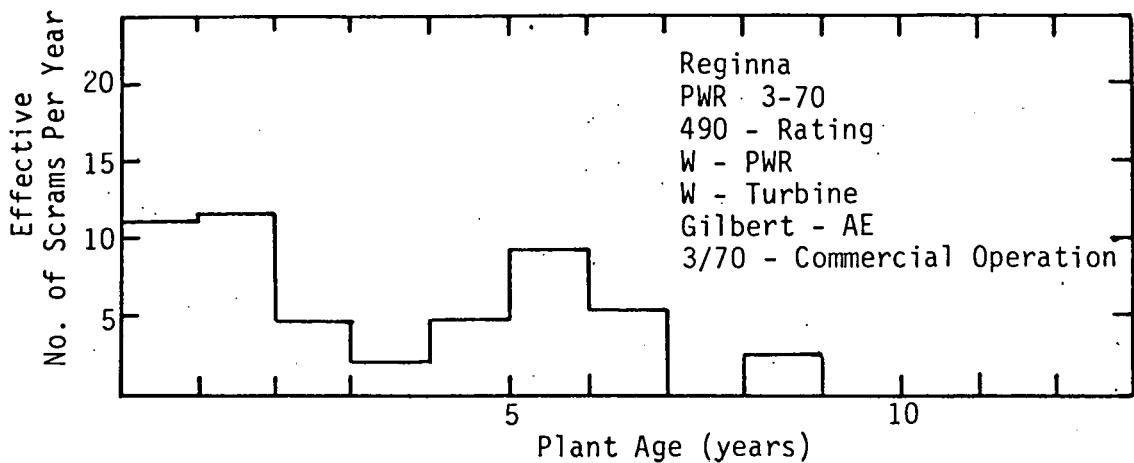
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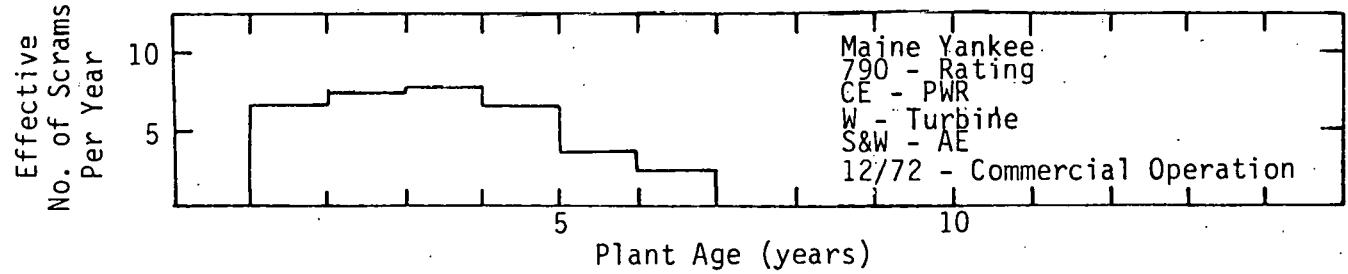
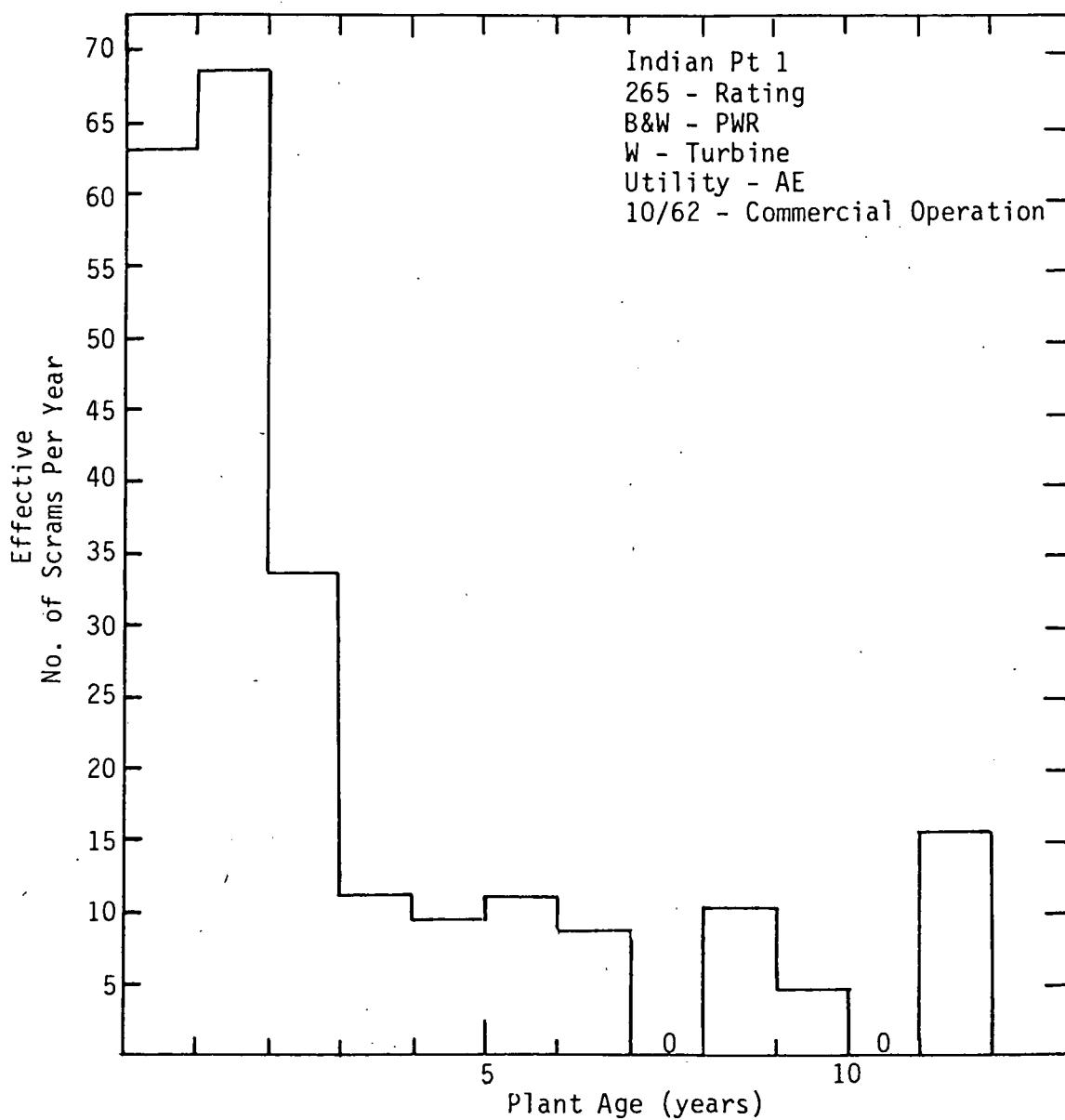
APPENDIX A

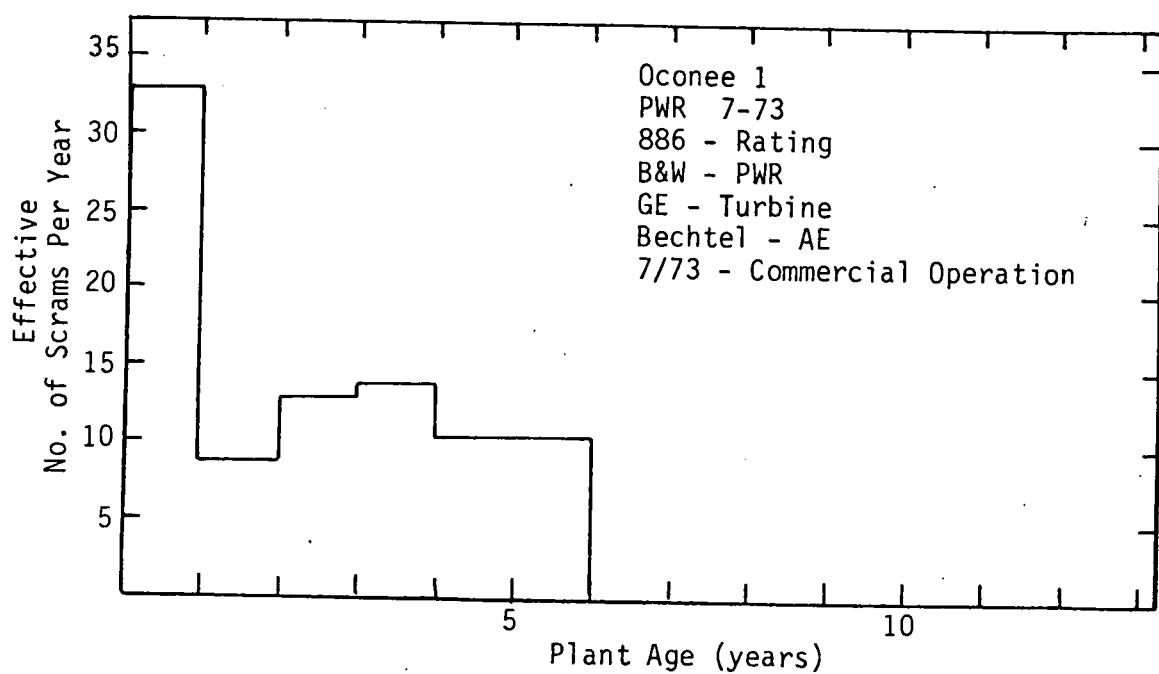
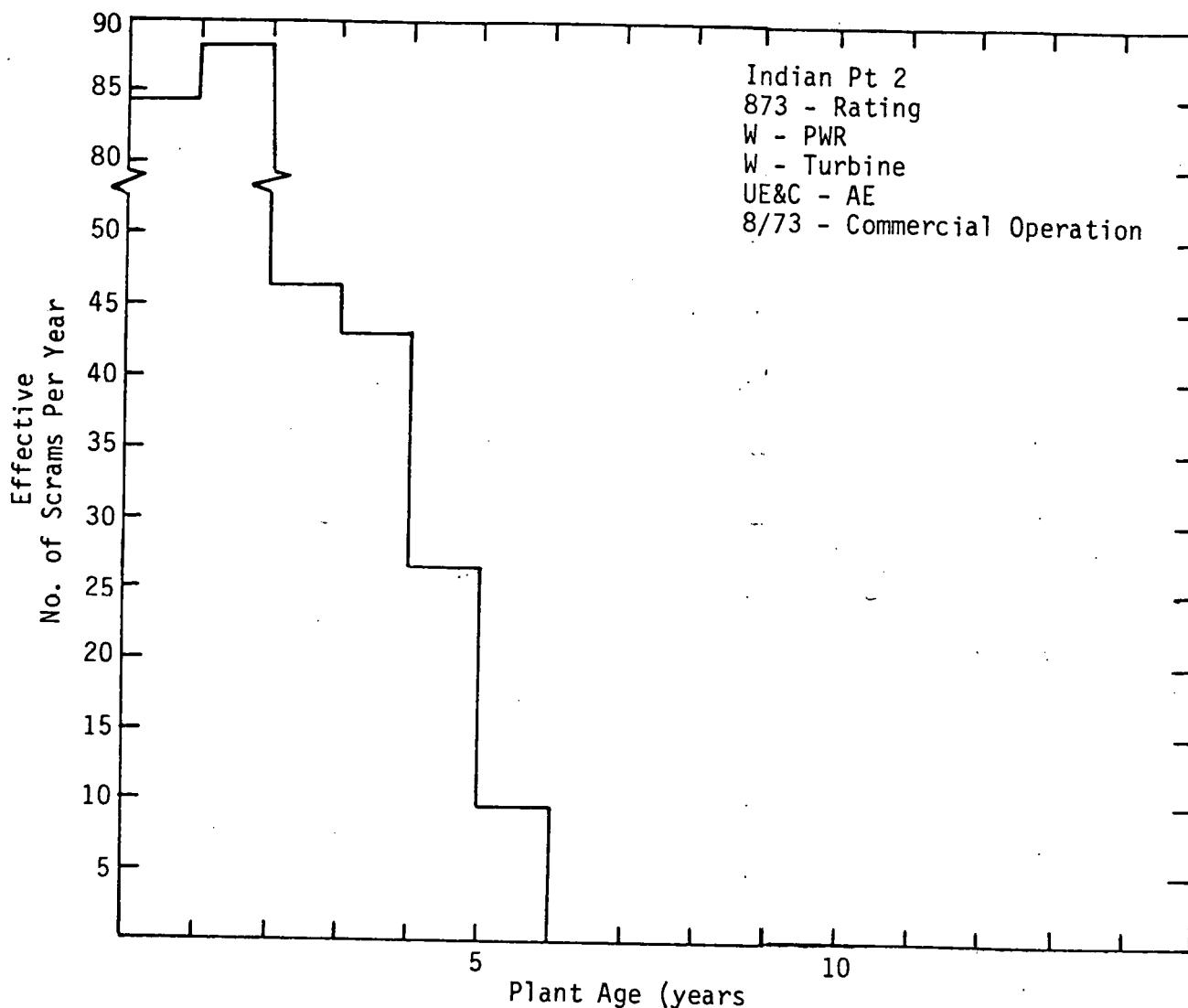
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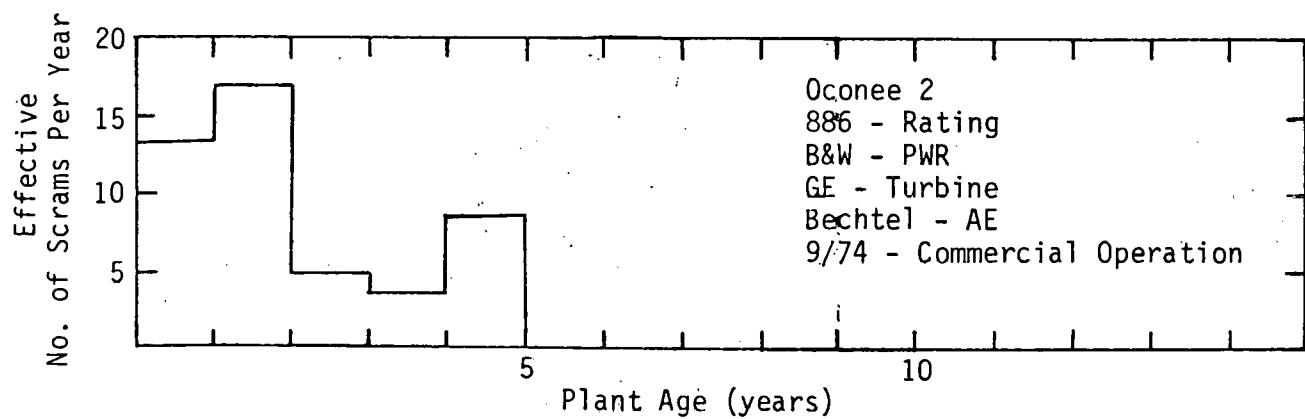
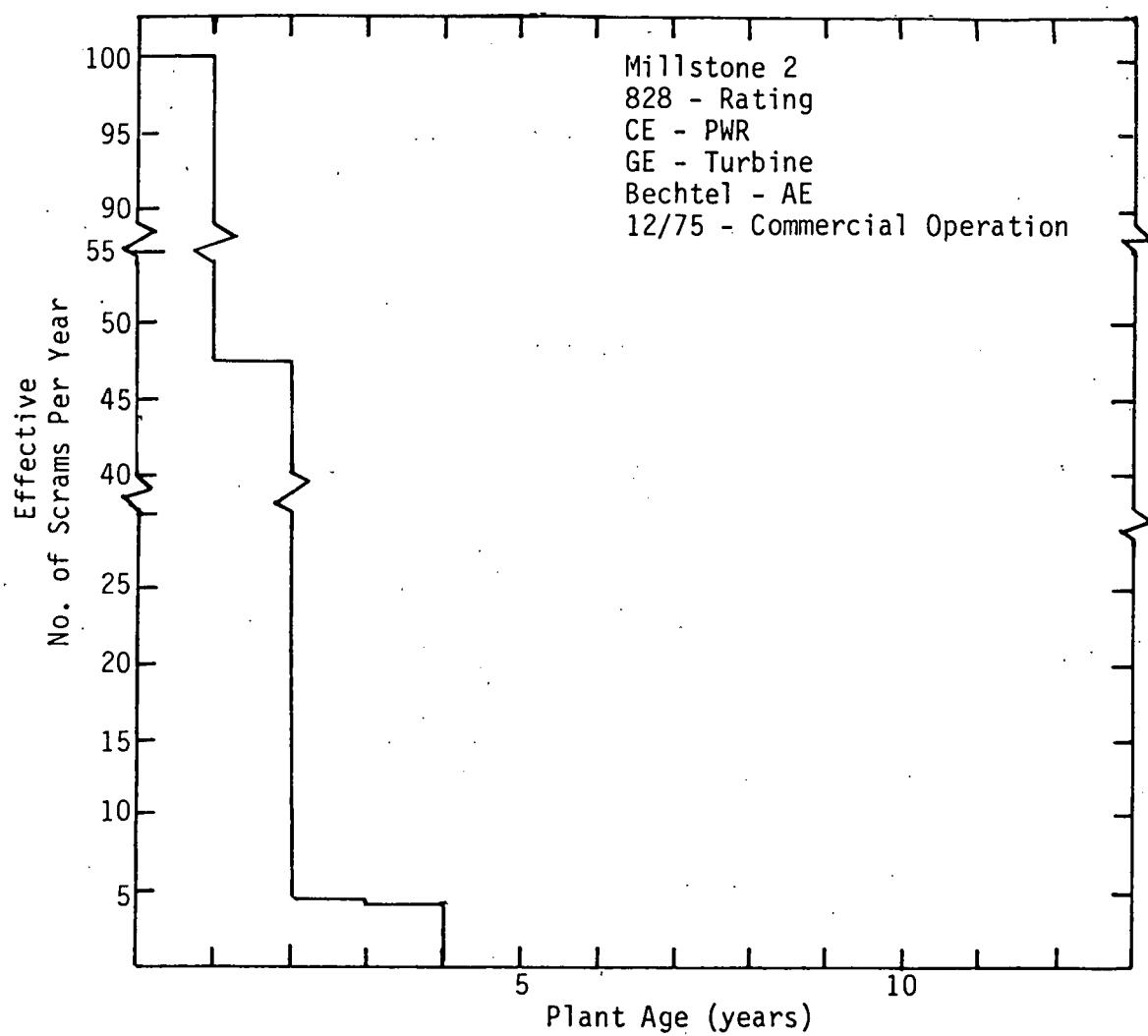
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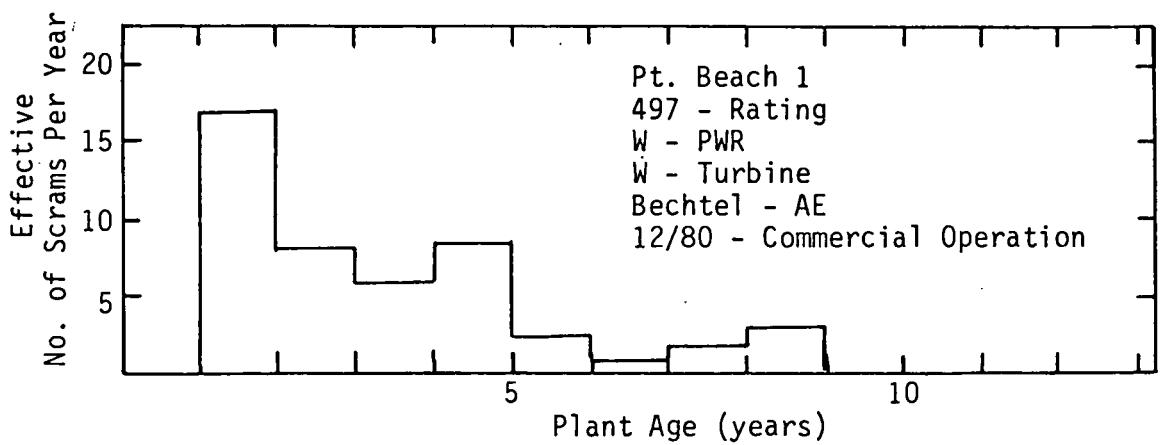
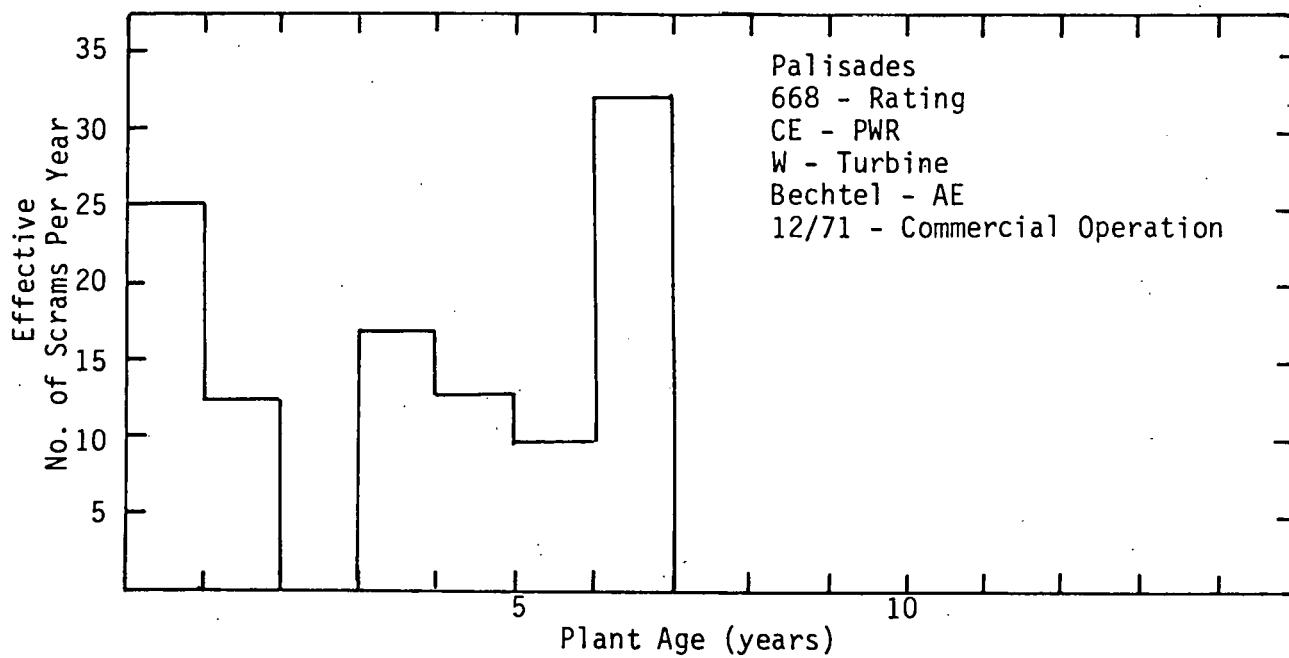
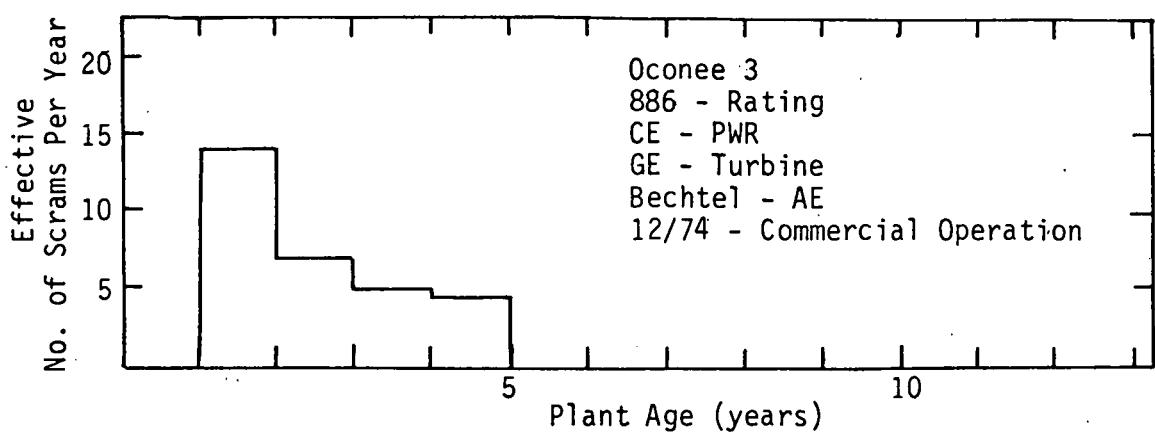


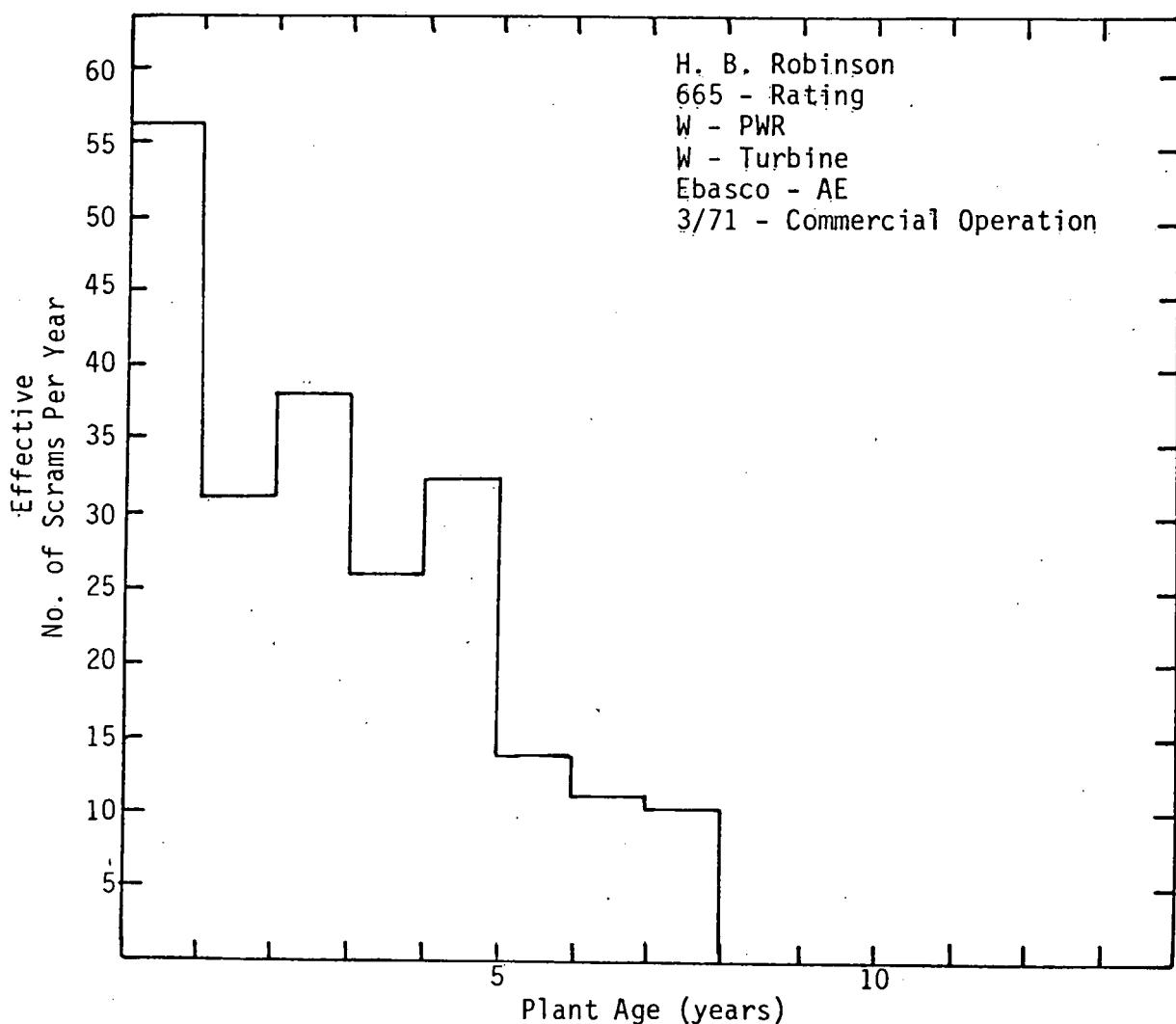
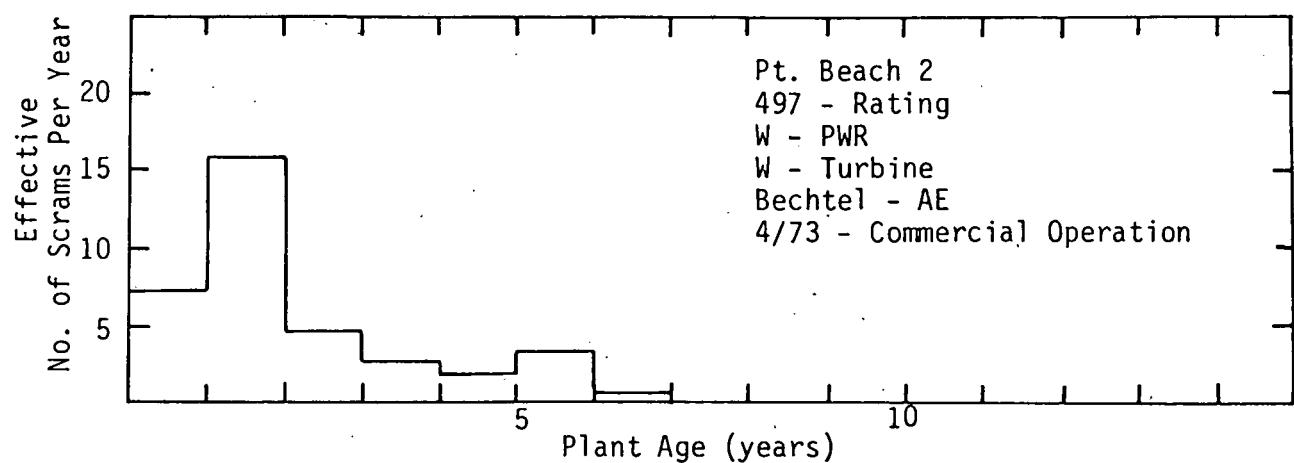


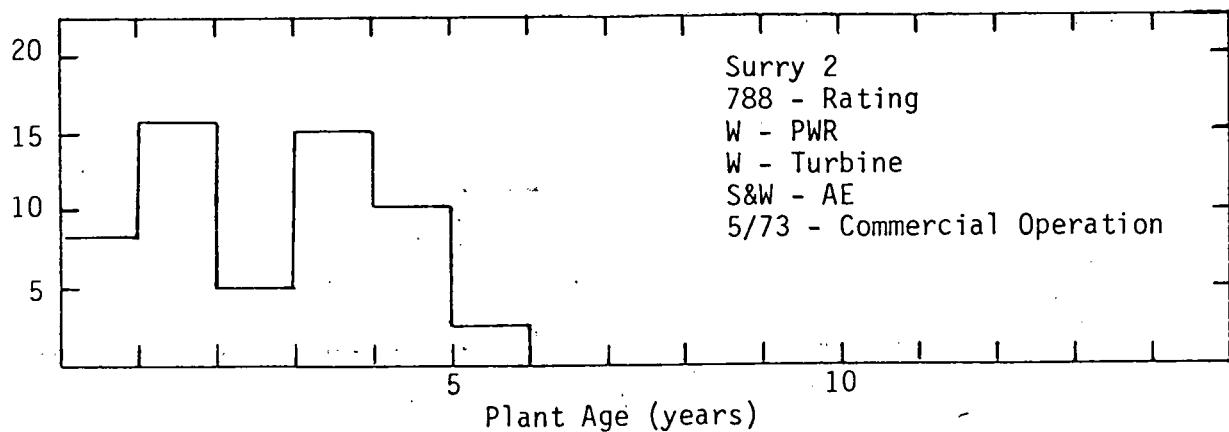
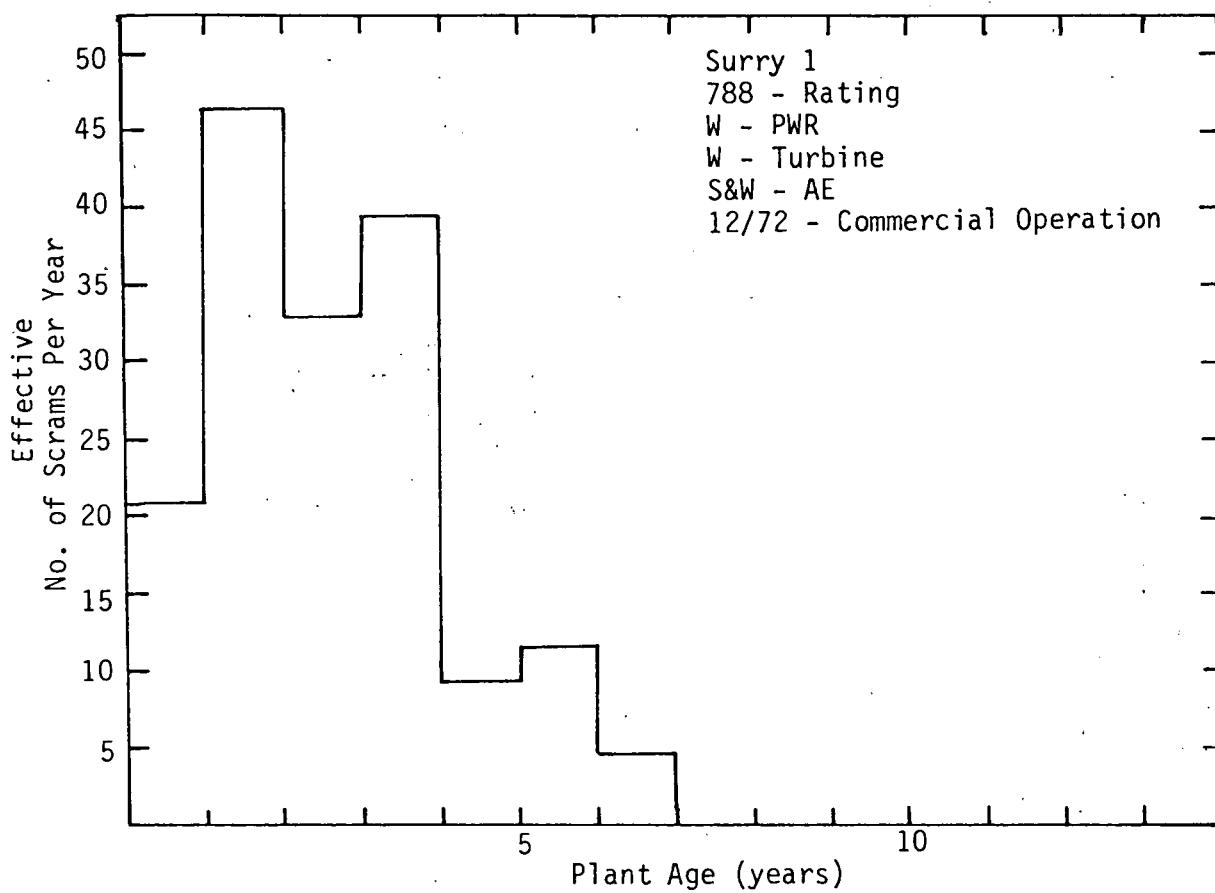
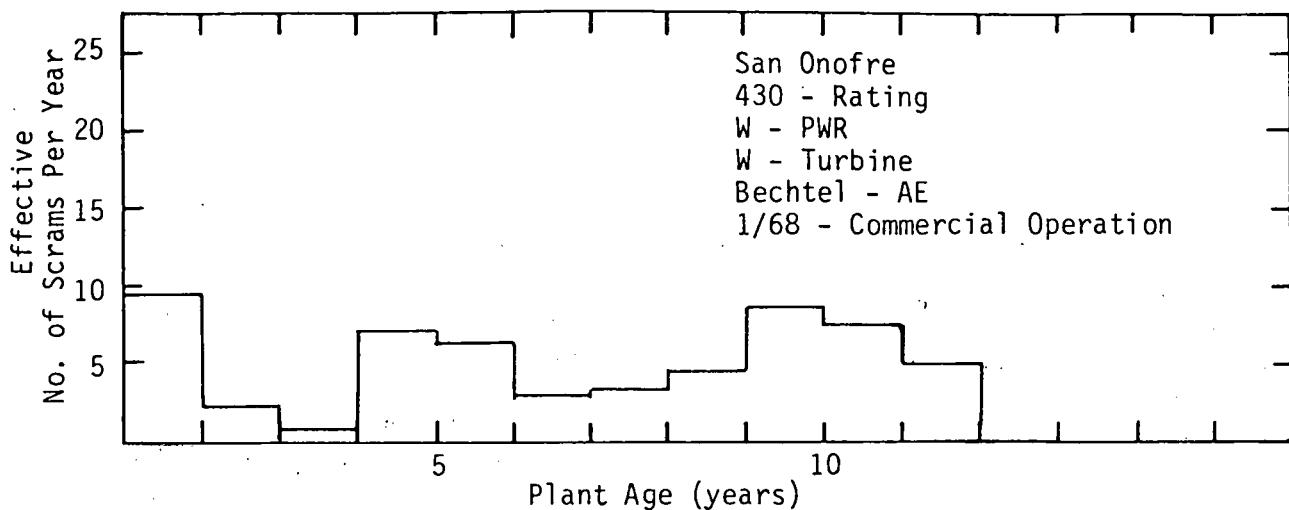


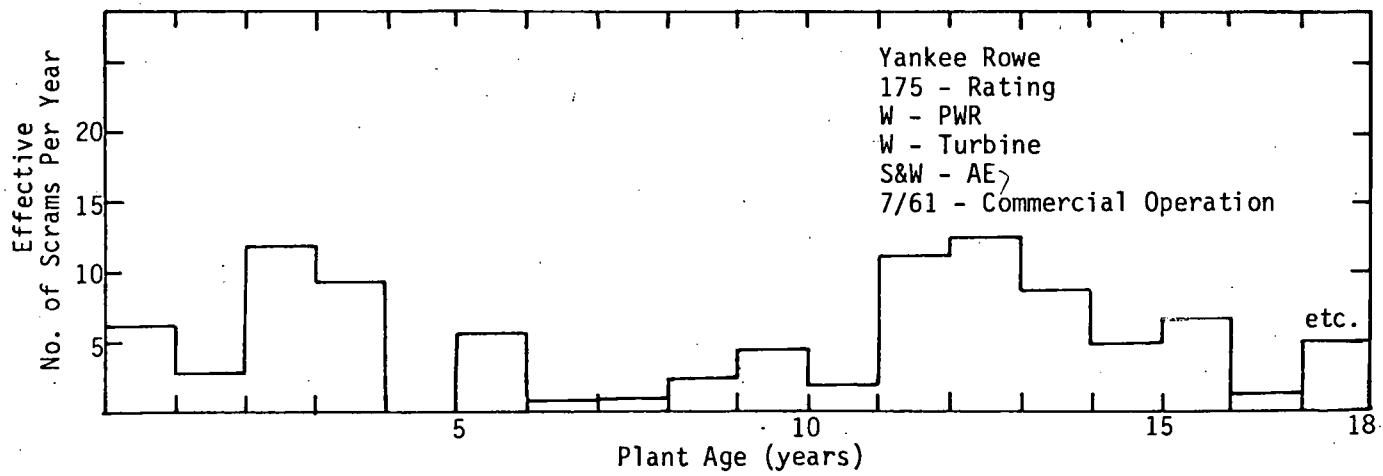
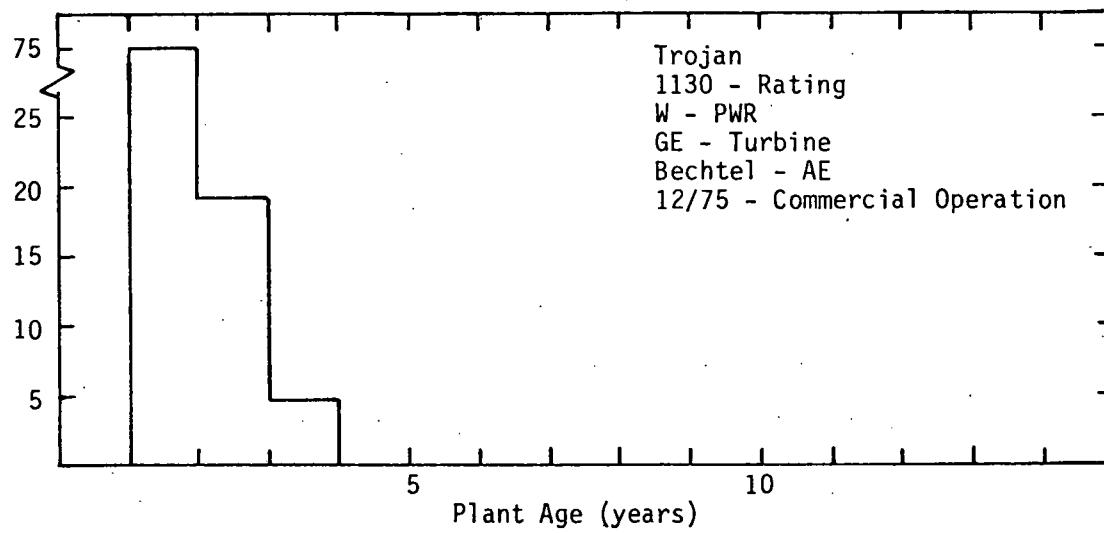
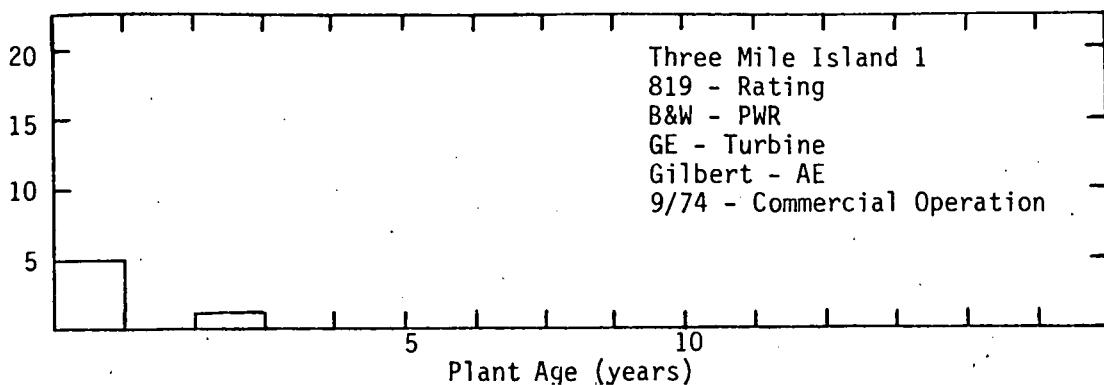


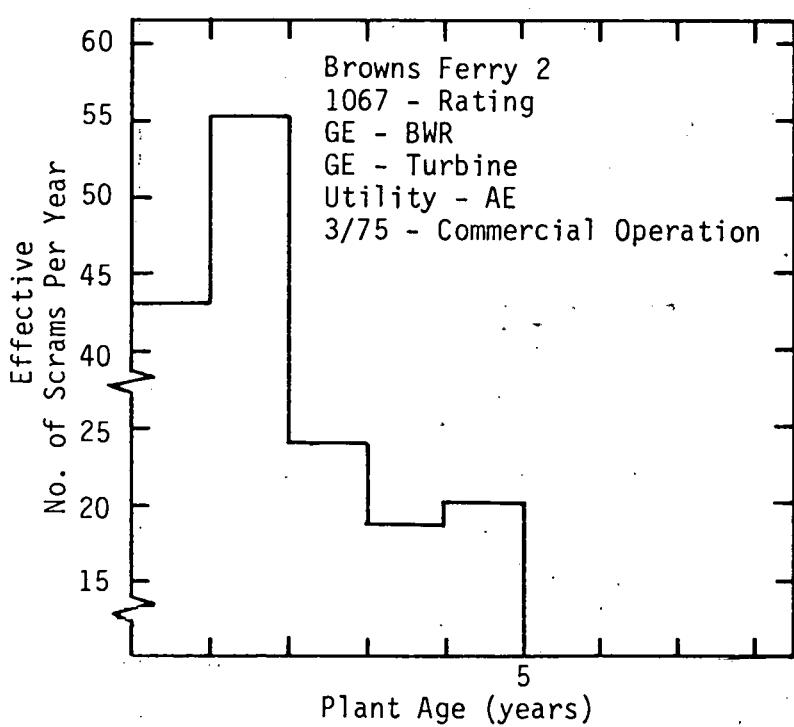
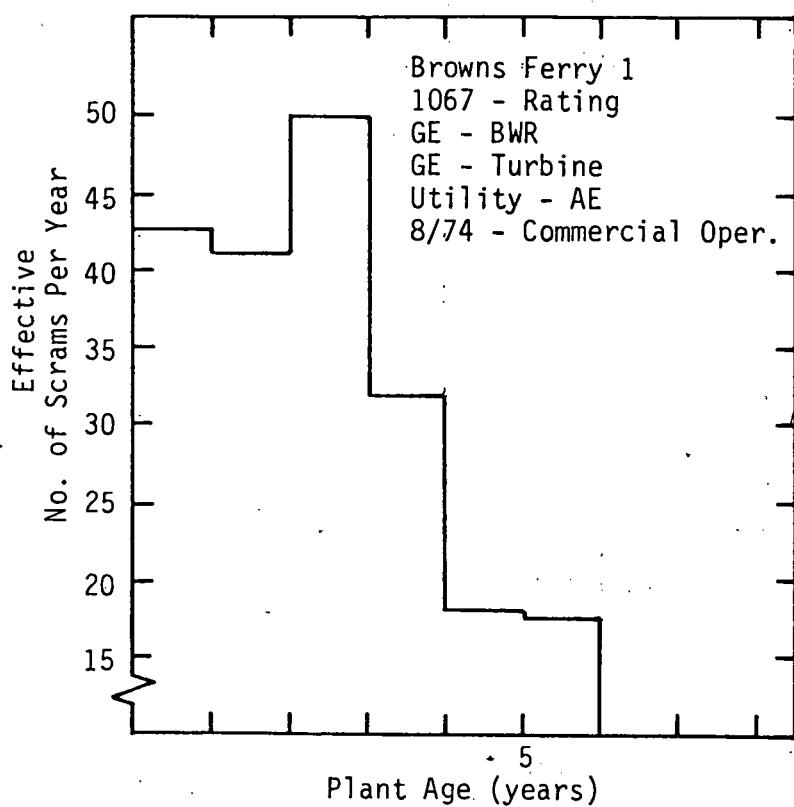


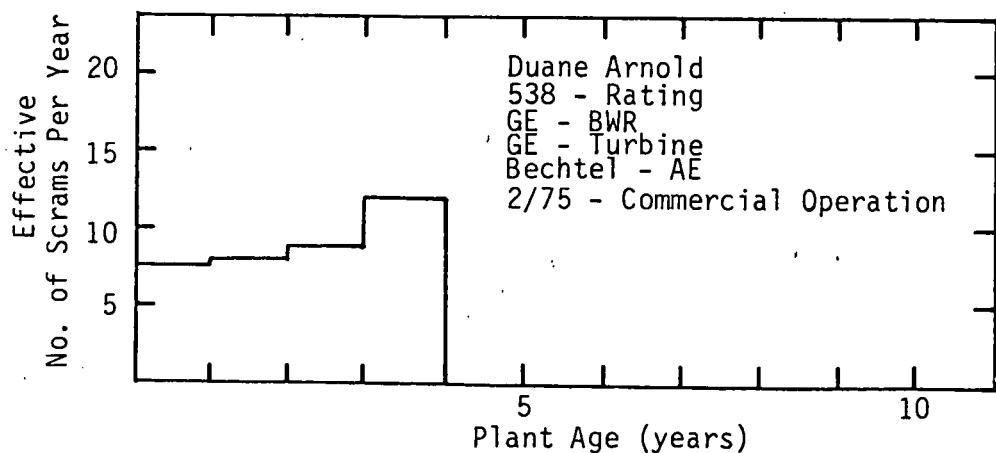
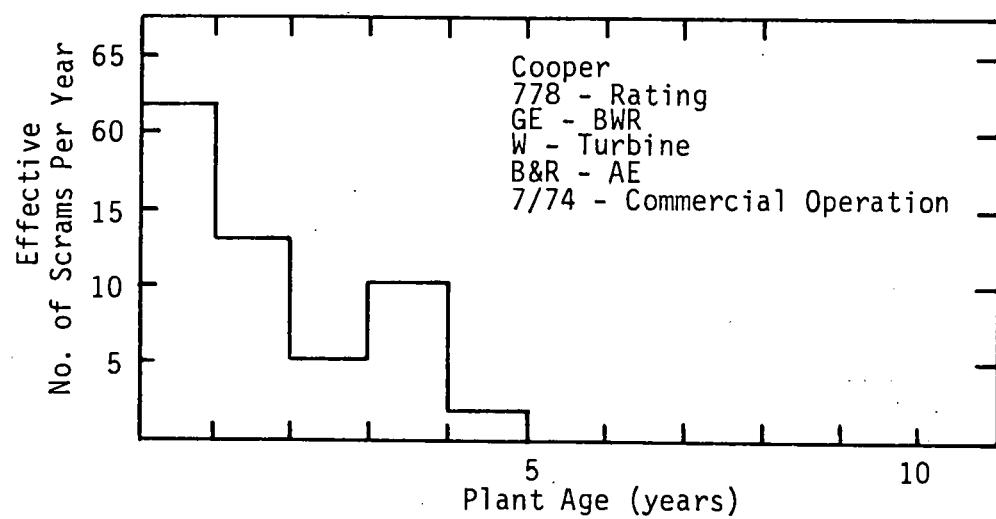
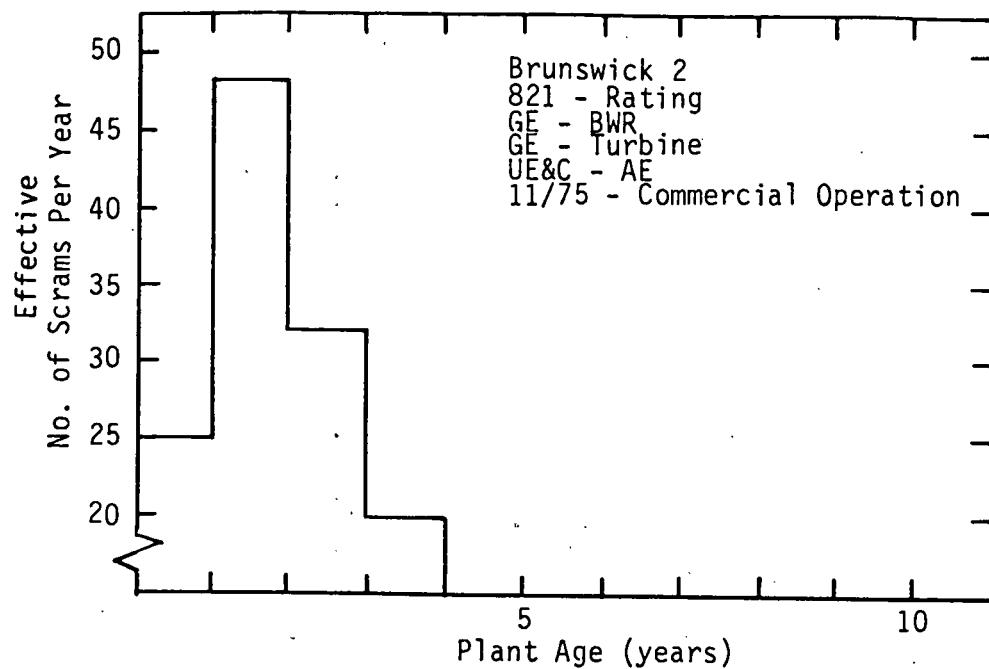


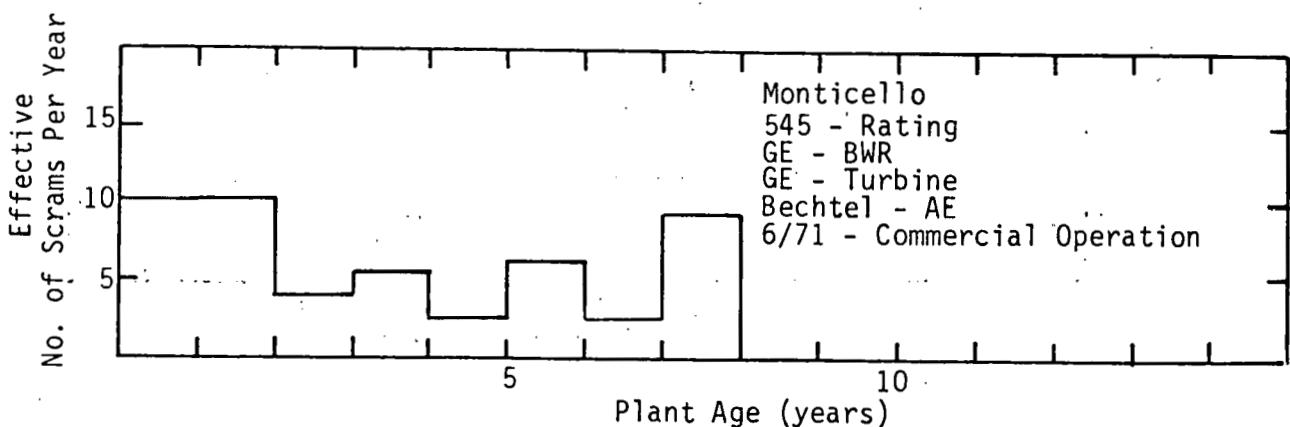
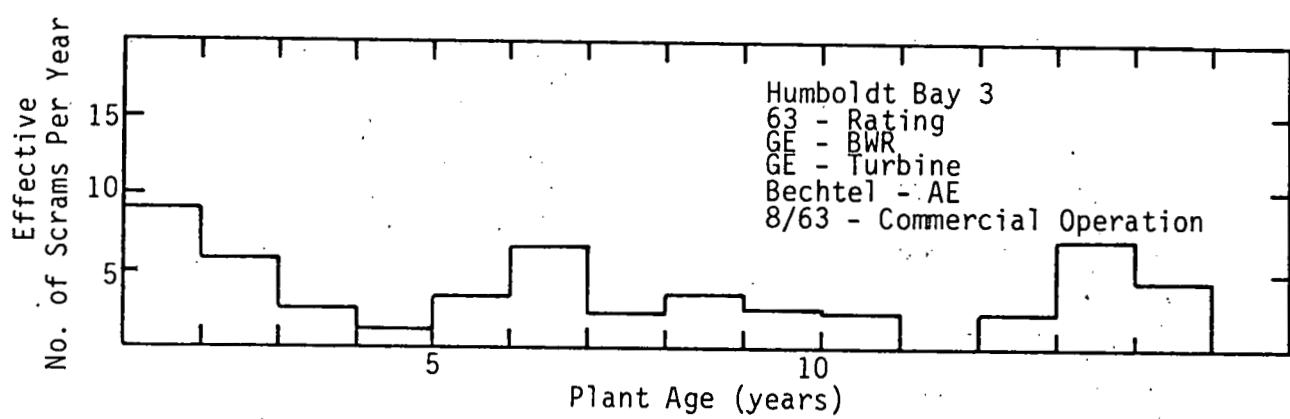
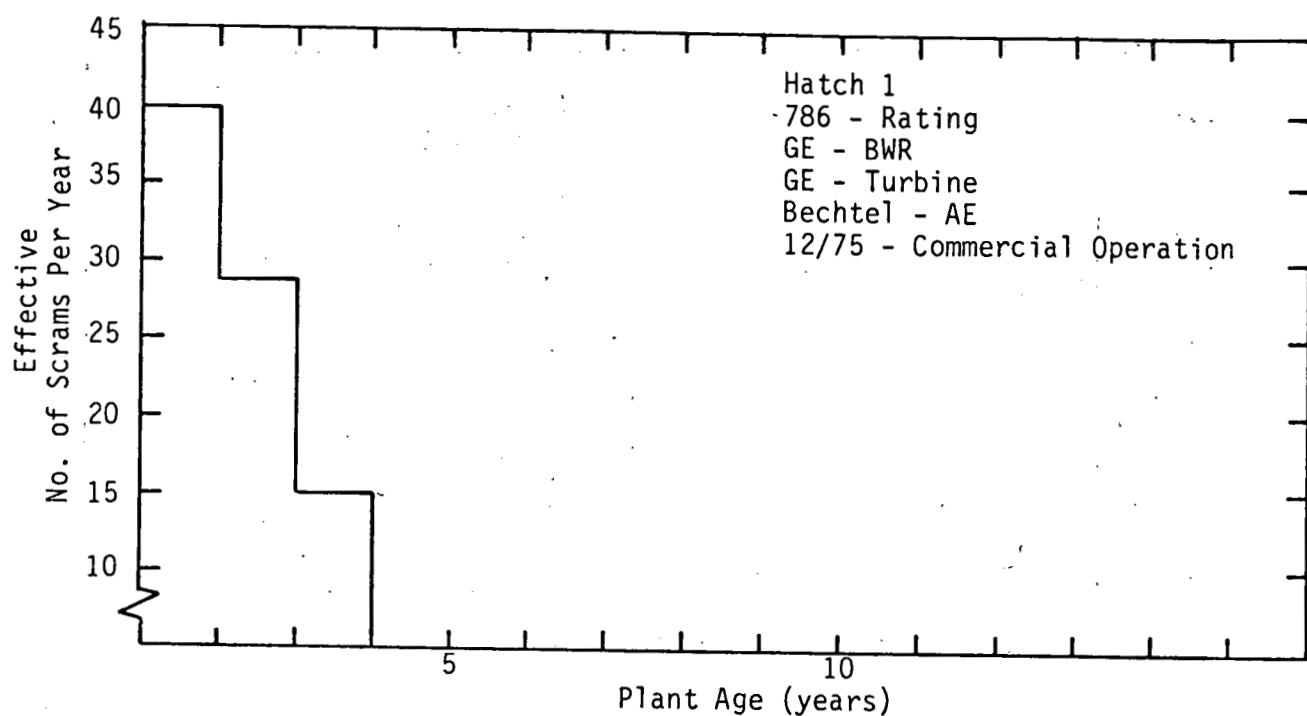


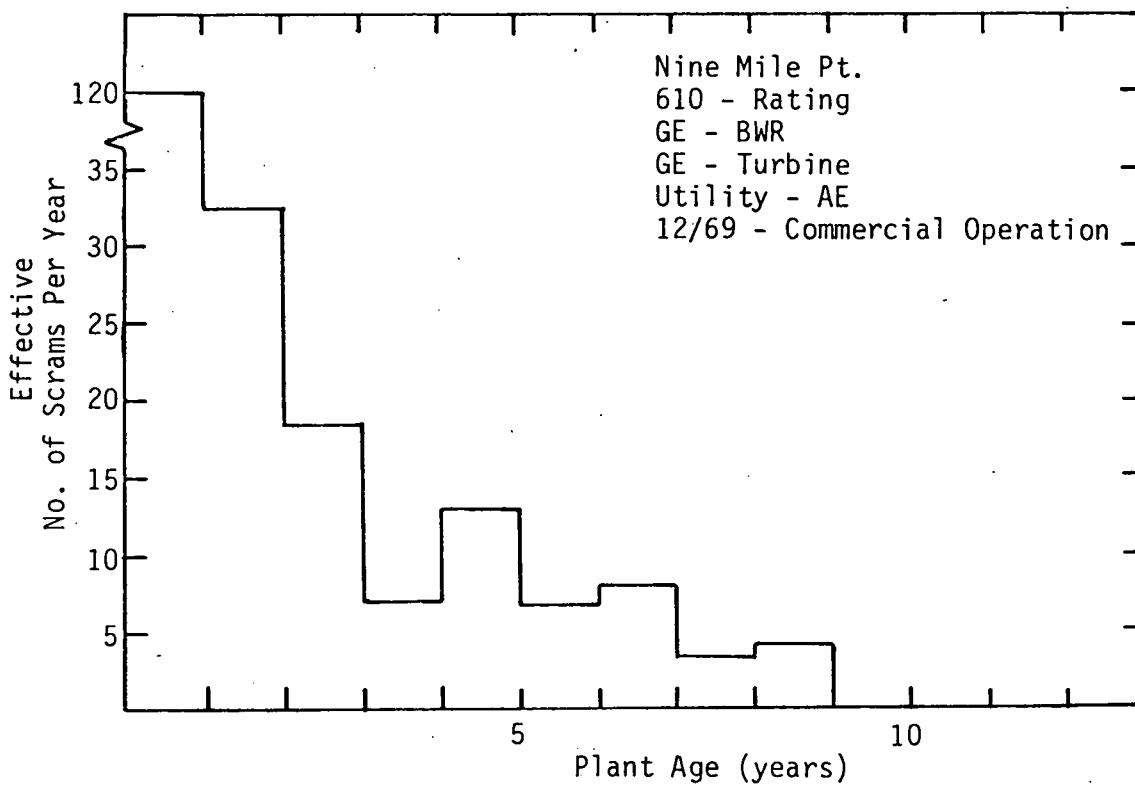
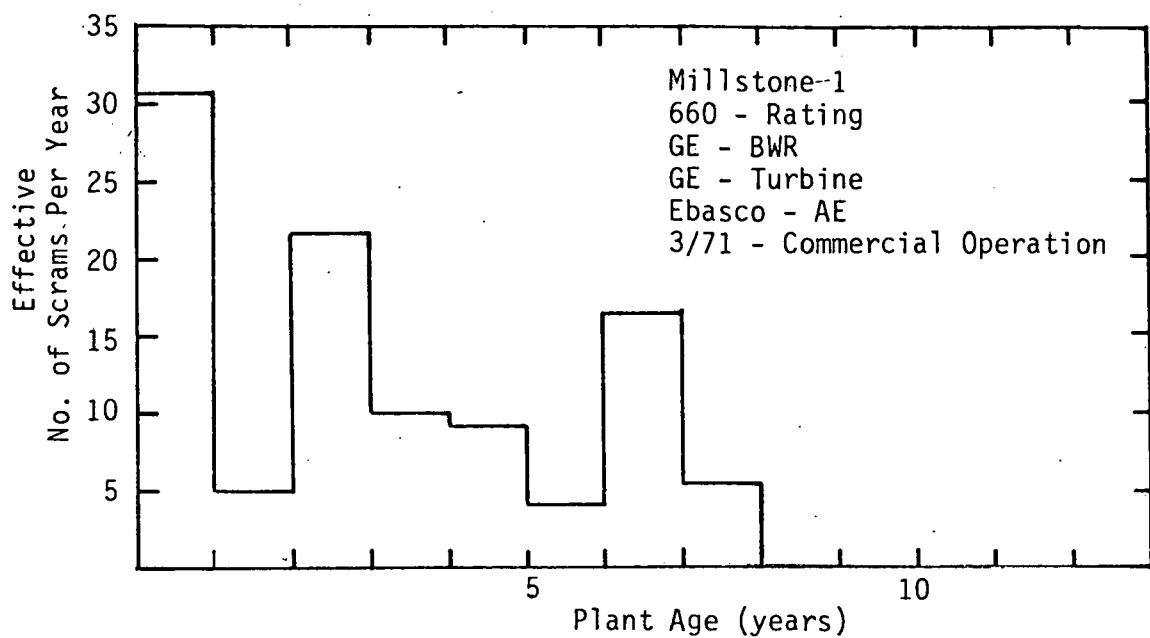


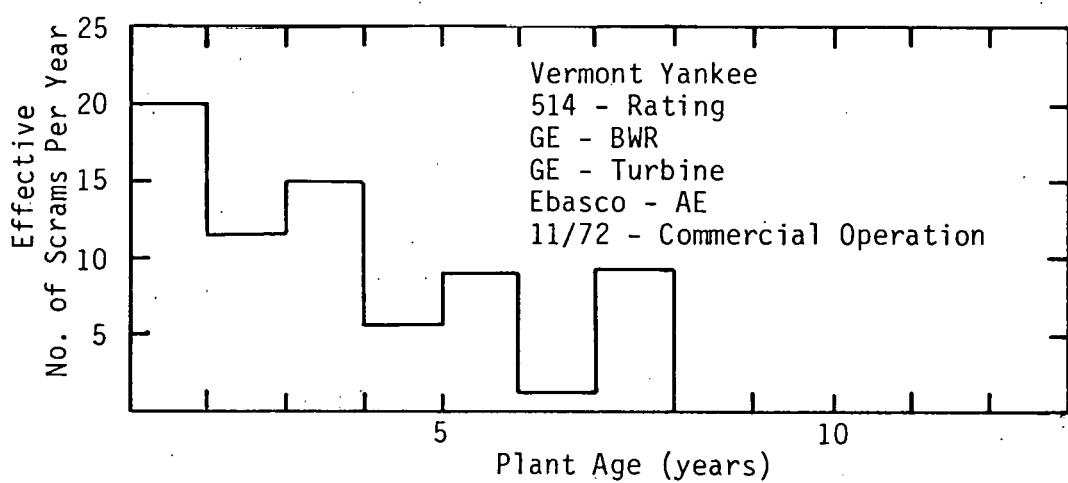
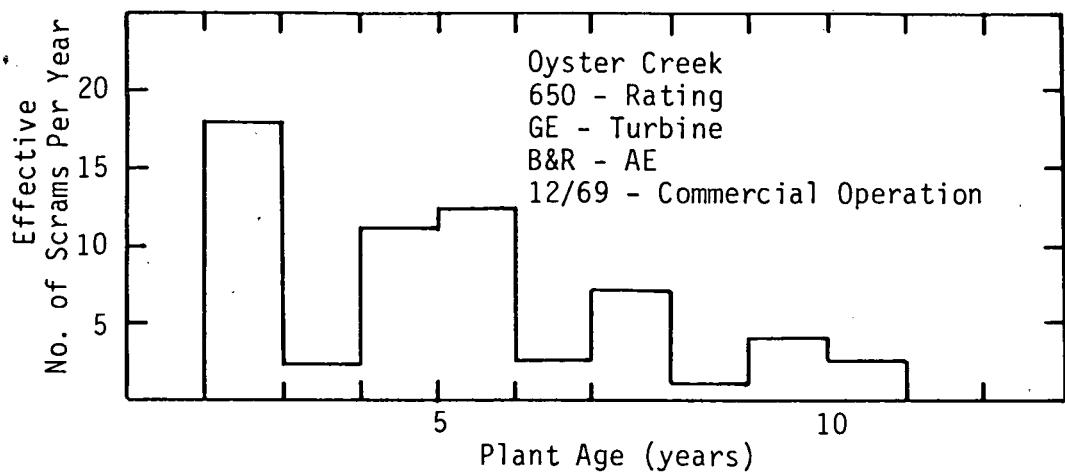












APPENDIX B

SUMMARY OF THE DATA TO INVESTIGATE PLANT SPECIFIC PROBLEMS LEADING TO REACTOR SCRAM FOR B&W PLANTS

This appendix summarizes the available operating experience data on reactor scrams for the operating B&W plants as discussed in Chapter 4.1.

CRYSTAL RIVER 3

DATE	SCRAM	DESCRIPTION
3-77	Auto	---
3-77	Auto	---
4-77	Manual	Failed CRD Motor
4-77	Auto	Surveillance Procedure Problem (Maint./Test)
4-77	Auto	Power Flow Calibration Problem (Maint./Test)
4-77	Manual	Loss of CRD Programmer Power
4-77	Manual	---
6-77	Auto	Loss of Power to Control Rod Group during Surveillance Test (Maint./Test)
7-77	Auto	Loss of Power to one Control Rod Group during Testing (Maint./Test)
9-77	Auto	During Surveillance Test - Power Flow Trip (Maint./Test)
10-77	Auto	Loss of A Invertor/'A' Vital Bus High R.C. Pressure tripped Reactor
10-77	Auto	Power Range Imbalance tripped Reactor
11-77	Auto	Loss of Feedwater Control while in Manual: Reactor tripped on High Pressure
12-77	Auto	Control Rod Group 5 Props due to Control Hardware Problem
1-78	Auto	Power Flow Mismatch: Operator Error
2-78	Manual	Loose Parts in Steam Generator (2 Burnable Poison Rods)
9-78	Manual	Seat Leakage on Main Steam Safety Valve
9-78	Manual	Replace Main Steam Drain Valve

RANCHO SECO

DATE	SCRAM	DESCRIPTION
11-74	Auto	Integrated Control System Tuning for Turbine of Mainfeed Pump "B". Resulted in Reactor Trip.
11-74	Auto	Loss of "J" Inverter
12-74	Auto	Human Error: Tripped Turbine during Instrument Check
12-74	Auto	Power Failure in Control Rod Drives
12-74	Auto	Power Failure in Control Rod Drives
12-74	Auto	Failure of Vital Power Bus "C"
12-74	Auto	Failure of Vital Power Bus "C"
12-74	Auto	Operator Error: Tripped Electrical Bus during Transfer to Alternate Power Source
2-75	Auto	Spurious Overspeed Protection Trip (Turbine)
2-75	Auto	Transient Testing: Slow Integrated Control System (ICS) Response for Feedwater Runback
2-75	Auto	Test: High Flux Imbalance
2-75	Auto	Test: High Flux Imbalance
3-75	Auto	Design Transient (Power Imbalance Trip)
3-75	Auto	Power Imbalance Trip
3-75	Auto	Design Transient (Power Imbalance Trip)
3-75	Manual	Turbine Bearing Inspection
4-75	Auto	Trip occurred due to Steaming Down on OTSG from 100" to 30" which caused <u>RCS Pressure to Increase</u>
4-75	Auto	Reactor Trip on High Pressure due to Loss of "B" Inverter Power

(continued)

RANCHO SECO, continued

DATE	SCRAM	DESCRIPTION
4-75	Auto	Reactor Tripped on Power Imbalance
6-75	Auto	Operator Error: I&C Technician adjusting Turbine Governor Valve caused Inadvertant Trip
		TURBINE BLADE PROBLEM
3-76	Auto	CRDM Power Supply Problem
		SURVEILLANCE TUBE MODIFICATION
10-76	Auto	Main Feed Pump Trip. Repairs made to Hydraulic Control System of Feed Pump
10-76	Manual	Test Scheduled Turbine Test
11-76	Auto	Turbine Trip when Placing Scot System in Service
1-77	Auto	Human Error: Caused Loss of Main Feedwater Causing High Pressure Trip of Reactor
3-77	Manual	100% Turbine Trip Test
7-77	Auto	Dropped Control Rod During Maintenance on CRD Programmer
1-78	Auto	Turbine Trip due to Turbine Oil Relief Valve Lifted. Reactor Tripped on High Pressure
3-78	Auto	Instrumentation Short Circuit
3-78	Auto	EHC System Plugged
9-78	Auto	Turbine Tripped during Maintenance on Autostop Block
12-78	Auto	Feed Pumps Tripped on Overspeed causing Reactor Trip on High Pressure

THREE MILE ISLAND-2

DATE	SCRAM	DESCRIPTION
3-78	Auto	Loss of Power Monitor on Reactor Coolant Pump 1A caused RPS to Believe that no Pumps were Running in one Loop.
4-78	Auto	Power to Flow Noise Spike led to Trip
4-78	Auto	Nonespike on Power Range Channel
4-78	Auto	Loss of Power Monitor on Reactor Coolant Pump 1A caused RPS to Believe that no Pumps were Running in one Loop.
9-78	Manual	Loss of Motor Driven Feed Pump
9-78	Auto	Control Problems with Feedwater Pump 1A
10-78	Auto	Turbine Tripped due to Feedwater Pump Problem. Reactor Tripped on <u>Low Pressure</u> due to over-compensating for the Turbine Trip.
11-78	Auto	Loss of Power to Feedwater System Condensate Polishing Valves
11-78	Auto	Condensate Booster Pump Tripped, thereby tripping the Feedwater Pumps
12-78	Auto	Loss of Feedwater Flow while Shifting Feedwater Pump Turbine 1A from Auxiliary to Main Steam.
12-78	Auto	Condensate Valve Inadvertently Positioned to Full Open Position. Human Error.
12-78	Auto	Feedwater Stopped to OTSGB due to Closure of Valve
12-78	Auto	Feedwater Pump 1B Failure Causes Turbine Trip and Scram.

DAVIS BESSE-1

DATE	SCRAM	DESCRIPTION
10-77	Auto	Reactor Tripped due to Depressurization of the the Reactor Coolant System. The Depressurization was caused During Recovery from a Feedwater Control System Trip.
11-77	Auto	Integrated Control System Increased Reactor Power following an Erroneous Load Demand Increase Signal. Reactor Tripped on Overpower.
12-77	Manual	Turbine Trip Test
12-77	Manual	Turbine Trip Test
12-77	Auto	Low Reactor Coolant System Pressure
12-77	Auto	Low Reactor Coolant System Pressure caused during Recovery of Plant from Loss of Main Feedwater Pump 1.
1-78	Auto	A Human Error led to a Feedwater System Pressure Oscillation which Initiated an Automatic Trip Signal. The Response of the Control System for Two Auxiliary Feed Pumps was Incorrect.
1-78	Auto	Feedwater System Pressure Oscillation which Initiated an Automatic Trip Signal.
2-78	Auto	Reactor Protection System Tripped due to Component Malfunction (High Temp. Spike of RCS)
3-78	Auto	Reactor Coolant System High Pressure
3-78	Auto	Reactor Coolant System Low Pressure. Operator Error during Calibration Testing.
9-78	Auto	Reactor Turbine Trip Test (TP 800.14) Scheduled
9-78	Auto	Reactor Coolant System Low Pressure. RCS Flow Transmitter Failed, causing Feedwater Flow to Increase to Steam Generators Leading to Lower Primary System Pressure.

(continued)

DAVIS BESSE-1, continued

DATE	SCRAM	DESCRIPTION
10-78	Auto	Low Electrohydraulic Control (EHC) Pressure of Turbine Control caused Trip.
11-78	Auto	Fuse in Reactor Coolant Pump Cabinet Blew and Tripped Reactor Coolant Pump 1-2. Reactor Tripped on Power Flow Mismatch.

OCONEE-1

DATE	SCRAM	DESCRIPTION
7-73	Auto	Control Rod Stator
8-73	Auto	Control Drive Stator
9-73	Auto	Control Rod Drive
9-73	Auto	Control Rod Drive
10-73	Auto	Control Rod Drive
10-73	Auto	Control Rod Drive Stator
11-73	Auto	Pressurizer Spray Valve
7-74	Auto	---
7-74	Auto	---
8-74	Auto	---
8-74	Auto	---
10-74	Auto	---
3-75	Auto	Steam Leak on Turbine Instrumentation Valve
3-75	Auto	Fault in the Integrated Control System Delta C(SIC) Instrumentation
3-75	Auto	Spurious Pressure/Temperature Trip
4-75	Auto	Integrated Control System Malfunction
4-75	Auto	Trip During Transient
6-75	Auto	Low Turbine Control Oil Pressure led to Turbine Trip
6-75	Auto	High Reactor Coolant Pressure While in Manual Control
8-75	Auto	Stator Cooling Switch Failed

(continued)

OCONEE-1, continued

DATE	SCRAM	DESCRIPTION
8-75	Auto	Test: Turbine Valves
8-75	Auto	Power/Flow Imbalance During Restart
1-76	Auto	Protective Relay on Generator Causes Unit Trip
5-76	Auto	Reactor Trip due to High Reactor Coolant System Pressure
6-76	Auto	Failure of Reactor Coolant System Flow Indicator
7-76	Auto	Test: Human Error: Steam Heaper Pressure Transmitter Valved Out
7-76	Auto	Maintenance: Reactor Trip during Integrated Control System Maintenance
8-76	Auto	Control Rod Drive Malfunction
8-76	Auto	Test: Control Rod Drive Power Supply Malfunction During Test
10-76	Auto	Test: Control Rod Drive Power Supply Malfunction During Scheduled Test
4-77	Auto	Failure in Integrated Control System Feedwater Control Module Circuitry
4-77	Auto	Reheater Drain Valve Failed to Open. Reheater High Level Trip
5-77	Auto	Test: During Emergency Feedwater Pump Test, lost Condenser Vacuum due to Failed Valve
6-77	Auto	Maintenance: Turbine Shaft Oil Pump Trip caused by Technician Error
6-77	Auto	Turbine Trip
10-77	Auto	Loss of "A" Feedwater Pump

(continued)

OCONEE-1, continued

DATE	SCRAM	DESCRIPTION
12-77	Auto	Personnel Error: Feedwater Valve Closed causing Unit Trip
3-78	Auto	During Shutdown Procedure Switch to Auxiliary Power was not made soon enough. Personnel Error.
3-78	Auto	Calibration Error. Caused Control System to give Trip Signal when Control Switched to Automatic
6-78	Auto	Emergency Pump Valve on "B" Moisture Separator Drain Tank failed to Function Properly causing High Level Trip
8-78	Auto	<u>Test:</u> Malfunction in Trip Mechanism while Performing test
9-78	Auto	MSIV "A" Failed Closed. Reactor Tripped on Overpower?
11-78	Auto	Power/Flow Mismatch
12-78	Auto	Short in T_{Avg} Transmitter Caused High Pressure 2x Trip
12-78	Auto	Loss of Condenser Vacuum caused Power/Flow Imbalance
12-78	Auto	<u>Calibration:</u> Power Imbalance perceived by RPS

OCONEE-3

DATE	SCRAM	DESCRIPTION
4-75	Auto	Aligning Demineralizer Valves led to Transient and Trip (Human Error?)
4-75	Auto	Unit Tripped while Operating Switch Gear
5-75	Auto	Turbine By-Pass Valve Circuitry
7-75	Auto	100% Turbine Trip <u>Test</u>
7-75	Auto	Turbine Tripped on Momentary loss of DC Power to EHC
10-75	Auto	Power/Flow Imbalance
1-76	Auto	Loss of DC Control Power to Turbine Controls
4-76	Auto	Power/Flow Imbalance
7-76	Auto	Spurious Turbine Trip
9-76	Auto	Maintenance: Reactor Trip During Maintenance on Power/Load Imbalance Runback Circuit
11-76	Auto	Test: Turbine Tripped during Turbine Thrust Bearing Wear Detector Testing
4-77	Auto	Momentary Loss of 125 V DC Power to the Turbine Control System
4-77	Auto	Spurious Power/Flow Trip
10-77	Auto	Indicated Loss of DC Power to EHC System caused Trip
12-77	Manual	<u>Test:</u> Manual Trip Test
4-78	Auto	<u>Test:</u> While Placing RCP BA 1 on, Reactor Tripped, Motor Control Center Breaker Tripped, Loss of Power to EHC System Pumps
8-78	Auto	Swing in Feedwater System Caused Power/Flow Imbalance Tripping Reactor
11-78	Auto	Swing in Feedwater System caused Unit Reactor Trip

OCONEE-2

DATE	SCRAM	DESCRIPTION
5-75	Auto	Loss of Condensor Vacuur
8-75	Auto	Failed Control Rod Drive Power Supply
8-75	Auto	Loss of Main Feedwater Pump
8-75	Auto	<u>Test</u>
7-76	Auto	Reactor Tripped during Shutdown
9-76	Auto	Test: Speed Amplifier Circuit Failed during a Scheduled Turbine Trip <u>Test</u>
12-76	Auto	Control Rod Drive Power Supply Relay Failure
3-77	Auto	Control Rod Dropped during Scheduled Rod Movement <u>Test</u>
11-77	Auto	Emergency Feedwater Pump Trip
1-78	Auto	<u>IP:</u> Control Rod Drive Fuse Failure
5-78	Auto	Valve 2HD-28 Failed Closed during Moisture Separator Reheater Drain <u>Test</u>
8-78	Auto	Operator Error: Power Flow Imbalance Following Tripping of 2B2 RCP
8-78	Auto	Low Pressure in Reactor Coolant System due to Pressurizer Spray Valve Malfunction when placed in Auto
8-78	Auto	Integrated Control System Increased Feedwater Flow, Reactor Coolant System low Pressure caused Trip Signal
10-78	Auto	Air Supply Leak caused 2 Feedwater Valves to Close, Reactor Tripped
10-78	Auto	Feedwater Pump 2B Tripped causing a Reactor Trip (Pump Control Problem)

THREE MILE ISLAND-1

DATE	SCRAM	DESCRIPTION
6-74	Manual	Thermal Overload Trip on Air Compressor
6-74	Auto	"B" Side Steam System By-Pass Valve Opened
6-74	Manual	Scheduled Test Procedure: Cooldown from Outside Control Room
6-74	Auto	Station Blackout
7-74	Manual	Reactor Trip Test
7-74	Auto	Faulty Relays on Generator caused Generator Trip. Reactor Tripped on Low Pressure
7-74	Auto	Oscillation on Secondary Plant Feed System caused the Reactor to Trip on Pressure-Temp.
7-74	Auto	Human Error: Technician removed T_{Avg} Signal from Integrated Control System which caused the Reactor to Trip
8-74	Auto	Power/Flow Imbalance
8-74	Auto	Test: Generator Trip Test. Reactor Tripped on High Pressure
8-74	Auto	Turbine Trip on High Vibration in #11 Bearing. Operator Placed Feedwater Demand in Manual. Reactor Tripped on Variable Temp./Pressure.
3-75	Auto	Erroneous Signal Tripped Turbine (i.e., Erroneous Loss of DC Power Signal to EHC). Reactor Tripped 4 seconds later on High Pressure.
5-75	Auto	Turbine Trip. Mechanical Failure in the Level Trip Device for the "B" Moisture Separator (High Moisture)
6-75	Auto	Turbine Electrohydraulic Controller caused Erroneous Signal Leading to Turbine Runback. Reactor Tripped on High Pressure.

(continued)

THREE MILE ISLAND-1, continued

DATE	SCRAM	DESCRIPTION
6-75	Auto	Dropped 1 Control Rod Group due to Electrical Fault
5-76	Auto	Integrated Control System Received Incorrect Input; began Pulling Control Rods until Core Power Exceeded Trip Point
11-77	Auto	Failure of Integrated Control System Module Signal Converter caused Reactor Power to Increase to 103%. Power/Flow Trip.
6-78	Auto	Dropped 3 Control Rods due to Shorted Diode in DC Power Supply.

ARKANSAS-1

DATE	SCRAM	DESCRIPTION
10-74	Auto	Main Feedwater Pump "B" Tripped for Unknown Reason, causing High Pressurizer Level, Thereby Tripping the Reactor
10-74	Auto	Main Feedwater Pump "A" Tripped due to High Vibration causing High Pressurizer Level, Thereby Tripping the Reactor
11-74	Auto	Momentary Interruption of Station Power
11-74	Auto	High Power Indication during Power Escalation Testing
11-74	Auto	Reactor Coolant Pump Under Power Relays Failed to Open Falsely Indicating Low Reactor Coolant Flow
11-74	Auto	Power Escalation Testing: Surge of Reactor Coolant Flow when Adding a Fourth Pump
12-74	Auto	High Pressure and Temperature in Reactor Coolant System
12-74	Auto	Test: Manual Trip of Turbine/Generator from 100% Power caused Reactor Trip
12-74	Auto	---
1-75	Auto	Generator Trip
2-75	Auto	Loss of Offsite Power due to Windstorm
5-75	Auto	Power/Flow Imbalance
5-75	Auto	Power/Flow Imbalance
6-75	Auto	Failed Cold Leg RTD Connector caused Power/Flow Imbalance Signal and Trip
6-75	Auto	Dropped Control Rod Group, due to Switch Failure. Reactor Tripped on Low Pressure

(continued)

ARKANSAS-1, continued

DATE	SCRAM	DESCRIPTION
7-75	Auto	Personnel Error: Bumped Inservice Instrumentation. ICS Reacted to False Indication to Increase Temperature, which resulted in Reactor Trip
12-75	Auto	Lightning Strike caused Voltage Fluctuation and CRDM Trip
12-75	Auto	CRDM Ratchet Trip
1-76	Auto	Ratchet Trip
2-76	Auto	Failed CRD Transfer Switch Dropped one Group
7-76	Auto	False Indication of High Reactor Coolant System Pressure due to Failure of one Nuclear Instrumentation Channel
9-76	Auto	Turbine Tripped on Bearing Vibration Alarm
11-76	Auto	Main Feedwater Pump Control Malfunction caused Reactor Trip
12-76	Auto	Partial Loss of Control System Power caused Automatic Control System Actions which resulted in Reactor Trip on High Pressure
4-77	Manual	False Hotwell Level Indication
11-77	Manual	Reactor Building Cooling Fan Failure Shutdown per Tech. Specs.
6-78	Auto	<u>Operator Error: Test:</u> During Emergency Diesel Generator Surveillance
10-78	Auto	Loss of RCS Loop A Flow to <u>ICS</u> Feedwater Flow Signal
12-78	Auto	Main Turbine Governor Valve. LVDT Failure

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