EPRI-NP-755

Major Outage Trends in Light Water Reactors

NP-755
Research Project 705

Interim Report, April 1978

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FOREWORD

This report is a product of an ongoing program at EPRI to analyze existing power industry data systems and to provide feedback on the analysis to the industry.

The report is designed to provide a better understanding of the causes and consequences of major outages (i.e., defined by the report to be outages greater than 100 hours in duration) in nuclear power plants. Refueling outages are not included in the analysis and will be addressed in a later report.

Major outages were chosen as a topic because as a group they contribute as much to nuclear plant unavailability as the yearly refueling outages and, as such, should be an important concern to utilities in their outage and maintenance planning. The report shows industry averages and trends which could be used as direct input to such planning.

Data analysis of this and other types will continue to be undertaken by EPRI in an effort to guide research and development work and provide useful information to the industry.

W. L. Lavallee
Project Manager

ABSTRACT

This report is a summary of the major outages which occurred in light water reactor plants during the period January 1971 through June 1977. Only those outages greater than 100 hours duration (exclusive of refueling outages) are included in this report. The trends in outages related to various reactor systems and components are presented as a function of plant age, and alternatively, calendar year. The principal contributors to major outages are ranked by their effect on the overall outage time for PWRs and BWRs. In addition, the outage history of each operating nuclear plant greater than 150 MWe is presented, along with a brief summary of those outages greater than two months duration.

TABLE OF CONTENTS

TABLE OF CONTENTS

ILLUSTRATIONS

ILLUSTRATIONS

ILLUSTRATIONS

TABLES

SECTION 1.0

INTRODUCTION AND SUMMARY

1.1 Background

This report provides a summary of the "major outages"* which occurred in light water reactors over the period January 1971 to June 1977. This six and one half year period of reactor operating experience provides a basis for assessing the trends in LWR plant performance over the initial years of plant operation.

Previous studies[1,2,3,4,5,6,7,8] have evaluated nuclear power plant operation. Each of these efforts has been limited to some extent by the lack of plant operating experience since the available data was based upon a small population of plants operating over a relatively short period of time. However, in the period 1975 to 1977, there has been a substantial increase in the amount of nuclear operating experience. To build on the previous work and to utilize the latest nuclear experience, a review of the major outages which impact on plant availability is given in this report.

The benefit to be gained from identifying those areas in need of improvement has been estimated by the former Federal Energy Administration (FEA)[9]. An increase of one percentage point in plant availability reduces the installed capacity requirement by approximately 6800 MWe (and therefore, the capital requirements by $1.8 billion) by 1980. In terms of reduced oil requirements, an increase in nuclear and coal availability of approximately 5% could reduce oil requirements by 500,000 barrels/day.

Several sets of categories of events can be chosen to classify the causes of reduced plant productivity (e.g., causal factors, length of outages, equipment involved). One set of categories which appears to be convenient is the following:

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* "Major outage", as used in this report, refers to an outage greater than 100 hours duration, exclusive of refueling outages.

a) Refueling Outages

b) Major Outages

c) Short Duration Outages

d) Power Restrictions

This classification is one vehicle that allows us to point out the areas which may require increased management attention. Through understanding the trends in nuclear operating experience, it is hoped that improvements can be made in component reliability, preventive maintenance schedules, and outage planning.

The results from previous works have pointed out that long duration outages have contributed significantly to the reduction in overall plant performance. However, there is a continual controversy over the issue of the trend of these outages. One argument is that the age of the plant does not have a significant effect on plant performance, while the counter argument suggests that each new plant has a "break-in" period during which a substantial number of its deficiencies are corrected. In an attempt to resolve this controversy, the current study shows the distribution of major outages which occurred over the initial seven years of plant operation as a function of plant age.

1.2 Objectives

The purpose of this study is to utilize operating experience data to identify trends of major plant outages in light water reactors (LWRs).

This report focuses on the initial seven years of LWR plant operation to obtain a sufficient statistical base to have confidence in the results. Even though seven years represents less than one fifth of a projected LWR lifetime, it is judged important to monitor the trends of key causes of plant unavailability in order to anticipate future plant performance and adjust planning for maintenance and equipment replacement. At the present time, each utility, component designer, and architect-engineer must make numerous key decisions on plant design, arrangement, and operation without the aid of adequate data. This summary report is aimed at providing a small piece of the data which can lead to better decisions for improving plant reliability.

1.3 Scope and Limitations

The scope of this report is limited in the following areas:

- Plant unavailability: only outages greater than 100 hours duration, exclusive of refueling, are considered.
- Population: all US light water reactors greater than 150 MWe which are in commercial operation are included.
- Time frame: the period January 1971 through June 1977 is considered.

This report focuses only on major outage trends. As pointed out in Section 1.2, additional categories of reduced plant performance must be considered in conjunction with this assessment to provide an accurate overall picture of reactor operating performance.

The data collection has been limited to the period 1971-1977 for two complementary reasons:

a) This data is the most reliable since it is generally from two or three sources[9,10,11].

b) The recent data is considered most applicable to the future trends in the nuclear industry.

The number of US plants greater than 150 MWe in commercial operation prior to 1971 is quite small, and data from them is judged not to significantly alter any of the conclusions of this study.

The plant size has been limited to plants larger than 150 MWe to focus on those plants which are most representative of the current and future generation of nuclear plants. Specifically, those plants which are eliminated from the current study are small prototype units which have had good records but may have incurred some unique problem (i.e., Humboldt Bay - 65 MWe, La Crosse - 50 MWe, Big Rock Point - 72 MWe). In addition, Indian Point 1 has not been included because it is presently shutdown and not scheduled for future operation.

The number of outages and their durations are well substantiated; therefore, there is little uncertainty associated with these aspects of the data. On the other hand, there are some cases of lack of precise definitions as to the cause of each outage or reasons for extension of the outage. The convention used in this report is to categorize outages according to their primary cause. If secondary causes for the outage or its extension are also reported, the outage is divided according to the amount of effort reported. In a few cases (~2%) an arbitrary decision is made to apportion outage time approximately based upon the judgement of the author. In addition, there are a few instances (~3%) where two maintenance/inspection jobs proceeded in parallel. These cases are treated as one outage in the overall summary, but are double counted when they are categorized by system or component for observation of trends. This later case involves primarily snubber or pipe restraint inspection and repair.

As will be shown, there are substantial data for light water reactors over the first years of commercial operation (0-4 years); however, in the period of 5, 6, and 7 years of commercial operation, the number of plants is quite limited, and therefore, there is a much larger uncertainty in the outage trends shown for these years.

Since the scope of this report is limited to the initial seven years of commercial operation, we are focusing on the nuclear power plant break-in period. There is insufficient data to develop any correlation for outages occurring in the time frame of 8-17 years, and there are no commercial plants operating in the period of 20-40 years. These latter two periods of operation cannot be addressed based upon available data. The classical "bath tub" curve of component failure rate suggests that at some point in the LWR lifetime, components will begin to wear out at an increasing rate. Identification of this point in time would be a valuable aid in utility planning; therefore, continued monitoring of plant performance is considered prudent.

Prior to commercial operation, a significant amount of outage time is caused by required testing, equipment start-up problems, and regulatory questions. These aspects of plant operation are important contributors to reduced energy output; however, since they are one time occurrences for a plant, and the focus of attention in this report is on the long term trends in plant operation, the data evaluation deals only with the events occurring after commercial operation has begun. In fact, this limitation of scope merely translates the origin of the trends from initial criticality to initial commercial operation. Experience indicates that the distribution of the major outages during the 3-6 month period from initial criticality to commercial operation are similar to those found in the initial year of commercial operation.

One area of outage management which has not been treated in this study is the impact of using outside contracted services in lieu of plant maintenance personnel. Past experience[34] with contracted services indicates that they rely heavily on craft people from local unions and that the quality of work and length of outages tends to be much more variable than if sufficient plant maintenance personnel could be brought to bear on the problem. However, information on the degree to which outside services are utilized is not readily available and is difficult to quantify; therefore, this area enters as an uncertainty in the analysis.

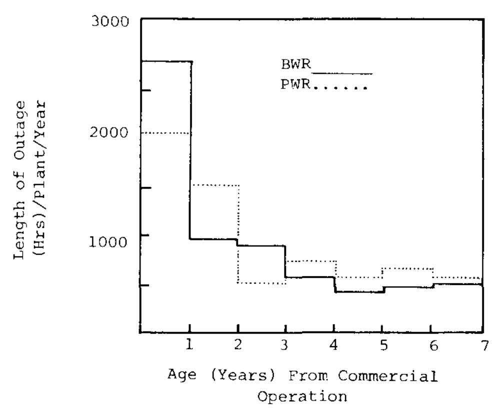
1.4 Summary of Conclusions

1. Over 50% of the major outage* time is related to one of the following four plant areas:

    - Steam System, principally the turbine related problems
    - Steam Generators, including inspection, testing, and tube plugging operations
    - Reactor In-Core Problems, caused by flow induced vibration
    - Reactor Coolant Pumps, principally PWRs

2. The major outages are approximately evenly divided between the Nuclear Steam Supply System (NSSS) and the balance of plant (BOP)

3. The distribution of major outages* as a function of plant age is shown in the figure below to have a distinctive trend. A high initial outage rate per plant is characteristic over the first two years, while subsequent years, third through the seventh, show a leveling off of the outage rate. The leveling off of the outage rate may be indicative of a maturation in the plant life cycle.



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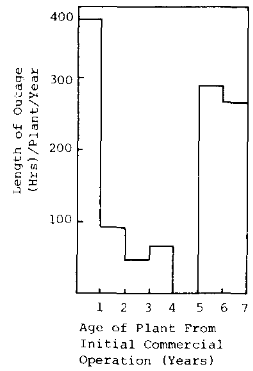
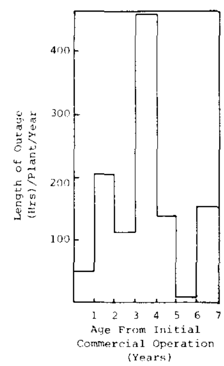
*Major outages as used in this report are outages >100 hours exclusive of refuelings

4. The outage trends for individual components may differ from the overall trend of the plants shown in item 3 above because the mode of failure may be quite different for each component, such as:

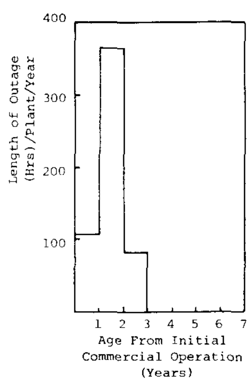
- Steam Turbine failures have caused a large contribution to the major outages required during the first year of commercial operation which is characteristic of uncovering a new design related problem. However, there has also been a relatively high contribution to the major outage time in the sixth and seventh years of operation which may indicate that turbine outages could be a constant long term concern.



- Steam Generators have caused major outages which have their highest percentage impact in the fourth year of commercial operation. Virtually all of the outages are related to failure of the steam generator tubes which appear to require an incubation period before the failures appear.



- Reactor Internals have been involved in several lengthy outages. The problems are generally generic in nature and affect an entire class of reactor. The actual failure mechanism has been flow-induced vibration. Note that these problems have characteristically surfaced during the second year of operation.



5. From Appendix A, a review of the individual plants indicates that many plants have recurring problems related to a particular system (e.g., Steam System) or component (e.g., Reactor Coolant Pumps). While the same problems do not occur at all plants, a limited number of other plants do face similar recurring problems and can benefit from the operating experience.

6. Over the period January 1971 to June 1977 there were 13 plants which incurred non-refueling major outages greater than two months duration. The principal causes of these exceptionally long "rare" events are as follows:

Event	No. Plants
In-Core Problem	5
Fire	2
Generator	2
Turbine Blade Failure	2
Steam Generator Inspection/ Repair	1
Feedwater Pipe Failure	1

SECTION 2.0

IMPACT OF LONG DURATION OUTAGES

2.1 Nuclear Plant Population

The population of nuclear plants considered in this report consists of fifty-six operationg LWRs of diverse size and design. Since each of the plants has been custom designed, caution must be exercised in the use of the data. The best that can be expected is that a characteristic trend can be identified which will dominate the differences in design, construction, and size. The fact that the plants are of unique design may lead to the belief that major outages are also unique to certain plants. In answer to this question, a summary of outages for each plant is included in Appendix A. These individual outage sheets are included to give a concise summary of the factors causing reduced plant availability in each plant.

One attempt toward highlighting differences in trends due to fundamental design differences is in the division of PWR and BWR populations; therefore, a profile of the nuclear plant population is divided between BWR and PWR plants as shown in Figures 2.1 and 2.2, respectively. In addition to design differences, the plants also vary in age from less than 6 months to more than 16 years; therefore, a comparison of trends in major outages as a function of calendar year would lead to mixing very young and very old plants in a single calendar year. A comparison among the plants based on the age of the units rather than on the calendar year of operation may be more useful. Such a comparison may indicate if there is an inherent variation in equipment outages as a plant increases in age from the initial "break-in" phase to a "mature" phase of operation. While it shall be shown that in some cases (e.g., snubber and pipe inspection) the calendar year of operation is important in determining trends in the industry, the key parameter in isolating trends in major outages is the age of a plant. (In this report the age is measured from initial commercial operation.) Figures 2.3 and 2.4 show the nuclear plant

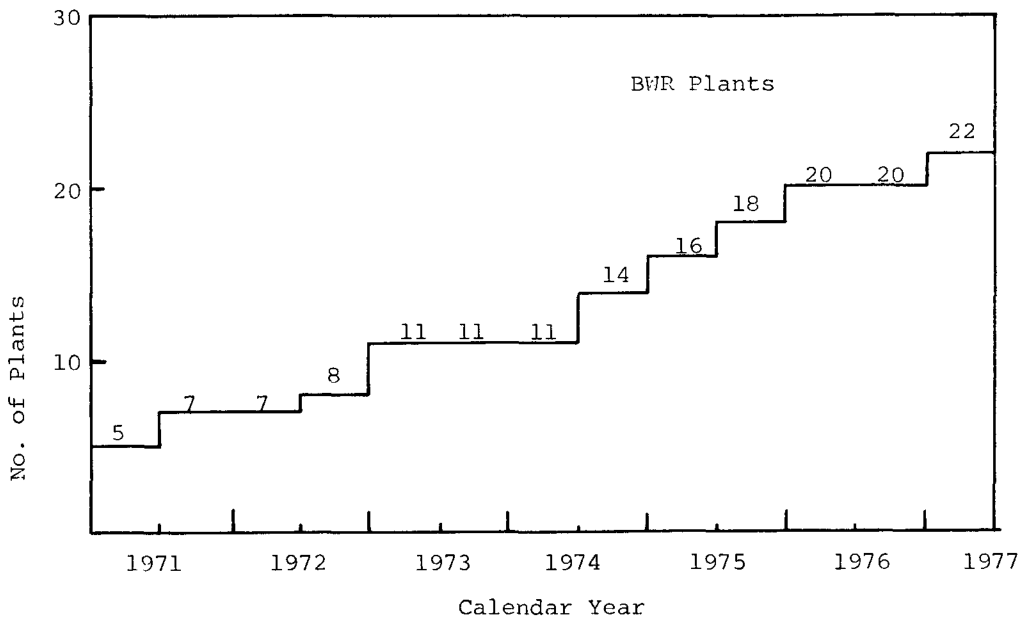


Figure 2.1. BWR Plants in Commercial Operation Versus Calendar Year

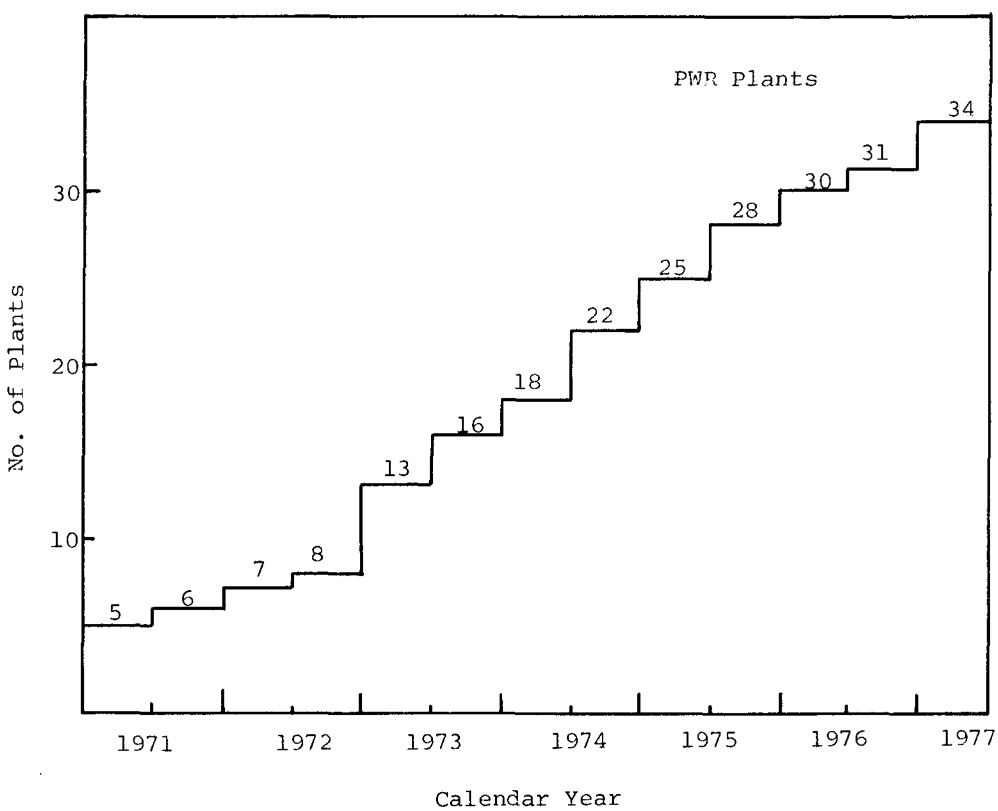


Figure 2.2. PWR Plants in Commercial Operation Versus Calendar Year.

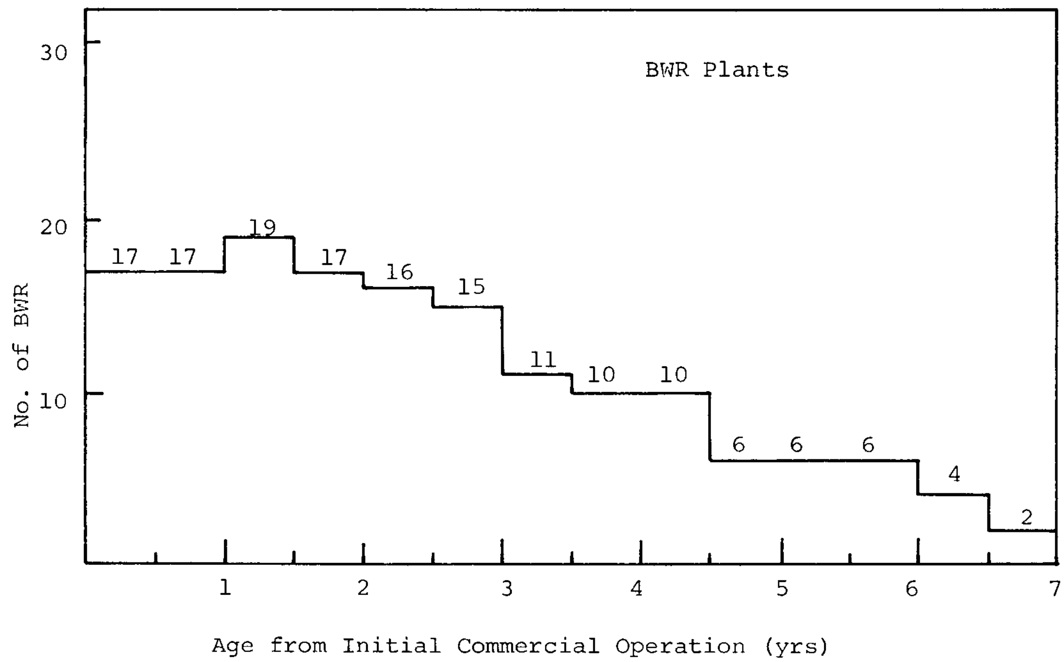


Figure 2.3. No. of BWR Plants for Which Data is Incorporated Into This Report Versus Plant Age

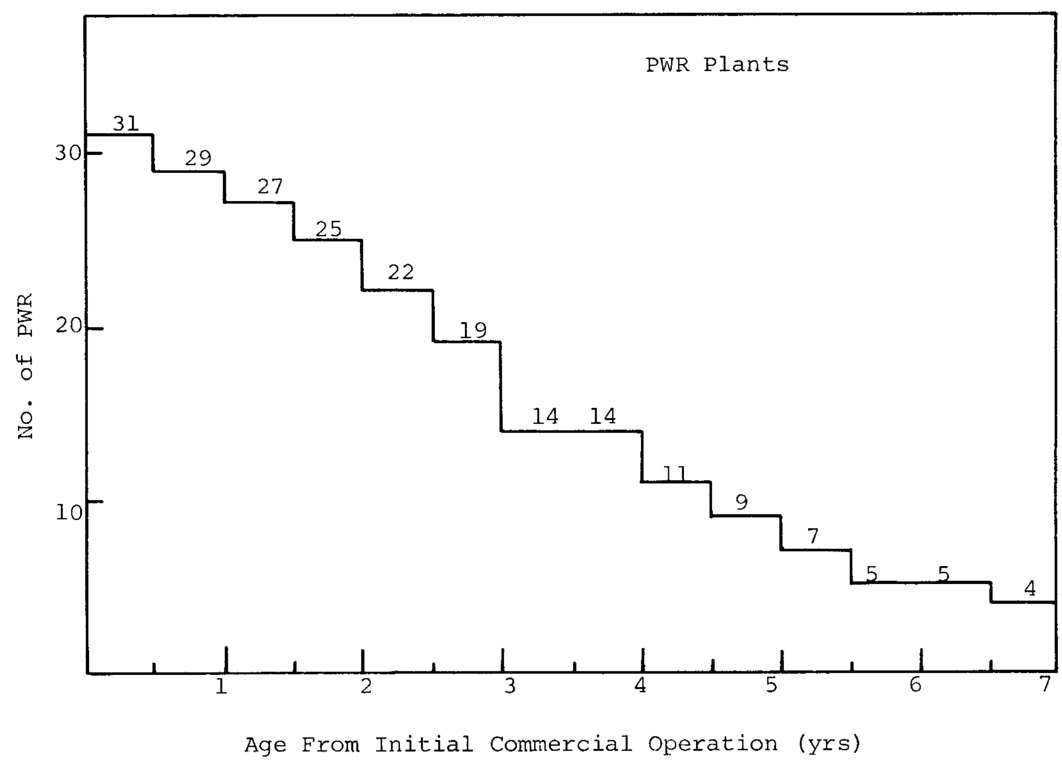


Figure 2.4. No. of PWR Plants for Which Data is Incorporated Into This Report Versus Plant Age

population which contributed data to this evaluation. These population distributions are important in the evaluation of the data since they portray the base line number of plants which are contributing to the outage hours for each operational year. Each of the plants of a given age included in the population is treated equivalently despite the variations in their size and design. (See Section 1.3).

An important note in this profile of operating plants (ages from zero to seven years) is that, in some cases, the initial seven years of data and the calendar period 1971-1977 are mutually exclusive; therefore, plants such as Yankee Rowe and Dresden 1 which do not have data in the range of one to seven years during the period of 1971 to 1977 are excluded de facto (see Section 1.3). An added note of caution is that the population contributing data in the period of five to seven years of age is relatively small; therefore outage fractions from this portion of the analysis have a larger uncertainty than in the initial four years where a larger population provides greater statistical confidence.

While most areas of the plant conveniently follow the obvious division of plants into PWRs and BWRs, one major piece of capital equipment does not lend itself to this same classification - the turbine.

As has been pointed out previously[6,7], turbines represent a significant portion of the outage time associated with nuclear plants. In addition, it has been pointed out[7] that there has been a difference in the performance of turbines manufactured by the two major suppliers, General Electric and Westinghouse. In the case of turbines, the usual division of PWR versus BWR does not tell the entire story; therefore, in the trend analysis on turbines, a separate population profile has been assembled for GE and Westinghouse turbines. Figures 2.5 and 2.6 show the population of turbines versus calendar year, and Figures 2.7 and 2.8 show the population versus plant age. As a rule of thumb, it can be said that all BWR plants contain GE turbines, while some recent PWR plants have incorporated GE turbines also.

2.2 Profile of Nuclear Plant Performance

Having established the population of nuclear plants to be considered in this report in Section 2.1, this subsection summarizes the past power plant performance for this population. With this background, Section 2.3 will show the relative impact of major outages on plant performance.

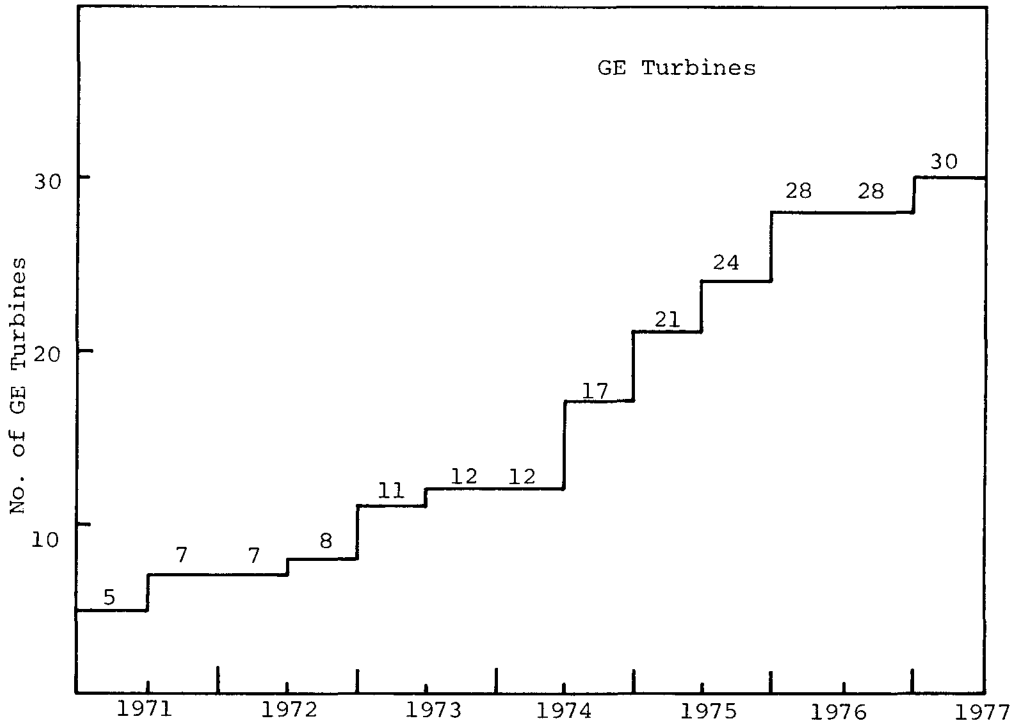


Figure 2.5. GE Turbine Population in LWRs Versus Calendar Year

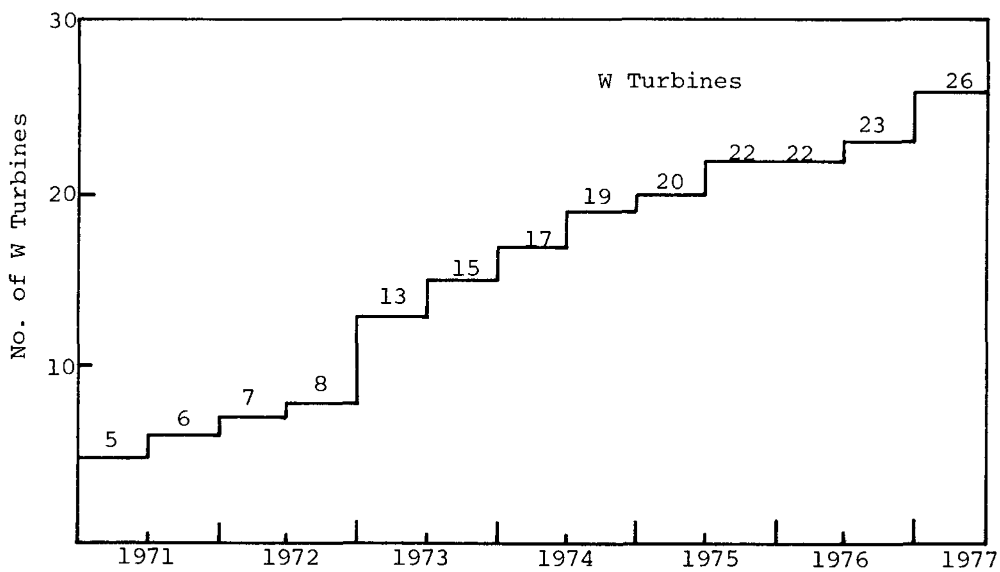


Figure 2.6 Westinghouse (W) Turbine Population in LWRs Versus Calendar Year

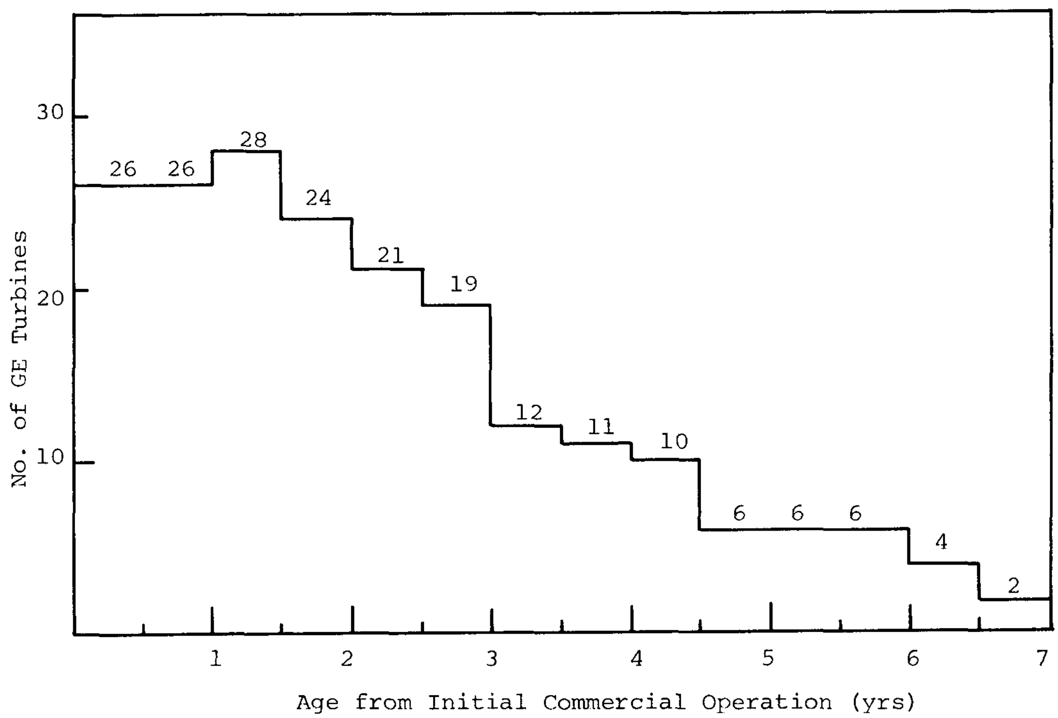


Figure 2.7 No. of GE Turbines with Data Included in This Report Versus Age From Initial Commercial Operation

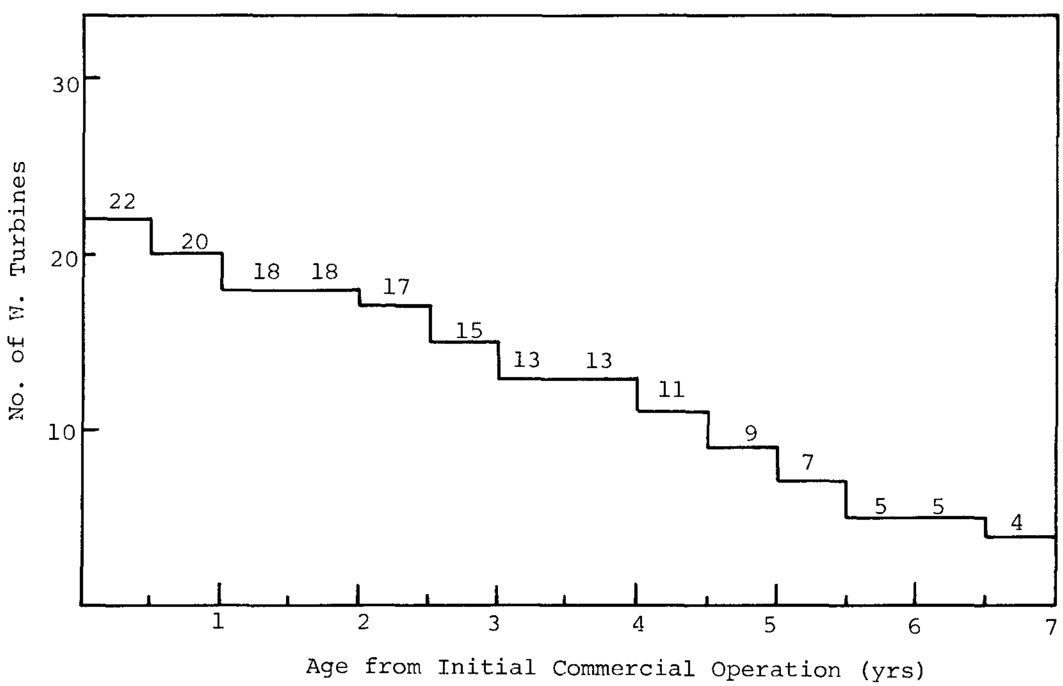


Figure 2.8 No. of W Turbines with Data Included in This Report Versus Age From Initial Commercial Operation

There are several measures of nuclear plant productivity currently in use; such as: plant availability, plant capacity factor (based upon maximum dependable capacity or design electrical rating), and forced outage rate. The capacity factor is the total amount of electricity actually produced by a unit in a year divided by the amount of electricity the unit could produce running at full capacity for the entire year.

Since the cost of nuclear generation of electricity is highly sensitive to plant availability and capacity factor, there has been a great deal of effort in the prediction of trends in these measures of plant productivity. However, there are two major problems in attempting to estimate future plant performance:

1. There are virtually no data on large, mature units, i.e., those in the 1,000 MWe range which have been operating for several years. Estimates must be made based largely on experience with units that are smaller than those now being built and that are in their second through fifth or sixth years of operation. In addition, the capacity factors are a strong function of the plant electrical capacity. The capacity factors used in this report are based on the plant design ratings. This does not account for seasonal variations due to differences in cooling water temperature, or deratings due to environmental or safety considerations.

2. Because of the diversity in plant design, size, and age, the method of averaging plant performance parameters for these different units is not clear. One approach is to weight each unit in proportion to its design rating. An alternative is to weight all units equally regardless of size. The latter method is used in this section. A less defensible method is to weight units according to the energy they actually generate; however, with this method a unit that is not operating (that has zero capacity factor) simply drops out of the calculation.

Recognizing these limitations, this subsection seeks only to crudely estimate the approximate magnitude of these plant performance parameters. Therefore, for the purposes of this summary profile, consider only those plants which have completed at least one refueling cycle. Plant productivity can be conveniently summarized with a comparison of availability and maximum dependable capacity (design) for PWR and BWR plants. Figures 2.9 and 2.10 compare the cumulative availability of PWR and BWR plants over their lifetimes. The observations are not weighted: equal weight is given to observations from young, old, large, and small plants. Note that this aggregate comparison indicates that PWR and BWR plants have approximately the same availability (~ 73%). The main focus of this report will be on the trends of the major outage contribution to plant unavailability.

Comparison of PWR and BWR Plant Availability for Which at Least One Refueling Cycle has been Completed.

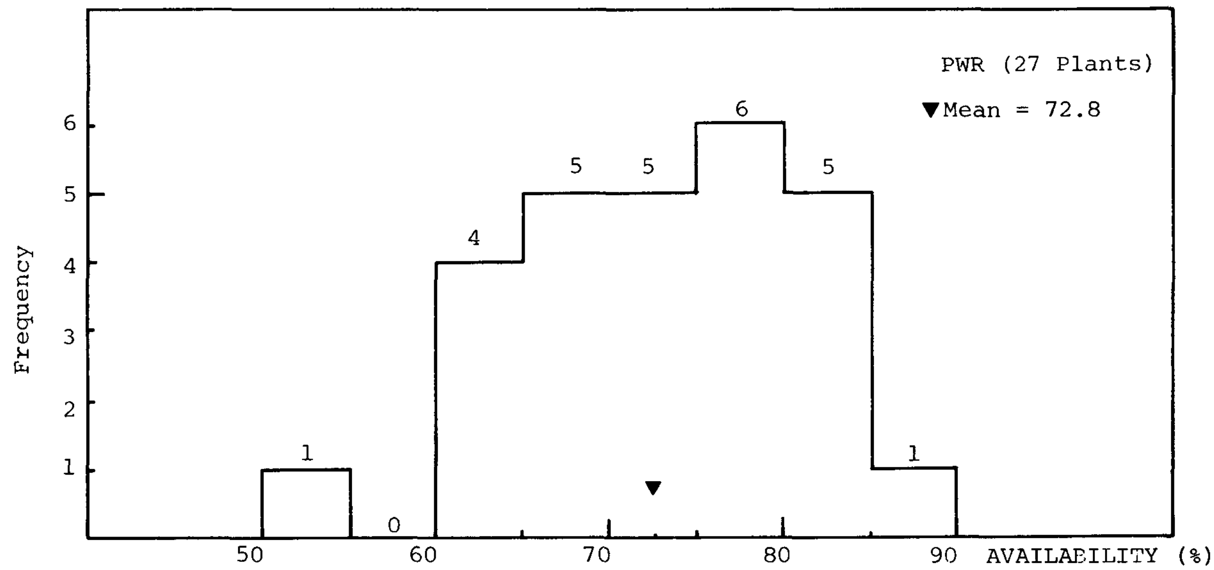


Figure 2.9. Frequency Histogram of PWR Plant Availability

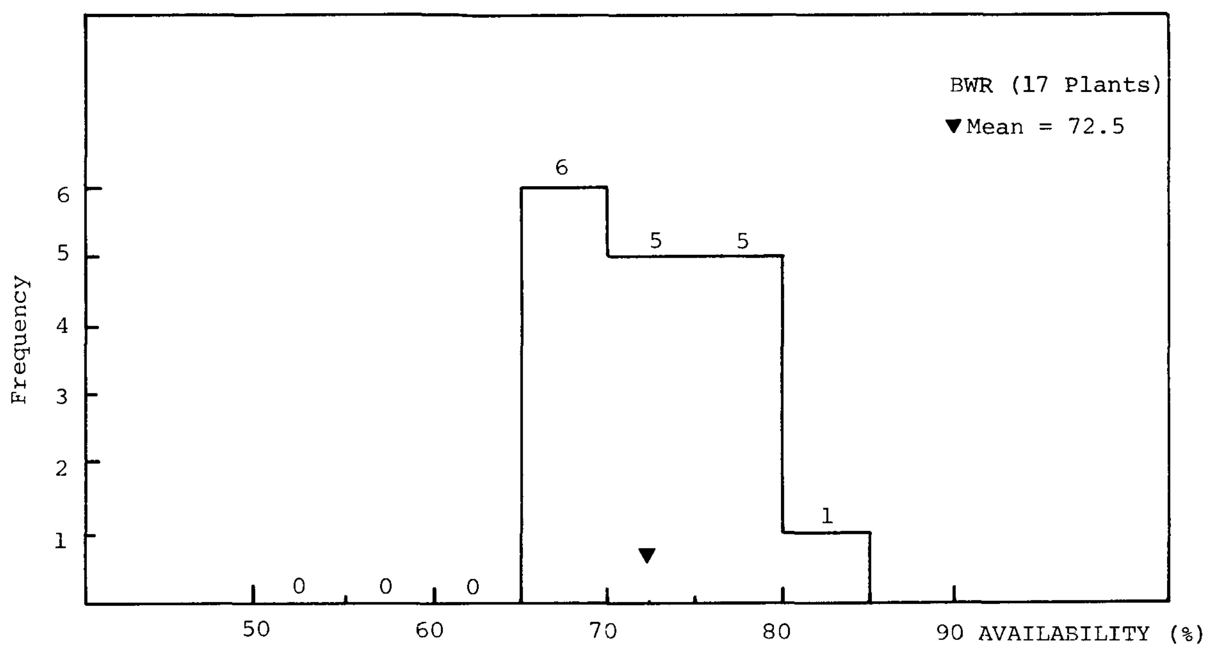


Figure 2.10. Frequency Histogram of BWR Plant Availability

Other factors also reduce plant performance. Figures 2.11 and 2.12 compare the capacity factor for the same plants as above; however, the comparison of PWR versus BWR does not exhibit the same distribution characteristics as shown for availability. Instead, it is shown that the mean BWR capacity factor is 6.6% less than that calculated for the PWR plants (remembering that they both have the same calculated availability). Appendix B discusses, in qualitative terms, the possible reasons for the reduction in capacity factor below the plant availability and why BWR plants have been more strongly affected.

2.3 Impact of Major Outages on Plant Availability

While major outages are intuitively judged to be important contributors to plant unavailability, the purpose of this section is to place the effects of major outages in more quantitative terms. Three comparisons will be made:

a) contribution to plant unavailability from major outages
b) ranking of key plant systems related to major outages
c) ranking of component types related to major outages

Previous estimates[13] of the effect of outages greater than 500 hours on a plant's capacity factor have been in the range of 5% based upon data through 1974. The present study has been expanded and updated to include: (a) outages greater than 100 hours in length; and (b) data accumulated through June 1977.

First, consider a gross comparison of the causes of plant unavailability over the three year period from May 1974 to June 1977. Over this period of time, we find that the fraction of unavailability time attributed to major outages is given in Table 2.1.

Table 2.1. Relative Contribution of Outages to Plant Unavailability From 1974 Through 1977

PERCENT OF TOTAL OUTAGE TIME				
	1974 (May-Dec)	1975 (Jan-Dec)	1976 (Jan-Dec)	1977 (Jan-June)
Refueling	42%	32%	39%	51%
Outages >100 Hrs	39%	61%	32%	28%
Outages <100 Hrs	19%	7%	29%	21%

Comparison of PWR and BWR Plant Capacity Factor for Those Plants Which Have at Least One Refueling Cycle. (Plants < 150 MWe Not Included)

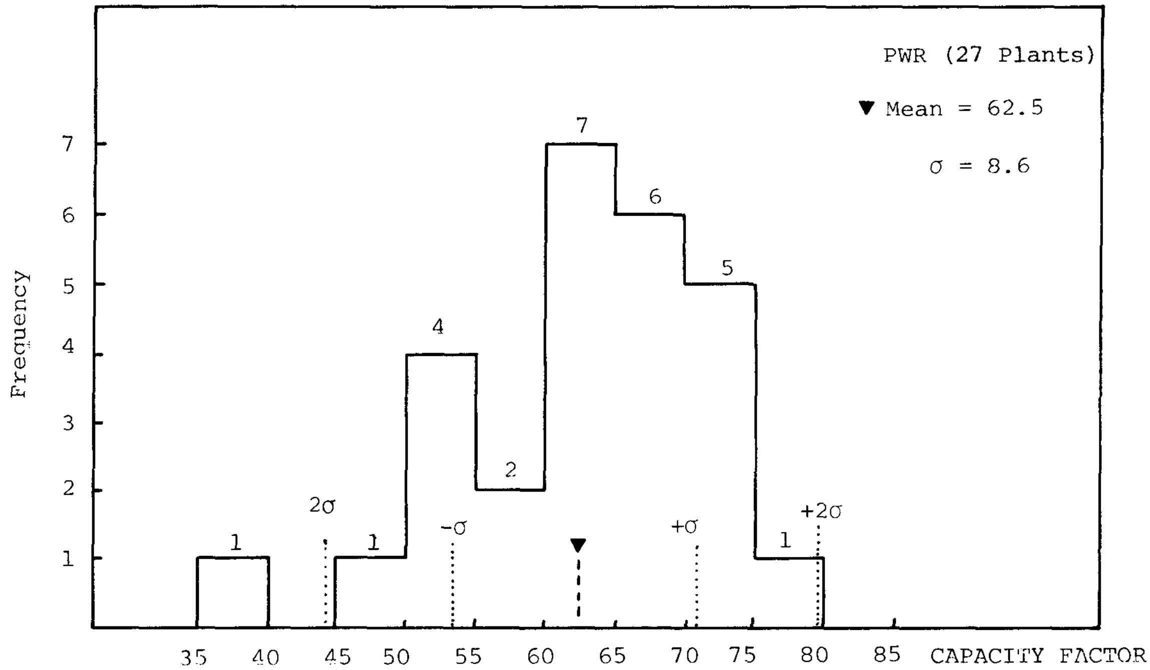


Figure 2.11. Frequency Histogram of PWR Plant Capacity Factors

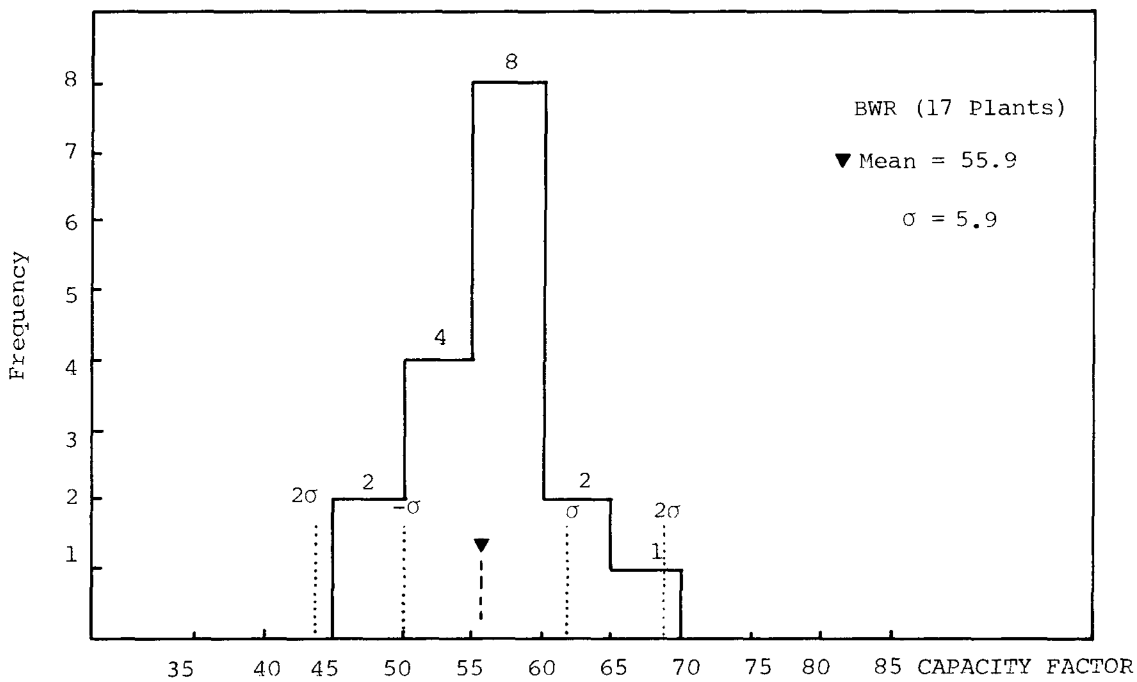


Figure 2.12. Frequency Histogram of BWR Plant Capacity Factors

On the average, refueling contributed approximately 39.5% to unavailability over this period and major outages contributed 40%. The characterization of refueling outage trends is the subject of a separate EPRI report[14]. From Section 2.2, the plant unavailability time represents approximately 27% of the reduction in capacity factor. Therefore, the major outages contribute 40% of this, or 11%, to the reduction in capacity factor. This loss in plant productivity is the target of this study.

A breakdown of the causes of all outages over this three year period is given in Appendix C and summarized in Table 2.2. A review of the data shows that the major outages are composed principally of "Equipment Failure" and "Maintenance and Test" categories. The short duration outages are associated with operator errors, administrative shutdowns, and operator training. In addition, there are frequent short duration outages related to the following:

a) pipe failures/repair[15]
b) instrumentation and control problems[16]
c) valve failure/repair[17]
d) pumps failure/repair[6]
e) condensers failure/repair[6]

A graphical summary of the breakdown of plant unavailability using this data is given in Figure 2.13.

With this quantitative measure of the impact on availability of major outages, let us now determine the systems and components which are involved in major plant outages. The operating data is insufficient to determine the root causes of the major outages; however, the identification of the components and/or systems involved in major outages will provide additional information needed by utility and designer for future decisions. First, the plant can be divided into the nuclear steam supply system (NSSS) and the balance of plant (BOP). While this division is somewhat arbitrary and in fact differs from plant to plant, it gives a general overview of where the problem areas are located. The definition used in this comparison is that all components inside of the main steam isolation valves are NSSS components. Using this convention, the outages greater than 100 hours reported over the period 1971-1977 are divided as shown in Table 2.3.

Table 2.2. Summary of Outages Over the Period May 1974 - June 1977 by Contributory Cause

Outage Categories	Outages (Unit Hours)					
	1974 (8 months)	1975 (12 months)	1976 (12 months)	1977 (6 months)	Total (38 months)	% of Total
Refueling	27,738	35,776	56,671	32,775	152,980	39.5%
Maintenance/Tests	13,943	36.138	31,655	9,138	90,857	23.5%
Equipment Failure	14,710	28,282	35,226	13,752	91,970	23.8%
Other/Multi	2,839	8,530	12,368	5,879	29,607	7.7%
Operator Error	2,110	1,817	2,645	562	7,134	1.8%
Regulatory	3,628	1,703	5,340	1,658	12,329	3.2%
Administrative	525	282	1,136	122	2,065	.5%
Operator Training	231	47	233	132	633	.2%
Avg/Month	5,477/mo	9,379/mo	12,109/mo	10,668/mo	10,199/mo	

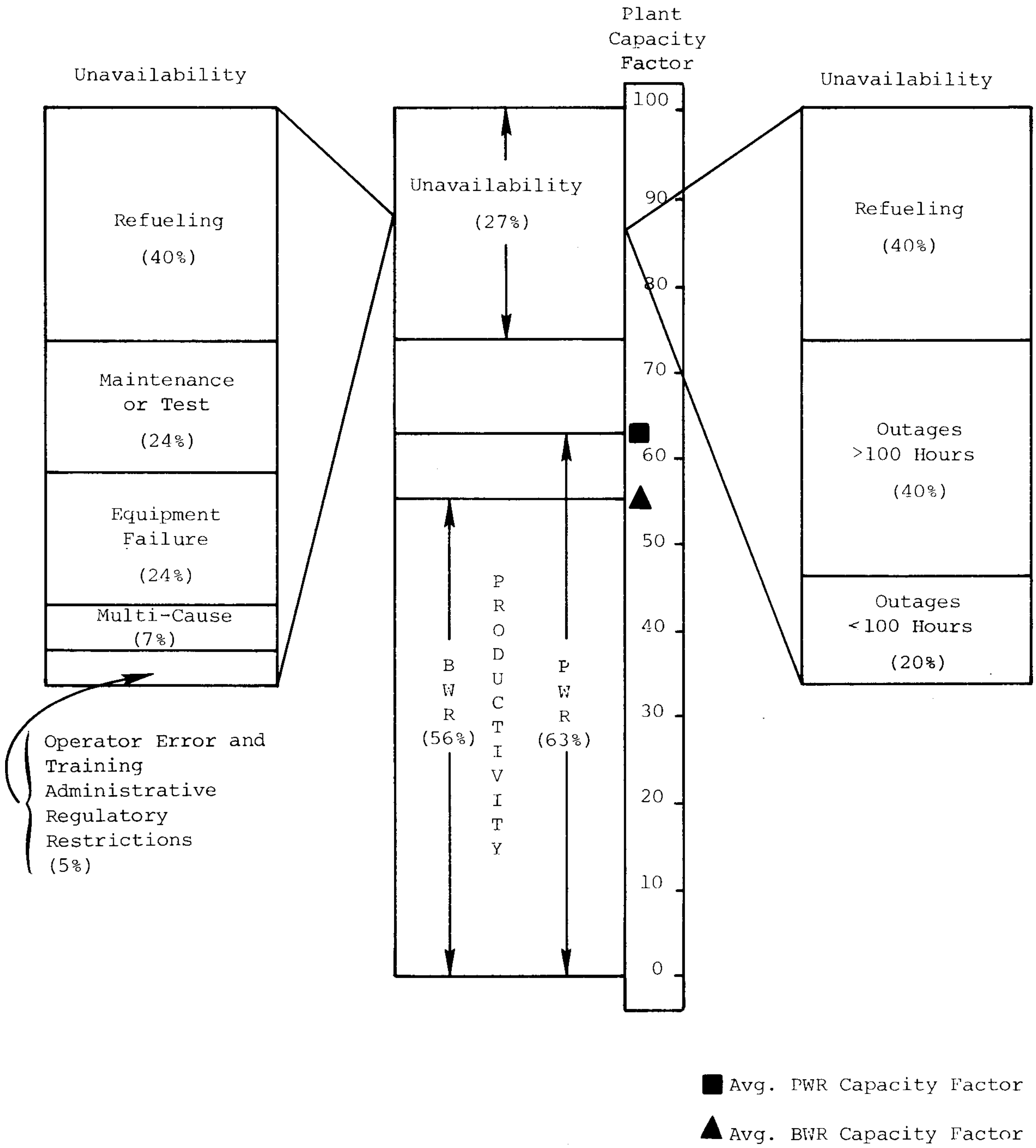


Figure 2.13. Summary of Plant Performance May 1974 - June 1977

Table 2.3 indicates that the major outage time in LWRs is approximately equally divided between NSSS and BOP related causes. Stated another way, more than one half of all major outages are related to equipment and systems located outside of containment where access for testing, monitoring, and some maintenance is possible.

Table 2.3. Comparison of All Outages >100 Hours in Duration Over the Period 1971-1977

	Outage Duration (Hrs)	% of Totals
NSSS (Nuclear Steam Supply System)	103,358	47%
BOP (Balance of Plant)	118,501	53%
TOTAL	221,859	100%

A major difficulty in the evaluation of trends in outages is the evaluation of the impact of rare occurrences. For example, the above outages include the impact of the Brown's Ferry fire, which occurred in March of 1975. That is, the outages associated with the balance of plant includes 26,160 hours of BWR outage associated with the loss of the two 1065 MWe plants for approximately 18 months. The argument can be made that these non-recurring problems should not be included in an assessment of power plant trends; however, historically it appears that some number of rare events, although of different character, do occur in large industrial operations. Therefore, the impact of the Brown's Ferry fire is included as a representative rare event which includes all phases of recovery from such an incident:

a) cleanup
b) design changes
c) repair
d) regulatory intervention/hearings
e) startup

A more detailed way of dissecting the contributions to unavailability due to outages greater than 100 hours is to perform a breakdown by major "systems". Table 2.4 is a ranking of those systems which have been shown to be the cause of outages greater than 100 hours in duration by operating experience over the time period January 1971 to June 1977.

Table 2.4. Summary of Major Outages Compared by System

Rank	System	Total Outage Duration (Hrs) for Outages >100 Hours	% of Totals*
1	Steam System (includes turbine)	42,955	18.9%
2	Steam Generators	31,586	13.9%
3	Reactor Related**	29,994	13.3%
4	Fire***	26,360	11.7%
5	Reactor Coolant Pumps	21,443	9.5%
6	Electrical Systems (includes generator)	18,973	8.4%
7	Safety Related Systems	12,023	5.3%
8	Condensate System	11,765	5.2%
9	Feedwater Systems	11,270	5.0%
10	CDM Systems	8,312	3.7%
11	Pipe Restraints/Snubbers	7,462	3.3%
12	Off-Gas System	2,192	1.0%
13	Unknown/not Specified	4,986	2.2%

*The total of outages >100 hrs duration excluding refueling

**Includes BWR feedwater sparger and core spray pipe problems

***Does not include the six month outage at San Onofre due to a cable tray fire which occurred prior to 1971

The outages cited here are associated with the systems indicated; however, the outages contributing to these totals may be related to an equipment failure, an inspection, a regulatory requirement, preventive maintenance, or some combination of these. A more in-depth understanding of the types of outages associated with each system can be obtained by a perusal of the individual summary sheets in Appendix A.

Table 2.4 is a composite summary of 6½ years of LWR experience on 56 nuclear plants. Appendix A shows that there is a wide diversity in the frequency of events of a given type at each plant. For example, the reactor coolant pump problems, which have caused significant amounts of outage and have primarily affected PWR plants, have varied from no reported incidents of outages greater than 100 hours to a large number of recurring problems over a period of years, such as those experienced by Oconee 3 and Robinson 2. Similarly, condenser tube problems have plagued some plants, such as Millstone 1 and 2, while other plants have encountered no major outages related to condenser problems. Each of the systems in Table 2.4 is mutually exclusive except for the pipe restraints/ snubber system, which is included as a separate category since it has received a great deal of regulatory and utility attention at various times in the past. However, many of the outages included under this category are also included elsewhere because of coincident work being carried out during snubber inspection and repair. The safety systems as used in the context of this report include a wide variety of equipment whose sole purpose is the safe operation of the reactor: containment, diesel generators, high and low pressure injection systems.

Perhaps a more meaningful breakdown of these outages would be a summary of the principal systems involved in major outages for PWRs versus BWRs. Table 2.5 points out the sharp distinction in systems causing major outages in the two reactor types. For PWR plants, the principal systems involved in major outages are: the steam system (including the turbine), steam generators, and reactor coolant pumps; while for BWRs the principal systems* are reactor core related problems, safety related systems, and electric systems (including the generator). None of the top three contributors are the same for PWRs and BWRs, indicating that there are significant differences in the causes and, therefore, the remedies to BWR and PWR unavailability.

Since there are fewer BWR reactor years of experience, Table 2.5 provides only a relative ranking of the contribution to major outages. A direct comparison of absolute magnitude of the outage durations for systems can be made if the numbers are normalized to approximately the same operating time (i.e., a normalizing factor of 1.4 times the BWR outage times will yield a comparable base of comparison). Section 3.2 provides a detailed discussion of the outage contributions due to each of the systems in Table 2.5 including a summary of the variation from plant to plant.

*Note that the outages due to fire are factored out of this discussion since it is judged that a fire or other rare event with high outage consequence could also occur in a PWR.

Table 2.5. Summary of Major Outages by System for PWR and BWR Plants

BWR Plants			PWR Plants		
Rank	System	Total Outage Duration (Hrs) for Major Outages	Rank	System	Total Outage Duration (Hrs) for Major Outages
1	Fire	26,160	1	Steam	38,366
2	Reactor Related	20,599	2	Steam Generators	31,375
3	Electrical	7,583	3	Reactor Coolant Pumps	16,483
4	Safety Related	7,101	4	Reactor Related	9,395
5	Steam	6,543	5	Condensate	7,862
6	Recirculation Pumps	4,960	6	Feedwater	7,638
7	Condensate	3,903	7	Safety Related	4,922
8	Feedwater	3,632	8	Pipe Restraints	4,605
9	Pipe Restraint	2,526	9	Electrical	4,411
10	Off-Gas	2,192			

The types of outages which lead to long duration outages (i.e., greater than 100 hours) can also be classified according to the type of component involved in the outage (see Table 2.6).

Table 2.6. Summary of Major Outages by Component Type

Component Types	Total Outage Duration (Hrs) Outages >100 hrs	% of Total
Turbine-Generators	42,046	18.5%
Steam Generators	31,586	13.9%
Pumps	22,343	9.9%
Reactor Core	22,198	9.8%
Valves	18,273	8.1%
Pipe*	16,077	7.1%
Condensers	11,765	5.2%

*Includes BWR core spray, recirculation, and feedwater sparger

SECTION 3.0

TRENDS IN MAJOR OUTAGES IN LWRS

The amount of LWR operating experience has increased quite rapidly in recent years: the number of years of operating experience increased by approximately 30% in 1975 and an additional 28% in 1976. Therefore, it appears fruitful at this point to determine whether a trend can be isolated in the complex operations of a nuclear power plant. This section focuses on trending major outages in three separate ways:

a) Overall trend for all major outages, plus a comparison of PWR and BWR plants (Section 3.1)

b) System trends (Section 3.2)

c) Component trends (Section 3.3)

If a trend can be established, one of the key elements in the planning sequence will be achieved - the identification of the problem, its magnitude, and its anticipated variation with time. As is evident from a review of the data, there is a wide diversity in the types of outages occurring; however, there are certain classes of high impact outages which can be isolated and, to a large extent, either prevented from occurring or adequate preparation made to minimize impact on plant performance. A known trend at one plant can aid management decisions at other utilites in the preparation of procedures, identification of sources of spare parts, and the training of personnel.

3.1 Overall Outage Trends as a Function of Plant Age: Comparison of PWR Versus BWR Plants

A continuing theme in the literature has been the call from both industry and utility management personnel [19,20,21,22,23] for additional planning and preparation for major outages, including collection of data and the application to special maintenance tasks. It is recognized that in order to minimize forced outage time, a comprehensive program of preventive maintenance must be incorporated into a work package. To establish a successful preventive maintenance program, utility management and engineering personnel must be informed of the outage trends of similar equipment throughout the industry. This section takes a broad-brush look at the overall trends in major outages. More specific information which may be required by utility planners and vendor design engineers to identify trends in specific types of equipment is provided in Section 3.2 and 3.3. Of course, this type of information can only alert the utility and designer to potential problem areas. For specific failure mechanisms and times to failure additional information is required which is only available through the collection, evaluation and sorting of detailed operating experience data.

The LWR operating experience outage data for the six and a half year period 1971-1977 includes 480 major outages. These outages are individually tabulated by plant in Appendix A. A yearly summary of the data is provided in Tables 3.1 through 3.3 on the basis of calendar year and plant age. The number of plant years in each analyzed category is calculated from the data in Section 2.1. Using this as the population base and using the major outage hours, the average outage time per plant year can be calculated. This is the parameter which is used in the remainder of the report to characterize potential trends. The average length of an outage is also calculated; however, the variation of this parameter is not as well interpreted and therefore has not been emphasized in this analysis.

Figure 3.1 summarizes the overall outage trend for U.S. LWRs during their initial seven years of commercial operation on a per plant year basis. The trend confirms a portion of previous predictions from the electric power industry[4] that as nuclear power plants mature, their productivity (availability) will increase. Clearly, however, this trend in major outages is only a segment of the plant productivity picture, which also includes refueling outages, short duration outages, and power restrictions. In the case of the major outage trends, there is a distinctive maturation trend for LWRs. After the second year of commercial operation, the major outage time required on a per plant basis is dramatically decreased to an approximately constant level. This indicates that after a high initial outage rate due to special "start-up" or "break-in" problems, nuclear plants settle into a constant background level of major outages. It must be understood

Table 3.1. Outage Data for BWR Plants for Outages >100 Hrs. in Duration (Jan. 1971 through June 1977)

BWR	Calendar Year						
	71	72	73	74	75	76	77
No. of Plant Years	6	7.5	11	12.5	17	20	22
No. of Outages**	19	19	32	33	51	41	30*
Total Outage** Time (Hrs.)	6266	4278	8418	12,318	40,854	13,219	9,286*
Length of Outage (Hrs)/Outage/Yr	330	225	263	373	801	322	309
Length of Outage (Hrs)/Plant/Yr	1044	570	765	985	2403	611	422

BWR	Age of Plant From Initial Commercial Operation (Years)						
	1	2	3	4	5	6	7
No. of Plant Years	17	18	15.5	10.5	8	6	3
No. of Outages**	69	43	43	22	16	16	9
Total Outage** Time (Hrs)	47,299	16,894	12,470	5,533	3,442	2,971	1,534
Length of Outage (Hrs)/Outage/Yr	685	393	290	252	215	186	171
Length of Outage (Hrs)/Plant/Yr	2782	938	804	527	430	495	511

*Projected based upon six months of data

**Only major outages (outages >100 hrs.) are included

Table 3.2. Outage Data for PWR Plants for Outage >100 Hrs. Duration (Jan. 1971 through June 1977)

PWR	Calendar Year						
	71	72	73	74	75	76	77
No. of Plant Years	5	7	13	18	25	30	34
No. of Outages**	7	14	36	44	68	87	72*
Total Outage Time (Hrs)**	4196	8891	28,693	26,609	27,604	33,926	26,650*
Length of Outage (Hrs)/Outage/Yr	599	635	929	605	406	390	370
Length of Outage (Hrs)/Plant/Yr	839	1270	2207	1478	1104	1131	784

PWR	Age of Plant From Initial Commercial Operation (Years)						
	1	2	3	4	5	6	7
No. of Plant Years	30	26	20.5	14	10	6	4.5
No. of Outages**	126	57	29	33	23	5	7
Total Outage Time (Hrs)**	61,332	39,047	11,270	10,510	6,577	4,128	3,057
Length of Outage (Hrs)/Outage/Yr	487	685	389	318	286	829	436
Length of Outage (Hrs)/Plant/Yr	2044	1501	550	751	657	691	671

*Projected based upon six months of data

**Only major outages (outages >100 hrs.) are included

Table 3.3. Outage Data for all LWR Plants for Outages >100 Hrs. Duration (Jan. 1971 through June 1977)

LWR Total	Calendar Year						
	71	72	73	74	75	76	77
No. of Plant Years	11	14.5	24	30.5	42	50	56
No. of Outages**	26	33	68	77	119	128	102*
Total Outage** Time (Hrs)	10,462	13,169	37,111	38,927	68,458	47,145	35,936*
Length of Outage (Hrs)/Outage/Yr	402	399	546	506	575	368	352
Length of Outage (Hrs)/Plant/Yr	951	908	1546	1276	1630	943	542

LWR Total	Age of Plant From Initial Commercial Operation (Years)						
	1	2	3	4	5	6	7
No. of Plant Years	47	44	36	24.5	18	12	7.5
No. of Outages**	195	100	72	55	9	21	16
Total Outage** Time (Hrs)	108,631	55,941	23,760	16,043	10,019	7,099	4,591
Length of Outage (Hrs)/Outage/Yr	557	559	330	292	257	338	287
Length of Outage (Hrs)/Plant/Yr	2311	1271	660	655	557	592	612

*Projected based upon six months of data

**Only major outages (outages>100 hrs.) are included

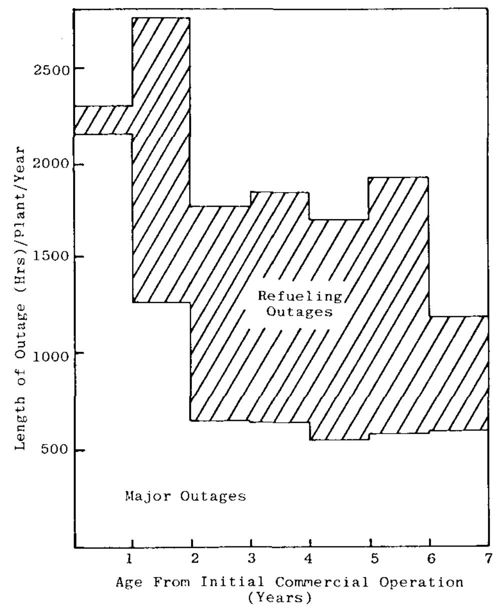


Figure 3.1a Trend in Major Outages* and Refueling Outages for LWRs

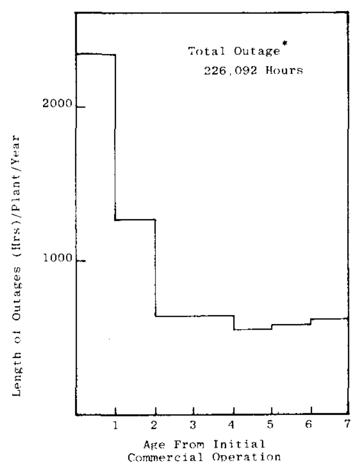


Figure 3.1b Trend in Major Outages* for LWRs

*Only non-refueling outages >100 hours are included here.

in interpreting these trends that the calculated outage rates for the sixth and seventh years of operation have greater uncertainty bands on the calculations since the number of data points (plants operating) is substantially less than during the initial years.

In order to emphasize the fact that major outages represent only one contributor to reduced plant availability, note that during the first year of commercial operation, generally less than 5% of the plants have a refueling outage. During the second year of commercial operation approximately 75% of the plants undergo a refueling outage. In subsequent years, 80-90% of the plants undergo one refueling each year. Since refueling outages have historically taken an average of 2.3 months, they represent a significant impact upon plant availability. Therefore, an accurate representation of trends in overall plant availability must account for both of these effects. Figure 3.1a shows that if refueling outages are included, the first two years of operation still represent the years with the highest outage rate per plant.

Notwithstanding this caution in the use of the major trends, the distribution of major outages represent one important contributor to the understanding of LWR performance. The understanding of each contributing cause of plant availability will lead to better planning for load requirements, improved maintenance preparation, and possible changes in the equipment design or arrangement.

To determine whether there are any distinctive trends for PWR and BWR outages, the overall trend of major outages from Figure 3.1 is divided into trends for PWR and BWR plants. Figure 3.2 shows the variation in the average outage time per plant year from initial commercial operation through seven years. BWRs and PWRs both exhibit the same tendency for high outage rates on a per plant basis during the initial two years of plant operation. However, in Section 3.2 and 3.3, it is pointed out that the causes of this high initial outage rate are different for PWRs and BWRs. Figure 3.3 is a display of the average length of an outage as a function of plant age. The variation of BWR average outages is a monotonically decreasing function, suggesting that as BWR plants mature, the length of each major outage is decreasing. However, for PWR plants this trend does not hold. Rather, it appears that while the total length of major PWR outages on a per plant basis is constant in the three to seven year period (see Figure 3.2), the average length of PWR outages (see Figure 3.3) is more variable. That is, there are indications that the PWR major outages may be less frequent but of longer duration in the sixth and seventh years of operation.

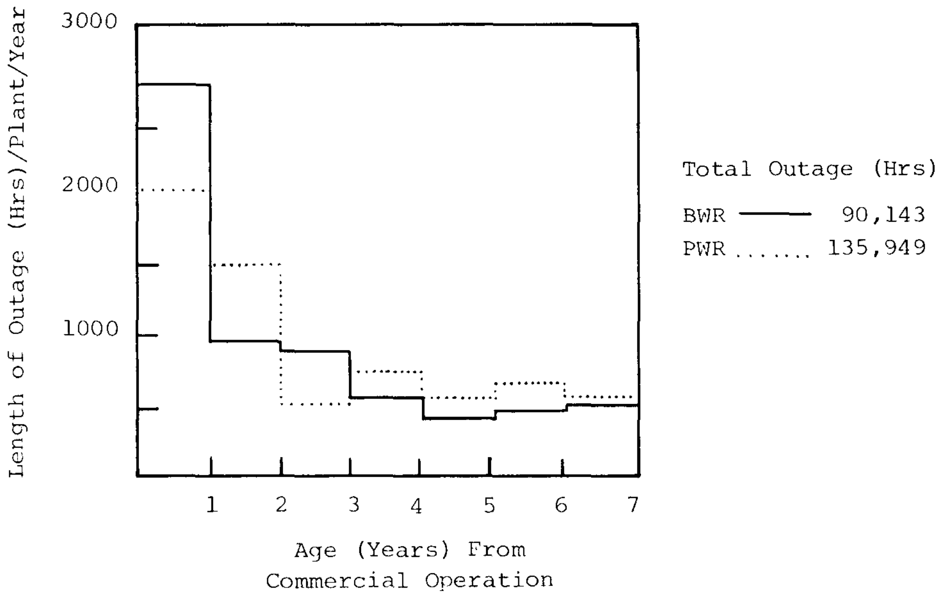


Figure 3.2 Average Yearly Major Outage Time Per Plant as a Function of Plant Age

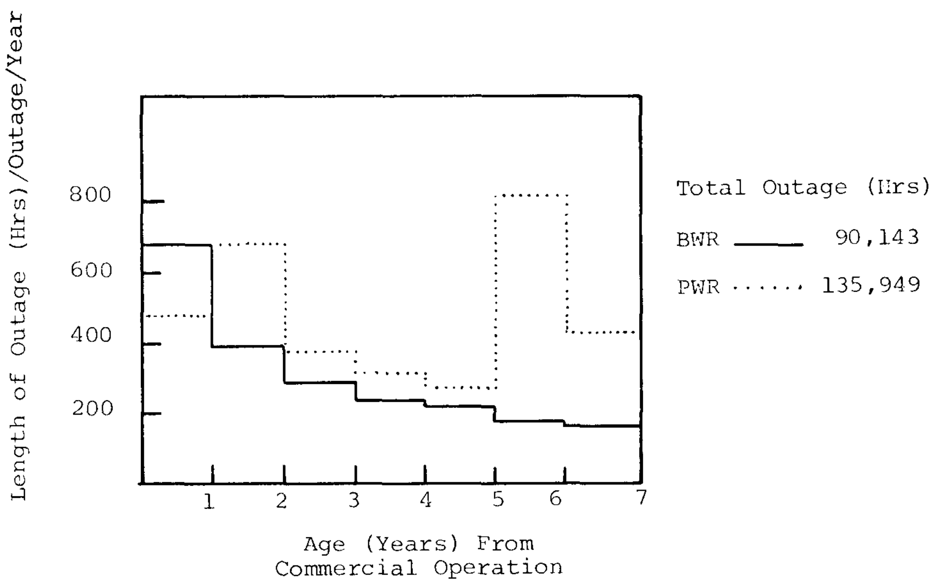


Figure 3.3 Average Length of Major Outages Per Outage Versus Plant Age From Initial Commercial Operation

Figures 3.4 and 3.5 are included for completeness. They display the length of outages per plant and per outage as a function of calendar year. The peaks in 1973 (PWR's) and 1975 (BWR's) are due mainly to anomalies in the data which have no particular significance. These years saw a high influx of new plants with the expected higher than average outage rates during first year operation. Also, a number of abnormally long duration outages occurred during these years such as the Brown's Ferry fire and the Palisades reactor internals outage.

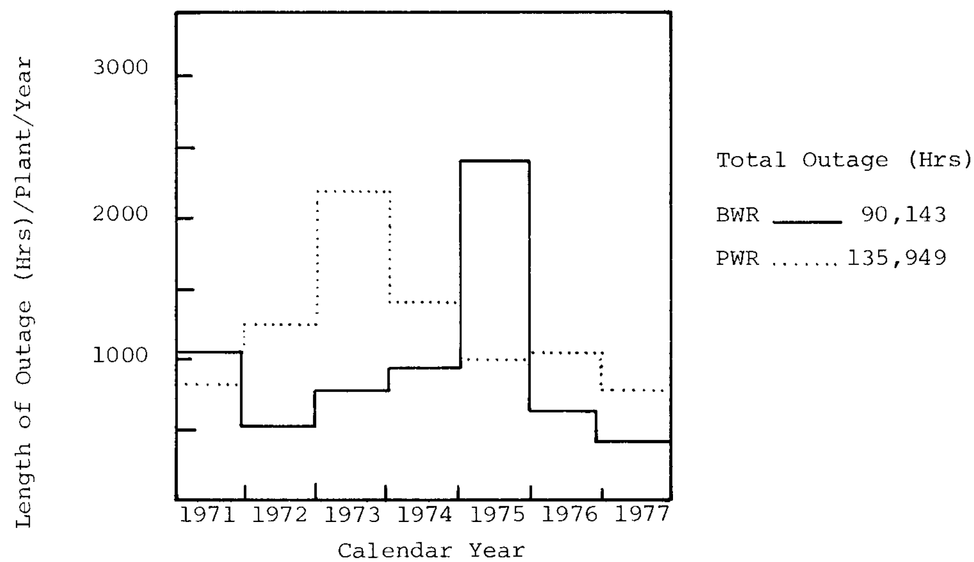


Figure 3.4 Average Major Outage Time Per Plant as a Function of Calendar Year

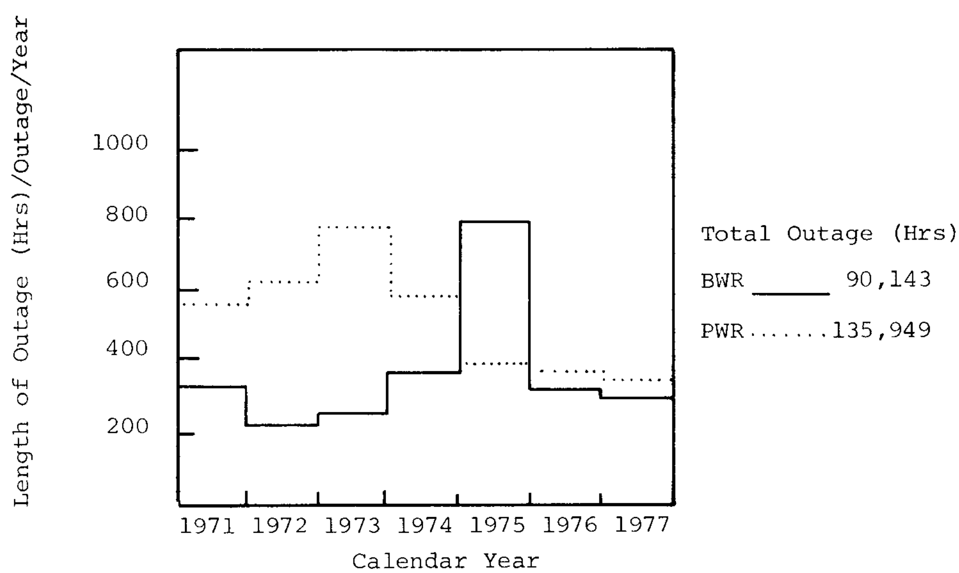


Figure 3.5 Average Length of Major Outages Per Outage Versus Calendar Year

However, before proceeding to the discussion of key systems and components, consider another method of displaying variations in major outages. It has been suggested[4] that one measure of plant maturity is the number of refuelings a plant has undergone. Figure 3.6 displays the average number of major outages per plant versus the number of the refueling cycle. Again, it is apparent that during the first fuel cycle there are a large number of problems. Following this period, a substantial decrease in the number of outages can be seen. However, it appears that there remains a constant number of approximately two major outages per plant during each refueling cycle.

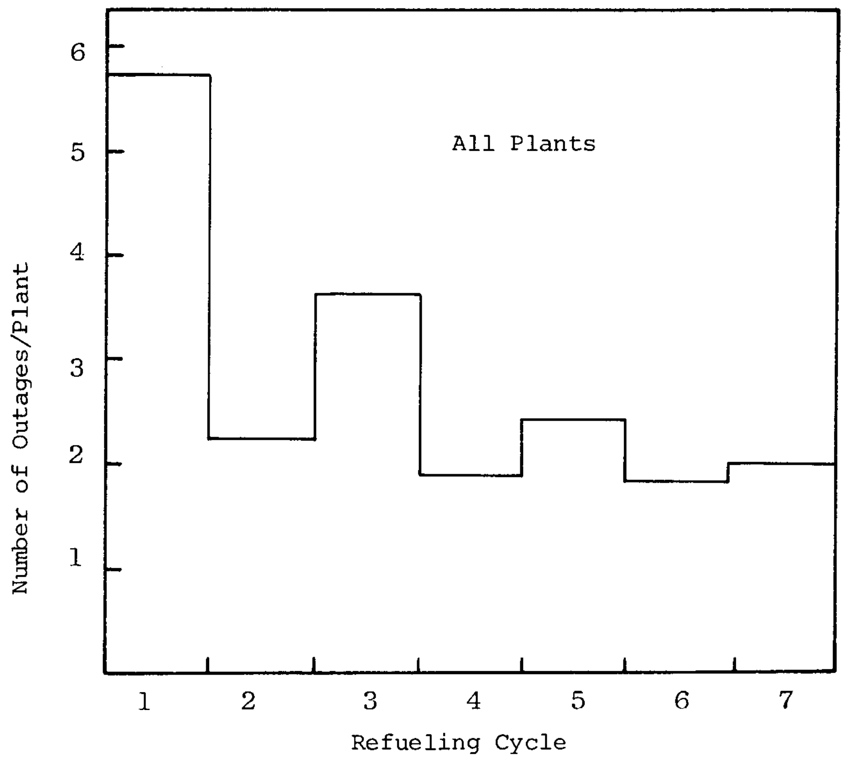


Figure 3.6. Distribution of the Number of Major Outage Occurrences Between Each Refueling

3.2 Outage Trends in Key LWR Systems

This section provides a perspective on the systems contributing to the major outages of LWRs. The system divisions used in this report are arbitrary, but are structured to present a clear picture of the principal systems involved in plant outages.

- Primary System (3.2.1)
- Steam System (3.2.2)
- Feedwater/Condensate Systems (3.2.3)
- Safety Related Systems (3.2.4)
- Electrical Distribution and Generation Systems (3.2.5)
- Fires Affecting Systems (3.2.6)

This section is an overview of the outage trends based upon systems. A more detailed evaluation of the causes of the trends is given in Section 3.3, which discusses each major component.

The components and subsystems of nuclear units are composed of many parts, and they exhibit different types of malfunctions at varying frequencies; however, a highly subdivided failure rate study of individual components may lack sufficient confidence levels to be meaningful. Since the malfunction rates for most nuclear generation equipment are quite low, statistical confidence accumulates only slowly with operating experience. Nevertheless, a summary of major outage trends by system has sufficient numbers of incidents to be statistically meaningful, although it does lack the degree of detail which many designers would like. It is hoped that an assessment of the major outages broken down by system will provide a basis for the "target" reliability requirements for components within specific systems. The literature indicates a growing interest in the methods of reliability engineering for increasing plant availability. This section is tailored to present the LWR operating experience in a way which will clarify those areas of nuclear plants which are involved in major outages. The pinpointing of the problem areas in LWRs is the first step in the reliability analysis by designers, architect engineers, and utilities to increase plant availability. The next steps involve such things as:

a) Introducing redundant back up systems to chronic problem systems in order to avoid outage time.

b) Modification to plant arrangements to allow a system to be maintained with a plant at power.

c) Optimizing space allowances for maintenance effort.

d) Upgrading or "over designing" equipment which has shown chronic patterns of failure in the past.

3.2.1 Outage Trends in Primary Systems

For the purposes of this study we shall define the following components as part of the primary system:

- Reactor
- Control Rod Drive Mechanisms (CRDMs)
- Main Coolant Pumps (PWR only)
- Recirculation Pumps (BWR only)
- Steam Generators (PWR only)

The primary system as defined here is the source of approximately 41.1% of the total outage time included in this study. Figure 3.7 shows the variation of the major outage contribution versus plant age. Note that the reactor in-core related outages dominate the second year of commercial operation. This is a notable difference from the overall trend which indicates that the first year of operation incurs a large percentage of the major outage problems. The majority of the in-core problems are generic in nature due to a design problem with the mode of failure related to core vibration which results from flow induced vibration (see Section 3.3.6). In-core problems appear to have a gestation period of more than one year before they are determined to be a problem.

The steam generator outages (PWRs only) represent a large percentage of the outage time beyond the third year of operation. The failure mechanism of steam generator tubes is usually some form of corrosion attack (e.g., wastage, denting, stress corrosion cracking, pitting). The tube corrosion characteristically has an exposure period of a few years before the inception of tube failures. Therefore, steam generator outages can be expected to occur after a few years of plant operation and may contribute substantially to outages occurring in the later years of plant operation (see Section 3.3.2.).

Reactor coolant pumps and control rod drive mechanisms (not shown) incur the largest percentage of major outages during the first two years of commercial operation. In subsequent years, repairs are generally incorporated into scheduled refuelings and therefore are not reflected in Figure 3.7. After the initial two years of operation, there are still a few plants which have recurring reactor coolant pump seal and motor problems for a longer period of time, and these few plants are the source of the major outage time shown in Figure 3.7 for the reactor coolant pumps in the third through the seventh years of commercial operation.

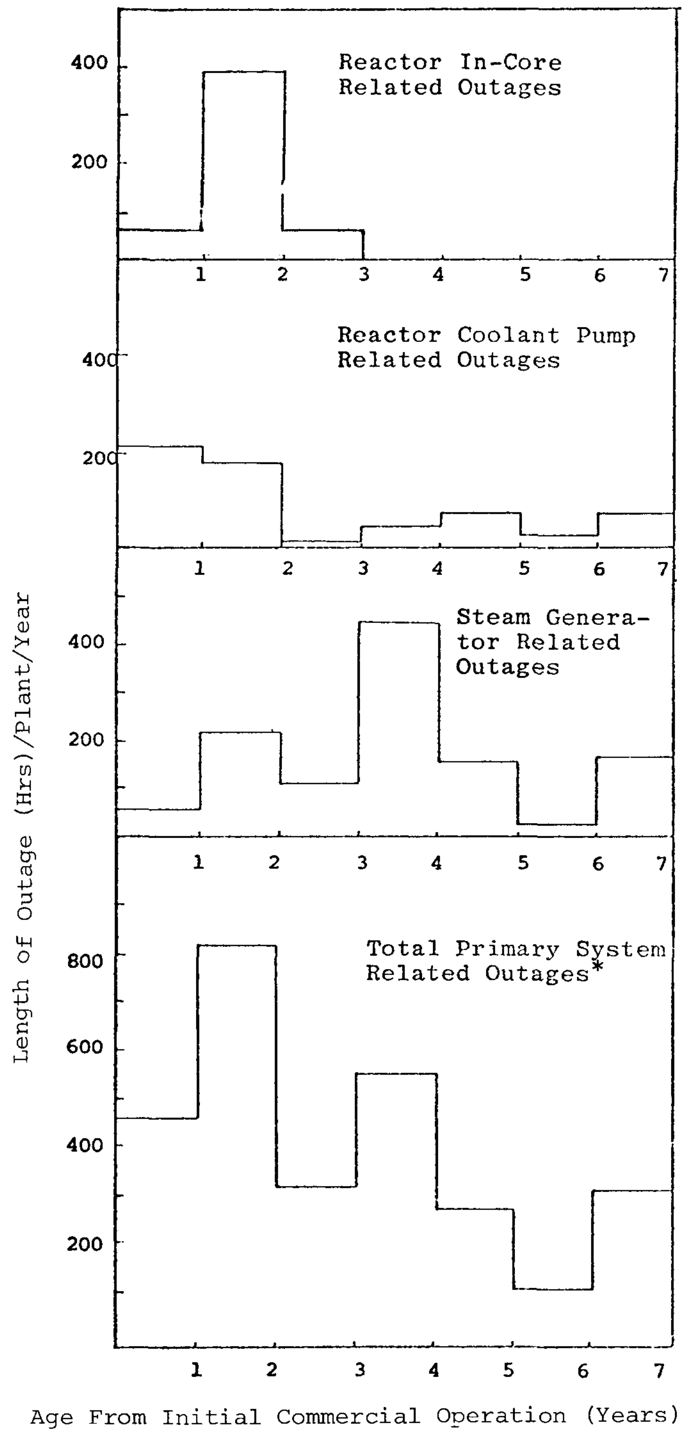


Figure 3.7. Major Outages Related to the Primary System as a Function of Commercial Age

*Includes CRDM which is not presented separately.

For the primary system, the overall variation in major outages on a per plant basis appears to decrease with increased plant age. However, certain contributors, i.e., reactor coolant pumps and steam generators, show signs of causing an increasing amount of outage time in later years of operation. For example, in the fourth year of operation, steam generators contributed over 50% of the total outage time. This was reduced to about 30% in the seventh year but still represents an inordinate proportion of lost productivity. The problem is quite complex. For steam generators, the solution may require extensive design modifications which have long lead times before impacting on plant performance. Therefore, it can be anticipated that steam generators may represent an increasing share of the necessary major outages in PWRs, especially in the later years of operation.

3.2.2 Outage Trends in Steam Systems

This section includes the major outages associated with the following systems:

- Steam Piping and Valves
- Steam Turbine

The steam system accounts for approximately 18.9% of the overall major outage time included in this study. Figure 3.8 shows the variation of the average major outage contribution as a function of plant age. The dominant contributor is the turbine related outages (see Section 3.3.1). The initial year of commercial operation contains the major share of the turbine related outage time (approximately 62%). Over the first five years of operation, the outage contribution decreases dramatically; however, in the sixth and seventh years there is an apparent reoccurrence of the turbine problems. The failure trend in later years is particularly disturbing since it indicates that turbine failures may be a continuing problem in "mature" plants as well as the new plants; in fact, over 50% of the outage time in the seventh year is accounted for by turbine outages.

A probably unexpected variation in major outages is the decreasing trend of steam system problems (including piping and valve problems) over the period covered by this study. A cautionary note is that steam system problems

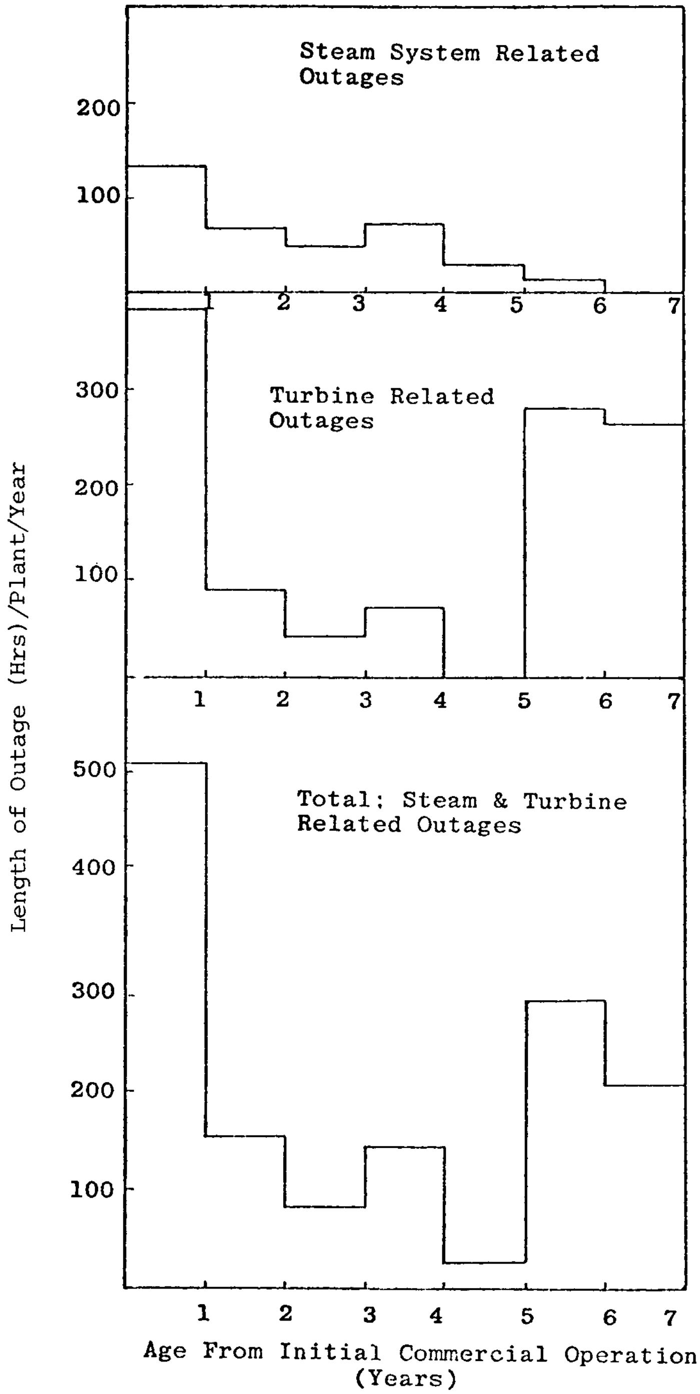


Figure 3.8. Major Outages of the Steam System Versus Plant Age

still occur, but they do not cause major outages and are remedied either during short duration outages or during refueling outages.

3.2.3. Trends in Outages Related to Feedwater and Condensate Systems

In this section the major outages related to the following systems are considered:

- Feedwater System
- Condensate System

These two systems account for approximately 10% of the major outage time included in the analysis, so it is important to understand whether the tendency for these outages is increasing or decreasing as plants mature.

In general, the PWR system is constructed in such a way as to allow secondary plant maintenance during reactor operation. However, similar repairs on a BWR normally must wait until shutdown because of the high radiation levels. This factor is offset in the overall maintenance picture since BWRs have fewer components to be maintained.

Each system contains valves, piping, heat exchanger tubing, and pumps. In Section 3.3 the systems are broken down and major outage trends are summarized by component, while this section emphasizes the systems. Understanding the trends for a composite system may lead to some inferences concerning the effects of operating environment, the number of demand cycles, or other factors on total system operation. Figure 3.9 compares PWR and BWR feedwater system major outage variations as a function of plant age. It should not be surprising that the variations with plant age for PWR and BWR systems are appreciably different since the systems themselves are designed to operate quite differently in terms of pressure, temperature, and demand cycles. Figure 3.9 indicates that PWR major feedwater system outages affect plant productivity early in plant life and that their effect rapidly decays after the first year. However, BWR outages related to the feedwater system occur at a constant rate throughout the first six years and may in fact show a tendency to increase in the seventh year. This, however, is a tentative conclusion based only on limited data for the seventh year and therefore has a large statistical uncertainty. The longer term BWR feedwater problems are primarily feedwater valve failures.

The PWR feedwater problems during the initial year of commercial operation involve such things as:

a) Failure of feedwater piping and associated modifications at Indian Point 2 and Beaver Valley

b) Water hammer causing pipe failure at Calvert Cliffs

c) Repair of feedwater pump turbine at Oconee

The four high impact items occurred at Indian Point and Beaver Valley.

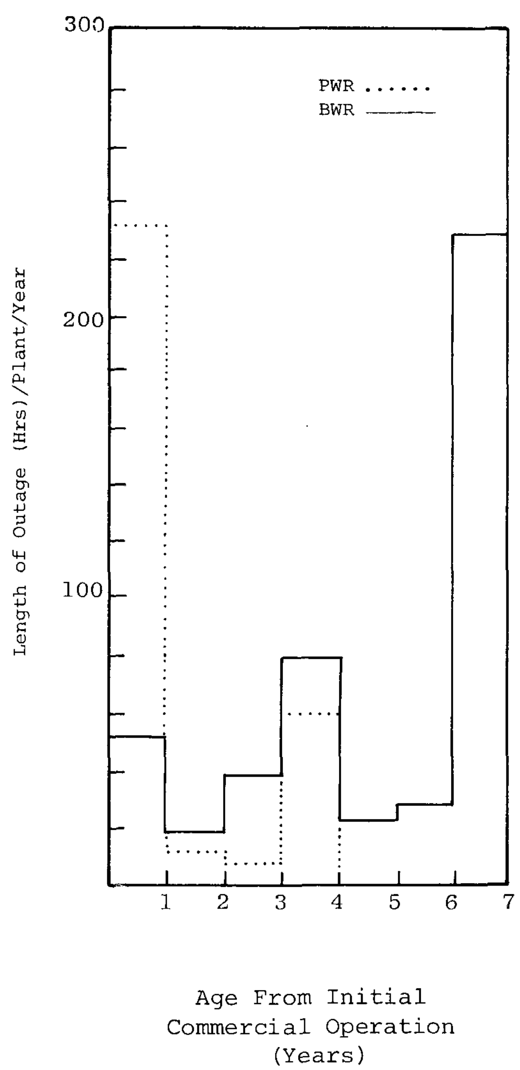


Figure 3.9. Comparison of PWR and BWR Feedwater System Major Outage Trends Versus Plant Age

Figure 3.10 is a composite of the feedwater system trends for both PWR and BWR systems and a summary of the condensate system related outages. It is difficult to pin-point any dominant trend in the condensate or combined feedwater-condensate system. The principal conclusions are as follows:

a) Feedwater systems in BWR and PWR plants have markedly different variations with plant age

b) Condensate systems (see Section 3.3.5) have an approximately constant outage rate per plant versus plant age indicating a potential continuing problem similar to steam generators which may continue to be a source of major outages in "mature" plants.

3.2.4 Outage Trends for Safety-Related Systems

For convenience in nomenclature, we classify the following components and systems under the general category, safety-related systems:

- Emergency Core Cooling System (ECCS) (high and low pressure injection systems)
- Containment Systems
- Emergency Power Sources (Diesels and Gas Turbines)
- Pipe Restraints and Snubbers

These four safety-related systems combine to contribute approximately 9% of the total major outage time involved in this analysis. Each of these systems has a safety-related function to perform which is designed to prevent or mitigate the consequences of an accident. For example, the pipe snubbers are designed to limit pipe motion and maintain pipe integrity in case of a seismic event, a potential pipe whip, or other transients. The ECCS is required to cool the core under conditions such as loss of coolant (LOCA) or loss of power. The containment systems serve to contain radiation in the event of a nuclear accident. Although the safety-related systems may not be required for normal operation, their unavailability, their need for modification or upgrading, and their requirements for surveillance testing have contributed to the reduction in overall plant performance.

Figure 3.11 portrays the variation in major outages for safety-related systems. Other than the distinctive peak in outages during the initial year of commercial operation, the dominant trend in outages due to safety-related

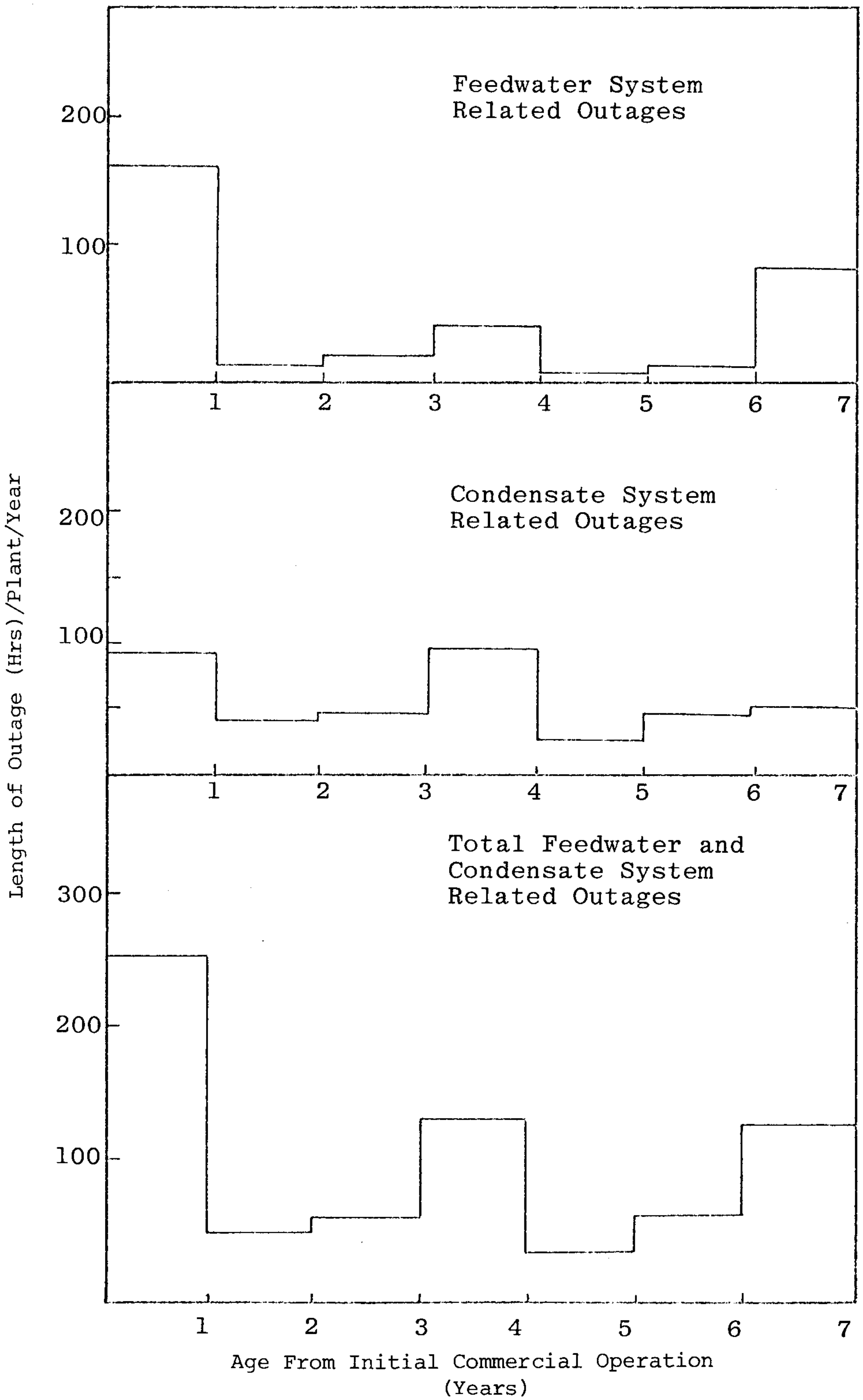


Figure 3.10. Composite of Major Outage Trends for Feedwater and Condensate Systems

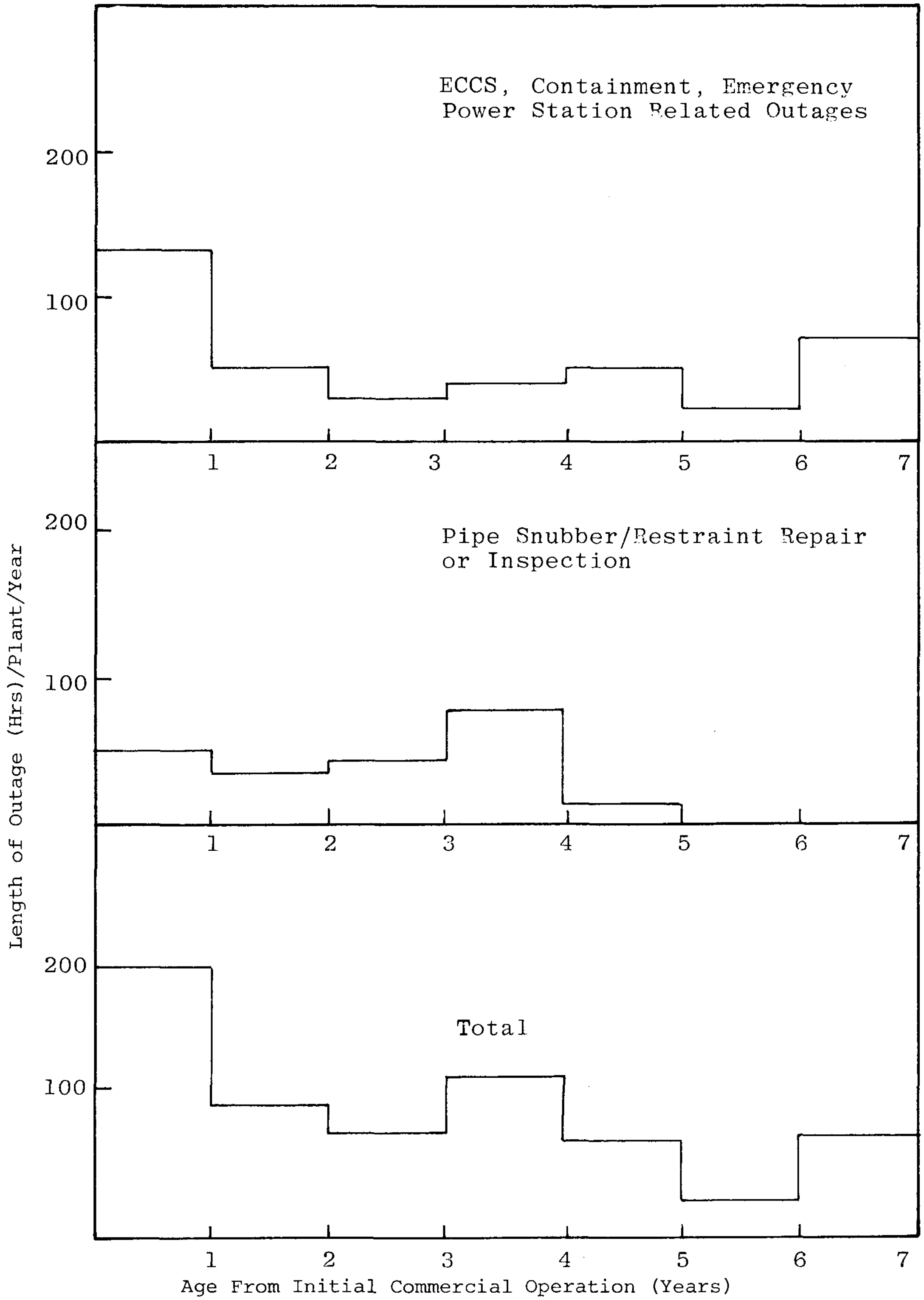


Figure 3.11. Major Outage Trends for Safety-Related Systems Versus Plant Age

systems is constancy versus time. Maintenance, testing, and repair requirements of these safety systems appear to continually affect LWRs throughout their life. A portion of this effect is due to the immediate nature of the repairs: even though the systems are not required for normal plant operation, the necessity of maintaining adequate safety systems to protect the reactor in the unlikely event of an accident requires reactor shutdown if safety systems are unavailable for a certain length of time as defined in the plant technical specifications.

In the case of snubbers, it is useful to display the major outage data as a function of calendar year. Outages which are associated with pipe restraint or snubber inspections have shown a definite trend over the past six years (see Figure 3.12). In 1973, a series of hydraulic snubber failures generated a strong regulatory interest in ensuring that pipe restraints and hydraulic snubbers were properly installed and operational. At that time, there was a great deal of activity to satisfy regulatory agencies that these systems were adequate. Since 1973 and early 1974, the amount of major outage time caused by pipe restraints and snubber maintenance and inspection has decreased dramatically. The reason for this decrease is due partially to solving the problems associated with snubbers and partially the result of scheduling maintenance and inspection work for performance during refueling outages.

3.2.5 Outage Trends for the Electrical Distribution and Electrical Generation System

This section includes the contribution to outages from the following:

- Electrical Power Distribution (principally transformer related)
- Generator/Exciter

Electrical system components contribute approximately 8% of all major outages. From Figure 3.13, the predominant features of the outage trends are:

a) A characteristic peak in the outage frequency in the first year of commercial operation.

b) A constant "background" tail contribution to plant unavailability for all years beyond the first.

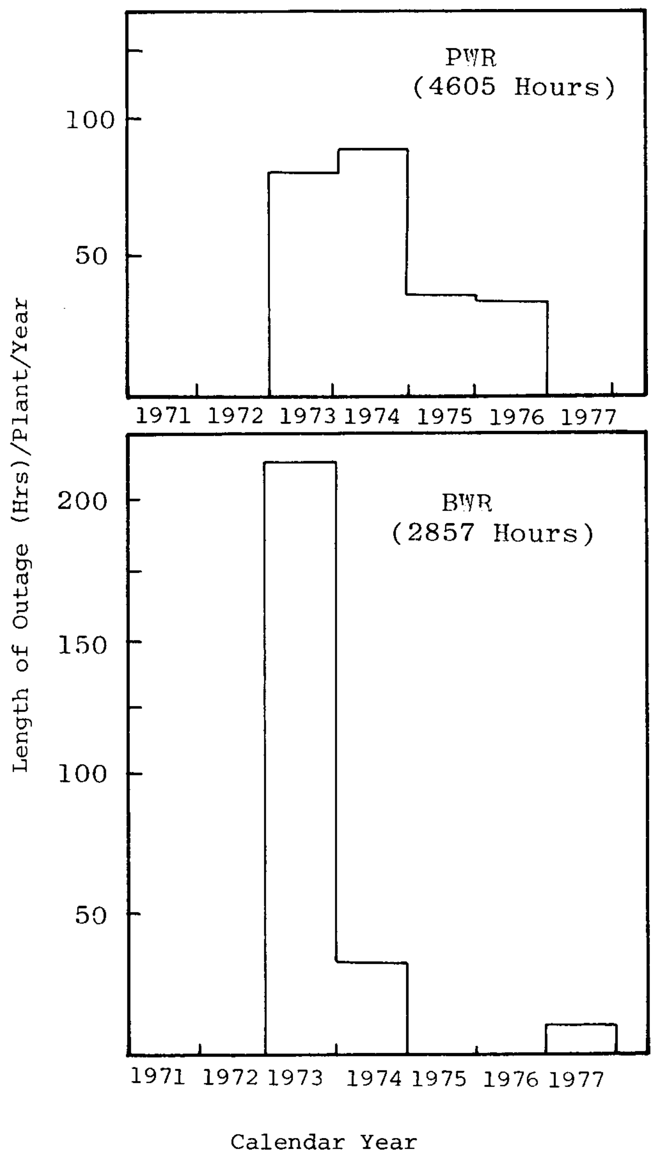


Figure 3.12. Major Outages Due to Pipe Restraint and Snubber Inspection and Repair on a Per Plant Basis (Total Outage Duration = 7462 Hrs)

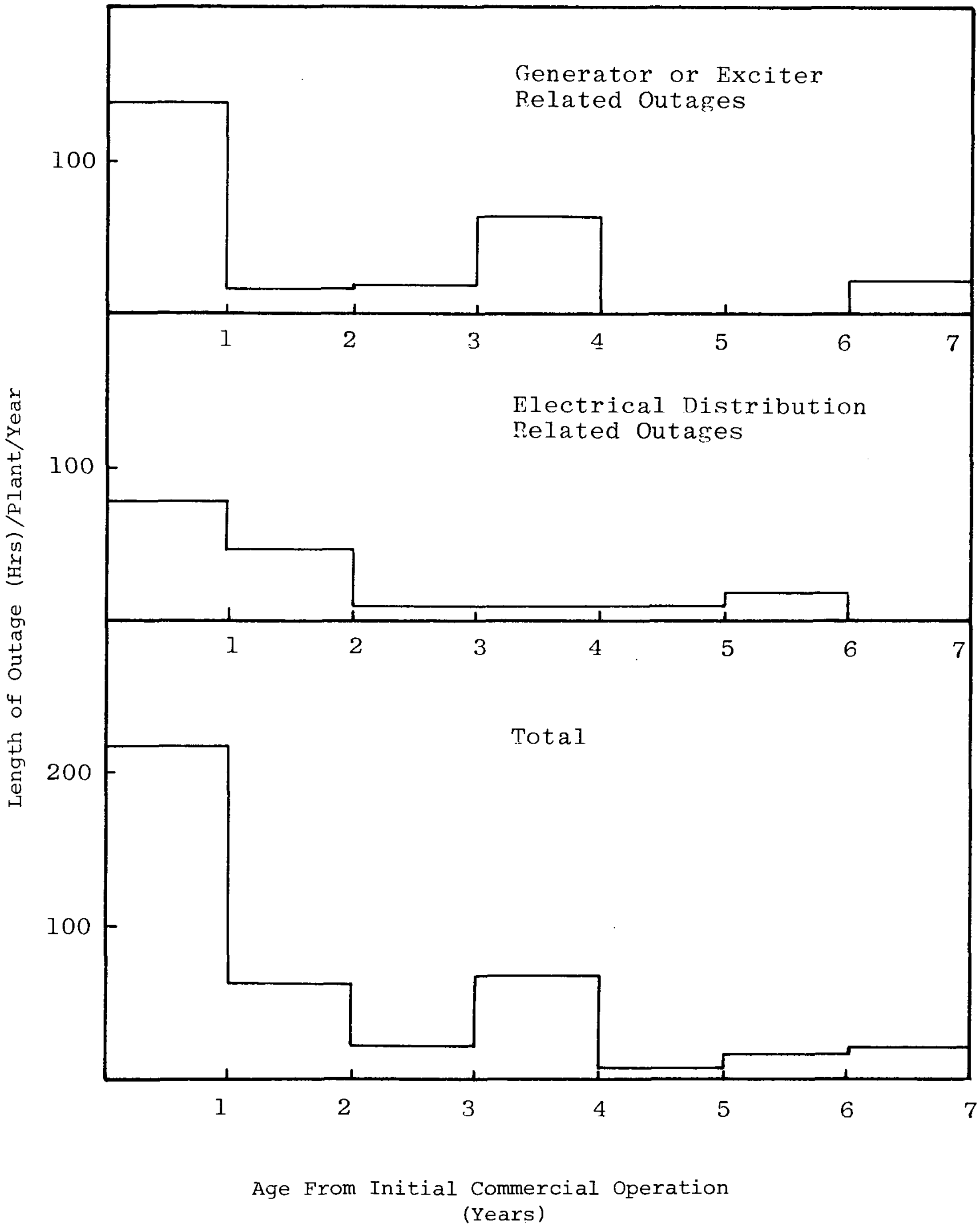


Figure 3.13. Major Outage Trends in LWR Electrical Equipment

3.2.6 Fires in Key Systems

One of the principal industrial hazards which can result in lost productivity is a fire. There have been two incidents of fire at LWRs which have caused outages greater than one month in duration:

a) A fire at Brown's Ferry in March 1975 which resulted in an extended outage of two 1000 MWe units for approximately 18 months

b) A fire at San Onofre in March 1968 which resulted in an outage of approximately 5 months (not included in the data for other sections since it occurred prior to 1971)

Each of the above fires were referred to as "cable tray fires" since they affected a portion of the extensive amount of electrical cabling required in a large power plant. The "combustible" involved in the fires was the cable insulation.

From the tabulation of outages by major cause (Section 2.3), it is noted that fire currently ranks as the fourth leading cause of outage by virtue of a single incident, i.e., the Brown's Ferry fire. Because of the apparent rarity of these events and their high impact, it is worthwhile to discuss other fire events which have occurred during nuclear plant operation as an indicator of the types of initiating events which could lead to very long duration outages as noted above. A list of some of the fires which have contributed to outages* at nuclear power plants follows:

Plant	Date	Outage Length	Location of Fire	Plant Status
San Onofre 1	2/68	288	Cable Tray	Operating
Quad Cities 2	7/72	192	Cable Tray	Startup
Peach Bottom 1	4/67	-	Heat Insulation	
St. Lucie 1	4/77	200	Generator	Operating
Vermont Yankee	11/72	360	Auxiliary Transformer	Operating
Point Beach 1	11/72	-	Steam Generator Turbine Piping	Refueling

*We have not included BWR off-gas system hydrogen explosions in this review, The off-gas system outages are reported separately (Section 3.3.8).

The above compilation of fires includes only those fires which have contributed to a major plant outage. A further investigation of fires in nuclear power plants indicates that there have been a number of smaller outages or delays attributed to fires in LWRs[36, 37]. A summary of fires which have occurred at various stages of plant operation by the major type of combustible is included here for completeness. This background data has been summarized in Table 3.4[36] which shows that there have been 53 fire incidents reported during approximately 260 plant years of operation, a frequency of .2 per year, or one incident every 5 operating years. However, as noted above, the population of serious fires at LWRs is limited to two events, each occurring during the initial year of commercial operation. Experience indicates that the time for a fire to cause the greatest impact on plant availability is during construction or extensive maintenance.

3.3 Outage Trends in LWR Components

An alternative approach to categorizing the variability of outages in LWRs would be to consider the effects of individual components on plant availability and performance versus plant age. This section is an attempt to obtain this level of detail, recognizing that the operating experience in the years beyond the fourth year of commercial operation is very limited.

Nuclear power plants have tens of thousands of components, and the accumulated operating experience is measured in hundreds of plant years. However, the number of major outages is comparatively small particularly in plants exceeding 4 years of commercial operation. The low rate of equipment-related outages in the small number of older plants results in an unacceptable statistical uncertainty in the calculated outage rate if the selection of components is reduced to a very detailed level (e.g., 1000 MWe GE turbines). Therefore, this section deals with general categories of components in the hope that sufficient data is available to characterize the overall performance of the class of components.

Table 3.4. Summary of Fires in Nuclear Power Plants

Combustible Materials Involved	Ignition Sources								Damage Estimate				Duration of Effects				Facility Condition					Plant Type		
	Spontaneous Combustion	Elect. Short Circuit	Overheating	Flame/Welding	Leak and Heat	Chemical Reactions	Lightning	Not Given	None	<$100 k	>$100 k	Not Given	<1 Day	<1 Month	>1 Month	Not Given	Construction	Startup	Shutdown	Operating	Not Given	BWR	PWR	Category Total
Oil					10				1			9			1	9	1	3	1	4	1	4	6	10
Diesel Oil					3				1			2				3			2	1		2	1	3
Wood			2	2				4			1	7	1			7	6			2†		4	4	8
Insulation			1	1	5*							7				7						3	4	7
Sealant				1							1				1					1		1		1
Elect. Equip.		3	1									4		1		3		1	1	1	1	2	2	4
Elect. Cables/Ins. Wire		3	1	1				1			1	5	2	2	1	1	1			4	1	2	4	6
Plastics		1		1				1				3	1			2	3						3	3
Adhesives				2								2	2				2					1	1	2
Rubber			1					1				2				2	1					1	1	2
Tarp				1								1				1	1						1	1
Paper			1									1				1					1		1	1
Other		2		2				1				5				5	2			1	2	2	3	5
TOTALS	0	9	7	11	18	0	0	8	2	0	3	48	6	3	3	41	17	4	4	14	6	22	31	53

* Oil soaked † Refueling

- Turbine-Generators (3.3.1)
- Steam Generators (3.3.2)
- Pumps (3.3.3)
- Valves (3.3.4)
- Condensers (3.3.5)
- Reactor Internals (3.3.6)
- Control Rod Drive Mechanisms (3.3.7)
- Off-Gas System (BWR) (3.3.8)
- Electrical Distribution Equipment (3.3.9)
- Pipe (3.3.10)

3.3.1 Outage Trends in Turbine-Generators

Although experience with 1800 RPM steam turbines dates back to the 1930's,[25] the bulk of experience in the 1950's and 1960's has been with 3600 RPM turbines using superheated steam in fossil fueled plants.

Since fossil fueled plants have had longer calendar time operation than nuclear plants, it is useful to mention the types of problems which have arisen in fossil units. However, the temperature and pressure conditions are dramatically different between fossil fueled and LWR steam turbines, leading to modes of failure which are seemingly unrelated. The times to failure of these expensive pieces of equipment are noteworthy. The problems which have arisen in fossil fueled steam turbine units[24] are:

- Blades: Blade failures occur primarily in the first stage of the high pressure turbine, with some failures occurring in the last stage of the low pressure turbine.
- Turbine Spindles: Creep rupture due to the high temperatures of superheated steam applications have caused several turbine units to be retired earlier than anticipated. Operating time in the range of 90,000 hours (15 to 20 calendar years) have been quoted for these problems to become serious.
- Turbine Casing: Steam chests have been observed to crack in the range of 55,000 operating hours or 8 years of calendar experience.
- Bolting: High temperature bolting application failures continue to plague turbine operation.

Nuclear power presents a different set of parameters for the turbine manufacturer to cope with. The low temperature and pressure of steam from LWRs contrasts sharply with the requirements of fossil fueled plants which use superheated steam. The saturated steam turbines used in LWRs require large volume flows over a relatively small enthalpy step. These characteristics have led to the use of low speed 1800 RPM turbines in LWRs.

The evolution of the saturated turbine for LWR application has incorporated many technological advancements to improve performance and overcome potential problem areas. These technological improvements are designed to eliminate problems which can be summarized[26] as uncertainties in the following:

a) Water droplet erosion from the high moisture content
b) Efficiency losses associated with the high moisture content
c) Effects of oxygenated steam
d) Use of large units for economy of scale

With this history as background, the LWR turbine generator experience can be characterized as a piece of developmental equipment exhibiting some engineering design problems.

From Section 2.3, note that LWR turbines plus the generator combine to be the leading cause of major outages, based upon the six and one half years of data included in this study. Turbine-generators are related to 18.5 % of the major outage time in LWRs.

Previous studies[7,8] have identified a difference in the performance of turbine-generator systems of the two major suppliers. In view of this, a distinction is made in this report between the turbine vendors. Figure 3.14 shows the trend in outages as a function of the age of the plant. The data shows that Westinghouse LWR turbines seem particularly troubled by blade failures, while the GE turbine-generator problems have been primarily generator related. Westinghouse units have incurred nearly six times the outage time for GE units, based both on a total outage time and on a per plant year basis.

Some problems which have occurred in the Westinghouse low pressure turbines (40 to 44 inch blades) have been previously observed in fossil fueled units. (As noted in the beginning of this subsection, TVA has reported[24] that the primary turbine-related problem has been turbine blade failure.) For the 1800 RPM nuclear turbine units under certain modes of operation at low load and high back pressure, the last row of blades in the low pressure turbine

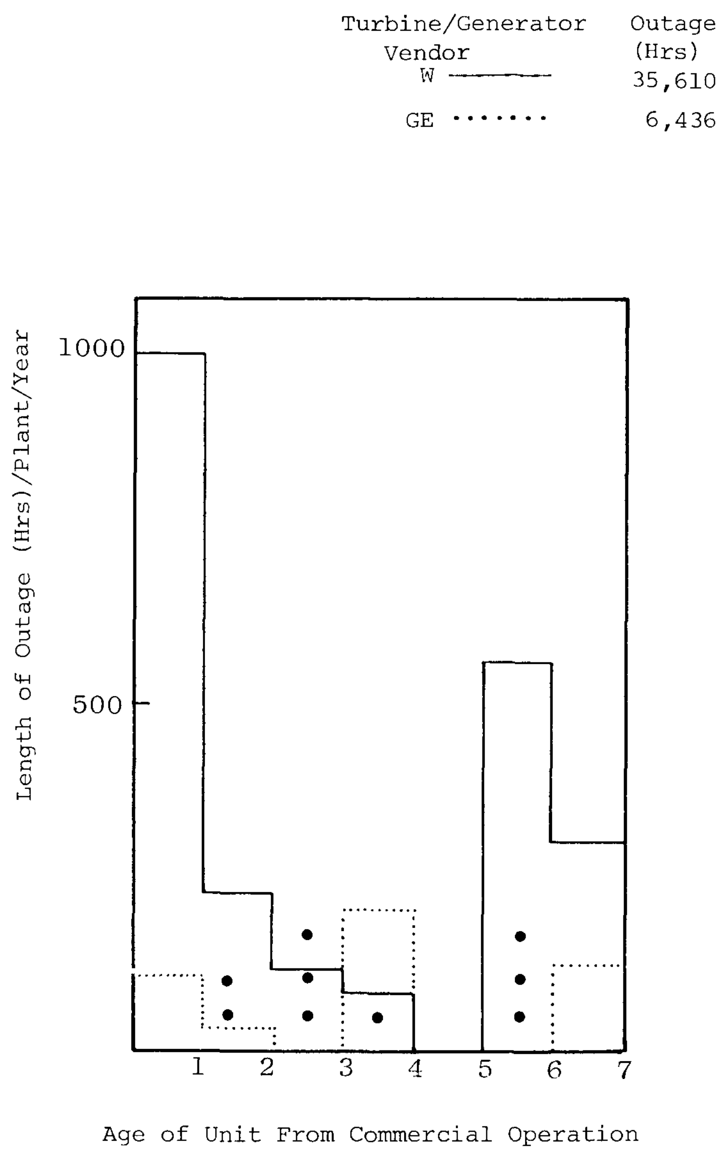


Figure 3.14. Trend in Average Major Outage Time per Plant Attributed to Turbine/Generator Components (Excluding Refueling Extensions)

● Instances of Turbine Blade Repair Occurring During Refueling is subjected to a self-excited vibration that results in relatively high stresses, causing blade failure. In addition, a resonant vibration has been observed in the 14 inch rotating blades. A summary of turbine blade failures in LWRs is presented in Figure 3.15 which is a time line displaying the plants and dates for the blade failures affecting plant availability. (Note that all units have Westinghouse turbines). Table 3.5 summarizes these turbine blade failures, their duration, and a brief description of their cause.

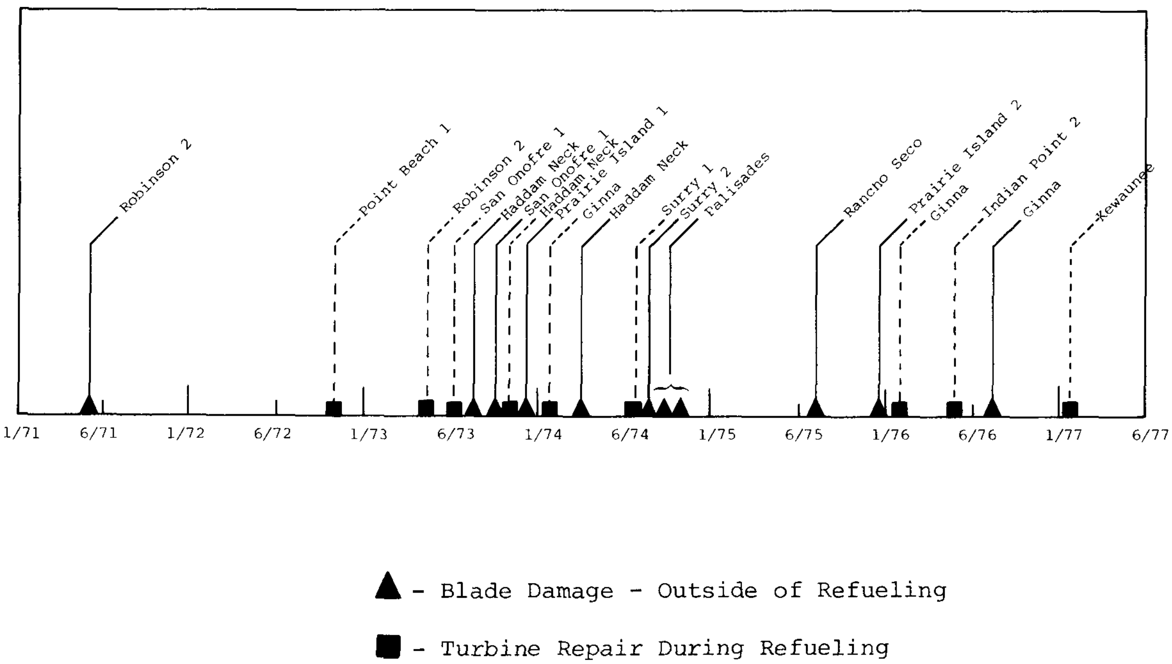


Figure 3.15. Major Turbine Blade Failures Causing Extended Nuclear Plant Outages (other turbine or generator outages not included)

Figure 3.15 indicates that major turbine blade failures are distributed throughout the six and one half year period included in this study; however, Figure 3.14 shows that the bulk of the outage time exclusive of refueling extensions occurs during the first year of commercial operation of most plants. The trend in the outage frequency per plant shows a significant drop in the second through the fifth years. Although the turbine-generator outages still make a significant contribution to the overall outage time after the second year of operation, the decreasing trend with plant age indicates that the problems are, in fact, design related and surface early in plant life. However, among the small plant population in the sixth and seventh years of operation,

Table 3.5a. Summary of Significant Turbine Outages

Plant	Date	Outage	Diagnosis
Robinson 2	May 1971	84 Days	14" blade thrown from 6th row of No. 2 low pressure turbine, resulting from resonant vibration
Haddam Neck	June 1973	15 Days	Increased turbine vibration caused shutdown for inspection. LP turbine blading was found broken
Prairie Is. 1	Dec. 1973	~180 Days	Repeated blade failures in low pressure turbine
Haddam Neck	Mar. 1974	29 Days	Blade broken in the 4th row of the low pressure
Palisades	Sep. 1974 Nov. 1974	~30 Days 30-60 Days	Reblade the low pressure turbine
Surry 2	Sep. 1974	119 Days	4 blades thrown in the low pressure turbine
Rancho Seco	July 1975	240 Days	Low pressure turbine blade failure. The failures were determined to be caused by stress corrosion cracking with NaOH as the corrosive agent
Prairie Is. 2	Dec. 1975	35 Days	Low pressure turbine blade failure. Baffles were used to replace the last 3 rows of blades
Ginna	Aug. 1976	29 Days	2 blades thrown in the low pressure turbine

Table 3.5b. Summary of the Significant Turbine Outages Occurring During Refueling Outages

Plant	Date	Diagnosis
Point Beach 1	Nov. 1972	800 40 in. low pressure turbine blades were replaced
Robinson 2	Apr. 1973	Replaced 2 LP rotors repair of cracks in stationary blading
San Onofre	June 1973	Cracked blades in last stage of LP turbine all 200 last stage blading was replaced; repair to HP turbine
Haddam Neck	Oct. 1973	Replaced 2 LP turbine rotors
Ginna	Jan. 1974	Blade thrown in the low pressure turbine
Surry 1	Oct. 1974	5 blades were cracked in the low pressure section. One stellite erosion shield was missing. No defects were found in the high pressure turbine blading
Ginna	Jan. 1976	Blade failure in low pressure turbine
Indian Pt. 2	Apr. 1976	Low pressure turbine blade failures due to cracking and erosion. Turbine modifications were made to avoid vibration induced failures
Kewaunee	Jan. 1977	Thrown blade in low pressure turbine

there have been some blade failures resulting in lengthy outages. This incidence of failures corresponds to the five year cycle of operation recommended by the turbine vendor, before complete overhaul is recommended. As operating experience begins to accumulate for operation beyond five years, turbine failures should be continually monitored to determine if a trend is developing as the plants become older which may affect LWR plant productivity.

The problem of turbine failures has been addressed by the turbine vendor and he feels that he has solved the difficulties that seemed to have plagued low pressure nuclear steam turbines, such as blade fatigue, vibration, and bearing failure. However, because of the long lead time involved in this equipment, the improvements are not fully reflected in the data from the existing units.

3.3.2 Outage Trends in Steam Generators

The summary of outages related to steam generators (primarily tube failures) indicates that steam generators are a major contributor to plant unavailability, causing 13.9% of the major outages in LWRs or 23% of the major outages in PWRs. The causes of the outages are, for the most part, related to a gradual deterioration of tubes over time; however, the outages can also be classified according to the actions required during each outage such as:

- Plug Tubes: To prevent leakage of radioactive primary coolant into the secondary system, failed tubes or those with incipient failure are plugged.
- Inspect Tubes: To determine the integrity of tubes and chart its variation with time, an Eddy Current test is performed on steam generator tubes.
- Change Secondary Chemistry: To reduce corrosive tube attack, secondary water chemistry has been changed from phosphate to all volatile chemistry treatment (AVT).
- Remove Sludge: To reduce the chloride stress corrosion cracking which is aided by large sludge accumulations, efforts are made to remove the sludge by flushing the secondary side of the steam generators.

The steam generator tubes are thin walled ($\sim$1-2mm) members that were expected to last the life of a plant with a small number of failures.* However,

*Each steam generator is fabricated with a small percentage of excess capacity (i.e., larger number of heat transfer tubes than required) in anticipation of minor problems necessitating plugging of a small number of tubes due to fabrication defects or accelerated corrosion.

experience to date indicates that these thin members are susceptible to a wide variety of failures. A summary of the dominant mechanisms of tube failure in LWRs includes the following:

1. Stress corrosion cracking of both stainless steel tubes and inconel tubes has been observed.

2. Tube vibration caused by cross flow from recirculating water has caused tube fretting at the support plate region and bend region.

3. Corrosion in the tube sheet crevice region, the classical concern of designers, has caused only a limited number of failures.

4. Condenser tube failures can be the cause of the introduction of impurities into the feedwater, thus perturbing the sensitive balance of steam generator water chemistry.

5. Secondary water chemistry control has been a point of major discussion in the prevention of steam generator tube failures. In 1974, the high incidence of wastage corrosion in steam generators prompted most suppliers to recommend a switch in secondary chemistry from phosphate treatment to all volatile treatment (AVT). Phosphate treatment has been blamed for heavy sludge formations in stagnant flow areas leading to tube wastage or thinning. However, a weakness of the AVT method is that it does not neutralize the attack of contaminants from condenser tube leaks. In addition, Surry 1 & 2 (see Section 4.0) and Turkey Point 3 & 4, all of which have been switched to AVT, have experienced significant swelling of carbon steel tube sheets which have caused "denting" of tubes and subsequent tube leaks.

   B&W has a different steam generator design and has always specified a high purity, all-volatile chemical treatment of the feedwater of their once-through-steam-generator (OTSG). In addition, B&W recommends high-purity feedwater from 100% full-flow condensate polishing demineralizers. However, the B&W Oconee steam generators have begun to show signs of an increasing number of tube defects which resulted in a number of outages for tube plugging.

Figure 3.16 summarizes the plants which have encountered failures of steam generator tubing in each year. In many of the plants (e.g., Surry 1 & 2, and Turkey Point 3 & 4) there have been multiple outages within a single year due to steam generator repairs; however, they are represented in this simple display as the total number of failures in a given year. Based on the individual plant data from Appendix I, one can note that during 1974 most plants had switched from phosphate treatment of the secondary water to an all

Legend

Plants
- □ - B&W
- △ - W
- ○ - CE

Failure Mode
- ● - Wastage
- ■ - SCC
- ▲ - Denting
- ⬬ - Fretting
- ▼ - Erossion/Corrosion
- ● - Defect
- ( ) - (No. of Tube Failures)

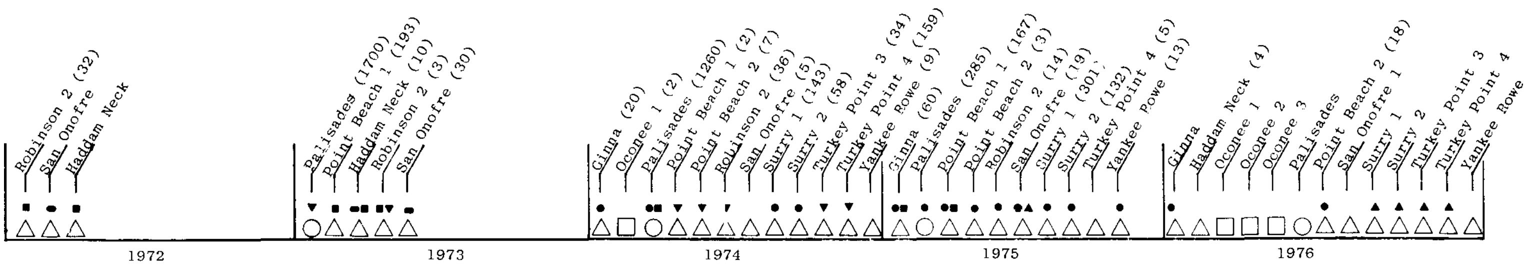


Figure 3.16. U.S. PWRs Requiring Steam Generator Tube Plugging by Year of Event[27,28,29,30,31] (Vendor, Reported Failure Mode, and No. of Tubes Repaired Also Included)

volatile chemistry (e.g., hydrazine). From Figure 3.16, this switch coincided with the reporting of fewer wastage failures and the emergence of the problem referred to as denting.

Also from Figure 3.16, there is an apparent high frequency of Westinghouse steam generator tube problems. This can be explained in part by the much larger population of installed Westinghouse steam generators. Figure 3.17 shows a composite plot of the contribution of steam generators to major outages, broken down by principal suppliers. Westinghouse units have accumulated the largest amount of operational experience and the highest amount of outage time attributed to steam generators. B&W and CE units have a much smaller amount of operational experience, that is, there are only 6 B&W units and 6 CE units operating. Therefore, if we use major outage time on a per plant year basis as a measure of performance, the Westinghouse steam generators are performing as well, or better than the B&W and CE units.

The distribution of major outages with plant age is similar for Westinghouse and B&W units despite their apparent marked difference in design. Each has experienced a dramatic increase in outage time in the fourth year of commercial operation. Less confidence can be applied to the CE distribution since all major outage time is associated with a single plant, Palisades.

A very limited data sample beyond the fourth year (virtually no B&W or CE experience) results in more uncertainty in the fifth, sixth and seventh year average outage numbers; however, the outages due to steam generators in those years represent a significant contribution to the overall outage rate and are a principal area of concern in long term PWR plant performance.

Figure 3.18 is a frequency histogram of the major outages related to steam generators during the period January 1971 to June 1977. It is apparent that there is a very high frequency of relatively short duration outages (100-200 hours) related to tube plugging; however, there are a substantial number of other outages with much longer outage durations. In fact, the mean outage is more than 500 hours.

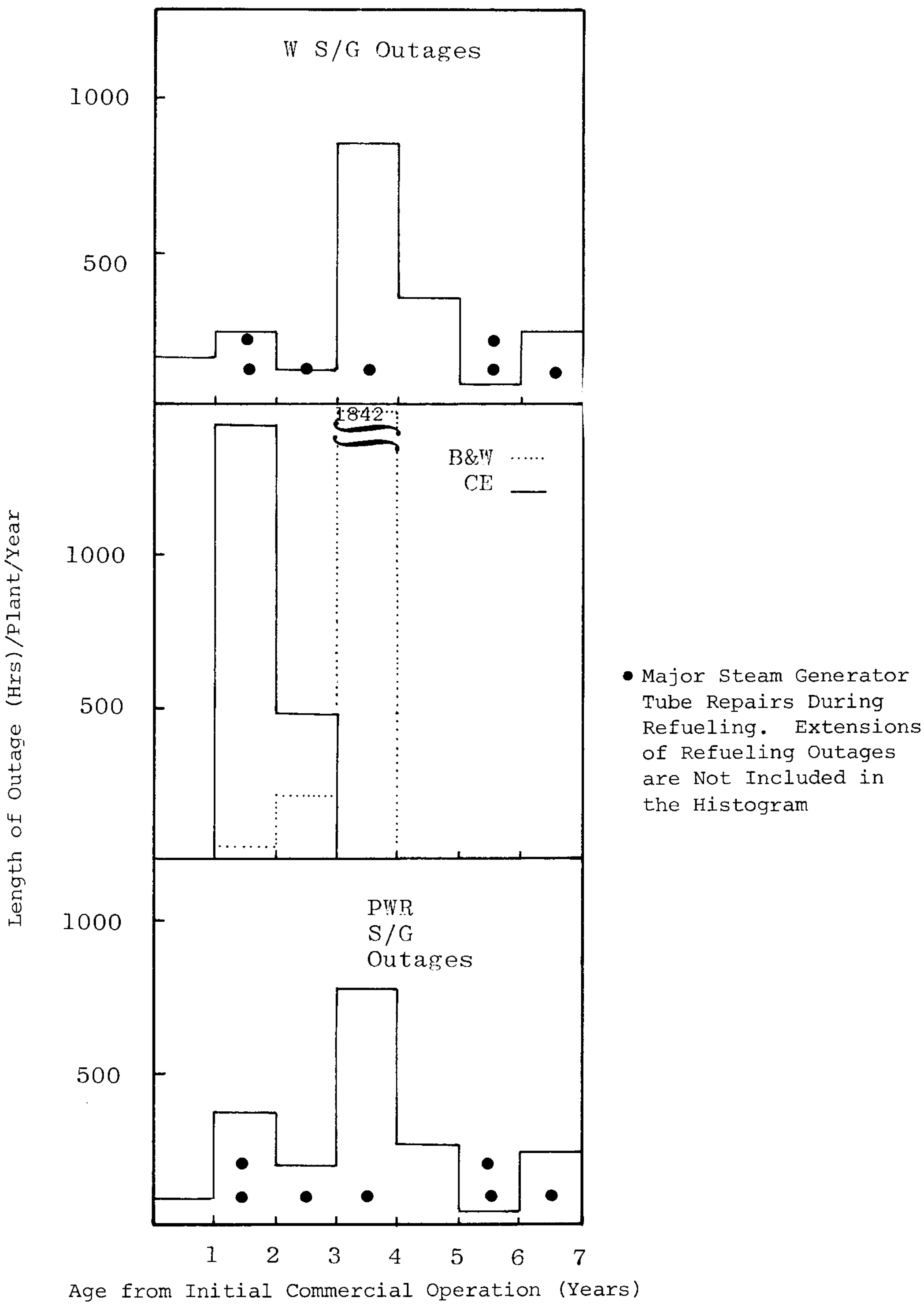


Figure 3.17. Comparison by Vendor of Major Outage Variations Related to Steam Generators

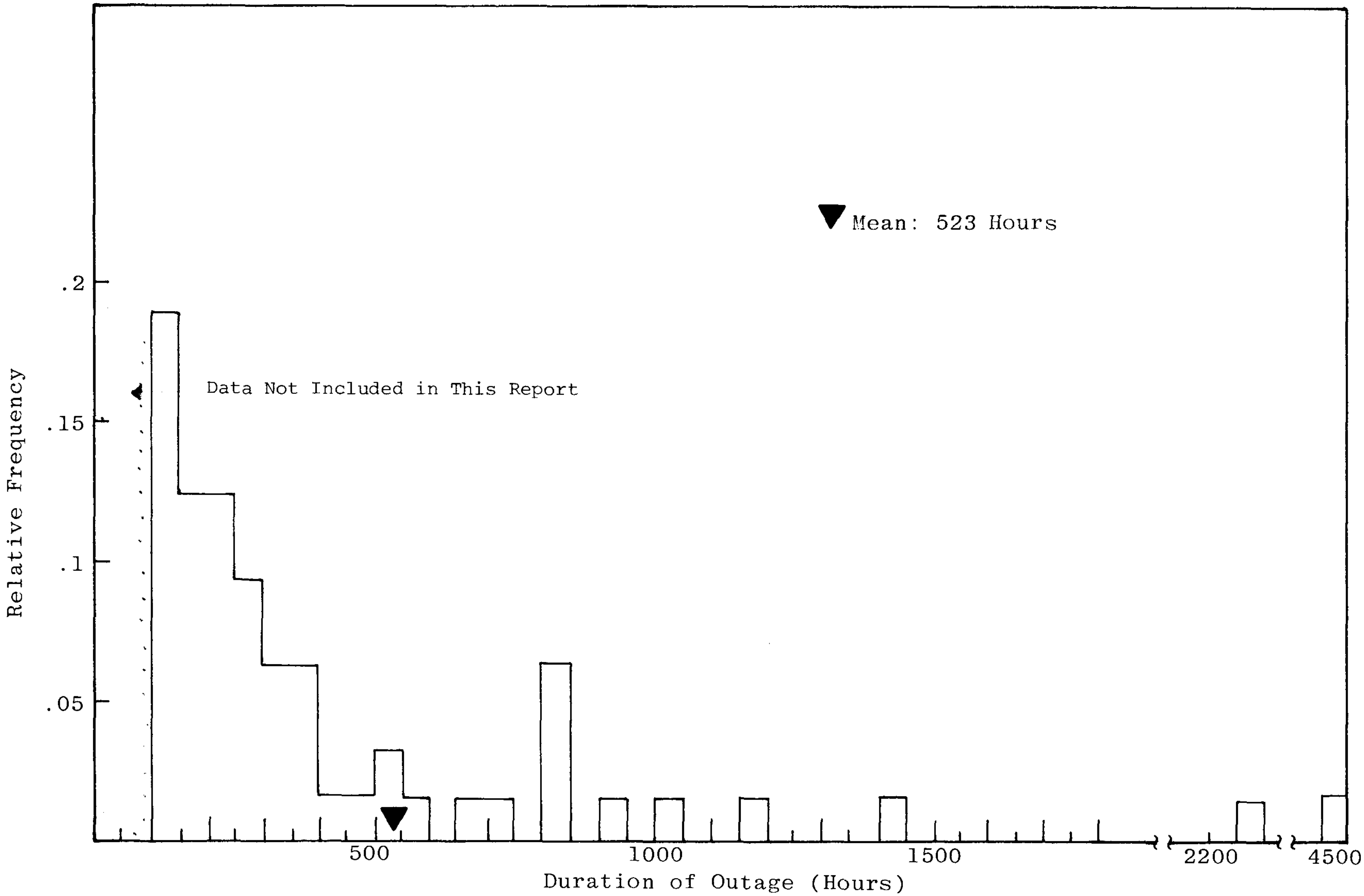


Figure 3.18. Frequency Histogram of Outages Involving Steam Generators

The Nuclear Regulatory Commission has taken a keen interest in steam generator tube failures and has taken steps to require periodic surveillance testing of steam generator tubes (e.g., Eddy Current testing). Some recent major outage time is related to this testing which includes obtaining base line data on all tubes and the subsequent periodic testing to chart any deterioration of tube wall thickness.

3.3.3 Outage Trends in Pumps: PWR Reactor Coolant Pumps and BWR Recirculation Pumps

Reactor coolant pumps account for a substantial portion of the total outage time in nuclear reactors. In PWR plants, the reactor coolant pumps are associated with major outages totaling 16,483 hours, or approximately 12% of the PWR major outage time. From Figure 3.19, we note that the trend of the PWR pump outages shows a very large peak in major outages during the initial two years of commercial operation. A few high impact incidents have accounted for a large percentage of the pump outage time:

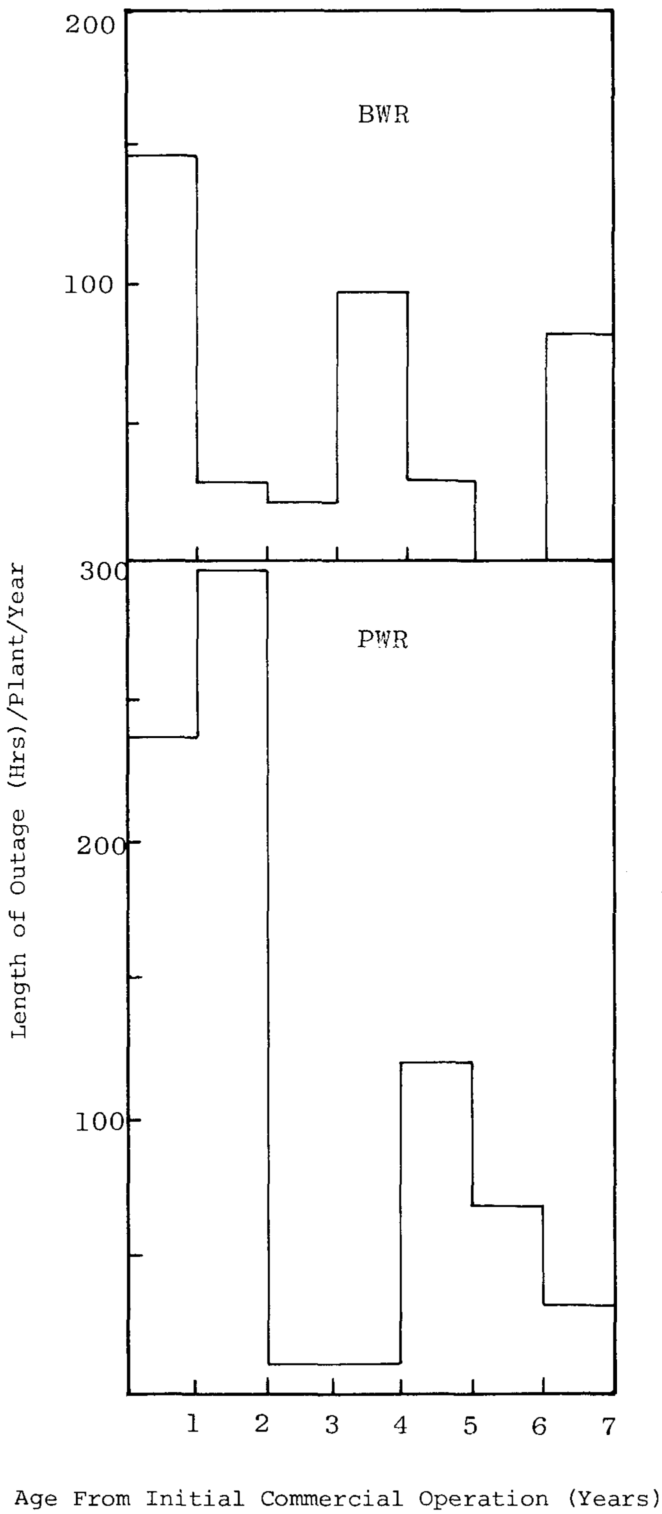
a) Main coolant pump shaft failure at Surry 1 resulted in an outage of 2529 hours.

b) Main coolant pump shaft replacement at Surry 2 to correct the design deficiency which caused the failure of Surry 1 resulted in an outage of 1552 hours.

c) Main coolant pump motor maintenance at Zion 1 and Oconee 2 resulted in outages of 1118 and 1142 hours respectively.

Most of the shorter duration outages are related to pump seal problems; in fact, these are the only major outages associated with reactor coolant pumps which have occurred in the fifth through seventh year of operation. There are also a number of these seal problems occurring in the first two years of operation. In summary, 88% of the PWR reactor coolant pump major outage time has occurred during the first two years of commercial operation.

The BWR recirculation pump performs an equivalent function to the PWR reactor coolant pump. In BWR plants, these pumps have accounted for approximately 4960 hours or 6% of the major outage time for BWRs: that is, the recirculation pumps have accounted for only ½ of the percentage of outage time in BWRs as the comparable PWR pumps. The distribution of major outages in time is similar for BWRs and PWRs. Both have an early period of high outages which falls off rapidly and then tends to increase in later years.



Total Outage Time For Pumps: Reactor Coolant (21,443 hours, or 9.5% of the major outages)

Figure 3.19. Trend in Average Major Outage Time per Plant for PWR Main Coolant Pumps and BWR Recirculation Pumps

The overall outage contribution from the PWR reactor coolant pumps and the BWR recirculation pumps is 21,443 hours, or 9.5% of the total major outage hours included in this study. There has been a small percentage effect on major outages due to other pumps, principally feedwater pumps. All pumps other than the PWR reactor coolant pumps and the BWR recirculation pumps have contributed only 899 hours, or .4% of the total major outage time; virtually all of these outages occurred in the first year of commercial operation. Combining all pumps, the percentage of major outages due to pumps is approximately 10%.

Consider a frequency histogram of the relative frequency of major pump-related outages (Figure 3.20). There is a relatively high mean value to the major outage time. This is principally due to the major outages cited above for PWR reactor coolant pumps (all greater than 1000 hours). An interesting observation is that 82% of the major outages less than 1000 hours in duration are related to pump seal problems.

The pump seal leakage problems are probably design-related failures; however a major contributing cause is that large numbers of plant transients (heatup and cooldown cycles) can lead to premature seal failure, reinforcing the belief that minimizing plant temperature transients will extend equipment operating life.

3.3.4 Outage Trends for Valves

Each power plant has numerous and diverse valves in the NSSS and BOP parts of the plants. It has been estimated[17] that the population of all valves approaches 10,000 for a 1000 MWe LWR plant. Fortunately, only a fraction of these, ~1/10, are located such that their failure or need of maintenance, test, or repair would result in a forced outage. In PWR plants, major outages totaling approximately 9578 hours, or 7% of the PWR major outage time, have been reported as being related to valve problems. The BWR major outage time attributed to valves is 8695 hours, or 9.6% of the reported major outages.

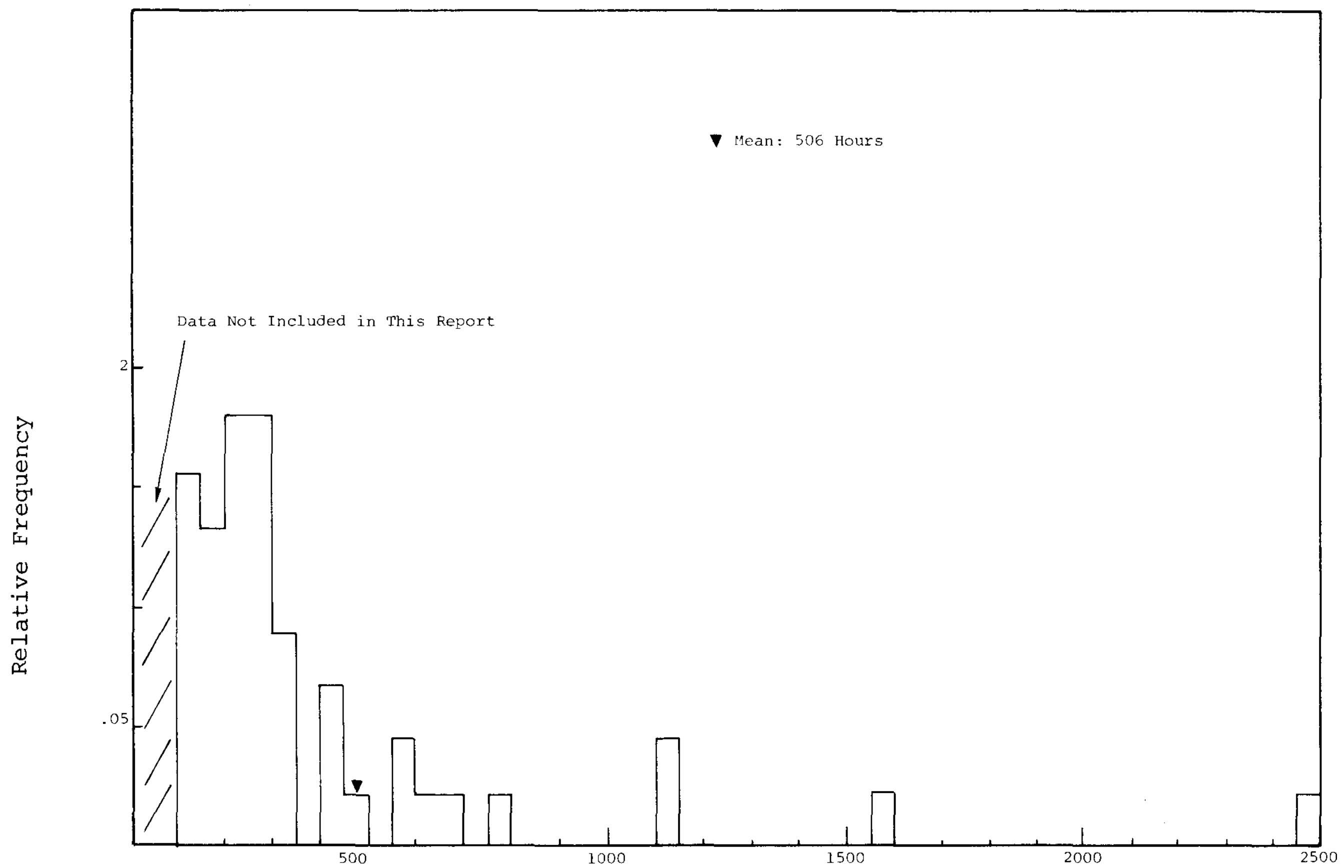


Figure 3.20. Frequency of Pump Related Major Outages

Figure 3.21 shows the trend in outages for PWR and BWR plants. A distinct difference in the BWR and PWR trends is evident. It appears that PWR valve problems causing major outages occur early in plant life and become less frequent as the plant matures. On the other hand, BWR plants show a different trend, indicating that major outages due to valve problems increase later in plant life. This phenomena has also been noted previously for BWR piping failures[15] although no connection between the two component failure rates has been established at this time.

Each time a nuclear plant is cycled from operating temperature to cold shutdown, the plant components undergo a severe test of their flexibility. During this $\sim 400^{\circ}F$ cooldown and heatup cycle, the seals, joints, and moving parts do not always respond as expected, and leaks or failures can occur. However, estimating the time required to fix these problems during startup and plant recovery is difficult because of the lack of available information; therefore, a true representation of the impact of valve and pump problems is not reflected in their contribution to major outages only. Lost plant availability due to valve leakages, failure to pass surveillance tests, or failure to operate, is judged to be appreciably larger than only the major outage contribution, however, this contribution is beyond the scope of this report.

Figure 3.22 presents a frequency histogram of the major valve outages as a function of outage duration. Compared with the other components considered in this report, valves have an unusually low mean major outage time $\sim$200 hours versus more than 500 hours for pumps, in-core problems, turbine generators, and steam generators. In addition, previous studies [6,13] have found that the most frequent valve failures are those causing outages of less than 100 hours, which, of course, are not included in this report.

It is very useful to identify the major types of valves which are causing the outages included in the current study even though previous work[17] concludes that the numbers of valves and types of failure modes are so diverse as to defy a generalized classification. Figures 3.21 and 3.22 have lumped all valve outages together. The exact pedigree of these valves can be better understood if we look at individual types of valves causing these long duration outages.

Total Outage Due to Valves (18,223 hours)

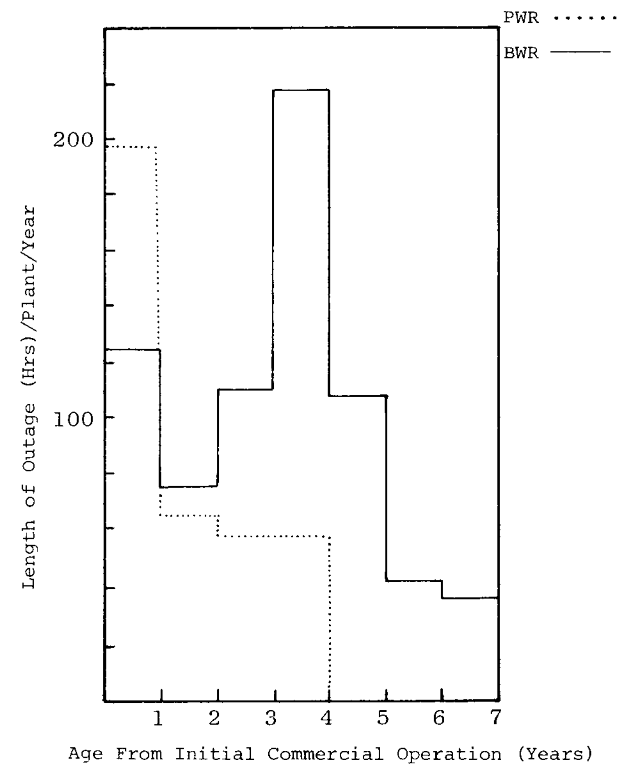


Figure 3.21. Trend in Average Major Outage Time per Plant for PWR and BWR Valves Versus Plant Age

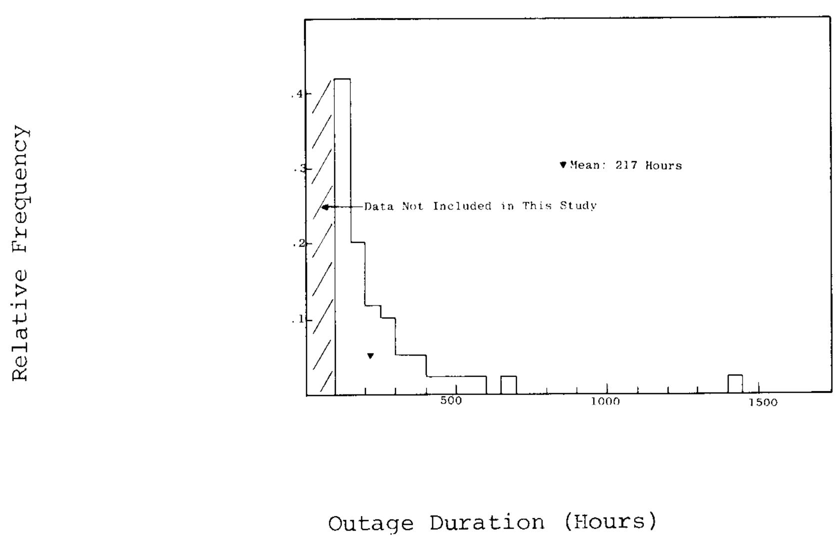


Figure 3.22. Frequency Histogram Outages Related to Valves in LWRs

The BWR valves contributing to major outages are summarized in the following table:

Valve	No. of Incidents of Outage >100 hrs.	Total Outage Hours	Mean Major Outage (Hours)
Main Steam Isolation Valve (MSIV)	9	1760	196
Relief & Safety Valves	12	1851	154
Feedwater:			
a) Regulating Valve	6	1223	204
b) All Feedwater	11	1901	173
ECCS Check Valves	2	263	132
Turbine (Bypass, Stop, Control)	5	910	221

PWR valves contributing to major outages are summarized in a similar manner in the following table:

Valve	No. of Incidents	Total Major Outage (Hours)	Mean Major Outage (Hours)
Pressurizer Spray Valves	5	880	176
Turbine (All Valves)	7	1803	243
ECCS (Check Valves)	4	764	191
MSIV	4	1020	255
Relief Valves	3	439	146
Trip Valves (Surry)	4	1646	412
Feedwater (All)	3	1757	586

A previous study[17] pointed out that the valves primarily responsible for outages are as follows:

1) BWR
   a) Relief Valves
   b) Main Steam Isolation Valves
   c) Turbine Valves

2) PWR
   a) Main Steam Isolation Valves (MSIV)
   b) Feedwater Control Valves
   c) Pressurizer Spray Valves
   d) Turbine Valves

The current report concludes that BWR feedwater regulating valves are also a significant contributor to major plant outages and should be added to the above list of BWR valves.

3.3.5 Outage Trends in Condensers

Condenser tube problems have received a significant amount of attention recently because of their increasing impact on plant performance through both plant power limitations* and plant outages. The contribution of condenser-related problems to major outages is 11,765 hours, or 5.2% of the total outages. Figure 3.23 shows the distribution of these outages versus plant age. The overall outage time, while significant, is not alarming in itself. However, the trend in outages versus plant age is indicative of a recurring problem similar to the steam generator tube failures which may continue to require plant outages to repair.

In addition to the explicit failures and outages attributable to condenser tube failures, it has been suggested that other outages can be traced back to condenser tube problems. Specifically, it has been suggested that: (a) PWR steam generator tube problems can be linked to secondary chemistry disturbances which are caused by in-leakage in the condenser; and (b) BWR chloride intrusion may be a contributing factor to accelerated stress corrosion cracking in the primary

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*Power limitations generally result from the reduction in power needed to repair or plug tubes in a portion of the condenser.

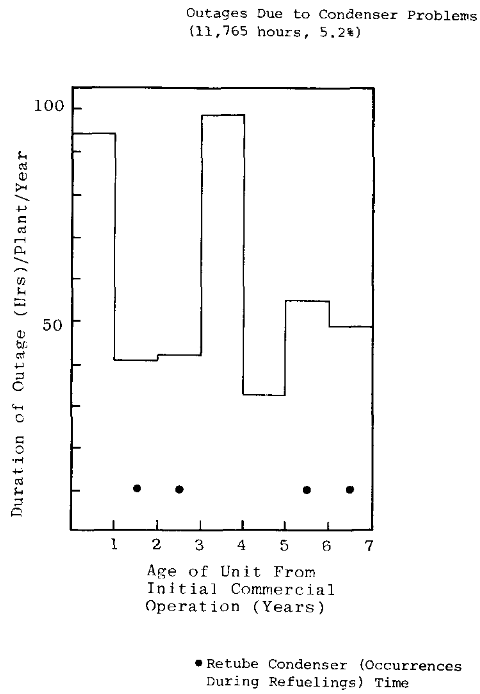


Figure 3.23. Trend in Average Major Outage Time per Plant for Condensers

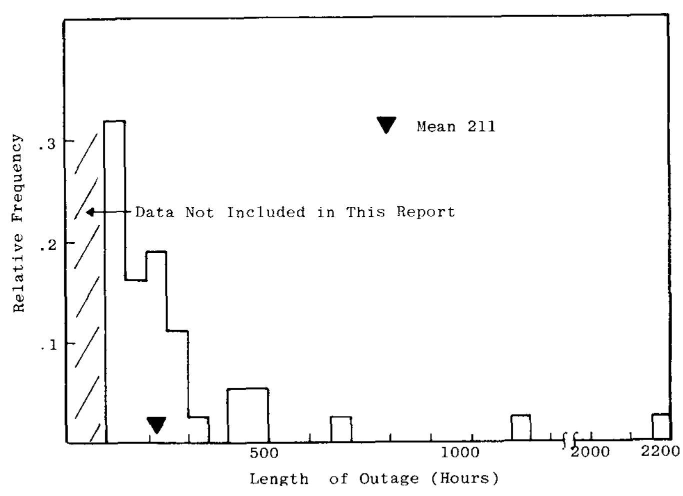


Figure 3.24. Frequency Histogram of Outages Associated With Condenser Problems

system. Note in Figure 3.23 that in those years in which there appears to be a reduction in the average major outage time due to condensers per plant, there was also a major retubing of a plant condenser during a refueling outage. The general conclusion then, is that condenser tube problems are showing an approximately constant trend with increasing plant age.

Figure 3.24 is an outage frequency histogram indicating that the predominant number of condenser problems are resolved with outages of less than 200 hours. The two outages of greater than one month duration are attributed to retubing condensers at Millstone 2 and Palisades.

3.3.6 Outage Trends Involving Reactor Internals

The nuclear reactor pressure vessel contains the heat source - the fuel - for the operation of an LWR power plant. However, in addition to the fuel there are a number of other components required in the reactor vessel for the safe operation of the plant. These components include such items as: a core barrel, core support structure control rods and control rod drive mechanism, material surveillance tubes, instrumentation, and portions of emergency core cooling systems.

Each of these components is exposed to the environment within the reactor vessel including: high radiation, high temperature, and high flow. The design rules used to ensure that each of these effects does not interfere with the component's functions require experienced engineering judgment to properly apply. Operating experience indicates that some of the peripheral components, such as instrumentation and surveillance tubes, have not been properly designed to withstand the high flows and have failed due to flow induced vibration.

Because of the location of these components adjacent to the fuel coolant channels, it is generally considered prudent to repair any failures or potential failures as quickly as possible. Therefore, it is not always possible to wait until the next refueling to perform the maintenance. Besides the immediacy of the repair, the repair also requires disassembly of the reactor closure head; therefore, the outage is usually quite long. Figure 3.25 gives the relative frequency of the major outages attributable to reactor core related repairs. The mean time to repair for the 15 major outages is 1480 hours. The median time to repair is approximately 1000 hours.

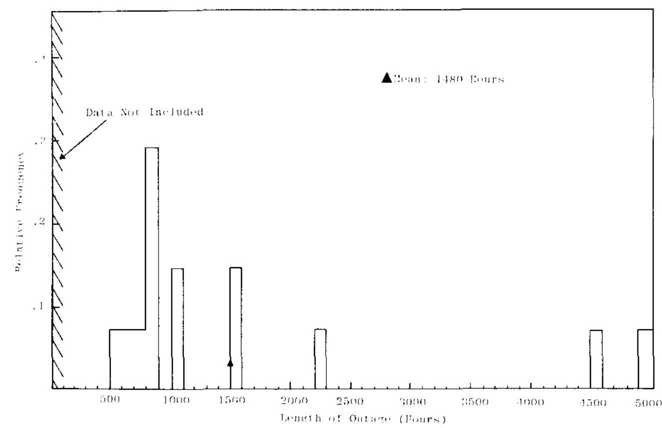


Figure 3.25. Frequency Histogram of Major Outages Related to Reactor Internals

The following generic in-core problems have accounted for 9.8% of the major outage time considered in this study (i.e., outages >100 hours exclusive of refueling):

a) Reactor Surveillance Specimen Tube Holders in B&W Reactors

b) Local Power Range Monitor Prlbmes in BWRs

c) Reactor Core Barrel Vibration in Palisades & Fort Calhoun (CE)

It is noteworthy that Westinghouse reactors did not encounter similar core related major outages in the data considered. However, Westinghouse has experienced some flow-induced vibration problems in the initial years of its commercial reactors (e.g., San Onofre and Haddam Neck both had thermal shield vibrations which required major outages).

In BWR plants, reactor internal problems have involved more than just fuel, instrumentation, and core support problems; it has also involved core spray piping and feedwater sparger piping located above the core. These additional outages are discussed in Section 3.3.10 and are not included in this section. Their history is thoroughly discussed in Reference 32.

The time distribution versus plant age of these problems is shown in Figure 3.26 and 3.27. Both the BWR and PWR trends indicate that the in-core related major

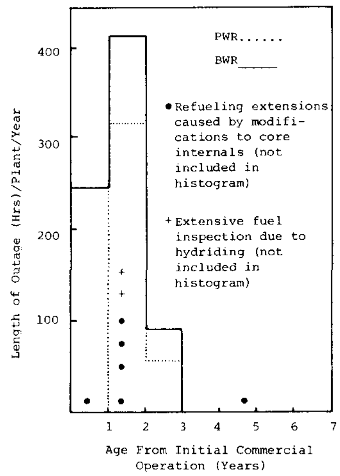


Figure 3.26 Trend in the Average Major Outage Time per Plant Related to In-Core Problems in PWRs and BWRs

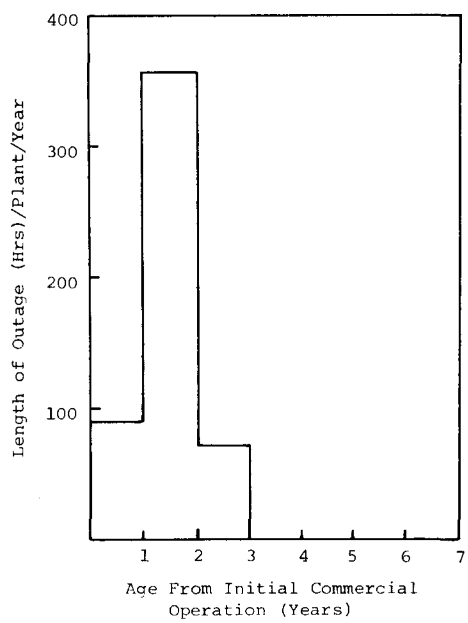


Figure 3.27 Trend in Average Major Outage Time Per Plant Related To In-Core Problems for All LWRs

outages have all occurred within the first 3 years of commercial operation, with the vast majority occurring in the first two years of commercial operation. In addition to the major outage time associated with core related components, there are a number of refueling extensions which are directly attributable to these problems. Figure 3.26 also gives the number of refueling outages which have been extended due to the generic core related repairs discussed above. It is important to note the trend in these compared with that noted in the major outage histogram. The trend appears quite similar with a peak in the second year of commercial operation and very little outage time beyond the second year.

The core related outages can be characterized in the following manner:

a) The basic cause of the problems has been an oversight in the design of core related equipment to not account for flow-induced vibration.

b) It appears that very few of the core vibration problems will remain unresolved as plants reach ages beyond 2 years.

c) Such generic problems should be avoided by proper design in the future.

3.3.7 Outage Trends in Control Rod Drive Mechanisms

Historically, control rod drive mechanisms (CRDM) have had a tendency to cause short duration outages. Occasionally, a problem is encountered which results in longer duration outages; however, the frequency of major outages related to CRDMs is relatively low.

The PWR outages associated with CRDMs are shown in Figure 3.28 and occurred over most of the time considered in this study (i.e., during the first five years of commercial operation).

It is important to emphasize that there is a large uncertainty in the fifth, sixth, and seventh years since only a limited amount of data is available. In addition, this report focuses only on seven years of commercial operation. Total commercial nuclear experience is limited to less than 17 years. Therefore, the conclusions drawn from these trends cannot hope to show the long term performance of CRDMs for a full plant life, i.e., 40 years.

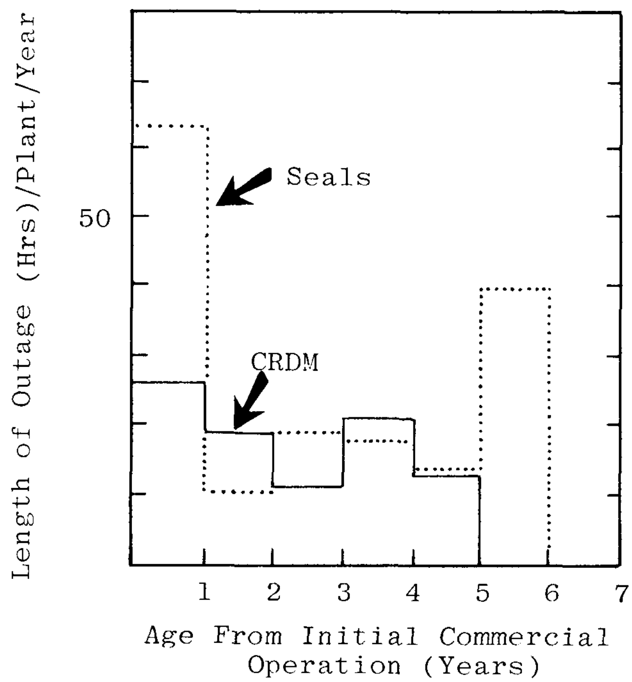


Figure 3.28. Trends in Average Major Outage Time Related to PWR Control Rod Drive Mechanisms and Closure Head Seals

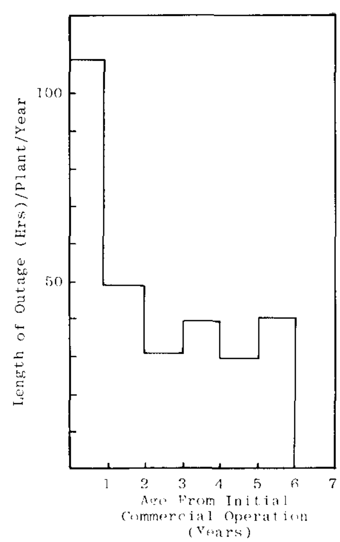


Figure 3.29. Trends in Average Major Outage Time Related to CRDMs or Closure Seals for PWRs (Combined)

The outages due to closure head seals and CRDM seals have also been included in this section. CRDM seals have caused recurring major outages at only one plant - Palisades. Figure 3.28 shows the comparison of the contributions to outage by CRDMs and seals. The trend in seal-related outages is similar to the CRDM trend with the exception of the higher outage rate during the first year of commercial operation. The combined outage rate for PWR CRDMs and seals is given in Figure 3.29.

BWR control rod and CRDM problems have not occurred with the same frequency as noted in PWR plants. In fact, the BWR outages are almost exclusively related to changing the control rod pattern and not to hardware problems. Figure 3.30 shows the trend in CRDM major outages for BWRs. The limited amount of data in the fifth, sixth, and seventh years of commercial operation indicate that the older BWR plants do not exhibit outages >100 hours due to control rod or CRDM problems.

Two control rod problems are not included in this analysis since they occurred in the time frame of 10-15 years from initial commercial operation:

a) Dresden 1 - control rod follower replacement - 1117 hours
b) Yankee Rowe - control rod replacement - 4104 hours

3.3.8 Outage Trends in BWR Off-Gas Components

BWR units have encountered a number of outages in the 1970's related to the upgrading of the radioactive off-gas system. The upgrading is needed to process the higher than anticipated stack releases due to fuel failures and to dispose of high fission gas levels. The system revisions have occurred during refueling and non-refueling periods. Figure 3.31 shows the outages greater than 100 hours due to off-gas system maintenance or upgrading. One hazard of the off-gas system is an explosion. Six explosions occurred in the off-gas systems of three different BWRs during a one year period:

Plant	Cause	Number
Quad Cities Station	Lightning	1
Dresden-2	Welding Torch	1
Vermont Yankee	Lightning (2) Unknown (2)	4

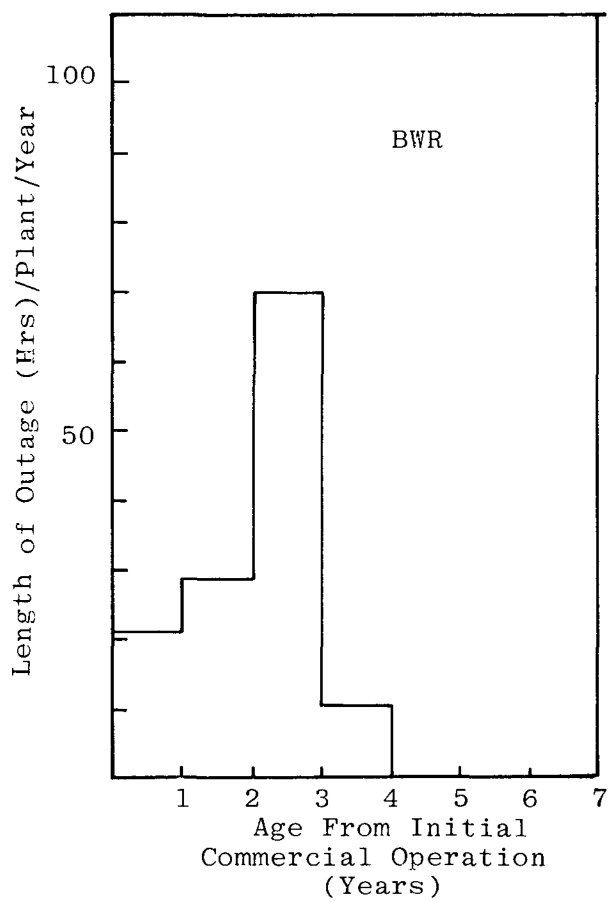


Figure 3.30. Outage Trend in BWRs Related to Control Rods and CRDMs

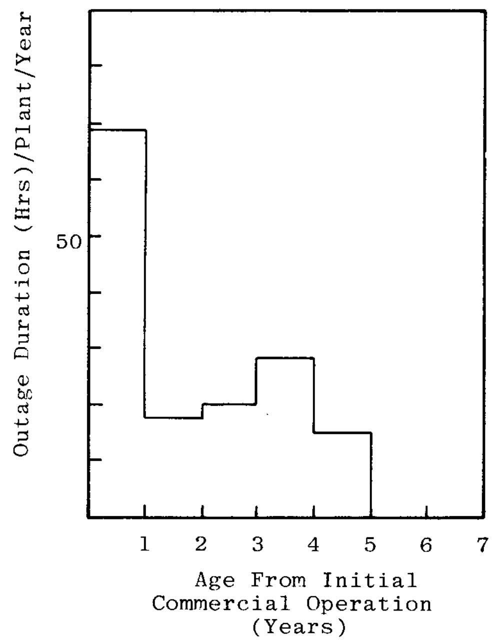


Figure 3.31. Major Outage Distribution in BWRs Due to Off-Gas System

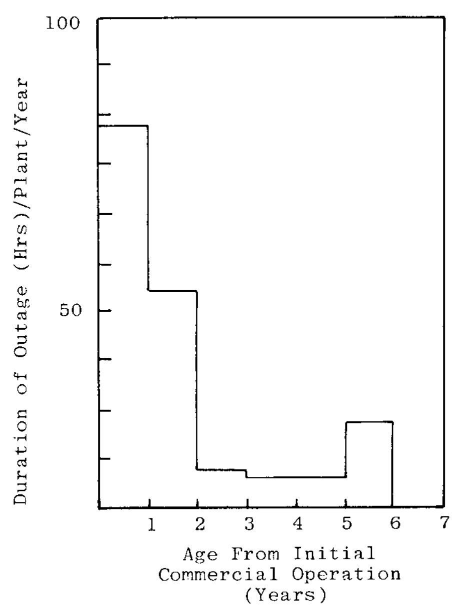


Figure 3.32. Major Outages Attributed to the Electrical Distribution System

The explosions resulted from the ignition of hydrogen normally present in the off-gas system. The hydrogen (and oxygen) result from a radiolytic decomposition of primary coolant water in the vicinity of the reactor core. In BWRs, these and other gases are carried by steam to the turbine where they are removed from the main turbine condenser and directed into an off-gas system. During operation, approximately 60% of the volume of the non-condensibles in the off-gas system is hydrogen, well above the 4% flammability limit for hydrogen in the presence of oxygen. The explosions have resulted in blown rupture disks (overpressure protection devices) and damage to filters in the off-gas system. The rupture disks provide a path for the gases to bypass the normal 30-minute delay line. This bypass results in an increase in the radioactivity available for release. However, the radiological effects of the explosions to personnel on and off site were not significant.

Of the ten events considered in Figure 3.31 only three were caused by failure of the system. The remaining major outage contributions are due to system modifications which require plant shutdown. In general, the off-gas system repairs are accomplished in less than 100 hours, and therefore, are not included in this report.

3.3.9 Outage Trends for Electrical Distribution Equipment

The major outages from electrical distribution equipment total 6,278 hours, or 2.8% of the total major outages included in this study. LWR electrical distribution systems exhibit a decreasing trend in outage duration per plant with increasing plant age (see Figure 3.32): The first two years of commercial operation have the highest outage time per plant. Transformers are the principal components causing these long duration outages.

3.3.10 Outage Trends in Nuclear Plant Piping

The major outages related to pipe problems* result in approximately 7.1% of all the major outages considered in this study. The loss in plant availability due to these major outages is approximately evenly split between PWRs and BWRs. However, the characteristics of the pipe related problems are quite different in the two plant types.

*This does not necessarily imply a failure of pipe, rather it includes outages for modifications to pipe arrangement, such as at Turkey Point.

The frequency distribution of major outages related to piping is significantly different between PWR and BWR plants (see Figure 3.33). The BWR outages are dominated by outages related to the austenitic stainless steel piping in the core spray, feedwater sparger, and recirculation bypass system and generally results in outages less than 500 hours in duration. These outages of 200-300 hours are usually associated with inspections. Actual repair work is usually performed during refueling outages. On the other hand, PWRs have a lower frequency of pipe related major outages but each outage tends to be of longer duration. The systems involved are quite diverse and the causes of the outages are generally related to resolving problems associated with vibration or pressure surges.

Figure 3.34 provides a comparison of the variation of pipe related major outages as a function of plant age from initial commercial operation. There is a marked difference in the time distribution between PWRs and BWRs. The BWR major outages related to piping are almost exclusively due to investigation and/or repair of cracking in austenitic stainless steel pipe. These outages are spread over each of the seven years included in this study. On the other hand, virtually all of the PWR major outages related to piping occur during the initial two years of operation.

The BWR pipe cracking problem of sensitized stainless steel can also be viewed as a function of calendar year. In 1973-1975 there was a great deal of activity to identify those plants which had a problem with sensitized stainless steel piping and the subsequent development of cracks. Figure 3.35 below summarizes the outage times for these inspections and repairs.

The trend is clear: At the initial recognition of the problem, a large amount of special outage time outside of refuelings was spent inspecting all potentially affected plants. However, follow-up inspections, repairs, and replacements have been scheduled during refueling outages. The result is an extension of refueling outages. Figure 3.35 notes the frequency of occurrences of these repairs during refueling outages.

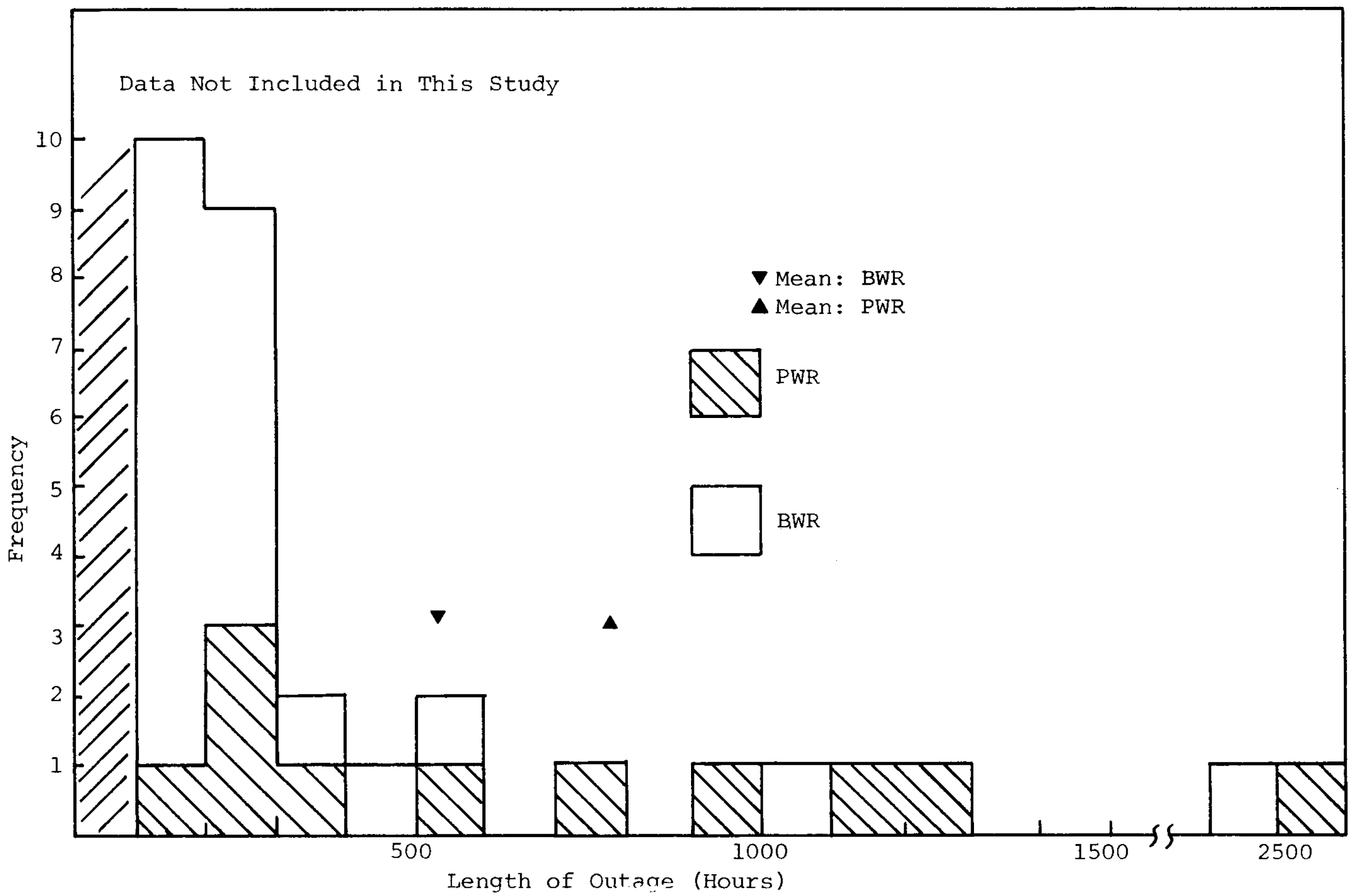


Figure 3.33. Frequency Histogram of Major Outages in PWRs and BWRs Related to Piping

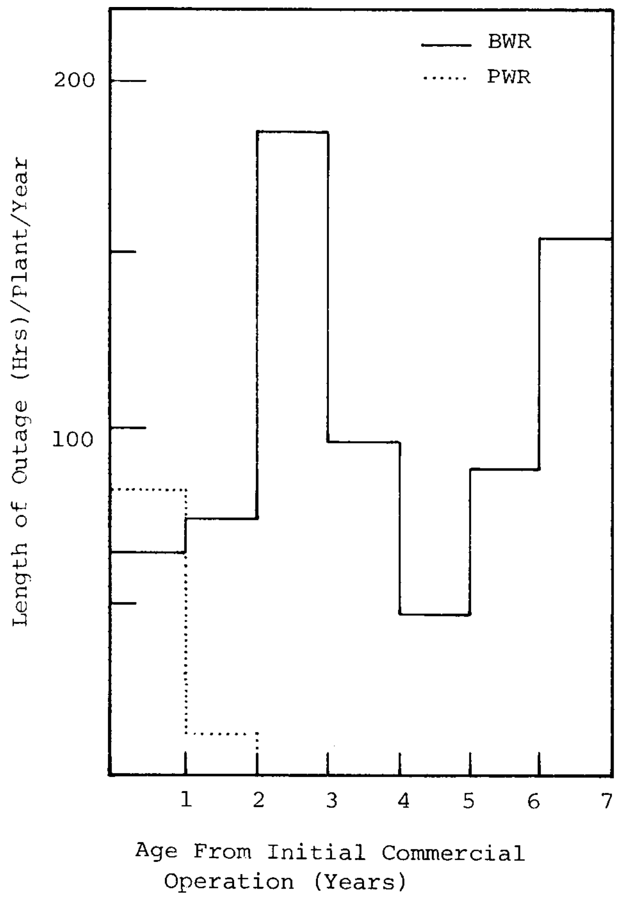


Figure 3.34. Comparison of the Major Outage Time Related to Piping on a Per Plant Basis as a Function of Age of Commercial Operation

The BWR pipe cracks, caused by stress corrosion cracking of the core spray pipe or feedwater sparger inside the reactor vessel, have led to outages of approximately 7643 hours or 3.4% of the total of long duration outages.

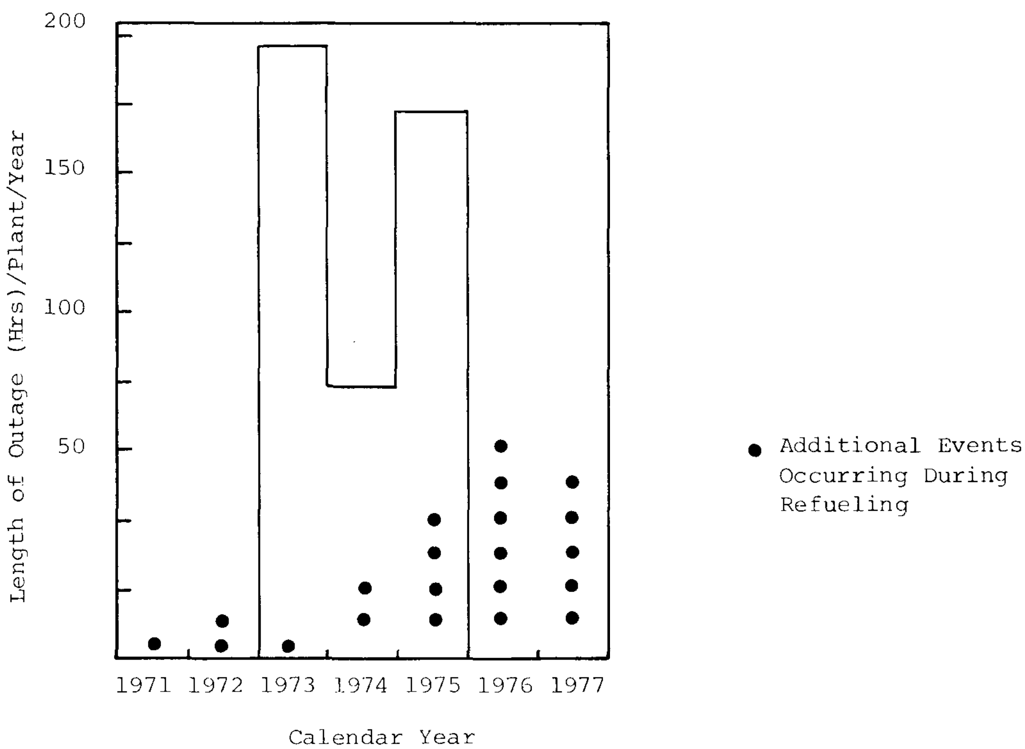


Figure 3.35. BWR Outages Due to Sensitized Stainless Steel Pipe

A previous report(15) on pipe failures considered all instances of pipe failures* whether they caused an outage or not. This current report focuses only on those related** incidents which have caused major outages.

*This does not necessarily imply a failure of pipe, rather it includes outages for pipe arrangement or modification such as at Turkey Point.

**Failures includes defects, anomalies, leaks or breaks.

SECTION 4.0

SUMMARY OF MAJOR OUTAGES GREATER THAN TWO MONTHS IN DURATION

"Rare" or "unique" occurrences in nuclear power plants, such as core modification, turbine failures, or steam generator failures, have caused lengthy outages in several nuclear plants. While these events may be new to a given plant, similar incidents may have occurred at other facilities.

One of the key factors in limiting the length of an outage is in proper planning of the work required. This planning includes the following general phases:

a) Preparation of procedures
b) Acquisition of spare parts
c) Availability of skilled and trained personnel
d) Monitoring of the critical path jobs

Unless plant management can learn from the experiences of other utilities, the learning cycle will be repeated in each plant which encounters a "rare" event. The events which are discussed in this section are shown in the frequency histograms (Figures 4.1 and 4.2) displaying all of the PWR and BWR events considered in this study. The following are summaries of the longest outages (excluding refuelings) which have impacted on plant availability.

<u>Arkansas 1 (PWR)</u>: March 1976. This was a 3 month (2221 hrs.) outage to inspect the failure of reactor in-vessel surveillance specimen holder tubes. Arkansas, which is a B&W PWR, encountered a generic problem occurring in a number of B&W plants (Three Mile Island, Oconee 1,2, and 3, and Rancho Seco). The problem affects the reactor's surveillance capsule tubes, which have been found to be worn or, as at Arkansas, broken off and fallen to the bottom of the reactor vessel. The surveillance tubes have been installed on the outside of the core internals structure so that samples of pressure vessel material could be exposed to a neutron flux similar to that which the vessel "sees".

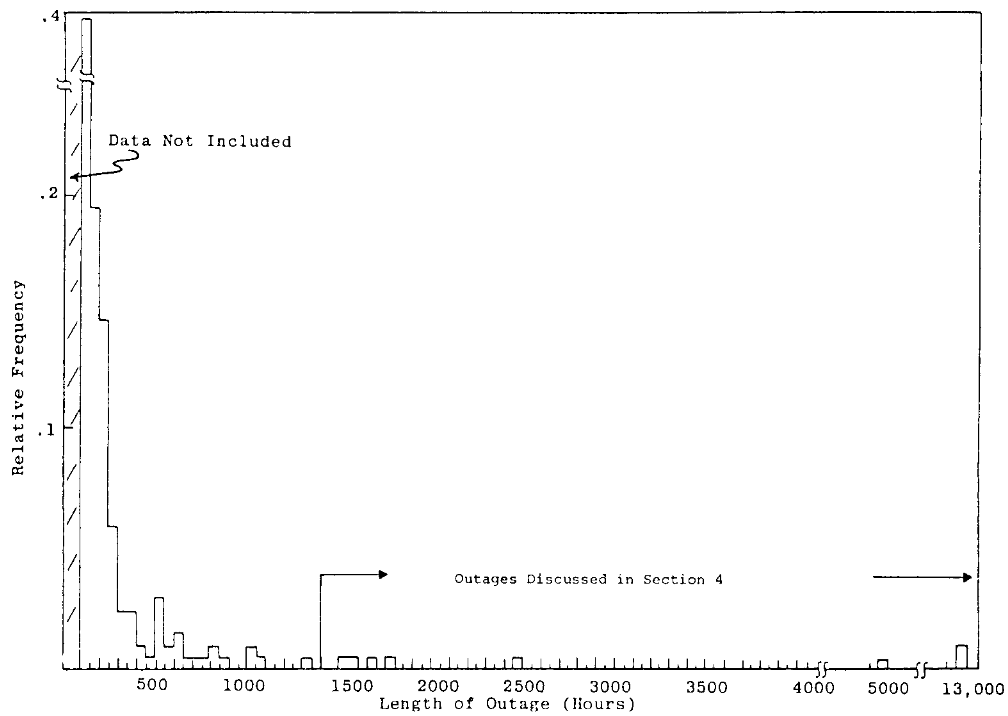


Figure 4.1 Frequency Histogram of BWR Major Outages Occurring From January 1971 through June 1977

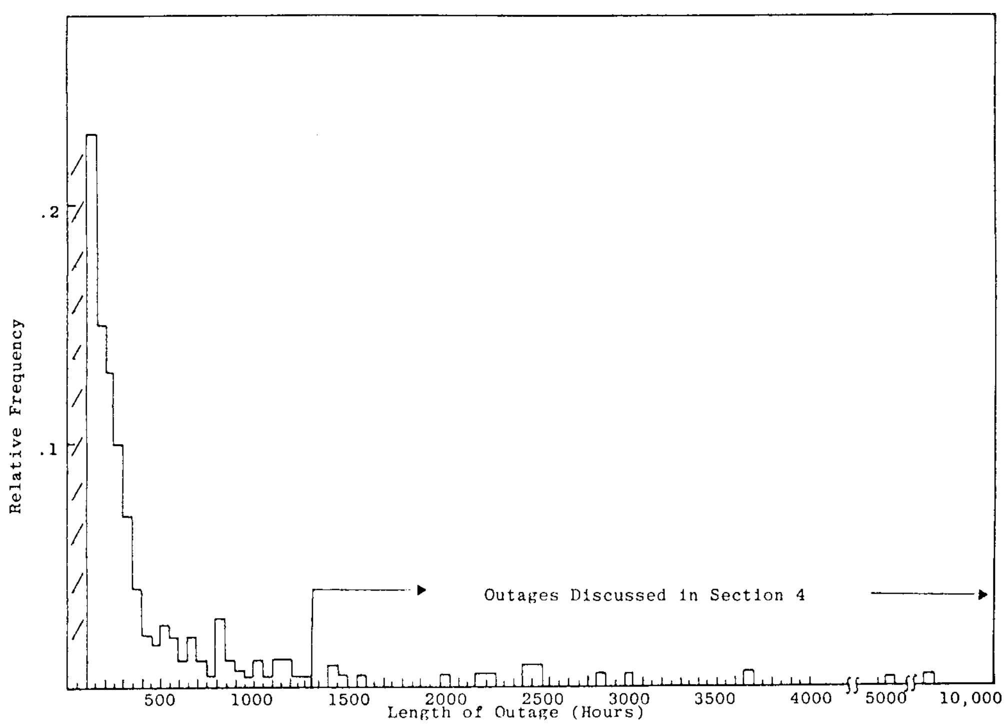


Figure 4.2 Frequency Histogram of Major PWR Outages Occurring From January 1971 through June 1977

The tubes serve no operational purpose and are used only to "hold" material samples for the purpose of monitoring the properties of the reactor pressure vessel materials. Flow-induced vibration caused excessive vibration, wear, and failures in the surveillance tubes.

Brown's Ferry 1 and 2 (BWR): March 1975 - September 1976. This eighteen month outage of two 1000 MWe plants was a major loss in electrical generating capacity and accounted for the loss of more than 6% of all outage time between June 1974 and June 1977. The equivalent energy loss is approximately 55 million barrels of oil. The outage was caused by a fire in the cable spreading room which contained cables for both units. The recovery procedure included extensive redesign of the cable routing to reduce the probability of such an extensive outage in the future.

The Brown's Ferry plant consists of three boiling water reactors, each designed to produce 1067 megawatts of electrical power. Units 1 and 2 were both operating at the time of the fire. Unit 3 was still under construction.

Units 1 and 2 shared a common control room with a cable spreading room located beneath the control room. Cables carrying electrical signals between the control room and various pieces of equipment in the plant including monitoring and control cabling were routed through the cable spreading room.

The immediate cause of the fire was the ignition of the polyurethane foam which was being used to seal cable penetrations between the Unit 1 reactor building and the cable spreading room located beneath the control room of Units 1 and 2. The material ignited when the flame from a candle which was being used to test the penetration for leakage was drawn into the foam by the air flow through the leaking penetration.

Following ignition of the polyurethane foam, the fire propagated through the penetration in the wall between the cable spreading room and the Unit 1 reactor building. In the cable room, the fire was of limited extent and was controlled by a combination of the installed carbon dioxide extinguishing system and manual fire fighting efforts.

In addition to the cable damage, the burning insulation created a dense soot which was deposited throughout the Unit 1 reactor building and in some small areas in the Unit 2 reactor building. The estimated 4,000 pounds of polyvinyl chloride insulated cable which burned also released an estimated 1400 pounds of chloride into the reactor building. Following cleaning, all exposed surfaces of piping, conduit, and other equipment were examined for evidence of damage. Piping surfaces where soot or other deposits were noted were examined by dye penetrant procedures. With the exception of some small (3 and 4 inch diameter), uninsulated carbon steel piping, one run of aluminum piping, heating and ventilation ducts, and copper instrument lines in or near the fire zone, no evidence of significant chloride corrosion was found. Where such evidence was found, the material affected was replaced. For some stainless steel instrument lines, an accelerated inspection program has been established to determine if effects of chloride may later appear.

The Brown's Ferry outage included all phases of a major outage which have potential safety overtones:

a) Clean-up and repair work
b) Redesign
c) Testing
d) Regulatory review to assure safe operation

Each of these phases takes detailed planning and aggressive management action to minimize the outage.

Brunswick 1 (BWR): April 1977. After entering commercial operation in March 1977, Brunswick 1 shut down for more than 2 months (1698 hours) due to a ground in the generator windings. A new stator was installed and the rotor rewound.

Fitzpatrick and Brunswick 2 (BWRs): Generic design deficiencies in the BWR/4 core design which could cause excessive vibration of the core instrument tubes (local power range monitors-LPRM) were identified. GE reported that the coolant flow through by-pass holes in the core support plate was inducing movement of the instrument components causing wear and cracking. The NRC directed 10 BWRs to review recent test results from their plants to determine if anomolous behavior was occurring. The design solution was to plug the core support plate bypass holes. Modifications to the core plate were made to prevent vibrations (outage: 1525 and 1578 hours).

The other plants affected by this generic problem were:

- Cooper
- Pilgrim
- Arnold
- Peach Bottom 2
- Peach Bottom 3
- Hatch 1
- Brown's Ferry 1
- Vermont Yankee

Indian Point 2 (PWR): October 1973. Shortly after beginning commercial operation, Indian Point 2 was shut down due to a relay failure. In the course of preparing the reactor for resumption of operation, a main feedwater line to one steam generator failed (180° circumferential crack). The steam impinging on the inside of the containment steel liner caused it to buckle. The outage extended for three and one-half months (2532 hours).

Unit 2 reactor at the Indian Point Station was critical at approximately 7% of rated power when a turbine trip occurred due to high feedwater level in the steam generator. The turbine trip caused the main boiler feed pump to trip which in turn caused the water level in all four steam generators to decrease. The two motor-driven auxiliary boiler feed pumps started. The reactor then tripped due to low level in a steam generator. Shortly thereafter, the feedwater line to one of the steam generators vibrated excessively. The operators were not able to maintain the proper water level in that steam generator and noted an increasing water level in the containment sump. An inspection revealed water on the containment floor and a 180° circumferential crack in the feedwater line to the steam generator. The crack was adjacent to the weld that attached the pipe to the containment penetration pipe sleeve.

The failed 18" diameter pipe was removed and examined at Consolidated Edison's metallurgical laboratory. There was no indication of material defects. The steam generator internals were inspected and found to be intact. Bulges in the containment liner, resulting from the steam released from the cracked pipe, were examined ultrasonically to determine the extent of the damage, and all piping in the steam generator system was examined and tested.

An inspection of the regulating valve for the feedwater system showed damage to the internal guide for the valve plug. It is postulated that excessive stresses were caused by water-steam interactions in the feedwater line.

Principal corrective actions included modifications of the feedwater inlet pipe, feedwater regulating valves, and control system. Additional pipe whip restraints and braces were installed, and the affected pipe was replaced. These modifications were tested prior to subsequent operations.

Palisades (PWR): August 1973. After operating for approximately 18 months at a capacity factor of 55%, the Palisades plant was shut down to repair leaks in steam generator tubing. Investigation of previous abnormal nuclear instrumentation fluctuations resulted in the identification of internal core vibration which required extensive repair. The core barrel support flange was extensively worn due to excessive flow-induced vibration.[17] Several fasteners inside the vessel were broken, and extensive inspections and repair action were required. The outage lasted 1.1 years (9890 hours). One "fix" to the core vibration problem was to increase the closure head preload by clamping the core barrel more securely (from ~4000 lb to ~7×10^6 lb). In this report, because of a lack of specific information on the percentage of critical path time assigned to each task, we have arbitrarily divided this outage into three outages: a) 4500 hours for internal core repair; b) 4500 hours for steam generator repair; and c) 890 hours for repair of the turbine blading which was damaged during recovery startup.

Palisades (PWR): November 1974. After less than two months of operation, Palisades was again shut down to:

a) Replace condenser tubing

b) Repair turbine blading (the low pressure turbine was rebladed)

This outage lasted five months (3660 hours).

Rancho Seco (PWR): July 1975. Beginning commercial operation in April 1975, Rancho Seco operated at approximately 90% capacity factor for 2.5 months. In July 1975, a thrown blade in the low pressure turbine caused a shutdown. Cracks were found in other low pressure turbine blades. The cause of the turbine damage was attributed to stress corrosion by sodium hydroxide carry-over from the steam generator. The total outage extended 9 months (5570 hours); however, the last 1.6 months was due to moisture in the electrical generator.

San Onofre 1 (PWR): October 1973. San Onofre operated for 5½ years at a capacity factor of approximately 70 to 75% until October 1973 when it was discovered that low pressure turbine blades had failed due to fatigue-induced fractures at the root of the blading. All last stage blading was replaced. The outage extended for 4 months (3006 hours).

Surry 1 (PWR): December 1973. Following one year of commercial operation at a capacity factor of approximately 50% (most of the power operation was above 80% power), Surry 1 was shut down due to the failure of a reactor coolant pump shaft. The outage extended 3½ months (2529 hours).

While operating at 95% of rated power, a reduction of primary coolant flow accompanied by excessive vibrations in the "A" coolant loop was noted in the Unit 1 reactor of the Surry Power Station.

Inspection of the pump internals revealed that the pump shaft had severed at a machined change in the shaft diameter. The diameter of the broken shaft was approximately 9 inches. Examination of the fracture surface showed characteristics of fatigue failure due to a relief groove radius which was too small (.040 in. versus a specified value of .2"). The broken shaft and impeller were sent to Westinghouse for metallurgical examination.

Virginia Electric Power Company (VEPCO) analysis indicated that the departure from nucleate boiling ratio (DNBR) for an unisolated idle pump with two operating loops was greater than 3.0 and that no fuel damage occurred. There was no increase in coolant activity following the occurrence.

VEPCO replaced all pump shafts of the same design at the Surry Station with those of a different design configuration in the area of the break. The only other pumps manufactured with the same design, located at another station, have been rebuilt.

Surry 2 (PWR): September 1976. Surry 2 operated for 3½ years at a capacity factor of approximately 60%. In September 1976, Surry 2 was shut down for steam generator tube repair and Eddy Current testing. The outage extended 3.2 months (2286 hours). This problem has culminated in the decision by Virginia Electric Power to plan on replacement of all the steam generators in the two Surry plants.

The trouble is caused by "denting" or circumferential pinching of the steam generator tubes by the baffle through which they pass. The pinching is caused by corrosion of the carbon steel of which the support plates are made. As the steel around the tubes corrodes, it expands and pinches the tubes. If the pinching is severe enough, the tubes develop ridges subject to high stress, which leads, in some cases, to cracked and leaking tubes. Leakage at Surry 1 reached 80 gal/min at one stage.

Thus far, the Turkey Point and Surry Westinghouse steam generators are the worst affected by denting. They were the first units of their vintage and operated for some time on phosphate water chemistry. When this was generally found to cause accelerated corrosion, pitting, and stress corrosion cracking of steam generator tubes at several units, the water chemistry was switched to volatile amine treatment (AVT). In addition, the four units most affected use brackish water for condenser cooling, and significant salt incursions have occurred. Westinghouse is considering changing its steam generator tube support plates from carbon steel to a material compatible with the tubes in order to prevent adverse interactions between the tubes and the support plates.

Surry 2: September 1974. The reactor was scrammed by a turbine trip due to high turbine vibration. During the outage, some of the low pressure turbine blading was replaced or modified.

Zion 1: January 1974. A generator short required reactor shutdown and disassembly of the generator. Investigation of the failure at the factory revealed that the cause of the short was the intrusion of moisture into the coil insulation. The moisture in-leakage occurred through a failed braze joint.

Yankee Rowe (July 1961): October 1972. Yankee Rowe, a Westinghouse reactor operated for more than 11 years until October 1972 when it shut down for control rod replacement. The outage extended for 4104 hours.

SECTION 5.0

CONCLUSIONS

The nuclear power experience data over the years 1971 through 1977 indicate a number of important concepts which are summarized as follows:

1) Plant unavailability can be attributed to three broad classes of events: refuelings, major outages greater than 100 hours, and short outages less than 100 hours. The approximate split in plant unavailability among these categories is, respectively, 40%, 40%, and 20%.

2) The major outages are approximately evenly divided between the Nuclear Steam Supply System (NSSS) and the balance of the plant (BOP).

3) The major plant outages observed in LWR operating experience from January 1971 through June 1977 can be ranked according to the principal system involved in the outage or cause as follows:

Rank	System	% of Major Outage Time
1	Steam System (includes Turbine)	18.9%
2	Steam Generators	13.9%
3	Reactor Related	13.3%
4	Fire	11.7%
5	Reactor Coolant Pumps	9.5%
6	Electrical Systems (includes generator)	8.4%
7	Safety-Related Systems	5.3%
8	Condensers	5.2%
9	Feedwater Systems	5.0%
10	CRDMs and Closure Head Seals	3.7%

4) Alternatively, the major outages can be categorized according to the type of component involved in the outage, as follows:

Rank	Component	% of Major Outage Time
1	Turbine Generator	18.5%
2	Steam Generator	13.9%
3	Reactor Core	9.8%
4	Pumps	9.8%
5	Valves	8.1%
6	Pipe	6.5%
7	Condensers	5.2%

5) From Appendix I, a review of the individual plants indicates that many plants have recurring problems related to a particular system (e.g., steam system) or component (e.g., reactor coolant pumps). While the same problems do not occur at all plants, a limited number of other plants do face similar recurring problems. These problem areas include:

a) steam generator tube leaks and failures
b) local power range monitor failures
c) recirculation/core spray pipe inspection/repair
d) reactor coolant pump seal failure
e) snubber inspection/repair
f) condenser tube failure

Recurring problems are not easily eliminated when they are related to high capital cost equipment such as the steam generators or turbines. Because of the long lead time of these pieces of equipment, design changes are not easily back fitted to components already on order.

6) Possibly more important than the above ranking is the trend of the major outage time. It was found that the most meaningful method of trending the outage data is versus the age of the plant. The result of this comparison is shown in Section 3 and indicates a dramatic decrease in major outages over the first three years of commercial operation. Figure 5.1 shows the overall trend in major outages versus plant age compared

with the refueling outage time. This variation is consistent with previous predictions by the industry that a "break-in" period for nuclear plants was to be expected. The limited data available from the fourth through the seventh year indicates that there may be a leveling off of the outage time per plant, and this constant outage rate should be an anticipated element of the plant maintenance and availability planning. That is, an ideal maintenance program would recognize that there may be some necessary work requiring a plant outage between refuelings. Anticipating this fact, the utility can properly plan for such an outage to deal with these items in an organized and prepared manner. The timing and length of such an outage is ideally set by the utility's backlog of work orders, desired preventative maintenance, and trouble shooting requirements.

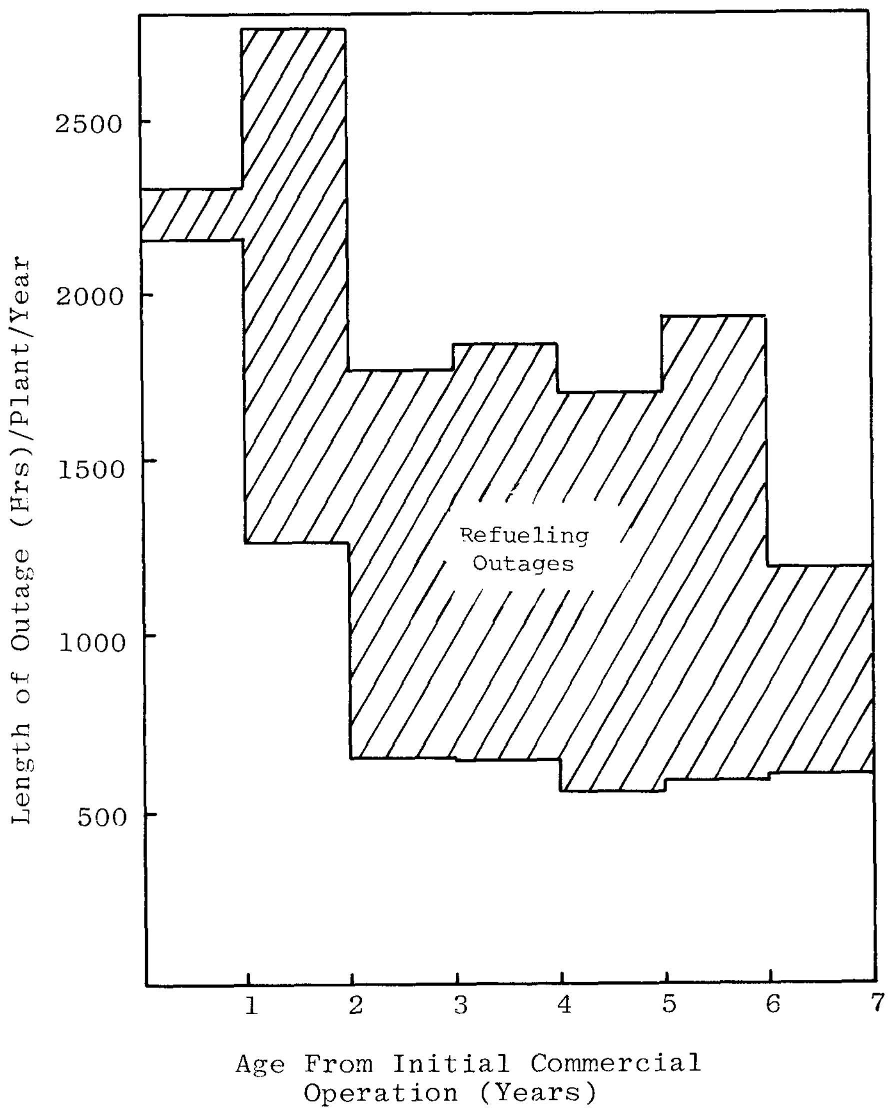
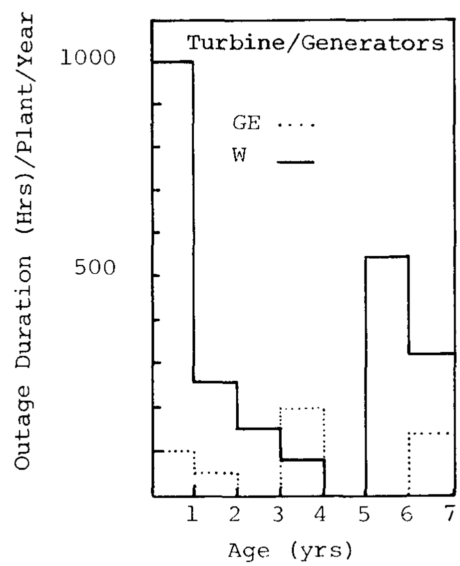


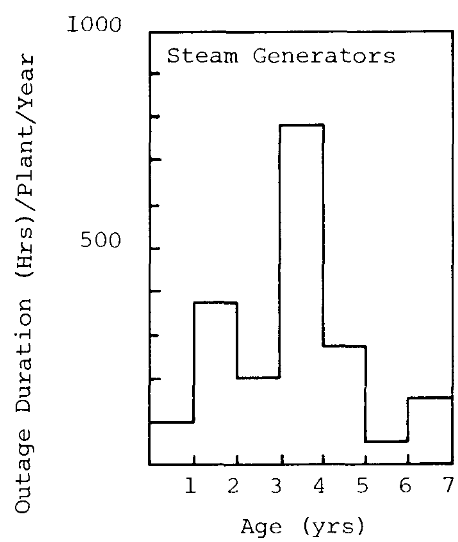
Figure 5.1. Distribution of LWR Outages as a Function of Plant Age

7) The overall trend determined for major outages during the period 1971 - June 1977 is a composite made up from many different contributing causes, each with its own distinctive trends as a function of plant age. The principal contributing components can be summarized as follows:

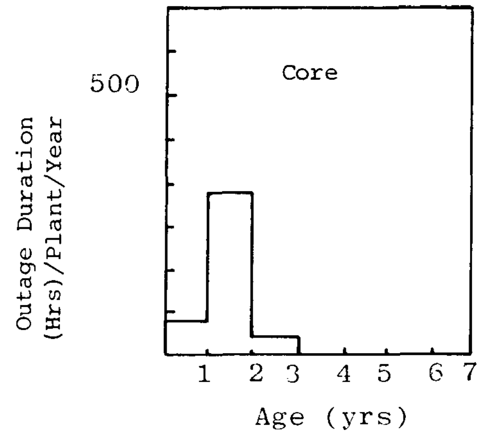
- TURBINE GENERATORS. A major portion of the outage time at nuclear plants has been due to turbine/generator related problems. This is similar to the experience in fossil fuel plants, and therefore it should not be a surprise, despite the wide disparity in operating conditions. In particular, steam turbine failures have caused a large percentage of the major outages required during the first year of commercial operation. However, of even more interest is the apparent trend of increasing turbine related problems in the sixth and seventh years.



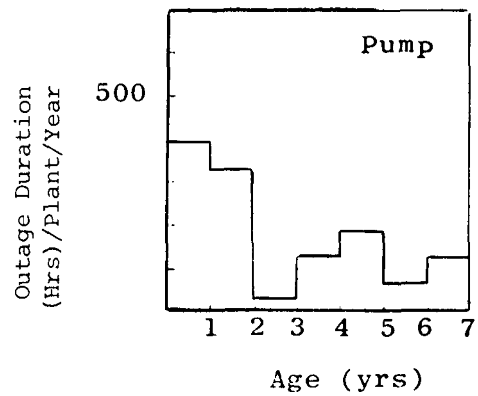
- STEAM GENERATORS. PWR plants use the thin-walled tubes in steam generators to separate the primary system (inside containment) from the secondary system (outside containment). Steam generator design for nuclear plant application is still an evolving technology, and a number of difficulties have plagued steam generators. Leakage of primary water (radioactive) into the secondary system through cracks in the thin-walled tubes has led to: a) tube plugging, b) extensive inspections (Eddy Current), and c) changes in secondary water chemistry. All of these operations have contributed to plant outages. From the data, it appears that for individual plants an incubation period measured in years is necessary before tube failures surface as a problem.



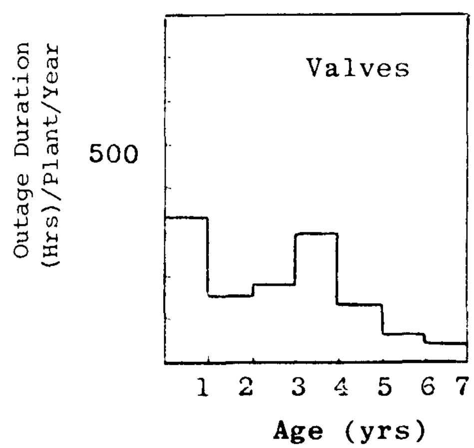
• CORES. Reactor core related problems have tended to be of a generic nature: they have generally been caused by design errors incorporated into several units. Most core related repairs have been performed during refueling operations. For those cores which were between refuelings, extended outages have been required to modify the design. The actual failure mechanism has been flow-induced vibration. Note that these problems have characteristically surfaced during the second year of operation.



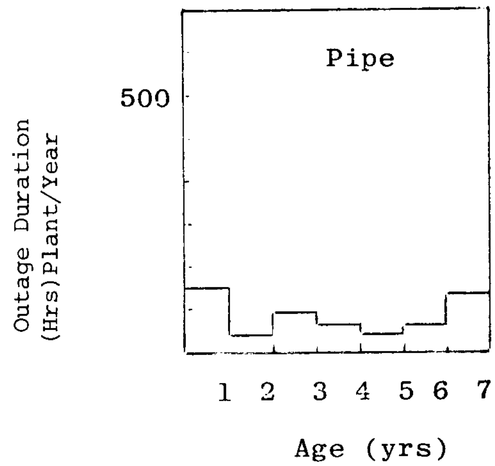
• PUMPS. Reactor coolant pumps in PWRs and recirculation pumps in BWRs have been the cause of a number of major outages, particularly early in plant life. Subsequent failures have been primarily related to pump seal problems.



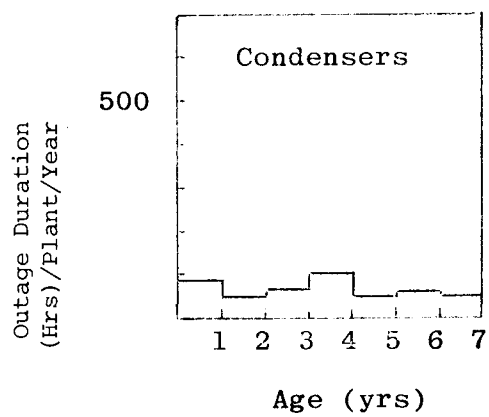
• VALVES. Considering the large number of valves in nuclear plants, the number of major outages related to valve problems has been small. However, there are a substantial number of short duration outages which are not included in this report.



• PIPE. Major outages attributable to piping have a low frequency but may have a high impact on the plant which is affected. All of the PWR major outages attributed to pipe related problems occurred within the first two years of commercial operation, while BWR problems generally occurred throughout the seven year period of this study. A large number of these BWR pipe failures were related to stress corrosion cracking of austenitic stainless steel.



• CONDENSERS. Condenser tube failures have not had a large, direct impact on plant performance in the form of major outages. However, as noted by the accompanying figure, the trend of major outages appears to be constant with the plant age, indicating a continuing problem.



8) Over the period January 1971 to June 1977, there were 13 plants which incurred non-refueling major outages greater than two months duration. The principal causes of these exceptionally long "rare" events were:

Event	No. of Plants
In-core problem	5
Fire	2
Generator	2
Turbine Blade Failure	2
Steam Generator Inspection/ Repair	1
Feedwater Pipe Failure	1

9) Based upon a review of each major outage, it can be stated qualitatively that much of the high outage time during the initial two years of commercial operation is the result of the introduction of a new design in an untested area. Typical problems include:

a) turbine blade vibration problems

b) core vibration problems

Because of limited operating experience, many of the problems are just now being solved and fed back into the design cycle, and therefore they may continue to cause outages for a number of years.

10) In assessing the trends in nuclear power plant outages, it is necessary to look at the variations in number and duration of outage as a function of a variety of parameters. This study has focused on the variation of outages at LWRs as a function of the age of the unit. This parameter seems to be the most reasonable criteria for assessing overall trends as a function of time. However, there have been some types of major outages which are primarily a function of the calendar year (i.e., recirculation bypass pipe inspection in BWRs in 1974 and 1975; and snubber inspection of PWRs and BWRs in 1973 and 1974 which were initiated as much by regulatory edict as by technical necessity). In general, the utility, if given the choice, chooses to avoid a special shutdown for operations not requiring immediate attention. The utility establishes inspections on a priority basis to be accomplished at the earliest available outage.

11) As has been discussed previously by various analysts[4,7,8] in assessing nuclear power plant performance, it is hypothesized that nuclear plant performance improves with age. From the current study it is clear that as nuclear plant age increases, the unavailability time due to major outages decreases. The trend of major outages is an important input for decisions relating to utility planning for maintenance, refuelings, and investment in additional power plant capacity.

SECTION 6.0

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APPENDIX A

SUMMARY OF MAJOR OUTAGE HISTORY BY PLANT

The following are profiles of long duration outages (hours) for each plant included in this report (i.e., plants with ratings larger than 150 MWe).

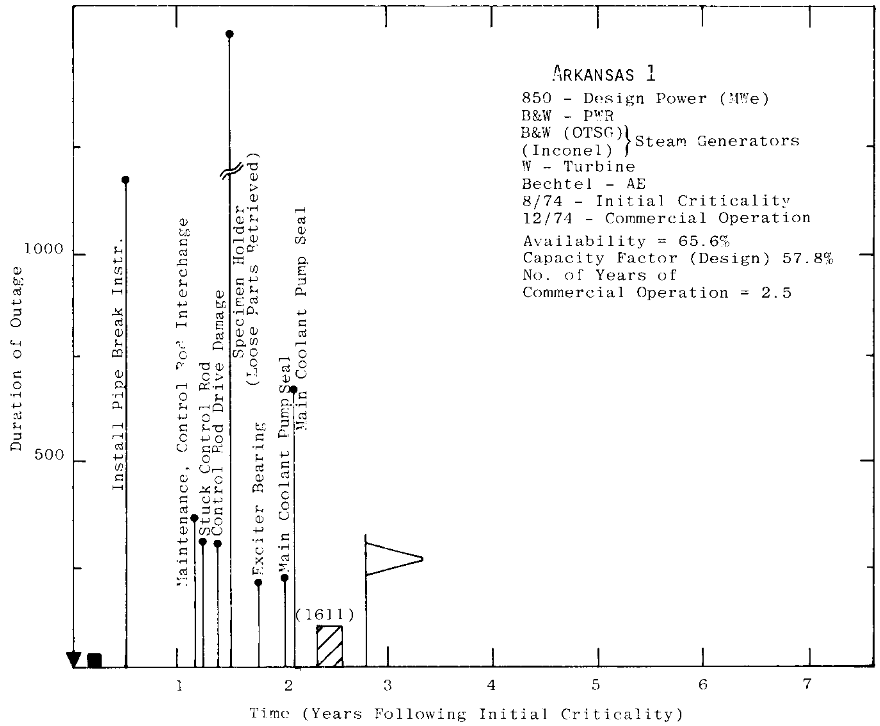
Legend of Symbols Used in This Appendix

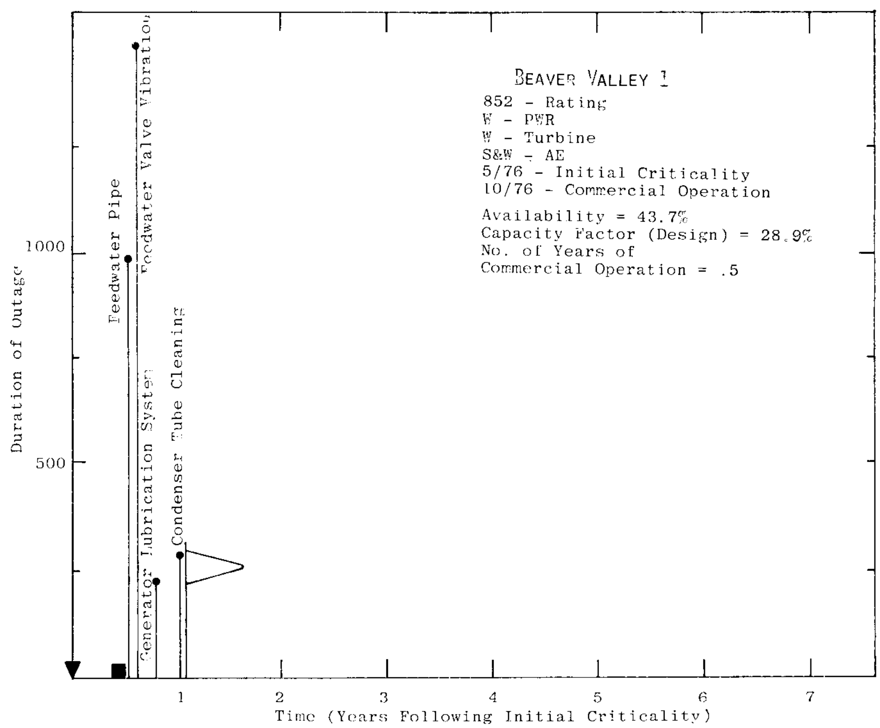
▼	- Initial Criticality
■	- Commercial Operation
	- Refueling
◆	- Beginning of the Use of All Volatile Chemistry (AVT) in the Secondary Water Chemistry (PWRs only)
	- End of Data Used in this Study; June 31, 1977
( )	- Duration of Refueling in Hours
< >	- Reasons for Extension of Refueling

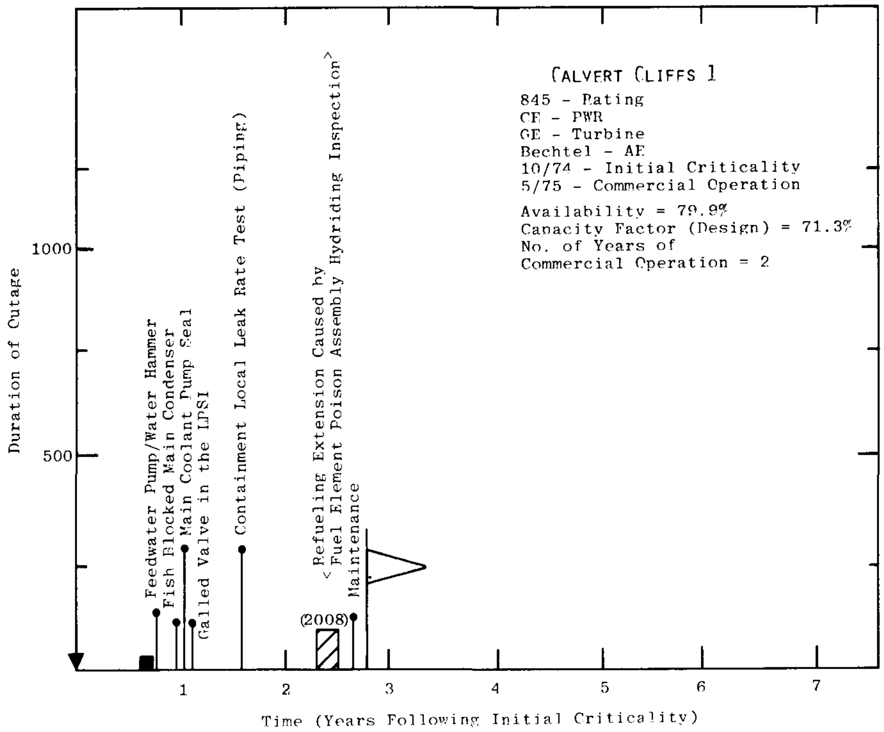
Vendor Abbreviations	
W	- Westinghouse
GE	- General Electric
CE	- Combustion Engineering
B&W	- Babcock and Wilcox

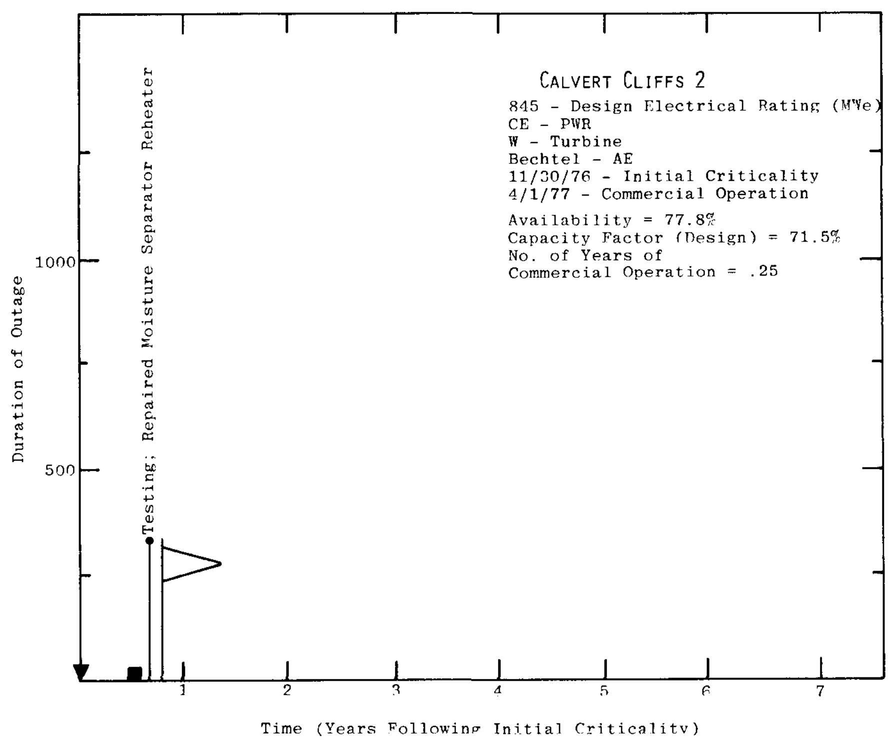
Plant Performance Parameters	
Availability	- Cumulative Plant Availability Through June 1977
Capacity Factor	- Cumulative Plant Capacity Factor Through June 1977 Based upon the Design Electrical Rating

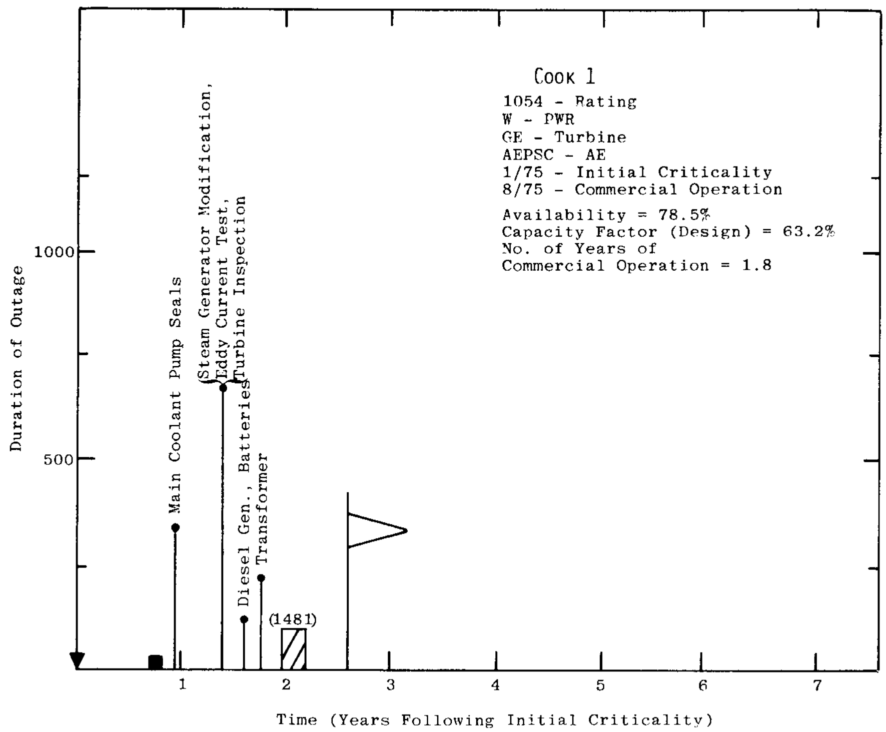
Architect Engineer Abbreviations		
AEPSC	-	American Power Service Corporation
B&R	-	Burns and Roe
Bechtel	-	Bechtel Corporation
Ebasco	-	Ebasco
Gilbert	-	Gilbert
G&H	-	Gibbs and Hill, Inc.
FPI	-	Flour Pioneer, Inc.
S&L	-	Sargent and Lundy Engineers
S&W	-	Stone and Webster
UE&C	-	United Engineers & Constructors

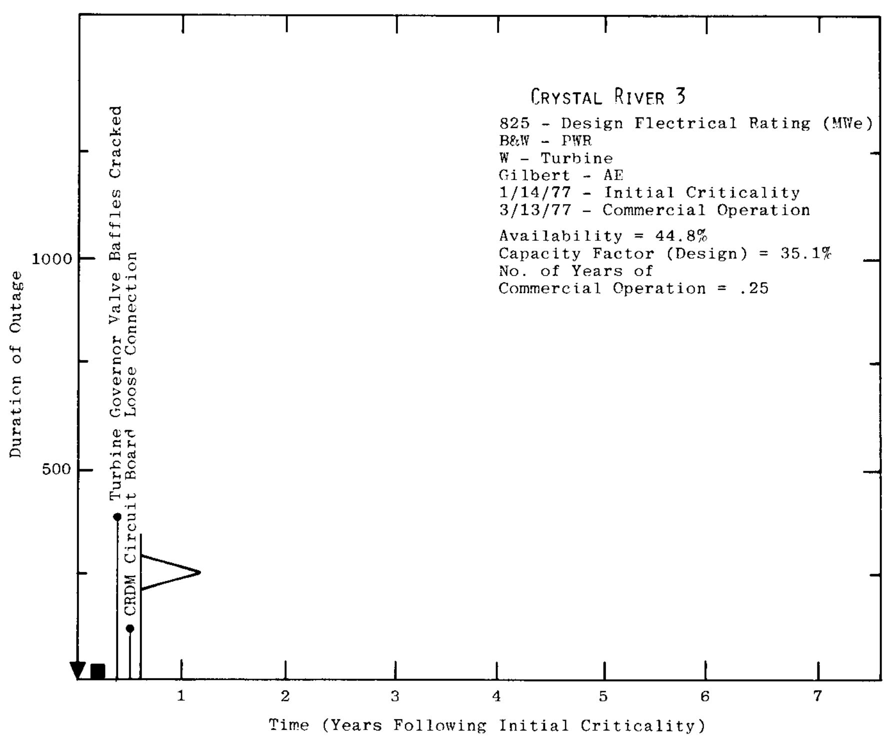

ARKANSAS 1
850 - Design Power (MWe)
B&W - PWR
B&W (OTSG) (Inconel) } Steam Generators
W - Turbine
Bechtel - AE
8/74 - Initial Criticality
12/74 - Commercial Operation
Availability = 65.6%
Capacity Factor (Design) 57.8%
No. of Years of
Commercial Operation = 2.5
Install Pipe Break Instr.
Maintenance, Control Rod Interchange
Stuck Control Rod
Control Rod Drive Damage
Specimen Holder
(Loose Parts Retrieved)
Exciter Bearing
Main Coolant Pump Seal
Main Coolant Pump Seal
(1611)
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

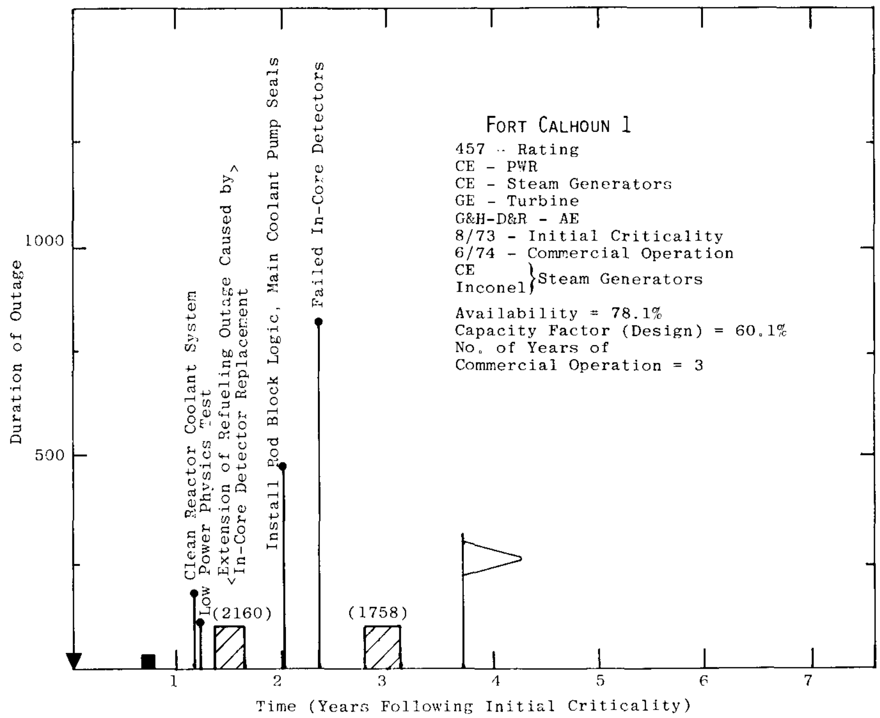

BEAVER VALLEY 1
852 - Rating
W - PWR
W - Turbine
S&W - AE
5/76 - Initial Criticality
10/76 - Commercial Operation
Availability = 43.7%
Capacity Factor (Design) = 28.9%
No. of Years of
Commercial Operation = .5
Feedwater Pipe
Feedwater Valve Vibration
Generator Lubrication System
Condenser Tube Cleaning
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


CALVERT CLIFFS 1
845 - Rating
CE - PWR
GE - Turbine
Bechtel - AE
10/74 - Initial Criticality
5/75 - Commercial Operation
Availability = 79.9%
Capacity Factor (Design) = 71.3%
No. of Years of
Commercial Operation = 2
Duration of Outage
1000
500
Feedwater Pump/Water Hammer
Fish Blocked Main Condenser
Main Coolant Pump Seal
Galled Valve in the LPSI
Containment Local Leak Rate Test (Piping)
(2008)
<Refueling Extension Caused by
Fuel Element Poison Assembly Hydriding Inspection>
Maintenance
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

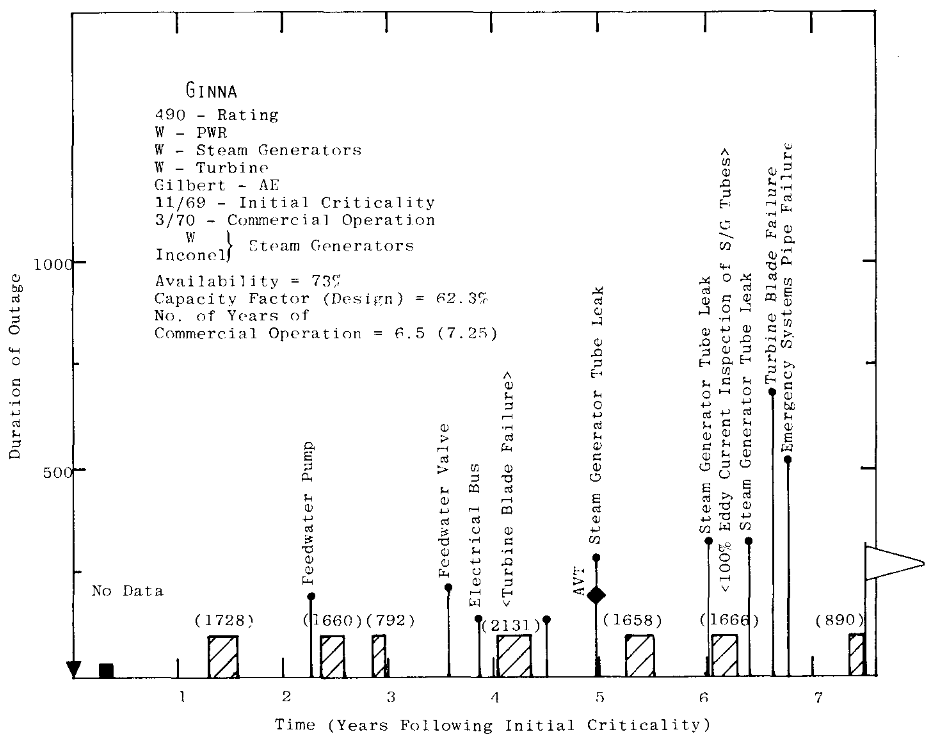

CALVERT CLIFFS 2
845 - Design Electrical Rating (MWe)
CE - PWR
W - Turbine
Bechtel - AE
11/30/76 - Initial Criticality
4/1/77 - Commercial Operation
Availability = 77.8%
Capacity Factor (Design) = 71.5%
No. of Years of
Commercial Operation = .25
Duration of Outage
1000
500
Testing; Repaired Moisture Separator Reheater
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


Cook 1
1054 - Rating
W - PWR
GE - Turbine
AEPSC - AE
1/75 - Initial Criticality
8/75 - Commercial Operation
Availability = 78.5%
Capacity Factor (Design) = 63.2%
No. of Years of
Commercial Operation = 1.8
Main Coolant Pump Seals
Steam Generator Modification,
Eddy Current Test,
Turbine Inspection
Batteries
Diesel Gen., Transformer
(1481)
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

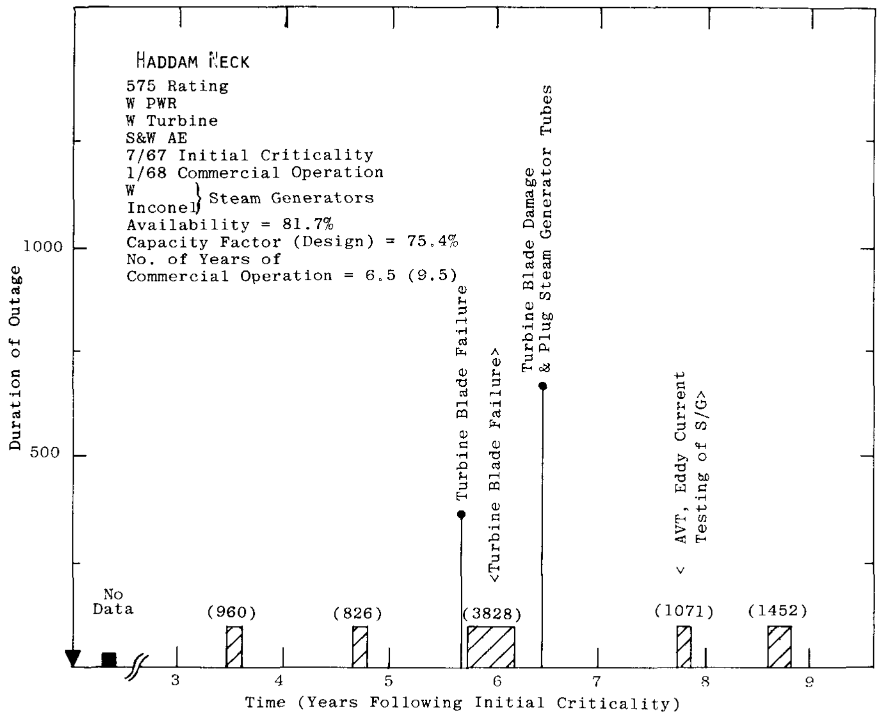

Crystal River 3
825 - Design Electrical Rating (MWe)
B&W - PWR
W - Turbine
Gilbert - AE
1/14/77 - Initial Criticality
3/13/77 - Commercial Operation
Availability = 44.8%
Capacity Factor (Design) = 35.1%
No. of Years of
Commercial Operation = .25
Turbine Governor Valve Baffles Cracked
CRDM Circuit Board Loose Connection
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

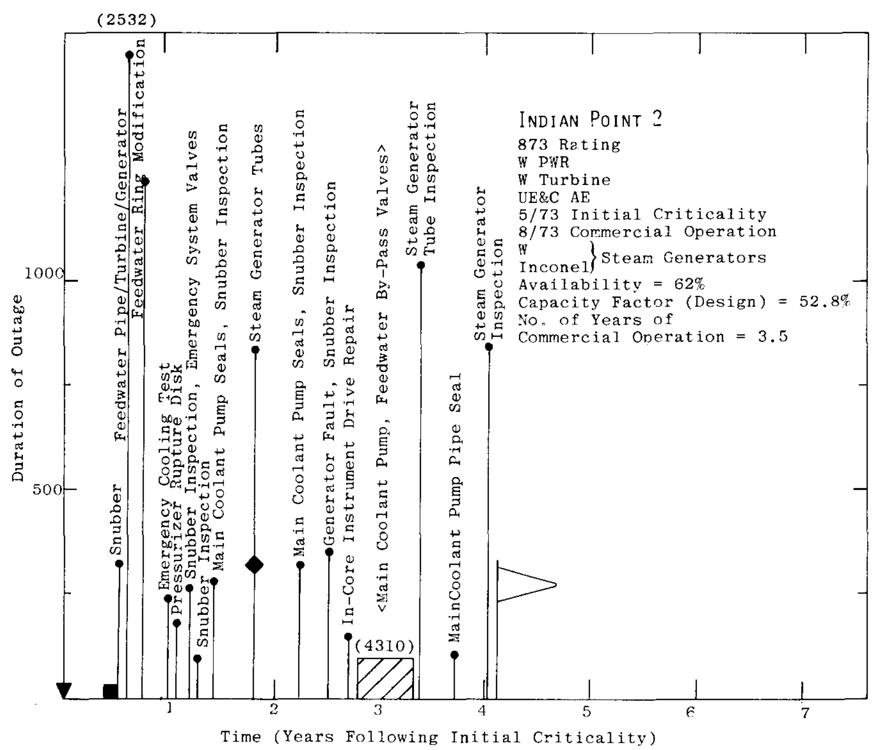


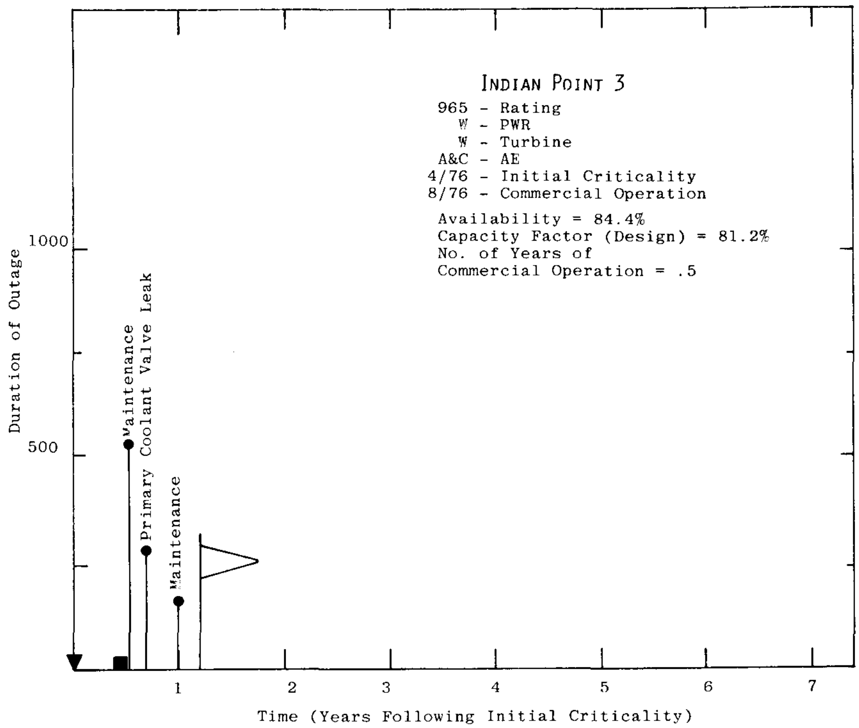
FORT CALHOUN 1
457 - Rating
CE - PWR
CE - Steam Generators
GE - Turbine
G&H-D&R - AE
8/73 - Initial Criticality
6/74 - Commercial Operation
CE
Inconel } Steam Generators
Availability = 78.1%
Capacity Factor (Design) = 60.1%
No. of Years of
Commercial Operation = 3
Clean Reactor Coolant System
Low Power Physics Test
<Extension of Refueling Outage Caused by>
<In-Core Detector Replacement
Install Rod Block Logic, Main Coolant Pump Seals
Failed In-Core Detectors
(2160)
(1758)
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

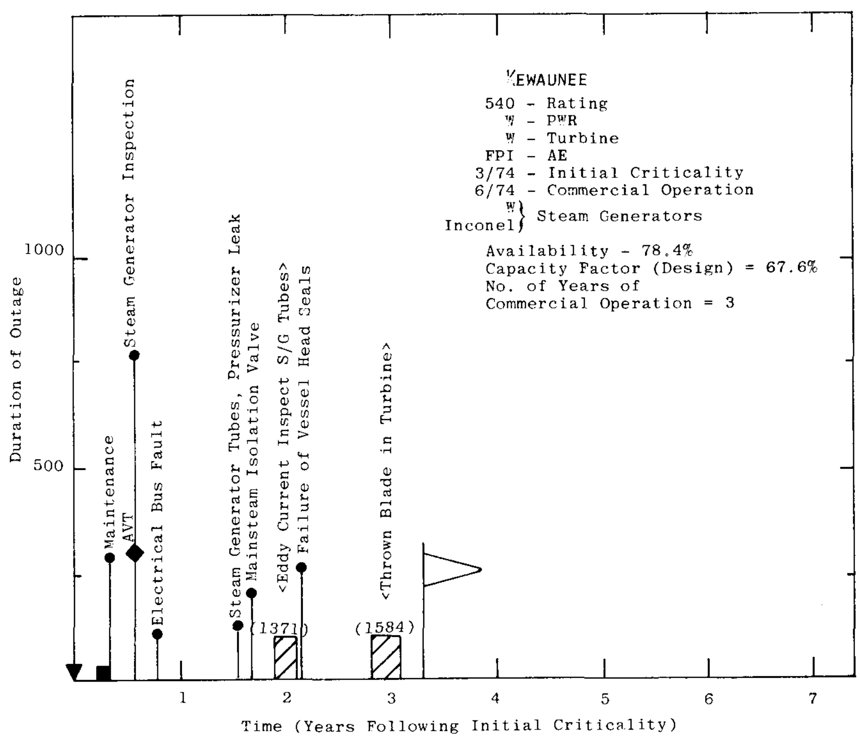


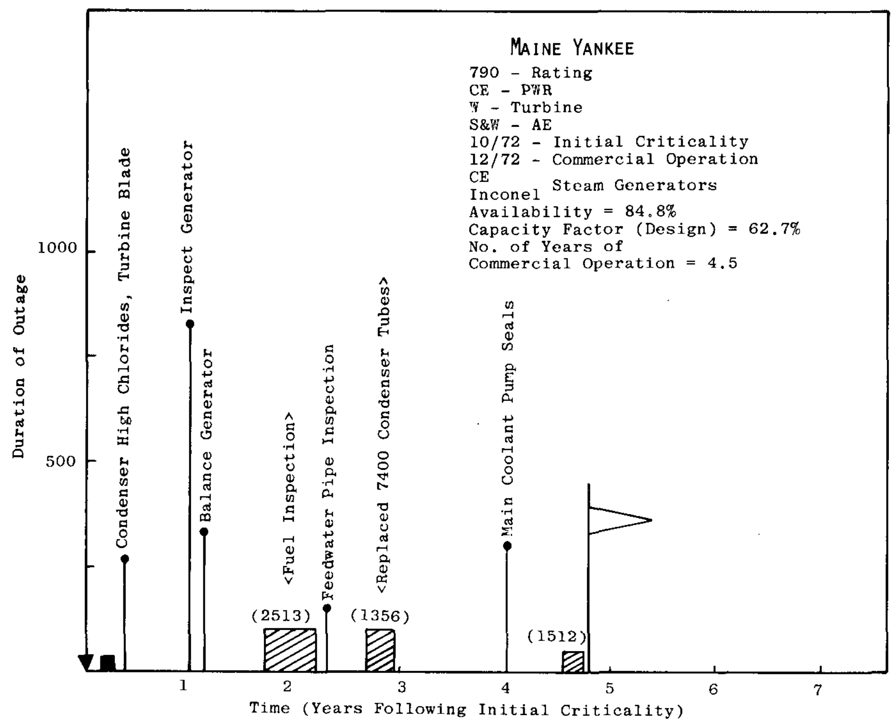
GINNA
490 - Rating
W - PWR
W - Steam Generators
W - Turbine
Gilbert - AE
11/69 - Initial Criticality
3/70 - Commercial Operation
W
Inconel } Steam Generators
Availability = 73%
Capacity Factor (Design) = 62.3%
No. of Years of
Commercial Operation = 6.5 (7.25)
No Data
Feedwater Pump
Feedwater Valve
Electrical Bus
<Turbine Blade Failure>
AVT
Steam Generator Tube Leak
Steam Generator Tube Leak
<100% Eddy Current Inspection of S/G Tubes>
Steam Generator Tube Leak
Turbine Blade Failure
Emergency Systems Pipe Failure
(1728)
(1660)
(792)
(2131)
(1658)
(1666)
(890)
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

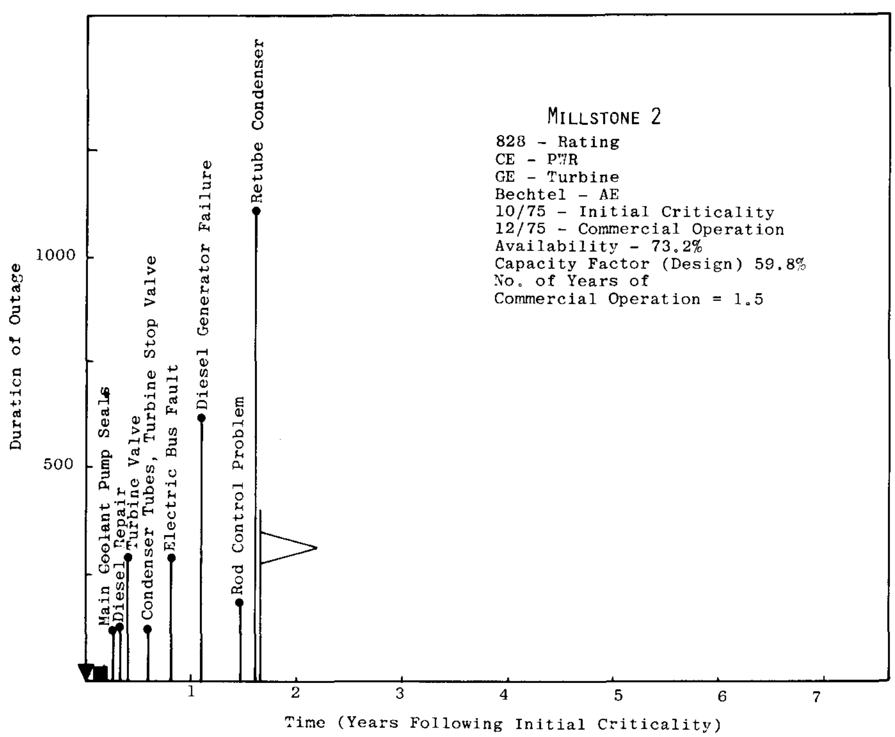

HADDAM NECK
575 Rating
W PWR
W Turbine
S&W AE
7/67 Initial Criticality
1/68 Commercial Operation
W
Inconel } Steam Generators
Availability = 81.7%
Capacity Factor (Design) = 75.4%
No. of Years of
Commercial Operation = 6.5 (9.5)
Duration of Outage
1000
500
Turbine Blade Failure
<Turbine Blade Failure>
Turbine Blade Damage & Plug Steam Generator Tubes
< AVT, Eddy Current Testing of S/G>
No Data
(960)
(826)
(3828)
(1071)
(1452)
3
4
5
6
7
8
9
Time (Years Following Initial Criticality)

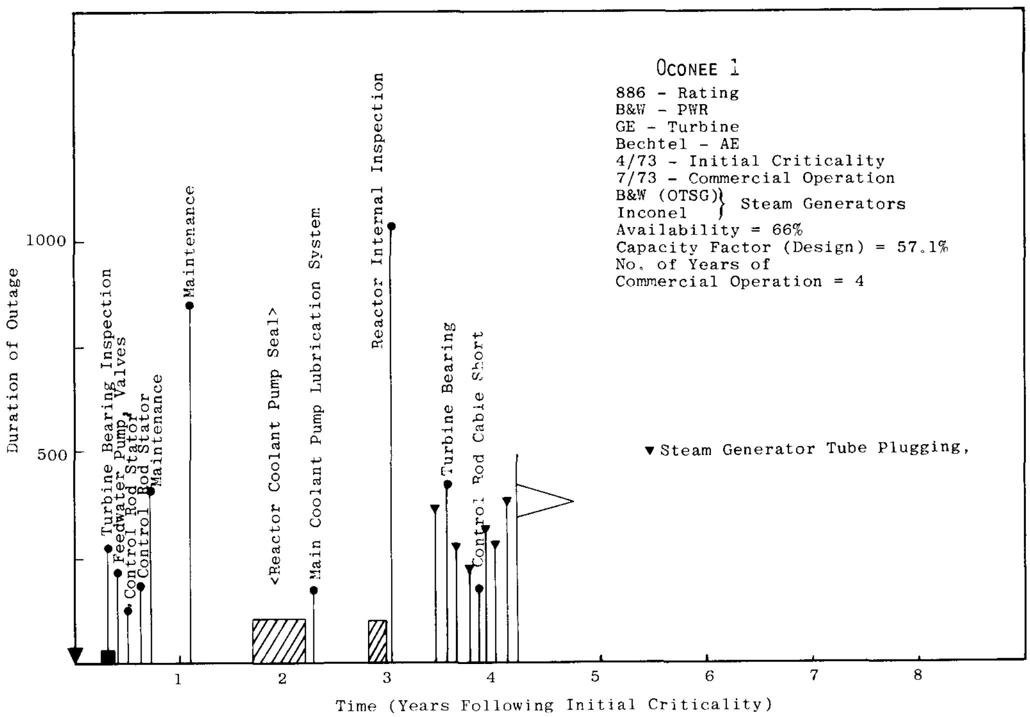

(2532)
INDIAN POINT 2
873 Rating
W PWR
W Turbine
UE&C AE
5/73 Initial Criticality
8/73 Commercial Operation
W
Inconel } Steam Generators
Availability = 62%
Capacity Factor (Design) = 52.8%
No. of Years of
Commercial Operation = 3.5
Duration of Outage
1000
500
Snubber
Feedwater Pipe/Turbine/Generator
Feedwater Ring Modification
Emergency Cooling Test
Pressurizer Rupture Disk
Snubber Inspection, Emergency System Valves
Snubber Inspection
Main Coolant Pump Seals, Snubber Inspection
Steam Generator Tubes
Main Coolant Pump Seals, Snubber Inspection
Generator Fault, Snubber Inspection
In-Core Instrument Drive Repair
<Main Coolant Pump, Feedwater By-Pass Valves>
Steam Generator Tube Inspection
MainCoolant Pump Pipe Seal
Steam Generator Inspection
(4310)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


INDIAN POINT 3
965 - Rating
W - PWR
W - Turbine
A&C - AE
4/76 - Initial Criticality
8/76 - Commercial Operation
Availability = 84.4%
Capacity Factor (Design) = 81.2%
No. of Years of
Commercial Operation = .5
Duration of Outage
1000
500
Maintenance
Primary Coolant Valve Leak
Maintenance
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

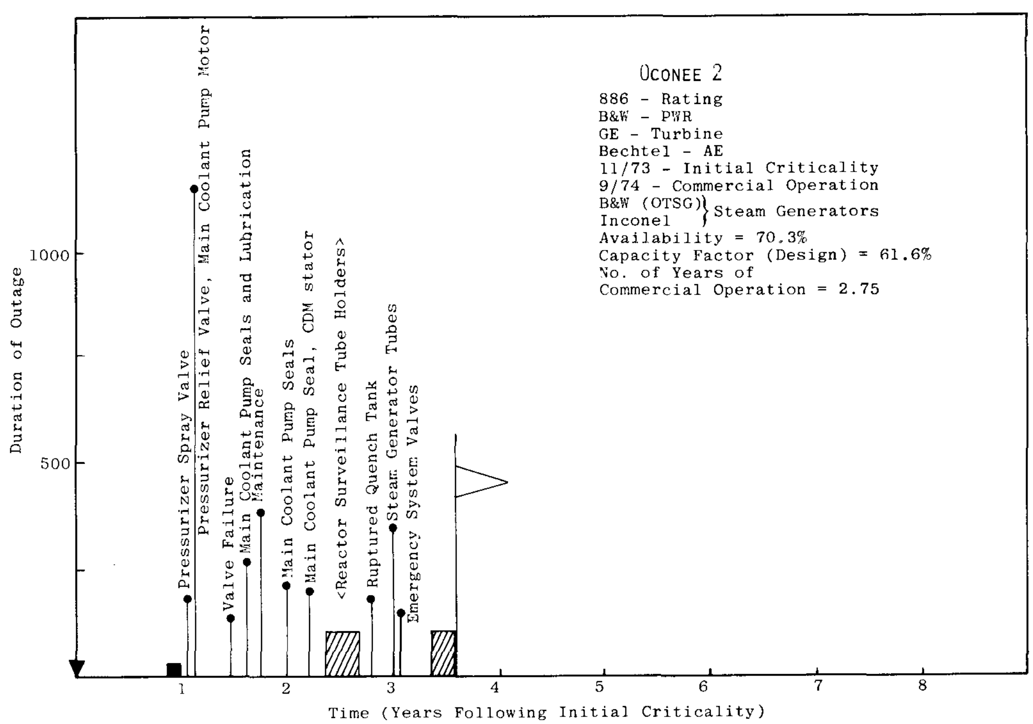

KEWAUNEE
540 - Rating
W - PWR
W - Turbine
FPI - AE
3/74 - Initial Criticality
6/74 - Commercial Operation
W
Inconel
Steam Generators
Availability - 78.4%
Capacity Factor (Design) = 67.6%
No. of Years of
Commercial Operation = 3
Duration of Outage
1000
500
Maintenance
AVT
Steam Generator Inspection
Electrical Bus Fault
Steam Generator Tubes, Pressurizer Leak
Mainsteam Isolation Valve
<Eddy Current Inspect S/G Tubes>
Failure of Vessel Head Seals
<Thrown Blade in Turbine>
(1371)
(1584)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


MAINE YANKEE
790 - Rating
CE - PWR
W - Turbine
S&W - AE
10/72 - Initial Criticality
12/72 - Commercial Operation
CE
Inconel Steam Generators
Availability = 84.8%
Capacity Factor (Design) = 62.7%
No. of Years of
Commercial Operation = 4.5
Duration of Outage
1000
500
Condenser High Chlorides, Turbine Blade
Inspect Generator
Balance Generator
<Fuel Inspection>
Feedwater Pipe Inspection
<Replaced 7400 Condenser Tubes>
Main Coolant Pump Seals
(2513)
(1356)
(1512)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

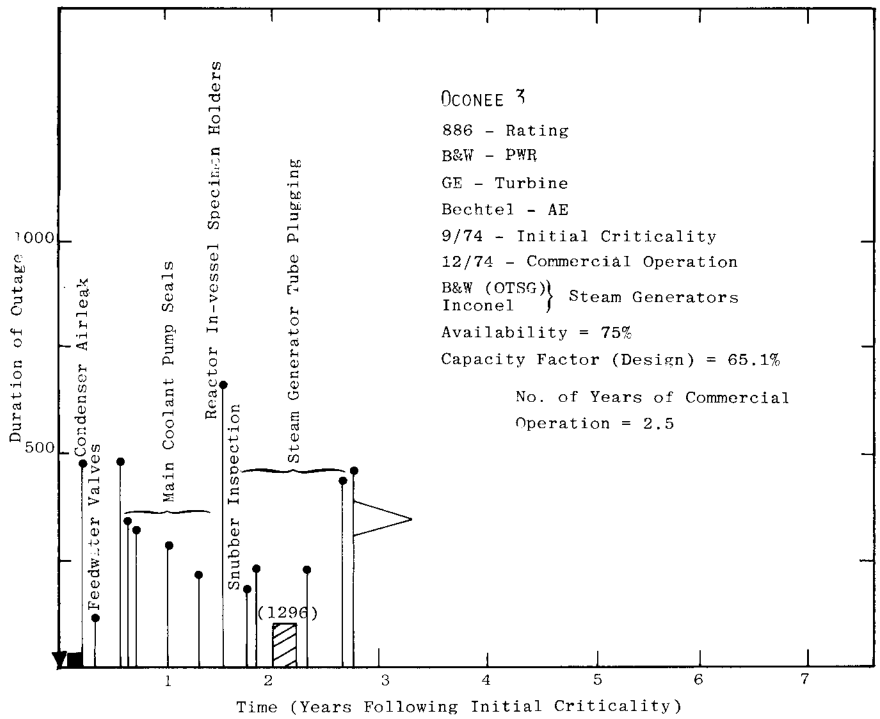

MILLSTONE 2
828 - Rating
CE - PWR
GE - Turbine
Bechtel - AE
10/75 - Initial Criticality
12/75 - Commercial Operation
Availability - 73.2%
Capacity Factor (Design) 59.8%
No. of Years of
Commercial Operation = 1.5
Duration of Outage
1000
500
Main Coolant Pump Seals
Diesel Repair
Turbine Valve
Condenser Tubes, Turbine Stop Valve
Electric Bus Fault
Diesel Generator Failure
Rod Control Problem
Retube Condenser
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)



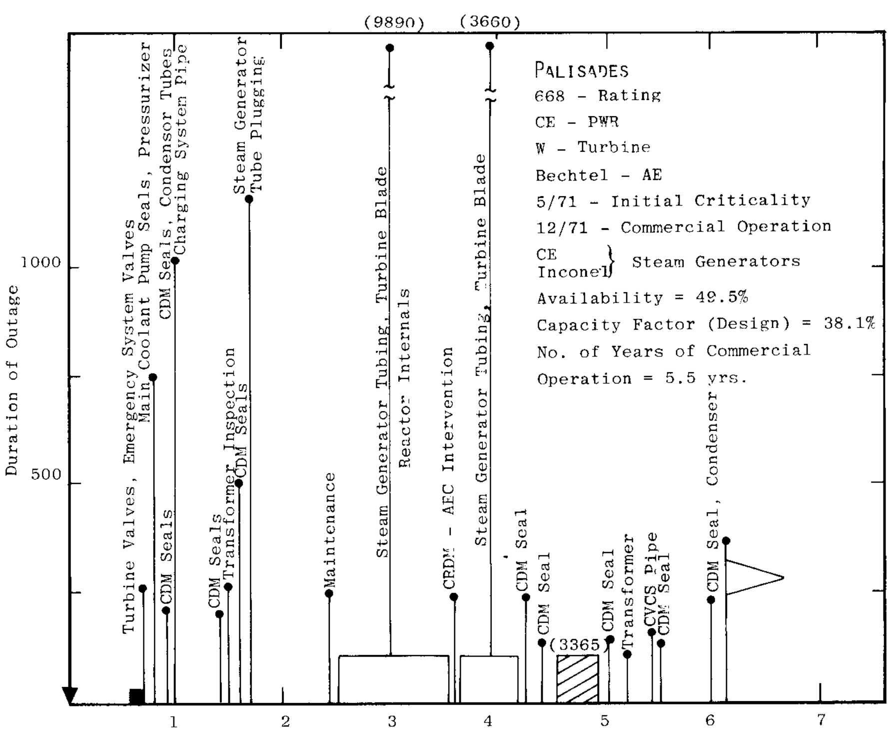
Oconee 1
886 - Rating
B&W - PWR
GE - Turbine
Bechtel - AE
4/73 - Initial Criticality
7/73 - Commercial Operation
B&W (OTSG)
Inconel
Steam Generators
Availability = 66%
Capacity Factor (Design) = 57.1%
No. of Years of
Commercial Operation = 4
Steam Generator Tube Plugging,
Duration of Outage
1000
500
Turbine Bearing Inspection
Feedwater Pump, Valves
Control Rod Stator
Control Rod Stator
Maintenance
Maintenance
<Reactor Coolant Pump Seal>
Main Coolant Pump Lubrication System
Reactor Internal Inspection
Turbine Bearing
Control Rod Cable Short
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)



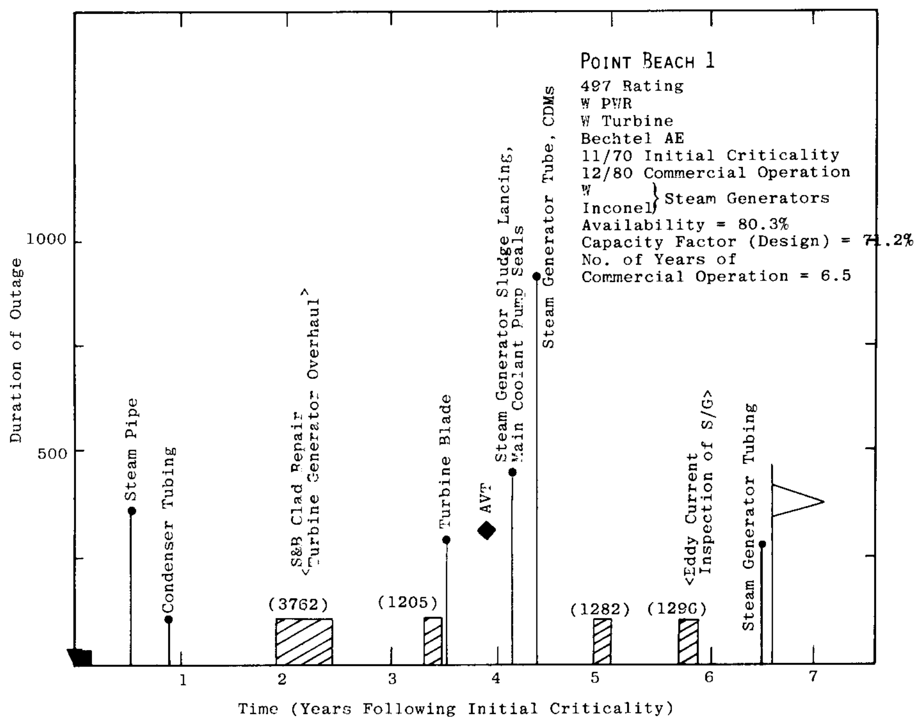
Oconee 2
886 - Rating
B&W - PWR
GE - Turbine
Bechtel - AE
11/73 - Initial Criticality
9/74 - Commercial Operation
B&W (OTSG)
Inconel
Steam Generators
Availability = 70.3%
Capacity Factor (Design) = 61.6%
No. of Years of
Commercial Operation = 2.75
Duration of Outage
1000
500
Pressurizer Spray Valve
Pressurizer Relief Valve, Main Coolant Pump Motor
Valve Failure
Main Coolant Pump Seals and Lubrication
Maintenance
Main Coolant Pump Seals
Main Coolant Pump Seal, CDM stator
<Reactor Surveillance Tube Holders>
Ruptured Quench Tank
Steam Generator Tubes
Emergency System Valves
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

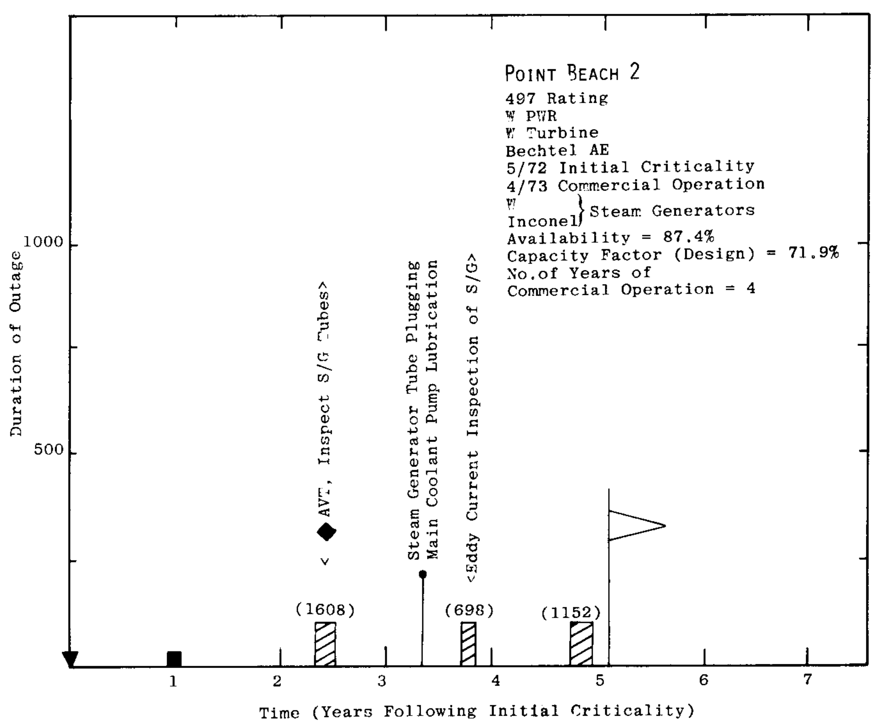


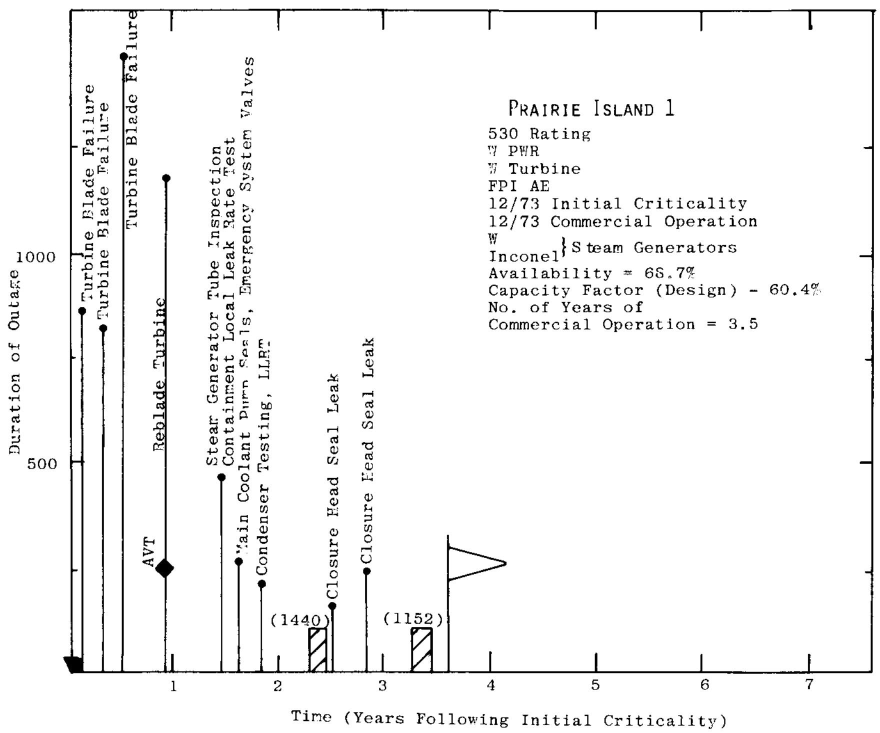
OCONEE 3
886 - Rating
B&W - PWR
GE - Turbine
Bechtel - AE
9/74 - Initial Criticality
12/74 - Commercial Operation
B&W (OTSG) Inconel } Steam Generators
Availability = 75%
Capacity Factor (Design) = 65.1%
No. of Years of Commercial Operation = 2.5
Duration of Outage
1000
500
Condenser Airleak
Feedwater Valves
Main Coolant Pump Seals
Reactor In-vessel Specimen Holders
Snubber Inspection
Steam Generator Tube Plugging
(1296)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

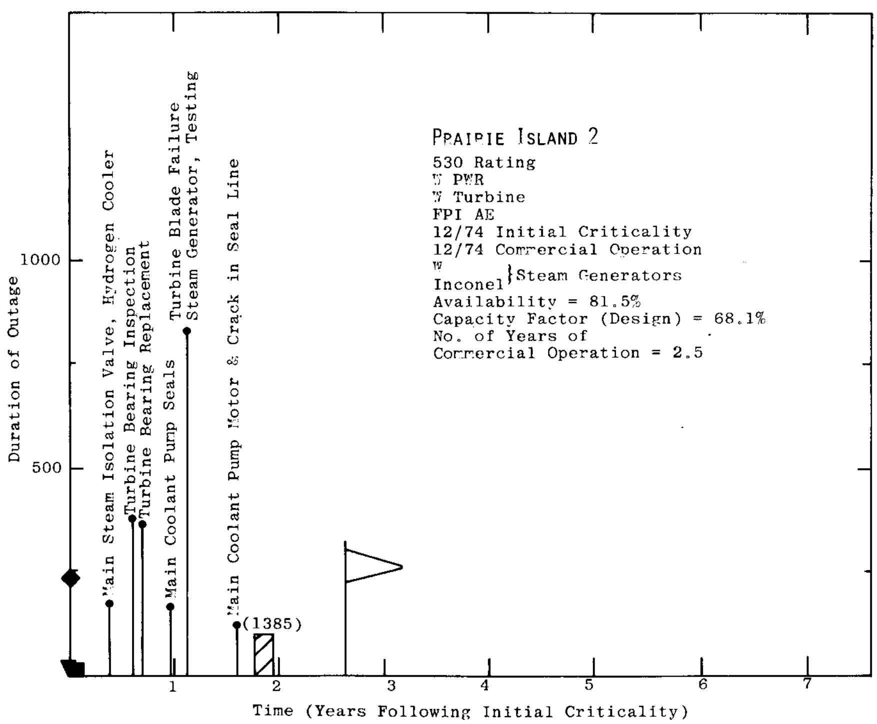


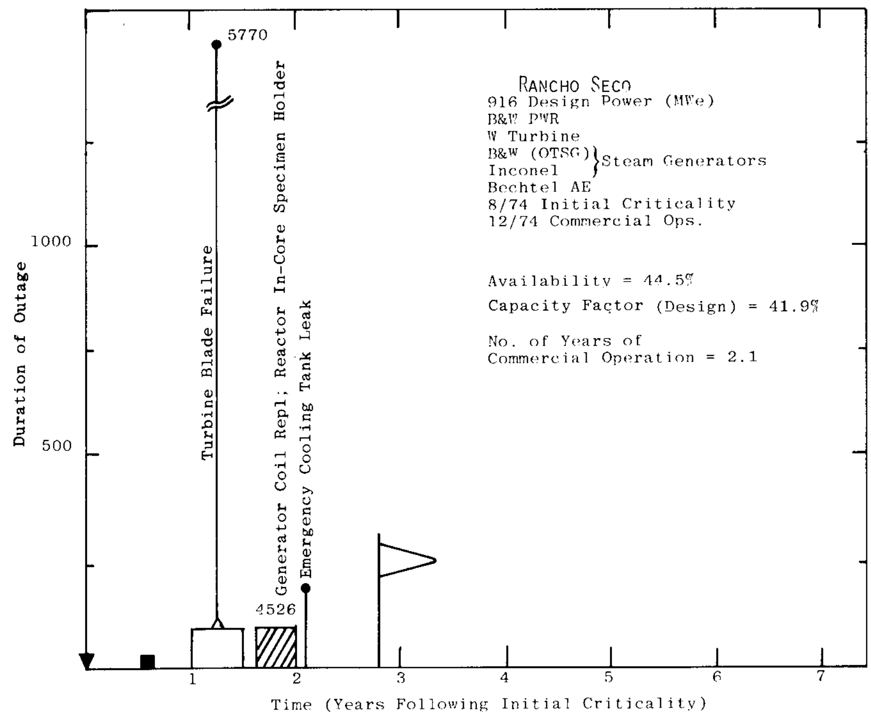
PALISADES
668 - Rating
CE - PWR
W - Turbine
Bechtel - AE
5/71 - Initial Criticality
12/71 - Commercial Operation
CE Inconel } Steam Generators
Availability = 49.5%
Capacity Factor (Design) = 38.1%
No. of Years of Commercial Operation = 5.5 yrs.
(9890)
(3660)
Duration of Outage
1000
500
Turbine Valves, Emergency System Valves
Main Coolant Pump Seals, Pressurizer
CDM Seals, Condensor Tubes
Charging System Pipe
CDM Seals
CDM Seals
Transformer Inspection
CDM Seals
Steam Generator Tube Plugging
Maintenance
Steam Generator Tubing, Turbine Blade
Reactor Internals
CRDM - AEC Intervention
Steam Generator Tubing, Turbine Blade
CDM Seal
CDM Seal
(3365)
CDM Seal
Transformer
CVCS Pipe
CDM Seal
CDM Seal, Condenser
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


POINT BEACH 1
497 Rating
W PWR
W Turbine
Bechtel AE
11/70 Initial Criticality
12/80 Commercial Operation
W
Inconel } Steam Generators
Availability = 80.3%
Capacity Factor (Design) = 71.2%
No. of Years of
Commercial Operation = 6.5
Steam Pipe
Condenser Tubing
S&B Clad Repair
<Turbine Generator Overhaul>
(3762)
(1205)
Turbine Blade
AVT
Steam Generator Sludge Lancing,
Main Coolant Pump Seals
Steam Generator Tube, CDMs
(1282)
(1290)
<Eddy Current
Inspection of S/G>
Steam Generator Tubing
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

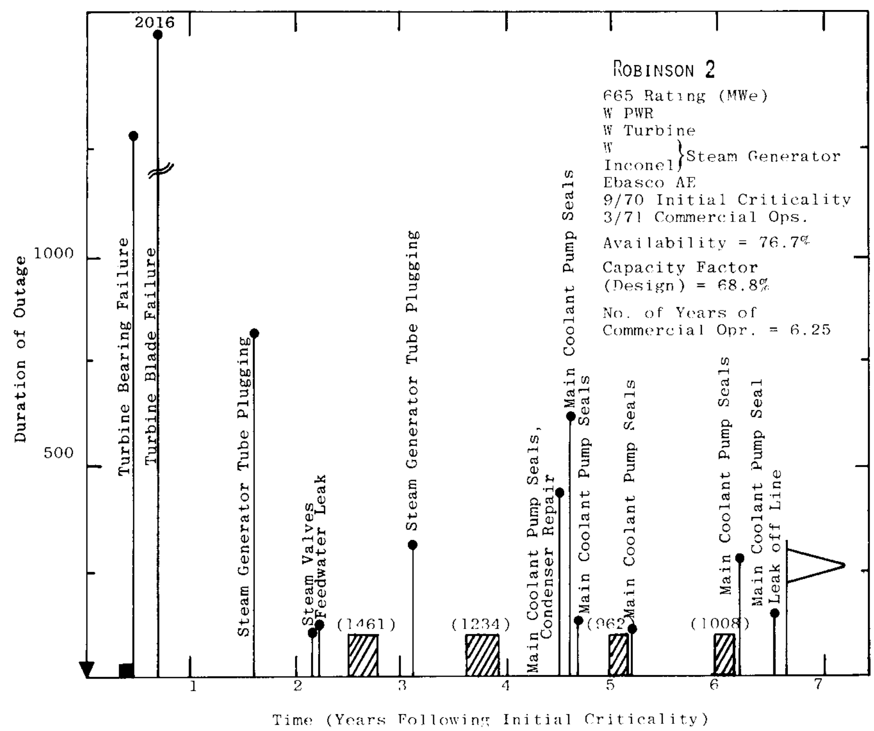

POINT BEACH 2
497 Rating
W PWR
W Turbine
Bechtel AE
5/72 Initial Criticality
4/73 Commercial Operation
W
Inconel } Steam Generators
Availability = 87.4%
Capacity Factor (Design) = 71.9%
No.of Years of
Commercial Operation = 4
< AVT, Inspect S/G Tubes>
(1608)
Steam Generator Tube Plugging
Main Coolant Pump Lubrication
<Eddy Current Inspection of S/G>
(698)
(1152)
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


PRAIRIE ISLAND 1
530 Rating
W PWR
W Turbine
FPI AE
12/73 Initial Criticality
12/73 Commercial Operation
W Inconel } Steam Generators
Availability = 68.7%
Capacity Factor (Design) - 60.4%
No. of Years of
Commercial Operation = 3.5
Turbine Blade Failure
Turbine Blade Failure
Turbine Blade Failure
Reblade Turbine
AVT
Steam Generator Tube Inspection
Containment Local Leak Rate Test
Main Coolant Pump Seals, Emergency System Valves
Condenser Testing, LLRT
(1440)
Closure Head Seal Leak
Closure Head Seal Leak
(1152)
Duration of Outage
500
1000
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

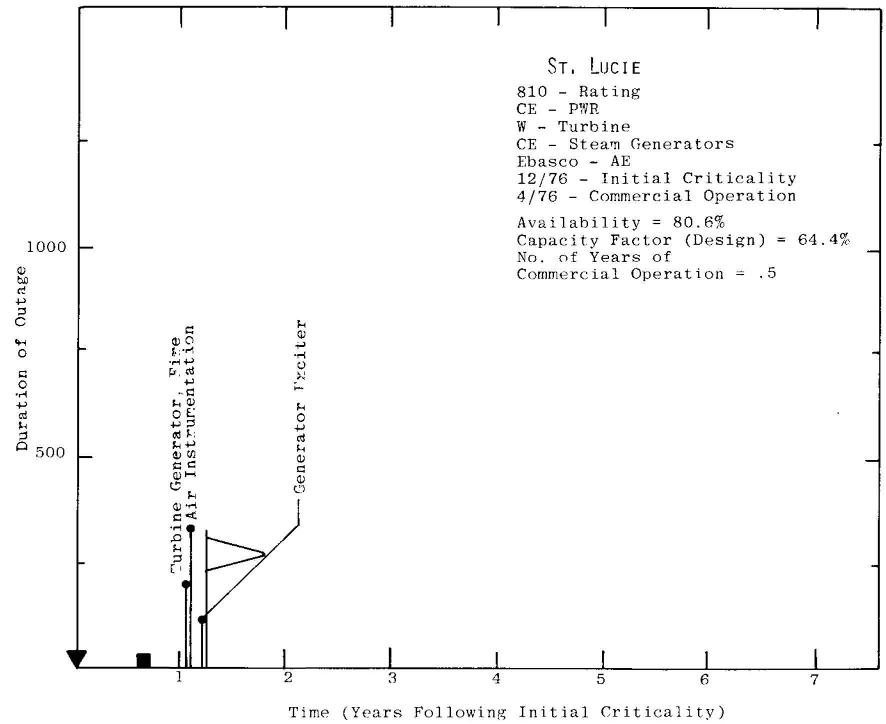

PRAIRIE ISLAND 2
530 Rating
W PWR
W Turbine
FPI AE
12/74 Initial Criticality
12/74 Commercial Operation
W Inconel } Steam Generators
Availability = 81.5%
Capacity Factor (Design) = 68.1%
No. of Years of
Commercial Operation = 2.5
Main Steam Isolation Valve, Hydrogen Cooler
Turbine Bearing Inspection
Turbine Bearing Replacement
Main Coolant Pump Seals
Turbine Blade Failure
Steam Generator, Testing
Main Coolant Pump Motor & Crack in Seal Line
(1385)
Duration of Outage
500
1000
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

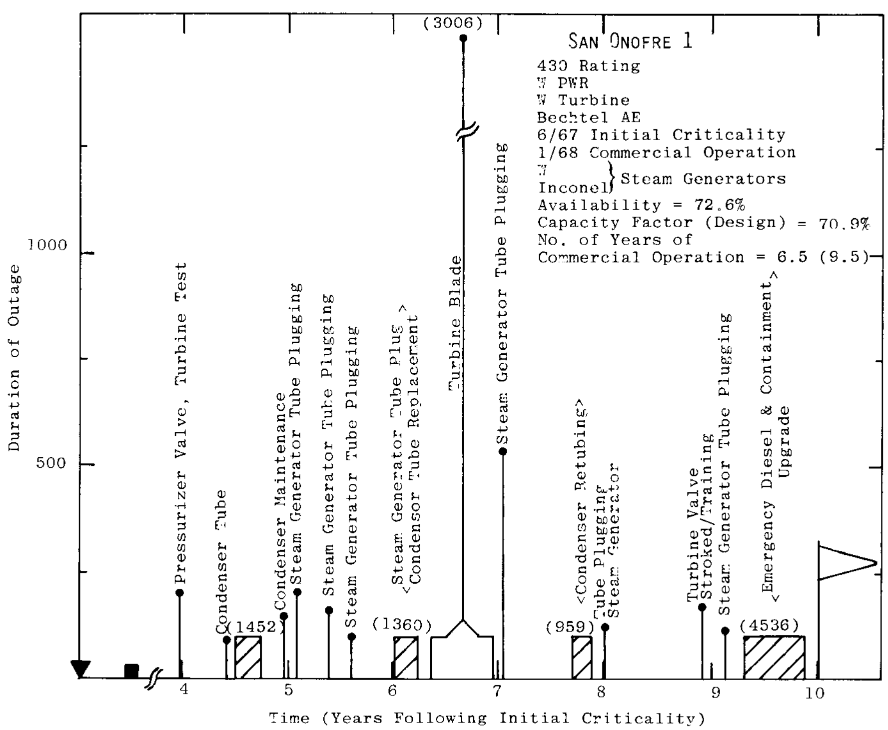


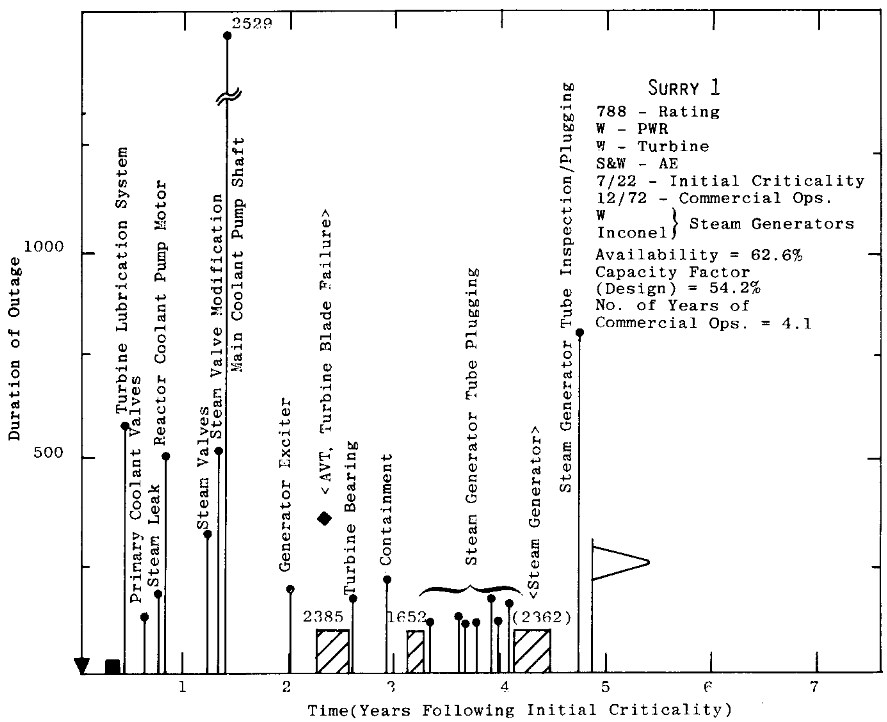
5770
Rancho Seco
916 Design Power (MWe)
B&W PWR
W Turbine
B&W (OTSG)
Inconel
Steam Generators
Bechtel AE
8/74 Initial Criticality
12/74 Commercial Ops.
Availability = 44.5%
Capacity Factor (Design) = 41.9%
No. of Years of
Commercial Operation = 2.1
Turbine Blade Failure
Generator Coil Repl; Reactor In-Core Specimen Holder
Emergency Cooling Tank Leak
4526
1000
500
Duration of Outage
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

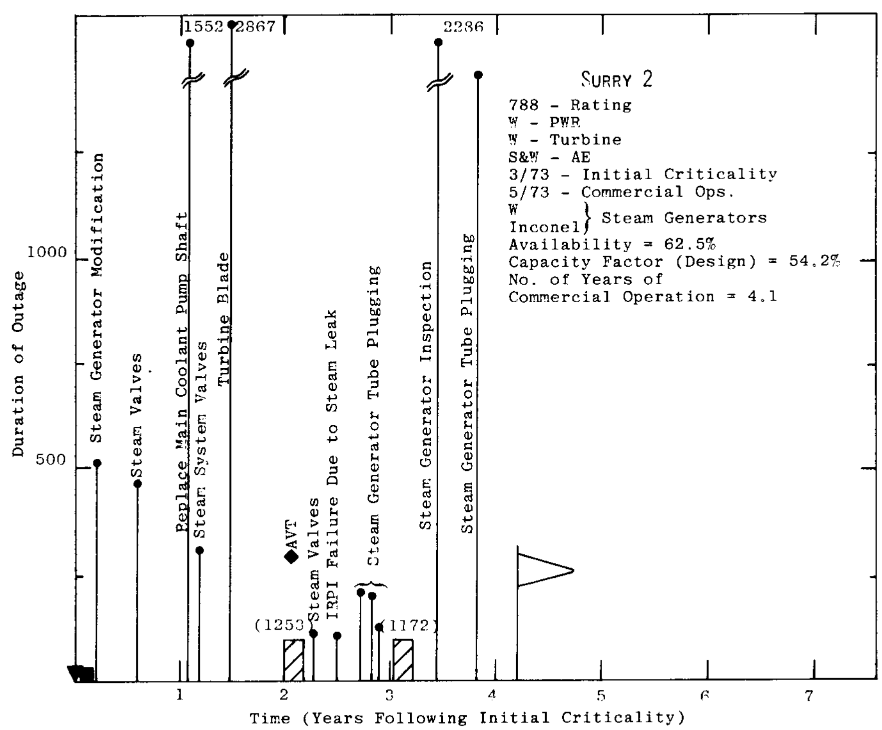


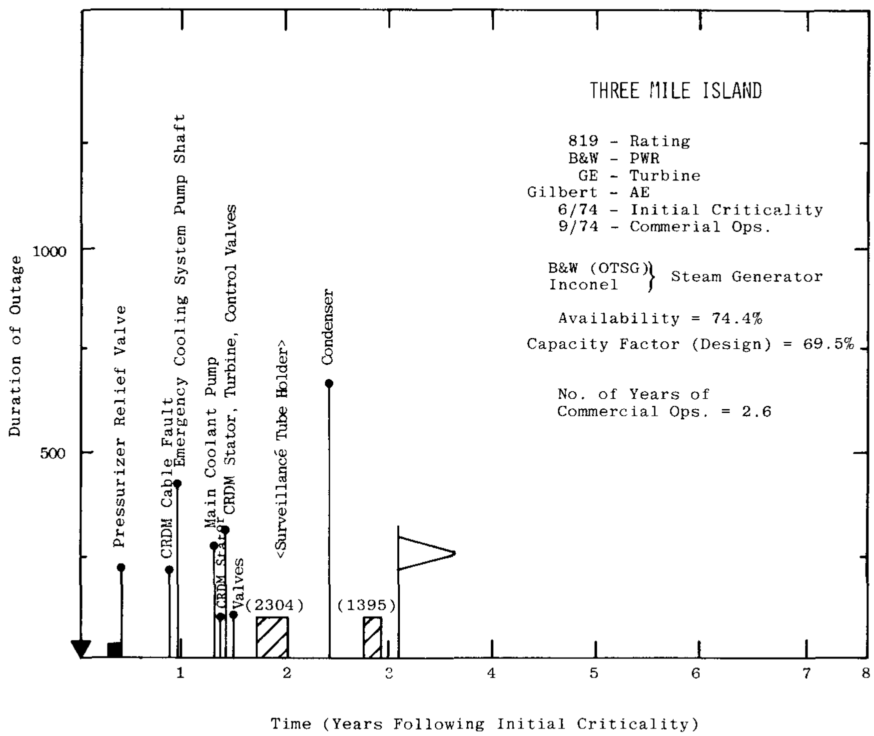
2016
Robinson 2
665 Rating (MWe)
W PWR
W Turbine
W
Inconel
Steam Generator
Ebasco AE
9/70 Initial Criticality
3/71 Commercial Ops.
Availability = 76.7%
Capacity Factor
(Design) = 68.8%
No. of Years of
Commercial Opr. = 6.25
Turbine Bearing Failure
Turbine Blade Failure
Steam Generator Tube Plugging
Steam Valves
Feedwater Leak
(1461)
Steam Generator Tube Plugging
(1234)
Main Coolant Pump Seals,
Condenser Repair
Main Coolant Pump Seals
Main Coolant Pump Seals
(962)
Main Coolant Pump Seals
(1008)
Main Coolant Pump Seals
Main Coolant Pump Seal
Leak off Line
1000
500
Duration of Outage
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

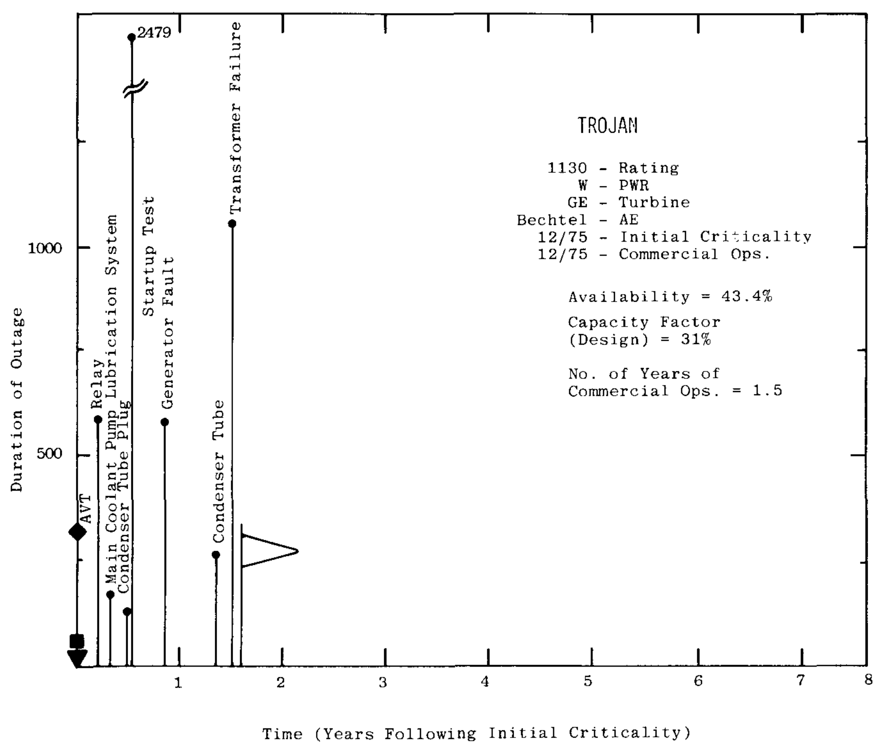

ST. LUCIE
810 - Rating
CE - PWR
W - Turbine
CE - Steam Generators
Ebasco - AE
12/76 - Initial Criticality
4/76 - Commercial Operation
Availability = 80.6%
Capacity Factor (Design) = 64.4%
No. of Years of
Commercial Operation = .5
Duration of Outage
1000
500
Turbine Generator, Fire
Air Instrumentation
Generator Exciter
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

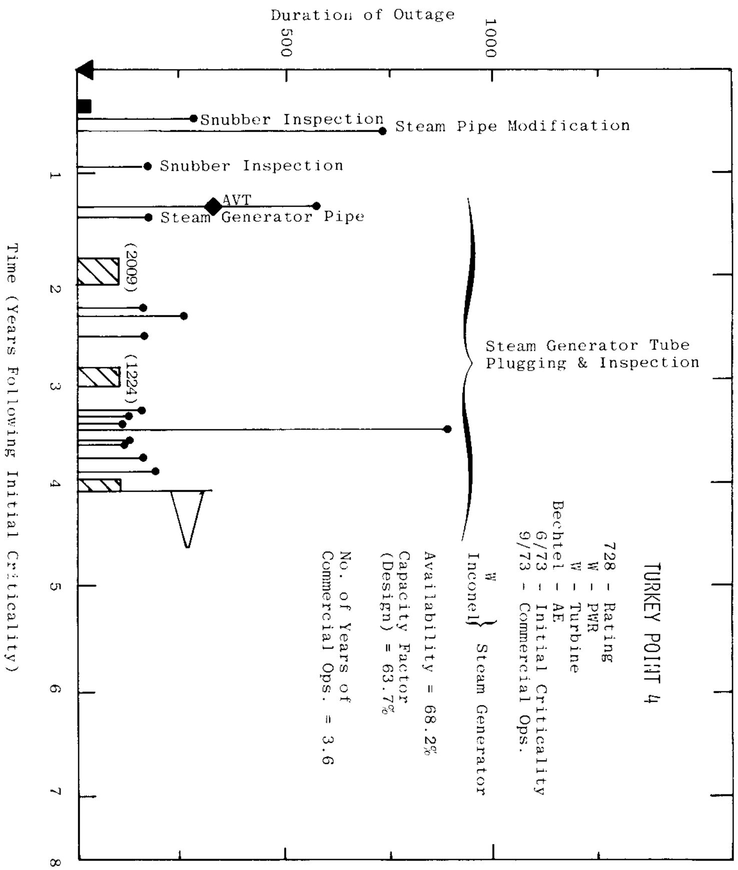

SAN ONOFRE 1
430 Rating
W PWR
W Turbine
Bechtel AE
6/67 Initial Criticality
1/68 Commercial Operation
W
Inconel
Steam Generators
Availability = 72.6%
Capacity Factor (Design) = 70.9%
No. of Years of
Commercial Operation = 6.5 (9.5)
Duration of Outage
1000
500
(3006)
Pressurizer Valve, Turbine Test
Condenser Tube
(1452)
Condenser Maintenance
Steam Generator Tube Plugging
Steam Generator Tube Plugging
Steam Generator Tube Plugging
Steam Generator Tube Plug
Condensor Tube Replacement
(1360)
Turbine Blade
Steam Generator Tube Plugging
Condenser Retubing
(959)
Tube Plugging
Steam Generator
Turbine Valve
Stroked/Training
Steam Generator Tube Plugging
Emergency Diesel & Containment
Upgrade
(4536)
4
5
6
7
8
9
10
Time (Years Following Initial Criticality)

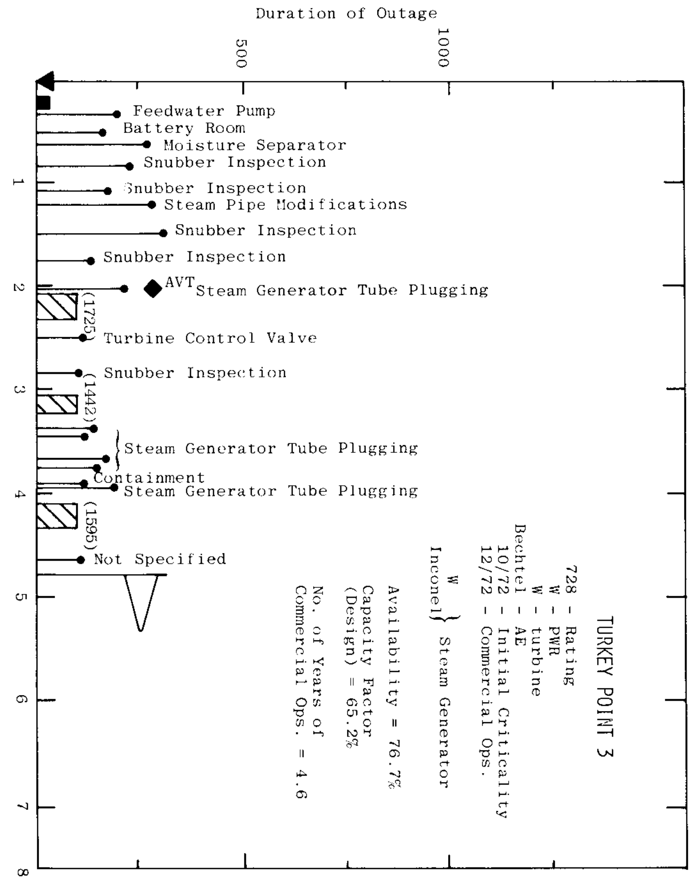

SURRY 1
788 - Rating
W - PWR
W - Turbine
S&W - AE
7/22 - Initial Criticality
12/72 - Commercial Ops.
W Inconel } Steam Generators
Availability = 62.6%
Capacity Factor (Design) = 54.2%
No. of Years of Commercial Ops. = 4.1
Duration of Outage
1000
500
2529
Turbine Lubrication System
Primary Coolant Valves
Steam Leak
Reactor Coolant Pump Motor
Steam Valves
Steam Valve Modification
Main Coolant Pump Shaft
Generator Exciter
<AVT, Turbine Blade Failure>
Turbine Bearing
Containment
Steam Generator Tube Plugging
<Steam Generator>
Steam Generator Tube Inspection/Plugging
2385
1652
(2362)
1
2
3
4
5
6
7
Time(Years Following Initial Criticality)

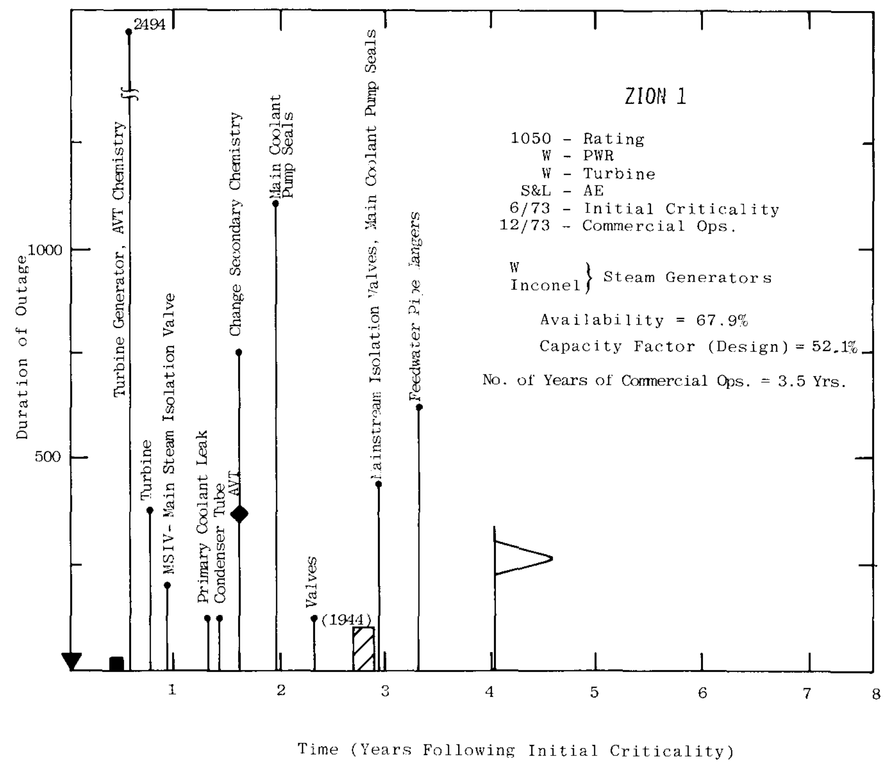

SURRY 2
788 - Rating
W - PWR
W - Turbine
S&W - AE
3/73 - Initial Criticality
5/73 - Commercial Ops.
W Inconel } Steam Generators
Availability = 62.5%
Capacity Factor (Design) = 54.2%
No. of Years of Commercial Operation = 4.1
Duration of Outage
1000
500
1552
2867
2286
Steam Generator Modification
Steam Valves
Replace Main Coolant Pump Shaft
Steam System Valves
Turbine Blade
AVT
Steam Valves
IRPI Failure Due to Steam Leak
Steam Generator Tube Plugging
Steam Generator Inspection
Steam Generator Tube Plugging
(1253)
(1172)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


THREE MILE ISLAND
819 - Rating
B&W - PWR
GE - Turbine
Gilbert - AE
6/74 - Initial Criticality
9/74 - Commerial Ops.
B&W (OTSG)
Inconel
Steam Generator
Availability = 74.4%
Capacity Factor (Design) = 69.5%
No. of Years of
Commercial Ops. = 2.6
Duration of Outage
1000
500
Pressurizer Relief Valve
CRDM Cable Fault
Emergency Cooling System Pump Shaft
Main Coolant Pump
CRDM Stator
CRDM Stator, Turbine, Control Valves
Valves
<Surveillancé Tube Holder>
Condenser
(2304)
(1395)
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

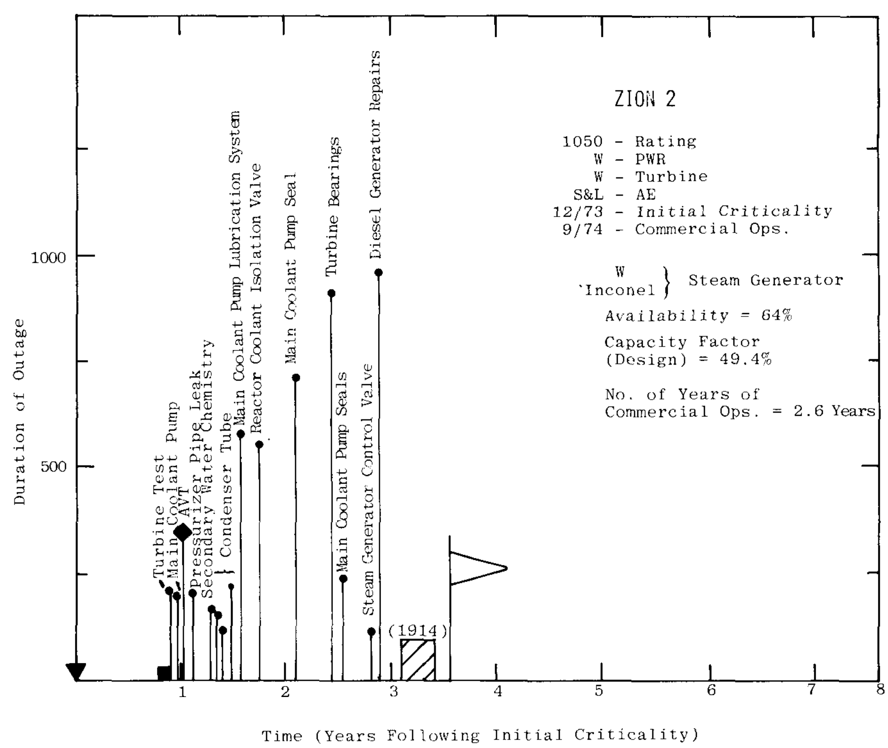

TROJAN
1130 - Rating
W - PWR
GE - Turbine
Bechtel - AE
12/75 - Initial Criticality
12/75 - Commercial Ops.
Availability = 43.4%
Capacity Factor
(Design) = 31%
No. of Years of
Commercial Ops. = 1.5
Duration of Outage
1000
500
2479
AVT
Relay
Main Coolant Pump Lubrication System
Condenser Tube Plug
Startup Test
Generator Fault
Condenser Tube
Transformer Failure
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)


Duration of Outage
500
1000
Snubber Inspection
Steam Pipe Modification
Snubber Inspection
AVT
Steam Generator Pipe
(2009)
(1224)
Steam Generator Tube
Plugging & Inspection
TURKEY POINT 4
728 - Rating
W - PWR
W - Turbine
Bechtel - AE
6/73 - Initial Criticality
9/73 - Commercial Ops.
W
Inconel Steam Generator
Availability = 68.2%
Capacity Factor
(Design) = 63.7%
No. of Years of
Commercial Ops. = 3.6
Time (Years Following Initial Criticality)
1
2
3
4
5
6
7
8

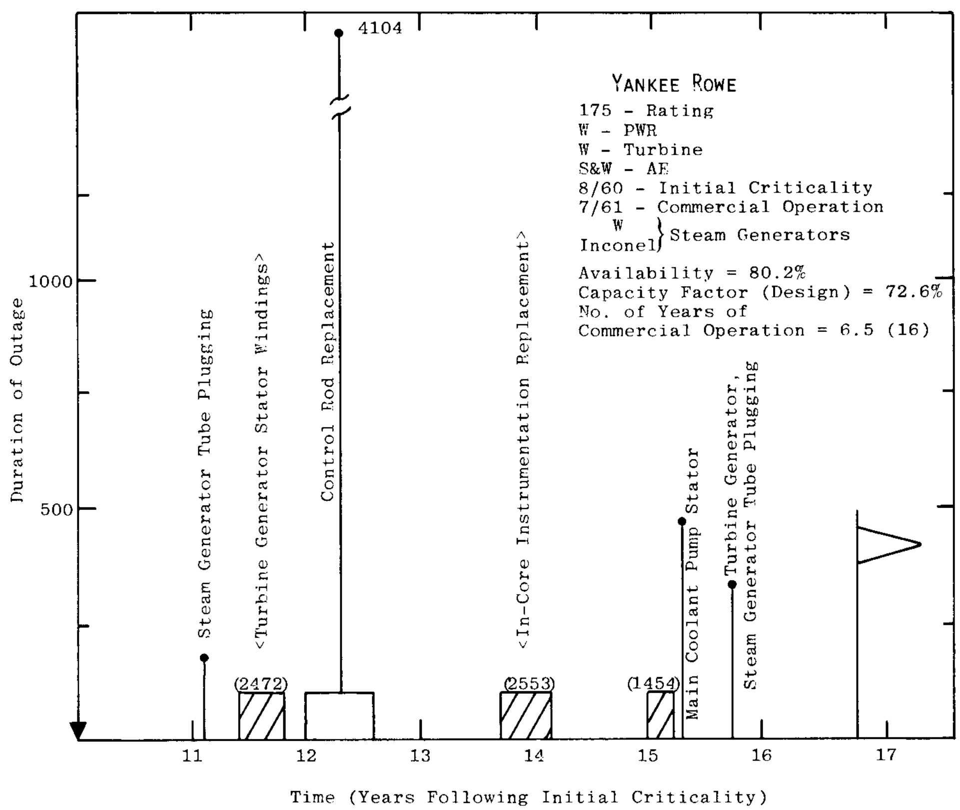

Duration of Outage
500
1000
Feedwater Pump
Battery Room
Moisture Separator
Snubber Inspection
Snubber Inspection
Steam Pipe Modifications
Snubber Inspection
Snubber Inspection
AVT
Steam Generator Tube Plugging
(1725)
Turbine Control Valve
Snubber Inspection
(1442)
Steam Generator Tube Plugging
Containment
Steam Generator Tube Plugging
(1595)
Not Specified
TURKEY POINT 3
728 - Rating
W - PWR
W - turbine
Bechtel - AE
10/72 - Initial Criticality
12/72 - Commercial Ops.
W
Inconel Steam Generator
Availability = 76.7%
Capacity Factor
(Design) = 65.2%
No. of Years of
Commercial Ops. = 4.6
1
2
3
4
5
6
7
8



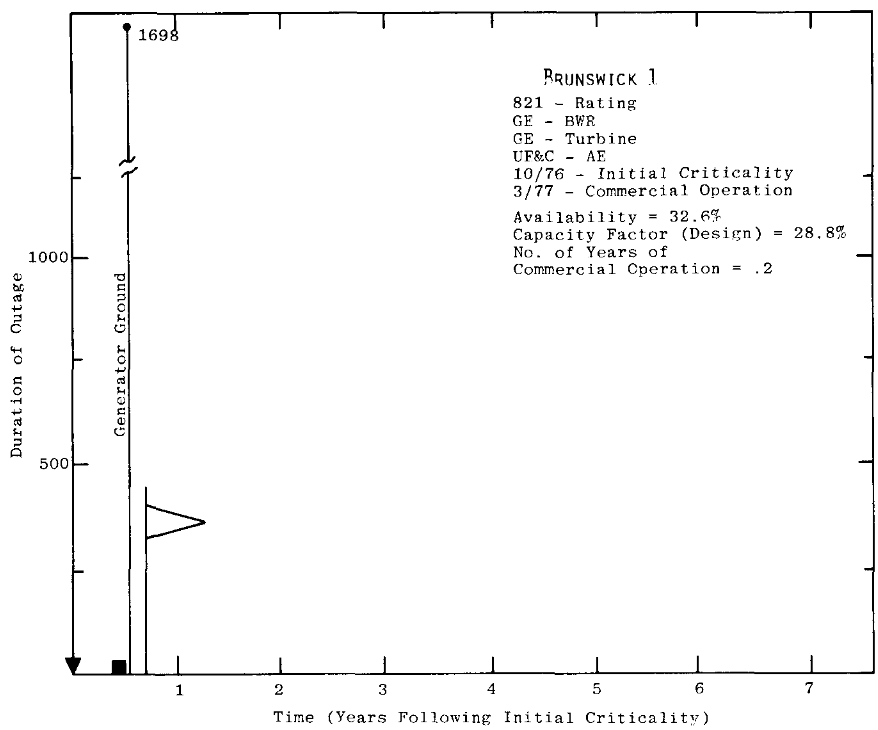
Duration of Outage
500
1000
2494
Turbine Generator, AVT Chemistry
Turbine
MSIV- Main Steam Isolation Valve
Primary Coolant Leak
Condenser Tube
AVT
Change Secondary Chemistry
Main Coolant Pump Seals
Valves
(1944)
Feedwater Pipe Hangers
ZION 1
1050 - Rating
W - PWR
W - Turbine
S&L - AE
6/73 - Initial Criticality
12/73 - Commercial Ops.
W
Inconel } Steam Generators
Availability = 67.9%
Capacity Factor (Design) = 52.1%
No. of Years of Commercial Ops. = 3.5 Yrs.
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

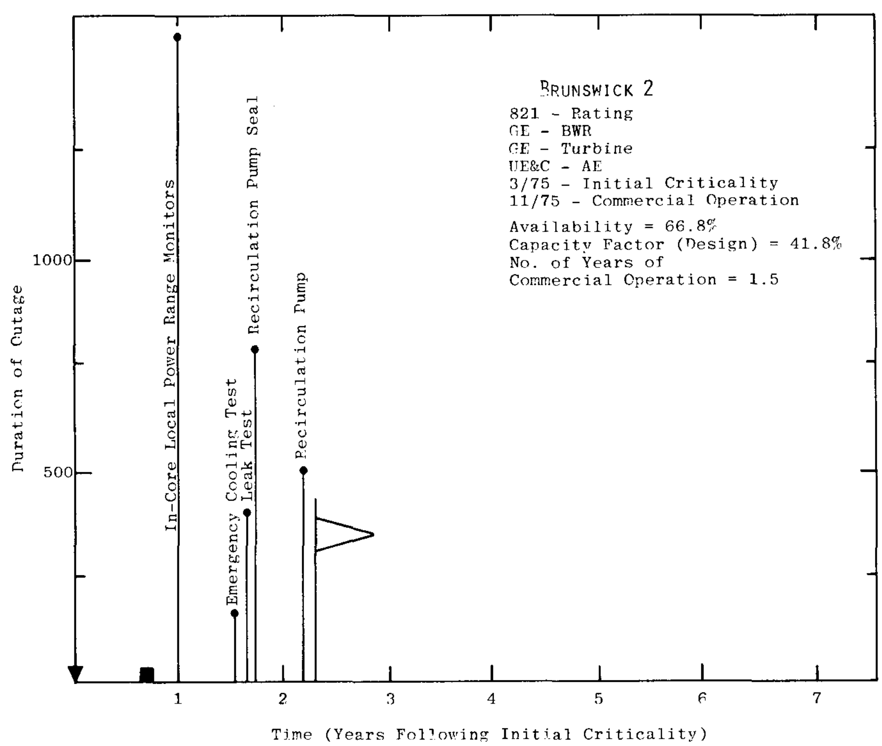


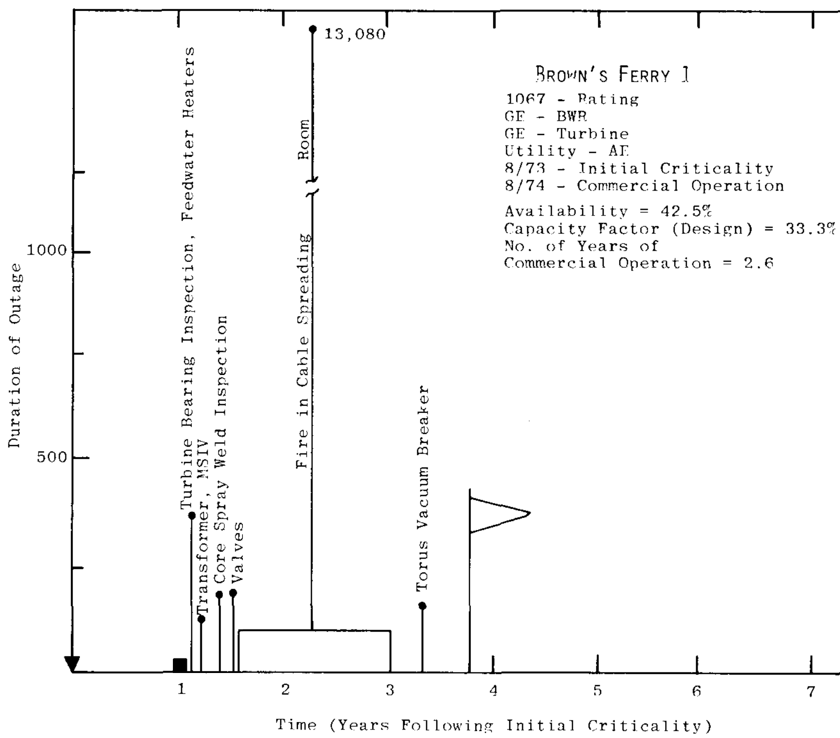
Duration of Outage
500
1000
Turbine Test
Main Coolant Pump
AVT
Pressurizer Pipe Leak
Secondary Water Chemistry
} Condenser Tube
Main Coolant Pump Lubrication System
Reactor Coolant Isolation Valve
Main Coolant Pump Seal
Turbine Bearings
Main Coolant Pump Seals
Steam Generator Control Valve
Diesel Generator Repairs
(1914)
ZION 2
1050 - Rating
W - PWR
W - Turbine
S&L - AE
12/73 - Initial Criticality
9/74 - Commercial Ops.
W
Inconel } Steam Generator
Availability = 64%
Capacity Factor
(Design) = 49.4%
No. of Years of
Commercial Ops. = 2.6 Years
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)



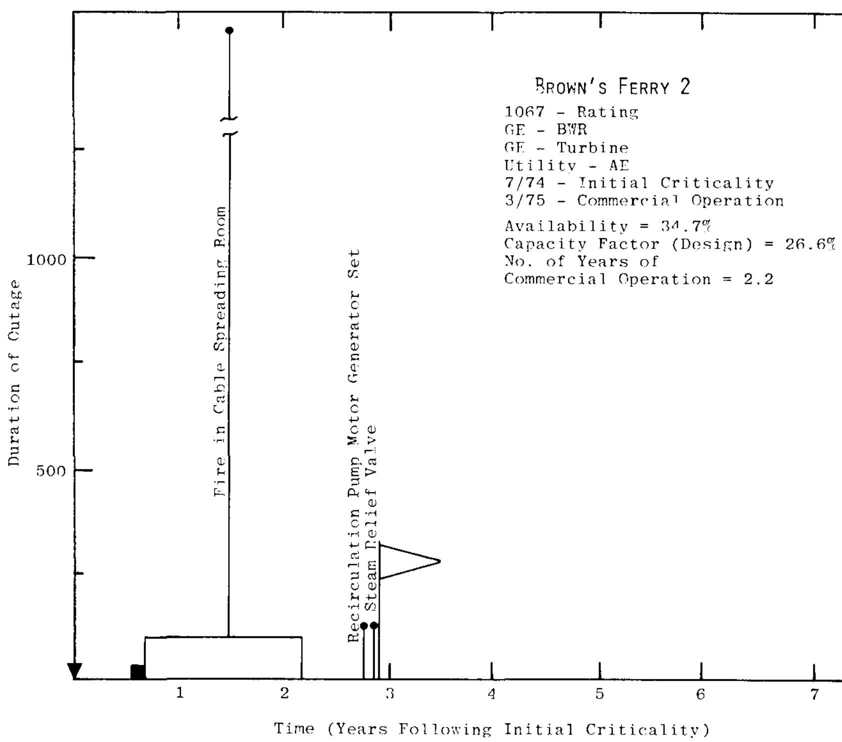
4104
YANKEE ROWE
175 - Rating
W - PWR
W - Turbine
S&W - AE
8/60 - Initial Criticality
7/61 - Commercial Operation
W
Inconel
Steam Generators
Availability = 80.2%
Capacity Factor (Design) = 72.6%
No. of Years of
Commercial Operation = 6.5 (16)
Duration of Outage
1000
500
Steam Generator Tube Plugging
<Turbine Generator Stator Windings>
Control Rod Replacement
<In-Core Instrumentation Replacement>
Main Coolant Pump Stator
Turbine Generator,
Steam Generator Tube Plugging
(2472)
(2553)
(1454)
11
12
13
14
15
16
17
Time (Years Following Initial Criticality)


1698
BRUNSWICK 1
821 - Rating
GE - BWR
GE - Turbine
UE&C - AE
10/76 - Initial Criticality
3/77 - Commercial Operation
Availability = 32.6%
Capacity Factor (Design) = 28.8%
No. of Years of
Commercial Operation = .2
Generator Ground
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

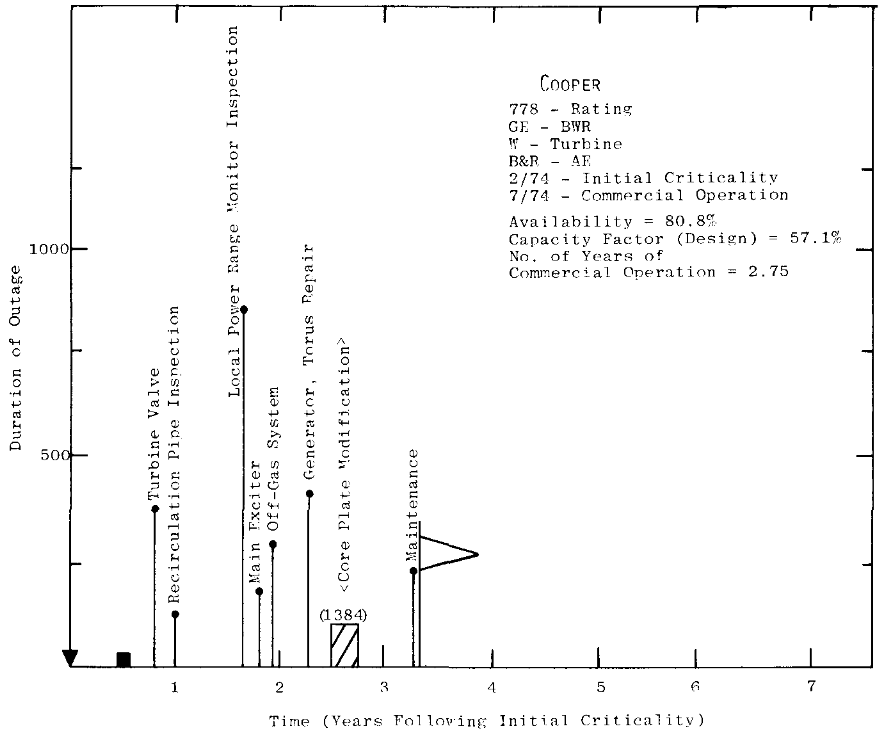

BRUNSWICK 2
821 - Rating
GE - BWR
GE - Turbine
UE&C - AE
3/75 - Initial Criticality
11/75 - Commercial Operation
Availability = 66.8%
Capacity Factor (Design) = 41.8%
No. of Years of
Commercial Operation = 1.5
In-Core Local Power Range Monitors
Emergency Cooling Test
Leak Test
Recirculation Pump Seal
Recirculation Pump
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

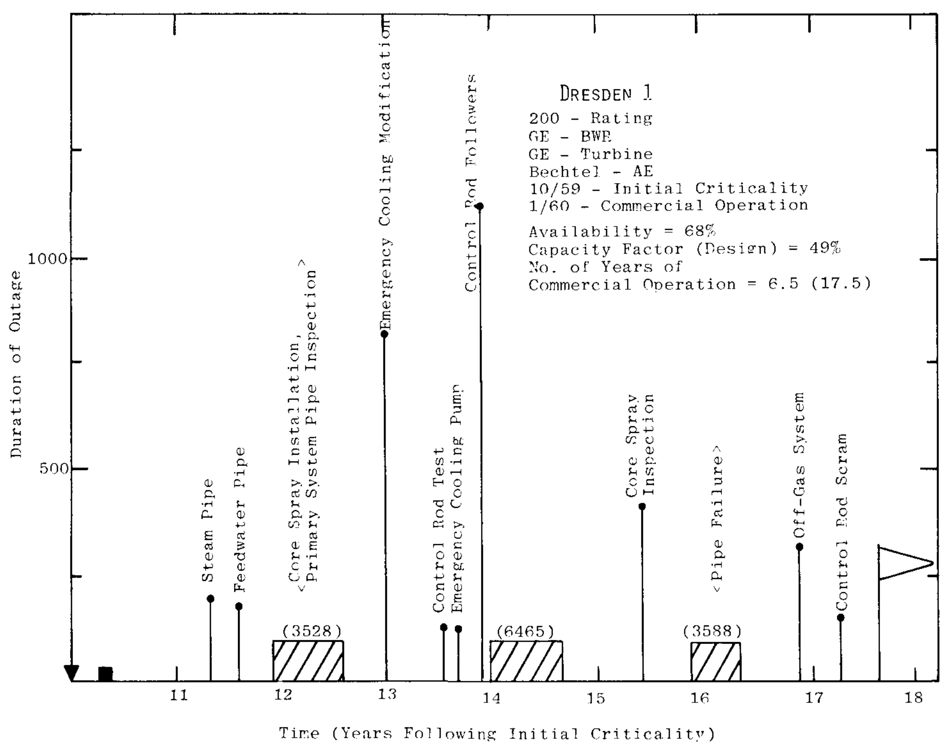


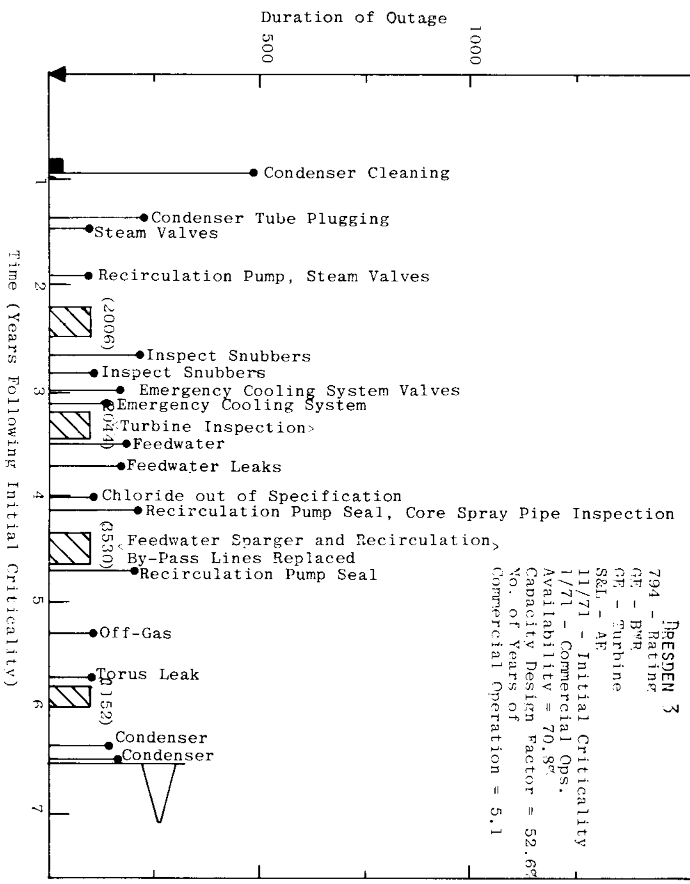
Brown's Ferry 1
1067 - Rating
GE - BWR
GE - Turbine
Utility - AE
8/73 - Initial Criticality
8/74 - Commercial Operation
Availability = 42.5%
Capacity Factor (Design) = 33.3%
No. of Years of
Commercial Operation = 2.6
13,080
Turbine Bearing Inspection, Feedwater Heaters
Transformer, MSIV
Core Spray Weld Inspection
Valves
Fire in Cable Spreading Room
Torus Vacuum Breaker
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)



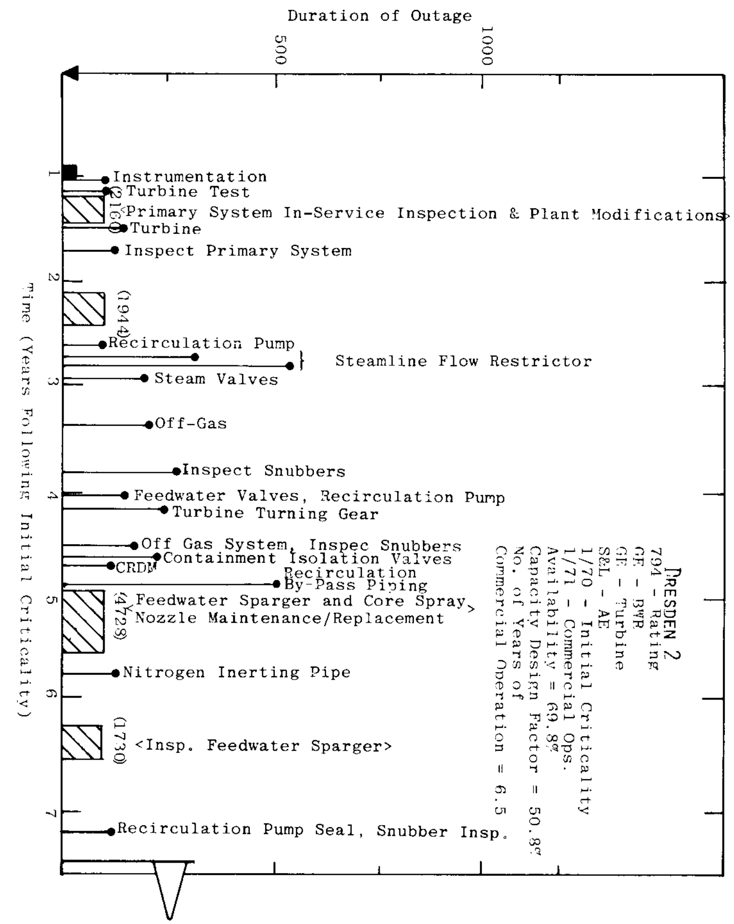
Brown's Ferry 2
1067 - Rating
GE - BWR
GE - Turbine
Utility - AE
7/74 - Initial Criticality
3/75 - Commercial Operation
Availability = 34.7%
Capacity Factor (Design) = 26.6%
No. of Years of
Commercial Operation = 2.2
Fire in Cable Spreading Room
Recirculation Pump Motor Generator Set
Steam Relief Valve
Duration of Outage
1000
500
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)


COOPER
778 - Rating
GE - BWR
W - Turbine
B&R - AE
2/74 - Initial Criticality
7/74 - Commercial Operation
Availability = 80.8%
Capacity Factor (Design) = 57.1%
No. of Years of
Commercial Operation = 2.75
Duration of Outage
Time (Years Following Initial Criticality)
Turbine Valve
Recirculation Pipe Inspection
Local Power Range Monitor Inspection
Main Exciter
Off-Gas System
Generator, Torus Repair
<Core Plate Modification>
(1384)
Maintenance
1000
500
1
2
3
4
5
6
7

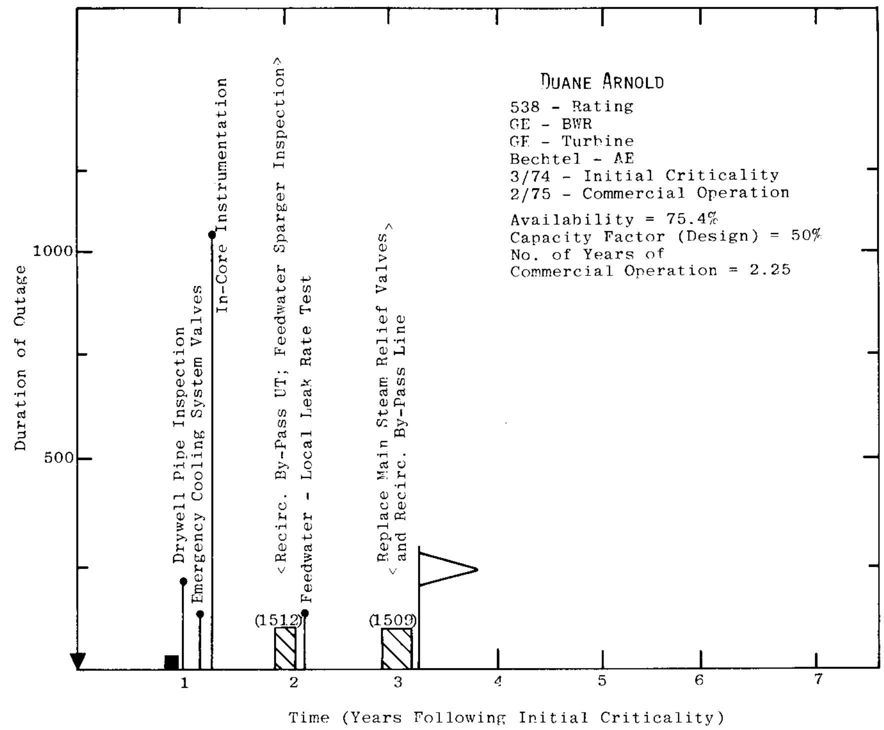

DRESDEN 1
200 - Rating
GE - BWR
GE - Turbine
Bechtel - AE
10/59 - Initial Criticality
1/60 - Commercial Operation
Availability = 68%
Capacity Factor (Design) = 49%
No. of Years of
Commercial Operation = 6.5 (17.5)
Duration of Outage
Time (Years Following Initial Criticality)
Steam Pipe
Feedwater Pipe
<Core Spray Installation,
Primary System Pipe Inspection>
(3528)
Emergency Cooling Modification
Control Rod Test
Emergency Cooling Pump
Control Rod Followers
(6465)
Core Spray
Inspection
<Pipe Failure>
(3588)
Off-Gas System
Control Rod Scram
1000
500
11
12
13
14
15
16
17
18

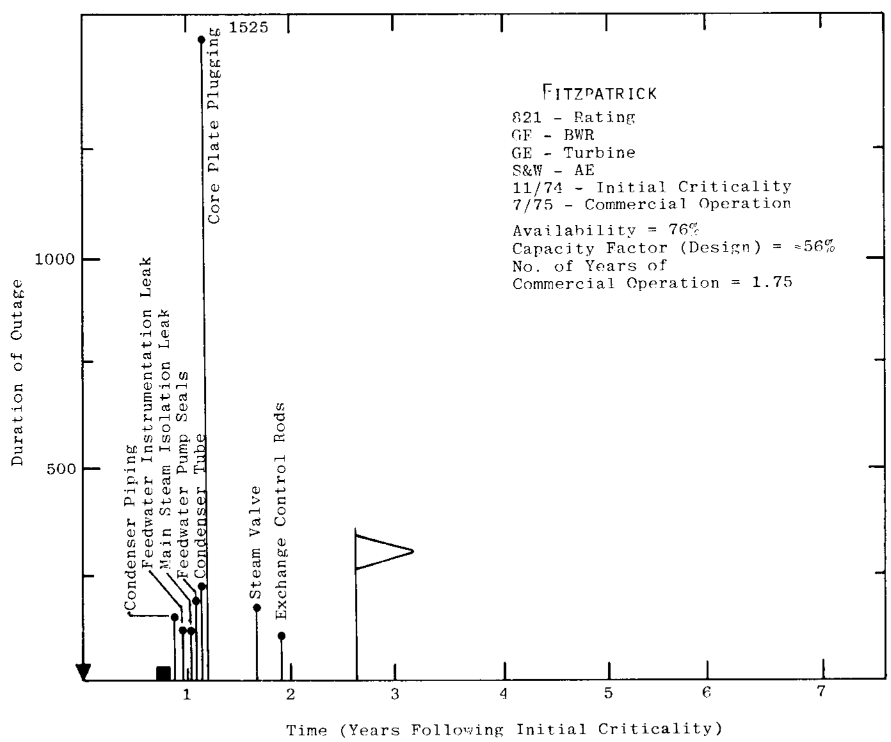


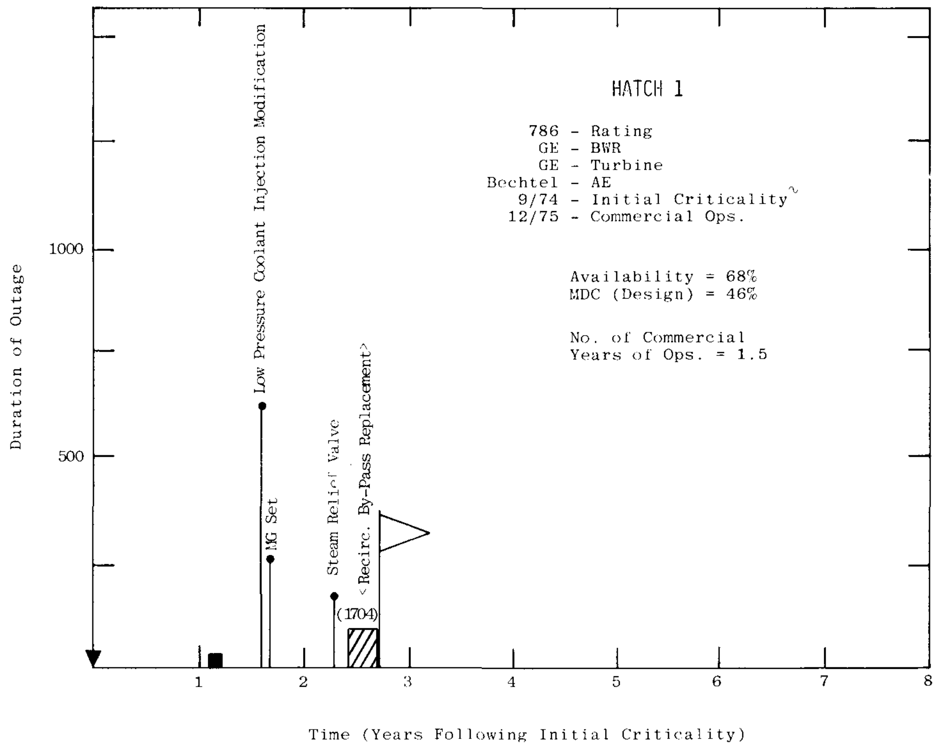
Duration of Outage
500
1000
Condenser Cleaning
Condenser Tube Plugging
Steam Valves
Recirculation Pump, Steam Valves
(2006)
Inspect Snubbers
Inspect Snubbers
Emergency Cooling System Valves
Emergency Cooling System
Turbine Inspection
Feedwater
Feedwater Leaks
Chloride out of Specification
Recirculation Pump Seal, Core Spray Pipe Inspection
(3530)
Feedwater Sparger and Recirculation
By-Pass Lines Replaced
Recirculation Pump Seal
Off-Gas
Torus Leak
(2152)
Condenser
Condenser
Time (Years Following Initial Criticality)
DRESDEN 3
794 - Rating
GE - BWR
GE - Turbine
S&L - AE
11/71 - Initial Criticality
1/71 - Commercial Ops.
Availability = 70.8%
Capacity Design Factor = 52.6%
No. of Years of
Commercial Operation = 5.1

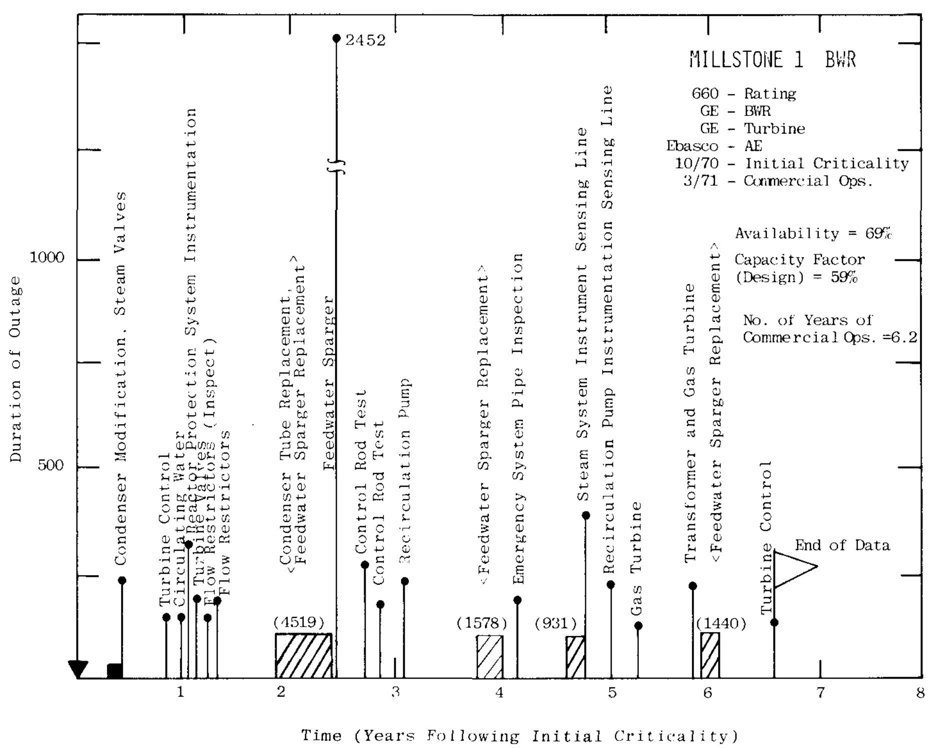


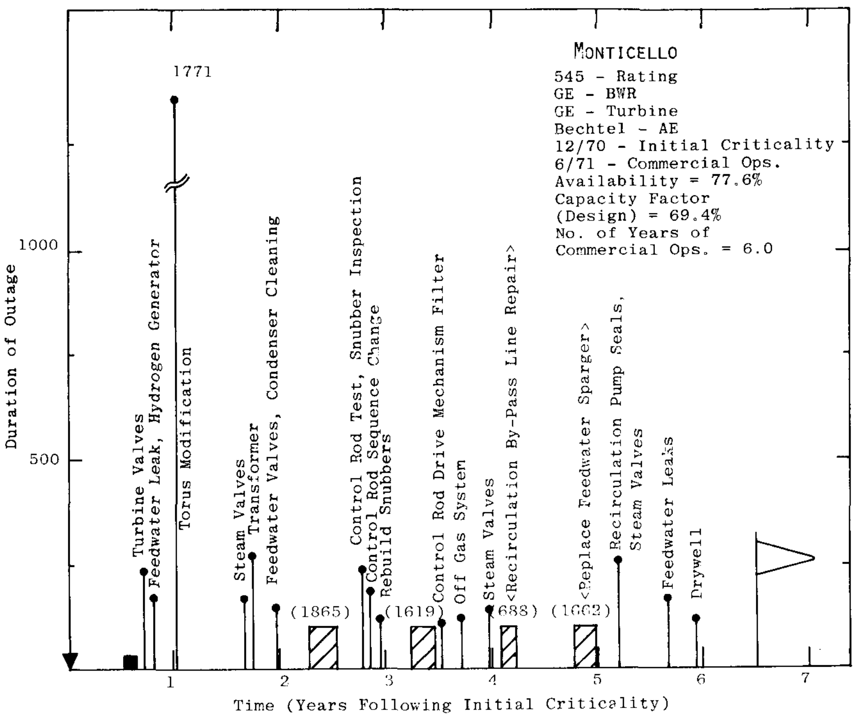
Duration of Outage
500
1000
Instrumentation
Turbine Test
Primary System In-Service Inspection & Plant Modifications
Turbine
Inspect Primary System
(1944)
Recirculation Pump
Steamline Flow Restrictor
Steam Valves
Off-Gas
Inspect Snubbers
Feedwater Valves, Recirculation Pump
Turbine Turning Gear
Off Gas System, Inspec Snubbers
Containment Isolation Valves
CRDM
Recirculation
By-Pass Piping
(4728)
Feedwater Sparger and Core Spray
Nozzle Maintenance/Replacement
Nitrogen Inerting Pipe
(1730)
Insp. Feedwater Sparger
Recirculation Pump Seal, Snubber Insp.
Time (Years Following Initial Criticality)
DRESDEN 2
794 - Rating
GE - BWR
GE - Turbine
S&L - AE
1/70 - Initial Criticality
1/71 - Commercial Ops.
Availability = 69.8%
Capacity Design Factor = 50.8%
No. of Years of
Commercial Operation = 6.5

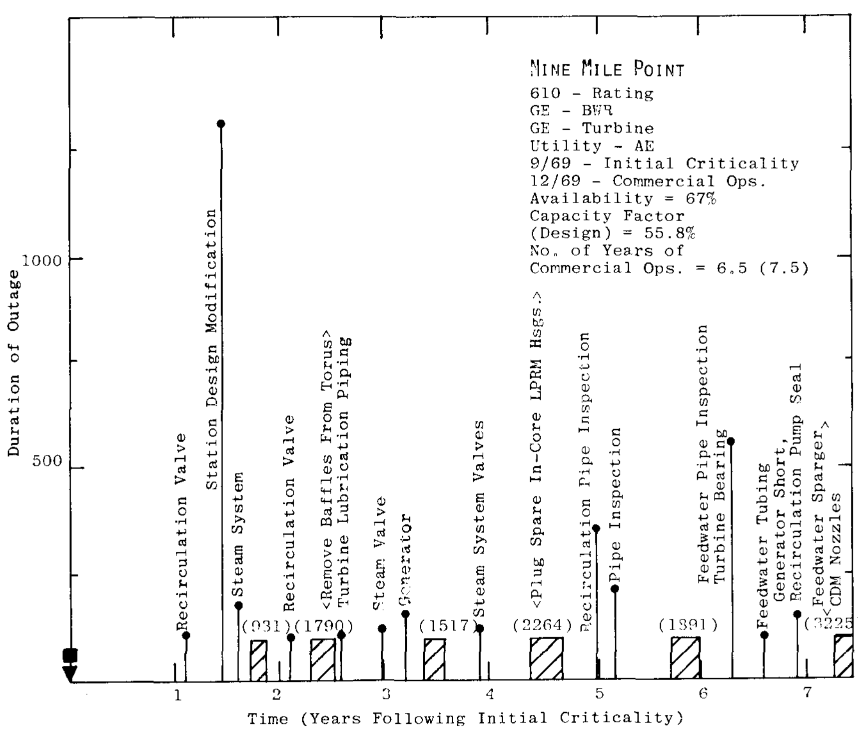

DUANE ARNOLD
538 - Rating
GE - BWR
GE - Turbine
Bechtel - AE
3/74 - Initial Criticality
2/75 - Commercial Operation
Availability = 75.4%
Capacity Factor (Design) = 50%
No. of Years of
Commercial Operation = 2.25
Duration of Outage
1000
500
Drywell Pipe Inspection
Emergency Cooling System Valves
In-Core Instrumentation
<Recirc. By-Pass UT; Feedwater Sparger Inspection>
Feedwater - Local Leak Rate Test
Replace Main Steam Relief Valves
and Recirc. By-Pass Line
(1512)
(1509)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

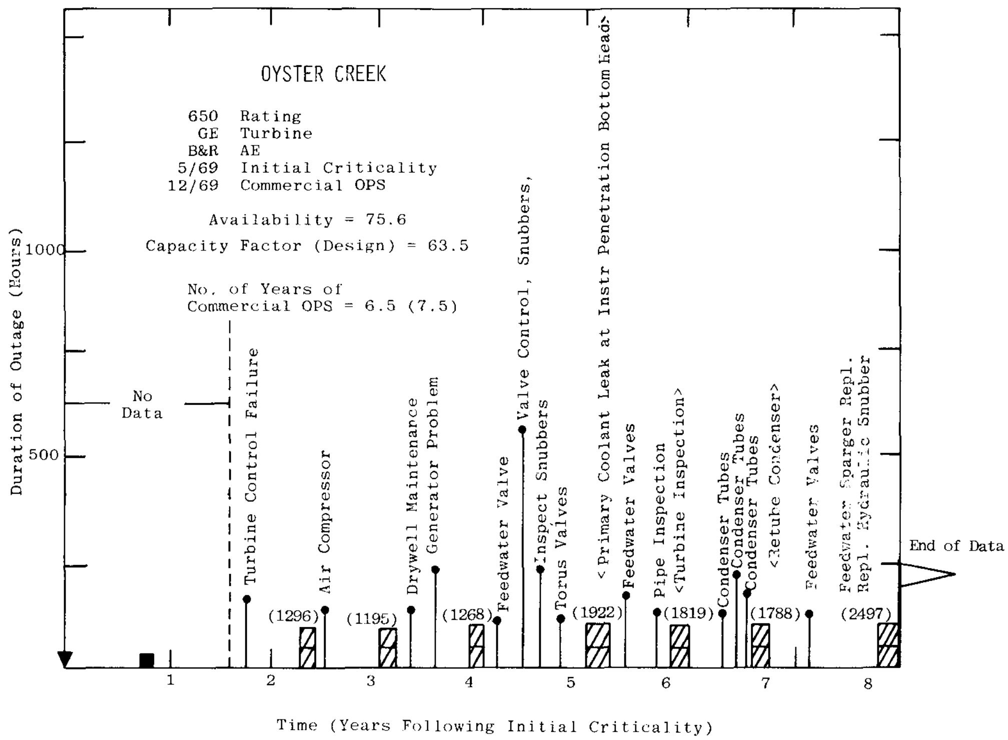

FITZPATRICK
821 - Rating
GE - BWR
GE - Turbine
S&W - AE
11/74 - Initial Criticality
7/75 - Commercial Operation
Availability = 76%
Capacity Factor (Design) = ≈56%
No. of Years of
Commercial Operation = 1.75
Duration of Outage
1000
500
1525
Core Plate Plugging
Condenser Piping
Feedwater Instrumentation Leak
Main Steam Isolation Leak
Feedwater Pump Seals
Condenser Tube
Steam Valve
Exchange Control Rods
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

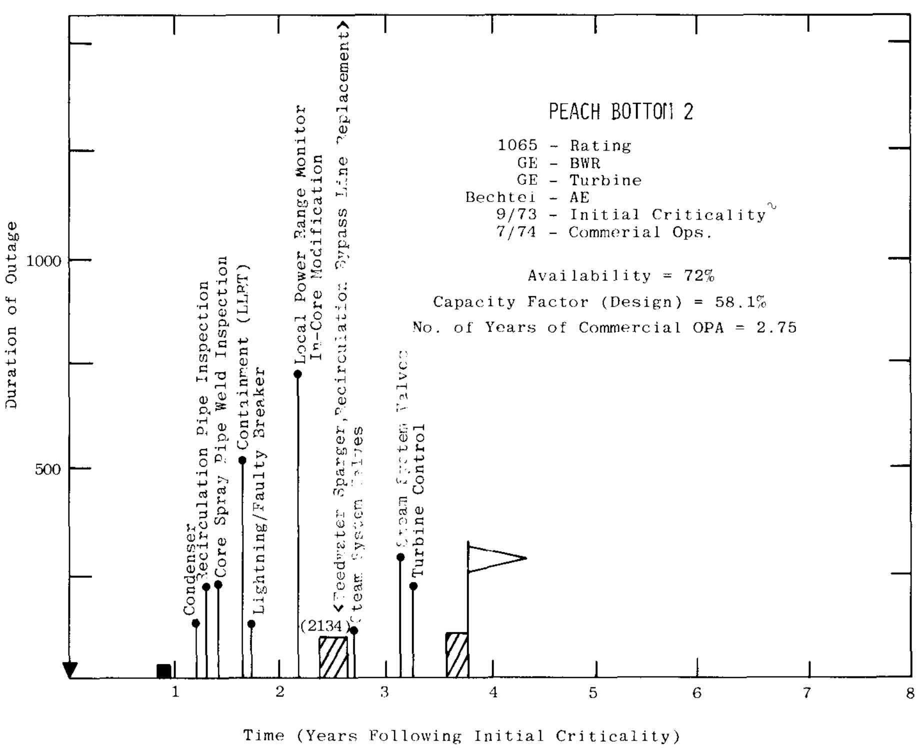

HATCH 1
786 - Rating
GE - BWR
GE - Turbine
Bechtel - AE
9/74 - Initial Criticality
12/75 - Commercial Ops.
Availability = 68%
MDC (Design) = 46%
No. of Commercial Years of Ops. = 1.5
Low Pressure Coolant Injection Modification
MG Set
Steam Relief Valve
<Recirc. By-Pass Replacement>
(1704)
Duration of Outage
1000
500
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

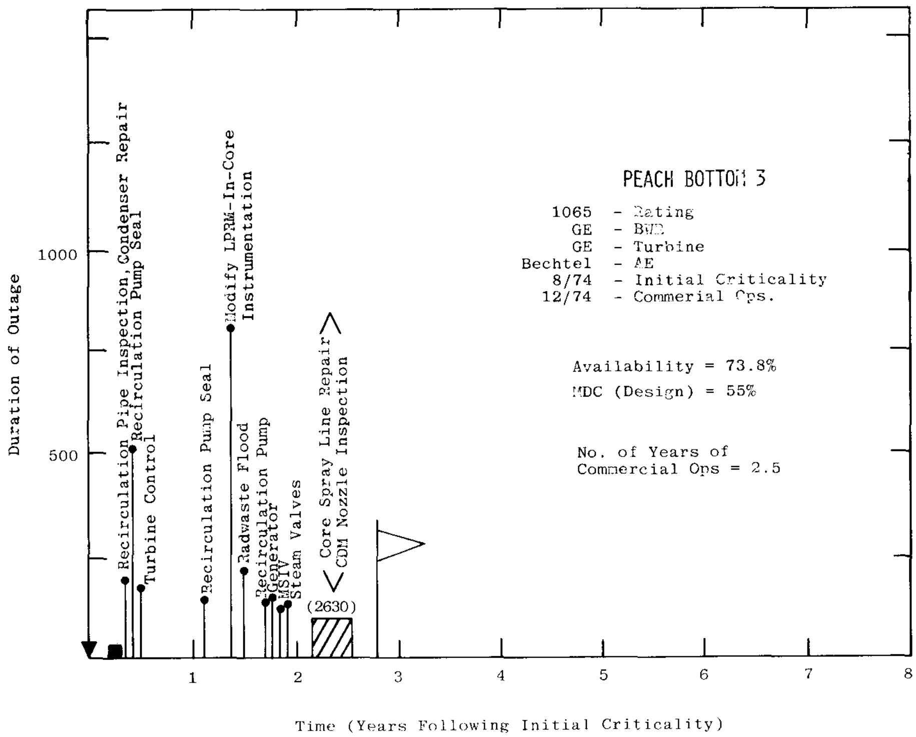

MILLSTONE 1 BWR
660 - Rating
GE - BWR
GE - Turbine
Ebasco - AE
10/70 - Initial Criticality
3/71 - Commercial Ops.
Availability = 69%
Capacity Factor (Design) = 59%
No. of Years of Commercial Ops. =6.2
2452
Condenser Modification, Steam Valves
Turbine Control
Circulating Water
Reactor Protection System Instrumentation
Turbine Valves
Flow Restrictors (Inspect)
Flow Restrictors
Condenser Tube Replacement,
<Feedwater Sparger Replacement>
Feedwater Sparger
Control Rod Test
Control Rod Test
Recirculation Pump
<Feedwater Sparger Replacement>
Emergency System Pipe Inspection
Steam System Instrument Sensing Line
Recirculation Pump Instrumentation Sensing Line
Gas Turbine
Transformer and Gas Turbine
<Feedwater Sparger Replacement>
Turbine Control
End of Data
(4519)
(1578)
(931)
(1440)
Duration of Outage
1000
500
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

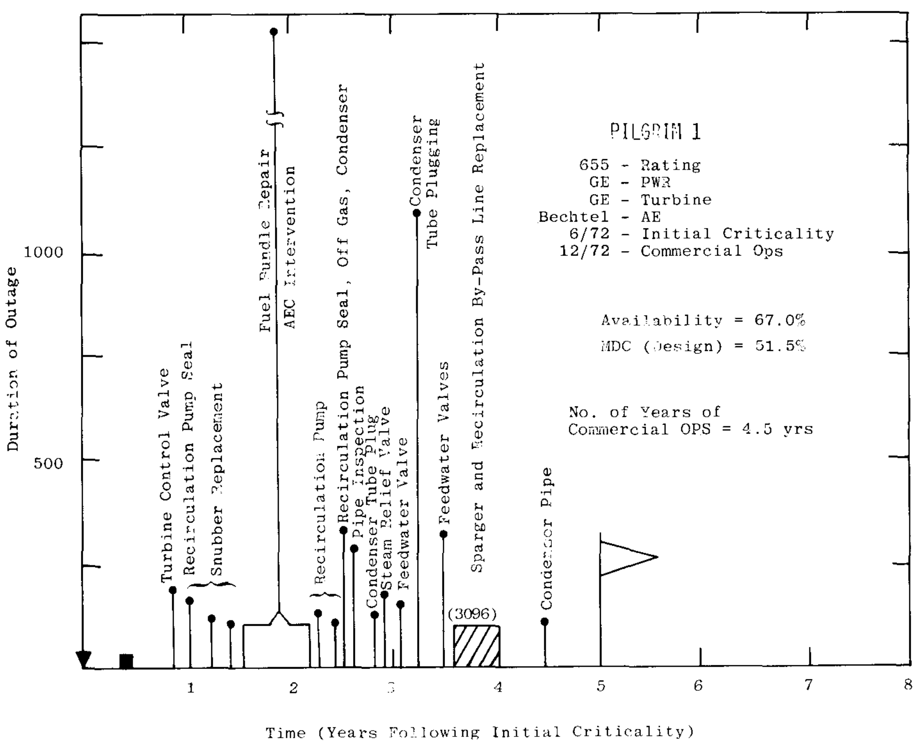

MONTICELLO
545 - Rating
GE - BWR
GE - Turbine
Bechtel - AE
12/70 - Initial Criticality
6/71 - Commercial Ops.
Availability = 77.6%
Capacity Factor
(Design) = 69.4%
No. of Years of
Commercial Ops. = 6.0
Duration of Outage
1000
500
1771
Turbine Valves
Feedwater Leak, Hydrogen Generator
Torus Modification
Steam Valves
Transformer
Feedwater Valves, Condenser Cleaning
(1865)
Control Rod Test, Snubber Inspection
Control Rod Sequence Change
Rebuild Snubbers
(1619)
Control Rod Drive Mechanism Filter
Off Gas System
Steam Valves
<Recirculation By-Pass Line Repair>
(688)
(1662)
<Replace Feedwater Sparger>
Recirculation Pump Seals,
Steam Valves
Feedwater Leaks
Drywell
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

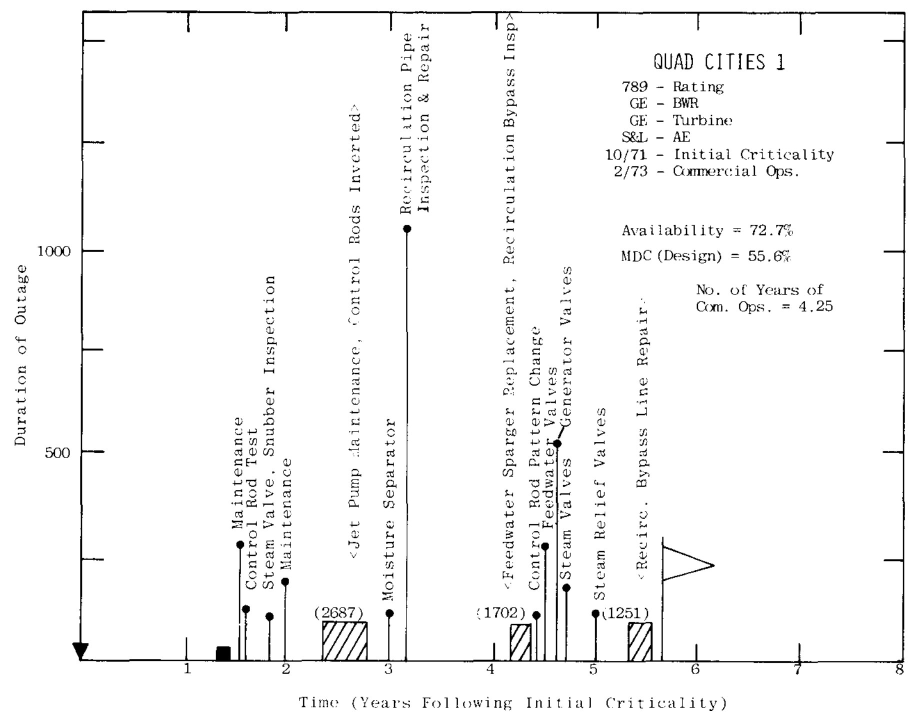

NINE MILE POINT
610 - Rating
GE - BWR
GE - Turbine
Utility - AE
9/69 - Initial Criticality
12/69 - Commercial Ops.
Availability = 67%
Capacity Factor
(Design) = 55.8%
No. of Years of
Commercial Ops. = 6.5 (7.5)
Duration of Outage
1000
500
Recirculation Valve
Station Design Modification
Steam System
(931)
Recirculation Valve
(1790)
<Remove Baffles From Torus>
Turbine Lubrication Piping
Steam Valve
Generator
(1517)
Steam System Valves
(2264)
<Plug Spare In-Core LPRM Hsgs.>
Recirculation Pipe Inspection
Pipe Inspection
(1391)
Feedwater Pipe Inspection
Turbine Bearing
Feedwater Tubing
Generator Short,
Recirculation Pump Seal
<Feedwater Sparger,
CDM Nozzles>
(3225)
1
2
3
4
5
6
7
Time (Years Following Initial Criticality)

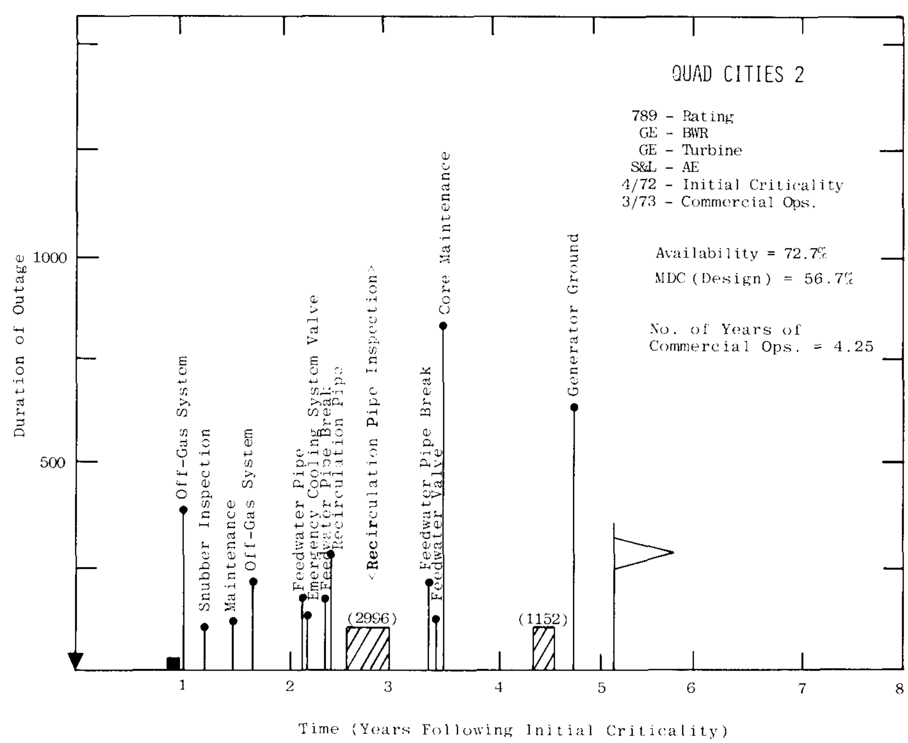

OYSTER CREEK
650 Rating
GE Turbine
B&R AE
5/69 Initial Criticality
12/69 Commercial OPS
Availability = 75.6
Capacity Factor (Design) = 63.5
No. of Years of Commercial OPS = 6.5 (7.5)
No Data
Turbine Control Failure
Air Compressor
Drywell Maintenance
Generator Problem
Feedwater Valve
Valve Control, Snubbers,
Inspect Snubbers
Torus Valves
< Primary Coolant Leak at Instr Penetration Bottom Head>
Feedwater Valves
Pipe Inspection
<Turbine Inspection>
Condenser Tubes
Condenser Tubes
Condenser Tubes
<Retube Condenser>
Feedwater Valves
Feedwater Sparger Repl.
End of Data
(1296)
(1195)
(1268)
(1922)
(1819)
(1788)
(2497)
Duration of Outage (Hours)
1000
500
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

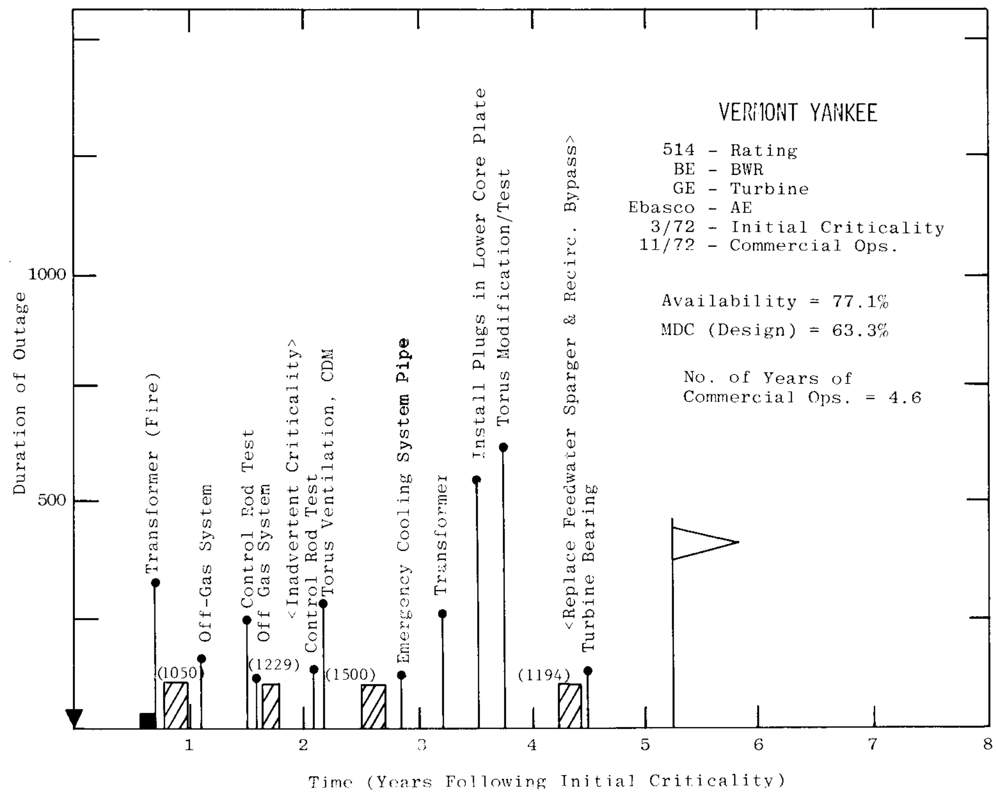

PEACH BOTTOM 2
1065 - Rating
GE - BWR
GE - Turbine
Bechtel - AE
9/73 - Initial Criticality
7/74 - Commerial Ops.
Availability = 72%
Capacity Factor (Design) = 58.1%
No. of Years of Commercial OPA = 2.75
Condenser
Recirculation Pipe Inspection
Core Spray Pipe Weld Inspection
Containment (LLRT)
Lightning/Faulty Breaker
Local Power Range Monitor
In-Core Modification
Turbine Control
(2134)
Duration of Outage
1000
500
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)


PEACH BOTTOM 3
1065 - Rating
GE - BWR
GE - Turbine
Bechtel - AE
8/74 - Initial Criticality
12/74 - Commerial Ops.
Availability = 73.8%
MDC (Design) = 55%
No. of Years of
Commercial Ops = 2.5
Duration of Outage
500
1000
Recirculation Pipe Inspection, Condenser Repair
Recirculation Pump Seal
Turbine Control
Recirculation Pump Seal
Modify LPRM-In-Core Instrumentation
Radwaste Flood
Recirculation Pump
Generator
MSIV
Steam Valves
Core Spray Line Repair
CDM Nozzle Inspection
(2630)
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)


PILGRIM 1
655 - Rating
GE - PWR
GE - Turbine
Bechtel - AE
6/72 - Initial Criticality
12/72 - Commercial Ops
Availability = 67.0%
MDC (Design) = 51.5%
No. of Years of
Commercial OPS = 4.5 yrs
Duration of Outage
500
1000
Turbine Control Valve
Recirculation Pump Seal
Snubber Replacement
Fuel Bundle Repair
AEC Intervention
Recirculation Pump
Recirculation Pump Seal, Off Gas, Condenser
Pipe Inspection
Condenser Tube Plug
Steam Relief Valve
Feedwater Valve
Condenser Tube Plugging
Feedwater Valves
Sparger and Recirculation By-Pass Line Replacement
(3096)
Condenser Pipe
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)


QUAD CITIES 1
789 - Rating
GE - BWR
GE - Turbine
S&L - AE
10/71 - Initial Criticality
2/73 - Commercial Ops.
Availability = 72.7%
MDC (Design) = 55.6%
No. of Years of
Com. Ops. = 4.25
Duration of Outage
500
1000
Time (Years Following Initial Criticality)
1
2
3
4
5
6
7
8
Maintenance
Control Rod Test
Steam Valve, Snubber Inspection
Maintenance
(2687)
Moisture Separator
Recirculation Pipe
Inspection & Repair
(1702)
Control Rod Pattern Change
Feedwater Valves
Generator Valves
Steam Valves
Steam Relief Valves
(1251)


QUAD CITIES 2
789 - Rating
GE - BWR
GE - Turbine
S&L - AE
4/72 - Initial Criticality
3/73 - Commercial Ops.
Availability = 72.7%
MDC (Design) = 56.7%
No. of Years of
Commercial Ops. = 4.25
Duration of Outage
500
1000
Time (Years Following Initial Criticality)
1
2
3
4
5
6
7
8
Off-Gas System
Snubber Inspection
Maintenance
Off-Gas System
Feedwater Pipe
Emergency Cooling System Valve
Feedwater Pipe Break
Recirculation Pipe
(2996)
Feedwater Pipe Break
Feedwater Valve
Core Maintenance
(1152)
Generator Ground


VERMONT YANKEE
514 - Rating
BE - BWR
GE - Turbine
Ebasco - AE
3/72 - Initial Criticality
11/72 - Commercial Ops.
Availability = 77.1%
MDC (Design) = 63.3%
No. of Years of
Commercial Ops. = 4.6
Duration of Outage
1000
500
Transformer (Fire)
(1050)
Off-Gas System
Control Rod Test
Off Gas System
(1229)
<Inadvertent Criticality>
Control Rod Test
Torus Ventilation, CDM
(1500)
Emergency Cooling System Pipe
Transformer
Install Plugs in Lower Core Plate
Torus Modification/Test
(1194)
<Replace Feedwater Sparger & Recirc. Bypass>
Turbine Bearing
1
2
3
4
5
6
7
8
Time (Years Following Initial Criticality)

APPENDIX B

REVIEW OF CAUSES OF REDUCED PLANT CAPACITY FACTOR

As noted in Section 2.2, the mean unweighted cumulative BWR plant capacity factor is approximately 6.6 percentage points less than the mean PWR plant capacity factor. Since the plant availability is approximately the same for PWR and BWR plants, the difference in capacity factors results from power restrictions. No attempt has been made to quantify the overall effect of these restrictions, an accurate assessment would require utility cooperation in identifying the causes and lengths of these restrictions. The following is a brief summary of some typical events which are causing power limitations on LWRs.

BWR Power Restrictions:

1) Off-gas activity: In some BWR plants the power has been limited in order to maintain the off-gas activity within that specified in the technical specifications. The range of power reductions is from 1% to 60% for at least 10 of the BWR plants. It is estimated that approximately 5-10% of the reduction in capacity factor is associated with controlling the high off-gas activity.

2) In-core vibration: Concern over potential in-core vibration resulted in power restrictions ranging from 50% to 90% of rated power on 5 plants.

3) Fuel densification*: Concerns over fuel densification resulted in derating two plants to 75 and 83% of rated power.

4) Equipment problems have also resulted in power deratings until repairs can be made. The following four examples were identified by the utilities as:

	% Derated Power
a) Feedwater Sparger Vibration	80%
b) Condenser Repair	50%
c) Recirculation Pump Repair	86%
d) Flow Restrictors Modification	90%

*It has been estimated by EPRI[33] that fuel considerations have accounted for an average of 3-6% lost capacity factor in nuclear plants. The principal reasons for this lost capacity are: (a) Restrictions on power maneuvers which require either slow ascents to full power or soak periods for the fuel; (b) Core deratings forced by high coolant activity due to leaking fuel elements or high off-gas activity.

5) Power peaking: Unusually high calculated power peaking in the lower part of the core. This caused power to be restricted to 70%.

6) Condenser cooling water temperature: Exceeding the state regulations on water returned to heat sink caused a power restriction of 40-75%.

7) Power not required: A limitation of 50% power on one plant.

PWR Power Restrictions:

1) Fuel Problems*: The following examples have been identified by the respective utilities:

		Derated Power	No. of Plants
a)	Unspecified	75-90%	2
b)	High coolant activity	80%	1
c)	Densification	75-90%	4
d)	Bowing	83%	1
e)	Incore detector failures	60%	1
f)	Extend fuel cycle	60-90%	1

2) Fuel License Limitation: This occurred during the initial year of commercial operation and was the cause of a substantial portion of lost capacity during that year.

3) Equipment Problems: The following examples have been identified by the respective utilities:

		Derated Power	No. of Plants
a)	Main Steam Isolation Valve	75%	3
b)	Condenser Tube Repair	50%	2
c)	Partial Turbine Repair	88-98%	2
d)	Feedwater Pump	65-75%	2
e)	Control Rods	80%	1

4) Power Not Required: A limitation on power of 50% on one plant was due to lack of need for power.

*See footnote on page 1 of this Appendix.

APPENDIX C

SUMMARY COMPARISON OF PLANT OUTAGES BY CONTRIBUTORY CAUSES MAY 1974 - JUNE 1977

Table C-1. Summary of Plant Outages (Total Hours) by Major Contributory Categories for 1977 (from Reference 10)

Category	Jan	Feb	Mar	April	May	June	6 Mo. Total
Maintenance or Test	883	1768	1617	1391	2485	994	9138
Refueling	4645	3905	6495	6328	5726	5676	32725
Equipment Failure	2245	1609	2888	2723	1996	2281	13752
Other/ Multi	125	985	906	1037	1490	1327	5870
Operator Error	78	156	0	118	121	89	562
Regulatory Restriction	0	36	0	951	671	0	1658
Administration	0	0	99	--	0	23	122
Operator Training	--	--	--	22	0	110	132
Total Outage (Hours/ Month)	7976	8459	12,004	12,570	12,489	10,571	64,009

Table C-2. Summary of Plant Outages (Total Hours) by Major Contributory Categories for 1976 (from Reference 10)

Category	Jan	Feb	Mar	April	May	June	July	Aug	Sept	Oct	Nov	Dec	Average Monthly Total	Yearly Total	Rank (by % Yearly Total)
Maintenance or Test	3919	3408	3013	3783	4476	2903	1658	1805	2558	973	2439	720	2638	31,655	3
Refueling	1667	5761	9403	7946	7259	3465	2173	908	2009	6425	6075	3600	4724	56,691	1
Equipment Failure	4382	2328	2173	2697	4050	2755	1922	2731	2143	4583	3918	2644	2936	35,226	2
Other/Multi	366	152	733	1951	136	819	2239	1716	1501	1151	763	841	1081	12,368	4
Operator Error	13	19	395	1425	138	167	20	102	313	21	0	32	220	2,645	6
Regulatory Restriction	24	58	0	719	767	620	2378	1247	--	--	27	8	496	5340	5
Administration	139	430	119	351	15	--	2	12	--	--	68	0	94	1136	7
Operator Training	8	95	0	45	24	--	65	--	16	--	--	0	21	253	8
Total Outage (Hours/Month)	10518	12251	15836	18917	16865	10829	10457	8521	8540	12053	13290	7845	12,160	145,319	

Table C-4. Summary of Plant Outages (Total Hours) by Major Contributory Categories for 1974 (from Reference 10)

Category	May	June	July	Aug	Sept	Oct	Nov	Dec	Average Monthly Total	Yearly Total
Maintenance or Test	3150	1902	1260	383	1872	2779	940	1657	1742	13,943
Refueling	6184	4474	2650	1187	1440	2538	5054	4211	3467	27,738
Equipment Failure	572	793	837	1225	3910	3033	2303	2037	1839	14,710
Other/ Multi	744	--	939	763	298	42	35	28	355	2,839
Operator Error	68	1441	76	42	78	317	78	10	264	2,110
Regulatory Restriction	744	761	650	--	8	25	1440	--	454	3,628
Administration	--	39	--	27	451	8	--	--	66	525
Operator Training	--	38	--	--	--	35	158	--	29	231
Total Outage (Hours/ Month)	11,462	9448	6412	3627	8057	8777	9998	7942	8037	65,723

Table C-3. Summary of Plant Outages (Total Hours) by Major Contributory Categories for 1975 (from Reference 10)

Category	Jan	Feb	Mar	April	May	June	July	Aug	Sept	Oct	Nov	Dec	Average Monthly Outage	Yearly Total	Rank (by % Yearly Total)
Maintenance or Test	1739	1220	2846	4020	3263	3958	2802	2475	4057	4469	3226	2046	3010	36,121	1
Refueling	3503	2031	2854	4692	4217	4415	1676	744	1286	2712	4774	2772	2982	35,776	2
Equipment Failure	2901	2205	1679	1417	2253	2144	1823	2220	3296	2853	2409	3082	2357	28,282	3
Other	676	1379	2423	1385	849	263	229	347	93	529	150	200	711	8,530	4
Operator Error	94	49	170	23	107	619	294	52	187	107	13	99	151	1,817	5
Regulatory Restriction	55	1073	0	35	0	0	0	540	0	0	0	0	142	1,703	6
Administrative	5	0	0	0	190	21	0	66	0	0	0	0	24	282	7
Operator Training	0	0	0	5	0	42	0	0	0	0	0	0	0	47	8
Total Outage (Hours/Month)	8973	7954	9972	11576	10879	11462	6824	6444	9018	10670	10582	8192	9377	112,558	