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NUCLEAR POWER PLANT OPERATING EXPERIENCE 1974 - 1975

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ABSTRACT

The operating experience of U. S. nuclear power plants in commercial operation during 1974 and 1975 is summarized. Power generation statistics, plant outages, reportable occurrences, fuel element performance, radiation exposure and radioactive effluent releases for each plant are presented. Summary highlights of these areas are discussed.

The 1974 data covers 40 plants -- 17 boiling water reactor plants and 23 pressurized water reactor plants; while the 1975 data includes 51 plants -- 23 boiling water reactor plants and 28 pressurized water reactor plants.

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NUCLEAR POWER PLANT OPERATING EXPERIENCE

1974 - 1975

1.0 INTRODUCTION

This report summarizes the operating experience of licensed nuclear power plants for the years 1974 and 1975. Operating statistics and data are presented for each plant that was in commercial operation at year end and had sufficient electrical generation for meaningful analyses.

On the end of 1974, forty-four licensed nuclear power plants were in "commercial operation."[1] Four of these, however, had less than 250 hours of generator-on-line time and were therefore not included in this summary. The operating experience of the 40 remaining plants was reviewed; 17 were boiling water reactor (BWR) plants and 23 were pressurized water reactor (PWR) plants. Peach Bottom 1, a small gas cooled reactor underg g decommissioning was not included. In comparison, the 1973 summary "Nucl Power Plant Operating Experience During 1973," OOE-ES-004, included 30 plants; 14 BWRs and 16 PWRs.

For the year 1975, twelve new plants were considered. A total of 54 plants were in commercial operation by the end of the year but only 51 are included in this study; 23 BWRs and 28 PWRs. Indian Point 1, a PWR, was not included because it had been defueled and had not operated during the entire year. Millstone Point 2 and Trojan started commercial operation in December and had too few hours of generator-on-line to be included.

The plants included in this report are presented in Table 1 with their commercial operation dates.

Operating statistics for each plant, such as plant availability and capacity factors and percent of scheduled and forced outages are presented. Because the definitions of these terms vary somewhat within the industry and Government, a glossary of these definitions is presented in Appendix A. Also included in this report are summaries of Licensee Event Reports, Abnormal Occurrences, Fuel Performance, Occupational Radiation Exposures, and Radioactive Releases.

Section 3 of this report on Plant Outages was prepared for the NRC by the Nuclear Safety Information Center of the Union Carbide Company at the Oak Ridge National Laboratory under Interagency Agreement ERDA Number 40-547-75, SOEW Number 80-76-01.

The primary sources of information for preparation of this report were the licensee's semiannual operating reports, Licensee Event Reports, Special Reports, and the NRC's Operating Units Status Report (the monthly "Gray Book"). These reports may be reviewed at the NRC's Public Document Room, located at 1717 H. Street, NW, Washington, D.C. Documents pertaining to specific plants are also available at the local public document rooms located in the vicinity of each plant.

1/ See Glossary, Appendix A.

NUCLEAR POWER PLANTS IN COMMERCIAL OPERATION – 12/31/75

1974 Summary	1975 Summary	PLANT NAME	COMMERCIAL DATE	REACTOR TYPE
INCLUDED IN 1974 SUMMARY OF OPERATING EXPERIENCE	INCLUDED IN 1975 SUMMARY OF OPERATING EXPERIENCE	Dresden 1	7/60	BWR
		Yankee Rowe	8/60	PWR
		Indian Point 1*	10/60	PWR
		Big Rock Point	3/63	BWR
		Humboldt Bay 3	8/63	BWR
		San Onofre 1	1/68	PWR
		Conn Yankee	1/68	PWR
		La Crosse	9/69	BWR
		Oyster Creek 1	12/69	BWR
		Nine Mile Point	12/69	BWR
		RE Ginna	3/70	PWR
		Point Beach 1	12/70	PWR
		H. B. Robinson	3/71	PWR
		Millstone Point 1	3/71	BWR
		Monticello	7/71	BWR
		Dresden 3	11/71	BWR
		Palisades	12/71	PWR
		Dresden 2	6/72	BWR
		Vermont Yankee	11/72	BWR
		Pilgrim 1	12/72	BWR
		Surry 1	12/72	PWR
		Turkey Point 3	12/72	PWR
		Surry 2	12/72	PWR
		Maine Yankee	12/72	PWR
		Quad Cities 1	2/73	BWR
		Quad Cities 2	3/73	BWR
		Point Beach 2	4/73	PWR
		Oconee 2	7/73	PWR
		Indian Point 2	8/73	PWR
		Turkey Point 4	9/73	PWR
		Oconee 1	9/73	PWR
		Prairie Island 1	12/73	PWR
		Zion 1	12/73	PWR
		Peach Bottom 2	5/74	BWR
		Fort Calhoun 1	6/74	PWR
		Kewaunee	6/74	PWR
		Cooper	7/74	BWR
		Browns Ferry 1	8/74	BWR
		Three Mile Island	9/74	PWR
		Zion 2	12/74	PWR
NOT INCLUDED		Oconee 3	12/74	PWR
		Arkansas 1	12/74	PWR
		Prairie Island 2	12/74	PWR
		Peach Bottom 3	12/74	BWR
		Duane Arnold	2/75	BWR
		Browns Ferry 2	3/75	BWR
		Rancho Seco	4/75	PWR
		Calvert Cliffs 1	5/75	PWR
		Fitzpatrick	7/75	BWR
		D. C. Cook	8/75	PWR
		Brunswick 2	11/75	BWR
		Hatch	12/75	BWR
	NOT INCLUDED	Trojan	12/75	PWR
		Millstone Point 2	12/75	PWR

*Not included in 1975 Summary
Peach Bottom 1 excluded from report

2.0 POWER GENERATION

2.1 Introduction

Tables 2.1 and 2.2 summarize the net electrical plant availability and capacity factors for BWR's and PWR's. This data covers the periods 1974 and 1975. Similar information was presented for 1972 and 1973 in a number of reports prepared by the Atomic Energy Commission.[1,2,3]

2.2 Electrical Output For 1974 and 1975

In 1974 the total net electrical output for 40 nuclear power plants in commercial operation was 92.2 billion kilowatt hours. This was approximately 5% of the total electrical energy generated in the United States for that year from all sources and represents a 19% increase in the total net electrical energy output generated by nuclear power plants over the previous year.

In 1975, the contribution made to the total electrical energy generated by nuclear power plants increased substantially. During that year, 51 nuclear power plants in commercial operation generated 167 billion kilowatt hours of electrical energy or approximately 8.7% of the total electrical energy generated in the United States. This represents a 118% increase over 1973 and a 82% increase over 1974 in the total net electrical output generated by nuclear power plants.

Of the total nuclear generation for 1974 i.e., 92.2 billion kilowatt hours, 49% was produced by 17 BWR's and 51% by 23 PWR's. The energy distribution for 1975 was 35% by 23 BWR's and 65% by 28 PWR's.

2.3 Plant Availability Factors For 1974

The average plant availability factor for 1974 was 68.2% for the 40 nuclear power plants in commercial operation. The average BWR and PWR availability factors for this period were 70.2% and 66.7%, respectively.

The BWR availability factors ranged from 35.5% to 90%. Two BWR plants had availability factors below 50% while 12 reported availability factors above 70%. The PWR availability factors ranged from 5.5% to 91.2%. Three PWR's had availability factors below 50% while 9 had availability factors above 70%. This information is presented in Table 2.1.

Overall, the average nuclear plant availability factor for 1974 decreased by 2.8% compared to 1973. Tables 2.3 and 2.4 summarize the causes which affected the plant availability factors for the five nuclear power plants discussed above which had availability factors below 50%. Additional information on individual plants is presented in Appendix B.

TABLE 2.1 - POWER GENERATION STATISTICS FOR 1974

Plants	Design Electrical Capacity (MWe-net)	Electrical Output (MWHe-net)	Plant Availability Factor (%)	Plant Capacity Factor (%)	Plant Age (1) (Years)
Boiling Water Reactors					
Big Rock Point	72	337,542	70.3	54.3	12.1
Browns Ferry 1	1098	5,168,631	74.5	55.4	1.2
Cooper Station	778	1,885,632	75.4	54.0	.6
Dresden 1	200	352,939	35.5	20.1	14.7
Dresden 2	809	3,379,588	64.1	48.2	4.7
Dresden 3	809	3,200,269	65.0	45.7	3.4
Humboldt Bay	65	365,930	83.8	66.3	11.7
LaCrosse	50	313,440	81.0	79.2	6.7
Millstone Point 1	690	3,604,240	79.1	63.1	4.1
Monticello	545	2,923,836	74.9	62.0	3.8
Nine Mile Point	610	3,296,654	70.5	61.7	5.2
Oyster Creek	650	3,673,489	70.4	67.6	5.3
Peach Bottom 2	1065	3,713,475	90.6	81.8	.9
Pilgrim 1	655	1,973,033	39.2	33.6	3.5
Quad Cities 1	809	3,562,941	61.9	50.8	2.7
Quad Cities 2	809	4,469,705	82.6	63.8	2.6
Vermont Yankee	514	2,482,564	74.1	56.2	2.3
BWR Total	11,28[illegible]	44,703,9[illegible]			
BWR Average			70.2	56.6	4.7
Pressurized Water Reactors					
Connecticut Yankee	575	4,350,932	91.2	91.9	7.4
Fort Calhoun	457	2,416,252	83.5	60.4	1.4

(1) Computed from date of first electrical generation through December 31, 1974.

Continuation of Table 2.1 - For 1974

Plants	Design Electrical Capacity (MWe-net)	Electrical Output (MWHe-net)	Plant Availability Factor (%)	Plant Capacity Factor (%)	Plant Age (1) (Years)
RE Ginna	490	2,097,216	62.4	51.7	5.1
Indian Point 1	265	1,232,560	63.6	55.8	12.3
Indian Point 2	873	3,324,048	59.4	43.5	1.5
Kewaunee	560	1,589,173	75.2	62.2	.8
Maine Yankee	790	3,574,301	63.7	51.6	2.1
Oconee 1	886	3,998,488	60.1	52.4	1.7
Oconee 2	886	1,387,526	68.5	58.2	1.8
Palisades	821	78,298	5.5	1.1	3.0
Point Beach 1	497	3,142,055	81.5	76.2	4.2
Point Beach 2	497	3,178,408	81.0	76.9	2.4
Prairie Island 1	530	1,432,750	43.9	31.5	1.1
Robinson 2	707	4,813,207	83.3	82.6	4.3
San Onofre 1	450	3,145,109	86.1	83.5	7.5
Surry 1	823	3,318,073	54.8	48.1	2.5
Surry 2	823	2,634,573	44.8	38.2	1.8
Three Mile Island 1	819	1,977,812	88.1	86.0	.5
Turkey Point 3	745	3,623,905	69.9	62.1	2.2
Turkey Point 4	745	4,293,374	77.1	74.1	1.5
Yankee Rowe	175	911,452	69.6	59.5	14.1
Zion 1	1050	3,477,561	57.2	45.1	1.5
Zion 2	1050	963,986	59.8	43.9	1.0
PWR total	15,514	47,540,051			
PWR Average			66.7	58.1	3.6
All plants total	26,798	92,243,959			
All plants average			68.2	57.5	4.2

Table 2.2 - POWER GENERATION STATISTICS FOR 1975

Plants	Design Electrical Capacity MWe-net)	Electrical Output (MWHe-net)	Plant Availability Factor (%)	Plant Capacity Factor (Using MDC) (%)	Plant Capacity Factor (Using Design) (MWE (%)	Plant Age(2) (Year)
Boiling Water Reactors						
Big Rock Point	72	290,532	59.8	46.7	46.1	13.1
Browns Ferry 1	1098	1,347,943	17.5	14.4	14.4	2.2
Browns Ferry 2	1098	1,374,133	18.0	14.7	14.7	1.3
Brunswick 2	821	1,405,366	93.2	59.7	58.8	.7
Cooper Station	778	3,853,630	83.6	57.6	56.5	1.6
Dresden 1	200	696,781	57.2	39.8	39.8	15.7
Dresden 2	809	2,960,092	55.1	42.3	41.2	5.7
Dresden 3	809	2,190,003	51.5	31.3	30.[illegible]	4.4
Duane Arnold	538	2,298,183	79.5	50.9	48.8	1.6
Fitzpatrick	821	2,154,564	70.1	100.9(3)	50.5	.9
Hatch 1	786	3,102,479	70.3	47.1	45.5	1.2
Humboldt Bay	65	382,938	83.9	69.4	69.4	12.7
La Crosse	50	263,368	69.6	62.6	60.1	7.7
Millstone Point 1	690	3,896,991	75.6	68.4	68.4	5.1
Monticello	545	2,879,458	72.[illegible]	61.1	60.3	4.8
Nine Mile Point 1	610	3,044,948	72.1	56.9	56.9	6.2
Oyster Creek	650	3,145,826	73.3	64.6	61.6	6.3
Peach Bottom 2	1065	5,082,479	75.8	55.2	54.5	1.9
Peach Bottom 3	1065	5,232,3[illegible]6	86.0	58.3	56.7	1.3
Pilgrim 1	670	2,587,248	71.3	44.1	42.9	3.5
Quad Cities 1	809	4,270,882	85.1	217 (3)	62.3	3.7
Quad Cities 2	809	2,475,331	51.7	126 (3)	36.2	3.6
Vermont Yankee	514	3,561,206	87.8	80.7	79.1	3.3
BWR total	15,372	55,507,749				
BWR average			67.8	63.8	50.3	4.5

(2) Computed from date of first electrical generation through December 31, 1975

(3) Plants with MDCs less than Design Electrical Capacities may have capacity factors greater than 100%

CONTINUATION OF TABLE 2.2 - FOR 1975

Plants	Design Electrical Capacity (MWe-net)	Electrical Output (MWHe-net)	Plant Availability Factor (%)	Plant Capacity Factor (Using MDC) (%)	Plant Capacity Factor (Using Design) (MWE (%)	Plant Age(2) (Year)
Pressurized Water Reactors						
Arkansas 1	850	4,879,862	76.5	66.6	65.5	1.4
Calvert Cliffs 1	845	4,386,319	90.4	78.8	74.6	1.0
Connecticut Yankee	575	4,121,428	89.9	87.9	81.8	8.4
Cook 1	1090	4,457,776	83.7	82.0	62.7	.9
Fort Calhoun	457	2,080,777	67.4	52.0	52.0	2.4
Indian Point 2	873	4,885,079	74.8	64.5	63.8	2.5
Kewaunee	560	3,341,153	88.2	71.3	68.1	1.8
Maine Yankee	790	4,502,452	79.9	67.6	65.1	3.1
Oconee 1	886	5,285,630	76.2	69.3	68.0	2.7
Oconee 2	886	4,967,625	73.1	65.1	63.9	2.1
Oconee 3	986	5,037,298	77.2	66.0	64.8	1.3
Palisades	821	2,427,933	64.5	40.5	33.7	4.0
Point Beach 1	497	2,921,849	71.9	69.3	67.6	5.2
Point Beach 2	497	3,741,304	93.9	87.9	85.8	3.4
Prairie Island 1	530	3,694,168	86.3	81.1	80.0	2.1
Prairie Island 2	530	3,176,256	80.3	69.7	68.4	1.0
Rancho Seco	913	1,326,506	27.5	26.2	25.3	1.2
RE Ginna 1	490	3,041,203	76.7	73.9	73.9	6.1
Robinson 2	707	4,170,774	72.7	71.6	67.3	5.3
San Onofre 1	450	3,245,108	87.4	86.2	82.4	8.5
Surry 1	823	3,916,527	62.0	56.7	54.3	3.5
Surry 2	823	5,053,082	79.6	73.2	70.1	2.8
Three Mile Island 2	819	5,541,523	82.2	79.9	77.3	1.5
Turkey Point 3	745	4,374,597	79.4	75.0	72.9	3.2
Turkey Point 4	745	3,989,524	70.5	68.4	65.7	2.5
Yankee Rowe	175	1,193,421	82.4	77.8	77.8	15.1
Zion 1	1050	4,909,363	70.0	65.9	54.1	2.5
Zion 2	1050	4,828,978	72.2	64.9	53.3	2.0
PWR total	20,463	109,497,497				
PWR Average			76.3	69.3	65.7	3.4

CONTINUATION OF TABLE 2.2 - FOR 1975

Plants	Design Electrical Capacity (MWe-net)	Electrical Output (MWHe-net)	Plant Availability Factor (%)	Plant Capacity Factor (Using MDC) (%)	Plant Capacity Factor (Using Design) (MWE (%)	Plant Age(2) (Year)
All Plants Total	36,291	167,086,023				
All Plants Average			72.4	66.5	58.6	4.0

TABLE 2.3 - PRIMARY CAUSES AFFECTING PLANT AVAILABILITY FACTORS FOR 1974

Boiling Water Reactors

Plant Name (Plant Availability Reactor)	Primary Causes
Dresden 1 (57.2%)	Facility had problems with containment leakage, control of canal water quality and coupling of control rod blades. Extended refueling also contributed to extended shutdown time.
Pilgrim 1 (39.2%)	Most of the shutdown time was caused by legal problems involving the replacement of twenty 7x7 fuel bundles with 8x8 fuel bundles. Questions were raised regarding the adequacy of the analysis performed by the licensee to justify replacement of bundles. Additional shutdown time was also required to replace defective seals on recirculation pumps.

TABLE 2.4 - PRIMARY CAUSES AFFECTING PLANT AVAILABILITY FACTORS FOR 1974

Pressurized Water Reactors

Plant Name (Plant Availability Factor)	Primary Causes
Palisades (5.5%)	Most of the shutdown time can be attributed to repair of primary to secondary tube leaks in the steam generator in addition to leaks in the condenser tubes. A fraction of the shutdown time also involved repair of damaged turbine blades.
Prairie Island I (43.9%)	The majority of the shutdown time was used to repair turbine blades. Some of the shutdown time was expended on turbine blade modifications.
Surry 2 (44.8%)	Considerable shutdown time required to repair broken turbine blades, reactor coolant pump shaft and loop isolation valve. Some time also required to perform maintenance on main steam non-return valves.

2.4 Plant Availability Factors For 1975

In 1975, the average plant availability factor was 72.4% for the 51 nuclear power plants in commercial operation. This was up by 4.2% compared to 1974. During this period the average BWR and PWR availability factors were 67.8% and 76.3%, respectively.

The BWR availability factors ranged from 17.5% to 93.2%. Two of the BWR's in commercial operation had availability factors below 50% while 15 had availability factors above 70%.

The PWR availability factors ranged from 27.5% to 93.9%. One PWR had an availability factor below 50%, while 24 plants had availability factors of 70% or greater.

The 1975 information is presented in Table 2.2. The circumstances which affected the plant availability factors for the two BWR's and one PWR with availability factors below 50% are summarized in Table 2.5. Additional information on individual plants is presented in Appendix B.

2.5 Plant Capacity Factors For 1974

In 1974, the average capacity factor (using design MWe) for the 40 commercial nuclear power plants was 58%.

The average capacity factor for 17 BWR's was 56.6% ranging from 20.1% to 81.8%. Four BWR's had capacity factors below 50% while two had capacity factors above 70%.

The average capacity factor for 23 PWR's was 57.5% ranging from 1.3% to 91.9%. Seven PWR's had capacity factors below 50% while seven had capacity factors above 70%.

Tables 2.6 and 2.7 present additional data pertaining to the distribution of availability and capacity factors.

2.6 Plant Capacity Factors For 1975

For 1975, the capacity factors were also computed using maximum dependable capacity (MDC)* and design capacity (MWe). The average capacity factors for 51 commercial nuclear power plants using MDC was 66.5% and 58.6% using MWe.

The average capacity factors for 23 BWR's were 63.4% and 50.3% using MDC and MWe, respectively. The MDC capacity factors range varied from 14.4% to over 100%. The MWe capacity factors range varied from 14.4% to 79.1%. Eight BWR's had capacity factors below 50% (using MDC) while four had capacity factors above 70%. Using design values (MWe), ten BWR's had capacity factors below 50% while one had a capacity factor above 70%.

*See Glossary, Appendix A for definition.

TABLE 2.5 - PRIMARY CAUSE AFFECTING PLANT AVAILABILITY FACTORS FOR 1975

Boiling Water Reactors

Plant Name (Plant Availability Factors)	Primary Causes
Browns Ferry 1 (17.5%) Browns Ferry 2 (18%)	Most of the shutdown time resulted from a cable fire which kept both plants down for the balance of the year for repairs.

Pressurized Water Reactor

Plant Name (Plant Availability Factors)	Primary Causes
Rancho Seco (27.5%)	Almost all of the down time was expended rebuilding turbine blades. Many of the original blades were cracked or had failed completely.

Table 2.6 - DISTRIBUTION OF PLANT AVAILABILITY
FACTORS AND PLANT CAPACITY FACTORS FOR 1974

Availability Factor %	Number of BWR's	Number of PWR's	Total No. of Plants
90 and over	1	1	2
80 - 90	3	6	9
70 - 80	8	2	10
60 - 70	3	7	10
50 - 60	0	4	4
less than 50	2	3	5
	17	23	40
Average Availability Factors	70.2%	66.7%	68.2%

Capacity Factor %	Number of BWR's	Number of PWR's	Total No. of Plants
90 and over	0	1	1
80 - 90	1	3	4
70 - 80	1	3	4
60 - 70	6	3	9
50 - 60	5	7	12
less than 50	4	6	10
	17	23	40
Average Capacity Factors	56.6%	58.1%	57.5%

Table 2.7 - DISTRIBUTION OF PLANT AVAILABILITY FACTORS AND PLANT CAPACITY FACTORS FOR 1975

Availability Factor %	Number of BWR's	Number of PWR's	Total No. of Plants
90 and over	1	2	3
80 - 90	5	8	13
70 - 80	9	14	23
60 - 70	1	3	4
50 - 60	5	0	5
less than 50	2	1	3
	23	28	51
Average Availability Factors	67.8%	76.3%	72.4%

Capacity Factor %	Number of BWR's	Number of PWR's	Total No. of Plants
90 and over	3	0	3
80 - 90	1	5	6
70 - 80	0	8	8
60 - 70	5	11	16
50 - 60	6	2	8
less than 50	8	2	10
	23	28	51
Average Capacity Factor (Using MDC)	63.8%	69.3%	66.5%

Capacity Factors % (using MWe)	Number of BWR's	Number of PWR's	Total No. of Plants
90 and over	0	0	0
80 - 90	0	4	4
70 - 80	1	6	7
60 - 70	6	12	18
50 - 60	7	4	11
less than 50	9	2	11
	23	28	51
Average Capacity Factors (Using MWE)	50.3%	65.7%	58.6%

The average capacity factors for 28 PWR's were 69.3% and 65.7% using MDC and MWe, respectively. The MDC range varied from 26.2% to 87.9%. The MWe range varied from 25.3% to 85.8%.

Two PWR's had capacity factors of less than 50% while 13 had capacity factors greater than 70% using MDC. Using MWe 2 PWR's had capacity factors less than 50% while 9 had capacity factors of 70% or greater.

Table 2.8 summarizes the plant availability and capacity factors for 1974 and 1975. No obvious trends are evident from this two year sample.

2.7 Summary For 1974

The most important problems affecting the availability factors during 1974 included tube leaks in steam generators and condensers, repair of turbine blades, repair of components in electrical power systems, repair of pumps and fuel channel-poison curtain problems. Some of these problems occurred at two or more facilities and had a marked affect on the availability factors of a significant number of BWR's and PWR's. Refueling was also a high consumer of plant shutdown time. It is interesting to note that minimal shutdown time in 1974 was attributed to repairing faulty equipment or components in engineered safeguard systems. In general the low availability factors noted in Table 2.1, caused by many extended shutdowns in 1974, adversely affected the average capacity factor (56.6%) for that year.

2.8 Summary For 1975

Extended shutdowns during 1975 were caused by failure and subsequent repair of, (a) reactor coolant pumps and seals, (b) pipe leaks, (c) condenser leaks, (d) turbine blades and (e) refueling of plants. The above problems were each experienced in at least three or more different reactor facilities.

Less significant shutdown outages involved pipe inspections to satisfy NRC Bulletin requirements, repair or replacement of, (a) fuel elements, (b) recirculation pump seals, (c) control rod drives and various types of heat exchangers. As in 1974 failed components in engineered safeguard systems were not responsible for extended shutdowns. Although there were some improvements in the availability and capacity factors for 1975, compared to 1974, it appears (as in previous years) that extended shutdowns had a significant affect on the capacity factors.

1. Nuclear Power Plant Availability and Capacity Statistics for 1973; May 1974, OOE-OS-002.

2. Evaluation of Nuclear Power Plant Availability; January 1974, OOE-ES-001.

3. Nuclear Power Plant Operating Experience During 1973: December 1974, OOE-ES-004.

Table 2.8 - PLANT AVAILABILITY AND CAPACITY FACTORS AS A FUNCTION OF PLANT AGE FOR 1974 and 1975

Plant Age Group (years)	1974 No. of Plants in Group	1974 Average Availability Factor (%)	1974 Average Capacity Factor (%)	1975 No. of Plants in Group	1975 Average Availability Factor (%)	1975 Average Capacity Factor (%)
0-1.9	14	68.2	56.0	16	73.9	55.8
2-3.9	11	61.6	50.2	17	69.1	55.2
4-5.9	7	73.1	64.4	6	67.8	57.6
6 and over	8	72.7	63.9	12	74.7	65.4

3.0 PLANT OUTAGES

3.1 Plant Outages - 1974

Review of the plant outages or shutdowns that occurred during 1974 provides a means of assessing the nature, number and extent of the operating problems experienced at nuclear power plants, as well as the principal systems and components involved. Most of the data were obtained either from the licensees periodic operating reports or from the data reported by the licensee for the NRC's monthly "Operating Units Status Report."

In some instances omissions or inconsistencies in the reported data necessitated checking with the licensee. In a few cases, outage type was classified differently than reported by the licensee. Where appropriate and sufficient information was available, major outages were subdivided to reflect more accurately the true nature of the work performed during the outage. The forced extension of a scheduled outage was generally reclassified by the NRC staff as a forced outage. In a few cases, work that had been scheduled for later in the year was rescheduled so it could be performed during an unexpected forced outage. These cases were classified as scheduled outages. Refueling of the reactor is classified as a scheduled outage.

In reviewing the outage data, it should be noted that there are significant differences in nuclear plant designs, even between plants of a given type. Therefore, care should be used in interpreting the data.

3.1.1 Plant Outages Statistics

There were 699 outages requiring 113,782 hours of plant downtime reported by the 40 nuclear power plants which were in commercial operation during 1974 as listed in Tables 3-1a and 3-1b. It should be noted that eight of the listed plants were not in commercial operation until long after the first of the year. The calculated percent outages and unit availability factors listed in Tables 3-1a and 3-1b consider only the period of time after the plants were in commercial operation. However, in subsequent outage analyses herein, all outages (including those reported as occurring before commercial operation began) were reviewed to ascertain the type of outage (forced or scheduled), cause, method of shutdown, duration, and the plant system and component primarily associated with the outage.

For the 40 plants thus reviewed, the average total outage time during commercial operation was 32.7% of the year, the average forced outage time was 14.0% and the average scheduled outage time was 18.6%. The average total unit availability for the 40 plants considering only the period when they were in commercial operation was 67.3%. The average performance of the BWRs was somewhat better than the average performance of the PWRs by all of the above measures. However, the PWR statistics were distorted by Palisades which was down 94.5% of the year due to major repairs to its steam generator, condenser tubes and turbine.

Table 3-1a. Summary of BWR Power Plant Outages During Commercial Operation in 1974

	Plant Name	Operational % of Year	Commercial Operation % of Year	Scheduled Outage During Commercial Operation		Forced Outage During Commercial Operation		Total Outage During Commercial Operation		Unit Availability Commercial Operation %	Fuel Inspection or Replacement	Reactor Vessel Internals	Control Rods and Control Rod Systems	Main Turbine	Condenser and Feedwater System	Auxiliary Systems (Electrical & Process)	Engineered Safety Systems	Hydraulic Restraints	Off-Gas Systems	Main Steam Systems	Valves and Valve Operators Except Main Steam Line Valves	Reactor Coolant or Recirculation Pumps	Miscellaneous
				hrs.	%	hrs.	%	hrs.	%	%													
1.	Big Rock Point 1	100	100	1555	17.8	1045	11.9	2600	29.7	70.3	1		1									1	
2.	Browns Ferry 1	100	41.9	488	13.3	71	1.9	559	15.2	84.8*					1		1						
3.	Cooper Station	>50	50	469	10.6	719	16.4	1188	27.1	72.9*					1						1	1	
4.	Dresden 1	100	100	3062	35.0	2588	29.5	5650	64.5	35.5	1		1	1		2							
5.	Dresden 2	100	100	1638	18.7	1509	17.2	3147	35.9	64.1	1			1					1		1	1	
6.	Dresden 3	100	100	2078	23.7	986	11.3	3064	35.0	65.0	1				1		1	1					
7.	Humboldt Bay	100	100	1388	15.8	28	0.4	1416	16.2	83.8	1												
8.	LaCrosse	100	100	1161	13.3	501	5.7	1662	19.0	81.0		1								1		1	
9.	Millstone Point 1	100	100	1646	18.8	186	2.1	1832	20.9	79.1	1												
10.	Monticello	100	100	1916	21.9	284	3.2	2200	25.1	74.9	1										1		
11.	Nine Mile Point 1	100	100	2265	25.9	319	3.6	2584	29.5	70.5	1										1		
12.	Oyster Creek	100	100	1470	16.7	1129	12.7	2599	29.6	70.4	1							2			1		
13.	Peach Bottom 2	60	48.9	77	1.8	331	7.7	408	9.5	90.5					1								
14.	Pilgrim 1	100	100	0	0	5326	60.8	5326	60.8	39.2													
15.	Quad Cities 1	100	100	2832	32.3	564	6.4	3396	38.	61.3	1	1											1
16.	Quad Cities 2	100	100	485	5.5	1044	11.9	1529	17.4	82.6	1				1	1			1		1	1	
17.	Vermont Yankee	100	100	2068	23.6	199	2.3	2267	25.9	74.1	1		1	1									
				24,599	17.4	16,829	11.9	41,427	30.7	69.3	12	2	3	3	5	3	2	3	2	1	6	5	1

*The unit availability factors presented here differ from those reported with the individual reactor since the latter were calculated starting at different times.

Table 3-1b Summary of PWR Power Plant Outages During Commercial Operation in 1975

Plant Name	Operational % of year	Commercial Operation % of Year	Scheduled Outage During Commercial Operation hrs.	Scheduled Outage %	Forced Outage During Commercial Operation hrs.	Forced Outage %	Total Outage During Commercial Operation hrs.	Total Outage %	Unit Availability Commercial Operation %	Fuel Inspection or Replacement	Reactor Vessel Internals	Control Rods and Control Rod Systems	Main Turbine	Condenser and Feedwater System	Steam Generators	Auxiliary Systems (Electrical & Process)	Engineered Safety Systems	Hydraulic Restraints	Off-Gas Systems	Main Steam Systems	Valves and Valve Operators Except Main Steam Line Valves	Reactor Coolant or Recirculation Pumps	Miscellaneous
1. Arkansas 1	100	100	1566	17.9	529	6.0	2095	23.9	76.1*			2										1	
2. Calvert Cliffs 1	100	65.2	135	3.3	801	14.0	986	17.3	82.7†					1								1	
3. Connecticut Yankee	100	100	1152	13.2	63	0.7	1215	13.9	86.1	1													
4. Cook 1	90	34.8	340	11.1	75	2.5	415	13.6	86.4†													1	
5. Fort Calhoun	100	100	2714	31.0	139	1.6	2853	32.6	67.4	1		1										1	
6. Ginna	100	100	1727	19.7	324	3.7	2051	23.4	76.6*	1													
7. Indian Point 2	100	100	1560	17.8	652	7.4	2212	25.2	74.8				1		1							1	
8. Kewaunee	100	100	289	3.3	743	8.5	1032	11.8	88.2						1						1		
9. Maine Yankee	100	100	1492	17.1	265	3.0	1757	20.1	79.9	1											1		
10. Oconee 1	100	100	1914	21.8	171	2.0	2085	23.8	76.2	1												2	
11. Oconee 2	100	100	442	5.1	1914	21.8	2356	26.9	73.1								1				2	2	
12. Oconee 3	100	100	1110	12.6	891	10.2	2001	22.8	77.2					1		1						3	
13. Palisades	100	100	351	4.0	2757	31.5	3108	35.5	64.5	1		2		1									
14. Point Beach 1	100	100	1199	13.7	1262	14.4	2461	28.1	71.9	1			1		2								
15. Point Beach 2	100	100	326	3.7	209	2.4	535	6.1	93.9						1								
16. Prairie Island 1	100	100	811	9.2	391	4.5	1202	13.7	86.3						1		1					1	
17. Prairie Island 2	100	100	531	6.1	1195	13.6	1726	19.7	80.3				5									1	
18. Rancho Seco	100	70.7	0	–	4491	72.5	4491	72.5	27.5†				1										
19. Robinson 2	100	100	1292	14.7	1097	12.6	2389	27.3	72.7	1					1							3	
20. San Onofre 1	100	100	1086	12.4	14	0.2	1100	12.6	87.4	1											1		
21. Surry 1	100	100	2602	29.7	732	8.4	3334	38.1	61.9*	2			1		1	1							
22. Surry 2	100	100	1332	15.2	458	5.2	1790	20.4	79.6	1											1		
23. Three Mile Island	100	100	637	7.3	922	10.5	1559	17.8	82.2			1				1					1	2	
24. Turkey Point 3	100	100	1597	18.2	196	2.3	1809[1]	20.6[1]	79.4	1								1			1		
25. Turkey Point 4	100	100	2559	29.2	23	0.3	2582	29.5	70.5	1					2								
26. Yankee Rowe	100	100	1610	18.4	27	0.3	1637	18.7	81.3*	1													
27. Zion 1	100	100	1313	15.0	1276	14.6	2632[2]	30.0[2]	70.0					1	1						1	1	
28. Zion 2	100	100	31	0.3	2335	26.7	2436[3]	27.8[3]	72.2					4				1			2		
			31,768	13.6	23,952	10.2	55,849	23.9	76.1	15	0	6	9	8	11	3	2	2	0	0	11	20	

[1]Includes 16 hours not identified as either Forced or Scheduled.

[2]Includes 43 hours not identified as either Forced or Scheduled.

[3]Includes 70 hours not identified as either Forced or Scheduled.

*Availability consistent with outage data but somewhat lower than previously published data.

†Calculated using commercial operation only.

3.1.2 Types of Outages

Identification of types and causes of major outages for each plant in commercial operation is contained in Tables 3-1a, 3-1b. To provide an overview of plant outages, these tables list plant availability, percent of scheduled, forced and total outages with duration, and the major systems and components causing outages lasting longer than five days. As shown in these tables, twenty-one of the plants experienced shutdowns for replacement of fuel, and fourteen plants had major outages caused by the main turbine generator. Pumps and valves continue to be a major source of trouble accounting for 14 and 11, respectively, of the major outages. Also of importance were 12 steam generator outages (from the 23 PWR plants). Other causes of more than five major outages include hydraulic restraints - 9 and the condenser and feedwater system - 8.

Table 3-2 summarizes the outages by type (i.e., scheduled or forced) and indicates the relative impact on plant operations. During 1974 (and considering all of the reported outage data) the average number of forced outages for each nuclear plant was 13, with each outage averaging 108 hours in duration. The average number of scheduled outages was 5 per plant, with each averaging 319 hours in duration. Excluding time for additional work, which was conducted during the same shutdown, the average duration of refueling outages was 1694 hours. However, it is not always possible to determine if the refueling outage was extended for other work. The times charged as refueling ranged from 1205 hours to 3000 hours.

3.1.3 Proximate Cause of Plant Outages

Outage events and proximate causes are summarized in Table 3-3. Outage cause was selected by the NRC staff to be in one of seven categories listed in order of importance (1) equipment failures (forced), (2) refueling (scheduled), (3) maintenance or test (both forced and scheduled), (4) regulatory restrictions (both forced and scheduled), (5) operator error (forced), (6) training and licensing (scheduled), and (7) other. The operator error category includes errors by plant personnel which caused a forced outage.

Equipment failure was the single largest cause of nuclear plant outages, accounting for 40% of the total outage time. Refueling outages took 30% of the total outage time; operational errors accounted for 89 outages, but only amounted to 2% of the total outage time. Scheduled maintenance accounted for 22% of the total outage time; regulatory restrictions accounted for 6% and training and licensing examinations amounted to less than 1% of the outage time. Nine outages were classified as "other" which consisted of tornado and storm damage to lines and equipment and other miscellaneous internal causes.

3.1.4 Systems and Components Associated with Plant Outages

Graphic representation of plant outages is shown in Tables 3-4 and 3-5. These tables classify outages by type, and identify system, component, plant, and

Table 3-2. Summary of Nuclear Plant Outages by Type for 1974

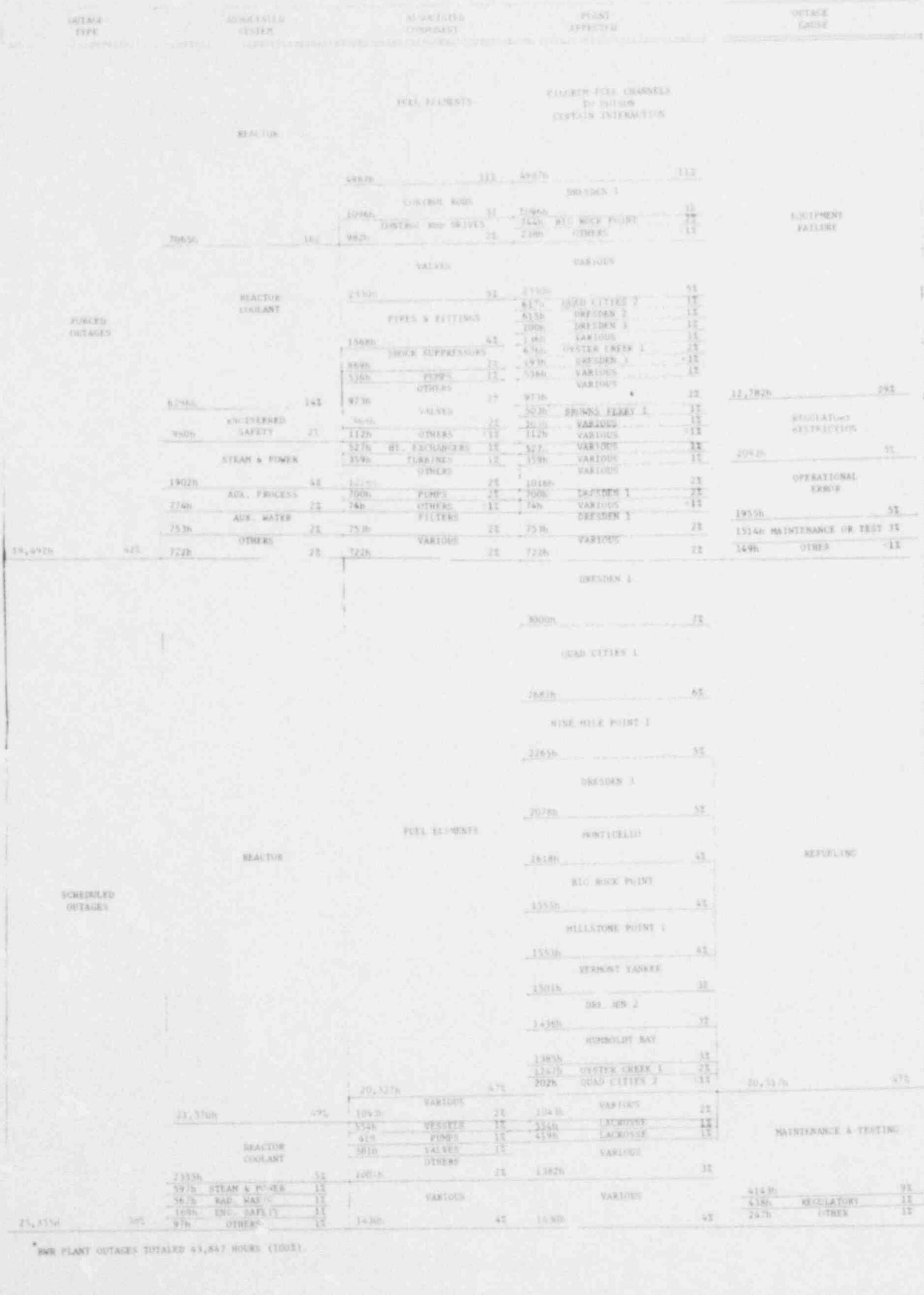
Plant Type (number)	Forced Outages		Scheduled Outages		Total Outages	
	Number of Events	Outages Duration (hours)	Number of Events	Outage Duration (hours)	Number of Events	Outage Duration (hours)
BWR Plants (17)	191	18,493	72	25,354	263	43,847
Average Per BWR Plant	11	1,088	4	1,491	15	2,579
PWR Plants (23)	331	34,884	109	35,051	440	69,935*
Average Per PWR Plant	14	1,517	5	1,524	19	3,041
All Plants (40)	522	53,377	181	60,405	703	113,782*
Overall Average Per Plant	13	1,334	5	1,510	18	2,845
Average Outage Duration Per Event	-	102	-	334	-	162

*There are 1981 additional shutdown hours in 4 shutdowns for Oconee No. 2 that are not included in this Table. Detailed data are not available.

Table 3-3. Proximate Cause of Outages During 1974

EVENTS		FORCED OUTAGE					SCHEDULED OUTAGE					TOTALS
		Equipment Failure	Maintenance or Test	Regulatory Restrictions	Operator Error	Other	Maintenance or Test	Refueling	Regulatory Restrictions	Training & Licensing	Other	
BWR	No. of Events	141	17	2	28	4	48	12	3	7	1	263
BWR	Hours of Outage	12,783	1514	2092	1955	149	5696	20,527	438	238	9	43,847
PWR	No. of Events	236	27	1	61	6	88 1/2	9 1/2	2	9	0	440
PWR	Hours of Outage	32,915	1057	21	834	57	19,110	14,165	1465	311	0	69,935
ALL	No. of Events	377	44	3	89	10	136 1/2	21 1/2	5	16	1	703
ALL	% of Total	54	06	01	13	01	19	03	01	02	>0	100
PLANTS	Total Outage Hours	45,698	2571	2113	2789	206	24,806	34,692	1903	549	9	113,789
PLANTS	% of Total	40	02	02	02	<01	22	26	02	<01	>0	100

TABLE 7-4
BOILING WATER PLANT OUTAGES,* 1976



*BWR PLANT OUTAGES TOTALED 43,847 HOURS (100%).

TABLE 3-5
PRESSURIZED WATER REACTOR PLANT OUTAGES,* 1974

OUTAGE TYPE	ASSOCIATED SYSTEMS	ASSOCIATED COMPONENTS	PLANT AFFECTED
FORCED OUTAGES	STEAM and POWER	TURBINES	SURRY 1 2787h 4%
			GINNA 2737h 4%
			PRAIRIE ISLAND 1 2294h 3%
			CONN. YANKEE 660h 1%
		9828h 14%	OTHERS 1350h 2%
		HEAT EXCHANGERS	PALISADES 7421h 10%
		8459h 12%	OTHERS 1038h 2%
		GENERATORS	ZION 2 3090h 4%
		3463h 5%	OTHERS 373h 1%
		PIPES and FITTINGS	INDIAN POINT 2 1843h 3%
		2870h 4%	OTHERS 1027h 1%
	26,280h 37%	OTHERS 1660h 2%	VARIOUS 1660h 2%
	REACTOR COOLANT	PUMPS	OCONEE 2 2907h 4%
		3393h 5%	OTHERS 486h 1%
		VALVES 1101h 2%	VARIOUS 1101h 2%
		762h PIPES & FITTINGS 1%	VARIOUS 762h 1%
	6083h 9%	827h OTHERS 1%	VARIOUS 827h 1%
	1078h REACTOR 1%	1078h VARIOUS 1%	VARIOUS 1078h 1%
34,884 49%	[illegible] 1443h 2%	VARIOUS 1443h 2%	VARIOUS 1443h 2%
SCHEDULED OUTAGES	REACTOR	FUEL ELEMENTS	YANKEE ROWE 2553h 4%
			MAINE YANKEE 2513h 3%
			OCONEE 1 1752h 2%
			TURKEY POINT 3 1704h 2%
			SURRY 1 1637h 2%
			POINT BEACH 2 1566h 2%
			ROBINSON 2 1235h 2%
		14,165h 20%	POINT BEACH 1 1205h 2%
	14,551h 20%	395h VARIOUS <<1%	395h VARIOUS 1%
	STEAM and POWER	HEAT EXCHANGERS	893h PRAIRIE ISLAND 1%
			711h KEWAUNEE 1%
		3944h 6%	OTHERS 2340h 4%
		GENERATORS	ZION 1 2492h 3%
		3262h 4%	OTHERS 770h 1%
		TURBINES	PRAIRIE ISLAND 1180h 2%
		1991h 3%	OTHERS 811h 1%
		862h VALVES 1%	862h VARIOUS 1%
	10,708h 15%	649h VARIOUS 1%	649h VARIOUS 1%
	REACTOR COOLANT	PUMPS	SURRY 1 1784h 3%
			SURRY 2 1557h 2%
		4344h 6%	OTHERS 1003h 1%
		SHOCK SUPPRESSORS	783h TURKEY POINT 4 1%
		2168h 3%	OTHERS 1385h 2%
		760h VALVES 1%	760h VARIOUS 1%
	7927h 11%	655h VARIOUS 1%	655h VARIOUS 1%
	ENGINEERED SAFETY 1514h 2%	VARIOUS 1514h 2%	INDIAN POINT 1 + OTHERS 1514h 2%
35,051h 49%	351h UNKNOWN & OTHERS <1%	351h UNKNOWN & VARIOUS <1%	351h VARIOUS <1%
UNKNOWN 1981h 2%	UNKNOWN 1981h 2%	UNKNOWN 1981h 2%	OCONEE 2 19[illegible] 2%

OUTAGE CAUSE		
EQUIPMENT FAILURE	32,907h	46%
MAINTENANCE	1057h	1%
OPER. ERROR	802h	1%
OTHER	118h	<1%
MAINTENANCE or TEST	19,129h	27%
REFUELING	14,165h	20%
REGULATORY	1465h	2%
TNG. & EXAMS	311h	<<1%
UNKNOWN	1981h	2%

*PWR PLANT OUTAGES TOTALED 71,916 HOURS (100%).

cause. Outage duration in hours and the percent of the total outage time is listed for major groupings. The size of each box in Tables 3-4 and 3-5 is proportional to the hours involved to the nearest 1%. The system and component classifications used in this report are defined in Appendix A; because of the fundamental differences between BWRs and PWRs, they are discussed separately.

3.1.5 Boiling Water Reactors

Forced Outages

BWR forced outages accounted for 42% of the total outage time, i.e., 18,492 of 43,847 hours. As indicated in Table 3-4, the reactor system was the dominant system associated with forced outages; however, most of the 16% of the total outage attributable to this cause was due to a unique problem at one plant, Pilgrim (fuel channel to poison curtain interaction). The reactor coolant system with 14% was also a major system involved in forced outages, but no one plant accounted for more than 995 hrs, i.e., ~2%.

The steam and power system with 4% outage time accounted for 1902 hours of the total 43,847 hours with no single event dominating. The auxiliary process and auxiliary water systems each accounted for ~2% of the total outage time.

The block designated as "others" accounted for 722 hours. This was comprised of: electric power - 214 hours; radwaste systems - 195 hours; instrumentation and controls - 177 hours; and other auxiliary systems - 136 hours.

Scheduled Outages

Scheduled outages in BWRs totaled 25,355 hrs or 58% of the total outage time. Refuelings accounted for most, i.e., 47%. Other activities, including major maintenance, often were carried out concurrently with refueling. However, in general, it was not feasible to prorate the outage time to other than the reactor system and fuel elements.

The reactor coolant system was involved in 5% (2355 hours) of the total BWR plant outage time. Activities associated with this system included both maintenance and testing.

The steam and power, radioactive waste, and engineered safety features each accounted for 1% of the outage time.

The time indicated as others represents 97 hours; of this subtotal, the following distribution occurred: electric power - 46 hours; auxiliary water - 33 hours; and instrumentation and controls - 18 hours.

3.1.6 Pressurized Water Reactors

Forced Outages

Forced outages in PWRs accounted for 49% of the total outage time, i.e., 34,884 of 71,916 hrs. This total outage time includes 1981 hrs or 2% of the total outage time at Oconee 1, which was undesignated and is listed separately herein. The largest portion of the forced outage time was the result of problems with the steam and power system. This includes the turbines which accounted for 9828 hours (14%), heat exchangers which accounted for 8459 hours (12%), electrical generators (5%), pipes and fittings (4%), and various others another 2%. The turbine outages were largely attributable to problems at three plants (Surry 2, Ginna, and Prairie Island 1). The heat exchanger outage was principally due to Palisades (10%), while Zion 2 was principally responsible for the electrical generator outage time.

Problems with the reactor coolant system accounted for 6083 hours of outage time, of which pumps and valves were the dominant components involved. The reactor system accounted for 1078 hours, of which control rod drive mechanisms required 906 hours for repair.

The 1443 hours listed as "others" include the following: undetermined - 507 hours; electric power - 457 hours; instrumentation and controls - 228 hours; auxiliary process - 191 hours; engineered safety features - 39 hours; and radiation protection - 21 hours.

Scheduled Outages

Scheduled outages in PWRs totaled 35,051 hrs or 49%. The reactor system accounted for 14,551 hours of outage time, of which 14,165 hours was for refueling. Problems with the steam and power system accounted for 10,708 hours due primarily to problems with heat exchangers, generators, and turbines.

The reactor coolant systems required 7927 hours, of which 4344 hours were due to problems with pumps and 2168 hours were for inspection and repair of shock suppressors. The engineered safety features required 1514 hours for maintenance; 123 hours were undeterminable, and 228 hours were accounted for by other systems.

Undetermined

There were 1981 hours of outage time at Oconee 2 for which no information was available.

3.1.7 Observation on BWR and PWR Outages

Forced Outages

Seventeen BWR plants experienced 18,492 hours of forced outage; an overall average of 1088 hours per plant. Twenty-three PWR plants experienced 34,884 hours of forced outage; an overall average of 1517 hours per plant. (The PWR plants also had 1981 hours of outage for which a determination of forced or scheduled could not be made.)

Additional insight as to the relative outages in BWRs and PWRs may be obtained by using the data in Tables 3-4 and 3-5, and comparing the outage percentage and the average number of hours attributable to various causes. This is done below where a listed component for either reactor type contributed more than 1% of the total outage time.

Outage Type	System	Component	PWR %	PWR Avg. hrs.	BWR %	BWR Avg. hrs.
Forced	Reactor	Fuel Elements	0		11	293
		Control Rod and Drive	<1	-	5	122
	Reactor Coolant	Pumps	5	148	1	33
		Valves	2	48	5	144
		Pipes and Fittings	2	33	4	92
		Shock Suppressors	<1	-	2	51
	Steam & Power	Turbines	14	427	1	21
		Heat Exchangers	12	368	1	31
		Generators	5	151	-	-
		Pipes and Fittings	4	125	-	-
	Auxiliary Systems	Pumps	<1	-	2	41
		Filters	<1	-	2	46

The dominant component contributing to BWR forced outage time was fuel elements - 4987 hours - because of the problem at Pilgrim with fuel channel to poison curtain interaction. At PWRs, the dominant component was the steam turbines accounting for 9828 hours or 427 hours per plant. This was followed closely by heat exchangers which averaged 368 hours per plant. Forced outage for both of these components was significantly lower at BWRs, but they do not employ steam generators which was the principal component in the heat exchanger classification. Both reactor types had comparable forced outage for total of pumps and valves, but BWRs had more outage for control rods (and drives), and auxiliary systems.

Scheduled Outages

The seventeen BWRs had 25,355 hours of scheduled outage time for an average of 1491 hours per plant. The twenty-three PWRs accumulated 35,051 hours for an average of 1524 hours per plant. The scheduled outages in the two types of reactors are compared below on the basis of percent outage and the average number of hours attributable to various causes (>1%).

Outage Type	System	Component	PWR %	PWR Avg. hrs.	BWR %	BWR Avg. hrs.
Scheduled	Reactor	Fuel Elements	20	616	47	1207
	Reactor Coolant	Pumps	6	189	1	25
		Valves	1	33	1	22
		Shock Suppressors	3	94	<1	-
	Steam & Power	Heat Exchangers	6	171	<1	-
		Generators	4	142	0	-
		Turbines	3	86	<1	-

Refueling at BWRs accounted for 20,527 hours (average 1207) and at PWRs 14,165 hours (average 616), reflecting the greater refueling undertaken at BWRs during the past year. However, each PWR plant also experienced an average of 831 hours for maintenance or test vs the BWR average of 243 hours. The scheduled outages for maintenance and testing in PWRs involved primarily pumps, shock suppressors, heat exchangers, generators and turbines.

3.1.8 Summary

During 1974 the 17 operating BWRs experienced an average of 2579 hours of outage compared to an average of 3127 hours for the 23 operating PWRs. This average total outage divided between forced and scheduled outages for both reactor types; 1088 and 1492 hours respectively for BWRs and 1517 and 1524 hours respectively (plus another 86 hours for the undesignated outage at Oconee 1) for PWRs.

The outage experience in the two reactor types is not sufficiently extensive and/or similar to permit meaningful, comparative analysis. The difference in the forced outage time could be attributed to the steam generators (which BWRs do not have), but there is little parallel in the remaining forced outage data except perhaps for the persistence of problems dealing with such things as pumps, valves, pipes and fittings.

The scheduled outage in BWRs was almost entirely attributable to refueling activities whereas in PWRs it was more equally divided between refueling and scheduled maintenance (or test). While refueling activities in BWRs average 1207 hours vs 616 hours in PWRs, these numbers reflect a higher frequency of refueling in BWRs (12 out of 17) than in PWRs (9 out of 23). However, a complete refueling operation averaged 1632 hours for 10 BWRs, and 17̄1 hours for 8 PWRs. (For both reactor types refueling outages occurred that were not considered in the averages, because they extended over other years.) Moreover the extreme (high and low) times required for refueling differed by a factor >2, which suggests both that some scheduled maintenance may be responsible for the longer times, and that the average time for refueling may be significantly reduced.

3.2 Plant Outages - 1975

Review of the plant outages or shutdowns that occurred during 1975 provides a means of assessing the nature, number and extent of the operating problems experienced at nuclear power plants, as well as the principal systems and components involved. Most of the data were obtained either from the data reported by the licensee for the NRC's monthly "Operating Units Status Report," although in some instances it was necessary to check information in the licensees periodic operating reports.

In some instances omissions or inconsistencies in the reported data necessitated checking with the licensee. In a few cases, outage type was classified differently than reported by the licensee. Where appropriate and sufficient information was available, major outages were subdivided to reflect more accurately the true nature of the work performed during the outage. The forced extension of a scheduled outage was generally reclassified by the NRC staff as a forced outage. In a few cases, work that had been scheduled for later in the year was rescheduled so it could be performed during an unexpected forced outage. These cases were classified as scheduled outages. Refueling of the reactor is classified as a scheduled outage.

Data sheets for all the plants considered are contained in Appendix B. In reviewing the outage data, it should be noted that there are significant differences in nuclear plant designs, even between plants of a given type. Therefore, care should be used in interpreting the data.

3.2.1 Plant Outage Statistics

There were 854 outages requiring 121,903 hours of plant downtime reported by the 51 nuclear power plants which were in commercial operation during 1975 as listed in Tables 3-6a and 3-6b. It should be noted that eight of the listed plants were not in commercial operation until after the first of the year. The calculated percent outages and unit availability factors listed consider only the period of time after the plants were in commercial operation. However, in subsequent outage analyses herein, all outages (including those reported as occurring before commercial operation began, were reviewed to ascertain the type of outage (forced or scheduled), cause, method of shutdown, duration, and the plant system and component primarily associated with the outage.

Table 3-6a

Summary of PWR Power Plant Outages During Commercial Operation in 1975

Plant Name	Operational % of Year	Commercial Operation % of Year	Scheduled Outage During Commercial Operation hrs.	%	Forced Outage During Commercial Operation hrs.	%	Total Outage During Commercial Operation hrs.	%	Unit Availability Commercial Operation %	Fuel Inspection or Replacement	Reactor Vessel Internals	Control Rods and Control Rod Systems	Main Turbine	Condenser and Feedwater System	Steam Generators	Auxiliary Systems (Electrical & Process)	Engineered Safety Systems	Hydraulic Restraints	Off Gas Systems	Main Steam Systems	Valves and Valve Operators Except Main Steam Line Valves	Reactor Coolant or Recirculation Pumps	Miscellaneous
1. Arkansas I	100	100	1566	17.9	529	6.0	2095	23.9	76.1*			2										1	
2. Calvert Cliffs I	100	65.2	185	3.3	801	14.0	986	17.3	82.7†					1								1	
3. Connecticut Yankee	100	100	1152	13.2	63	0.7	1215	13.9	86.1	1													
4. Cook 1	90	34.8	340	11.1	75	2.5	415	13.6	86.4†													1	
5. Fort Calhoun	100	100	2714	31.0	139	1.6	2853	32.6	67.4	1		1										1	
6. Ginna	100	100	1727	19.7	324	3.7	2051	23.4	76.6*	1													
7. Indian Point 2	100	100	1560	17.8	652	7.4	2212	25.2	74.8				1		1							1	
8. Kewaunee	100	100	289	3.3	743	8.5	1032	11.8	88.2						1						1		
9. Maine Yankee	100	100	1492	17.1	265	3.0	1757	20.1	79.9	1											1		
10. Oconee 1	100	100	1914	21.8	171	2.0	2085	23.8	76.2	1												2	
11. Oconee 2	100	100	442	5.1	1914	21.8	2356	26.9	73.1								1				2	2	
12. Oconee 3	100	100	1110	12.6	891	10.2	2001	22.8	77.2					1		1						3	
13. Palisades	100	100	351	4.0	2757	31.5	3108	35.5	64.5	1		2		1									
14. Point Beach 1	100	100	1199	13.7	1262	14.4	2461	28.1	71.9	1			1		2								
15. Point Beach 2	100	100	326	3.7	209	2.4	535	6.1	93.9						1								
16. Prairie Island 1	100	100	811	9.2	391	4.5	1202	13.7	86.3			-			1		1					1	
17. Prairie Island 2	100	100	531	6.1	1195	13.6	1726	19.7	80.3				5									1	
18. Rancho Seco	100	70.7	0	-	4491	72.5	4491	72.5	27.5†				1										
19. Robinson 2	100	100	1292	14.7	1097	12.6	2389	27.3	72.7	1					1							3	
20. San Onofre 1	100	100	1086	12.4	14	0.2	1100	12.6	87.4	1											1		
21. Surry 1	100	100	2602	29.7	732	8.4	3334	38.1	61.9*	2			1		1	1							
22. Surry 2	100	100	1332	15.2	458	5.2	1790	20.4	79.6	1											1		
23. Three Mile Island	100	100	637	7.3	922	10.5	1559	17.8	82.2			1				1					1	2	
24. Turkey Point 3	100	100	1597	18.2	196	2.3	1809[1]	20.6[1]	79.4	1								1			1		
25. Turkey Point 4	100	100	2559	29.2	23	0.3	2582	29.5	70.5	1					2								
26. Yankee Rowe	100	100	1610	18.4	27	0.3	1637	18.7	81.3*	1													
27. Zion 1	100	100	1313	15.0	1276	14.6	2632[2]	30.0[2]	70.0					1	1						1	1	
28. Zion 2	100	100	31	0.3	233[illegible]	26.7	2436[3]	27.8[3]	72.2					4				1			2		
			31,768	13.6	23,952	10.2	55,849	23.9	76.1	15	0	6	9	3	11	3	2	2	0	0	11	20	

[1]Includes 16 hours not identified as either Forced or Scheduled.

[2]Includes 43 hours not identified as either Forced or Scheduled.

[3]Includes 70 hours not identified as either Forced or Scheduled.

*Availability consistent with outage data but somewhat lower than previously published data.

†Calculated using commercial operation only.

Table 3-6b

Summary of BWR Power Plant Outages During Commercial Operation in 1975

Plant Name	Operational % of Year	Commercial Operation % of Year	Scheduled Outage During Commercial Operation hrs.	Scheduled Outage During Commercial Operation %	Forced Outage During Commercial Operation hrs.	Forced Outage During Commercial Operation %	Total Outage During Commercial Operation hrs.	Total Outage During Commercial Operation %	Unit Availability Commercial Operation %	Fuel Inspection or Replacement	Reactor Vessel Internals	Control Rods and Control Rod Systems	Main Turbine	Condenser and Feedwater System	Steam Generators	Auxiliary Systems (Electrical & Process)	Engineered Safety Systems	Hydraulic Restraints	Off-Gas Systems	Main Steam Systems	Valves and Valve Operators Except Main Steam Line Valves	Reactor Coolant or Recirculation Pumps	Miscellaneous
1. Arnold, Duane	100	91.8	1080	13.4	640	8.0	1720	21.4	78.6[†]	1													
2. Big Rock Point	100	100	50	0.6	3472	39.6	3522	40.2	59.8								1						
3. Browns Ferry 1	100	100	180	2.1	7045	80.4	7225	82.5	17.5							1	1				1		
4. Browns Ferry 2	100	83.8	0	0	6828	93.0	6828	93.0	7.0[†]							1							
5. Brunswick 2	58.3	16.2	0	0	105	7.4	105	7.4	93.2[†]														
6. Cooper	100	100	1010	11.5	426	4.9	1436	16.4	83.6				1			1	1						
7. Dresden 1	100	100	2687	30.7	1064	12.1	3751	42.8	57.2	1							1					1	
8. Dresden 2	100	100	3591	41.0	339	3.9	3930	44.9	55.1	1							1						
9. Dresden 3	100	100	3530	40.3	692	7.9	4252[1]	48.5[1]	51.5	1				1		1							
10. Fitzpatrick	50.4	42.7	439	11.7	659	17.6	1098	29.3	70.7[†]				1	3						2			
11. Hatch 1	100	0	1526[2]	17.5[2]	1062[2]	12.2[2]	2588[2]	29.7[2]	70.3[2]														
12. Humboldt Bay	100	100	126d	14.5	145	1.6	1413	16.1	83.9	1												1	
13. La Crosse	100	100	2384	27.2	318	3.6	2702	30.8	~9.2*	1												1	
14. Millstone Point 1	100	100	613	7.1	1516	17.3	2135	24.4	75.6						NOT APPLICABLE	1	1			1		1	
15. Monticello	100	100	2401	27.4	35	0.4	2436	27.8	72.2	2													
16. Nine Mile Point	100	100	2200	25.1	322	3.7	2522	28.8	71.2	1							1						
17. Oyster Creek	100	100	1394	15.9	944	10.8	2338	26.7	73.3	2				2								1	1
18. Peach Bottom 2	100	100	1218	13.9	905	10.3	2123	24.2	75.8		1					1	2					1	
19. Peach Bottom 3	100	100	397	4.5	831	9.5	1228	14.0	86.0					1								3	
20. Pilgrim	100	100	617	7.0	1900	21.7	2517	28.7	71.3					2			1			1	2	1	
21. Quad Cities 1	100	100	242	2.7	1067	12.2	1309	14.9	85.1													1	
22. Quad Cities 2	100	100	3691	42.1	539	6.2	4230	48.3	51.7	2				1							1		
23. Vermont Yankee	100	100	715	8.2	358	4.0	1073	12.2	87.8		1					1							
			29,713	16.7	30,150	16.9	59,893[1]	33.6[1]	66.4	13	2	0	2	10	-	7	12	0	0	4	4	11	1

*Availability consistent with outage data but somewhat lower than previously published data.

[†]Calculated using commercial operation only.

[1]These numbers do not include a discrepancy of 30 hrs additional hours outage which could not be identified.

[2]The Hatch reactor was not placed in commercial operation until after the first of January 1976. Consequently, this outage data is not included in the totals of this table.

For the 51 plants thus reviewed, the average total outage time during commercial operation was 28.1% of the year, the average forced outage time was 13.1%, and the average scheduled outage time was 14.9%. (The difference of 0.1% between the total and the sum of the scheduled and forced outages is due to 159 hours of unclassified outage at four plants.) The average total unit availability for the 51 plants considering only the period when they were in commercial operation was 71.9%. The average performance of the PWRs was somewhat better than the average performance of the BWRs by all of the above measures. However, the BWR statistics were distorted by Browns Ferry 1 and 2, both of which were out of service for the remainder of the year following the fire of March 22, 1975. See Section 4.3 for details of the fire.

3.2.2 Types of Outages

Identification of types and causes of major outages for each plant in commercial operation is contained in Tables 3-6a, 3-6b. To provide an overview of plant outages, these tables list plant availability, percent of scheduled, forced and total outages with duration, and the major systems and components causing outages lasting longer than five days. As shown in these tables, these plants experienced 28 major outages for refueling with four plants having more than one such outage during the year. There were 35 major outages attributable to reactor coolant or recirculating pumps and they affected half (i.e., 25) of the operating plants. Problems in the condenser and feedwater system accounted for 18 major outages, while valves accounted for 15. Also of note were 11 turbine outages (mostly in PWRs), 11 steam generator outages (all in PWRs) and 10 each auxiliary system and engineered safety system outages (mostly in BWRs).

Table 3-7 summarizes the outages by type (i.e., scheduled or forced) and indicates the relative impact on plant operations. During 1975 (and considering all of the reported outage data) the average number of forced outages for each nuclear plant was 13, with each outage averaging 90 hours in duration. The average number of scheduled outages was four per plant, with each averaging 301 hours in duration. Excluding time for additional work, which was conducted during the same shutdown, the average duration of refueling outages was 1451 hours. (This was obtained by considering only those refueling shutdowns which were started and completed within the calendar year.) However, it is not always possible to determine if the refueling outage was extended for other work. The times charged as refueling ranged from 938 hours to 2508 hours.

3.2.3 Proximate Cause of Plant Outages

Outage events and proximate causes are summarized in Table 3-8. Outage cause was selected by the NRC staff to be in one of eight categories as follows: [1] refueling (scheduled), [2] equipment failure (forced), [3] maintenance or test (primarily scheduled), [4] operational error (forced), [5] regulatory restriction (forced and scheduled), [6] administrative (forced and scheduled), [7] training and licensing (scheduled), and [8] other. The operational error category includes any plant personnel errors which caused a forced outage.

Scheduled refuelings required the most outage time of all causes with 37,735 hours (31%). This was followed closely by equipment failures (forced) with 35,694 hours or 29% of all outage time. Scheduled maintenance or testing consumed 22,951 hours - 19%.

Table 3-7. Summary of Nuclear Plant Outages by Type for 1975

Plant Type (number)	Forced Outages		Scheduled Outages		Total Outages	
	Number of Events	Outage Duration (hours)	Number of Events	Outage Duration (hours)	Number of Events	Outage Duration (hours)
BWR Plants (23)	251	33,487	81	31,490	332	64,977
Average Per BWR Plant	11	1,456	4	1,369	14	2,825
PWR Plants (28)	390	24,197	132	32,729	522	56,926
Average Per PWR Plant	14	864	5	1,169	19	2,033
All Plants (51)	641	57,684	213	64,219	854	121,903
Overall Average Per Plant	13	1,131	4	1,259	17	2,390
Average Outage Duration Per Event	-	90	-	301	-	143

Table 3-8

Proximate Cause of Outages During 1975

EVENTS		FORCED OUTAGE						SCHEDULED OUTAGE						TOTALS
		Equipment Failure	Maintenance or Test	Regulatory Restrictions	Administrative	Operational Error	Other	Maintenance or Test	Re-fueling	Regulatory Restrictions	Training & Licensing	Administrative	Other	
BWR	No. of Events	192	2	5	2	46	4	57	11	10	1	1	1	332
BWR	Hrs of Outage	13,680	31	1308	183	14,670	3,615	9,653	19,210	1,627	20	31	949	64,977
PWR	No. of Events	308	1	0	1	68	12	110	15	0	3	3	1	522
PWR	Hrs of Outage	22,014	14	0	28	1,487	654	13,298	18,525	0	95	51	760	56,926
ALL PLANTS	No. of Events	500	3	5	3	114	16	167	26	10	4	4	2	854
ALL PLANTS	% of Total	59	<1	1	<1	13	2	20	3	1	<1	<1	<1	100
ALL PLANTS	Total Outage Hours	35,694	45	1308	211	16,157	4,269	22,951	37,735	1,627	115	82	1,709	121,903
ALL PLANTS	% of Total	29	<1	1	<1	13	4	19	31	1	<1	<1	1	100

There was a total of 114 operational errors - 46 at BWRs and 68 at PWRs resulting in 16,157 hours of forced outage time. However, 13,656 hours was due to one incident. The Browns Ferry fire which disabled two units for 6828 hours each was attributed to an operational error because the fire was started by a candle held by a technician checking containment penetration leakage. The other 112 plant outages attributed to operational errors accounted for only 2501 hours which is an average of 22 hours per operational error.

There were 4269 hours of forced outage classified as other. The basic causes were varied but included such things as power distribution system disturbance (5 outages), lightning (4), windstorm (1), condenser loaded with fish (1), and seaweed on intake (1). Each of the other causes of outages (regulatory restriction, administrative training and licensing) resulted in 1% or less of the total outage time.

3.2.4 Systems and Components Associated with Plant Outages

Graphic representation of plant outages is shown in Tables 3-9 and 3-10. These tables classify outages by type, and identify system, component, plant, and cause. Outage duration in hours and the percent of the total outage time is listed for major groupings. The size of each box is proportional to the hours involved to the nearest 1%. The system and component classifications used in this report are listed in Appendix B.

Because of the fundamental differences between BWRs and PWRs, they are discussed separately.

3.2.5 Boiling Water Reactors

Forced Outages

BWR forced outages accounted for 52% of the total outage time, i.e., 33,487 of 64,977 hours. As indicated in Table 3-9, the electric power system was the dominant system associated with forced outages. There was a fire at the Browns Ferry plant on March 22, 1975, which destroyed electrical cables and resulted in both Units 1 and 2 being shut down for the remainder of the year. Each unit accumulated 6828 hours of outage time. Thus, 13,656 hours of outage time was due to one incident. This represents 21% of the total outage time accumulated by all BWR plants.

The reactor coolant system accounted for 9112 hours. The major components involved were pipes and fittings - 3769 hours, pumps - 2991 hours, and valves - 1700 hours.

There was 5285 hours required for engineered safety feature problems. The dominant component was instruments and controls with 3450 hours. Big Rock Point accounted for 3421 hours after it was found that design and QA deficiencies existed in the instrumentation for the post incident cooling system.

Table 3-9

BOILING WATER PLANT OUTAGES,* 1975

OUTAGE TYPE	ASSOCIATED SYSTEM	ASSOCIATED COMPONENTS	PLANTS AFFECTED	OUTAGE CAUSE
FORCED OUTAGES 33487h 52%	ELECTRIC POWER 15508h 24%	ELECTRICAL CONDUCTORS 13746h 21%	BROWNS FERRY 1 CABLE FIRE 6828h 10%	OPERATIONAL ERROR† 14670h 23%
			BROWNS FERRY 2 CABLE FIRE 6828h 10%	
			VARIOUS 90h <1%	
		TRANSFORMERS 1423h 2%	MILLSTONE PT 1 961h 1%	
			VARIOUS 462h 1%	
		OTHERS 339h 1%	VARIOUS 339h 1%	
	REACTOR COOLANT 8460h 13%	PIPES, FITTINGS 3117h 5%	QUAD CITIES 1 472h 1%	EQUIPMENT FAILURE [illegible]680h 21%
			PILGRIM 1 751h 1%	
			DRESDEN 1 408h 1%	
			VARIOUS 986h 2%	
		PUMPS 2991h 5%	BRUNSWICK 2 1496h 2%	
			PILGRIM 1 370h 1%	
			VARIOUS 1125h 2%	
		VALVES 1700h 2%	VARIOUS 2352h 3%	
		OTHERS 653h 1%		
	ENGINEERED SAFETY 5937h 9%	I & C 3450h 5%	BIG ROCK POINT - 3421, QA DEFICIENCY FITZPATRICK-29h 3450h 5%	
		PIPES, FITTINGS 1257h 2%	VARIOUS 1257h 2%	
		VALVES 1141h 2%	VARIOUS 1141h 2%	
		OTHERS 89h 1%	VARIOUS 89h <1%	
	STEAM & POWER 2769h 4%	HEAT EXCHANGERS 920h 1%	VARIOUS 2769h 4%	OTHER (DESIGN & QA DEFICIENCIES FOUND AT BIG ROCK POINT - 3421 HRS) 3615h 6%
		PIPES, FITTINGS 406h 1%		
		OTHERS 1443h 2%		
	I & C 438h 1%	VARIOUS 438h 1%	VARIOUS 438h 1%	1390h REG. RESTRICT. 2%
	OTHERS 375h 1%	VARIOUS 375h 1%	VARIOUS 375h 1%	214h ADM. & TESTING <1%
SCHEDULED OUTAGES 31490h 48%	REACTOR 21344h 33%	FUEL ELEMENTS 21109h 32%	QUAD CITIES 2 3595h 5%	REFUELING 19215h 30%
			DRESDEN 2 3342h 5%	
			DRESDEN 1 2508h 4%	
			MONTICELLO 2363h 4%	
			LaCROSSE 2153h 3%	
			NINE MILE POINT 1927h 3%	
			DRESDEN 3 1823h 3%	
			OYSTER CREEK 1362h 2%	
			ARNOLD 1062h 2%	
			HUMBOLDT BAY 938h 1%	
		CONTROL ROD DRIVES 235h 1%	VARIOUS 237h 1%	
	REACTOR COOLANT 4662h 7%	PIPES, FITTINGS 2488h 4%	DRESDEN 3 INSTALL NEW FW SPARGER 1701h 2%	MAINTENANCE OR TESTING 9619h 15%
			VARIOUS 787h 1%	
		ELEC. CONDUCTORS 858h 1%	COOPER 858h 1%	
		VALVES 643h 1%	VARIOUS 1316h 2%	
		VARIOUS 673h 1%		
	I & C 2374h 4%	I & C (REL'D) 2283h 4%	HATCH 1 949h 1%	
			PEACH BOTTOM 2 (CLAMP'D VALVE) 786h 1%	
			548h 1%	
		OTHERS 91h	VARIOUS 167h 1%	
	STEAM & POWER 1498h 2%	[illegible] 1498h 2%	[illegible] 1498h 2%	1627h REG. RESTRICT. 2%
	ENGINEERED SAFETY 1348h 2%	1348h FIRE-1241, OTHER-107 2%	VARIOUS 1348h 2%	1[illegible]h OTHER 1%
	ELEC. POWER/RADWASTE 261h 1%	VARIOUS 261h 1%	VARIOUS 261h 1%	47h TEST/INSPECTIONS 1%

*BWR Plant Outages totalled [illegible] hours (100%).

†The Browns Ferry fire was attributed to operational error. This one incident accounted for [illegible] hours or 21% of the total BWR outage time. All other operational errors combined to only [illegible] which is less than 2% of the total outage time.

Table 3-10

PRESSURIZED WATER PLANT OUTAGES,* 1975

OUTAGE TYPE	ASSOCIATED SYSTEM	ASSOCIATED COMPONENT	PLANT AFFECTED	OUTAGE CAUSE
FORCED OUTAGE	STEAM AND POWER	TURBINES	RANCHO SECO 4438h 8%	EQUIPMENT FAILURE
			PRAIRIE ISLAND 597h 1%	
			VARIOUS 658h 1%	
		5693h 10%		
		HEAT EXCHANGERS	PALISADES 2208h 4%	
			POINT BEACH 1 910h 1%	
			ZION 2 556h 1%	
			VARIOUS 1066h 2%	
		4740h 8%		
		VALVES 1503h 3%	VARIOUS--P. ISLAND-303, ZION 2-278, KEWANEE 271, OTHERS 651 1503h 3%	
		603h SHOCK SUPPRESSOR 1%	ZION 2 603h 1%	
		575h GENERATORS 1%	VARIOUS	
		398h I & C 1%		
		392h PUMPS 1%		
		701h OTHERS 1%	2066h 4%	
	14605h 26%			
	REACTOR COOLANT	PUMPS	ROBINSON 2 874h 1%	
			ZION 1 839h 1%	
			OCONEE 3 359h 1%	
			VARIOUS 925h 2%	
		2997h 5%		
		VALVES	OCONEE 2 1280h 2%	
			ZION 2 640h 1%	
			VARIOUS 372h 1%	
		2292h 4%		
		575h PIPES, FITTINGS 1%	VARIOUS	
		125h OTHERS 1%	[illegible]h 2%	
	5989h 11%			
	REACTOR	CONTROL ROD DRIVES	PALISADES 527h 1%	
		1053h 2%	VARIOUS 526h 1%	
		434h OTHERS 1%	VARIOUS 434h 1%	
	1487h 3%			22014h 39%
	880h ELECTRIC PWR 1%	880h VARIOUS 1%	VARIOUS 880h 1%	OPERATIONAL ERROR
	379h AUX PROCESS 1%	379h VARIOUS 1%	VARIOUS 379h 1%	1482h 3%
24197h 43%	857h OTHERS 1%	857h VARIOUS 1%	VARIOUS 857h 1%	659h OTHER 1%
				42h TYING-14/ADM-28 1%
SCHEDULED OUTAGES	REACTOR	FUEL ELEMENTS	SURRY 1 2429h [illegible]	REFUELING
			FORT CALHOUN 2160h [illegible]	
			TURKEY POINT 4 2009h [illegible]	
			GINNA 1658h 3%	
			[illegible] 1459h 3%	
			TURKEY POINT 3 1449h 3%	
			MAINE YANKEE 1336h 2%	
			SURRY 2 1253h 2%	
			CONN. YANKEE 1071h 2%	
			ROBINSON 2 962h 2%	
			SAN ONOFRE [illegible] 2%	
			OCONEE 1 764h 1%	
			POINT BEACH 1 744h 1%	
		18525h 33%	PALISADES 280h 1%	
		562h CR DRVE 1%	VARIOUS 562h 1%	
	19534h 34%	447h OTHERS 1%	VARIOUS 447h 1%	18525h 33%
	STEAM AND POWER	HEAT EXCHANGERS	ZION 1 1068h 2%	MAINTENANCE OR TESTING
			INDIAN POINT 2 823h 1%	
			COOK 1 478h 1%	
			PRAIRIE ISLAND 1 461h 1%	
			POINT BEACH 1 430h 1%	
			TURKEY POINT 4 403h 1%	
		4728h 8%	OTHERS 1061h 2%	
		897h TURBINES 1%	VARIOUS 897h 1%	
		370h PIPES, FITTINGS 1%	VARIOUS 370h 1%	
	6474h 11%	479h OTHERS 1%	VARIOUS 479h 1%	
	REACTOR COOLANT	PUMPS	OCONEE 1 1138h 2%	
			OCONEE 2-193, INDIAN PT 2-[illegible], COOK 1-340, OCONEE 3-309, OTHERS 1386h 2%	
		3453h 6%	931[illegible] 2%	
		713h VALVES 1%	COOK 1-379, OTHERS 334 713h 1%	
		OTHERS	VARIOUS	
	5672h 10%	1501h 3%	1501h 3%	13298h 23%
	524h ENG SAFETY 1%	524h VARIOUS 1%	VARIOUS 524h 1%	360h OTHER 1%
32729h 57%	525h OTHERS 1%	525h VARIOUS 1%	VARIOUS 525h 1%	144h [illegible]/ADM-31 1%

*PWR plant outages totalled 56926 hours (100%).

The steam and power system accounted for 2769 hours of which 920 hours were required for heat exchangers and 406 hours were required for pipes and fittings.

The instrumentation and controls system outages accounted for 438 hours. The 375 hours listed as others is comprised of reactor - 195, radioactive waste - 96, auxiliary water - 58, and auxiliary process - 26.

Scheduled Outages

Scheduled outages in BWRs total 31,490 hours or 48% of the total outage time. Refuelings accounted for 19,210 hours or 30%. Other activities, such as maintenance, often were carried out concurrently with refueling. However, in general, it was not feasible to prorate the outage time to other than the reactor system and fuel elements.

The reactor coolant system required 5132 hours. The dominant component was pipes and fittings which required 2956 hours. Dresden 3 installed a new feedwater sparger which accounted for 1705 hours.

The instrumentation and controls system required 2374 hours which was primarily due to local power range monitor vibration problems at Hatch 1, Peach Bottom 2, and Vermont Yankee.

The steam and power system required 1498 hours, the engineered safety features required 775 hours, and the electric power and radwaste systems combined accounted for 261 hours.

3.2.6 Pressurized Water Reactors

Forced Outages

Forced outages in PWRs accounted for 43% of the toal outage time, i.e., 24,197 of 56,926 hours. The largest portion of the forced outage time was the result of problems with the steam and power system. This includes the turbines which accounted for 5693 hours (10%), heat exchangers, which accounted for 4740 hours (8%), valves (3%), shock suppressors (1%), generators (1%), instrumentation and controls (1%), pumps (1%), and various others, another 1%. The turbine outage time was primarily due to problems at Rancho Seco which had 4438 hours of outage due to turbines. The outage time due to heat exchanger problems was principally due to Palisades, which accounted for 2208 hours.

Problems with the reactor coolant system accounted for 5993 hours of outage time. The dominant components were pumps, accounting for 2997 hours, and valves, accounting for 2292 hours. The reactor system accounted for 1487 hours of which the major component involved was control rod drive mechanisms, accounting for 1053 hours. The Palisades plant required 527 hours for this component.

The electric power system accounted for 880 hours, and the auxiliary process system accounted for 379 hours. The 857 hours listed as others include the following: engineered safety features - 347 hours, instrumentation and controls - 286 hours, auxiliary water system - 219 hours, and other auxiliary systems - 3 hours.

Scheduled Outages

Scheduled outages in PWRs totaled 32,729 hours or 57%. The reactor system accounted for 19,534 hours, of which 18,525 hours was for refueling. The steam and power system required 6474 hours, and the dominant component was heat exchangers which accounted for 4722 hours. The reactor coolant system accounted for 5672 hours, with pumps accounting for 3453 hours.

The engineered safety features required 524 hours and various other systems accounted for the remaining 525 hours.

3.2.7 Observations on BWR and PWR Outages

Forced Outages

Twenty-three BWR plants experienced 33,487 hours of forced outage; an overall average of 1456 hours per plant. Twenty-eight PWR plants experienced 24,197 hours of forced outage - an overall average of 864 hours per plant.

Additional insight as to the relative outages in BWRs and PWRs may be obtained by using the data in Tables 3-9 and 3-10, and comparing the outage percentage and the average number of hours per plant. This is done below where a listed component for either reactor type contributed 1% or more of the total outage time.

The dominant component contributing to PWR forced outage time was turbines accounting for 5693 hours or 203 hours per plant. This was followed closely by heat exchangers with 4740 hours or 169 hours per plant. At BWRs, a disproportionate amount of time was attributed to electrical conductors because of the fire which occurred at Browns Ferry, shutting down two reactor units for a combined total of 13,656 hours.

The second most dominant component at BWRs was pipes and fittings with 3769 hours or 164 hours per plant, followed closely by pumps with 130 hours per plant.

Outage Type	System	Component	PWR %	PWR Avg hrs	BWR %	BWR Avg hrs
Forced	Reactor	Control Rod Drive	2	38	-	-
	Reactor Coolant	Pumps	5	107	5	130
		Valves	4	82	2	74
		Pipes & Fittings	1	21	6	164
	Steam & Power	Turbines	10	203	-	-
		Heat Exchangers	8	169	1	40
		Valves	3	54	-	-
		Shock Suppressors	1	22	-	-
		Generators	1	21	-	-
		Instrumentation and Control	1	14	-	-
		Pumps	1	14	-	-
		Pipes & Fittings	1	-	1	18
	Electric Power	Electrical Conductors	-	-	21	598
		Transformers	-	-	2	62

Scheduled Outages

The 23 BWRs had 31.490 hours of scheduled outage time for an average of 1369 hours per plant. The 28 PWRs accumulated 32,729 hours for an average of 1169 hours per plant. The scheduled outages in the two types of reactors are compared below on the basis of percent outage and average number of hours per plant. The comparison is made where a listed component for either reactor type contributed 1% or more of the total outage time.

Obviously fuel elements, the component involved in refuelings, was the dominant component at both types of reactors. The percentage of outage time was nearly the same, but on the average BWRs required 256 hours (39%) more than PWRs.

Other than fuel elements, the dominant component at PWRs was heat exchangers requiring 8% of the total outage time, followed closely by pumps, requiring 6%. At BWRs the dominant component was pipes and fittings, requiring 5% of the total outage time.

Outage Type	System	Component	PWR %	PWR Avg hrs	BWR %	BWR Avg hrs
Scheduled	Reactor	Fuel Elements	33	662	32	918
		Control Rod Drives	1	20	<1	-
	Reactor Coolant	Pumps	6	123	-	-
		Valves	1	26	-	-
		Pipes & Fittings	-	-	4	108
		Electrical Conductor	-	-	1	37
		Valves	-	-	1	28
	Steam & Power	Heat Exchanger	8	169	1	25
		Turbines	1	32	-	-
		Pipes & Fittings	1	13	-	-
	Engineered Safety Features	Pipes & Fittings	-	-	2	54

3.2.8 Summary

During 1975, the 23 operating PWRs experienced an average of 2825 hours of outage time compared to an average of 2033 hours for the 28 operating PWRs. The percent forced outage at BWRs was 52% compared to 43% at PWRs. This was due primarily to the Browns Ferry fire which disabled two units and accounted for 21% of the total outage time accumulated at BWRs. The effect of the fire is also reflected in the percent scheduled outage with 48% for BWRs, while at PWRs it is 57%.

At PWRs, the primary cause of forced outages was equipment failures, while at BWRs the primary cause was operational error. Again, this is due to the Browns Ferry fire which was started by a candle held by a technician checking containment penetration leakage. Equipment failures accounted for 21% of the total outage time at BWRs.

Refueling was the primary cause of scheduled outages at both BWRs and PWRs requiring 30% and 33%, respectively, of the total outage time. Maintenance or testing also accounted for a large percentage of the scheduled outage time at both types of plants.

4.0 REPORTABLE OCCURRENCES

4.1 Introduction

The Nuclear Regulatory Commission collects and evaluates operational information concerning licensed nuclear facilities to assess safety, and to form the basis for comparing performance with design intent. Incidents or events occur that involve system, component or structural failure or malfunction, personnel error, design deficiencies, management deficiencies and other matters that are related to plant safety in various ways. Because of the multiple levels of protection, or defense-in-depth, including the provision of redundant safety systems and components, such events do not, in general, affect safety directly. Therefore, they do not have an actual impact or consequence on the health and safety of the public. However, information regarding them is useful to improve safety. Therefore, these events are brought to the attention of the NRC through a variety of reporting requirements or by NRC inspection, and appropriate enforcement and corrective measures are taken.

Plant technical specifications include a section on reporting requirements detailing the types of events that should be reported (a) as expeditiously as possible (within 24 hours) or (b) within 30 days. The data from these reports are stored in the Commission's License Event Report (LER) File for further analysis and evaluations and public dissemination. In general the reporting requirements for these two types of events may be briefly summarized as follows:

Prompt notification:

(1) Failure of the reactor protection system or other systems subject to limiting safety-system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified in the technical specifications or failure to complete the required protective function.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

(4) Reactivity anomalies involving disagreement with the predicted value under steady-state conditions during power operation greater than or equal to 1% $\Delta k/k$; a claculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

(7) Conditions arising from natural or manmade events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than that assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Thirty Day Reports:

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation, or plant shutdown required by a limiting condition for operation.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems designed to contain radioactive material resulting from the fission process.

The NRC started a program to standardize technical specifications including reporting requirements. However, the standardization was not completed during the period covered by this report and therefore the plants operated under different reporting requirements. It would be inappropriate therefore to compare the performance of the plants on the basis of the number of LER's submitted.

4.2 Licensee Event Report (LERs)

4.2.1 Introduction

The data for the 1974 and 1975 LER file have been tabulated by reactor type, i.e., BWR's and PWR's. The events have been categorized by the plant system involved in accordance with the lists of systems presented in Appendix C. In addition, components identified in the licensee report were categorized in accordance with the components list presented in Appendix B-2. In general, these system and component categories correspond to those developed by subcommittee N18-20 of the American National Standards Institute, Nuclear Plant Reliability Data. The differences in the system categorizations of 1974 and 1975 reflect the evolutionary changes made by the subcommittee.

Tables 4.1 and 4.2 summarize the 1974 LERs by reactor type and system while 1975 data are summarized in Tables 4.3 and 4.4.

The reports from which the data were derived may be reviewed at the NRC's Public Document Room (all reports are filed in the NRC's Public Document Room located at 1717 H Street, N. W., Washington, D. C. Documents relating to particular power plants are available at Local Public Document Rooms located in the vicinity of each plant).

4.2.2 Discussion

LERs only cover off-normal conditions and by themselves convey only negative impressions of plant operations. Extensive knowledge of normal operations, which is the situation most of the time is needed to put these events in their proper perspective. A large number of events of a particular type may not be significant to safety whereas a single event may be very significant in terms of its safety implications. The LER data should be considered as only one of several inputs to the overall evaluation of plant performance.

4.2.3 LERS - 1974

There were a total of 1,566 LERs submitted by 57 plants during the 1974 calendar year. This report, however, covers only those plants which had been in commercial operation more than 250 hours by December 31, 1974. There were, in this case 40 plants, 17 BWRs and 23 PWRs filing 1,253 LERs, 660 by BWRs and 593 by PWRs.

TABLE 4.1
BWR LERs 1974

	No Sys. Spec.	Aux. Sys.	Containment	Control Rods	Core Support	ECC Sys.	Fuel Sys.	Liquid Waste	Off Gas	Onsite Power	Other	Output Elec.	Pers Prot	Pri Cool	Proc Inst	React Prot	Sec. Sys.
Big Rock Point	–	5	4	6	2	4	4	–	3	–	2	–	1	–	–	4	1
Browns Ferry 1	1	8	6	3	–	21	1	–	3	–	–	–	2	12	1	3	2
Cooper Station 1	1	4	5	–	–	19	1	1	1	–	1	–	1	12	1	11	1
Dresden 1	–	3	4	6	–	3	–	4	–	–	–	–	–	3	–	2	1
Dresden 2	1	3	10	6	–	10	–	4	4	–	1	–	–	14	–	4	5
Dresden 3	2	–	5	1	–	10	1	1	3	–	2	–	–	10	–	2	–
Humboldt Bay	–	4	1	–	–	2	–	–	–	1	–	–	–	2	–	3	1
La Crosse	–	1	–	2	–	–	–	–	1	1	1	–	–	2	–	1	–
Mill-Stone 1	–	2	2	–	–	3	1	1	–	–	–	–	–	8	–	–	2
Monti-Cello 1	–	2	3	–	–	13	–	–	5	–	–	–	–	10	–	2	–
9 Mile Point 1	–	1	2	1	–	4	1	1	–	–	–	–	–	4	1	3	–
Oyster Creek	1	2	10	–	–	9	1	–	5	2	2	–	–	30	–	5	1
Peach Bottom 2	1	3	–	–	–	27	–	–	1	–	–	–	1	12	–	8	3
Pilgrim 1	1	15	2	1	–	16	–	–	4	2	–	–	–	8	–	1	–
Quad Cities 1	2	4	11	1	–	7	–	6	3	–	–	–	–	8	1	6	2
Quad Cities 2	1	–	5	–	–	8	–	1	1	–	–	–	–	10	–	2	–
Vermont Yankee 1	–	1	4	1	–	7	–	–	4	–	–	–	–	7	–	1	–

TABLE 4.2
PWR LERs - 1974

	No Sys. Spec.	Aux. Sys.	Containment	Control Rods	Core Support	ECC Sys.	Fuel Sys.	Liquid Waste	Off Gas	Onsite Power	Other	Output Elec.	Pers Prot	Pri Cool	Proc Inst	React Prot	Sec. Sys.
Conn Yankee	–	3	1	–	–	2	–	–	–	1	–	–	–	3	–	1	4
Fort Calhoun	–	4	3	–	–	2	–	–	–	2	–	–	–	2	1	4	2
H. B. Robinson	–	3	6	3	–	3	1	–	–	–	2	–	–	16	–	3	13
Indian Point 1	–	–	–	4	–	–	–	4	1	–	1	–	–	3	–	8	6
Indian Point 2	–	9	1	1	–	5	–	–	4	–	3	–	–	7	1	2	6
Kewaunee 1	–	4	2	2	–	–	1	–	–	–	–	–	–	9	–	2	4
Maine Yankee	–	1	1	–	–	–	–	–	–	–	–	–	1	2	–	1	–
Oconee 1	1	1	1	1	–	2	–	1	2	6	–	–	–	9	–	2	1
Oconee 2	–	1	3	–	–	6	2	1	1	2	–	–	1	8	1	2	–
Palisades 1	1	6	2	1	1	2	–	1	1	2	1	–	–	8	–	3	2
Point Beach 1	–	6	3	1	–	–	–	2	2	–	1	–	2	5	–	–	2
Point Beach 2	–	1	4	–	–	1	–	–	–	–	–	–	–	1	–	2	4
Prairie Island 1	–	7	6	–	–	2	–	1	–	2	3	–	–	9	–	1	3
Ginna 1	–	1	4	–	–	2	–	–	–	2	–	–	–	9	1	–	6
San Onofre 1	–	1	–	–	–	–	–	–	–	–	–	–	–	–	–	1	
Surry 1	–	9	–	–	–	3	–	–	–	–	–	–	–	9	–	–	1
Surry 2	–	1	1	–	–	1	–	–	–	–	–	–	–	3	–	1	3
3 Mile Island 1	–	11	9	2	2	4	–	5	–	–	3	–	1	15	–	3	4
Turkey Point 3	1	3	–	1	–	1	–	–	–	–	–	1	1	4	1	2	5
Turkey Point 4	–	1	–	–	–	1	–	–	–	–	–	–	1	5	–	–	2
Yankee Rowe	–	–	1	1	–	1	–	–	1	–	–	–	–	2	–	1	–
Zion 1	–	8	9	1	–	3	–	1	2	1	1	–	–	12	–	4	6
Zion 2	1	8	8	1	–	7	–	–	–	–	–	–	1	10	–	6	10

TABLE 4.3
BWR LERs - 1975

	Big Rock Point 1	Browns Ferry 1	Browns Ferry 2	Brunswick 2	Cooper Station 1	Dresden 1	Dresden 2	Dresden 3	Duane Arnold	Fitz Patrick	Hatch 1
Other Auxiliary Systems	2	–	–	–	1	–	–	–	1	1	–
Radiation Protection Systems	–	–	–	–	–	–	–	–	–	–	–
Reactor Coolant & Connected Sys.	6	3	5	83	14	3	5	16	19	32	19
Electric Power Systems	3	2	–	12	5	–	10	4	3	3	10
Fuel Storage & Handling Systems	–	–	–	–	–	–	–	–	–	1	–
Steam & Power Conversion Sys.	1	–	–	5	3	–	–	–	–	3	2
Instrumentation and Control	–	1	2	7	4	–	1	5	6	14	17
Radioactive Wst. Management Sys.	2	–	–	9	4	2	2	–	2	4	2
Auxiliary Process Systems	–	–	–	1	–	–	–	1	–	3	–
Reactor	8	–	1	4	2	–	3	3	6	1	3
Engineered Safety Features	6	3	–	46	16	6	23	16	33	25	31
Auxiliary Water Sys.	–	–	–	3	3	1	–	–	2	–	1
Other	–	–	–	–	–	–	–	–	–	–	–
System Code Not Applicable	4	1	–	2	–	2	1	–	3	4	2

TABLE 4.3 (Continued)
BWR LERs - 1975

	Humboldt Bay	LaCrosse	Millstone Point 1	Monti Cello	9 Mile Point 1	Oyster Creek 1	Peach Bottom 2	Peach Bottom 3	Pilgrim 1	Quad Cities 1	Quad Cities 2	Vermont Yankee
Other Auxiliary Systems	–	–	1	–	–	–	–	–	–	–	–	–
Radiation Protection Systems	–	–	1	–	–	1	1	6	–	–	–	2
Reactor Coolant & Connected Sys.	3	1	8	10	4	3	32	20	17	7	20	2
Electric Power Systems	1	1	3	2	–	3	2	1	2	1	3	–
Fuel Storage & Handling Systems	1	–	–	–	–	–	–	–	–	–	–	–
Steam & Power Conversion Sys.	–	–	–	–	1	1	–	–	–	2	–	–
Instrumentation and Control	–	1	8	4	21	2	13	10	2	5	4	4
Radioactive Wst. Management Sys.	–	–	2	–	3	3	5	1	7	2	1	4
Auxiliary Process Systems	–	–	–	–	–	–	1	3	–	–	–	1
Reactor	–	3	–	2	–	2	–	1	1	4	2	–
Engineered Safety Features	3	3	4	18	6	19	23	26	11	3	16	8
Auxiliary Water Sys.	–	–	1	–	2	1	1	2	1	–	–	–
Other	–	–	–	–	–	–	–	–	–	–	–	1
System Code Not Applicable	–	–	1	–	2	–	–	–	–	–	1	2

TABLE 4.4
PWR LERs 1975

	Arkansas 1	Calvert Cliffs 1	D. C. Cook 1	Fort Calhoun 1	H. B. Robinson	Conn Yankee	Indian Point 2	K unee	Maine Yankee	Oconee 1	Oconee 2	Oconee 3	Palisades
Other Auxiliary Systems	–	–	1	–	–	–	–	1	–	–	–	–	–
Radiation Protection Systems	–	2	4	2	–	–	–	–	–	–	–	–	–
Reactor Coolant & Connected Sys.	2	4	6	3	5	3	3	8	–	3	6	4	1
Electric Power Systems	1	2	4	4	1	–	1	3	–	3	1	2	2
Fuel Storage Handling Systems	–	–	–	–	–	–	1	–	–	–	–	–	–
Steam & Power Conversion Sys.	–	5	1	1	1	2	1	–	–	–	2	1	3
Instrumentation and Control	1	21	17	4	2	–	5	1	2	2	4	3	4
Radioactive Wst. Management Sys.	1	5	1	5	–	–	–	–	–	3	1	–	5
Auxiliary Process Systems	–	3	3	1	1	–	12	2	2	1	2	1	1
Reactor	2	–	6	1	3	–	–	–	–	4	4	2	7
Engineered Safety Features	4	16	9	7	3	1	5	6	7	4	17	9	7
Auxiliary Water Sys.	1	2	3	1	2	–	–	2	2	–	1		3
Other	–	1	–	–	–	–	–	–	–	–	–	–	1
No System Specified	–	–	11	–	1	–	–	–	1	–	–	–	1

TABLE 4.4 (Continued)
PWR LERs 1975

	Point Beach 1	Point Beach 1	Prairie Island 1	Prairie Island 2	Rancho Seco	R. E. Ginna	San Onofre	Surry 1	Surry 2	3 Mile Island	Turkey Point 3	Turkey Point 4	Yankee Rowe	Zion 1	Zion 2
Other Auxiliary Systems	–	–	–	–	1	–	–	–	–	2	–	–	–	–	–
Radiation Protection Systems	–	–	–	–	–	–	–	–	–	–	–	–	–	2	3
Reactor Coolant & Connected Sys.	2	3	4	1	4	6	–	4	3	8	1	–	3	7	5
Electric Power Systems	–	–	1	–	1	2	3	1	1	3	2	1	2	2	7
Fuel Storage & Handling Systems	–	–	–	–	–	1	–	–	–	–	1	2	–	–	–
Steam & Power Conversion System	2	1	3	1	–	2	–	1	3	3	1	2	–	1	1
Instrumentation and Control	3	1	2	6	2	4	–	6	4	8	4	1	4	5	12
Radioactive Wst. Management Sys.	2	1	5	1	1	–	–	6	–	2	–	1	–	1	–
Auxiliary Process Systems	–	1	–	–	3	2	–	3	–	4	–	–	2	2	[illegible]
Reactor	1	–	–	–	–	3	–	3	3	7	–	3	–	–	1
Engineered Safety Features	6	1	10	8	3	4	1	9	7	18	–	2	7	6	13
Auxiliary Water System	–	–	1	–		–	–	5	2	2	2	–	–	–	–
Other	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
No System Specified	–	–	3	1	–	1	–	–	–	1	–	–	–	1	–

Systems (See Appendix C-1 for list of systems)

Of the 17 system categories used in 1974, the five systems which were most often reported on were:

Type	System	Percent of Reports
BWR	ECCS	24.7
	PCS	23.0
	Containment	11.8
	Auxiliary	8.8
	RPI	8.8
PWR	PCS	25.5
	Auxiliary	15.0
	Secondary	14.2
	Containment	11.0
	RPI	8.3

The performance of these systems is of interest to the NRC. As previously noted, the reports cover a wide range of safety related occurrences with the vast majority involving events of minor consequences.

Reactor Status

The most frequent Reactor Status during reported occurrences were:

Type	Status	Percent of Reports
BWR	Steady state power operation	40.3
	Refueling	17.4
	Shutdown	16.7
PWR	Steady State power operation	38.6
	Shutdown	22.9
	No status specified	12.5

Proximate Cause

The three most frequently reported proximate causes for the occurrences were:

Type	Causes	Percent of Reports
BWR	Component Failure	66.4
	Personnel Error	10.9
	Defective Procedures	8.5
PWR	Component Failur	45.2
	Personnel Error	19.4
	Other	13.0

Method of Discovery

The most frequently reported method of discovery is listed below. It is interesting to note that the method of discovery of events were revered in BWRs and PWRs. It is significant that 52% of all events reported were the result of tests.

Type	Method of Discovery	Percent of Reports
BWR	Routine Tests	54.1
	Operational Events	36.5
PWR	Routine Tests	36.6
	Operational Events	52.3

4.2.4 LERs - 1975

The 51 commercially operating plants covered in this report submitted 1799 LERs during 1975. The 23 BWRs submitted 1072 LERs while the 28 PWRs submitted 727 LERs. The increased reporting is due primarily to an increase in the number of plants since new plants generally submit a greater number of reports during startup than do the older plants during routine operations. A contributing factor may be the increased standardization of reporting requirements resulting in the broadening of the scope of reportable occurrences.

The six new BWR plants which came on line in 1975 (Browns Ferry 2, Brunswick 2, Duane Arnold, Fitzpatrick, Hatch 1, and Peach Bottom 3) submitted 47% of the LERs filed by all BWR licensees. The six PWR plants that went into commercial operation in 1975 (Arkansas 1, Calvert Cliffs 1, DC Cook, Oconee 3, Prairie Island 2 and Rancho Seco) submitted 27% of the LERs filed by PWR licensees.

Systems (See Appendix C-2 for list of systems)

For 1975, system reporting was classified into 72 subsystems under 14 major system headings. The most frequently reported system of the 14 were:

Type	System	Percent of Reports
BWR	Engineered Safety Features	32.2
	Reactor Coolant & Connected Systems	31.0
	Instrumentation & Control	12.2
PWR	Engineered Safety Features	26.1
	Reactor Coolant & Connected System	13.6
	Instrumentation & Control	17.6

Proximate Cause

The 3 most frequently noted proximate causes were:

Type	Proxinate Cause	Percent of Reports
BWR	Component Failure	61.8
	Personnel Error	17.9
	Design Error	10.9
PWR	Component Failure	52.0
	Personnel Error	19.5
	Design Error	12.4

Method of Discovery

The two most frequent methods of discovery were:

Type	Method of Discovery	Percent of Report
BWR	Routine Tests	61.7
	Operational Event	30.8
PWR	Operational Event	48.4
	Routine Tests	41.4

Reactor Status

The most frequent reactor status at the time of the reportable occurrences were:

Type	Status	Percent of Reports
BWR	Steady State Power Operations	48.8
	Shutdown	15.1
	Preop. Testing Initial Startup and Power Ascension	14.5
PWR	Steady State Power Operations	56.0
	Shutdown	18.7
	Preop. Testing Initial Startup and Power Ascension	7.2

4.3 Abnormal Occurrences

Section 208 of the Energy Reorganization Act of 1974 directs the NRC to "... submit to the Congress each quarter a report listing for that period any abnormal occurrences at or associated with any facility whih is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954 as amended, or pursuant to this Act. For the purposes of this section, an abnormal occurrence is an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety...."

In Section 208, the information to be reported was specified: date and place; nature and probable consequence; cause or causes; and action taken to prevent recurrence.

The NRC developed in 1975 two major interim criteria by which abnormal occurrences were to be determined: (1) events involving an actual loss of the protection provided for the health and safety of the public, and (2) events involving major reduction in the degree of protection provided for the health and safety of the public.[1]

None of the events occurring at nuclear power plants in 1975 had any direct impact on public health and safety, although some involved temporary but significant reductions in the levels of protection provided. From over 1,900 reportable occurrences in 1975, seven were considered to be abnormal occurrences under the interim criteria. Three of these were "single" incidents, one of a recurrent nature, and three were generic with implications for a number of facilities. There were no offsite exposures to radiation caused by these events and there were no releases of radioactive materials that exceeded regulatory limits. Table 5 lists the abnormal occurrences. A summary of each follows.

[1] See Appendix D for interim criteria with examples.

Table 5

ABNORMAL OCCURRENCES AT NUCLEAR POWER PLANTS

Date	Event Type	Event	Facility
February 26, 1975	Single	Steam Generator Tube Failure	Point Beach 1
March 22, 1975	Single	Fire in Electrical Cable Trays	Browns Ferry 1 & 2
May 1, 1975	Single	Loss of Main Coolant Pump Seals	H. B. Robinson 2
January 25, 1975 and May 3, 1975	Recurring	Improper Control Rod Withdrawals-Maintenance	Dresden 2 Quad-Cities 1
Various: September 1974 to December 1975	Generic	Cracks in Pipes at Boiling Water Reactors	Dresden 2, Quad-Cities 1 & 2, Millstone 1, Monticello, Peach Bottom 3, Pilgrim 1, Hatch 1 and Monticello
April 1975	Generic	Fuel Channel Box Wear at Boiling Water Reactors	Duane Arnold, Cooper, Peach Bottom 2 & 3, Browns Ferry 1 & 2, Brunswick 2, Hatch 1, FitzPatrick, and Vermont Yankee
Various: October 1972 thru 1975	Generic	Steam Generator Feedwater Flow Instability at Pressurized Water Reactor	Surry 1, Turkey Point 3 & 4, Indian Point 2, and Calvert Cliffs 1

(Note: For the recurring and generic events, the circumstances surrounding the events varied from Plant to plant.)

4.3.1 Steam Generator Tube Failure

This event occurred on February 26, 1975, at the Point Beach Nuclear Plant, Unit 1. This unit is a pressurized water reactor using primary system water to transfer heat from the reactor fuel to the secondary water system. Water in the secondary system, which is nonradioactive, is converted to steam in two steam generators by the heat from some 3,200 tubes in each generator through which primary system (radioactive) water flows. A hole developed in a 1-inch diameter tube in the "B" steam generator, resulting in a contamination of the secondary system. The leak rate from the primary to the secondary system reached an estimated maximum of 125 gallons-per-minute in about three-quarters of an hour. Licensee personnel manually shut down the unit. Offsite radioactive releases during the event were within the NRC regulatory limits and liquid releases were below maximum permissible concentration values.

The event is significant because it involved a large primary-to-secondary leak rate, the first tube failure to occur during plant operation with leakage in excess of 25 gallons-per-minute. The rate was, however, only one-fifth of the rate postulated for safety design purposes and was handled within the capability of the normal primary coolant makeup system. Several radiation monitors designed to diagnose such an event did not perform as intended and caused a delay in determining that a tube failure had occurred.

The cause of the leak was identified as a buildup of sludge around the tube from a phosphate water treatment used to control tube corrosion. The build-up resulted in accelerated corrosion and subsequent tube failure. The sludge was removed, inspections performed on the tubes in both steam generators, and the defective tube was plugged, as were tubes with excessive wall thinning. For the long term, continuous removal of sludge, modification to the steam generators to minimize build-up, continuing use of a new all-volatile treatment for secondary water chemistry control, and more frequent in-service inspections have been adopted. Radiation monitoring systems and operational procedures have also been modified, and NRC has changed the specified primary-to-secondary leak rate limits to require earlier corrective action. Generic aspects of incidents of this kind were previously identified and a plant-by-plant review of the operating experience of steam generators is continuing. The Unit resumed operation on April 5, 1975.

4.3.2 Fire in Electrical Cable Trays

A fire occurred on March 22, 1975, at the Browns Ferry Nuclear Plant of the Tennessee Valley Authority. The plant contains three nuclear units. One of the plants was under construction. The two operational units, each powered by a boiling water reactor with a net electrical capacity of over 1,000 megawatts, and each in full power operation at the time of the fire, were shut down for an extended period as a result of this occurrence.

The fire started in an electrical cable penetration between the cable spreading room and the reactor building; the cable spreading room is located beneath the common control room for Units 1 and 2. The fire burned for about seven hours, spreading horizontally and vertically to all 10 cable trays within the penetration, into the cable spreading room for several feet, and along the cables through the penetration about 40 feet into the reactor building. The fire damage, confined to an area roughly 40 feet by 20 feet in the Unit 1 secondary containment building, affected about 1,600 electrical power and control cables.

While both units were shut down safely, normally used shutdown cooling systems and other components which comprise the emergency core cooling system (ECCS) for Unit 1 were inoperable for several hours. Other installed equipment was employed to maintain sufficient cooling capability to protect the nuclear fuel from overheating. There were no significant problems with the shutdown cooling of the Unit 2 reactor. Even though normal and emergency core cooling systems were unavailable in Unit 1 for a time, at least five alternative methods were available to provide adequate core cooling within the required time frame. There was no adverse impact on the public, plant personnel or the environment as a result of the fire; sampling indicated that airborne release rates were less than 10% of the Technical Specification limit. Several cases of smoke inhalation and other minor injuries were incurred by fire-fighting personnel.

The cause of the fire was the ignition of cable penetration sealing material by a candle flame, being used by a construction worker checking for air leaks. The flexible polyurethane foam sealing material being used had not been specifically approved by the licensee's design department, nor had it been tested for this kind of application. The dangers involved in using flammable material in this manner were evidently not recognized by plant management, even though several small fires had occurred during similar testing activities at the plant. Personnel inspecting, sealing and testing the cable penetrations had not been provided with an adequate written procedural guide. Another contributing factor may have been the plant's fire-fighting techniques and equipment.

Following the fire, the TVA removed all the fire damaged cable and equipment. Facility modification (such as the addition of permanently installed water spray systems, fire detection systems, and fire barriers) were made. New designs for the electrical penetration seal and fire stop were tested and used to replace the flammable polyurethane. Additional insulating material was installed to form a fire stop.

Administrative controls and procedures were revised including plans for training of personnel in fire fighting, emergency procedures and communications, and clarification of responsibilities in emergency situations.

A special review group within NRC studied the incident and have issued a report including recommendations to prevent or mitigate the consequences of similar events at other plants. In addition, the NRC instructed all licensees to review overall policies and procedures related to the possible effect of construction work on reactor operation, fire protection and emergency shutdown, and to reevaluate electrical system design.

Following completion of the modifications and renovation, a hearing was held in early August of 1976. As a result of the hearing, the Commission on August 20, 1976 authorized the licensee to resume operation of both Units 1 and 2.

4.3.3 Loss of Main Coolant Pump Seals

This event occurred on May 1 and 2, 1975, at the Carolina Power and Light Company's H. B. Robinson S. E. Nuclear Power Plant, Unit 2. The nuclear power unit at the site is a pressurized water reactor with a net capacity of about 700 MWe. Three primary coolant loops circulate pressurized water from the nuclear core to the steam generators. Each loop has one main coolant pump, and each pump shaft has three seals arranged in series to prevent any coolant leakage to the containment structure. The system of seals is lubricated and cooled by a water source other than the primary coolant water; normal leakage of the seal water system, called "seal water leakoff," runs through leakoff lines from the three pumps to a common line. When a first stage pump seal was found to be leaking primary coolant water, reactor power was quickly reduced to about 35% and the leaking pump was shut down. Shortly thereafter, an automatic shutdown of the plant occurred because a signal indicated that the water level in one of the steam generators was too high. Four hours later--with the other two pumps shut down and, despite attempts, not restarting--the licensee restarted the leaking pump. It was operated for about 90 minutes, during which time all three seals failed, resulting in a discharge of about 132,500 gallons of radioactive primary coolant water into the containment structure. The structure contained the leakage. However, the leak could not be isolated from the primary coolant system because the coolant loops were not designed with isolation valves. A maximum leak rate estimated at 400 gallons-per-minute occurred, greatly exceeding the postulated leakage from seals of this design. Coolant makeup was provided by a safety injection system designed for the purpose, and the fuel was adequately cooled.

Steam from the failed seals affected the cooling system for the other pumps, which were then shut down could not be restarted. Leakage from the failed seals was not stopped until some 16 hours after the initial shutdown of the leaking pump. During the forced plant cooldown, the normal cooldown rate prescribed in specifications of the license was exceeded. The plant was safely shut down, however, and maintained in a safe shutdown condition. Offsite release of radioactivity was within Technical Specification limits.

The event was caused by the complete failure of the seal system on a main coolant pump. That failure was compounded by failure of the pump radial bearing which damaged other seals. Procedural errors were contributing factors. The reason for the failure of the first seal, which set the train of seal failures in motion, has not been identified, although improper maintenance may explain it.

The licensee replaced all the failed parts on the affected coolant pump and all three pumps were inspected and tested before returning to service. New procedures were implemented to prevent events of this kind, with closure of the pump seal water leakoff isolation valve as the immediate action to be taken upon indicated damage to a No. 1 seal. The plant resumed operation in June 1975.

The NRC sent a Notice of Violation to the licensee, citing noncompliance with regulations, including failure to adhere to the approved procedure for plant cooldown and failure to meet commitments of the quality assurance program. A critical review of the pump seal design was performed by NRC because the reported leak rate exceeded previously stated values for that pump seal design. The review concluded that the design was adequate in that, should a No. 1 pump fail completely, the No. 2 is designed for full system pressure, thus serving as a total backup. The failure of No. 2 pump seal system in the occurrence cited was the result of continued operation under abnormal conditions, causing mechanical damage and a large leak rate. Such operation was not in accord with established procedures and more stringent controls have been applied to prevent recurrence.

4.3.4 Improper Control Rod Withdrawals

These two occurrences took place on January 25, 1975, at the Dresden Nuclear Power Station, Unit 2 and on May 3, 1975, at the Quad Cities Station, Unit 1. The licensee for the Dresden plant is the Commonwealth Edison Co. and for the Quad-Cities station the licensees are Commonwealth Edison and the Iowa-Illinois Gas & Electric Co.

The Dresden 2 facility is a boiling water reactor plant with a rating of 809 MWe net. While the unit was shut down for refueling, maintenance was being performed on the control rod drives. Through personnel errors and inadequate procedures, two adjacent control rods were withdrawn to the full-out position, in violation of the minimum separation criterion for such maintenance. The criterion is intended to prevent an unintended self-sustained nuclear chain reaction, or "inadvertent criticality." The reactor remained shutdown by a safe margin; instrumentation verified that no criticality existed during the control rod withdrawal, and there was no release of radioactivity or damage to the facility. Satisfactory control was demonstrated during the performance of the shutdown margin test that is required after each refueling.

Quad-Cities 1 is a boiling water reactor plant with a rating of 809 MWe net. This unit was shut down and maintenance on the control rods were being performed much as in the Dresden incident. Two control rods were withdrawn in violation of the minimum separation criterion. In this case, the rods were separated by one inserted rod. The reactor was shut down by a safe margin; instrumentation verified that no criticality existed during the control rod withdrawals, and there was no release of radioactivity or damage to the facility.

Even if criticality had occurred in either of these two events, it is extremely unlikely that any impact on the general public would have resulted. Automatic safety features were available and functioning, such as automatic control rod insertion activated by criticality monitoring systems. Both incidents involved personnel error and procedural inadequacy; automatic protection devices can mitigate the larger possible consequences of criticality, but, during refueling and rod maintenance plant personnel are usually on the refueling floor and could be subjected to radiation exposure in the event of inadvertent criticality. For this reason and others, the NRC places serious emphasis on maintaining strict personnel controls during refueling.

The cause of these events was, as noted, both human error and procedural deficiency which represent a potentially recurring problem. The licensees in these instances have instituted programs to improve personnel performance and management control and have modified rod maintenance procedures accordingly The NRC investigated both events, issuing a Notice of Violation to the licensee and imposing a civil penalty.

4.3.5 Pipe Cracks in BWR

A series of events raising questions of a generic nature cons titute this occurrence in boiling water reactors. Small hairline cracks were discovered in the bypass lines in September 1974 at the Dresden Unit 2, Quad-Cities Unit 2 and the Millstone Unit 1. Again in December 1974, similar defects were observed at the first five facilities, and yet again in January 1975 at those two plants and at the Peach Bottom Station Unit 3 and at Monticello. Hairline cracks were also discovered in the core spray pipes at Dresden Unit 2. Eventually, similar cracks were discovered at other boiling water reactors including Hatch Unit 1 and Pilgrim Unit 1.

The existence of small hairline cracks was first detected by a leakage monitoring system at Dresden 2. Subsequent inspection disclosed similar cracks at the other installations and in 10-inch lines in the core spray system. The cracks were determined to be of the type which propagate slowly and are readily detectable before they could lead to large leaks or pipe rupture. All pipes involved were either type 304 or 316 stainless steel.

Though no immediate hazard was presented by the cracks, the fact that they affect one of the primary boundaries for the containment of radioactive material warranted prompt study and action.

After the first cracks were detected in September 1974, the Atomic Energy Commission directed that all boiling water reactor licensees with bypass systems similar in design to those found defective conduct examinations of welds in the bypass lines within 60 days. More stringent coolant leakage limits were also imposed. No new cracks were found. When new cracks that did not exist in September were found in December at the Dresden plant, the AEC directed all BWR licensees to reevaluate their September findings, conduct more examinations at the next scheduled shutdown, and observe even more stringent leakage limits. Cracks were found in four reactors. In early January 1975, the AEC formed a special study group to coordinate and intensify the investigation of causes.

After small cracks were found in two 10-inch core spray system pipes at Dresden 2 the NRC directed the operators of all operating BWRs to conduct an inspection within 20 days of all circumferential welds in each core spray loop within the boundary of the reactor coolant system, plus a representative sampling of welds in other stainless steel piping. No cracks were found.

The cause of the cracking, as determined by the special NRC study group in late 1975, was corrosion resulting from a combination of stress, water chemistry and the type of material used: a type of austenitic stainless steel which loses some of its resistance to corrosion in heat-affected zones adjacent to welds in relatively small diameter, thin-walled applications. The study group recommended a continuing program of surveillance for cracking; replacement of cracked pipes with others made of material less susceptible to the kind of corrosion that caused the initial cracking; further investigation into possible changes in operating procedures to reduce the relatively high level of oxygen contributing to corrosion in the pipes. Although additional cracks may develop in the future, the study group reported that they do not pose a threat to public health and safety because they can be detected by periodic inspection or sensitive leak detection equipment. In no instance was the structural integrity of the cracked pipes affected. There had been no releases of radioactivity as a result of the cracks and, even in the remote case of pipe failure, redundant core cooling systems are available and functioning at all plants involved.

All affected pipes were replaced by the licensees and, where system operation was a major factor, operational procedures were revised.

In-depth investigations of the problem initiated by the NRC are continuing, including research on corrosion susceptibility of structural materials, residual and operating stress measurements, welding and fabrication practices, and nondestructive testing. The Energy Research and Development Administration is sponsoring efforts in the same area. Foreign countries have been apprised of the action taken by NRC licensees and asked to convey results of similar examinations conducted at their BWR facilities. Cracks have been found in the core spray piping of two reactors located in Japan; NRC is in touch with other nations currently conducting tests.

4.3.6 Fuel Channel Box Wear

Another series of events, with generic implications, involves fuel channel box wear at boiling water reactors. First notice of a problem came on April 17, 1975, from the Duane Arnold Energy Center, Unit 1. Other plants subsequently affected are: Cooper Nuclear Station; Peach Bottom Atomic Power Station, Units 2 and 3; Browns Ferry Nuclear Power Plant, Units 1 and 2; Brunswick Steam Electric Plant, Unit 2; Edwin I. Hatch Nuclear Plant, Unit 1; James A. FitzPatrick Nuclear Power Plant; Vermont Yankee Generating Station.

The General Electric Co. reported to the NRC that excessive wear and damage to some fuel channel boxes adjacent to the incore instrument tubes had been found in one class of boiling water reactor (BWR-4) by a foreign operator. The thin-walled metal fuel channel box encloses a bundle of fuel rods; one of its purposes is to guide the flow of coolant water around the fuel rods. It was determined that the wear was occurring as the result of the vibration of the in-core instrument tubes. Operation of a plant for extended periods with high wear rates could lead to penetrations of the channel wall, allowing too much of the reactor coolant to bypass certain fuel rods and thereby reducing thermal safety margins. Loose channel box fragments could also cause local coolant flows blockage and possible overheating of some of the fuel rods.

Prompt corrective action was taken in all instances cited, and there was no impact on public health and safety. The margin of safety was assured in all cases by reducing local power generation and permissible thermal-hydraulic operating limits and by reducing r- -or coolant flow to decrease instrument tube vibration.

Surveillance of anomalous noise on the in-core instrument readings, indicative of vibration, was increased, and limits were placed on the permissible magnitude of such noise.

The problem is significant with respect to maintaining safety margins for the reactor fuel cladding, a principal barrier for the retention of fission products (radioactive material formed with the fuel cladding during the fission process). There was no release of radioactivity in any of these occurrences.

The cause of the channel box wear is, as noted, vibration over a period of time; the vibrations are set up in the incore instrument tubes by water turbulence. A high velocity flow through holes in the lower core support plant, which are intended to permit a certain amount of coolant to flow outside the channel boxes, is the basic cause.

All affected licensees of operating BWRs have plugged the holes in the lower core support and replaced the fuel channel boxes where excessive wear had been discovered. These actions have eliminated the cause of the fuel channel box wear problem using interim corrective measures. General Electric has proposed a further corrective measure designed to resolve any operational problems, such as thermal-hydraulic limitations, resulting from the interim measure.

The NRC has monitored the activities of all licensees affected by this problem. Licensing actions have been taken covering operations with the bypass holes plugged. Surveillance programs will be used to monitor for any unanticipated operational anomalies.

4.3.7 Feedwater Flow Instability--Water Hammer

Also of generic importance was a series of events involving the phenomenon called "water-hammer" in pressurized water reactors. The problem was experienced as far back as 1972 at the Surry Power Station, in 1973 at the Turkey Point Station, Unit 3; the Robert Emmett Ginna Nuclear Power Plant; and the Indian Point Station, Unit 2; in 1974, again at the Turkey Point Station, in Unit 4; and in May 1975 at the Calvert Cliffs Nuclear Power Plant, Unit 1, in Calvert County, Maryland.

In pressurized water reactors, an essential part of the secondary water system (nonradioactive) is the feedwater system. This system returns water from the main condenser to the steam generators and maintains the water inventory in the secondary system. Each PWR has at least two steam generators. Loss of the feedwater system by pipe or valve failure could affect the ability of the plant to cool down after a reactor shutdown, though auxiliary systems are provided as backup.

Water-hammer occurs when steam replaces water in the feedwater distribution piping (sparger) or in the feedwater inlet nozzle of the steam generator. This happens when the steam generator water level drops below the level of these components. Restarting feedwater flow causes condensation of the steam and is one of the factors inducing water-hammer. Other factors may be involved and are being sought. Feedwater flow instability, leading to water-hammer, can damage feedwater system piping and associated components; it occurred with varying severity at the above-named plants, usually after restarting feed flow following an operational adjustment required by some abnormal condition, such as a rapid change in the steam generator water level. In the remote instance that both the normal and auxiliary feedwater systems should be lost to several steam generators at once, the capability for plant cooldown would be affected. The development of design and operational modifications to reduce water-hammer to a minimum is clearly indicated. Termination of feedwater flow to several steam generators, however, has not occurred and, in none of the events cited, was radioactivity released or satisfactory safety margins compromised.

At plants where the phenonemon has occurred, corrective actions have been taken. These include changing the feedwater piping arrangement, modifying the feedwater distribution ring or steam generator refilling procedures, or limiting refill flow rate to reduce condensation. In early 1975, the NRC contacted all PWR reactor licensees requesting a review of the potential for water-hammer in their systems and its potential consequences.

The licensees of operating pressurized water reactors have responded to an NRC request for information.

Many licensees are making modifications to the feedwater system to either alleviate the consequences of water-hammer or prevent its occurrence, e.g., modification of the feedwater inlet line and the installation of J-tubes on the feedwater distributor ring located inside the steam generator.

The utilities, vendors, and architect/engineers are working to reduce the occurrence of these type events. Field tests are being conducted to demonstrate the adequacy of system modifications.

Tests conducted in September at one pressurized water reactor plant involving a feedwater system modification, including J-tubes on the distribution ring, verified a reduced occurrence of severe water-hammer over the range of feedwater flow tested.

The NRC is continuing to review information submitted by licensees, including field test data. The generic aspects of feedwater flow instability will continue to be studied.

5.0 FUEL PERFORMANCE

5.1 Introduction

For the purpose of this report, fuel performance is defined in terms of the number or percent of fuel elements that failed. For completeness, actual or potential damage to the fuel rods is also discussed. A fuel failure generally means the perforation of the cladding which normally protects against fission products entering the primary coolant system.

This section describes the fuel performance for 1974 and 1975, as well as consequences on reactor operation. The information for 1974 was taken from NUREG-0032, dated January 1976.[1] The information for 1975 was taken from the semiannual or annual operating reports and Licensee Event Reports (LERs) submitted by the licensees. Only those plants are listed in which specific fuel performance information was submitted by the licensees in their operating and licensee event reports. If known, the causes of the failures are also described.

5.2 BWR Fuel Experience

A summary of BWR fuel failure experience in 1974 and 1975 is listed in Tables 5-1 and 5-2, respectively. Most of the fuel rod failures, for both years, were apparently caused by two basic mechanisms: localized internal hydriding and fuel pellet-clad interaction. During 1975, a problem pertaining to fuel channel box wear was identified.

5.2.1 Internal Hydriding

Internal hydriding has been a continuing cause of BWR fuel failures for several years. Such failures are the result of excessive moisture or hyrogenous material left in fuel rods during fabrication. The fuel failures attributed to hydriding are generally for fuel manufactured several years ago. In the past few years, changes in fabrication techniques and specifications have been made to prevent excessive internal hydrogenous impurities from being introduced into BWR fuel rods during manufacture. Therefore, failure due to hydriding should decrease as the older fuel is replaced by the improved designs.

5.2.2 Pellet-clad Interaction

The pellet-clad interaction (PCI) mechanism involves localized mechanical loading of the cladding adjacent to cracks in the fuel pellets and at pellet interfaces. The PCI effects are reduced by the use of "fuel preconditioning." The latter is in the form of procedures for a periodic, slow ascent to full power, which preconditions the fuel for subsequent normal full power generation. Such procedures result in the reduction of power generation (capacity factor) due to the longer periods of time required to reach full power operation.

Table 5-1

SUMMARY OF BWR FUEL FAILURE EXPERIENCE IN 1974

REACTOR	EXPERIENCE
Big Rock Point	In the March refueling, 9 assemblies out of 84 were found to be leakers. Most probable cause was accelerated cladding corrosion induced by crud spalling and the resulting localized heating. Crud buildup on one-cycle assemblies was minimal. After startup, off-gas rates continued at high levels. Power derated to 63 MWe in May. After encountering other plant problems in June, decision was made to refuel once again. Dry sipping of 71 assemblies showed 15 leakers. Cause was likely to be the same as described for the March refueling.
Dresden 2	Off-gas activity during 1974 indicated several fuel rod failures. During the Fall refueling, 615 assemblies were wet sipped out-of-core. Thirty-eight defective assemblies were detected.
Dresden 3	During the March refueling, in-core and out-of-core sipping showed 27 definite leaker assemblies plus 6 probable defective assemblies. Most probable causes were hydriding and pellet-clad interactions. On October 31, a sudden increase in off-gas radiation occurred, indicating that several fuel rods had ruptured. The sudden increase in off-gas followed after rapid local power changes were allowed to occur, probably resulting in several pellet-clad interaction failures. Plant was then limited to lower power levels to reduce the off-gas rates.
Humboldt Bay 3	During the October refueling, 60 assemblies were selectively dry sipped. Eleven leakers were identified; these assemblies were all in high power density regions.

Table 5-1 (Cont'd)

REACTOR	EXPERIENCE
Millstone 1	Plant was restricted frequently to 80% power due to off-gas activity. During the Summer refueling, of about 460 assemblies dry sipped, about 25 were leakers.
Monticello	During Cycle 2, power was administratively reduced to lower the stack off-gas activity. In the March refueling, in-core and out-of-core wet sipping identified 83 leaking assemblies out of 484. Pellet-clad interaction was the predominant failure mechanism. During Cycle 3, power was again administratively limited at various times to reduce stack off-gas activity. Another refueling was planned for early 1975 to sip and replace defective assemblies.
Nine Mile Point 1	During the refueling starting in late March, wet sipping identified 28 leakers.
Oyster Creek 1	During the April refueling, in-core sipping identified 27 leakers out of 560 assemblies.
Pilgrim 1	During refueling outage, fuel sipping began on 1/18/74. Sixteen fuel assemblies showed indications of cladding perforations. In addition, 4 other assemblies were damaged. In the last half of December, station operation limited to about 95% of rated power due to high airborne effluent release rates and unexplained perturbations in the Augmented Off-Gas System.
Quad-Cities 1	The plant was administratively limited in power level at times, starting in the last half of 1973, to maintain stack rates at acceptable levels. During the refueling outage starting 3/31/74, in-core and out-of-core sipping identified 29 leaker assemblies out of 724. Cause of the failures attributed to cladding hydriding and pellet-clad interactions.

Table 5-1 (Cont'd)

REACTOR	EXPERIENCE
Vermont Yankee	Plant was administratively limited to lower power during 1974 due to excessive off-gas activity at the steam jet air ejectors. Problem was attributed to "faulty cladding", probably caused by hydriding. During the Fall refueling, 328 assemblies were replaced by the new 8x8, design. The remaining 40 (of the improved 7x7 design) were wet sippped out-of-core and no defects found; these 40 were reinserted into the core.

Table 5-2

SUMMARY OF BWR FUEL FAILURE EXPERIENCE IN 1975

REACTOR	EXPERIENCE
Big Rock Point	During an outage starting in January 1975, channel No. 98, an old zirc model, was found slightly damaged in three places on the lower edge of the support tube. The damaged section was reformed and the channel was returned to its core position. At the end of the year, the 13th fuel cycle continued smoothly. Effluent releases were well within limits.
Browns Ferry 1	As a result of the cable tray fire on March 22, 1975, the plant remained down for the rest of the year. The plant was defueled. During August, the fuel channel boxes were inspected. Inspection results of the 248 channels adjacent to instrument tubes were as follows: 1 channel with perforations, 125 rejectable wear, 98 probable acceptable (minor wear), and 24 acceptable (no wear). On December 22, a program was commenced to plug the bypass flow holes in the bottom core plate.
Browns Ferry 2	As a result of the cable tray fire on March 22, 1975, the plant remained down for the rest of the year. The plant was defueled. During August, the fuel channel boxes were inspected. Inspection results of the 248 channels adjacent to instrument tubes were as follows: 159 rejectable wear, 75 probable acceptable (minor wear), and 14 acceptable (no wear). On December 8, a program was commenced to plug the bypass flow holes in the bottom core plate.
Cooper Station	In response to notification of possible LPRM vibration and associated channel box damage, power and flow were limited. During the October outage, all 192 fuel channels were inspected; 125 channels were considered rejects and were replaced. The bypass holes in the lower core plate were plugged and the plant resumed operation in early November 1975. Off-gas activity showed no increases indicative of significant fuel failures.

Table 5-2 (Cont'd)

REACTOR	EXPERIENCE
Dresden 1	Cycle 9B was completed on September 1, 1975. During the outage, a complete out-of-core sipping program identified 27 failed fuel assemblies.
Dresden 3	The unit was 50% derated on October 31, 1974 due to fuel damage following control rod movements. The derating continued until the end of Cycle 3 on April 16, 1975. During the outage, the entire core of 724 fuel assemblies were wet sipped out-of-core. The sipping program identified 113 defected fuel assemblies.
Duane Arnold	On June 6, 1975, the plant was shut down for an inspection of fuel channel boxes surrounding in-core instrument tubes. During the shutdown, 134 fuel channel boxes were inspected; of these, 63 were rejected and replaced, 54 were acceptable for use in locations not adjacent to in-core instrument tubes, and 17 were acceptable. During the shutdown, an interim fix to the vibration problem was accomplished by plugging all 49 bypass holes in the lower core plate. Reactor operation resumed on July 19, 1975 with an amendment to the facility operating license allowing a minimium core power ratio of 1.34 (approximately 85% power). During the fuel movement operations associated with the inspection and fix, fuel assembly AR 156 dropped from its grapple into the core, impacting directly onto fuel assembly AR 356. Both fuel assemblies were damaged and replaced. Two fuel assemblies (AR 149 and 174) adjacent to assembly AR 356 sustained channel box damage; the damaged channels were replaced.
Edwin I. Hatch 1	On November 16, 1975, the unit was brought down to inspect the fuel channel boxes and to plug 96 bypass flow holes in the lower core support plate. The results of the

Table 5-2 (Cont'd)

REACTOR	EXPERIENCE
	inspection showed 3 channels with no visible wear, 66 channels with acceptable wear, and 125 channels with rejectable wear. There were no perforated channels observed. Plant startup commenced December 24, 197-
Humboldt Bay 3	The unit was shut down for refueling on May 30, 1975. Due to the increase in off-gas activity during Cycle 10, a program to dry sip the fuel was performed. The majority of elements sipped were from the central region of the core and high exposure elements to be retained for Cycle 11. Forty-seven assemblies were sipped and 11 leakers were detected. The core size for Cycle 11 was increased over that of Cycle 10. Off-gas activity for Cycle 11 indicated that all failed fuel was removed and no additional failures had occurred by the end of 1975.
La Crosse	Fuel Cycle 3 ended May 9, 1975, after 16 months of operation. All fuel assemblies were removed from the core for visual examination; in addition, 68 assemblies were dry sipped. Underwater TV detected defective fuel rods in 4 fuel assemblies. In addition, dry sipping identified 5 other fuel assemblies which contained leaking fuel rods plus one other possible leaker assembly. The 4 assemblies with visible defects are believed to have failed due to significant power increases early in the cycle. While shuffling fuel around in the core to facilitate cleaning the shroud can inlets and reactor upper grid, fuel assembly 2-12 suffered damage to the top intermediate spacer grid when it hung up on the top edge of the shroud can in core position K-8. It was replaced by assembly 2-3, even though the latter contains a leaky

Table 5-2 (Cont'd)

REACTOR	EXPERIENCE
	corner pin. The only alternative to using this assembly would have been to use an assembly with high burnup.
Millstone 1	During the refueling commencing September 14, 1975, 513 fuel assemblies out of 580 were sipped. Thirty-nine leakers were identified (contained cladding perforations). None of the reload 2 assemblies (Type GEB and MSB) were identified as leakers.
Monticello	Power was limited to about 66% until shutdown for refueling on January 9, 1975, to minimize in-plant background and contamination levels. Fuel sipping during the outage identified 42 leaking fuel assemblies (all part of the original core). In addition, 12 possible leaker assemblies were identified and replaced. Off-gas activity upon startup was significantly less; however, by 5/7/75, off-gas levels again had increased such that administrative limits on reactor power were again initiated. Ten of the positively identified leaker assemblies were inspected by underwater TV and visible cr[illegible] were observed in several corner fuel pins of the assemblies. In September 1975, another refueling was performed to replace all remaining initial 7x7 fuel in the core. Fuel sipping operations identified 77 leaker assemblies. All leakers to date were of the initial 7x7 fuel assemblies. The fuel pin failures are believed to be caused by the hydride and pellet-clad interaction mechanisms.
Nine Mile Point 1	The unit was shutdown for refueling and overhaul on September 13, 1975. During in-core sipping operations, 154 control rod cells, including 25 resipped, were examined. Fifty fuel assemblies were identified as having fuel rods with perforated cladding. One-hundred

Table 5-2 (Cont'd)

REACTOR	EXPERIENCE
	ninety-four original fuel bundles along with 6 GEA fuel bundles were removed from the core.
Oyster Creek	The annual refueling/maintenance outage ran from March 29 to May 25, 1975. In-core sipping at the end of Cycle IV identified 19 failed Type 1 fuel assemblies, of which only 2 had exposures below the batch cycle discharge exposure. Cycle IV core contained 184 Type 1 fuel assemblies. A further refueling, together with the main condenser retubing, commenced on December 26, 1975.
Peach Bottom 2	In response to notification of possible LPRM vibration and associated channel box damage, power and flow were limited. On October 31, the unit was removed from service to plug the core plate bypass flow holes. All fuel bundle channels adjacent to instrument strings were inspected and rechanneled as necessary. There were indications of channel corner wear, but no through-wall holes or missing pieces. The unit was back on line in early December. Except for the channel wear problem, fuel performance was good during the year.
Peach Bottom 3	Fuel performance was good during the year. At the end of December, the unit was down to inspect for channel box damage and to plug the core plate bypass flow holes. Unit power and flow were limited more than half the year due to possible LPRM vibration and associated channel box damage
Pilgrim 1	No irradiated examinations were performed during the year. Reactor power was frequently reduced during the year to limit effluent release.
Quad Cities 1	At the end of 1975, fuel performance appeared good. No significant off-gas increase was observed during the year.

Table 5-2 (Cont'd)

REACTOR	EXPERIENCE
Quad Cities 2	During the first refueling outage, which lasted from January through April 1975, the entire core (724 fuel assemblies) was wet sipped out-of-core. Seventy-four assemblies were identified as leakers. These and 70 other high exposure bundles were replaced by the new 8x8 type assemblies. There was no indication of flow induced in-core monitor vibration wear on any of the channels inspected. On May 22, 1975, after a rapid ramp to power, a sudden increase in off-gas was experienced. Therefore, on October 3, 1975, the plant was shutdown once more for fuel maintenance. Seven-hundred and eight fuel assemblies were in-core wet sipped, followed by the out-of-core wet sipping of 121 assemblies made up of periphery assemblies and assembly resips. Ninety-four assemblies were identified as leakers (all of the old 7x7 design). At the end of the year, the unit was operating without derating due to off-gas. Failures during the year were attributed to hydriding and pellet-clad interactions.
Vermont Yankee	The plant was shutdown on August 7, 1975 to inspect the channels from fuel bundles next to in-core instruments and to plug the in-core instrumentation coolant holes in the lower core support plate. No bundles were replaced, but 36 bundles exchanged locations with symmetrical bundles fr[illegible] [illegible]ther qu[illegible]rants due to restrictions on the l[illegible]cati[illegible] of wo[illegible] channels. Startup commen[illegible]d on A[illegible]st 2[illegible] 1975. During startup, the in-core ins[illegible]rumen[illegible]ation was checked for vibration at [illegible] [illegible]alues of core flow. Vibration was negligible.

The effects of PCI can also be reduced by lowering the thermal duty of the fuel rods; e.g., increasing the number of fuel rods per fuel assembly and decreasing the fuel loading per fuel rod. GE's latest fuel assembly design incorporating an 8x8 array of fuel rods (as compared to the previous 7x7 array) should reduce the number of PCI failures. However, such failures should be expected to continue until the older fuel is phased out.

5.2.3 Fuel Channel Box Wear

This problem is discussed in more detail in Section 4.3. The problem was discovered during 1975 in one class of boiling water reactor, the BWR-4, by the operator of a foreign reactor. The problem is caused by instrument tube vibration against the adjacent fuel channel boxes, causing wear and eventual perforation of the boxes. If uncorrected, damage to the fuel rods could result. The problem varied from plant to plant depending upon the time the boxes are exposed to the tube vibration and the magnitude of tube vibration. The short term solution to the problem included reduction in primary coolant flow, plugging the bypass holes in the lower core plate, and inspecting and replacing damaged fuel channel boxes with indications of excessive wear. A summary of the actions taken by the plants affected is described in Table 5-2.

5.3 PWR Fuel Experience

A summary of PWR fuel failure experience in 1974 and 1975 is listed in Tables 5-3 and 5-4, respectively. Overall experience for both years indicated relatively good performance in that most PWR's had few fuel failures.

5.3.1 Pellet-Clad Interaction

Some failures occurred due to the PCI mechanism, generally as the result of rapid changes in power level. Mitigation of the effects of this mechanism is similar to that described earlier for BWR's.

5.3.2 Densification

The effects of densification (axial gaps in the fuel column causing local power peaking and possible cladding collapse) have been considerably reduced by development of conservative models to account for the effects of densification. Changes in fuel rod fabrication techniques (e.g., use of pressurized rods and fuel pellets sintered at higher temperatures) also appear to be effective. During both 1974 and 1975, some densification induced power spikes were observed, but the spikes were relatively small.

Table 5-3

SUMMARY OF PWR FUEL FAILURE EXPERIENCE IN 1974

REACTOR	EXPERIENCE
Fort Calhoun 1	Radioactivity levels indicated a failure rate < 0.01%.
H. B. Robinson 2	During Cycle 2, the number of blips per monitored assembly (an indication of densification) increased to about 2.0. Increased iodine activity indicated some fuel clad failures.
Kewaunee 1	On 9/4/74, primary coolant activity level increased suddenly, due to a leaking rod. No apparent cause of the leaking rod could be ascertained. There were no indications of clad creep.
Maine Yankee	Plant shutdown on 6/28/74 (earlier than anticipated) due to high iodine release rates. All assemblies were sipped and 43 leakers were identified. Most likely causes were pellet-clad interactions and/or hydriding. In addition, some problems were identified concerning fuel pin bowing and spacer-grid damage. In the last two months of 1974, a factor of 10-15 increase in I-131 levels were experienced indicating some fuel failures. Licensee planned to reduce power level until scheduled refueling in May 1975.
Oconee 1	Coolant activity levels observed correspond to fission gas escape through small pinholes. During the Fall refueling, visual examinations and physical measurements were made on a few fuel assemblies; no defective assemblies were detected of those examined.
Palisades	Radioactivity levels indicate a failure rate < 0.1%.
Point Beach 1	During the June startup of Cycle 3, higher than expected main coolant radioactivity indicated some rod defects. Cause was attributed to pellet-clad interactions in conjunction with a rapid rate of reactor power increase after the refueling shutdown.

Table 5-3 (Cont'd)

REACTOR	EXPERIENCE
Point Beach 2	During the Fall refueling, the assemblies which were to be reinserted were visually inspected. No defective assemblies were observed.
Surry 1	During 1974, I-131 activity level in primary coolant indicated 2-4 defective fuel rods. Densification induced power spikes observed in all regions of the core. The number is increasing, but all spikes are relatively small. During the Fall refueling, visual (binocular) inspection was performed on all 157 assemblies and TV inspection was performed on 44 assemblies; no defects were observed. Very little crud was present. Some sight bowing was observed in some assemblies.
Surry 2	Radioactivity levels indicate about 1-2 defective fuel rods Densification induced power spikes observed in all regions of the core.
Turkey Point 3	During the Fall refueling, 157 assemblies were visually (binocular) inspected. Sipping was not done due to equipment problems. Some bowing of fuel rods was observed.
Yankee (Rowe)	During refueling shutdown starting 5/10/74, 12 Core X predetermined fuel assemblies were given close surveillance. Some crud, discoloration, and abrasions were noted. No failures were detected. The 12 assemblies were considered acceptable for continued second and third cycles of operation in Core XI.

Table 5-4

SUMMARY OF PWR FUEL FAILURE EXPERIENCE IN 1975

REACTOR	EXPERIENCE
D. C. Cook 1	At the end of 1975, coolant activity indicates only a nominal number of fuel rod failures. No fuel inspections were performed during the year.
Fort Calhoun	Cycle I terminated February 7, 1975. Inspections and measurements of batches A, B, and C fuel showed no abnormalities or indications of fuel wear or failure. Neither excessive crud buildup nor bundle distortions due to growth were observed. At the end of the year, fuel performance indicated close agreement between predicted core parameters and measured values.
H. B. Robinson 2	Cycle 3 ended on October 31, 1975. The agreement between predicted and actual core parameters during Cycle 3 was excellent. Cycle 4 operation commenced on December 7, 1975.
Indian Point 1	The unit remained in shutdown the entire year in compliance with the NRC Interim Acceptance Criteria for ECCS. The reactor was defueled commencing in the beginning of December 1975, when the first assembly was removed from the core. By the end of the year, 68 out of 120 fuel assemblies were transferred from the core to the Fuel Handling Building.
Oconee 1	In April, during post-irradiation examination of once-burned Oconee-1 fuel assemblies, a defected fuel rod was found in assembly 1A10 and in assembly 1A19. No other defected fuel rods were observed. The two assemblies were permanently discharged from the core.
Palisades	Based on primary chemistry data, about 0.05% failed fuel was indicated. The plant was shut down for refueling and steam generator eddy current testing on December 20, 1975.

Table 5-4 (Cont'd)

REACTOR	EXPERIENCE
Point Beach 1	Refueling commenced 11/23/75. During refueling operations, spent fuel element DO-3 in core position K-6 was found to be damaged. Fretting wear had occurred to the cladding of 3 fuel rods at several of the 7 grid assembly locations along the length of the fuel rods. Holes through the clad, clad splitting and loose fuel pellet fragments were observed. An 11 inch section of one rod was broken out of the rod during the handling of this element. The cause of the fretting wear was attributed to water impingement through the baffle plate. The damaged assembly was replaced. Restrictions on rate of power escalation following a cold shutdown of the reactor will be followed; also, careful monitoring of reactor coolant activity during startup and later operation will be done. Investigation will continue to determine a final fix on the potential problem.
Rancho Seco	At the end of 1975, activity release data did not indicate any fuel cladding failures.
R. E. Ginna 1	Refueling commenced March 10, 1975. Visual (underwater TV and binocular) examinations revealed no failed fuel.
San Onofre 1	Refueling commenced 3/14/75. Fifty-two new fuel assemblies (containing prepressurized fuel rods) were installed.
Surry 1	Primary coolant activity indicates about 15 to 20 fuel defects at the end of 1975. There are 8 confirmed and 8 suspected locations with power spikes due to pellet gap formation; however, the spikes are relatively small.
Surry 2	Primary coolant activity indicates about 3 to 5 fuel defects at the end of 1975. There is 1 confirmed location with a power spike due to pellet gap formation; however, the spike is relatively small.

Table 5-4 (Cont'd)

REACTOR	EXPERIENCE
Turkey Point 4	On April 26, 1975, during fuel reloading for Cycle 2, a Region 3 assembly (No. P41) was damaged. When the spent fuel pit (SFP) side lifting frame was upended, the lifting frame struck the fuel assembly and pushed it into the lifting frame pulley mounted on the west wall of the SFP transfer canal. Underwater TV showed that the first grid above the bottom nozzle was damaged and the seventh and eighth fuel rods from the southwest corner of the fuel assembly were distorted. There was no evidence of a breach of fuel cladding integrity. The fuel assembly was replaced, together with its three symmetric assemblies.

5.4 Summary

Fuel performance in 1974 and 1975 ranged from fair to good. Power levels at several plants had to be reduced because of fuel failures, to limit radioactive gaseous releases to acceptable levels.

Fuel failures continued to be caused by internal hydriding, pellet-clad interactions and corrosion. Even so, failure rates were generally below 1%. Even though failures had some effect on reactor operations, there were no adverse effects on the health and safety of the public.

6.0 RADIATION EXPOSURE AND RADIOACTIVE RELEASES

6.1 Occupational Radiation Exposure

Two NRC reports[1] have been recently published that compiled and summarized annual occupational radiation exposures at commercial light water cooled power reactors. The following data and observations are taken from those reports.

Table 6-1 and Table 6-2 summarize the radiation exposure information reported for personnel whose annual exposures exceeded 100 mrems by those plants that had been commercially operating for at least one full year as of December 31, 1974 and December 31, 1975, respectively. The average exposure per individual was 0.79 rems in 1975, a slight decrease from the average value of 0.83 rems reported for 1974. However, the average number of personnel per reactor increased to 578 as compared to the 1974 average number of 515 workers.

Comparison of the more recent data with that of previous years indicates that occupational radiation exposures, in terms of man-rem per reactor-year continue to rise. In 1969 when there were 7 reactors that had been operating for at least one year, the average man-rem/reactor-year was 178. In 1974 the figure had risen to 427 and in 1975 it was 457. Based on data submitted by about 50% of the power reactors in the format described in Reg. Guide 1.16, the percentage of the cumulative dose (man-rems) attributed to routine and special maintenance continues to exceed 65%. A further breakdown is shown in Table 6-3.

6.2 Radioactive Releases

Releases of radioactive effluents from nuclear power generating facilities are restricted by Title 10, Code of Federal Regulations, Part 20 (10 CFR Part 20, "Standards for Protection Against Radiation"), and by limits established in the technical specifications of each plant. Paragraph (a)(2) of 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors", of 10 CFR Part 50 provides that technical specifications for each licensee will include a requirement that the licensee submit a report to the Commission within 60 days after January 1 and July 1 of each year which specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 6 months of operation.

Table 6-4 summarizes the airborne noble and halogen gaseous releases as well as the electrical power generation (net electrical MWH_{th}) for each operating plant. Table 6-5 presents data showing the tritium and mixed fission and activation products released in the liquid effluent. The data are presented for the calendar years 1974 and 1975.

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[1]"Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974," NUREG-75/032 and "Occupational Radiation Exposures at Light Water Cooled Power Reactors," NUREG-0109.

TABLE 6-1

Annual Radiation Exposures to Plant Workers 1974

Plant Name	No. of Personnel Exposed (No. of Man-rems)			Average Exposure Rem/Person
	Licensee	Contractor	Totals	
Big Rock Point	- -	- -	281 (276)	0.98
Dresden 1, 2, 3	1,274 (1,605)	318 (57)	1,594 (1,662)	1.04
R.E. Ginna 1	- -	- -	884 (1,224)	1.38
Haddam Neck	- -	- -	550 (201)	0.37
Humboldt Bay 3	75 -	221 -	296 (318)	1.07
Indian Point 1,2	905 -	114 -	1,019 (910)	0.89
LaCrosse	94 (133)	21 (6)	115 (139)	1.21
Maine Yankee	135 (232)	485 (188)	620 (420)	0.68
Millstone Point 1	- -	- -	2,477 (1,430)	0.58
Monticello	365 (258)	477 (91)	842 (349)	0.41
Nine Mile Point	277 (545)	463 (279)	740 (824)	1.11
*Oconee 1	591 (373)	253 (144)	844 (517)	0.61
Oyster Creek 1	589 (822)	346 (162)	935 (984)	1.05
Palisades	- -	- -	774 (627)	0.81
Pilgrim 1	- -	- -	454 (415)	0.91
Point Beach 1, 2	- (214)	- (81)	400 (295)	0.74
*Prairie Island 1	[illegible]4 (13)	56 (5)	150 (13)	0.12
Quad Cities 1, 2	190 (446)	488 (36)	678 (482)	0.71
H.B. Robinson	- -	- -	853 (672)	0.78
San Onofre 1	- -	- -	219 (71)	0.32
Surry 1, 2	- -	- -	1,715 (884)	0.52
Turkey Point 3, 4	- (252)	- (202)	794 (454)	0.57
Vermont Yankee	- (113)	- (103)	357 (216)	0.61
Yankee Rowe	94 (106)	149 (99)	243 (205)	0.84
*Zion 1	219 (43)	87 (13)	306 (56)	0.18

*1974 was the first full year of operation for these plants.

TABLE 6-2

Annual Radiation Exposures to Plant Workers 1975

Plant Name	No. of Personnel Exposed (No. of Man-rems)			Average Exposure Rem/Person
	Licensee	Contractor	Totals	
*Arkansas 1	- -	- -	147 (46)	0.31
Big Rock Point	- (20)	- (160)	216 (180)	0.83
*Brown's Ferry 1	- -	- -	2,380 (325)	0.14
*Cooper Station	104 (80)	71 (16)	175 (96)	0.55
Dresden 1, 2, 3	596 (1,098)	3,076 (2,111)	3,671 (3,209)	0.87
*Fort Calhoun	100 (205)	369 (93)	469 (298)	0.63
Ginna	- -	- -	558 (496)	0.89
Haddam Neck	- -	- -	795 (669)	0.84
Humboldt Bay	73 (222)	230 (110)	303 (332)	1.10
Indian Point 1,2	407 (479)	73 (42)	480 (626)	1.30
*Kewaunee	41 (14)	23 (11)	54 (25)	0.50
LaCrosse	94 (133)	21 (6)	165 (234)	1.42
Maine Yankee	159 (150)	418 (197)	577 (347)	0.60
Millstone Point 1	- -	- -	2,587 (2,022)	0.78
Monticello	- -	- -	1,353 (1,353)	1.0
Nine Mile Point	320 (478)	329 (203)	649 (681)	1.04
Oconee 1 ,2*,3*	429 (374)	112 (83)	541 (457)	0.84
Oyster Creek	- (863)	- (269)	1,210 (1,132)	0.94
Palisades	- -	- -	474 (292)	0.62
*Peach Bottom 2, 3	- -	- -	971 (228)	0.24
Pilgrim	- (360)	- (384)	473 (744)	1.60
Point Beach 1, 2	- -	- -	339 (456)	1.30
Prairie Island 1,2*	- -	- -	477 (123)	0.26
Quad Cities 1, 2	554 (793)	1,418 (592)	1,972 (1,385)	0.70
H.B. Robinson	- -	- -	849 (1,142)	1.35
San Onofre 1	- -	- -	424 (292)	0.75
Surry 1, 2	- (549)	- (1,000)	808 (1,549)	1.91
*Three Mile Island	- (62)	- (21)	168 (83)	0.49
Turkey Point 3, 4	- (317)	(558)	1,175 (875)	0.74
Vermont Yankee	83 (82)	164 (57)	247 (139)	0.56
Yankee Rowe	76 (60)	134 (78)	210 (138)	0.66
Zion 1,2*	495 (72)	938 (45)	1,433 (117)	0.08

*1975 was the first full year of operation for these plants.

TABLE 6-3

Percentages of Personnel Dose by Work Function

Work Function	Percent of Dose 1974	1975
Reactor Operations and Surveillance	14.0%	10.8%
Routine Maintenance	45.4%	52.5%
Inservice Inspection	2.7%	2.9%
Special Maintenance	20.4%	19.0%
Waste Processing	3.5%	6.9%
Refueling	14.0%	7.7%

TABLE 6-4 AIRBORNE EFFLUENTS

TYPE	NUCLEAR POWER PLANT: NAME	NOBLE GAS 1974[a] Release (Curies)	NOBLE GAS 1975 Release (Curies)	HALOGEN 1974 Release (Curies)	HALOGEN 1975 Release (Curies)	ELECTRICAL POWER GENERATION (NET ELECTRICAL MWHe) 1974	ELECTRICAL POWER GENERATION (NET ELECTRICAL MWHe) 1975
	Big Rock Point 1	1.88(5)[b]	5.06(4)	3.55(-1)	2.67(-1)	3.4(5)	2.9(5)
	Browns Ferry 1 & 2[c]	6.40(4)	9.24(4)	4.05(1)	5.96(-1)	5.2(6)	2.8(6)
	Brunswick 2	C	1.85(2)	C	2.67(-3)	C	1.4(6)
	Cooper Station	2.00(3)	1.97(4)	3.54(0)	4.18(-1)	1.8(6)	3.9(6)
	Dresden 1	9.80(4)	5.20(5)	1.35(1)	5.70(0)	3.5(5)	7.0(5)
	Dresden 2 & 3	6.27(5)	3.69(5)	3.80(1)	1.17(1)	6.6(6)	5.1(6)
	Duane Arnold 1	C	1.58(3)	C	4.07(-1)	C	2.3(6)
	J. A. Fitzpatrick	C	4.08(3)	C	1.77(-2)	C	2.2(6)
	E. I. Hatch	C	2.70(2)	C	6.42(-3)	C	3.1(6)
	Humboldt Bay 3	5.72(5)	2.97(5)	1.70(0)	1.47(0)	3.8(5)	3.8(5)
BWR	Lacrosse	4.90(4)	5.71(4)	6.33(-2)	1.32(-1)	3.3(5)	2.6(5)
	Millstone Point 1	9.12(5)	2.97(6)	3.18(0)	6.28(1)	3.6(6)	3.9 (6)
	Monticello	1.48(6)	1.55(5)	2.88(1)	1.52(1)	2.9(6)	2.9(6)
	Nine Mile Point 1	5.58(5)	1.30(6)	<2.55(0)	5.88(0)	3.3(6)	3.0(6)
	Oyster Creek	2.79(5)	2.06(5)	2.33(1)	4.13(0)	3.7(6)	3.1(6)

TABLE 6-4 AIRBORNE EFFLUENTS (Continued)

TYPE	NUCLEAR POWER PLANT: NAME	NOBLE GAS 1974[a] Release (Curies)	NOBLE GAS 1975 Release (Curies)	HALOGEN 1974 Release (Curies)	HALOGEN 1975 Release (Curies)	ELECTRICAL POWER GENERATION (NET ELECTRICAL MWHe) 1974	ELECTRICAL POWER GENERATION (NET ELECTRICAL MWHe) 1975
	Peach Bottom 2 & 3	<1.00(3)	1.30(4)	6.6(-3)	1.11(-1)	3.9(6)	1.0(7)
	Pilgrim 1	5.46(5)	4.60(4)	1.45(0)	6.92(0)	2.0(6)	2.6(6)
	Quad Cities 1 & 2	9.50(5)	1.1(5)	3.60(1)	2.91(0)	8.1(6)	6.7(6)
	Vermont Yankee	6.40(4)	4.08(3)	4.72(-1)	1.16(-1)	2.5(6)	3.6(6)
						C	
	Arkansas 1	1.96(2)	1.03(3)	5.3(-2)	7.43(-1)	5.7(5)	4.9(6)
	Calvert Cliffs 1	C	7.72(3)	C	3.56(-2)	C	4.4(6)
	Connecticut Yankee	7.00(0)	4.80(2)	4.57(-8)	8.92(-4)	4.4(6)	4.1(6)
	Cook 1	C	2.64(0)	C	1.65(-4)	C	4.5(6)
	Fort Calhoun	3.03(2)	4.29(2)	5.14(-4)	6.89(-3)	2.4(6)	2.1(6)
	H. B. Robinson 2	2.31(3)	1.17(3)	5.15(-2)	2.34(-2)	4.8(6)	4.2(6)
	Indian Point 1	6.11(2)	3.67(2)	9.44(-2)	1.04(-2)	1.2(6)	Shut Down
	Indian Point 2	5.58(3)	8.20(2)	2.88(-1)	4.00(-1)	3.3(6)	4.9(6)
	Kewaunee	3.35(3)	2.45(3)	2.87(-2)	1.99(-2)	1.6(6)	3.3(6)
	Maine Yankee	6.36(3)	4.09(3)	5.49(-2)	5.90(-3)	3.6(6)	4.5(6)

TABLE 6-4 AIRBORNE EFFLUENTS (Continued)

TYPE	NUCLEAR POWER PLANT: NAME	NOBLE GAS 1974[a] Release (Curies)	NOBLE GAS 1975 Release (Curies)	HALOGEN 1974 Release (Curies)	HALOGEN 1975 Release (Curies)	ELECTRICAL POWER GENERATION (NET ELECTRICAL MWHe) 1974	ELECTRICAL POWER GENERATION (NET ELECTRICAL MWHe) 1975
PWR	Millstone Point 2	C	ND[d]	C	4.59(-5)	C	1.3(5)
	Oconee 1, 2 & 3	1.94(4)	1.51(4)	3.21(-2)	1.07(-2)	5.5(6)	1.5(7)
	Palisades	<1.00(0)	2.61(3)	2.30(-2)	4.27(-1)	7.8(4)	2.4(6)
	Point Beach 1 & 2	9.74(3)	4.45(4)	1.27(-1)	1.87(-1)	6.7(6)	6.7(6)
	Prairie Island 1 & 2	3.58(2)	2.17(3)	6.03(-4)	2.10(-2)	1.4(6)	6.9(6)
	R. E. Ginna	7.57(2)	1.04(4)	4.47(-4)	2.69(-2)	2.1(6)	3.0(6)
	Rancho Seco	C	1.18(2)	C	1.89(-4)	C	1.3(6)
	San Onofre 1	1.78(3)	1.11(3)	2.31(-4)	2.45(-1)	3.1(6)	3.2(6)
	Surry 1 & 2	5.50(4)	8.04(3)	1.22(-1)	5.22(-2)	5.9(6)	9.0(6)
	Three Mile Island	9.16(2)	3.63(3)	3.11(-3)	9.36(-4)	2.1(6)	5.5(6)
	Turkey Point 3 & 4	4.66(4)	1.34(4)	3.45(0)	4.65(-1)	7.9(6)	8.4(6)
	Yankee Rowe	4.00(1)	2.24(1)	1.01(-3)	2.73(-3)	9.1(5)	1.2(6)
	Zion 1 & 2	2.99(3)	4.88(4)	1.53(-2)	2.17(-1)	4.7(6)	9.7(6)

[a] Source: "Radioactive Materials Released From Nuclear Power Plants, 1974, NUREG-0077," USNRC, Washington, D.C. (June 1976).

[b] Notation: $1.88(5) = 1.88 \times 10^5$.

[c] Units not in commercial operation prior to 1975.

[d] NR = Not Reported; ND = Not Detected.

TABLE 6-5 LIQUID EFFLUENTS

TYPE	NUCLEAR POWER PLANT: NAME	TRITIUM 1974[a] Release (Curies)	TRITIUM 1975 Release (Curies)	MIXED FISSION AND ACTIVATION PRODUCTS 1974 Release (Curies)	MIXED FISSION AND ACTIVATION PRODUCTS 1975 Release (Curies)
	Big Rock Point 1	5.1	5.73	1.1	2.02
	Browns Ferry 1 & 2	2.8	10.42	0.8	2.70
	Brunswick 2	C	3.20	C	1.89
	Cooper Station	1.7	8.25	1.4	1.74
	Dresden 1	18.8	0.27	6.9	0.84
	Dresden 2 & 3	22.6	54.00	33.1	0.81
	Duane Arnold 1	C	0.33	C	2.07(-3)[b]
	J. A. Fitzpatrick	C	5.03	C	5.32
	E. I. Hatch	C	6.12	C	0.06
	Humboldt Bay	31.7	20.1	4.4	3.79
	Lacrosse	115.0	127.0	13.1	14.20
	Millstone Point 1	24.1	80.30	198.0	199.00
BWR	Monticello	0.0	0.00	0.0	0.00
	Nine Mile Point	18.7	28.10	25.6	21.00
	Oyster Creek	14.1	17.87	0.7	0.41

TYPE	NUCLEAR POWER PLANT: NAME	TRITIUM 1974[a] Release (Curies)	TRITIUM 1975 Release (Curies)	MIXED FISSION AND ACTIVATION PRODUCTS 1974 Release (Curies)	MIXED FISSION AND ACTIVATION PRODUCTS 1975 Release (Curies)
	Peach Bottom 2 & 3	10.0	30.80	0.9	0.93
	Pilgrim 1	10.5	18.20	4.2	8.01
	Quad Cities 1 & 2	34.0	53.70	38.8	17.14
	Vermont Yankee	0.0	0.00	0.0	4.06(-6)
	Arkansas 1	25.6	460.00	6.5	3.11
	Calvert Cliffs 1	C	262.70	C	1.44
	Connecticut Yankee	2240.0	5670.00	2.2	1.20
	Cook 1	C	56.40	C	0.26
	Fort Calhoun	124.0	110.60	2.3	0.36
	H. B. Robinson 2	449.0	624.00	2.5	0.45
	Indian Point 1	684.0	287.00	2.9	1.30
	Indian Point 2	47.9	79.36	4.2	4.93
	Kewaunee	92.4	277.00	0.4	0.72
PWR	Maine Yankee	219.0	177.30	4.0	3.21
	Millstone Point 2	C	7.60	C	0.02
	Oconee 1, 2 & 3	350.0	3550.00	1.9	5.05
	Palisades	8.1	41.59	5.9	3.45

TABLE 6-5 LIQUID EFFLUENTS (Continued)

NUCLEAR POWER PLANT: TYPE	NAME	TRITIUM 1974[a] Release (Curies)	TRITIUM 1975 Release (Curies)	MIXED FISSION AND ACTIVATION PRODUCTS 1974 Release (Curies)	MIXED FISSION AND ACTIVATION PRODUCTS 1975 Release (Curies)
	Point Beach 1 & 2	833.0	885.00	0.2	2.34
	Prairie Island 1 & 2	142.0	763.00	<0.1	0.45
	R. E. Ginna	195.0	260.90	0.1	0.42
	Rancho Seco	C	132.00	C	2.87(-4)
	San Onofre 1	3810.0	4000.00	5.0	1.22
	Surry 1 & 2	245.0	442.0	3.8	9.27
	Three Mile Island	130.0	463.0	1.3	0.07
	Turkey Point 3 & 4	580.0	797.0	1.6	3.07
	Yankee Rowe	314.0	247.0	<0.1	0.02
	Zion 1 & 2	2.3	40.0	<0.1	0.01

[a]Source: "Radioactive Materials Released From Nuclear Power Plants, 1974, NUREG-0077," USNRC, Washington, D. C. (June 1976).

[b]Notation: $2.0(-3) = 2.0 \times 10^{-3}$.

[c]Unit not in commercial operation prior to 1975.

There was a wide variation in the amounts of radioactivity released due to differences in fuel performance, power produced and the extent to which effluent treatment systems were used and improved. The bulk of the radioactivity releases were in the form of noble gases from boiling water reactors.

In all cases, the radioactivity released from nuclear power plants during 1974 and 1975 was only a small fraction of the permissible limits set forth in applicable regulations.

APPENDIX A - GLOSSARY

Definitions

- Abnormal Occurrences and Unusual Events

  See Section 4.3

- Commercial Operation - plant status, declared by the utility-owner when unit is available for the regular production of electricity.

- Design Electrical Capacity (Net) - the nominal net electrical output of the plant (unit) used for the purpose of plant design.

- Forced Outage - the occurrence of an equipment malfunction, operational error or plant condition which requires or causes a plant shutdown.

- Net Electrical Output - the gross electrical output measured at the output terminals of the main generator(s) less the normal station service load(s) and transformer losses.

- Outage - when the main generator is not connected to the output transmission facilities (off-line).

- Outage Duration - the length of time the main generator is off-line during an outage. (When the outage duration was not referenced by licensees, it was estimated from power production graphs, chronologies and outage and maintenance information.)

- Plant Age - the elapsed time from the date of first electrical generation through December 31.

- Plant Availability Factor (PAF) - the quotient of time (hours) that the plant was operated with the main generator on-line during a given period, divided by the total time (hours) in the given period expressed as a percent.

  $$PAF = \frac{\text{Time (Hrs) Generator On-Line}}{\text{Time period (Hrs)}} \times 100$$

  (EEI defines service factor in the same manner. EEI has a definition for Plant Operating Availability which takes cognizance of the Reserve Shutdown Hours when the plant is shut down for economic reasons, but still considered available to be used.)

- Plant Capacity Factor (PCF/MWE) - the quotient of the net electrical output produced by the plant in a given period divided by the net electrical output the plant would have produced had it been operated at its design electrical capacity (net) for the given period, expressed as a percent.

$$PCF/MWE = \frac{\text{Net Electrical Output (MWHe)}}{\text{Design Electrical Capacity (Net) x Time (Hrs)}} \times 100$$

PCF/MDC - utilizes maximum dependable capacity, the maximum dependable main unit capacity, winter or summer, whichever is smaller, rather than using the design electrical capacity.

- Scheduled Outage (Planned) - the removal of the main generator from service for plant activities normally planned in advance. The activities include refueling, periodic inspections, major equipment preventive maintenance, reactor operator training and examinations, and plant modifications.

APPENDIX B

Summary of Plant Operating Experience

Data sheets for each plant are included in this appendix. Information is provided on plant operating and outage statistics, highlights, and details on each outage.

Symbols used in the tables are as follows: Under TYPE of outage, F is used for Forced and S is used for Scheduled. Under CAUSE, the following symbols were used:

- A - Equipment failure
- B - Maintenance or testing
- C - Refueling
- D - Regulatory restriction
- E - Operator training and license exams
- F - Administrative
- G - Operational error
- H - Other

Under method of shutdown, the symbols used are: 1 - Manual, 2 - Manual scram, and 3 - Automatic scram.

The system and component classifications used are defined in Appendix B-1 and B-2.

- B-1 System Description
- B-2 Component Types
- B-3 Individual Plant Summaries - 1974
- B-4 Individual Plant Summaries - 1975

Appendix B-1

System Description	Code
Reactor	RX
Reactor Vessel Internals	RA
Reactivity Control Systems	RB
Reactor Core	RC
Reactor Coolant System & Connected Systems	CX
Reactor Vessels & Appurtenances	CA
Coolant Recirculation Systems & Controls	CB
Main Steam Systems & Controls	CC
Main Steam Isolation Systems & Controls	CD
Reactor Core Isolation Cooling Systems & Controls	CE
Residual Heat Removal Systems & Controls	CF
Reactor Coolant Cleanup Systems & Controls	CG
Feedwater Systems & Controls	CH
Reactor Coolant Pressure Boundary Leakage Detection Systems	CI
Other Coolant Subsystems & Their Controls	CJ
Engineered Safety Features	SX
Reactor Containment Systems	SA
Containment Heat Removal Systems & Controls	SB
Containment Air Purification & Cleanup Systems & Controls	SC
Containment Isolation Systems & Controls	SD
Containment Combustible Control Systems & Controls	SE
Emergency Core Cooling Systems & Controls	SF
Control Room Habitability Systems & Controls	SG
Other Engineered Safety Feature Systems & Their Controls	SH
Instrumentation and Controls	IX
Reactor Trip Systems	IA
Engineered Safety Feature Instrument Systems	IB
Systems Required for Safe Shutdown	IC
Safety Related Display Instrumentation	ID
Other Instrument Systems Required for Safety	IE
Other Instrument Systems Not Required for Safety	IF
Electric Power Systems	EX
Offsite Power Systems & Controls	EA
AC Onsite Power Systems & Controls	EB
DC Onsite Power Systems & Controls	EC
Onsite Power Systems & Controls (Composite AC & DC)	ED

System Description	Code
Emergency Lighting Systems & Controls	EF
Other Electric Power Systems & Controls	EG
Fuel Storage and Handling Systems	FX
New Fuel Storage Facilities	FA
Spent Fuel Storage Facilities	FB
Spent Fuel Pool Cooling & Cleanup Systems & Controls	FC
Fuel Handling Systems	FD
Auxiliary Water Systems	WX
Station Service Water Systems & Controls	WA
Cooling Systems for Reactor Auxiliaries & Controls	WB
Demineralized Water Make-up Systems & Controls	WC
Potable & Sanitary Water Systems & Controls	WD
Ultimate Heat Sink Facilities	WE
Condensate Storage Facilities	WF
Other Auxiliary Water Systems & Their Controls	WG
Auxiliary Process Systems	PX
Compressed Air Systems & Controls	PA
Process Sampling Systems	PB
Chemical, Volume Control, & Liquid Poison Systems & Controls	PC
Failed Fuel Detection Systems	PD
Other Auxiliary Process Systems & Their Controls	PE
Other Auxiliary Systems	AX
Air Conditioning, Heating, Cooling & Ventilation Systems & Controls	AA
Fire Protection Systems & Controls	AB
Communication Systems	AC
Other Auxiliary Systems & Their Controls	AD
Steam and Power Conversion Systems	HX
Turbine-Generators & Controls	HA
Main Steam Supply System & Controls (Other Than CC)	HB
Main Condenser Systems & Controls	HC
Turbine Gland Sealing Systems & Controls	HD
Turbine Bypass Systems & Controls	HE

System Description	Code
Circulating Water Systems & Controls	HG
Condensate and Feedwater Systems & Controls (Other Than CH)	HH
Steam Generator Blowdown Systems & Controls	HI
Other Features of Steam & Power Conversion Systems (Not included eleswhere)	HJ
Radioactive Waste Management Systems	MX
Liquid Radioactive Waste Management Systems	MA
Gaseous Radioactive Waste Management Systems	MB
Process & Effluent Radiological Monitoring Systems	MC
Solid Radioactive Waste Management Systems	MD
Radiation Protection Systems	BX
ARea Monitoring Systems	BA
Airborne Radioactivity Monitoring Systems	BB

Appendix B-2

COMPONENT TYPES

Component Type	Component Type Includes:
Accumulators	Scram Accumulators Safety Injection Tanks Surge Tanks
Air Dryers	
Annunciator Modules	Alarms Bells Buzzers Claxons Horns Gongs Sirens
Batteries & Chargers	Chargers Dry Cells Wet Cells Storage Cells
Blowers	Compressors Gas Circulators Fans Ventilators
Circuit Closers/Interrupters	Circuit Breakers Contactors Controllers Starters Switches (other than sensors) Switchgear
Control Rods	Poison Curtains
Control Rod Drive Mechanisms	
Demineralizers	Ion Exchangers
Electrical Conductors	Bus Cable Wire
Engines, Internal Combustion	Butane Engines Diesel Engines Gasoline Engines Natural Gas Engines Propane Engines

Component Type	Component Type Includes:
Filters	Strainers Screens
Fuel Elements	
Generators	Inverters
Heaters, Electric	
Heat Exchangers	Condensers Coolers Evaporators Regenerative Heat Exchangers Steam Generators Fan Coil Units
Instrumentation and Controls	
Mechanical Function Units	Mechanical Controllers Governors Gear Boxes Varidrives Couplings
Motors	Electric Motors Hydraulic Motors Pneumatic (Air) Motors Servo Motors
Penetrations, Primary Containment	Air Locks
Pipes, Fittings	
Pumps	
Recombiners	
Relays	
Shock Suppressors and Supports	
Transformers	
Turbines	Steam Turbines Gas Turbines Hydro Turbines
Valves	Valves Dampers

Component Type	Component Type Includes:
Valve Operators	
Vessels, Pressure	Containment Vessels Drywells Pressure Suppression Pressurizers Reactor Vessels

APPENDIX B - 3

INDIVIDUAL PLANT SUMMARIES

1974

BIG ROCK POINT

I. Summary

Description

Location: Big Rock Point, Michigan
Docket No: 50-155
Reactor Type: Boiling Water
Capacity (MWe-Net): 72
Commercial Operation: 3/63
Plant Age: 12.1 Years

Performance

Net electrical energy generated (MWH):	337,541
Unit availability factor (%):	70.3
Unit capacity factor (%): (Using Design MWe)	54.3

Outages

Total No.	3		
Forced	2		
Scheduled	1		
Total:	2,600	Hours,	29.7%
Forced	1,044	Hours,	11.9%
Scheduled	1,556	Hours,	17.8%

Cause: Equipment Failure 3
Refueling 2
Method of shutdown:
Manual 3

II. Highlights

A. General :

The plant was base loaded at 53 MWe during the period January — June except for a one-week run at 71 MWe. A refueling outage began on March 23 and lasted until May 5. On June 2 a forced outage due to steam leaks and stuck control rod drives was extended for refueling through July 26. The unit was then base loaded at 63 MWe and operated consecutively for 157 days.

B. Outages :

1. Forced: Two forced outages consumed 1044 hours. The first 253 hrs of the year were a continuation of a 1973 shutdown, in which the emergency condenser was repaired. Seven hundred ninety two hours were consumed for repair of steam leaks and stuck control rod drives.

2. Scheduled: Refueling consumed 1556 hours.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	253	F	Repair emergency condenser. Modified baffle plates in the inlet water box.	A	1	Reactor Coolant (CE)	Heat Exchangers
2)	3/23	1044	S	Refueling	C	1	Reactor (RC)	Fuel Elements
3a)	6/2	48	F	Steam leak on 3 in. drain line from HP section of turbine to HP feedwater heater.	A	1	Steam and Power (HH)	Pipes, Fittings
3b)	6/5	744	F	Continuation of 3a. Control rod drives stuck Maintenance also performed.	A		Reactor (RB)	Control Rod Drives
3C)	7/6	511	S	Continuation of 3a and 3b. Refueling.	C		Reactor (RC)	Fuel Elements

BROWNS FERRY 1

I. Summary

Description

Location: Decatur, Alabama
Docket No: 50-259
Reactor Type: BWR
Capacity (MWe-Net): 1,065
Commercial Operation: 8/1/74
Plant Age: 1.2 Years

Performance

Net electrical energy generated (MWH):	5,168,631
Unit availability factor (%):	74.5
Unit capacity factor (%): (Using Design MWe)	55.4

Outages

Total No.	36		
Forced	26		
Scheduled	10		
Total:	2,138	Hours,	25.5%
Forced	1,369	Hours,	15.6%
Scheduled	769	Hours,	9.9%

Cause:

Equipment Failure	18
Maintenance/Test	12
Operational Error	5
Tornado	1

Method of Shutdown:

Manual	12
Manual Scram	6
Auto Scram	16

II. Highlights

A. General:

The reactor operated at a nominal power of 80% for the first half of the year, and for the second half of the year the unit operated at a nominal 97% of power.

B. Outages:

1. Forced: Twenty-six forced outages occurred during the year consuming 1369 hours. The longest outage consumed 445 hours to replace relief valves and HPCI valves and to perform general maintenance.

2. Scheduled: Ten scheduled outages consumed 769 hours. The longest scheduled outage was for 365 hours to perform general maintenance on feedwater heaters and the residual heat exchangers.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/2	14	F	Faulty underfrequency relay caused breaker trip.	A	3	Electric Power (EA)	Relays
2)	1/18	112	S	Inspection of reactor pressure relief system.	B	2	Reactor Coolant (CA)	Valves
3)	1/24	31	F	Reactor protection system MG set tripped.	A	2	Instrumentation & Controls (IA)	Generators
4)	1/29	6	S	Repair traversing in-core probe (TIP).	B	1	Instrumentation & Controls (ID)	Instrumentation & Controls
5)	2/12	136	F	Failure to provide adequate ventilation to steam tunnel.	G	3	Other Auxiliary (AA)	NA
6)	2/28	101	S	Relief valve maintenance and replacement.	B	1	Reactor Coolant (CA)	Valves
7)	3/11	13	S	Work on electrical penetration in drywell.	B	1	Engineered Safety (SA)	Penetrations, Primary Containment
8)	3/15	24	F	Test-reactor low water level. Cond. booster pump trip.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	3/24	7	F	MSIV closure — (steam low pressure) — press. switch review.	A	1	Reactor Coolant (CD)	Circuit Closers
10)	3/26	10	F	Condensate low vacuum — operator error.	G	1	Steam & Power (HH)	NA
11)	3/27	34	S	Test-turbine control valve closure.	B	1	Reactor Coolant (CC)	Valves
12)	4/1	14	F	MSIV closure — main steam line low pressure switches apparent cause.	A	3	Reactor Coolant (CD)	Circuit Closers
13)	4/3	19	F	Lost 500 KV lines during tornado.	H	3	Electric Power (EA)	Electrical Conductors
14)	4/3	226	F	HPCI system and main condenser repair.	B		Steam & Power (HC)	Heat Exchangers
15)	4/15	18	F	MSIV closure — pressure switches apparent cause.	A	1	Reactor Coolant (CD)	Circuit Closers
16)	4/17	2	S	Loss of condensate booster suction. Demineralizer problems.	B	1	Steam & Power (HH)	Demineralizers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	4/27	21	F	Lost condensate booster pump. Repaired flow control system.	A	3	Steam & Power (HH)	Instrumentation & Controls
18)	5/5	10	F	Reactor low water level — vessel level controller adjusted.	A	3	Reactor Coolant (CH)	Circuit Closers
19)	5/6	15	F	Reactor low water level RFP suction valve failed to open.	A	3	Reactor Coolant (CH)	Valves
20)	5/7	445	F	Relief valve and HPCI valve maintenance.	A	1	Engineered Safety (SF)	Valves
21)	5/26	58	F	HPCI valve maintenance.	A	1	Engineered Safety (SF)	Valves
22)	6/2	15	F	EHC oil line leak on main steam stop valve.	A	2	Reactor Coolant (CC)	Pipes, Fittings
23)	6/6	24	F	APRM high flux trip. Recirculation master controller readjusted.	A	3	Reactor Coolant (CB)	Circuit Closers
24)	6/9	13	S	Test-turbine stop valve closure.	B	3	Reactor Coolant (CC)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
25)	6/21	211	F	Main condenser work and steam line welding.	A	3	Steam & Power (HC)	Pipes, Fittings
26)	8/1	10	F	Loss of control air.	A	2	Auxiliary Process (PA)	Air Dryers
27)	8/20	11	F	Control valve closure — operator switching error.	G	3	Reactor Coolant (CC)	NA
28)	8/25	6	F	MSIV closure — operator error during test.	G	3	Reactor Coolant (CD)	NA
29)	8/26	7	F	MSIV closure.	A	3	Reactor Coolant (CD)	Valves
30)	8/28	7	F	EHC oil press. spike. Recalibrated pressure switch.	A	3	Reactor Coolant (CC)	Instrumentation & Controls
31)	9/5	11	F	Inadvertent movement of turbine oil tank low level switch.	G	3	Steam & Power (HA)	NA
32)	9/8	1	S	Turbine overspeed trip test.	B		Steam & Power (HA)	Mechanical Function Units

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
33)	9/19	365	S	General maint. — feed-water HTR., RHR heat Exch., and pipe hangers.	B	2	Reactor Coolant (CH)	Heat Exchangers
34)	10/6	12	F	EHC oil leak.	A	2	Reactor Coolant (CC)	Pipes, Fittings
35)	10/9	7	F	Reactor low water level. Feedwater heater isolation.	A	3	Reactor Coolant (CH)	Valves
36)	11/18	122	S	Maintenance — HPCI modification and inspections.	B	1	Engineered Safety (SF)	Shock Suppressors

I. Summary

Description	Performance		Outages		
Location: Haddam Neck, Connecticut	Net electrical energy generated (MWH):	4,350,932	Total No.	14	
Docket No: 50-213			Forced	13	
Reactor Type: Pressurized Water	Unit availability factor (%):	91.2	Scheduled	1	
Capacity (MWe-Net): 575			Total:	770	Hours, 8.8%
Commercial Operation: 1/68	Unit capacity factor (%): (Using Design MWe)	91.9	Forced	743	Hours, 8.5%
Plant Age: 7.4 Years			Scheduled	27	Hours, 0.3%

Cause:

Maintenance Test	7
Operational Error	2
Equipment Failure	2
Op. Tng. & License Exam	1
Lightning Strike	1
Weather	1

Method of Shutdown:

Manual	5
Auto Scram	5

II. Highlights

A. General:

For the period January through June, plant performance was normal. Plant efficiency continues to be lower than expected because of poor condenser performance. The cleanliness factor for "A" condenser is 13.75% lower than the expected value. For the second half of the year, plant performance continued to be normal except for condenser performance. Unit efficiency and electric power generation have been slightly reduced by use of the modified "Robinson" low pressure rotors installed in 1973.

B. Outages:

1. Forced: Thirteen forced outages consumed 743 hours during the year. The longest outage consumed 660 hours to investigate and correct turbine vibration.

2. Scheduled: There was one scheduled outage for 27 hours for maintenance and operator licensing examinations.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/18	7	F	Sensing lines froze on steam line flow transmitters.	H	3	Steam & Power (HC)	Pipes, Fittings
2)	1/19	5	F	Incorrect settings of protective relays resulted in loss of off-site power and turbine trip.	G	3	Electric Power (EA)	Relays
3)	1/19	4	F	Operator shut down two circulating water pumps supplying the same condenser, satisfying trip logic.	G	3	Steam & Power (HF)	NA
4)	2/16	12	F	Secondary plant shutdown to repair leaking flange on feedwater control valve.	B	1	Steam & Power (HH)	Valves
5)	3/23	660	F	Investigate turbine vibration. Found broken blade and missing shroud. Repaired.	B	1	Steam & Power (HA)	Turbines
6)	4/20	4	F	For balance move on turbine.	B	1	Steam & Power (HA)	Turbines
7)	4/20	4	F	For balance move on turbine.	B		Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/20	4	F	For balance move on turbine.	B		Steam & Power (HA)	Turbines
9)	4/20	8	F	For balance move on [illegible]ine.	B		Steam & Power (HA)	Turbines
10)	4/20	6	F	For balance move on turbine.	B		Steam & Power (HA)	Turbines
11)	6/24	9	F	Defective filter capacitor in amplifier module for loop 2 flow transmitter caused trip.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
12)	9/8	8	F	Repaired turbine eccentricity plate.	A	1	Steam & Power (HA)	Turbines
13)	10/19	27	S	Maintenance performed concurrent with operator licensing.	E	1	Reactor (RB)	NA
14)	12/8	12	F	Lightning faulted transmission lines causing generator load rejection.	H	3	Electric Power (EA)	Electrical Conductors

COOPER

I. Summary

Description

Location: Brownville, Nebraska
Docket No: 50-298
Reactor Type: Boiling Water
Capacity (MWe-Net): 730
Commercial Operation: 7/1/74
Plant Age: 0.6 Years

Performance

Net electrical energy generated (MWH):	1,885,632
Unit availability** factor (%):	75.4
Unit capacity factor (%): (Using Design MWe)	54.0

**This factor added to the percent outage time is 102.3%. Discrepancy of 2.3% exists but it is based on best available data.

Outages

Total No.	22		
Forced	19		
Scheduled	3		
Total	1,188	Hours,	27.1%*
Forced	719	Hours,	16.4%*
Scheduled	469	Hours,	10.7%*

Cause:

Equipment Failure	14
Maintenance/Test	3
Reg. Restriction	1
Operational Error	6

Method of Shutdown:

Manual	1
Manual Scram	5
Auto Scram	15
Undesignated	1

*Data based on date of first commercial operation — July 1.

II. Highlights

A. General:

The unit began the report period at 50% of rated power. Startup testing was continuing. In the latter part of October, the power level was maintained between 75% and 100% of rated. Operation in December was at a nominal 80% of rated power.

B. Outages:

1. Forced: There were 19 forced outages during the report period which consumed 719 hours. The only outage which lasted longer than 100 hours occurred in July. It lasted 176 hours and was initiated by a relief valve which failed to close after a test. All eight relief valves were inspected and repaired.

2. Scheduled: There were 3 scheduled outages which consumed 469 hours. The longest outage was 285 hours to repair condenser tube leaks. Another outage for 173 hours was required to inspect the recirculation pump discharge valve bypass piping.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	7/1	70	F	Feedwater turbine steam governor valve malfunction.	A	3	Reactor Coolant (CH)	Valves
2)	7/6	11	S	Startup test. Turbine trip from 50% power.	B	2	Steam & Power (HA)	Turbines
3)	7/14	11	F	Relief valve failed to close after test.	A	2	Reactor Coolant (CC)	Valves
4)	7/19	14	F	Feedwater control signal upscale. Power supply malfunction due to faulty fuse connection.	A	3	Reactor Coolant (CH)	Circuit Closers
5)	7/20	18	F	Oil in generator due to sticking float in drain tank.	A	2	Steam & Power (HA)	Instrumentation & Controls
6)	7/22	176	F	One relief valve failed to close after test and one prematurely lifted. Inspected all 8 relief valves and repaired.	A	2	Reactor Coolant (CC)	Valves
7)	7/29	16	F	Feedwater control signal malfunction.	A	3	Reactor Coolant (CH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	8/12	56	F	Turbine control system malfunction. Modified governor valve pressure control curve.	A	3	Steam & Power (HA)	Valves
9)	8/27	63	F	Scram due to trip of main turbine oil pressure switch. Cause setpoint drift. Minor maintenance performed.	A	3	Steam & Power (HA)	Instrumentation & Controls
10)	8/30	9	F	Turbine control system inadequate during stop valve test. Pressure control feature unsuitable.	A	3	Steam & Power (HA)	Instrumentation & Controls
11)	9/10	61	F	Faulty press. switch caused partial closure of turbine control valves.	A	3	Steam & Power (HA)	Instrumentation & Controls
12)	9/14	3	F	Failure of position transducer on main turbine.	A		Steam & Power (HA)	Instrumentation & Controls
13)	9/14	173	S	Inspected recirc pump discharge valve bypass piping.	D	1	Reactor Coolant (CB)	Pipes, Fittings
14)	10/7	16	F	Scram from false low water level signal induced during testing.	G	3	Instrumentation & Controls (IA)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	10/8	8	F	Loss of bypass valve control fluid pressure due to operator switching error.	G	3	Steam & Power (HE)	NA
16)	10/16	13	F	False high main steam line radiation signal due to operator error in performance of procedure.	G	3	Reactor Coolant (CD)	NA
17)	10/22	24	F	Inadvertent trip of both recirc pumps during surveillance test.	G	3	Reactor Coolant (CB)	NA
18)	10/31	11	F	Apparent speed control failure in recirc system during test.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
19)	11/6	18	F	False high main steam steam line radiation signal due to personnel error during test.	G	3	Reactor Coolant (CD)	NA
20)	11/14	19	F	Erroneous high reactor pressure signal trip.	G	3	Instrumentation & Controls (IA)	NA
21a)	12/8	285	S	Repair condenser tube leaks.	B	3	Steam & Power (HC)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
21b)	12/19	71	F	Pilot stage leakage on relief valves extended this outage.	A		Reactor Coolant (CC)	Valves
21c)	12/22	31	F	Testing of new turbine bypass valves and repair of turbine control fluid leaks again extended this outage.	B		Steam & Power (HE)	Valves
22)	12/26	11	F	Repair leak in EHC fluid piping at a bypass valve.	A	2	Steam & Power (HE)	Pipes, Fittings

*Data for this table covers the period July 1 through December 31.

DRESDEN 1

I. Summary

Description	Performance		Outages		
Location: Morris, Illinois	Net electrical energy generated (MWH):	352,939	Total Number	5	
Docket No: 50-010			Forced	2	
Reactor Type: Boiling Water	Unit availability factor (%):	35.5	Scheduled	3	
Capacity (MWe-Net): 200			Total:	5,650	Hours, 64.5%
Commercial Operation: 7/60	Unit Capacity factor: (Using Design MWe)	20.1	Forced:	2,588	Hours, 29.5%
Plant Age: 14.7 years			Scheduled	3,062	Hours, 35.0%

Cause:	
Maintenance/Test	2
Refueling	1
Operator Error	1
Equipment Failure	3
Method of Shutdown:	
Manual	5

II. Highlights

A. General

At the beginning of the year, the refueling and maintenance outage started in 1973 was continuing. On July 5, repairs were completed and the unit placed on-line, ending a 270 day outage. On August 31, another extended outage of 1096 hours was required to remove the vessel head and recouple the control rod blades properly. In November and December operation was uninterrupted.

B. Outages:

1. Forced: Two forced outages occurred; one for 1096 hours to recouple control rod blades and one for 39 hours to repair steam sample line leaks.

2. Scheduled: A continuation of the 1973 refueling outage during which extensive maintenance and repair was performed. One hour was required for a turbine overspeed trip test; and 61 hours were required for routine maintenance and miscellaneous steam leaks.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1a)	1/1	3000	S	Continuation of refueling outage which started in 1973. Extensive maintenance and modifications performed	C		Reactor (RC)	Fuel Elements
1b)	1/1	753	F	Continuation of 1a. Delayed because of canal water quality.	A		Auxiliary Water (WC)	Filters
1c)	1/1	700	F	Continuation of 1a. Poison system pump seal leakage contaminated water system.	A		Auxiliary Process (PC)	Pumps
2)	7/5	1	S	Turbine overspeed trip test.	B	1	Steam & Power (HA)	Turbines
3)	8/2	61	S	Maintenance on miscellaneous steam leaks.	B	1	Reactor Coolant (CC)	Pipes, Fittings
4)	8/31	1096	F	Vessel head was removed because control rod blades had not been properly latched.	G	1	Reactor (RB)	Control Rods
5)	10/23	39	F	Repair leaks on primary steam sample lines	A	1	Reactor Coolant (CC)	Pipes, Fittings

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-237
Reactor Type: Boiling Water
Capacity (MWe-Net): 800
Commercial Operation: 6/9/72
Plant Age: 4.7 Years

Performance

Net electrical energy generated (MWH):	3,379,588
Unit availability factor (%):	64.1
Unit capacity factor (%): (Using Design MWe)	48.2

Outages

Total No.	20	
Forced	16	
Scheduled	4	
Total:	3,147	Hours, 35.9%
Forced	1,509	Hours, 17.2%
Scheduled	1,638	Hours, 18.7%

Cause	Equipment Failure	10
	Maintenance/Test	7
	Operational Error	2
	Refueling	1

Method of Shutdown:

Manual	12
Auto Scram	4
Manual Scram	3
Undesignated	1

II. Highlights

A. General:

The unit began the period at 720 MWe. Operation in January was uninterrupted but in February problems were experienced with recirc pump seal leakage and T/G turning gear damage. Operation in April was uninterrupted and in May operation was interrupted only to reverse circulating water flow. In June operation was interrupted for about 6 days to tie in the new modified off-gas system. In July containment isolation valve leakage was experienced and in August uncoupling problems with the control rod drives occurred. Most of September was devoted to inspection and repair of hairline cracks found in both recirc pump discharge valve bypass lines. November and December were devoted to refueling and maintenance.

B. Outages:

1. Forced: Sixteen forced outages required 1509 hours in 1974. Five of these outages were for replacement and testing of the generator reverse power relay and none exceeded 19 minutes duration. Five outages required over 100 hours. The longest was 580 hours to repair leaks on the recirculation system piping.

2. Scheduled: Four scheduled outages required 1638 hours. Two of these outages were to reverse the circulating water flow. One outage for 169 hours was to tie in the new off-gas system. The longest outage, for 1436 hours, was for refueling and maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/12	144	F	Repair excessive leaks on feedwater check valve seal ring and recirculation pump seal	A	1	Reactor Coolant (CH)	Valves
2)	2/18	329	F	Repair damaged turbine generator turning gear	G	1	Steam & Power (HA)	Turbines
3)	3/9	57	F	Repair excessive leakage of standby liquid control system valve and pilot valve of a relief valve in drywell	A		Auxiliary Process (PC)	Valves
4)	3/16	12	F	Spurious Hi-Hi moisture separator level signal	A	3	Reactor Coolant (CC)	Instrumentation & Controls
5)	5/4	19	S	Reverse circulating water flow and maintenance	B	2	Steam & Power (HC)	Heat Exchangers
6)	5/25	14	S	Circulating water flow reversal	B	2	Steam & Power (HC)	Heat Exchangers
7)	6/8	169	S	Scheduled to tie in the new modified off-gas system. Inspect shock suppressors	B	3	Radioactive Waste (MB)	Filters
8)	7/27	116	F	Repair leaking containment isolation valves	A	1	Reactor Coolant (CD)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	8/3	(18 min)	F	Replaced generator reverse power relay	A	1	Steam & Power (HA)	Relays
10)	8/11	(11 min)	F	Tested generator reverse power relay	B	1	Steam & Power (HA)	Relays
11)	8/11	(14 min)	F	Tested generator reverse power relay	B	1	Steam & Power (HA)	Relays
12)	8/11	(16 min)	F	Tested generator reverse power relay	B	1	Steam & Power (HA)	Relays
13)	8/11	(12 min)	F	Tested generator reverse power relay	B	1	Steam & Power (HA)	Relays
14)	8/22	112	F	Replace control rod drives because of uncoupling problems	A	1	Reactor (RB)	Control Rod Drives
15)	9/1	35	F	Repair ruptured cooling water line to condensate booster pump	A	2	Reactor Coolant (CH)	Pipes, Fittings
16)	9/3	14	F	Valving error caused generator/turbine mis-match signal	G	3	Steam & Power (HA)	NA
17)	9/12	580	F	Repair leaks on recir-culation system piping	A	1	Reactor Coolant (CB)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
18)	10/8	9	F	Repair pressure regulator circuitry	A	1	Reactor Coolant (CC)	Instrumentation & Controls
19)	10/19	99	F	Repair seal leak on recirculation pump	A	1	Reactor Coolant (CB)	Pumps
20)	11/2	1436	S	Refueling outage	C	3	Reactor (RC)	Fuel Elements

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-249
Reactor Type: Boiling Water
Capacity (MWe-Net): 800
Commercial Operation: 11/71
Plant Age: 3.4 Years

Performance

Net electrical energy generated (MWH):	3,200,269
Unit availability factor (%):	65.0
Unit capacity factor (%): (Using Design MWe)	45.7

Outages

Total No.	19	
Forced	18	
Scheduled	1	
Total:	3,064	Hours, 35.0%
Forced	986	Hours, 11.3%
Scheduled	2,078	Hours, 23.7%

Cause: Equipment Failure 14
Maintenance/Test 1
Refueling 1
Operational Error 3

Method of Shutdown:
Manual 9
Manual Scram 1
Auto Scram 9

II. Highlights

A. General:

Dresden 3 operated at about 650 MWe until March 11 when the unit was taken off line for refueling. The unit resumed operation on June 6 and operated at about 700 MWe the remainder of the year except for November and December. In November the power level was about 500 MWe. In December the average power level was 400 MWe.

B. Outages:

1. Forced: Eighteen forced outages required 986 hours. Repair of HPCI system and an electromatic relief valve required 164 hours. One hundred ninety three hours were expended to repair damaged pipe restraints on the feedwater lines. One hundred eighty three hours were needed to repair a leak on the feedwater discharge header.

2. Scheduled: The only scheduled outage for refueling required 2078 hours.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/17	164	F	Failure of the high pressure coolant injection system and an electromatic relief valve.	A	1	Engineered Safety (SF)	Valves
2)	1/25	6	F	High flux scram caused by pressure spike during turbine valve testing.	A	3	Reactor Coolant (CC)	NA
3)	2/20	37	F	Hydrogen explosion in off gas system damaged filter and ruptured SJAE rupture disc.	A	2	Radio-active Waste (MB)	Filters
4)	3/11	2078	S	Refueling outage plus containment leak testing, turbine inspection and off gas system tie-in.	C	1	Reactor (RC)	Fuel Elements
5)	6/10	7	F	Repair crack on feed-water line instrument tap.	A	1	Reactor Coolant (CH)	Pipes, Fittings
6)	6/17	19	F	Scram on hi flux due to recirc pump mismatch.	G	3	Reactor Coolant (CB)	NA
7)	6/24	193	F	High vibration in feed-water line caused pipe restraint damage.	A	1	Reactor Coolant (CH)	Shock Suppressors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	7/11	95	F	Repair recirc pump seal.	B	1	Reactor Coolant (CB)	Pumps
9)	7/22	19	F	Reactor Scram caused by Instrumentation vibration due to water hammer in core spray system.	G	3	Engineered Safety (SF)	Instrumentation & Controls
10)	7/27	62	F	Failure of primary containment isolation valve in the pressure suppression system to pass leak test.	A	1	Engineered Safety (SD)	Valves
11)	8/15	7	F	Loss of instrument air due to inadvertent valve closure.	G	3	Auxiliary Process (PA)	NA
12)	9/8	14	F	Repair Steam leaks on HP turbine.	A	1	Steam & Power (HA)	Turbines
13)	9/20	183	F	Repair leak on feedwater discharge header low pressure switch tap.	A	1	Reactor Coolant (CH)	Pipes, Fittings
14)	9/28	13	F	Low water level scram caused by failure of 3 element feedwater control system.	A	3	Reactor Coolant (CH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	11/4	72	F	Malfunction of pilot valve caused MSIV to trip.	A	3	Reactor Coolant (CD)	Valve Operators
16)	11/8	6	F	Power Spike caused by recirc pump speed spike.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
17)	11/9	32	F	Pressure spike in Recombiner system caused rupture of disc on SJAE. Low Cond. Vacuum.	A	3	Steam & Power (HC)	Pipes, Fittings
18)	11/27	12	F	Spurious MSIV closure.	A	3	Reactor Coolant (CD)	Valves
19)	11/30	35	F	Loss of secondary containment due to blow out of blow out panels.	A	1	Engineered Safety (SA)	Other

FORT CALHOUN

I. Summary

Description

Location: Fort Calhoun, Nebraska
Docket No: 50-285
Reactor Type: PWR
Capacity (MWe-Net): 457
Commercial Operation: 6/20/74
Plant Age: 1.4 Years

Performance

Net electrical energy generated(MWH):	2,416,252
Unit availability factor (%):	83.5
Unit capacity factor (%): (Using Design MWe)	60.4

Outages

Total No.	13		
Forced	4		
Scheduled	9		
Total:	1,451	Hours,	16.6%
Forced	613	Hours,	7.0%
Scheduled	838	Hours,	9.6%

Cause:
- Equipment Failure 3
- Maintenance/Test 9
- Operational Error 1

Method of Shutdown:
- Manual 6
- Auto Scram 7

II. Highlights

A. General:

There were 13 shutdowns during 1974 which accounted for 1451 hours of generator down time. Three of the shutdowns were related to problems with valves, three were related to problems with electrical equipment, three were related to performing tests and/or obtaining special data, and one was caused by an operational error while

B. Outages:

1. Forced: There were four forced outages during the year; these accounted for a total of 613 hours of generators down time. Two exceeded 100 hours in duration. One was for 304 hours (caused by failure of the MSIVS) and one for 284 hours (caused by a broken air supply line to a feedwater regulating valve).

2. Scheduled: There were nine scheduled outages occurred during the year and accounted for a total of 838 hours of generator down time. The scheduled outages exceeding 100 hours duration were: (1) 288 hours because of MSIV failure; (2) 189 hours because of an electrical power trip test; and (3) 173 hours for maintenance and cleaning of the Reactor coolant system.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (1974)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/19	288	S	Main steam isolation valve repair.	B	3	Reactor Coolant (CD)	Valves
2)	3/6	16	S	Scheduled complete loss of flow trip test.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls
3)	3/7	189	S	Complete loss of off-site A.C. power trip test followed by General Maintenance.	B	3	Reactor Coolant (CX)	NA
4)	3/29	304	F	MSIV failure.	A	1	Reactor Coolant (CD)	Valves
5)	4/11	25	S	Perform turbine overspeed trip tests and adjustments.	B	1	Steam & Power (HA)	Turbines
6)	4/17	284	F	Failure of main feedwater regulating valve (broken control air line).	A	3	Reactor Coolant (CH)	Pipes, Fittings
7)	5/10	33	S	100% power generator trip test.	B	3	Steam & Power (HA)	Generators
8)	8/13	19	F	Electrical system interference due to electrical storm.	A	3	Electric Power (EA)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	8/14	6	F	Operator error while borating.	G	3	Auxiliary Process (PC)	NA
10)	10/28	49	S	Containment surveillance test.	B	1	Engineered Safety (SA)	NA
11)	11/6	20	S	Outage to measure core delta P.	B	1	Reactor (RC)	NA
12)	11/9	173	S	Maintenance outage for cleaning of Reactor Coolant System.	B	1	Reactor Coolant (CX)	Pipes, Fittings
13)	12/30	45	S	Maintenance outage and low power physics testing for refueling.	B	1	Reactor (RB)	NA

GINNA

I. Summary

Description

Location: Ontario, New York
Docket No: 50-244
Reactor Type: Pressurized water
Capacity (MWe-Net): 490
Commercial Operation: 3/70
Plant Age: 5.1 Years

Performance

Net electrical energy generated (MWH):	2,097,216
Unit availability factor (%):	62.4
Unit capacity factor (%): (Using Design MWe)	51.7

Outages

Total No.	10	
Forced	7	
Scheduled	3	
Total:	3,292	Hours, 37.6
Forced	3,000	Hours, 34.2
Scheduled	292	Hours, 3.4

Cause: Equipment Failure 7
Maintenance/Test 3
Method of shutdown:
Manual 6
Auto Scram 4

II. Highlights

A. General:

During the first 4 months, overhauling of the turbine was conducted and refueling was accomplished. Power level for the remainder of the year ranged from 0% to 100% of full licensed power (1520 MWT).

B. Outages:

1. Forced: There were seven forced outages requiring a total of 3000 hrs; the longest for 2737 hrs to overhaul the turbine and conduct refueling.

2. Scheduled: There were 3 scheduled outages requiring 292 hours; the longest for 270 hours to inspect the steam generator tubing.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	2737	F	Blade failure on No. 2 LP turbine. Refueling was accomplished while maintenance overhaul of turbine was in progress.	A	1	Steam & Power (HA)	Turbines
2)	4/25	1	S	Turbine overspeed trip test.	B	1	Steam & Power (HA)	Turbines
3)	4/27	9	F	Instrument bus inverter failed resulting in trip.	A	3	Electric Power (ED)	Generators
4)	5/18	9	F	Steam leak on main steam to 1A reheater.	A	3	Steam & Power (HB)	Pipes, Fittings
5)	6/21	139	F	Repair gasket leak on pressurizer manway.	A	1	Reactor Coolant (CB)	Vessels, Pressure
6)	6/29	43	F	Repair leak in charging pump filter vent line.	A	1	Reactor Coolant (CH)	Pipes, Fittings
7)	7/2	47	F	Repair leak in charging pump filter vent pipe socket weld.	A	1	Reactor Coolant (CH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	7/26	16	F	Instrument Bus Inverter failed resulting in trip.	A	3	Electric Power (ED)	Generators
9)	8/24	21	S	Repair feedwater heater tube leaks.	B	3	Steam & Power (HH)	Heat Exchangers
10)	11/2	270	S	Inspect steam generator tubing.	B	1	Steam & Power (HB)	Heat Exchangers

HUMBOLDT BAY

I. Summary

Description

Location: Eureka, California
Docket No: 50-133
Reactor Type: Boiling Water
Capacity (MWe-Net): 65
Commercial Operation: 8/63
Plant Age: 11.7 Years

Performance

Net electrical energy generated (MWH):	365,930
Unit availability factor (%):	83.8
Unit capacity factor (%): (Using Design MWe)	66.3

Outages

Total No.	5		
Forced	3		
Scheduled	2		
Total:	1,416	Hours,	16.2%
Forced	28	Hours,	0.4%
Scheduled	1,388	Hours,	15.8%

Cause:

Equipment Failure	2
Op. Tng. & License Exam	1
Operational Error	1
Refueling	1

Method of Shutdown:

Manual	3
Auto Scram	2

II. Highlights

A. General:

The unit began the period at 65 MWe but in February and March operated base loaded at about 45 MWe. On August 1, the end of cycle coast down began and on October 30 the unit was shut down for the annual refueling. On December 27, the unit was returned to service.

B. Outages:

1. Forced: Three forced outages required 28 hours.

2. Scheduled: Two scheduled outages consumed 1388 hours, of this 1385 hours were for refueling and maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/23	1	F	Load fluctuations caused by turbine bypass valve operation. Rotary inverter brushes were stuck.	A	1	Reactor Coolant (CC)	Generators
2)	3/25	3	S	Operator license exam and turbine overspeed trip test.	E	1	Reactor (RB)	NA
3)	5/2	18	F	Reactor trip when linkage between turbine control valves and the operating cylinder sheared.	A	3	Reactor Coolant (CC)	Valve Operators
4)	6/26	9	F	115 KV system disturbance due to an improper wiring change in 1974.	G	3	Electric Power (EA)	Circuit Closers
5)	10/31	1385	S	Annual refueling and maintenance.	C	1	Reactor (RC)	Fuel Elements

INDIAN POINT 1

I. Summary

Description

Location: Indian Point, New York
Docket No: 50-003
Reactor Type: Pressurized Water
Capacity (MWe-Net): 265
Commercial Operation: 10/62
Plant Age: 12.3 Years

Performance

Net electrical energy generated (MWH):	1,232,560
Unit availability factor (%):	63.6
Unit capacity factor (%): (Using Design MWe)	55.8

Outages

Total No.	26	
Forced	20	
Scheduled	6	
Total:	3,191	Hours, 36.4%
Forced	1,173	Hours, 13.4%
Scheduled	2,018	Hours, 23.0%

Cause:

Equipment Failure	19
Maintenance/Test	5
Operational Error	1
Regulatory Restriction	1

Method of Shutdown:

Manual	15
Auto Scram	10

II. Highlights

A. General:

At the beginning of the period, the plant was in the final stages of an extended refueling and maintenance outage that had started December 29, 1972. On January 19, the plant was returned to service and operated until October 31 when it was shut down to perform installation of an ECCS to comply with the AEC Interim Acceptance Criteria for ECCS.

B. Outages:

1. Forced: Twenty forced outages caused the plant to be out of service for 1173 hours during 1974. Four outages exceeded 100 hours in duration. At the beginning of the period 383 hours were required to complete repairs begun on Dec. 29, 1972. At the end of January, 113 hours were required for cable repairs on a 13.8 KV feeder. In July 236 hours were required to repair a leaking downcomer on a nuclear boiler. In October another leaking downcomer required 113 hours for repairs.

2. Scheduled: There were six scheduled outages which required 2018 hours during the year. In May, 355 hours were required to investigate and repair a leak on a downcomer of a nuclear boiler and to repair tube leaks. On October 31, the plant was shut down for an estimated 2-1/2 years for installation of an ECCS.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	383	F	Continuation of Dec. 29, 1972 shutdown for refueling and general maintenance.	A		Steam and Power (HB)	Heat Exchangers
2)	1/16	54	F	Oil leak on the No. 6 bearing at the turbine turning gear.	A	1	Steam and Power (HA)	Pipes, Fittings
3)	1/19	21	F	Spurious scram caused by false indication from the gross gamma monitor.	A	3	Radiation Protection (BA)	Instrumentation & Controls
4)	1/20	11	S	Turbine overspeed trip test.	B	1	Steam and Power (HA)	Turbines
5)	1/23	17	F	Scram occurred due to a defective scram solenoid valve on the No. 12 control rod.	A	3	Reactor (RB)	Valve Operators
6)	1/28	113	F	13.8 KV Feeder grounding troubles.	A	1	Electric Power (EA)	Electrical Conductors
7)	2/5	42	F	Spurious trip of channel 16 flux amplifier while channel 11 was being recalibrated. An open circuit to the drive motor windings for CRD No. 6 delayed restart.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	2/21	20	F	Another spurious trip from channel 16 flux amplifier while channel 11 was being recalibrated.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
9)	2/28	7	F	Trouble with pothead on outgoing 138 KV feeder. Transferred to companion feeder.	A	1	Electric Power (EA)	Electrical Conductors
10)	3/22	56	S	Scheduled weekend maintenance.	B	1		
11)	3/25	7	F	Unit trip due to lightning arrestor failure on a 138 KV feeder.	A	3	Electric Power (EA)	Electrical Conductors
12)	5/4	355	S	Scheduled maintenance. Investigation and repair of weld leak on downcomer of steam generator. Repaired tube leaks on both superheaters.	B	1	Steam and Power (HB)	Pipes, Fittings
13)	6/3	6	F	Scram inadvertantly initiated while technicians were checking the cause of a scram pulser failure.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
14)	7/7	236	F	Leakage of the No. 5 downcomer on the No. 11 steam generator.	A	1	Steam and Power (HB)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	7/16	14	F	Repaired disconnect switch on the generator side of the generator breaker.	A	1	Steam and Power (HA)	Circuit Closers/ Interrupters
16)	7/19	24	F	Failed rupture discs in the excess makeup system.	A	1	Reactor Coolant (CH)	Pipes, Fittings
17)	7/28	59	F	Replaced a defective lower limit switch on the No. 20 control rod.	A	3	Reactor (RB)	Circuits Closers/ Interrupters
18)	8/9	35	S	Repair leaking pressure connection on the main feedwater supply line and overhaul of controllers on deaerator level control valves.	B	1	Steam and Power (HH)	Pipes, Fittings
19)	8/11	21	F	Repair deaerator regulator controls.	A	1	Reactor Coolant (CG)	Instrumentation & Controls
20)	8/13	10	F	Disturbances on the 13.8 KV system caused flux flow computer scram.	A	3	Electric Power (EA)	Instrumentation & Controls
21)	8/14	7	F	Disturbances on the 13.8 KV system caused flux flow computer scram.	A	3	Electric Power (EA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
22)	8/22	15	F	Repair a control oil leak on the main turbine.	A	1	Steam and Power (HA)	Pipes, Fittings
23)	9/2	96	S	Repair leaking tubes on steam generator and feedwater heater.	B	1	Steam and Power (HB)	Heat Exchangers
24)	10/1	113	F	Repair leaking downcomer on No. 12 steam generator.	A	1	Reactor Coolant (CC)	Pipes, Fittings
25)	10/18	7	F	A bus tie fault resulted in scram.	A	3	Electric Power (EB)	Circuit Closers/ Interrupters
26)	10/31	1465	S	To comply with AEC interim acceptance criteria for ECCS	D	1	Engineered Safety (SF)	NA

INDIAN POINT 2

I. Summary

Description	Performance		Outages		
Location: Indian Point, New York	Net electrical energy generated (MWH):	3,324,048	Total No.	63	
Docket No: 50-247			Forced	52	
Reactor Type: Pressurized Water	Unit availability factor (%):	59.4	Scheduled	11	
Capacity (MWe-Net): 873	Unit capacity factor (%): (Using Design MWe)	43.5	Total:	3,552	Hours, 40.6%
Commercial Operation: 8/73			Forced	2,405	Hours, 27.5%
Plant Age: 1.5 Years			Scheduled	1,147	Hours, 13.1%

Cause:	
Equipment Failure	46
Maintenance/Test	11
Operational Error	5
Other	1

Method of Shutdown:	
Auto Scram	54
Manual Scram	1
Manual	7

II. Highlights

A. General:

At the beginning of the year the plant was still shut down to repair a crack in a steam generator main feedwater line. Following the repairs, the plant operated for a short period and on January 29, a waterhammer occurred in a feedwater line. Results indicated the phenomenon was a function of the rate of auxiliary boiler feedwater flow during recovery from low steam generator level trips when the feedwater rings were exposed to a steam atmosphere. Internal modification to the Steam Generator rings were made requiring 1238 hours of down time. Operation for the remainder of the year was sporadic primarily due to spurious main steam isolation valve closures, and losses of main boiler feed pumps. Feedwater control problems at low loads also produced several trips.

B. Outages:

1. Forced: Fifty-two forced outages caused 2405 hours of plant downtime. Six hundred and five hours were a continuation of a 1973 outage to modify a main feedwater line. One thousand two hundred and thirty-eight hours were required to correct a water hammer problem in the SG feedwater rings. Eleven shutdowns were caused by the loss of main boiler feed pumps. Five shutdowns were caused by spurious closure of main steam isolation valves, and 5 shutdowns were the result of feedwater control problems at low loads.

2. Scheduled: Eleven scheduled outages required 1147 hours. In April 247 hours were required to perform a loss of coolant flow test, inspect seismic restraints, and perform general maintenance. A portion of 6 scheduled outages were devoted to inspection of seismic restraints.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	605	F	Continuation of 1973 shutdown to modify main feedwater line to the No. 22 steam generator.	A		Steam & Power (HH)	Pipes, Fittings
2)	1/26	3	F	Trip from under-power relay because power output to system was not attained within specified time interval.	G	3	Steam & Power (HA)	Relays
3)	1/27	4	S	Check turbine overspeed trip mechanism.	B	1	Steam & Power (HA)	Mechanical Function Units
4)	1/28	3	F	High drum level on steam generator due to feedwater oscillation on failure of heater drain tank level controller.	A	3	Steam & Power (HH)	Instrumentation & Controls
5)	1/29	3	F	Low drum level on SG No. 21 due to feedwater oscillation on reduction of condenser bypass flow.	A	3	Steam & Power (HH)	Valves
6)	1/29	1238	F	Modification of steam generator feedwater rings to eliminate water hammer.	A	3	Steam & Power (HH)	Pipes, Fittings
7)	3/22	1	F	Low drum level on SG No. 21.	A	3	Steam & Power (HB)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	3/22	4	F	Main steam isolation valve closure resulted in low drum level on SG No. 23.	A	3	Reactor Coolant (CD)	Valves
9)	3/26	3	F	Both main steam generator feed pumps were lost. False signal due to electrical impulse from auto selector switch.	A	3	Steam & Power (HH)	Circuit Closers/ Interrupters
10)	3/26	11	F	High drum level on SG No. 22 due to level control problems.	A	3	Steam & Power (HB)	Instrumentation & Controls
11)	4/7	16	F	Spurious trip signal from Buchanan Substation.	A	3	Electric Power (EA)	NA
12)	4/8	1	F	Leaking flow transmitter drain valve resulted in false low reactor coolant flow indication on loop No. 24.	A	3	Instrumentation & Controls (IA)	Valves
13)	4/8	1	F	Leaking flow transmitter drain valve resulted in false low reactor coolant flow indication on loop No. 24.	A	3	Instrumentation & Controls (IA)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	4/9	2	F	Steam generator feed pump problems. Pump suction header pressure cut back controller setting was reduced to correct problem.	A	3	Steam & Power (HH)	Pumps
15)	4/10	4	F	Low level in SG due to trip of both main SG feed pumps.	A	3	Steam & Power (HH)	Pumps
16)	4/18	247	S	To perform loss of coolant flow test, monthly seismic restraint inspection, and maintenance. An early trip occurred due to trip of SG feed pump.	B	3	Reactor Coolant (CX)	Shock Suppressors
17)	4/29	2	F	Low level in SG due to trip of No. 21 boiler feed pump.	A	3	Steam & Power (HB)	Pumps
18)	4/29	4	F	Operator inadvertantly placed high source range flux trips in service causing trip.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
19)	5/3	58	F	Loss of No. 21 main SG feed pump. Pump governor oil piping leaks were repaired.	A	3	Steam & Power (HH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
20)	5/5	5	F	Loss of No. 21 main SG feed pump.	A	3	Steam & Power (HH)	Pumps
21)	5/6	9	F	Loss of No. 21 main SG feed pump. Oil fitting leaks were repaired.	A	3	Steam & Power (HH)	Pipes, Fittings
22)	5/10	67	S	Miscellaneous maintenance.	B	1		
23)	5/13	184	F	Blown rupture disc on the pressurizer relief tank. New disc installed and repaired tank foundation.	A	3	Reactor Coolant (CJ)	Vessels, Pressure
24)	5/21	3	F	High drum level in SG No. 23 due to difficulty in controlling levels.	A	3	Steam & Power (HB)	Instrumentation & Controls
25)	5/22	18	F	Repair air leak to valve operator of main steam isolation valve.	A	3	Reactor Coolant (CD)	Valve Operators
26)	5/23	1	F	Tripped as a result of over temp delta T protection circuitry due to axial offset.	A	3	Instrumentation & Controls (IA)	NA
27)	6/1	20	F	Loss of instrument bus No. 22 due to failure of static inverter.	A	3	Electric Power (ED)	Generators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
28)	6/6	3	F	Lost No. 21 SG feed pump.	A	3	Steam & Power (HH)	Pumps
29)	6/10	5	F	Closure of main steam isolation valve due to low air pressure to valve operating cylinder.	A	3	Reactor Coolant (CD)	Valve Operators
30)	6/12	2	F	Closure of main steam isolation valve due to low air pressure to valve operating cylinder.	A	3	Reactor Coolant (CD)	Valves Operators
31)	6/14	61	S	Inspect seismic pipe restraints.	B	3	Reactor Coolant (CD)	Shock Suppressors and Supports
32)	6/21	9	F	Malfunction of heater drain pump discharge regulator.	A	3	Steam & Power (HH)	Instrumentation & Controls
33)	6/24	25	F	Closure of main steam isolation valve due to low air pressure to valve operating cylinder.	A	3	Reactor Coolant (CD)	Valve Operators
34)	7/2	7	F	Main SG feed pump tripped due to oil leakage at loose coupling on control oil header.	A	3	Steam & Power (HH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
35)	7/2	2	F	Repair malfunction of feed-water regulator.	A	3	Steam & Power (HH)	Instrumentation & Controls
36)	7/3	7	F	Repair loose terminal block connection for loop 22 cold leg temperature input.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
37)	7/4	17	F	Repair loose connection on loop 24 cold leg RTD amplifier.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
38)	7/22	32	F	Truck driver performed valving to change nitrogen tank trucks. Low pressure resulted in MSIV closure.	G	3		Valves
39)	7/26	265	S	Perform 100% plant trip test & scheduled maintenance.	B	2	Steam & Power (HA)	Generators
40)	8/6	9	F	High drum level on SG 22 due to feedwater control problems at low loads.	A	3	Steam & Power (HH)	Instrumentation & Controls
41)	8/8	8	F	Trip due to insufficient reactivity to keep up with Xenon burnout.	G	3	Reactor (RB)	Control Rods
42)	8/8	7	F	Low drum level caused by main steam isolation valve closure.	A	3	Reactor Coolant (CD)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
43)	8/28	15	S	Repair weld leak on vent valve associated with boiler feedwater discharge piping.	B	1	Steam & Power (HH)	Pipes, Fittings
44)	8/30	12	S	Inspect main SG feedwater pump and repair nipple on discharge line.	B	1	Steam & Power (HH)	Pipes, Fittings
45)	8/31	2	F	Spurious trip of No. 21 main SG feed pump.	A	3	Steam & Power (HH)	Pumps
46)	9/3	14	F	Steam generator mismatch signal caused trip due to failure of No. 21 static inverter.	A	3	Steam & Power (HB)	Generators
47)	9/6	97	S	Inspect seismic pipe restraints.	B	1	Reactor Coolant (CX)	Shock Suppressors and Supports
48)	9/13	1	F	Trip due to insufficient reactivity to keep up with Xenon burnout.	G	3	Reactor (RB)	Control Rods
49)	9/13	1	F	Trip due to feedwater control problems at low loads.	A	3	Steam & Power (HH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
50)	9/13	3	F	Trip due to feedwater control problems at low loads.	A	3	Steam & Power (HH)	Instrumentation & Controls
51)	9/14	3	F	Turbine governor failure caused large load swings resulting in trip.	A	3	Steam & Power (HA)	Mechanical Function Units
52)	9/27	7	F	Trip resulted from electrical fault external to plant which opened generator output breakers.	A	3	Electric Power (EA)	Breakers
53)	9/30	278	S	Inspect seismic pipe restraints and maintenance.	B	3	Reactor Coolant (CX)	Shock Suppressors and Supports
54)	11/6	4	F	Trip due to shorted test lead on No. 21 hot leg RTD.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
55)	11/7	11	F	Trip due to transient during 3 loop operation.	A	3	Steam & Power (HB)	Valves
56)	11/9	54	S	Inspect seismic pipe restraints and maintenance.	B	1	Reactor Coolant (CX)	Shock Suppressors and Supports

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
57)	11/13	16	F	Inverter failed. Switched instrument bus to backup.	A	3	Electric Power (ED)	Generators
58)	11/28	4	F	Lost No. 21 main SG feed pump causing steam generator mismatch.	A	3	Steam & Power (HH)	Pumps
59)	12/2	2	F	Electrical system disturbance caused drop in instrument bus voltage.	A	3	Electric Power (EA)	NA
60)	12/5	1	F	Operator accidentally hit trip lever on main SG feed pump.	G	3	Steam & Power (HH)	NA
61)	12/6	47	S	Inspect seismic restraints and perform maintenance.	B	1	Reactor Coolant (CX)	Shock Suppressors and Supports
62)	12/8	1	F	Spurious signal on steam generator level channel.	A	3	Steam & Power (HB)	Instrumentation & Controls
63)	12/15	3	F	Governor torque motor grounded causing No. 22 SG feed pump to run down.	A	3	Steam & Power (HH)	Mechanical Function Units

KEWAUNEE

I. Summary

Description

Location: Carlton, Wisconsin
Docket No: 50-305
Reactor Type: PWR
Capacity (MWe-Net): 535
Commercial Operation: 6/16/74
Plant Age: 0.8 Years

Performance

Net electrical energy generated (MWH):	1,589,173
Unit availability** factor (%):	75.2
Unit capacity factor** (%): (Using Design MWe)	62.2

**Based on date of commercial operation — June 16.

Outages

Total No.	38		
Forced	31		
Scheduled	7		
Total:	1,768	Hours,	27.5%*
Forced	550	Hours	8.6%*
Scheduled	1,218	Hours	18.9%*

Cause:
- Equipment Failure 19
- Maintenance/Test 8
- Operator Training 1
- Operational Error 10

Method of Shutdown:
- Manual 9
- Manual Scram 2
- Auto Scram 27

*Base is 6430 hrs from time of first electrical generation to end of year.

II. Highlights

A. General:

The units initial electrical power generation occurred on April 8. In June the unit operated up to 100% power. In September a 711 hour outage was incurred to test the steam generator and to prepare them for all volatile treatment for chemistry control.

B. Outages:

1. Forced: There were 31 forced outages during the year in which 550 hours were expended. The longest forced outage was for 135 hours to repair a cracked casing on a feedwater pump.

2. Scheduled: Seven scheduled outages required 1218 hours. The longest outage required 711 hours for testing steam generators and preparation for the all volatile treatment. Another scheduled outage consumed 278 hours to repair a leaking pressurizer manway and leaking valves.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	4/8	8	F	Feedwater control problems. Trip from hi S/G level.	A	3	Steam & Power (HH)	Instrumentation & Controls
2)	4/8	12	F	Feedwater control problems.	A	3	Steam & Power (HH)	Instrumentation & Controls
3)	4/9	15	F	Feedwater control problems.	A	3	Steam & Power (HH)	Instrumentation & Controls
4)	4/9	135	F	Repair feedwater pump casing.	A	1	Steam & Power (HH)	Pumps
5)	4/15	30	F	Repair cracked weld on feedwater pump suction line.	A	1	Steam & Power (HH)	Pipes, Fittings
6)	4/16	29	F	Drain lines to condenser plugged with debris caused moisture separator relief valve to lift.	A	3	Steam & Power (HB)	Pipes, Fittings
7)	4/19	16	F	Leaking flanges on turbine intercept and reheat Stop valves.	A	1	Steam & Power (HB)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/20	9	F	Repair weld failure on feedwater pump suction line hangar.	A	1	Steam & Power (HH)	Shock Suppressors
9)	4/23	12	F	I & C testing lamp on safeguard rack caused trip of FW isolation.	G	3	Instrumentation & Controls (IB)	NA
10)	4/24	6	F	Operator reduced heater drain pump speed causing low suction press to FW pump which then tripped.	G	3	Steam & Power (HH)	NA
11)	4/24	3	F	Manual turbine trip because of water hammer in reheater relief valve line. Lines not draining properly.	A	2	Steam & Power (HB)	Pipes, Fittings
12)	4/24	75	S	AEC licensing exams.	E	3	Reactor (RB)	NA
13)	4/28	4	S	Test turbine overspeed trip.	B	3	Steam & Power (HA)	NA
14)	4/28	4	F	Operator valved in main FW control valves which was slightly open. Trip on SG Hi level.	G	3	Steam & Power (HH)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	5/2	19	F	Unit tripped while I & C was changing rate trip setpoints. Various maintenance performed.	G	3	Instrumentation & Controls (IA)	NA
16)	5/3	7	F	Failed power supply in EH turbine control.	A	3	Steam & Power (HA)	Turbines
17)	5/4	3	F	Dropped rod giving negative rate trip.	A	3	Reactor (RB)	Control Rods
18)	5/7	9	F	Trip due to lost feedwater pump suction pressure.	A	3	Steam & Power (HB)	Filters
19)	5/8	40	F	Pressure loss across air ejector too great at higher loads. Bypass line installed.	A	3	Steam & Power (HC)	Pipes, Fittings
20)	5/11	28	F	Scale buildup in condensate pump strainers tripped feed pump which tripped unit.	A	3	Steam & Power (HH)	Filters
21)	5/20	4	F	Technician failed to reset rate trip before working on second unit.	G	3	Instrumentation & Controls (IA)	NA
22)	5/25	78	S	Maintenance on feedwater heaters.	B	3	Steam & Power (HH)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
23)	6/3	7	F	Electrician caused condensate pump trip which tripped turbine.	G	3	Steam & Power (HH)	NA
24)	6/8	28	F	Replaced leaking annubar on heater drain pipe.	B	1	Steam & Power (HH)	Pipes, Fittings
25)	6/19	6	F	Technician inadvertently tripped turbine.	G	3	Steam & Power (HA)	NA
26)	6/21	2	F	Failed relay in turbine overspeed system.	A	3	Steam & Power (HA)	Relays
27)	6/27	4	F	Turbine EH controller failed.	A	3	Steam & Power (HA)	Circuit Closers
28)	6/28	278	S	Scheduled maintenance on leaking pressurizer manway and leaking valves.	B	3	Reactor Coolant (CB)	Vessels, Pressure
29)	8/27	12	F	Broken air line to main steam isolation valve.	A	3	Steam & Power (HB)	Pipes, Fittings
30)	8/27	5	F	Steam flow feed flow mismatch.	G	3	Steam & Power (HB)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
31)	9/8	13	F	Misaligned turbine control valves.	A	2	Steam & Power (HA)	Valves
32)	9/10	12	F	Broken airline to main steam isolation valve broke.	A	1	Steam & Power (HB)	Pipes, Fittings
33)	9/19	711	S	Prepared steam generators for all volatile treatment.	B	3	Steam & Power (HB)	Heat Exchangers
34)	10/26	20	S	Repaired feedwater pump suction valve.	B	1	Steam & Power (HH)	Valves
35)	10/31	2	F	Technician work on pressurizer level inst. caused trip.	G	3	Reactor Coolant (CB)	NA
36)	11/8	52	S	Feed pump repair.	E	1	Steam & Power (HH)	Pumps
37)	11/28	62	F	Operator inadvertently opened generator output breaker.	G	3	Electric Power (EA)	NA
38)	12/22	8	F	Secondary side steam leak-reweld.	B	1	Steam & Power (HB)	Pipes, Fittings

LA CROSSE

I. Summary

Description	Performance		Outages		
Location: Genoa, Wisconsin	Net electrical energy generated (MWH):	313,440	Total No.	14	
Docket No: 50-409			Forced	8	
Reactor Type: BWR	Unit availability factor (%):	81.0	Scheduled	6	
Capacity (MWe-Net): 48			Total:	1,662	Hours, 19.0%
Commercial Operation: 9/13/69	Unit capacity factor (%): (Using Design MWe)	79.2	Forced	501	Hours, 5.7%
Plant Age: 6.7 Years			Scheduled	1,161	Hours, 13.3%

Cause:
- Equipment Failure 7
- Maintenance/Test 5
- Operator Training 2
- Operational Error 1
- Other 1

Method of Shutdown:
- Manual 7
- Auto Scram 7

II. Highlights

A. General:

Problems with control rod drive mechanisms accounted for five of the 16 outages, and problems with the MSIV's accounted for 2 outages.

B. Outages:

1. Forced: Eight forced outages caused the generator to be out of service for a total of 501 hours. One outage of 294 hours was for the purpose of inspecting and repairing a main steam isolation valve.

2. Scheduled: Six scheduled outages caused the generator to be out of service for a total of 1161 hours. The outage of longest duration was for 554 hours to conduct maintenance and perform the reactor vessel stress analysis.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/10	29	S	Operator license, exam, and maintenance on control rod insert-withdraw switch.	E	3	Reactor (RB)	Control Rod Drives
2)	2/26	18	F	Both recirc pump tripped while shifting seal injection pumps.	A	3	Reactor Coolant (CB)	Pumps
3)	3/2	8	F	Scram from closure of MSIV due to vibration of a control relay.	A	3	Reactor Coolant (CD)	Relays
4a)	5/6	419	S	Semiannual maintenance, primarily on recirc pumps.	B	1	Reactor Coolant (CB)	Pumps
4b)	5/6	113	S	Continuation of outage due to mechanical seal leakage on CRD.	B		Reactor (RB)	Control Rod Drives
5)	6/5	26	F	Adjusted controller of the steam turbine initial pressure regulator.	A	1	Reactor Coolant (CC)	Instrumentation & Controls
6)	6/16	2	F	Operator erroneously adjusted a trip signal.	G	3	Instrumentation & Controls (IA)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	7/15	25	S	Replaced a leaking seal on a CRD hydraulic accumulator.	B	1	Reactor (RB)	Control Rod Drives
8)	8/10	4	S	Repaired oil leak on CRDM hydraulic accumulator.	B	1	Reactor (RB)	Control Rod Drives
9a)	8/28	2	F	A partial interruption of electrical power to the I & C regulated bus occurred due to failure of transformer.	A	3	Electric Power (EB)	Transformers
9b)	8/28	554	S	Continuation of 9a to perform fall maintenance. Reactor remained shutdown most of the month until the reactor vessel stress analysis were completed.	B		Reactor Coolant (CA)	Vessels, Pressure
10)	9/24	93	F	Repair oil leak on CRDM.	A	1	Reactor (RB)	Control Rod Drives
11)	9/28	294	F	Malfunction of main steam isolation valve caused scram. Valve was disassembled, inspected, and repaired.	A	3	Reactor Coolant (CD)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
12)	10/11	33	F	Repaired a hydraulic oil leak on a control rod drive mechanism.	A	1	Reactor (RB)	Control Rod Drives
13)	10/23	25	F	A power operated post hole digger severed underground control wiring for 69 KV breaker which caused load rejection and subsequent scram.	H	3	Electric Power (EB)	Electrical Conductors
14)	11/7	17	S	Performed 8 critical demonstrations for operator license applicants.	E	1	Reactor (RB)	NA

MAINE YANKEE

I. Summary

Description

Location: Wiscasset, Maine
Docket No: 50-309
Reactor Type: Pressurized Water
Capacity (MWe-Net): 790
Commercial Operation: 12/72
Plant Age: 2.1 Years

Performance

Net electrical energy generated (MWH):	3,574,301
Unit availability factor (%)	68.7
Unit capacity factor (%) (Using Design MWe)	51.6

Outages

Total No.	11		
Forced	8		
Scheduled	3		
Total:	2,748	Hours,	31.3%
Forced	137	Hours,	1.5%
Scheduled	2,611	Hours,	29.8%

Cause: Equipment Failure 8
Maintenance/Test 2
Refueling 1

Method of shutdown:
Manual 6
Auto Scram 5

II. Highlights

A. General :

The unit operated at power most of the year except for the months of August, September, and October which were devoted to fuel inspection and refueling.

B. Outages :

1. Forced: There were 8 forced outages during the year which required 137 hours to effect repairs. The longest of these outages was for 57 hours resulting from a failure of a heater drain tank level control valve.

2. Scheduled: There were three scheduled outages, consuming 2611 hours, of which the longest was for 2513 hours devoted to fuel inspection and refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/14	7	F	Test switch failure during monthly surveillance testing.	A	3	Instrumentation & Controls (IA)	Circuit Closers
2)	2/16	23	S	To repair minor steam leaks on turbine drain lines and misc. maintenance.	B	1	Steam & Power (HB)	Pipes, Fittings
3)	3/8	75	S	To repair H_2 leak in gen. leads box and AEC Operator License.	B	1	Steam & Power (HA)	Generators
4)	4/5	57	F	Failure of heater drain tank level control valve.	A	3	Steam & Power (HH)	Valves
5)	4/8	5	F	Ruptured diaphragm on the interface/dump valve (turbine oil system).	A	3	Steam & Power (HA)	Valves
6)	6/28	2513	S	Refueling and inspection.	C	1	Reactor (RC)	Fuel Elements
7)	11/6	7	F	Voltage spike in RPS — turbine runback and trip.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
8)	11/13	31	F	Repair HP turbine steam leak.	A	1	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	11/15	7	F	Voltage spike in RPS — turbine runback trip.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
10)	12/6	10	F	Condenser tube leak.	A	1	Steam & Power (HC)	Heat Exchangers
11)	12/26	13	F	Condenser tube leak.	A	1	Steam & Power (HC)	Heat Exchangers

MILLSTONE POINT 1

I. Summary

Description

Location: Waterford, Connecticut
Docket No: 50-248
Reactor Type: Boiling Water
Capacity (MWe-Net): 652
Commercial Operation: 3/71
Plant Age: 4.1 Years

Performance

Net electrical energy generated (MWH):	3,604,240
Unit availability factor (%):	79.1
Unit capacity factor (%): (Using Design MWe)	63.1

Outages

Total No.	17		
Forced	14		
Scheduled	3		
Total	1,832	Hours,	20.9%
Forced	186	Hours,	2.1%
Scheduled	1,646	Hours,	18.8%

Cause:

Equipment Failure	13
License Exams	1
Reg. Restriction	1
Refueling	1
Operational Error	1

Method of Shutdown:

Manual	12
Auto Scram	5

II. Highlights

A. General:

The plant operated satisfactorily except that it was limited to 80% power to prevent inducing vibration in the feedwater distribution spargers. On August 30, the unit shut down for refueling, and the feedwater spargers were replaced. Following the sparger replacement, a power level of 97% was obtained.

B. Outages:

1. Forced: There were fourteen forced outages during the year consuming 186 hours. The longest outage was for 52 hours to repair pilot seats on the pressure relief valves.

2. Scheduled: There were 3 scheduled outages consuming 1646 hours. The longest outage consumed 1553 hours in which refueling was accomplished and the feedwater spargers were replaced.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/11	52	F	Repair pilot seats on auto pressure relief valves.	A	1	Reactor Coolant (CB)	Valves
2)	2/8	52	S	AEC license exams.	E	1	Reactor (RB)	NA
3)	3/6	23	F	Packing of reactor recirculation equalizer valve leaking. Repaired.	A	1	Reactor Coolant (CB)	Valves
4)	3/20	4	F	Steam leak weld connection on instrument top of main steam line.	A	1	Reactor Coolant (CC)	Pipes, Fittings
5)	6/11	7	F	Recirculation pump control signal malfunction.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
6)	6/28	41	S	Inspection of seismic shock suppressors, replaced pump seal on reactor recirculation pump, plugged leaking condenser.	D	1	Reactor Coolant (CB)	Shock Suppressors
7)	8/30	1553	S	Refueling, maintenance, and feedwater sparger replacement.	C	1	Reactor (RC)	Fuel Elements
8)	11/3	15	F	Turbine trip from high water level caused by maintenance on feedwater transmitter.	G	3	Reactor Coolant (CH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	11/4	9	F	While reducing power to investigate MSIV malfunction, pressure regulator also malfunctioned and resulted in scram.	A	3	Reactor Coolant (CC)	Valves
10)	11/5	5	F	During pressure regulator testing, pressure oscillation caused scram from APRM hi flux.	A	3	Reactor Coolant (CC)	Instrumentation & Controls
11)	11/15	8	F	MSIV failure to close. Debris in air slide-valve.	A	1	Reactor Coolant (CD)	Valves
12)	11/18	15	F	Main generator ground fault.	A	1	Steam & Power (HA)	Generators
13)	12/16	8	F	Main feedwater control valve failed.	A	3	Reactor Coolant (CH)	Valves
14)	12/27	34	F	Repair main feedwater control valve.	A	1	Reactor Coolant (CH)	Valves
15)	12/27	1	F	Malfunction of level controller on moisture separator.	A	1	Steam & Power (HB)	Circuit Closers
16)	12/29	2	F	Malfunction of level controller on moisture separator.	A	1	Steam & Power (HB)	Circuit Closers
17)	12/29	3	F	Malfunction of level controller on moisture separator.	A	1	Steam & Power (HB)	Circuit Closers

MONTICELLO

I. Summary

Description

Location: Monticello, Minnesota
Docket No: 50-263
Reactor Type: Boiling Water
Capacity (MWe-Net): 545
Commercial Operation: 7/04/71
Plant Age: 3.8 Years

Performance

Net electrical energy generated (MWH):	2,923,836
Unit availability Factor (%):	74.9
Unit capacity factor (%): (Using Design MWe)	62.0

Outages

Total No.	13		
Forced	8		
Scheduled	5		
Total:	2,200	Hours,	25.1%
Forced	284	Hours,	3.2%
Scheduled	1,916	Hours,	21.9%

Cause:

Equipment Failure	8
Maintenance/Test	2
Refueling	1
Op. Tng. & License Exam	2

Method of Shutdown:

Manual	9
Scram	4

II. Highlights

A. General:

Five of the thirteen outages were directly related to the off-gas system (e.g., detonation of hydrogen in the recombiner and modification to the system). Three of the thirteen outages were related to problems with electrical systems. A refueling outage lasted for 1617 hours. Two outages were for operator training and licensing examinations. Administrative limits on reactor power were continued during the period to minimize off-gas release rates and to minimize plant radiation levels. Fuel leakage increased to the extent that, prior to resumption of test operation of the modified off-gas system (in November), the off-gas release rate of 85% of rated power was at 25% of Tech. Spec. limits for the stack.

B. Outages:

1. Forced: Eight forced outages caused the plant to be out of service for 284 hours during 1974. Only one outage exceeded 100 hours duration (141 hours; resulting from the malfunction of a safety-relief valve). The next longest outage (63 hrs.) was necessitated by a malfunctioning main steam line isolation valve.

2. Scheduled: There were five scheduled outages which required a total of 1916 hours during the year. Those of the longest duration were: (1) 1617 hours, for refueling; (2) 117 hours, for the modification of the off-gas recombiner and repair of the generator's outboard hydrogen seal; and (3) 77 hours, for licensing exams (this shutdown was extended because of safety-relief valve malfunction).

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/16	63	F	MSIV failed to close during test due to design error in solenoid.	A	1	Reactor Coolant (CD)	Valve Operators
2)	3/15	1618	S	Refueling	C	1	Reactor (RC)	Fuel Elements
3)	6/6	76	S	Tie in modified off-gas system.	B	1	Radioactive Waste (MB)	Pipes, Fittings
4)	6/10	29	F	Low condenser vacuum due to rupture of air ejector pressure relief discs caused by hydrogen explosion.	A	3	Radioactive Waste (MB)	Recombiners
5)	6/11	5	F	Repaired a generator field ground.	A	1	Steam & Power (HA)	Generators
6)	6/14	10	F	Repaired a leak on a restricting orifice coupling on feed pump warmup line.	A	1	Reactor Coolant (CH)	Pipes, Fittings
7)	6/19	10	F	Replaced failed insulator on 3?5 KV transmission line.	A	3	Electric Power (EA)	Other

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	7/3	117	S	Modified off-gas recombiner system and repair generator outboard hydrogen seal.	B	1	Radioactive Waste (MB)	Pipes, Fittings
9)	7/8	15	F	Hydrogen detonation occurred in modified off-gas system. System was removed from service.	A	1	Radioactive Waste (MB)	Recombiners
10)	7/15	10	F	Plugged drain line in off gas caused unstable air ejector operation. Result was low suction pressure trip of feedwater pumps and scram.	A	3	Radioactive Waste (MB)	Pipes, Fittings
11)	8/30	28	S	Training and control rod sequence exchange.	E	1	Reactor (RB)	Control Rods
12)	11/8	77	S	Licensing Exams. Inspection of recirc piping, and tie in of modified off gas system also performed.	E	1		
13)	11/15	142	F	Repair and clean safety relief valves.	A	3	Reactor Coolant (CA)	Valves

NINE MILE POINT I

I. Summary

Description

Location: Scriba, New York
Docket No: 50-220
Reactor Type: Boiling water
Capacity (MWe-Net): 625
Commercial Operation: 12/69
Plant Age: 5.2 Years

Performance

Net electrical energy generated (MWH):	3,296,654
Unit availability factor (%):	70.5
Unit capacity factor (%): (Using Design MWe)	61.7

Outages

Total No.	4	
Forced	3	
Scheduled	1	
Total:	2,584 Hours,	29.5%
Forced	319 Hours,	3.6%
Scheduled	2,265 Hours,	25.9%

Cause:
- Equipment Failure 1
- Maintenance/Test 1
- Refueling 1
- Operational Error 1

Method of shutdown:
- Manual 1
- Manual Scram 1
- Auto Scram 2

II. Highlights

A. General:

On March 30, a 121 day continuous power run was terminated to conduct annual refueling and maintenance. Wet sipping of fuel bundles identified 28 (5.2%) leaking fuel bundles. On July 2, the unit was returned to service and operated for the remainder of the year at about 550 MWe until December 21 when an inadvertent shutdown occurred, so the unit remained shutdown the rest of the period to effect repairs and perform maintenance.

B. Outages:

1. Forced: Three forced outages occurred during the year consuming 319 hours. One outage for 261 hours was to repair a feedwater control valve and to eliminate leakage on 2 electromatic relief valves.

2. Scheduled: One scheduled outage occurred, lasting 2265 hours, for refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	3/30	2265	S	Annual refueling. Activities included 5 year inspection program and HPCI system installation and testing.	C	1	Reactor (RC)	Fuel Elements
2)	10/12	42	F	Operator manually scrammed when he received alarms on loss of some instrumentation in control room.	G	2	Instrumentation & Controls (ID)	Instrumentation & Controls
3)	12/9	16	F	Feedwater control problem due to failure of diaphragm in feedwater valve air relay.	A	3	Reactor Coolant (CH)	Valve Operators
4)	12/21	261	F	Scram due to improper ranging of IRM's by operator. Maintenance performed on feedwater control valve and electromatic relief valve.	B	3	Reactor Coolant (CH)	Valves

I. Summary

Description

Location: Seneca, South Carolina
Docket No: 50-269
Reactor Type: Pressurized Water
Capacity (MWe-Net): 871
Commercial Operation: 7/15/73
Plant Age: 1.7 Years

Performance

Net electrical energy generated (MWH):	3,998,488
Unit availability** factor (%):	60.1
Unit capacity factor (%): (Using Design MWe)	52.4

**This factor plus the outage factor is 99.7% which is a discrepancy of 0.3% and not considered significant. The cause is unverifiable data.

Outages

Total No.	16		
Forced	12		
Scheduled	4		
Total:	3472	Hours,	39.6%*
Forced	403	Hours,	4.6%*
Scheduled	3069	Hours,	35.0%*

Cause:

Equipment Failure	6
Maintenance/Test	5
Refueling	1
License Exam	1
Operational Error	2
External Cause	1

Method of shutdown:

Manual	9
Auto Scram	7

*Data was not verifiable although best available.

II. Highlights

A. General :

Following a 3 week maintenance outage in January, the unit operated at 100% power until May 2 when another maintenance outage started. This outage lasted until June 6 when operations were resumed. Operations continued until October 19 when an extended outage was experienced lasting the remainder of the year.

B. Outages :

1. Forced: There were 12 forced outages during the year consuming 403 hours. None of these outages exceeded 100 hours duration.

2. Scheduled: There were 4 scheduled outages during the year requiring 3069 hours. Three of the 4 outages required over 100 hours, 2 were for general maintenance and one was not determined.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	413	S	General maintenance and turbine acceptance test.	B	1	Steam & Power (HA)	Turbines
2)	1/22	36	F	A differential relay in the main generator control circuitry failed.	A	3	Steam & Power (HA)	Relays
3)	2/11	70	F	2 control rod drive stators failed.	A	1	Reactor (RB)	Control Rod Drives
4)	3/11	12	F	External cause - undetermined.	H	3	Not determinable	Not determinable
5)	4/5	45	F	Clean reactor coolant pump oil coolers and replace oil.	B	1	Reactor Coolant (CB)	Pumps
6)	4/25	42	S	Operator training & license examinations.	E	1	Reactor (RB)	NA
7)	5/2	862	S	General plant maintenance. Included repairs to emergency feedwater pump.	B	1	Reactor Coolant (CX)	Pumps
8)	6/28	40	F	Leakage from instrument line valve packing.	A	1	Reactor Coolant (CB)	Valves
9)	6/29	53	F	Repair control rod indicator tube & stator.	A	1	Reactor (RB)	Control Rod Drives

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	7/5	9	F	Information not available.	G	3	Not determinable	Not determinable
11)	7/5	2	F	Information not available.	G	3	Not determinable	Not determinable
12)	8/23	20	F	Information not available.	B	3	Not determinable	Not determinable
13)	8/26	25	F	Information not available.	A	3	Not determinable	Not determinable
14)	10/5	74	F	Information not available.	B	1	Not determinable	Not determinable
15)	10/16	17	F	Information not available.	A	3	Not determinable	Not determinable
16)	10/19	1752	S	Refueling	C	1	Reactor (RC)	Fuel Elements

OCONEE 2

I. Summary

Description

Location: Seneca, South Carolina
Docket No. 50-270
Reactor Type: Pressurized Water
Capacity (MWe-Net): 871
Commercial Operation: 9/9/74
Plant Age: 1.1 Years

Performance

Net electrical energy generated (MWH):	1,387,526
Unit availability** factor (%):	68.5
Unit capacity factor**(%): (Using Design MWe)	58.2

**Based on experience after commercial operation was declared.

Outages

Total No.	16	
Forced	11	
Scheduled	1	
Undetermined	4	
Total:	6,062	Hours, 69.2%*
Forced	3,940	Hours, 45.0%*
Scheduled	141	Hours, 1.6%*
Undetermined	1,981	Hours, 22.6%

Cause: Equipment Failure 10
Maintenance/Test 2
Undetermined 4 (or more)

Method of shutdown:
Manual 7
Manual Scram 1
Auto Scram 4
Undetermined 4 (or more)

*Values given here are not authoritative. They include unreliable but best available data.

II. Highlights

A. General :

The unit experienced a total of 6062 hours of outages. The unit was in the startup phase of operation most of the year and declared commercial operation on September 9.

B. Outages :

1. Forced: There were 11 known forced outages during the year requiring 3940 hours.

2. Scheduled: The 1 known scheduled outage consumed 141 hours for maintenance to the reactor coolant pumps.

3. Unknown: Four outages for unstated reasons consumed 1981 hours.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	32	F	Repaired valves in the feedwater and condensate systems.	B	1	Steam & Power (HH)	Valves
2)	1/4	441	F	Malfunction in switchyard and foreign object in bottom of reactor vessel.	A	3	Reactor Coolant (CB)	Pumps
3)	1/22	2907	F	Maintenance primarily on reactor coolant pumps. Seals replaced.	A	2	Reactor Coolant (CB)	Pumps
4)	5/XX	91	F	Information not available.	A	3	Not determinable	Not determinable
5)	5/XX	91	F	Information not available.	A	3	Not determinable	Not determinable
6)	5/XX	92	F	Information not available.	A	3	Not determinable	Not determinable
7)	6/2	141	S	Maintenance primarily to reactor coolant pumps.	B	1	Reactor Coolant (CB)	Pumps
8)	6/28	40	F	Information not available.	A	1	Not determinable	Not determinable
9)	6/29	34	F	Information not available.	A	1	Not determinable	Not determinable

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	7/XX	414		Information not available.			Not determinable	Not determinable
11)	8/XX	734		Information not available.			Not determinable	Not determinable
12)	9/XX	279		Information not available.			Not determinable	Not determinable
13)	10/XX	554		Information not available			Not determinable	Not determinable
14)	12/4	8	F	Unidentified reactor coolant leakage greater than 1 gpm.	A	1	Reactor Coolant (CB)	Valves
15)	12/5	10	F	Unidentified reactor coolant leakage greater than 1 gpm.	A	1	Reactor Coolant (CB)	Valves
16)	12/11	194	F	Pressurizer spray valve leakage.	A	1	Reactor Coolant (CB)	Valves

OYSTER CREEK I

I. Summary

Description

Location: Toms River, New Jersey
Docket No.: 50-219
Reactor Type: Boiling water
Capacity (MWe-Net): 650
Commercial Operation: 12/69
Plant Age: 5.3 Years

Performance

Net electrical energy generated (MWH):	3,673,489
Unit availability factor (%):	70.4
Unit capacity factor (%): (Using Design MWe)	67.6

Outages

Total No.	7	
Forced	5	
Scheduled	2	
Total:	2,599	Hours, 29.6%
Forced	1,129	Hours, 12.9%
Scheduled	1,470	Hours, 16.7%

Cause:

Equipment Failure	2
Maintenance/Test	2
Refueling	1
Regulatory Restriction	2
Offsite Power Disturbance	1

Method of shutdown:

Manual	6
Auto Scram	1

II. Highlights

A. General:

For the period January through June, there were seven load reductions (other than the outages). The most significant load reduction resulted in a 4% loss in generation from March 29 to April 13. The cause was a combination of high condensate polisher differential pressure and fuel limitations imposed by GE interim operating recommendations to minimize fuel failures induced by fuel pellet-clad interactions. For the period July - December, there were 21 load reductions in addition to the outages. Eleven of the load reductions were for repairs on leaking main condenser tubes; seven were to accomodate core neutron flux shaping.

B. Outages:

1. Forced: There were five forced outages during the year requiring 1129 hours. The longest outage required 676 hours for maintenance and refurbishing all 65 hydraulic shock and sway arrestors. (This was an extension of a refueling outage.) Two outages requiring 260 hours were for the purpose of repairing a leak on a 1 in. bypass valve around a feedwater shut off valve.

2. Scheduled: Two scheduled outages required 1470 hours. One outage consumed 224 hours to inspect hydraulic shock and sway arrestors in accordance with an AEC directive. The second scheduled outage was for refueling and consumed 1246 hrs.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/12	224	S	Inspect hydraulic shock and sway arrestors in accordance with AEC directive	D	1	Reactor Coolant (CX)	Shock Suppressors
2)	3/7	105	F	Maintenance performed on 3 drywell to torus vacuum breaker valves that failed to pass surveillance test	D	1	Engineered Safety (SA)	Valves
3a)	4/13	1246	S	Refueling	C	1	Reactor (RC)	Fuel Elements
3b)	6/4	676	F	Maintenance including refurbishing all 65 hydraulic shock and sway arrestors	B		Reactor Coolant (CX)	Shock Suppressors
4)	7/13	58	F	Investigated air leak in drywell which prevented adequate containment inertment. Instrument air piping to check valve replaced.	B	1	Engineered Safety (SA)	Pipes, Fittings
5)	9/25	30	F	Generator load rejection scram caused by malfunction of transformer at an offsite substation.	H	3	Electric Power (EA)	Transformers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
6)	10/8	176	F	Shut down because of high drywell unidentified leak rate. The cause was a leaking bonnet gasket on a 1 in. bypass valve around feedwater shutoff valve.	A	3	Reactor Coolant (CH)	Valves
7)	11/11	84	F	The same valve as reported in 6) was leaking. Shut down because of leak rate.	A	1	Reactor Coolant (CH)	Valves

I. Summary

Description	Performance		Outages		
Location: South Haven, Michigan	Net electrical energy generated (MWH):	78,298	Total No.	5	
Docket No: 50-225	Unit availability factor (%):	5.5	Forced	4	
Reactor Type: Pressurized Water	Unit capacity factor (%): (Using Design MWe)	1.3	Scheduled	1	
Capacity (MWe-Net): 821			Total:	8,275	Hours, 94.5%
Commercial Operation: 12/31/71			Forced	8,262	Hours, 94.3%
Plant Age: 3.0 Years			Scheduled	13	Hours, 0.2%

Cause: Equipment Failure 5
Maintenance/Test 1
Method of Shutdown:
Manual 4
Auto Scram 1

II. Highlights

A. General: The plant came on line on October 1 for the first time since August 11, 1973. The period was devoted to repairing the primary to secondary leaks on the steam generator. The plant operated only for the month of October and in November it was again shut down for repair of condenser tube leakage.

B. Outages:

1. Forced: There were 4 forced outages. 5970 hrs of the year were devoted to repairs on the steam generator. 1451 hrs were for repair of leaking condenser tubes. 600 hours were devoted to repair of the turbine, and 236 hours were required for repair of several different items.

2. Scheduled: One scheduled outage for turbine overspeed testing consumed 13 hours.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1a)	1/1	5970	F	Continuation of shutdown began 8/11/73 to repair primary to secondary tube leakage in steam generator.	A	1	Steam and Power (HB)	Heat Exchangers
1b)	9/6	600	F	While rolling turbine, blade damage occurred due to leakage of feed-water heaters.	A		Steam and Power (HA)	Turbines
2)	10/2	13	S	Turbine overspeed test.	B	3	Steam and Power (HA)	Turbines
3)	10/7	5	F	Pilot wire and anti-motoring relay tripped, no deficiencies found.	A	1	Steam and Power (HA)	Relays
4)	10/17	236	F	Repaired CRDM seal, leaking condenser tubes, and pipe fitting leak on PCP seal leak-off line.	A	1	Steam and Power (HC)	Heat Exchangers
5)	11/1	1451	F	Condenser tube leakage.	A	1	Steam and Power (HC)	Heat Exchangers

PEACH BOTTOM 2

I. Summary

Description

Location: Peach Bottom, Pennsylvania
Docket No: 50-277
Reactor Type: Boiling Water
Capacity (MWe-Net): 1,055
Commercial Operation: 7/5/74
Plant Age: 0.9 Years

Performance

Net electrical energy generated (MWH):	3,713,475
Unit availability** factor (%):	90.6
Unit capacity factor**(%): (Using Design MWe)	81.8

**Based on date of commercial operation — July 5.

Outages

Total No.	28		
Forced	18		
Scheduled	10		
Total*	1,249	Hours,	16.8%
Forced	697	Hours,	9.4%
Scheduled	552	Hours,	7.4%

Cause:

Equipment Failure	10
Maintenance/Test	12
Operational Error	6
Op. Tng. & License Exam	1

Method of Shutdown:

Manual	6
Manual Scram	3
Auto Scram	19

*Base is 7440 hours of operation after initial electric power generation on 2/16/74.

II. Highlights

A. General:

Initial electrical power was generated on February 16 which was followed by startup testing. On July 5, the unit was declared commercial. The unit operated near full power for the remainder of the year except for power reductions in all months but September because of condenser tube leaks.

B. Outages:

1. Forced: There were 18 forced outages requiring 697 hours. The longest forced outage was 181 hours to effect repairs to the steam relief valves.

2. <u>Scheduled</u>: There were 10 scheduled outages requiring 552 hours. The longest scheduled outage was 122 hours to perform general maintenance including the control rod discharge filter valves.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/20	4	S	Turbine overspeed trip testing.	B	1	Steam & Power (HA)	Turbines
2)	2/21	9	F	Loose cams on the thrust bearing wear detector instrumentation resulted in turbine trip.	A	3	Steam & Power (HA)	Turbines
3)	3/5	122	S	Maintenance including control rod drive discharge filter valves.	B	3	Reactor (RB)	Control Rod Drives
4)	3/11	46	S	Testing — loss of off site power.	B	3	Electric Power (EA)	NA
5)	3/18	181	F	Maintenance on all main steam relief valves.	A	3	Reactor Coolant (CC)	Valves
6)	3/26	78	F	Failure of relief valves.	A	1	Reactor Coolant (CC)	Valves
7)	4/7	56	F	Turbine trip due to improper draining of a moisture separator drain tank.	G	3	Steam & Power (HB)	NA
8)	4/16	25	S	Startup testing.	B	3	Reactor (RB)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	4/28	52	S	Startup testing.	B	3	Reactor (RB)	NA
10)	5/2	17	F	Steam leak at orifice in recombiner.	A	1	Radio-active Waste (MB)	Recombiners
11)	5/17	32	S	AEC licensing tests for operators.	E	1	Reactor (RB)	NA
12)	5/27	33	S	Testing from 100% power and general maintenance.	B	3	Reactor (RX)	NA
13)	6/1	115	S	Testing followed by scheduled maintenance	B	3	Reactor (RX)	NA
14)	6/10	16	F	Main steam high radiation spike caused trip.	A	3	Reactor Coolant (CC)	Instrumentation & Controls
15)	6/22	46	S	Repair actuators on turbine stop and control valves.	B	1	Steam & Power (HA)	Valve Operators
16)	7/2	9	F	Lost condenser vacuum while repairing leak on water box.	B	3	Steam & Power (HC)	Heat Exchangers
17)	7/22	6	F	Repair steam leak in moisture separator.	B	1	Reactor Coolant (CC)	Vessels, Pressure

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
18)	7/23	20	F	Blown fuse in vessel level controller.	G	3	Reactor Coolant (CH)	Circuit Closers
19)	9/13	46	S	Repair EHC leaks.	B	2	Steam & Power (HA)	Pipes, Fittings
20)	9/16	13	F	Condensate pump casting and shaft failure.	A	3	Steam & Power (HH)	Pumps
21)	9/23	10	F	Operational error — lost instrument nitrogen to MSIV's.	G	3	Reactor Coolant (CD)	Valves
22)	10/15	13	F	Core drilling adjacent to instrument panel caused vibration and scram signal.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
23)	10/16	74	F	Relief valve opened and failed to close.	A	3	Reactor Coolant (CC)	Valves
24)	10/23	9	F	Operational error during surveillance testing caused false scram signal.	G	3	Instrumentation & Controls (IA)	NA
25)	10/24	13	F	Operational error caused trip of feedpump turbine.	G	3	Reactor Coolant (CH)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
26)	11/4	136	F	Repairs to condènser.	A	2	Steam & Power (HC)	Heat Exchangers
27)	11/16	9	F	Condenser vacuum leak.	A	3	Steam & Power (HC)	Heat Exchangers
28a)	11/29	28	F	Recirc valve gland seal leak.	A	2	Reactor Coolant (CB)	Valves
28b)	12/1	31	S	Continuation of 28a but extended for work on drywell.	B		Engineered Safety (SA)	Valves

*Information for this table was obtained from the Semiannual reports, Operating Units Status Report (Grey book) and from telephone communication with the utility.

PILGRIM

I. Summary

Description	Performance		Outages		
Location: Plymouth, Mass.	Net electrical energy generated (MWH):	1,973,033	Total No.	7	
Docket No.: 50-293			Forced	7	
Reactor Type: Boiling water	Unit availability factor (%):	39.2	Scheduled	0	
Capacity (MWe-Net): 664	Unit capacity factor (%): (Using Design MWe)	33.6	Total:	5,326 Hours,	60.8%
Commercial Operation: 12/72			Forced	5,326 Hours,	60.8%
Plant Age: 2.5 Years			Scheduled	0 Hours,	0%

Cause:
- Equipment Failure 5
- Maintenance/Test 1
- Regulatory Restriction 1

Method of shutdown:
- Manual 3
- Manual Scram 3
- Auto Scram 1

II. Highlights

A. General:

For the first seven months of the year, the plant shutdown which had started on 12-28-73, continued. The primary purpose of the shutdown was AEC intervention concerning the replacement of twenty 7 × 7 fuel bundles with 8 × 8 fuel bundles. The reactor was also refueled during this time.

B. Outages:

1. Forced: Seven forced outages occurred; two exceeded 100 hours (115 and 102 hours) both for the replacement of recirculation pump seals.

2. Scheduled: There were no scheduled outages during the report period.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	4987	F	Continuation of shutdown began 12/28/73. Legal intervention by AEC to shut down until any fuel channel-to-poison-curtain interaction damage had been repaired and corrected. Refueling accomplished.	D	1	Reactor (RC)	Fuel Elements
2)	7/28	3	F	To perform turbine overspeed test.	B	1	Steam & Power (HA)	Turbines
3)	9/17	115	F	Replacement of recirculation pump seal and inspection of bypass piping.	A	2	Reactor Coolant (CB)	Pumps
4)	10/25	66	F	Replacement of recirculation pump seal.	A	2	Reactor Coolant (CB)	Pumps
5)	11/2	33	F	Replacement of two safety-relief valves.	A	1	Reactor Coolant (CA)	Valves
6)	11/11	20	F	An accumulation of salt spray on the insulators in the 345 KV switchyard caused flash over and subsequent loss of the unit.	A	3	Electric Power (EA)	Other
7)	12/13	102	F	Replacement of seal on recirculation pump.	A	2	Reactor Coolant (CB)	Pumps

POINT BEACH 1

I. Summary

Description

Location: Two Creeks, Wisconsin
Docket No: 50-301
Reactor Type: PWR
Capacity (MWe-Net): 497
Commercial Operation: 12/21/70
Plant Age: 4.2 Years

Performance

Net electrical energy generated (MWH):	3,142,055
Unit availability factor (%):	81.5
Unit capacity factor (%): (Using Design MWe)	76.2

Outages

*Total No.	8		
Forced	5		
Scheduled	3		
Total:	1,626	Hours,	18.5%
Forced	61	Hours,	0.7%
Scheduled	1,565	Hours,	17.8%

Cause:

Equipment Failure	4
Maintenance/Test	2
Refueling	1
Operational Error	1

Method of Shutdown:

Manual	2
Manual Scram	1
Auto Scram	5

II. Highlights

A. General:

The unit operated on load follow until the refueling outage on April 6. From January through June 30, unit availability was 65.2%. For the remainder of the year the unit operated on load follow at an average capacity factor of 92.13% and unit availability of 97.4%.

B. Outages:

1. Forced: There were 5 forced outages during the year consuming 61 hours. The longest outage was for 49 hours to repair a main steam stop valve bypass valve.

2. Scheduled: There were 3 scheduled outages consuming 1565 hours. The longest outage was for 1205 hours to conduct refueling. One outage for 295 hours was required to repair damage to the turbine blades caused by a broken spacer washer.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/11	3	F	Low-low steam gen. level -- Loss of inverter in instrument power.	A	3	Electric Power (ED)	Generators
2)	1/18	4	F	Low steam gen. & steam flow - Feed mismatch - Loss of inverter in instrument power supply.	A	3	Electric Power (ED)	Generators
3)	2/3	5	F	Operations testing fuses in rod control system.	A	3	Reactor (BB)	Instrumentation & Controls
4)	4/6	1205	S	Refueling, inspection of turbine, mis. repairs.	C	1	Reactor (RC)	Fuel Elements
4a)	5/26	295	S	Repair of turbine - Spacer washer broke loose and damaged blades.	B		Steam & Power (HA)	Turbines
5)	8/2	16	S	Test of crossover steam dump system for turbine overspeed.	B	3	Steam & Power (HB)	NA
6)	9/25	1	F	Tripped due to testing error during periodic analog input test.	G	3	Instrumentation & Controls (IA)	NA
7)	10/4	49	S	Repair 3" main steam stop valve bypass valve.	B	2	Steam & Power (HB)	Valves
8)	11/23	48	F	Blowdown steam generator to alleviate secondary water chemistry problem.	A	1	Steam & Power (HB)	Heat Exchangers

POINT BEACH 2

I. Summary

Description

Location: Green Bay, Wisconsin
Docket No: 50-301
Reactor Type: PWR
Capacity (MWe-Net): 497
Commercial Operation: 4/73
Plant Age: 2.4 Years

Performance

Net electrical energy generated (MWH):	3,178,408
Unit availability factor (%):	81.0
Unit capacity factor (%): (Using Design MWe)	76.9

Outages

Total No.	4	
Forced	0	
Scheduled	4	
Total	1,661	Hours, 19.0%
Forced	0	Hours, 0.0%
Scheduled	1,661	Hours, 19.0%

Cause: Maintenance/Test 3
Refueling 1
Method of Shutdown:
Manual 2
Auto Scram 2

II. Highlights

A. General:

For the first 6 months of the year, the unit operated on load follow with a capacity factor of 94.72%. During the last six months of the year a refueling outage took place reducing the capacity factor for this period to 59.25%.

B. Outages:

1. Forced: There were no forced outages during the year.

2. Scheduled: There were four scheduled outages for the year; the longest for 1566 hours was to refuel.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	7/1	88	S	To correct explosive gas condition in B phase transformer.	B	1	Electric Power (EA)	Transformers
2)	10/16	1566	S	Refueling	C	1	Reactor (RC)	Fuel Elements
3)	12/21	3	S	Turbine overspeed testing.	B	3	Steam & Power (HA)	NA
4)	12/27	4	S	Test of crossover steam dump system for turbine overspeed.	B	3	Steam & Power (HA)	NA

PRAIRIE ISLAND I

I. Summary

Description

Location: **Red Wing, Minnesota**
Docket No: 50-282
Reactor Type: Pressurized Water
Capacity (MWe-Net): 520
Commercial Operation: 12/5/73
Plant Age: 1.1 Years

Performance

Net electrical energy generated (MWH):	1,432,750
Unit availability factor (%):	43.9
Unit capacity factor (%): (Using Design MWe)	31.5

Outages

Total No.	30		
Forced	20		
Scheduled	10		
Total:	4,912	Hours,	56.1%
Forced	2,746	Hours,	31.3%
Scheduled	2,166	Hours,	24.8%

Cause:
- Equipment Failure 17
- Maintenance/Test 10
- Operational Error 3

Method of Shutdown:
- Manual 8
- Manual Scram 3
- Auto Scram 19

II. Highlights

A. General:

The unit began the year with a continuation of a shutdown started in 1973 for the purpose of modifying the steam generators and to repair the turbine. In February, the shutdown ended and operations proceeded routinely until March 9 when an outage was required to repair the turbine. Turbine blade damage had occurred due to vibration. On April 27 another outage was started, and lasted into July, to effect turbine blade repair. In September, another long outage was required to reblade the turbine.

B. Outages:

1. Forced: There were 20 forced outages during the year consuming 2746 hours. The two longest outages to reblade the turbine consumed 2294 hours.

2. Scheduled: There were 10 scheduled outages during the year consuming 2746 hours. One outage lasted 1180 hours to reblade the turbine.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	893	S	Continuation of shutdown began in 1973 for steam generator modification and turbine repair.	[illegible]	1	Steam & Power (HA)	Heat Exchangers
2)	2/8	2	S	To obtain data on feedwater pump.	B	1	Steam & Power (HH)	Pumps
3)	2/15	5	F	Repaired oil line in the speed increaser of a feedwater pump.	A	1	Steam & Power (HH)	Pipes, Fittings
4)	2/19	15	S	Test — low reactor coolant pump flow.	B	3	Reactor Coolant (CB)	NA
5)	2/22	9	S	Test — loss of off-site power.	B	3	Electric Power (EA)	NA
6)	2/23	9	S	Test — negative flux rate trip.	B	3	Instrumentation & Controls (IA)	NA
7)	2/24	11	S	Test — negative flux rate trip.	B	3	Instrumentation & Controls (IA)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	3/1	75	F	Repair leak in feedwater pump casing.	A	1	Steam & Power (HH)	Pumps
9)	3/6	8	F	Operator error in valve lineup for warmup of feedwater pump.	G	3	Steam & Power (HH)	NA
10)	3/9	704	F	Turbine blade failure due to vibration.	A	1	Steam & Power (HA)	Turbines
11)	4/12	37	F	Feedwater regulating valve closed when its position cover was opened.	A	3	Steam & Power (HH)	Valves
12)	4/17	13	F	Defective fuse resulted in drop of control rod assembly.	A	3	Instrumentation & Controls (IA)	Circuit Closers/ Interruptors
13)	4/19	75	F	Repair leaking feedwater check valve shaft seal.	A	2	Steam & Power (HH)	Valves
14)	4/25	3	F	Spurious instrument malfunction.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	4/27	1590	F	Turbine blade failure.	A	2	Steam & Power (HA)	Turbines
16)	7/5	15	F	Spurious MSIV closure.	A	3	Reactor Coolant (CD)	Valves
17)	7/7	9	S	Test – turbine trip.	B	3	Steam & Power (HA)	NA
18)	7/12	65	F	Repair leaking feed-water check valve shaft seal.	A	3	Steam & Power (HH)	Valves
19)	7/27	21	S	Test – generator trip from 100% power.	B	3	Steam & Power (HA)	NA
20)	8/16	87	F	Installed new shafts in MSIV's.	A	1	Reactor Coolant (CD)	Valves
21)	8/28	13	F	Turbine trip from MSIV movement.	A	3	Reactor Coolant (CD)	Valves
22)	8/29	10	F	Turbine trip from MSIV movement.	A	3	Reactor Coolant (CD)	Valves

DETAILS OF PLANT OUTACES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
23)	8/30	5	F	Turbine trip from MSIV movement.	A	3	Reactor Coolant (CD)	Valves
24)	9/5	1180	S	Reblade turbine.	B	1	Steam & Power (HA)	Turbines
25)	10/26	17	S	Test — turbine trip from 100% power for telemetry testing.	B	2	Steam & Power (HA)	Turbines
26)	10/27	10	F	Checklist error — condenser vacuum lost.	G	3	Steam & Power (HC)	NA
27)	11/7	11	F	Spurious turbine trip from spike in EH control system.	A	3	Steam & Power (HA)	Instrumentation & Controls
28)	11/8	3	F	Condenser vacuum lost — both circulating water pumps were accidentally tripped.	G	3	Steam & Power (HF)	NA
29)	11/29	6	F	Plugged condenser tube leak.	A	1	Steam & Power (HC)	Heat Exchangers
30)	12/26	11	F	Solenoid valve failed causing one feedwater regulating valve to close.	A	3	Steam & Power (HH)	Valve Operators

QUAD CITIES 1

I. Summary

Description	Performance		Outage		
Location: Cordova, Illinois	Net electrical energy generated (MWH):	3,562,941	Total No.	19	
Docket No: 50-254	Unit availability* factor (%):	61.9	Forced	14	
Reactor Type: Boiling Water	Unit capacity factor (%):	50.8	Scheduled	5	
Capacity (MWe net): 800	(Using Design MWe)		Total	3,396	Hours, 38.7%*
Commercial Operation: 2/73			Forced	564	Hours, 6.4%
Plant Age: 2.7 Years			Scheduled	2,832	Hours, 32.3%

*These two factors exceed 100% by 0.6% because of a 60 hr discrepancy in the Details table. It was allowed because it was less than 1%.

Cause: Equipment Failure 13
Maintenance/Test 5
Refueling 1

Method of Shutdown:
Manual 15
Auto Scram 4

II. Highlights

A. General:

Nineteen outages occurred during the year consuming 3396 hours. Thirteen of these were due to equipment failure, five were due to maintenance or testing, and there was a refueling outage.

B. Outages:

1. Forced: There were 14 forced outages consuming 564 hours. The longest forced outage required 147 hours to repair condenser tube leaks.

2. Scheduled: There were 5 scheduled outages requiring 2832 hours. A refueling outage consumed 2687 hours.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	147	F	To repair condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
2)	1/7	1	S	Shutdown margin tests.	B	1	Reactor (RB)	Control Rod Drives
3)	1/20	22	F	To repair air ejectors — condenser vacuum was low.	A	1	Steam & Power (HC)	Other
4)	1/31	12	F	EHC oil leak.	A	1	Steam & Power (HA)	Pipes, Fittings
5)	3/1	33	S	Tie in auxiliary transformer for spray canal.	B	1	Auxiliary Water (WE)	Transformers
6)	3/2	6	F	Explosion in off gas holdup line.	A	3	Radioactive Waste (MB)	Pipes, Fittings
7)	3/24	12	F	Turbine stop valve closed.	A	3	Steam & Power (HA)	Valves
8)	3/31	2687	S	Refueling	C	1	Reactor (RC)	Fuel Elements
9)	7/22	43	S	Control rod sequence changeover.	B	1	Reactor (RB)	Control Rods

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	7/27	30	F	Steam leak at turbine control valve.	A	1	Steam & Power (HA)	Valves
11)	8/10	12	F	Repair EHC fluid leak on control valve.	A	1	Steam & Power (HA)	Pipes, Fittings
12)	10/4	79	F	Inspect recirc system piping.	B	1	Reactor Coolant (CB)	Pipes, Fittings
13)	10/8	8	F	Load reject signal occurred during turbine testing.	A	3	Steam & Power (HA)	NA
14)	10/10	66	F	Off-gas explosion — replaced off gas filters.	A	1	Radioactive Waste (MB)	Filters
15)	10/13	124	F	Vibration damage on moisture separator drain tank.	A	1	Steam & Power (HB)	Vessels, Pressure
16)	11/2	34	F	Repair packing leak on core spray isolation valve.	A	1	Engineered Safety (SF)	Valves
17)	11/4	2	F	Repair steam leak on restricting orifice.	A	1	Reactor Coolant (CC)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
18)	11/4	10	F	Lost essential services bus.	A	3	Electric Power (ED)	NA
19)	12/14	68	S	Repair leaks on main steam line.	B	1	Reactor Coolant (CC)	Pipes, Fittings

*Data was furnished by the plant technical staff, supplemented by the Operating Unit Status Report and Semiannual reports. The outage hours tabulated above exceed that which would correspond to the indicated generator on-line hours by 60 hours. Efforts to find the discrepancy were unsuccessful.

QUAD CITIES 2

I. Summary

Description

Location: Cordova, Illinois
Docket No: 50-265
Reactor Type: Boiling Water
Capacity (MWe-Net): 800
Commercial Operation: 3/73
Plant Age: 2.6 Years

Performance

Net electrical energy generated (MWH)	4,469,705
Unit availability factor (%)	82.6
Unit capacity factor (%) (Using Design MWe)	63.8

Outages

Total No.	20		
Forced	16		
Scheduled	4		
Total:	1,529	Hours,	17.4%
Forced	1,044	Hours,	11.9%
Scheduled	485	Hours,	5.5%

Cause

Equipment Failure	16
Maintenance/Test	2
Refueling	1
Change Rod Sequence	1

Method of shutdown:

Manual	12
Auto Scram	8

II. Highlights

A. General :

Unit operated at about 750 MWe except for periods twice a week when power was reduced to 550 MWe to perform MSIV surveillance. During the last quarter, electrical output was reduced because of high off-gas release rates. In December, refueling began.

B. Outages :

1. Forced: There were 16 forced outages during the year consuming 1,044 hours. The longest outage occurred in August, in which 288 hours were expended to repair weld failures on recirculation system pump discharge valve bypass piping.

2. Scheduled: Four scheduled outages consumed 485 hours, of which the longest was for 205 hours to tie in the modified off-gas system piping. At the end of the year, 202 hours were allotted to refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/5	36	F	Repaired steam leaks on steam seal valve.	A	1	Steam & Power (HD)	Valves
2)	1/25	69	S	Inspect all hydraulic shock suppressors.	B	1	Reactor Coolant (CX)	Shock Suppressors
3)	3/23	205	S	Tie in modified off-gas system piping.	B	1	Radio-active Waste (MB)	Pipes, Fittings
4)	4/12	56	F	Repaired recirculation pump motor.	A	3	Reactor Coolant (CB)	Motors
5)	5/24	36	F	Repaired steam leak on turbine control valve.	A	1	Reactor Coolant (CC)	Valves
6)	6/2	11	F	Excessive leakage on feedwater low flow regulating valve.	A	3	Reactor Coolant (CH)	Valves
7)	6/10	173	F	Severed feedwater low flow line to regulating valve.	A	3	Reactor Coolant (CH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	6/18	136	F	Seal leak on reactor water cleanup check valve.	A	1	Reactor Coolant (CG)	Valves
9)	6/30	22	F	Turbine overspeed relay malfunction.	A	3	Steam & Power (HA)	Relays
10)	7/1	6	F	Severed instrument air line on feedwater minimum flow valve.	A	1	Reactor Coolant (CH)	Pipes, Fittings
11)	7/1	35	F	Failure of control valves to open with bypass valves closed.	A	1	Steam & Power (HA)	Valves
12)	8/31	18	F	Recirculation MG set oil pump tripped.	A	3	Reactor Coolant (CB)	Pumps
13)	8/31	150	F	Broken instrument air line on low flow feedwater regulating valve.	A	3	Reactor Coolant (CH)	Pipes, Fittings
14)	9/13	288	F	Weld failures on recirc pump discharge valve bypass piping.	A	1	Reactor Coolant (CB)	Pipes, Fittings
15)	9/26	11	F	Spurious trip signal during testing.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
16)	10/14	23	F	Repair packing leak on RHRS valves.	A	1	Reactor Coolant (CF)	Valves
17)	10/17	6	F	Flux spike when recirc MG set oil pump tripped and standby pump started.	A	3	Reactor Coolant (CB)	Pumps
18)	11/16	9	S	Changed control rod pattern sequence.	H	1	Reactor (RB)	Control Rods
19)	12/22	37	F	A recirc pump tripped. Other pump out of service.	A	1	Reactor Coolant (CB)	Pumps
20)	12/23	202	S	Started refueling outage.	C	1	Reactor (RC)	Fuel Elements

H. B. ROBINSON 2

I. Summary

Description

Location: Hartsville, S.C.
Docket No: 50-261
Reactor Type: Pressurized Water
Capacity (MWe-Net): 707
Commercial Operation: 3/7/71
Plant Age: 4.3 Years

Performance

Net electrical energy generated (MWH):	4,813,207
Unit availability factor (%):	83.3
Unit capacity factor (%): (Using Design MWe)	82.6

Outages

Total No.	19		
Forced	15		
Scheduled	4		
Total:	1,460	Hours,	16.7%
Forced	128	Hours,	1.5%
Scheduled	1,332	Hours,	15.2

Cause:

Equipment Failure	8
Maintenance/Test	5
Operational Error	3
Refueling	1
Other	2

Method of shutdown:

Manual	7
Manual Scram	2
Scram	10

II. Highlights

A. General :

A total of 19 outages occurred during 1974; 3 were directly related to problems with steam generators; 5 were related to problems with instrumentation; 2 were related to problems with the turbine valves; 2 were related to problems with electrical equipment; and 3 were related to problems with the control rods.

B. Outages :

1. Forced: Fifteen forced outages occurred; only one exceeded 16 hours in duration (48 hrs for work on MSR heater drains).

2. Scheduled: Of the four scheduled outages, only one was of considerable duration (1235 hours for refueling and plant maintenance).

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/9	2	F	High level in steam generator while increasing load after weekly turbine valve test.	G	3	Steam & Power (HB)	NA
2)	1/9	2	F	High level in steam generator while recovering from previous trip.	G	3	Steam & Power (HB)	NA
3)	1/9	2	F	High level in steam generator while recovering from previous trip.	G	3	Steam & Power (HB)	NA
4)	1/20	15	S	Repair Bus Ducts to reduce heating of supports between B and C phases. Repair secondary leaks.	B	1	Electric Power (EA)	Electrical Conductors
5)	1/26	3	F	Turbine trip caused by voltage regulator failing when a fuse blew.	A	3	Electric Power (ED)	Circuit Closers
6)	2/23	13	F	While shutting down for maintenance, received intermediate range trip due to large current caused by power produced in top of core.	H	3	Instrumentation & Controls (IA)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	2/24	(14 min)	F	When returning to power received steam flow feed flow mismatch when main steam isolation valve was opened.	H	3	Steam & Power (HB)	NA
8)	2/24	3	S	Manually tripped to verify control rod bottom bistable operation.	B	1	Reactor (RB)	NA
9)	5/6	1235	S	Refuel and general plant maintenance.	C	1	Reactor (RC)	Full Elements
10)	7/18	16	F	Dropped shutdown rod and and could not relatch. Shut down to inspect rod control cables on vessel head.	A	1	Reactor (RB)	Control Rods
11)	7/19	4	F	Radial tilt greater than specified limits.	A	1	Reactor (RB)	Control Rods
12)	8/11	11	F	Loss of instrument bus.	A	3	Electric Power (ED)	NA
13)	8/27	12	F	Loss of instrument bus causing runback and turbine trip.	A	3	Electric Power (ED)	NA
14)	8/31	48	F	Repaired leaks in moisture separator reheater and feedwater heater drains.	B	1	Steam & Power (HH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	9/22	2	F	Turbine valves closed during valve test causing reactor trip.	B	2	Steam & Power (HB)	Valves
16)	9/30	7	F	Spurious indication on loop flow while one of the transmitters was out of service.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
17)	10/11	79	S	To repair items on secondary system, clean condenser tubes, and plug condenser tubes.	B	1	Steam & Power (HA)	Heat Exchangers
18)	11/4	3	F	Loss of instrument buses caused reactor runback and subsequent high level in steam generator.	A	3	Electric Power (ED)	Electrical Conductors
19)	11/20	3	F	Manual trip after turbine valves closed.	A	2	Steam & Power (HA)	Valves

SAN ONOFRE I

I. Summary

Description

Location: San Clemente, California
Docket No: 50-206
Reactor Type: Pressurized Water
Capacity (MWe-Net): 430
Commercial Operation: 1/68
Plant Age: 7.5 Years

Performance

Net electrical energy generated (MWH):	3,145,109
Unit Availability factor (%):	86.1
Unit capacity factor (%): (Using Design MWe)	83.5

Outages

Total No.	8		
Forced	4		
Scheduled	4		
Total:	1,220	Hours,	13.9%
Forced	583	Hours,	6.7%
Scheduled	637	Hours,	7.2%

Cause:

Equipment Failure	4
Maintenance/Test	3
Op. Eng. & License Exam	1

Method of Shutdown:

Manual	3
Manual Scram	1
Auto Scram	3
Undesignated	1

II. Highlights

A. General:

The first 516 hours of the year was a continuation of a shutdown begun in 1973 to repair steam generator and reheater tube leaks. Operation during the year was at 450 MWe.

B. Outages:

1. Forced: There were four forced outages consuming 583 hours; 516 hours were a continuation of a 1973 shutdown due to turbine blade failure.

2. Scheduled: There were four scheduled outages consuming 637 hours; 546 hours were for repair of steam generator and reheater tube leaks.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	516	F	This outage began on Oct. 21, 1973 with a turbine blade failure.	A	3	Steam & Power (HA)	Turbines
2)	1/23	1	S	Turbine overspeed test.	B		Steam & Power (HA)	Turbines
3)	4/27	546	S	Repair of steam generator and reheater tube leaks and repair of leaking pressurizer safety valve.	B	1	Steam & Power (HB)	Heat Exchangers
4)	7/7	55	F	Trip from indicated overpower condition caused by water intrusion into detectors of two power range channels due to gasket failure on cooler of rod drive cooling fan.	A	3	Reactor (RB)	Heat Exchangers
5)	8/20	5	F	Spurious trip on indicated pressurizer high level while testing level channels.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
6)	9/4	24	S	Repair power range detection package and install axial-offset monitoring system.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls
7)	10/18	66	S	Reheaters were leak tested and operator licensing exams were given.	E	1	Reactor (RB)	NA
8)	10/21	7	F	While resuming operation from 10/18 shutdown control bank No. 2 slipped into core. No cause found.	A	2	Reactor (RB)	Control Rod Drives

SURRY 1

I. Summary

Description

Location: Surry, Virginia
Docket No: 50-280
Reactor Type: Pressurized Water
Capacity (MWe net): 788
Commercial Operation: 12/72
Plant Age: 2.5 Years

Performance

Net electrical energy generated (MWH):	3,318,073
Unit availability factor (%):	54.8
Unit capacity factor (%): (Using Design MWe)	48.1

Outages

Total No.	21	
Forced	19	
Scheduled	2	
Total	3,961	Hours, 45.2%
Forced	540	Hours, 6.1%
Scheduled	3,421	Hours, 39.1%

Cause: Equipment Failure 9
Maintenance/Test 3
Refueling 1
Operational Error 8

Method of Shutdown:
Manual 1
Auto Scram 20

II. Highlights

A. General:

The unit was shut down the first 2 1/2 months of the year to repair the reactor coolant pumps and system isolation valves. The unit operated satisfactorily until shut down in October for refueling.

B. Outages:

1. Forced: There were 19 forced outages during the year of which 5 were attributed to steam flow/feed flow mismatch. The longest outage was for 263 hours caused by failure of the generator exciter.

2. Scheduled: Two scheduled outages consumed 3421 hours. Replacement of the reactor coolant pump shafts required 1784 hours, and refueling consumed 1637 hours.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	1784	S	Reactor coolant pump shaft replacement.	B	3	Reactor Coolant (CB)	Pumps
2)	3/20	23	F	Malfunction of EHC system.	A	3	Steam & Power (HA)	Mechanical Function Units
3)	4/6	8	F	Fuses blew in over-temp delta T circuit.	A	3	Instrumentation & Controls (IA)	Circuit Closers
4)	4/6	10	F	Operator error. Steam flow/feed flow mismatch.	G	3	Steam & Power (HH)	NA
5)	4/25	38	F	Broken air line to feed-water regulator valve.	A	3	Steam & Power (HH)	Pipes, Fittings
6)	5/26	71	F	Feed/steam mismatch.	G	3	Steam & Power (HH)	NA
7)	6/9	11	F	Loss of power to EH control system.	A	3	Steam & Power (HA)	Electrical Conductors
8)	6/10	2	F	Feed/steam flow mis-match.	G	3	Steam & Power (HH)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	6/20	15	F	Loss of instrument air.	A	3	Auxiliary Process (PA)	Valves
10)	6/21	2	F	Loss of instrument air.	A	3	Auxiliary Process (PA)	Valves
11)	6/22	2	F	Operator error on feed system.	G	3	Steam & Power (HH)	NA
12)	7/3	263	F	Failure of generator exciter.	A	3	Steam & Power (HA)	Generators
13)	7/11	1	F	Steam/feed flow mismatch.	G	3	Steam & Power (HH)	NA
14)	7/14	1	F	Steam/feed flow mismatch.	G	3	Steam & Power (HH)	NA
15)	7/16	2	F	Gen. differential lockout when efforts made to set field forcing.	C	3	Steam & Power (HA)	Generators
16)	9/4	48	F	Primary leakage through valve packing gland.	A	1	Reactor Coolant (CB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	9/29	2	F	Turbine trip from feed-water heater. Level setpoints reevaluated.	B	3	Steam & Power (HH)	Instrumentation & Controls
18)	9/29	2	F	Lo Steam Generator level. Operator error.	G	3	Steam & Power (HB)	NA
19)	9/29	20	F	Switch failed to make up on MOV. Safety injection occurred.	A	3	Engineered Safety (SF)	Circuit Closers
20)	10/3	19	F	Safety injection occurred during test of isolation valve.	B	3	Engineered Safety (SF)	Valves
21)	10/24	1637	S	Refueling	C	3	Reactor (RC)	Fuel Elements

*Information for this table was obtained from the plant Semiannual reports, the Operator Units Status Reports, and from data supplied by the plant staff via telephone to NSIC.

SURRY 2

I. Summary

Description

Location: Gravel Neck, Virginia
Docket No: 50-281
Reactor Type: Pressurized Water
Capacity (MWe net): 788
Commercial Operation: 5/1/73
Plant Age: 1.8 Years

Performance

Net electrical energy generated (MWH):	2,634,573
Unit availability* factor (%):	44.8
Unit capacity factor (%): (Using Design MWe)	38.2

*These two factors total 100.8% but the discrepancy could not be resolved.

Outages

Total No.	11		
Forced	7		
Scheduled	4		
Total	4,905	Hours,	56.0%*
Forced	2,987	Hours,	34.1%
Scheduled	1,918	Hours,	22.0%

Cause:

Equipment Failure	6
Maintenance/Tests	2
Op. Tng. & License Exam	2
Operational Error	1

Method of Shutdown:

Manual	3
Manual Scram	1
Auto Scram	7

II. Highlights

A. General:

The unit operated satisfactorily until April 13 when a 1557 hour shutdown was required to replace the reactor coolant pump shafts. A major outage (313 hours) was again required in June to repair the main steam system non-return valves. In September a shutdown was required because of high turbine vibration caused by broken blades. This shutdown lasted through the end of the year.

B. Outages:

1. Forced: There were seven forced outages during the year consuming 2987 hours. The longest forced outage was for 2787 hours because of turbine vibration caused by broken blades.

2. Scheduled: There were four scheduled outages during the year consuming 1918 hours. Of this amount, 1557 hours were consumed replacing the shafts on the reactor coolant pumps.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	8	F	Repair leaks in main condenser.	A	3	Steam & Power (HC)	Heat Exchangers
2)	2/3	9	S	During examination, operator caused feed/steam flow mismatch	E	3	Steam & Power (HH)	NA
3)	4/13	1557	S	Reactor coolant pump and system isolation valve maintenance.	B	3	Reactor Coolant (CB)	Pumps
4)	6/22	39	S	AEC operator exams.	E	1	Reactor (RB)	NA
5)	6/25	313	S	Repair main steam non-return valves.	B	1	Steam & Power (HB)	Valves
6)	7/8	12	F	Solenoid failed on main steam stop valve.	A	3	Steam & Power (HB)	Valve Operators
7)	7/9	1	F	Operator failed to reset P-10 when load exceeded 10%.	G	3	Instrumentation & Controls (IA)	NA
8)	8/3	85	F	Main [illegible]eam trip va[illegible] close[illegible]	A	3	Steam & Power (HB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	8/12	1	F	Overpower delta T caused trip.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
10)	8/18	93	F	Excessive reactor coolant leakage in valve packing.	A	1	Reactor Coolant (CB)	Valves
11)	9/4	2787	F	Turbine vibration due to broken blades.	A	2	Steam & Power (HA)	Turbines

*Data for this table was obtained from the plant Semiannual reports, the Operating Units Status Reports, and from the plant staff via telephone to NSIC.

THREE MILE ISLAND I

I. Summary

Description	Performance		Outages			
Location: Goldsboro, Pennsylvania	Net electrical energy generated (MWH):	1,577,812	Total No.	2		
Docket No: 50-289			Forced	1		
Reactor Type: Pressurized Water	Unit availability factor (%):	88.1	Scheduled	1		
Capacity (MWe-Net): 792			Total	346	Hours,	11.9%*
Commercial Operation: 9/2/74	Unit capacity factor (%): (Using Design MWe)	86.0	Forced	97	Hours,	3.3%*
Plant Age: 0.5 Years			Scheduled	249	Hours,	8.6%*

Cause: Equipment Failure 1
Maintenance/Test 1
Method of Shutdown:
Manual 2

*Data is based on date of first commercial operation — September 2.

II. Highlights

A. General:

Initial Electrical Power generation occurred on June 9 and commercial operation was declared September 2. Operations were uninterrupted in September and December, and at essentially full power in December.

B. Outages:

1. Forced: There was one forced outage lasting 97 hours to replace a faulty control rod drive motor stator winding.

2. Scheduled: There was one scheduled outage for 249 hours to repair leaking relief valves on the pressurizer.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	10/20	249	S	Repaired leaking relief valves on the pressurizer.	B	1	Reactor Coolant (CB)	Valves
2)	11/17	97	F	Replaced a faulty control rod drive motor stator winding which had caused a dropped rod condition.	A	1	Reactor (R3)	Control Rod Drives

*Data for this table covers the period from the date of commercial operation, Sept. 2, through December 31, 1974.

TURKEY POINT 3

I. Summary

Description

Location: Florida City, Florida
Docket No: 50-250
Reactor Type: Pressurized Water
Capacity (MWe-Net): 666
Commercial Operation: 12/72
Plant Age: 2.2 Years

Performance

Net electrical energy generated (MWH):	3,623,905
Unit availability factor (%):	69.9
Unit capacity factor (%): (Using Design MWe)	62.1

Outages

Total No.	32	
Forced	25	
Scheduled	7	
Total:	2,640 Hours,	30.1%
Forced	198 Hours,	2.2%
Scheduled	2,442 Hours,	27.9%

Cause:

Equipment Failure	11
Maintenance/Test	13
Refueling	1
Administration	1
Operational Error	5
Other	1

Method of Shutdown:

Manual	16
Manual Scram	5
Auto Scram	11

II. Highlights

A. General:

Unit 3 reached rated power of 2200 MWT for the first time on March 9. During the second half of the year, the unit was base loaded at rated power. The first refueling of Unit 3 took place from October 6 to December 15.

B. Outages:

1. Forced: Twenty-five forced outages consumed 198 hours. All of these outages were of relatively short duration, the longest was 31 hours to repair tube leaks in a moisture separator reheater.

2. Scheduled: Seven scheduled outages consumed 2442 hours. The longest scheduled outage was for 1704 hours to perform refueling. 289 hours were expended to inspect seismic restraints, modify feedwater lines, and repair a turbine control valve.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/6	11	F	A power source was inadvertently shorted which caused a loss of power to the instrument bus. A high steam generator delta P resulted along with a reactor trip.	G	3	Electric Power (ED)	Electrical Conductors
2)	2/7	6	F	Voltage spikes on vital instrument bus caused voltage spikes on auto control system.	A	2	Electric Power (ED)	
3)	2/7	4	F	Steam flow greater than feed flow with low steam generator level during startup.	G	3	Steam & Power (HB)	N.A.
4)	2/11	1	F	Voltage spike on vital instrument bus caused a spurious low pressurizer pressure signal.	A	3	Electric Power (ED)	Electrical Conductors
5)	3/1	12	F	Manual turbine trip associated with load transient caused by unit 4 trip.	F	2	Electric Power (EA)	NA
6)	3/18	289	S	Inspection of seismic restraints, feedwater line modification, and turbine control valve repair.	B	2	Reactor Coolant (CX)	Shock Suppressors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	4/5	7	F	Dropped control rod, accompanied by loss of rod control. Capacitor failure.	A	2	Reactor (RB)	Instrumentation & Controls
8)	4/6	9	F	A level switch failure indicated an off-normal condition in reactor coolant pump motor oil reservoir.	A	1	Reactor Coolant (CB)	Circuit Closers
9)	4/23	17	F	Loss of instrument power supply caused steam flow/ feed flow mismatch.	A	3	Electric Power (ED)	Instrumentation & Controls
10)	4/25	2	F	Generator breaker system inadvertently energized. Several control system wiring errors	G	3	Steam & Power (HA)	Electrical Conductors
11)	5/11	31	S	Repair tube leak in moisture separator reheater and other maintenance.	B	1	Steam & Power (HB)	Heat Exchangers
12)	6/5	6	F	A capacitor failure in the rod control system caused a loss of power to 4 control rod drive mechanisms.	A	1	Reactor (RB)	Control Rod Drives
13)	6/7	145	S	Inspection of seismic restraints and other maintenance.	B	1	Reactor Coolant (CX)	Shock Suppressors
14)	6/15	10	S	Nuclear instrumentation system modification.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	6/17	2	F	Reactor trip breaker was opened in error.	G	2	Electric Power (ED)	NA
16)	6/24	20	F	Malfunction in pressurizer spray valve control system. Reactor trip from decreasing pressurizer pressure.	A	3	Reactor Coolant (CB)	Vessels, Pressure
17)	6/28	2	F	Low voltage on 4160 volt auxiliary bus associated with system disturbance.	A	3	Electric Power (EA)	NA
18)	7/8	37	S	Repair tube leaks in moisture separator reheater.	B	1	Steam & Power (HB)	Heat Exchangers
19)	7/30	9	F	Maintenance trouble shooting to correct problem in rod control system caused a loss of power to control rod drive mechanism. Reactor trip caused by low pressurizer level.	G	3	Reactor (RB)	Control Rod Drives
20)	7/31	7	F	Repair leak in drain line to heater drain tank.	B	1	Steam & Power (HB)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
21)	8/19	2	F	Replaced faulty component in pressurizer spray valve control system.	A	1	Reactor Coolant (CB)	Instrumentation & Controls
22)	9/1	13	F	Steam generator low level caused by cycling of switch on 125 V DC system, while trouble shooting to locate DC ground.	H	3	Electric Power (EC)	Circuit Closers
23)	9/5	7	F	Added balancing weights to turbine to correct excessive vibration.	B	1	Steam & Power (HA)	Turbines
24)	9/14	226	S	Repair steam generator leak.	B	1	Steam & Power (HB)	Heat Exchangers
25)	10/1	2	F	Failure of No. 3A condensate pump motor. Trip from low level in steam generator.	A	3	Steam & Power (HH)	Motors
26)	10/5	1704	S	Refueling, maintenance and inspection.	C	1	Reactor (RC)	Fuel Elements
27)	12/16	5	F	Turbine-generator removed from service to add balancing weights to turbine.	B	1	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
28)	12/16	4	F	Turbine-generator removed from service to add balancing weights.	B	1	Steam & Power (HA)	Turbines
29)	12/17	4	F	Turbine-generator removed from service to add balancing weights to the turbine.	B	1	Steam & Power (HA)	Turbines
30)	12/17	22	F	Turbine-generator removed from service to perform turbine overspeed trip test.	B	1	Steam & Power (HA)	Turbines
31)	12/18	1	F	Turbine-generator removed from service to perform turbine overspeed trip test.	B	1	Steam & Power (HA)	Turbines
32)	12/26	23	F	Turbine-generator removed from service to correct excessive vibration on turbine caused by loss of shroud-band from L. P. Turbine blade.	A	3	Steam & Power (HA)	Turbines

TURKEY POINT 4

I. Summary

Description

Location: Florida City, Florida
Docket No: 50-251
Reactor Type: Pressurized Water
Capacity (MWe-Net): 666
Commercial Operation: 9/7/73
Plant Age: 1.5 Years

Performance

Net electrical energy generated (MWH):	4,292,374
Unit availability factor (%)	77.1
Unit capacity factor (%): (Using Design MWe)	74.1

Outages

Total No.	25		
Forced	17		
Scheduled	8		
Total:	2,009	Hours,	22.9%
Forced	454	Hours,	5.2%
Scheduled	1,555	Hours,	17.7

Cause:

Equipment Failure	10
Maintenance/Test	9
Refueling	6

Method of shutdown:

Manual	14
Manual Scram	1
Auto Scram	10

II. Highlights

A. General :

At the beginning of the year, Unit 4 was limited to 80% of rated power because of main steam line vibration problems. After repairs, the unit escalated and reached 100% rated power (2200 MWT) on March 11. For the period July 1 — December 31, the unit was essentially base loaded.

B. Outages :

1. Forced: There were 17 forced outages during the year consuming 454 hours. The longest forced outage was for 153 hours to repair a leak in a steam generator sample line and to repair a leak in a vent line upstream of a check valve.

2. Scheduled: Eight scheduled outages consumed 1555 hours, of which, the longest was for 782 hours to inspect seismic restraints and to perform routine maintenance. One outage lasted 565 hours to repair steam generator tube leaks.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/2	783	S	Inspection of seismic restraints and routine maintenance.	B	2	Reactor Coolant (CX)	Shock Suppressors
2)	2/4	8	S	Turbine overspeed trip test.	B	1	Steam & Power (HA)	Turbines
3)	2/6	3	F	Maintenance of condensate pump valves.	B	1	Steam & Power (HH)	Valves
4)	2/7	1	F	Low steam generator level during start-up.	G	3	Steam & Power (HB)	Heat Exchangers
5)	2/20	5	S	Preoperational test program required a turbine trip from 100% power.	B	1	Steam & Power (HA)	Turbines
6)	2/21	3	S	Preoperational test program required a generator trip from 100% power.	B	1	Steam & Power (HA)	Generators
7)	3/1	12	F	Generator experienced a loss of field while voltage regulator was in manual.	A	3	Steam & Power (HA)	Generators
8)	3/13	10	F	Turbine reheat and intercept valve control system repairs.	A	1	Steam & Power (HB)	Valve Operators
9)	4/2	82	F	Turbine control oil system governor impeller was found to have excessive clearance.	A	1	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	4/14	4	F	Selector switch malfunctioned in feedwater control system. Trip from generator high level.	A	3	Steam & Power (HH)	Circuit Closers
11)	4/16	35	F	Generator hydrogen gas cooler leak.	A	1	Steam & Power (HA)	Heat Exchangers
12)	4/25	14	F	Generator breaker protection system was inadvertently energized during test of startup transformer protection.	G	3	Steam & Power (HA)	Circuit Closers
13)	5/25	167	S	Inspection of seismic restraints and maintenance.	B	1	Reactor Coolant (CX)	Shock Suppressors
14)	6/5	16	S	Replaced motor on control rod drive mechanism cooling system.	B	1	Reactor (RB)	Motors
15)	6/17	8	S	Modify nuclear instrumentation system.	B	1	Instrumentation & Controls (IB)	Instrumentation & Controls
16)	6/17	1	F	Operating with feedwater system in MANUAL. Received high water level trip from steam generator.	G	3	Steam & Power (HH)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	6/28	2	F	Low voltage on 4160 volt auxiliary bus associated with system disturbance.	A	3	Electric Power (EA)	NA
18)	6/29	1	F	Trip from steam generator low level while operating with feedwater system in manual.	G	3	Steam & Power (HH)	NA
19)	7/19	2	F	Workers removing scaffolding from turbine housing inadvertently actuated turbine trip device.	G	3	Steam & Power (HA)	NA
20	8/17	565	S	Steam generator tube leak repair.	B	1	Steam & Power (HB)	Heat Exchangers
21)	10/26	153	F	Repair leak in steam generator sample line and a leak in a vent line upstream of a check valve.	A	1	Steam & Power (HB)	Pipes, Fittings
22)	11/7	6	F	Operator error caused loss of power supply to vital instrument bus. Trip from high level in steam generator.	G	3	Electric Power (ED)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
23)	11/23	21	F	Repair restricted flow path from boric acid tanks to reactor coolant system.	A	1	Auxiliary Process (PC)	Pipes, Fittings
24)	12/1	20	F	Turbine control valve spring adjusting bolt failed.	A	3	Steam & Power (HA)	Valves
25)	12/3	87	F	Expansion joint on extraction steam line to feedwater heater failed.	A	1	Steam & Power (HB)	Pipes, Fittings

VERMONT YANKEE

I. Summary

Description

Location: Vermon, Vermont
Docket No: 50-271
Reactor Type: BWR
Capacity (MWe-Net): 514
Commercial Operation: 11/72
Plant Age: 2.3 Years

Performance

Net electrical energy generated (MWH):	2,482,564
Unit availability factor (%):	74.1
Unit capacity factor (%): (Using Design MWe)	56.2

Outages

Total No.	16		
Forced	7		
Scheduled	9		
Total:	2,267	Hours,	25.9%
Forced	199	Hours,	2.3%
Scheduled	2,068	Hours,	23.6%

Cause:

Equipment Failure	3
Maintenance/Test	10
Refueling	1
Operational Error	1
Lightning	1

Method of Shutdown:

Manual	7
Manual Scram	3
Auto Scram	6

II. Highlights

A. General:

Five of the 16 shutdowns were directly associated with the turbine. A refueling shutdown lasted for 1501 hours. Following a shutdown in April, the plant was restricted to 80% power because of excessive off-gas activity levels; the levels were attributed tc faulty fuel cladding.

B. Outages:

1. Forced: Seven forced outages caused the plant to be out of service for 199 hours during 1974. There were no forced outages that exceeded 100 hours duration. The duration of the longest forced outage was 75 hours (caused by multiple lightning strikes).

2. Scheduled: Nine scheduled outages caused 2068 hours down time during the year. Those of the longest duration were: (1) 1501 hours; for refueling; (2) 263 hours for repairs of a leak in the drywell, and (3) 139 hours for a full load reject test.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/5	20	F	Steam leak on the HP turbine inlet flange.	A	1	Reactor Coolant (CC)	Pipes, Fittings
2)	1/24	41	F	Feedwater system test resulting in high reactor vessel level.	B	3	Reactor Coolant (CH)	Instrumentation & Controls
3)	2/17	24	S	Full load reject test resulting in a turbine trip.	B	3	Steam & Power (HA)	Turbines
4)	2/23	79	S	MSIV isolation test.	B	3	Reactor Coolant (CD)	Valves
5)	2/26	5	S	Turbine overspeed test.	B	3	Steam & Power (HA)	Turbines
6)	2/27	4	F	Turbine thrust bearing adjustment.	B	1	Steam & Power (HA)	Turbines
7)	3/3	34	F	Steam trap on main steam line malfunctioned.	A	1	Reactor Coolant (CC)	Pipes, Fittings
8)	3/29	139	S	Full load reject test resulting in high reactor water level.	B	2	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	4/4	3	S	Test operation of HPCI system.	B	1	Engineered Safety (SF)	NA
10)	4/19	10	F	Accidental jarring of reactor water level indicator.	G	3	Instrumentation & Controls (IA)	NA
11)	4/27	42	S	Main steam line relief valve repair.	B	1	Reactor Coolant (CC)	Valves
12)	5/9	15	F	Augmented off gas condensate booster pump tripped resulting in high turbine back pressure.	A	2	Radio-active Waste (MB)	Pumps
13)	5/25	263	S	Increase in drywell temperature and leak in CRD return to vessel.	B	1	Reactor (RB)	Pipes, Fittings
14)	7/5	75	F	Multiple lightning strikes.	H	3	Electric Power (EA)	None
15)	10/12	1501	S	Refueling outage.	C	1	Reactor (RC)	Fuel Elements
16)	12/14	12	S	Test - shutdown from outside of control room.	B	2	Instrumentation & Controls (IC)	None

I. Summary

Description

Location: Rowe, Massachusetts
Docket No: 50-029
Reactor Type: Pressurized Water
Capacity (MWe-Net): 175
Commercial Operation: 2/61
Plant Age: 14.1 Years

Performance

Net electrical energy generated (MWH):	911,452
Unit availability factor (%):	69.6
Unit capacity factor (%): (Using Design MWe)	59.5

Outages

Total No.	7	
Forced	2	
Scheduled	5	
Total:	2,659	Hours, 30.4%
Forced	15	Hours, 0.2%
Scheduled	2,644	Hours, 30.2%

Cause:

Operator Tng. & License Exam	3
Maintenance/Test	2
Equipment Failure	1
Refueling	1

Method of Shutdown:

Manual	4
Auto Scram	2
Manual Scram	1

II. Highlights

A. General:

Unit availability for January – April exceeded 96%. A refueling and maintenance outage commenced on May 10 and lasted until August 25. Unit availability for the remainder of the year was above 97%.

B. Outages:

1. Forced: Two forced outages consumed 15 hours; one occurred during a turbine overspeed test.

2. Scheduled: Five scheduled outages consumed 2644 hours. The extended refueling and maintenance outage lasted 2553 hours. Three outages consuming 53 hours were devoted to operator training and license examinations.

DETAILS OF PLANT OUTAGES

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/11	20	S	Control rod drop time testing and operator training.	E	1	Reactor (RB)	Control Rod Drives
2)	3/16	23	S	Control rod drop time testing and operator license examinations.	E	1	Reactor (RB)	Control Rod Drives
3)	4/18	9	F	Ground on nuclear recorder caused coolant pressure scram.	A	3	Instrumentation & Controls (ID)	Instrumentation & Controls
4)	5/10	2553	S	Refueling and maintenance.	C	2	Reactor (RC)	Fuel Elements
5)	8/25	6	F	Scram during performance of turbine overspeed trip test.	B	3	Steam and Power (HA)	Turbines
6)	9/28	10	S	Control rod drop time test and NRC license exam.	E	1	Reactor (RB)	Control Rod Drives
7)	11/30	38	S	Repair nuclear detectors and feedwater heater leaks, and control rod drop time test.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls

ZION 1

I. Summary

Description

Location: Zion, Illinois
Docket No: 50-295
Reactor Type: PWR
Capacity (MWe net): 935, but changed to 880 in Nov. 1974
Commercial Operation: 12/31/73
Plant Age: 1.5 Years

Performance

Net electrical energy generated (MWH):	3,477,561
Unit availability factor (%):	57.2
Unit capacity factor (%): (Using Design MWe)	45.1

Outages

Total No.	21		
Forced	16		
Scheduled	5		
Total	3,746	Hours,	42.8%
Forced	897	Hours,	10.3%
Scheduled	2,849	Hours,	32.5%

Cause:	
Equipment Failure	9
Maintenance/Test	6
Reg. Restriction	1
Operational Error	5

Method of Shutdown:	
Manual	8
Manual Scram	2
Auto Scram	11

II. Highlights

A. General:

The first 3 1/2 months of the year were devoted to repairs and modifications to the generator. After completing repairs, the startup and power ascension program was resumed. On July 23, the unit reached its licensed power limit of 85%.

B. Outages:

1. Forced: There were 16 forced outages during the year that required 897 hours. The longest outage was for 305 hours to repair pressurizer spray valve bellows. Condenser tube leaks required 146 hours in November.

2. Scheduled: Five scheduled outages consumed 2849 hours during the year. Most of this, 2492 hours, were to effect repairs to the generator which was damaged by a ground fault. Another scheduled outage required 213 hours for testing and replacement of a main steam line check valve.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	2492	S	Generator repairs and modifications.	B	3	Steam & Power (HA)	Generators
2)	4/30	43	S	Generator inspection.	B	3	Steam & Power (HA)	Generators
3)	5/2	10	F	Trip due to steam flow spike.	A	3	Steam & Power (HB)	Instrumentation & Controls
4)	5/5	8	F	Trip -- dirty governor oil caused feedwater pump to oscillate.	G	3	Steam & Power (HH)	Pumps
5)	5/6	18	F	Steam/feed flow mismatch.	G	3	Steam & Power (HB)	NA
6)	5/25	213	S	Testing and main steamline check valve replacement.	B	2	Steam & Power (HB)	Valves
7)	6/30	30	F	Boric acid line plugged.	A	1	Auxiliary Process (PC)	Pipes, Fittings
8)	7/4	97	F	To repair rupture disc on heater drain tank.	A	1	Steam & Power (HH)	Heat Exchangers
9)	7/10	16	S	Turbine trip test.	B	2	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	7/12	68	F	Intercept valve work on turbine.	A	1	Steam & Power (HB)	Valves
11)	7/29	21	F	Low boric acid concentration.	D	1	Auxiliary Process (PC)	NA
12)	8/15	28	F	Main steam line isolation valve repair.	A	1	Reactor Coolant (CD)	Valves
13)	8/24	22	F	Plugged condenser tubes.	B	1	Steam & Power (HC)	Heat Exchangers
14)	8/25	11	F	Low steam gen. level — feedwater pump.	A	3	Steam & Power (HH)	Pumps
15)	8/26	305	F	Leakage thru bellows of pressurizer spray valve.	A	1	Reactor Coolant (CB)	Valves
16)	9/19	85	S	85% generator trip — scheduled outage — repair feedwater pump.	B	3	Steam & Power (HH)	Pumps
17)	10/8	83	F	Instrument mech. caused reactor trip — low boron concen. in accum. corrected.	G	3	Auxiliary Process (PC)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
18)	10/28	23	F	Turbine tripped due to high steam generator level.	A	3	Steam & Power (HB)	Heat Exchangers
19)	11/6	146	F	Condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
20)	11/27	16	F	Instrument mech. caused reactor trip.	G	3	Instrumentation & Controls (IA)	NA
21)	12/3	11	F	Reactor trip due to short in instrument mech. test leads.	G	3	Instrumentation & Controls (IA)	Electrical Conductors

*Data for the table was obtained from the plant Semiannual report, monthly data tables, the Operating Units Status report, and information supplied by the plant staff to NSIC. A discrepancy of 40 hours exists in the outage hours.

ZION 2

I. Summary

Description

Location: Zion, Illinois
Docket No: 50-304
Reactor Type: Pressurized Water
Capacity (MWe-net): 935, but changed to 880 in Nov. 1974
Commercial Operation: 9/17/74
Plant Age: 1.0 Year

Performance

Net electrical energy generated (MWH):	963,986
Unit availability* factor (%):	59.8
Unit capacity factor*(%): (Using Design MWe)	43.9

*Based on date of declaration of commercial operation - Sept. 17.

Outages

Total No.	44		
Forced	38		
Scheduled	6		
Total	6,190	Hours,	70.7%
Forced	4,952	Hours,	56.5%
Scheduled	1,238	Hours,	14.2%

Cause:

Equipment Failure	30
Maintenance/Test	6
Operational Error	8

Method of Shutdown:

Manual	5
Manual Scram	3
Automatic Scram	36

II. Highlights

A. General:

The unit began the year with a continuation of a maintenance outage which started in December 1973. In February, the startup and power ascension program began. In April, a phase to phase fault in the main generator caused a four month outage. Following the outage, the power testing program was resumed, and in November the licensed power level of 85% was obtained.

B. Outages:

1. Forced: There were 38 forced outages during the year consuming 4952 hours. The longest outage began in April and lasted 3090 hours. The cause was a phase to phase short in the generator. On January 16, another maintenance outage began which consumed 664 hours.

2. Scheduled: There were six scheduled outages during the year consuming 1238 hours. The longest outage was 391 hours in November to convert the secondary water system to an all volatile treatment system.

DETAILS OF PLANT OUTAGES*

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	330	S	Continuation of outage started in Dec. 1973. Generator short. Replaced stator bar and modified stator cooling system.	B	1	Steam & Power (HA)	Generators
2)	1/14	4	F	Feedwater pump trip. Auto controls erratic due to plugged oil filters.	A	3	Steam & Power (HH)	Filters
3)	1/15	22	F	Operator error. Tripped feed pump.	G	3	Steam & Power (HH)	NA
4)	1/16	664	F	High steam line flow and low temperature on all loops. Maintenance performed. Rewelded crack in CRD housing.	A	3	Reactor (RB)	Control Rod Drives
5)	2/15	8	F	Trip due to turbine vibration.	A	2	Steam & Power (HA)	Turbines
6)	2/15	6	S	Startup testing — 25% turbine trip test.	B	2	Steam & Power (HA)	Turbines
7)	2/15	36	F	Steam flow/feed flow mismatch.	G	3	Steam & Power (HH)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	2/25	13	F	Boric acid concentration low in tank.	G	1	Auxiliary Process (PC)	NA
9)	3/2	267	S	Inspection of main steam check valves.	B	3	Steam & Power (HB)	Valves
10)	3/14	7	F	High steam generator level.	A	3	Steam & Power (HB)	Heat Exchangers
11)	3/29	82	F	Excessive H_2 leak on generator.	A	2	Steam & Power (HA)	Generators
12)	4/2	3	F	Steam gen. level trip when feedpump transferred to auxiliary.	A	3	Steam & Power (HB)	Heat Exchangers
13)	4/2	4	F	Feedwater pump trip.	A	3	Steam & Power (HH)	Pumps
14)	4/3	13	F	While testing, test relay de-energized faster than safeguards relay. Feedpumps tripped.	A	3	Steam & Power (HH)	Relays
15)	4/5	251	F	Excessive leakage in reactor coolant system.	A	1	Reactor Coolant (CB)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
16)	4/17	3090	F	Phase to phase short in generator requiring repair.	A	3	Steam & Power (HA)	Generators
17)	8/24	100	F	EHC system problem. Repaired transducer.	A	1	Steam & Power (HB)	Instrumentation & Controls
18)	8/29	93	F	EHC system leak. Repaired pin hole weld leak.	A	3	Steam & Power (HB)	Pipes, Fittings
19)	9/2	4	F	Steam gen. low feed flow. Adjusted setpoints of atmospheric relief valves.	A	3	Steam & Power (HB)	Valves
20)	9/4	72	F	Replaced pilot valve for MSIV.	A	3	Reactor Coolant (CD)	Valves
21)	9/11	21	F	EHC system leak.	A	1	Steam & Power (HB)	Pipes, Fittings
22)	9/12	20	F	Link pin in turbine trip mechanism fell out.	A	3	Steam & Power (HA)	Mechanical Function Units
23)	9/13	14	F	Feed pump overspeed trip while adjusting speed changer.	G	3	Steam & Power (HH)	Circuit Closers
24)	9/14	11	F	Feedwater pump problems. Adjusted pump speed control.	A	3	Steam & Power (HH)	Circuit Closers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
25)	9/14	7	F	Steam/feed flow mismatch. Adjusted pump speed control.	A	3	Steam & Power (HH)	Circuit Closers
26)	9/17	21	F	Replaced up/down counter on EHC system valve position limit.	A	3	Steam & Power (HA)	Valves
27)	9/18	15	F	Steam/feed flow mismatch. Operator error.	G	3	Steam & Power (HH)	NA
28)	9/18	6	F	EHC system leak.	A	3	Steam & Power (HA)	Pipes, Fittings
29)	9/25	11	F	Steam/feed flow mismatch. No problems found.	A	3	Steam & Power (HH)	NA
30)	9/28	27	F	Steam/feed flow mismatch.	A	3	Steam & Power (HH)	NA
31)	9/29	5	F	Steam generator hi level. Operator error.	G	3	Steam & Power (HB)	NA
32)	10/2	25	F	Low level in steam generator - faulty limit switch - main steam line isola. valve closure.	A	3	Steam & Power (HB)	Instrumentation & Controls
33)	10/6	14	F	Operator error during refilling of battery.	G	3	Electric Power (EC)	Batteries & Chargers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
34)	10/12	223	S	Scheduled maintenance. Repaired cracked bellows on pressurizer spray valves.	B	3	Reactor Coolant (CB)	Valves
35)	10/22	198	F	Instrument mech. shorted test leads — reactor coolant pump tripped.	G	3	Reactor Coolant (CB)	NA
36)	11/25	5	F	High steam generator level.	A	3	Steam & Power (HB)	Circuit Closers
37)	11/25	14	F	Low steam generator level and low feed pump flow. FW control problems.	A	3	Steam & Power (HB)	Circuit Closers
38)	11/28	391	S	Conversion to AVT Chem System.	B	3	Steam & Power (HH)	Heat Exchangers
39)	12/14	11	F	Steam generator low level with mismatch of flows. Recalibrated level trans.	A	3	Steam & Power (HB)	Instrumentation & Controls
40)	12/14	5	F	Stuck relay caused flow mismatch.	A	3	Steam & Power (HB)	Relays
41)	12/15	20	F	Loss or feedwater pump.	A	3	Steam & Power (HH)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1974)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
42)	12/27	21	S	85% generator trip test.	B	3	Steam & Power (HA)	Generators
43)	12/28	2	F	Flow mismatch on steam generator. FW control problems.	A	3	Steam & Power (HH)	Instrumentation & Controls
44)	12/30	34	F	Flow mismatch due to broken connector on control relay.	A	3	Steam & Power (HH)	Relays

*Data for this table was obtained from the plant Semiannual reports, the Operating Units Status Reports, and from the plant staff via telephone to NSIC.

APPENDIX B - 4

INDIVIDUAL PLANT SUMMARIES

1975

I. Summary

Description

Location: Russellville, Arkansas
Docket No: 50-313
Reactor Type: PWR
Capacity (MWe-Net): 850
Commercial Operation: 12/19/74
Plant Age: 1.4 Years

Performance

Net electrical energy generated (MWH):	4,879,862
Unit availability factor (%):	76.5
Unit capacity factor (%) (using MDC):	66.6
Unit capacity factor (%) (using Design MWe):	65.5

Outages

Total No.	18	
Forced	13	
Scheduled	5	
Total:	2,095	Hours, 23.9%
Forced	529	Hours, 6.0%
Scheduled	1,566	Hours, 17.9%

Cause:

Equipment Failure	10
Maintenance or Testing	5
Operational Error	1
Other	2

Method of Shutdown:

Manual	8
Automatic Scram	7

II. Highlights

A. General:

A total of 18 outages occurred during 1975. Of these, 6 were related to problems with instrumentation; 2 were related to problems with heat exchangers; and 3 were associated with the control rod drive mechanisms. The longest single shutdown was for 1168 hours, to permit maintenance and completion of the outstanding construction items.

B. Outages:

1. Forced: Thirteen forced outages occurred in 1975. The ones of longest duration were: (1) 258 hours, for control rod drive problems; and (2) 50 hours, due to a CRD transfer switch failure.

2. <u>Scheduled</u>: Five scheduled outages occurred during the report period. The ones of longest duration were: (1) 1168 hours, for maintenance and construction; and (2) 315 hours for control rod drive interchange and maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	48	F	Continued from 1974. Cracked resistance temperature detector well in main steam line.	A	1	Steam & Power (HB)	Pipes, Fittings
2)	1/3	29	F	Control rod drive assembly failed to withdraw due to burned stator windings.	A	1	Reactor (RB)	Control Rod Drive Mechanisms
3)	1/6	18	F	Generator differential current relay tripped because drive belt came off isophase bus cooling system pulley.	A	3	Steam & Power (HA)	Blowers
4)	2/22	28	F	Loss of load due to destruction of power lines by windstorm resulted in trip.	H	3	Electric Power (EA)	Electrical Conductors
5)	3/1	8	F	Three condenser tubes were severed due to unknown cause.	A	NA	Steam & Power (HC)	Heat Exchangers
6)	3/19	1168	S	Unit shut down for maintenance and completion of outstanding construction items.	B	1	Reactor Coolant (CX)	NA
7)	5/9	9	F	Reactor trip from improper power imbalance/flux flow relationship indicated by nuclear instrumentation.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	5/15	16	F	Reactor trip from improper power imbalance/flux flow relationship indicated by nuclear instrumentation.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
9)	5/17	29	S	Repair tube leak in feedwater heater.	B	NA	Steam & Power (HH)	Heat Exchangers
10)	5/30	16	S	Power escalation test for generator separation at power.	B	NA	Steam & Power (HA)	Generators
11)	6/6	11	F	RPS trip due to flux-flow imbalance resulting from failed cold leg RTD connector.	A	1	Instrumentation & Controls (IA)	Instrumentation & Controls
12)	6/20	50	F	RPS trip due to low RCS pressure resulting from dropped CRD group due to failed CRD transfer switch.	A	1	Reactor (RB)	Circuit Closers
13)	7/3	5	F	Reactor trip from flux/flow imbalance. Accidental disturbance of inservice instrumentation resulted in a false indication of low temperature. The ICS response to increase the temperature resulted in a reactor trip.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	7/23	41	F	Trip from heater drain tank. Trip point of main feedwater pump was reduced.	A	1	Steam & Power (HH)	Instrumentation & Controls
15)	9/14	38	S	Shut down to repair steam leaks on level tap valves on steam generator.	B	1	Steam & Power (HB)	Valves
16)	10/24	315	S	Control rod drive system interchange.	B	1	Reactor (RB)	Control Rod Drive Mechanisms
17)	12/5	8	F	A lightning strike on the 161 transmission line caused a voltage fluctuation that resulted in a CRDM trip.	H	3	Electric Power (EA)	Electrical Conductors
18)	12/21	258	F	One control rod failed to withdraw after a CRDM ratchet trip.	A	3	Reactor (RB)	Control Rod Drive Mechanisms

BIG ROCK POINT 1

I. Summary

Description

Location: Big Rock Point, Michigan
Docket No: 50-155
Reactor Type: BWR
Capacity (MWe-Net): 72
Commercial Operation: 3/63
Plant Age: 13.1 Years

Performance

Net Electrical Energy Generated (MWH):	290,532.1
Unit Availability Factor (%):	59.8
Unit Capacity Factor (%) (Using MDC):	46.7
Unit Capacity Factor (%) (Using Design MWE):	46.1

Outages

Total No.	4	
Forced	3	
Scheduled	1	
Total:	3,522 Hours,	40.2%
Forced	3,472 Hours,	39.6%
Scheduled	50 Hours,	0.6%

Cause: Equipment Failure	2
Maintenance or Testing	1
Other	1
Method of Shutdown:	
Manual	2
Automatic Scram	1

II. Highlights

A. General: A total of 4 outages occurred during 1975. An investigation of the post-incident cooling system, with respect to its operability during LCCA conditions, was the major item causing generator downtime during the report period.

B. Outages:

1. Forced: Three forced outages occurred during 1975; the one of longest duration (3421 hours) was attributed to investigating the deficiencies in the design and quality of the instrumentation for the Post Incident Cooling System:

2. Scheduled: There was only one scheduled shutdown during the report period; this (for 50 hours) was for semiannual control rod drive testing and various maintenance items.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/16	3421	F	The unit was shut down when it was found that design and QA deficiencies existed in instrumentation for the Post Incident Cooling System.	H	1	Engineered Safety (SF)	Instrumentation & Controls
2)	10/30	6	F	Leak on stage drain line from HP turbine to HP heater. Initial pressure regulator failed during power reduction.	A	NA	Steam & Power (HB)	Pipes, Fittings
3)	11/13	45	F	Unit off line to plug leaking tubes in main condenser. During routine instrument check, an auto scram was initiated. E-4 control rod drive could not be withdrawn in fully inserted position.	A	3	Steam & Power (HC)	Heat Exchangers
4)	12/6	50	S	Semiannual control rod drive testing and various maintenance items.	B	1	Reactor (RB)	Control Rod Drive Mechanisms

I. Summary

Description

Location: Decatur, Alabama
Docket No: 50-259
Reactor Type: BWR
Capacity (MWe-Net): 1098
Commercial Operation: 8/1/74
Plant Age: 2.2 Years

Performance

Net Electrical Energy Generated (MWH):	1,347,943
Unit Availability Factor (%):	17.5
Unit Capacity Factor (%) (Using MDC):	14.4
Unit Capacity Factor (%) (Using Design MWE):	14.4

Outages

Total No.	5	
Forced	4	
Scheduled	1	
Total:	7,225 Hours,	82.5%
Forced	7,045 Hours,	80.4%
Scheduled	180 Hours,	2.1%

Cause:

Equipment Failure	2
Maintenance or Testing	1
Regulatory Restriction	1
Operational Error	1

Method of Shutdown:

Manual Scram	2
Automatic Scram	3

II. Highlights

A. General: A total of 5 outages occurred during the early part of 1975. Of these, the shutdown resulting from the cable tray fire, which occurred in March and remained in effect for the remainder of the year, was the most significant.

B. Outages:

1. Forced: Four forced outages occurred during 1975. The cable tray fire accounted for 6828 hours in the report period. The forced shutdown of next longest duration was for 195 hours to conduct maintenance on relief valves.

2. Scheduled: Only one scheduled outage occurred during 1975, requiring 180 hours for the purpose of inspecting core spray welds.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/2	180	S	Core spray weld inspection.	D	2	Engineered Safety (SF)	Pipes, Fittings
2)	2/23	13	F	Core shutdown margin test.	B	3	Reactor (RB)	Control Rods
3)	2/26	195	F	Blown fuse in feedwater inverter and relief valve maintenance.	A	3	Reactor Coolant (CC)	Valves
4)	3/11	9	F	Oxygen level in drywell too high.	A	3	Engineered Safety (SE)	——
5)	3/22	6828	F	Cable tray fire during containment penetration leak test.	G	2	Electric Power (ED)	Electrical Conductors

BROWNS FERRY 2

I. Summary

Description

Location: Decatur, Alabama
Docket No: 50-260
Reactor Type: BWR
Capacity (MWe-net): 1098
Commercial Operation: 3/1/75
Plant Age: 1.3 Years

Performance

Net Electrical Energy Generated (MWH):	1,374,133
Unit Availability Factor (%):	18.0
Unit Capacity Factor (%) (Using MDC):	14.7
Unit Capacity Factor (%) (Using Design MWE):	14.7

Outages

Total No.	9		
Forced	7		
Scheduled	2		
Total:	7,181	Hours,	82.0%
Forced	7,055	Hours,	80.5%
Scheduled	126	Hours,	1.5%

Cause:	
Equipment Failure	4
Maintenance or Testing	2
Regulatory Restriction	1
Operational Error	2

Method of Shutdown:	
Manual	1
Automatic Scram	8

II. Highlights

A. General:

A total of 9 outages occurred during early 1975. Of these, the shutdown resulting from the cable tray fire, which occurred in March and lasted throughout the remainder of the year, was the most significant.

B. Outages:

1. Forced: Seven forced outages occurred during 1975. Prior to the cable tray fire which occurred during a penetration leak test and which lasted for the remaining 6828 hours in the report period, the other forced shutdowns of long duration were: (1) 124 hours, the result of a relief valve malfunction; and (2) 25 hours, the result of an attempt at balancing recirculation flows.

2. Scheduled: Two scheduled outages occurred during the report period. Of these, the one of longest duration was for 103 hours to inspect the core spray piping welds.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/7	23	S	Startup testing.	B	3	Reactor (RB)	Control Rod Drive Mechanisms
2)	1/11	25	F	Attempted recirc pump flow balance.	G	3	Reactor Coolant (CB)	Instrumentation & Controls
3)	1/16	43	F	High moisture separator level.	A	3	Steam & Power (HB)	Vessels, Pressures
4)	1/19	124	F	Relief valve malfunction.	A	1	Reactor Coolant (CC)	Valves
5)	1/24	10	F	Maintenance error.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
6)	1/28	13	F	EHC control failure.	A	3	Steam & Power (HA)	Turbines
7)	2/3	12	F	Condensate demineralizer pressure fluctuations.	A	3	Steam & Power (HG)	Demineralizers
8)	2/11	103	S	UT Inspection of core spray piping welds.	D	3	Engineered Safety (SF)	Pipes, Fittings
9)	3/22	6828	F	Cable tray fire during penetration leak test.	G	3	Electric Power (ED)	Electrical Conductors

I. Summary

Description	Performance		Outages		
Location: Southport, North Carolina	Net electrical energy generated (MWH):	1,405,366	Total No.*	29	
Docket No: 50-324	Unit availability factor (%)**	93.2	Forced	28	
Reactor Type: BWR	Unit capacity factor (%)** (using MDC):	59.7	Scheduled	1	
Capacity (MWe-Net): 821	Unit capacity factor (%)** (using Design MWe):	58.8	Total*:	2,112	Hours, 41.1%
Commercial Operation: 11/3/75			Forced	2,092	Hours, 40.7%
Plant Age: 0.7 Year			Scheduled	20	Hours, 0.4%

Cause:
- Equipment Failure 20
- License Exam 1
- Administrative 1
- Operational Error 7

Method of shutdown:
- Manual 13
- Manual Scram 2
- Automatic Scram 14

*Data is for the period June 1 through December 31, 1975.

**Data is from date of commercial operation (11/3/75) to December 31, 1975.

II. Highlights

A. General:

At the beginning of June, the plant was continuing the startup test program. In October, 98% of design power was attained. On November 3, the plant was declared commercial. Subsequently, power was reduced to 50% because of an LPRM vibration problem. In December, the nominal power level was 75% with a 94% availability factor.

B. Outages:

1. Forced: There were 28 forced outages during the period which required 2102 hours. Twenty were due to equipment failures. The longest outage required 842 hours to replace the seals on the recirculation pumps. Another outage for the same reason, consumed 472 hours.

2. Scheduled: The only scheduled outage required 20 hours for the purpose of administering NRC license examinations.

DETAILS OF PLANT OUTAGES*

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	6/1	6	F	Continuation from May outage. While testing, a premature bus lockout occurred resulting in loss of offsite power.	A	3	Electric Power (EA)	Circuit Closers/ Interrupters
2)	6/1	9	F	Poor reactor water quality.	F	1	Reactor Coolant (CG)	Demineralizers
3)	6/7	20	S	NRC operator exams.	E	1	Reactor (RB)	Control Rod Drive Mechanisms
4)	6/9	24	F	During test, valving of instrument caused scram.	G	2	Instrumentation & Controls (IA)	Valves
5)	6/10	16	F	Broken spring in valve controller for feed flow diverted it to condenser.	A	1	Reactor Coolant (CH)	Valve Operators
6)	6/11	9	F	HPCI was being used for low flow feed. Speed spike caused cold water scram.	A	1	Engineered Safety (SF)	Pumps
7)	6/12	842	F	Seal leakage on recirculation pump. Replaced seals and repaired thermal barrier.	A	1	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	7/20	20	F	During test of main steam line radiation monitors, failed to reset before second channel tripped.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
9)	7/22	10	F	Technician improperly removed switch from service. Transient caused scram.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
10)	7/22	20	F	Operator on startup failed to open FW heater valves.	G	1	Reactor Coolant (CH)	Valves
11)	7/25	60	F	Main steam valve packing leak.	A	1	Reactor Coolant (CC)	Valves
12)	8/5	25	F	Steam jet air ejector first stage steam inlet valve closed and would not open.	A	3	Steam & Power (HC)	Valves
13)	8/16	124	F	Repairs to diesel generator.	A	1	Electric Power (EE)	Generators
14)	8/20	7	F	Loss of feedwater flow due to malfunctioning valves.	A	2	Reactor Coolant (CH)	Valves
15)	8/21	14	F	Condensate pumps lost suction. Undetermined reason.	A	3	Steam & Power (HH)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
16)	8/24	182	F	Replaced recirculation pump seals.	A	3	Reactor Coolant (CB)	Pumps
17)	9/1	4	F	High generator vibration faulty ground wire found and bearings were damaged.	A	3	Steam & Power (HA)	Generators
18)	9/2	34	F	Inspection and repair of generator bearings.	A	1	Steam & Power (HA)	Generators
19)	9/5	472	F	Replaced seals on recirc pumps.	A	3	Reactor Coolant (CB)	Pumps
20)	9/29	38	F	Repaired valve packing leaks on RHR loop.	A	1	Reactor Coolant (CF)	Valves
21)	9/30	13	F	Repaired valve motor of the RCIC system.	A	1	Engineered Safety (SF)	Valve Operators
22)	10/16	28	F	Leaks on main steam valve.	A	1	Reactor Coolant (CC)	Valves
23)	10/19	12	F	During test, operator failed to reset scram channel before tripping second channel.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
24)	10/26	18	F	Lighting inverter tripped. Reason unknown. Circulating water pumps tripped.	A	3	Electric Power (EB)	Generators
25)	11/9	26	F	During stop valve tests, turbine high vibration occurred tripping unit.	A	3	Steam & Power (HA)	Turbines
26)	11/15	11	F	During testing of trip channels a wrench slipped causing scram.	G	3	Instrumentation & Controls (IA)	None
27)	11/23	12	F	Turbine high vibration.	A	3	Steam & Power (HA)	Turbines
28)	12/2	32	F	Leaking valve packing in recirculation system.	A	1	Reactor Coolant (CB)	Valves
29)	12/27	24	F	During test, technician operated wrong valve on level transmitter, scram resulted.	G	3	Instrumentation & Controls (IA)	None

*Data covers the period June 1—December 31, 1975.

I. Summary

Description

Location: Lusby, Maryland
Docket No: 50-317
Reactor Type: PWR
Capacity (MWe-Net): 845
Commercial Operation: 5/8/75
Plant Age: 1.0 Year

Performance

Net electrical energy generated (MWH):	4,386,319
*Unit availability factor (%):	90.0
*Unit capacity factor (%) (using MDC):	78.8
*Unit capacity factor (%) (using Design MWe):	74.6

*Data covers the period from date of commercial operation (5/8/75) to end of the year. Total hours considered is 5712.

Outages

Total No.*	18	
Forced	14	
Scheduled	4	
Total*:	986	Hours, 17.3%
Forced	801	Hours, 14.0%
Scheduled	185	Hours, 3.3%

Cause:
- Equipment Failure 10
- Maintenance or Testing 4
- Operational Error 3
- Other 1

Method of Shutdown:
- Manual 7
- Manual Scram 5
- Automatic Scram 5

II. Highlights

A. General:

A total of 18 outages occurred during the period May 8—December 31, 1975. Ten were the result of equipment failures; 4 were for maintenance and/or tests; and 3 were the result of operational errors. Four of the outages were related to problems with pumps; 3 were related to problems with pumps; 3 were related to problems with instrumentation; 3 were related to problems with valves; 2 were related to problems with control rod drive mechanisms; and 1 was related to a problem with the turbine-generator.

B. Outages:

1. Forced: There were 14 forced outages in 1975. Of these, the ones of longest duration were: (1) 2[illegible]8 hours, due to leakage in the reactor cooling system and seal replacement in a reactor cooling pump; (2) 140 hours, due to the loss of a feedwater pump; (3) 105 hours, due to the saltwater system and condenser becoming loaded with small fish; and (4) 111 hours, because of a leak to the reactor cooling drain tank.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the one of longest duration was for 83 hours, due to leaking tubes in moisture separator reheater.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	5/10	33	S	Power testing. 100% trip test and refine balance of turbine.	B	3	Steam & Power (HA)	Turbines
2)	5/12	140	F	Trip due to loss of feed pump and subsequent water hammer in feed system.	A	2	Steam & Power (HH)	Pumps
3)	6/6	83	S	Plugged leaking tubes in moisture separator reheater. Installed turbine heat rate test instrumentation.	B	1	Steam & Power (HB)	Heat Exchangers
4)	6/13	15	F	Reactor trip due to high level in feedwater heater.	A	3	Steam & Power (HH)	Heat Exchangers
5)	6/14	8	F	Plant tripped manually because of stratification of concentrated boron in the volume control tank.	A	2	Auxiliary Process (PC)	Electrical Conductors
6)	6/15	37	F	Pressurizer relief isolation valve gland leakage.	A	1	Reactor Coolant (CB)	Valves
7)	7/8	18	F	Reactor tripped while conducting daily nuclear instrumentation calibration. Operator failed to reset turbine runback bistable resulting in trip on high pressurizer pressure.	G	3	Steam & Power (HA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	7/12	66	S	Removed test connections and flow nozzles in condensate system, installed for initial heat rate test.	B	1	Steam & Power (HH)	Instrumentation & Controls
9)	7/14	3	S	Scheduled maintenance delayed because of problems with reactor cooling pump oil lift system and loss of auxiliary boiler.	B	NA	Reactor Coolant (CB)	Pumps
10)	8/4	105	F	Salt water system and condensers became loaded with small fish which caused increase in condenser delta T and sounded high temp. alarm on the stator cooling system. Replaced 5 intake screens.	H	2	Auxiliary Water (WE)	Filters
11)	8/11	288	F	Leak in reactor cooling system. Installed new seal in reactor cooling pump.	A	1	Reactor Coolant (CB)	Pumps
12)	9/6	25	F	Unit shut down due to sparks eminating from permanent magnetic generator on the main unit.	A	1	Steam & Power (HA)	Generators
13)	9/15	9	F	Reactor tripped due to high level in feedwater heater. Time delay installed in FW heater level trip circuit.	A	3	Steam & Power (HH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	9/23	111	F	Shut down due to leak to reactor cooling drain tank caused by valve gland. Replaced valve gland.	A	1	Reactor Coolant (CB)	Valves
15)	9/29	16	F	Reactor tripped due to closure of steam generator feedwater regulating valve.	G	1	Steam & Power (HH)	Valves
16)	10/2	7	F	Maintenance personnel opened the second set of control element drive mechanism power supply breakers prior to closing the first set while conducting monthly surveillance test.	G	3	Reactor (RB)	Control Rod Drive Mechanisms
17)	11/19	12	F	Manual reactor trip due to excessive vibration on steam generator feed pump due to cracked coupling.	A	2	Steam & Power (HH)	Pumps
18)	12/17	0	F	Control element assembly dropped while performing routine test.	A	2	Reactor (RB)	Control Rod Drive Mechanisms

CONNECTICUT YANKEE

I. Summary

Description

Location: Haddam Neck, Connecticut
Docket No: 50-213
Reactor Type: PWR
Capacity (MWe-Net): 575
Commercial Operation: 1/1/68
Plant Age: 8.4 Years

Performance

Net electrical energy generated (MWH):	4,121,427.8
Unit availability factor (%):	89.9
Unit capacity factor (%) (using MDC):	87.9
Unit capacity factor (%) (using Design MWe):	81.8

Outages

Total No.	6		
Forced	4		
Scheduled	2		
Total:	1,215	Hours,	13.9%
Forced	63	Hours,	0.7%
Scheduled	1,152	Hours,	13.2%

Cause: Equipment Failure 4
Maintenance or Testing 1
Refueling 1

Method of Shutdown:
Manual 3
Automatic Scram 2

II. Highlights

A. General:

A total of 6 outages occurred during 1975. Of these, the major one was for refueling and maintenance. Two shutdowns were related to problems with pipes and fittings, one was related to a problem with a valve, one was related to problems with a pump, and one was related to problems with feedwater heaters.

B. Outages:

1. Forced: Four forced outages occurred during 1975. The one of longest duration was for 39 hours, due to excessive leaking from a valve packing within the containment.

2. Scheduled: Two scheduled shutdowns occurred during the report period. The one of longest duration was for 1071 hours, for refueling and maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/1	16	F	Trip from low feed pump suction pressure.	A	3	Steam & Power (HH)	Pumps
2)	3/26	39	F	Packing gland leakage from the letdown system stop valve to the valve stem leakoff header in the containment was in excess of administrative limits.	A	3	Steam & Power (HI)	Valves
3a)	5/17	1071	S	Refueling and maintenance.	C	1	Reactor (RC)	Fuel Elements
3b)	7/1	6	S	Continuation of refueling. Turbine overspeed trip adjustment after reaching thermal equilibrium conditions.	B	NA	Steam & Power (HA)	Instrumentation & Controls
4)	7/5	5	F	Unit forced off line by broken oil pressure gauge line on turbine.	A	NA	Steam & Power (HA)	Pipes, Fittings
5)	7/14	3	F	Unit forced off line by leaking auto stop oil line on turbine.	A	NA	Steam & Power (HA)	Pipes, Fittings
6)	12/6	75	S	Repaired leaking tubes in feedwater heaters.	B	1	Steam & Power (HH)	Heat Exchangers

I. Summary

Description

Location: Bridgman, Michigan
Docket No: 50-315
Reactor Type: PWR
Capacity (MWe-net): 1090
Commercial Operation: 8/27/75
Plant Age: 0.9 Year

Performance

Net Electrical Energy Generated (MWH):	4,457,776
Unit Availability Factor (%):**	83.7
Unit Capacity Factor (%)** (Using MDC):	82.0
Unit Capacity Factor (%)** (Using Design (MWE):	62.7

Outages

Total No.*	22		
Forced	16		
Scheduled	6		
Total:*	1,709	Hours,	22.1%
Forced	318	Hours,	4.1%
Scheduled	1,391	Hours,	18.0%

Cause: Equipment Failure 8
Maintenance or Testing 6
Operational Error 8

Method of Shutdown:
Manual 7
Automatic Scram 13

*Data is for the period from initial electrical generation (2/10/75) to end of year. Total hours considered was 7734.

**Data is based on period from commercial operation, 8/23/75, to end of year.

II. Highlights

A. General:

A total of 22 outages occurred during the period from initial electrical generation (2/10/75) to the end of the year. Eight were the result of equipment failure; 6 were for maintenance; and 8 were the result of operational errors. Six of the outages were related to problems with instrumentation; 4 were related to problems with heat exchangers; 3 were related to problems with valves; 2 were related to problems with the turbine; 2 were related to problems with electrical conductors; 3 were related to problems with piping; and 2 were related to problems with pumps.

B. Outages:

1. Forced: There were 16 forced outages in 1975. Of these, the one for longest duration was for 85 hours, for a failed cable splice in the reserve power feed.

2. Scheduled: There were 6 scheduled outages in 1975. Of these, the ones of longest duration were: (1) 478 hours due to leakage in the main condenser; (2) 379 hours for general maintenance; (3) 340 hours for surveillance of the ice condenser; and (4) 97 hours for surveillance and maintenance of the ice condenser.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/14	74	S	Repair turning gear on main turbine.	B	1	Steam & Power (HA)	Turbines
2)	2/17	85	F	Repair of failed cable splice in reserve power feed.	A	3	Electric Power (ED)	Electrical Conductors
3)	2/25	10	F	Repair to sample point on steam lead to high pressure turbine.	A	1	Steam & Power (HA)	Pipes, Fittings
4)	3/5	19	F	Main turbine CIV valve stuck in closed position.	A	NA	Steam & Power (HA)	Valves
5)	3/13	23	S	Negative rate trip on reactor as part of power test program.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls
6)	3/15	5	F	High temperature in moisture separator reheater caused turbine trip and reactor trip.	G	3	Steam & Power (HB)	Heat Exchangers
7)	3/18	379	S	Tripped generator at 50% power level as part of power test program. Operator licensing exams on 19, 20, and 21. General maintenance was performed.	B	3	Reactor Coolant (CX)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/8	11	F	Inadvertent closure of steam generator stop valve caused low level in steam generator.	G	3	Steam & Power (HB)	Valves
9)	4/10	8	F	Turbine removed from service to repair leak in EHC fluid supply line to left inner turbine stop valve.	A	NA	Steam & Power (HA)	Pipes, Fittings
10)	4/11	9	F	One channel of feedwater flow on steam generator out of service for surveillance. Low level spike received and the coincidence of the two caused reactor trip.	G	3	Steam & Power (HH)	Instrumentation & Controls
11)	4/12	18	F	One channel of steam flow out for transmitter repair. Work started on wrong transmitter which caused feed pump speed control to go to full open. High feed flow caused level shrink to low level. Coincidence of low level with feed-steam flow mismatch caused reactor trip.	G	3	Steam & Power (HB)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
12)	6/24	59	F	After testing of an air re-circ-hydrogen skimmer fan, a backdraft damper was unlocked too soon. This resulted in slight pressurization of lower volume causing 45 of 48 lower ice condenser inlet doors to open. Inspection after coming down revealed leak in reactor coolant pump seal bypass line.	G	1	Reactor Coolant (CB)	Pipes, Fittings
13)	7/3	478	S	Shut down due to main condenser leak and maintenance installation of additional service water valves and modification of steam dump valves.	B	1	Steam & Power (HC)	Heat Exchangers
14)	8/14	12	F	Pressurizer level channel in trip mode. False high level in other channel caused reactor trip.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
15)	8/14	7	F	High level in steam generator caused feedwater isolation and turbine trip and reactor trip.	G	3	Steam & Power (HB)	Heat Exchangers
16)	10/3	18	F	Unit tripped and received safety injection. Experienced momentary loss of power to reactor protection system.	A	3	Electric Power (ED)	Electrical Conductors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	10/10	12	F	One of two operating hot well pumps tripped out causing low suction pressure to main feed pump which caused reactor trip.	A	3	Steam & Power (HC)	Pumps
18)	10/11	97	S	Unit tripped from turbine front standard to start hot shutdown panel test. Unit remained out to conduct ice condenser surveillance and maintenance.	B	1	Engineered Safety (SB)	Heat Exchangers
19)	10/31	340	S	Power reduced for ice condenser surveillance. Unit removed from service to inspect reactor coolant pump seals.	B	1	Reactor Coolant (CB)	Pumps
20)	12/13	29	F	Inadvertently removed control input to S-G level controlling channel during channel test caused low flow on feedwater header and reactor trip.	G	3	Steam & Power (HH)	Instrumentation & Controls
21)	12/14	9	F	During startup turbine was tripped manually because of high vibration.	A	1	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
22)	12/17	7	F	Blowing down of pressurizer level instrument line caused reduction of pressure to the pressure instruments. This resulted in a reactor trip on low pressurizer pressure.	G	3	Reactor Coolant (CB)	Instrumentation & Controls

COOPER STATION

I. Summary

Description

Location: Brownville, Nebraska
Docket No: 50-298
Reactor Type: BWR
Capacity (MWe-Net): 778
Commercial Operation: 7/1/74
Plant Age: 1.6 Years

Performance

Net electrical energy generated (MWH):	3,853,630
Unit availability factor (%):	83.6
Unit capacity factor (%) (using MDC):	57.6
Unit capacity factor (%) (using Design MWe):	56.5

Outages

Total No.	13		
Forced	10		
Scheduled	3		
Total:	1,436	Hours,	16.4%
Forced	426	Hours,	4.9%
Scheduled	1,010	Hours,	11.5%

Cause:

Equipment Failure	8
Maintenance or Testing	3
Regulatory Restriction	1
Operational Error	1

Method of Shutdown:

Manual	2
Manual Scram	2
Automatic Scram	8

II. Highlights

A. General

A total of 13 outages occurred during 1975. Of these, 5 were related to problems with instrumentation, 2 were related to problems with the generator, 2 were related to problems with the off-gas system, and 2 were related to problems with valves.

B. Outages:

1. Forced: Ten forced outages occurred during 1975. The ones of longest duration were: (1) 178 hours, to correct problems with the main exciter; (2) 135 hours, to perform ultrasonic testing of pipes, etc. (as required by NRC); and (3) 23 hours, to repair a sump line which had been damaged by an explosion in the off-gas system.

2. Scheduled: Three scheduled shutdowns occurred during the report period: (1) 859 hours, for repairs of a 4160V bus and investigation into problems with the LPRM channels; and (2) 93 hours, for maintenance; and (3) 58 hours, for installation of accelerometers for LPRM vibration testing.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/5	11	F	High flux level trip caused by a rapid pressure increase. The pressure spike was induced by the closure of the main turbine control valves resulting from a malfunction of the DEH system.	A	3	Steam & Power (HA)	Instrumentation & Controls
2)	2/3	135	F	Shut down for ultrasonic testing of core spray piping as required by NRC Bulletin IE 75-01.	D	2	Engineered Safety (SF)	Pipes, Fittings
3)	3/15	13	F	Inadvertent scram while checking for troubles in malfunctioning DEH system.	G	3	Steam & Power (HA)	Instrumentation & Controls
4)	5/14	93	S	High pressure scram during implementation of minor design change which caused the main steam bypass valves to close. Sch. maint. performed.	B	3	Reactor Coolant (CC)	Valves
5)	5/27	9	F	Shut down from a false turbine control valve fast closure scram signal caused by failure of a DEH calibration valve.	A	3	Steam & Power (HA)	Valves
6)	7/21	12	F	Reactor scrammed due to power increase caused by pressure spike induced by failure of DEH system.	A	3	Steam & Power (HA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	8/7	58	S	Shut down to install accelerometers for LPRM vibration testing.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls
8)	9/27	859	S	Scram from high water level following loss of feedwater system control due to a loss of a 4160 V bus. Investigated problem with the LPRM channels.	B	3	Reactor Coolant (CH)	Electrical Conductors
9)	11/5	22	F	Investigation of explosion in off-gas system.	A	2	Radioactive Waste (MB)	Recombiners
10)	11/6	23	F	Repair sump liner that had been damaged during the explosion in the off-gas system.	A	1	Radioactive Waste (MA)	Pipes, Fittings
11)	11/7	8	F	Unit was taken off line to add weights to main exciter.	A	NA	Steam & Power (HA)	Generators
12)	12/10	178	F	Scram from main generator field breaker trip while testing power system stabilizer. Main exciter failed on subsequent start-up and unit remained out while a temporary exciter was installed.	A	3	Steam & Power (HA)	Generators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
13)	12/23	15	F	Scram from turbine control system pressure transmitter failure causing main steam bypass valves to open.	A	3	Steam & Power (HA)	Instrumentation & Controls

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-010
Reactor Type: BWR
Capacity (MWe-net): 200
Commercial Operation: 7/4/60
Plant Age: 15.7 Years

Performance

Net Electrical Energy Generated (MWH):	696,781.3
Unit Availability Factor (%):	57.2
Unit Capacity Factor (%) (Using MDC):	30.8
Unit Capacity Factor (%) (Using Design MWE):	30.8

Outages

Total No.	12	
Forced	8	
Scheduled	4	
Total:	3,751	Hours, 42.8%
Forced	1,064	Hours, 12.1%
Scheduled	2,687	Hours, 30.7%

Cause:

Equipment Failure	7
Maintenance or Testing	3
Refueling	1
Regulatory Restriction	1

Method of Shutdown:

Manual	11
Automatic Scram	1

II. Highlights

A. General:

There were 12 outages during the year consuming 3751 hours. Three outages involved pipes and fittings; 2 involved instrument cables; 1 involved valves; 1 involved the demineralizers; 1 involved the turbine; 1 involved the control rods; 1 was a spurious scram; and 1 was a refueling.

B. Outages:

1. Forced: There were 8 forced outages during 1975. The one of longest duration was 462 hours for inspection of the ECCS piping.

2. Scheduled: There were 4 scheduled outages during the period. The longest shutdown for refueling required 2508 hours.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/8	87	S	Repaired leaks in the turbine cross under pipe.	B	1	Steam & Power (HA)	Pipes, Fittings
2)	1/14	23	F	Service water bay in the crib house froze which caused a reduction in flow.	A	1	Auxiliary Water (WA)	NA
3)	1/17	2	F	Spurious scram.	A	1	Instrumentation & Controls (IA)	Instrumentation & Controls
4)	1/24	33	S	Rod pattern interchange.	B	1	Reactor (RB)	Control Rods
5)	3/19	462	F	Shut down for an inspection of ECCS piping.	D	1	Engineered Safety (SF)	Pipes, Fittings
6)	4/7	34	F	Shut down because of high chlorides in primary water system caused by problems with the condensate demineralizers.	A	1	Reactor Coolant (CG)	Demineralizers
7)	4/11	12	F	Incore monitor cables failed.	A	3	Instrumentation & Controls (IA)	Electrical Conductors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/18	59	S	Repair of incore monitor cables.	B	1	Instrumentation & Controls (IA)	Electrical Conductors
9)	5/28	84	F	Leak in stack nozzle of the turbine.	A	1	Steam & Power (HA)	Turbines
10)	6/5	39	F	Unit removed from service because steam generators had leaking valve.	A	1	Steam & Power (HB)	Valves
11)	9/1	2508	S	Refueling	C	1	Reactor (RB)	Fuel Elements
12)	12/15	408	F	A section of the unloader heat exchanger return to reactor line was found to be cracked and was replaced.	A	1	Reactor Coolant (CF)	Pipes, Fittings

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-237
Reactor Type: BWR
Capacity (MWe-net): 809
Commercial Operation: 6/72
Plant Age: 5.7 Years

Performance

Net Electrical Energy Generated (MWH):	2,966,092
Unit Availability Factor (%):	55.1
Unit Capacity Factor (%) (Using MDC):	42.3
Unit Capacity Factor (%) (Using Design MWE):	41.2

Outages

Total No.	15		
Forced	10		
Scheduled	5		
Total:	3,930	Hours,	44.9%
Forced	339	Hours,	3.9%
Scheduled	3,591	Hours,	41.0%

Cause:

Equipment Failure	8
Maintenance or Testing	4
Refueling	1
Op. Tng. and License Exams	2

Method of Shutdown:

Manual	10
Manual Scram	2
Automatic Scram	3

II. Highlights

A. General:

The unit started the year with a continuation of a refueling outage which lasted until May 18. Operation then continued for the remainder of the year with minor outages occurring for EHC oil leaks and turbine control problems.

B. Outages:

1. Forced: There were 10 forced outages during the year resulting in 339 hours of outage time. The only forced outage lasting more than 100 hours was a 126 hour outage to repair cracks in the nitrogen inerting piping.

2. Scheduled: There were 5 scheduled outages consuming 3591 hours. The longest was for 3342 hours which was a continuation of a refueling outage that started Nov. 2, 1974.

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	3342	S	Continuation of refueling outage started Nov. 2, 1974.	C	1	Reactor (RC)	Fuel Elements
2)	5/23	66	S	Problems with electromatic relief valve & MSIV.	B	3	Reactor Coolant (CC)	Valves
3)	6/13	85	S	Repair electromatic relief valves.	B	1	Reactor Coolant (CC)	Valves
4)	7/8	15	F	Instrument mech. scrammed unit while surveillance testing.	G	3	Instrumentation & Controls (IA)	None
5)	8/2	10	F	Back-up pressure regulator failed.	A	1	Steam & Power (HA)	Instrumentation & Controls
6)	9/20	31	F	Repairs to turbine control valve.	A	2	Steam & Power (HA)	Valves
7)	9/24	93	S	Snubber, inspection & maintenance.	B	2	Reactor Coolant (CX)	Shock Suppressors
8)	9/29	48	F	Nitrogen bypass valve left open causing high drywell pressure.	G	3	Engineered Safety (SE)	None

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	10/8	126	F	Nitrogen inerting pipe crack.	A	1	Engineered Safety (SE)	Pipes, Fittings
10)	10/16	41	F	H.P. turbine inlet steam leak.	A	1	Steam & Power (HA)	Pipes, Fittings
11)	10/25	8	F	EHC oil leak.	A	1	Steam & Power (HA)	Pipes, Fittings
12)	11/15	24	F	Turbine EHC oil leak.	A	1	Steam & Power (HA)	Pipes, Fittings
13)	11/16	5	S	EHC oil leak.	B	1	Steam & Power (HA)	Pipes, Fittings
14)	11/22	10	F	Repair turbine control valve.	A	1	Steam & Power (HA)	Valves
15)	11/28	26	F	Leakage — drywell pneumatic system.	A	1	Auxiliary Process (PA)	Pipes, Fittings

DRESDEN 3

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-249
Reactor Type: BWR
Capacity (MWe-Net): 809
Commercial Operation: 11/16/71
Plant Age: 4.4 Years

Performance

Net Electrical Energy Generated (MWH):	2,190,003
Unit Availability Factor (%):	51.5
Unit Capacity Factor (%) (Using MDC):	31.3
Unit Capacity Factor (%) (Using Design MWE):	30.9

*This data reflects a discrepancy of 30 hours which could not be reconciled.

Outages

Total No.	9
Forced	8
Scheduled	1
Total:*	4,222 Hours, 48.2%
Forced	692 Hours, 7.9%
Scheduled	3,530 Hours, 40.3%

Cause:

Equipment Failure	6
Maintenance or Testing	1
Refueling	1
Operational Error	2

Method

Manual	5
Automatic Scram	4

II. Highlights

A. General: A total of 9 outages occurred in 1975. Six were the result of equipment failure; 1 was for maintenance (actually an extension of the refueling shutdown); 1 was for refueling; and 2 were the result of operational errors. Three of the outages were related to problems with piping; 1 was related to a problem with valves; 2 were related to problems with instrumentation; 1 was related to a problem with a demineralizer; 1 was related to a pump; and 1 was due to a loose connection in a control circuit.

B. Outages:

1. Forced: There were 8 forced outages during 1975. Of these, the ones of longest duration were: (1) 105 hours, due to chloride concentration being over the Tech. Spec. limit; (2) 225 hours, for repair of the primary containment isolation valves; (3) 217 hours, for repairs of leaking recirc pump seals; and (4) 95 hours, to investigate air leaks in the drywell.

2. Scheduled: There was 1 scheduled outage during the report period; 1825 hours for refueling, and 1705 hours for maintenance during the extended refueling outage.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/4	105	F	Chloride tech. spec. limit was exceeded due to the cleanup system being out of service.	A	1	Reactor Coolant (CG)	Demineralizers
2)	1/24	8	F	Spurious scram due to operator error.	G	3	Reactor (RB)	Instrumentation & Controls
3)	2/16	225	F	Repair primary containment isolation valves and check core spray lines.	A	1	Engineered Safety (SA)	Valves
4a)	4/16	1825	S	Refueling and maintenance.	C	1	Reactor (RC)	Fuel Elements
4b)	4/16	1705	S	Installation of new feedwater sparger and piping to enlarge the scram discharge volume. Overhauled control rod drives. Replaced collet housings on drives where cracked.	B	NA	Reactor Coolant (CH)	Pipes, Fittings
5)	9/10	217	F	Leaking recirculation pump seals.	A	1	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
6)	10/1	15	F	Operator saw a 1/2 scram and thought it to be a full scram. He then switched to the shutdown mode which caused a reactor scram.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
7)	11/11	10	F	APRM high level trip caused by loose connections in the recirculation pump MG set speed control circuit.	[illegible]	3	Reactor Coolant (CB)	Electrical Conductors
8)	12/3	95	F	Shut down for drywell air leaks. Repaired fitting on MSIV.	A	3	Reactor Coolant (CD)	Pipes, Fittings
9)	12/17	17	F	Repairs of HPCI steam line flange leak.	A	1	Engineered Safety (SF)	Pipes, Fittings

I. Summary

Description

Location: Palo, Iowa
Docket No: 50-331
Reactor Type: BWR
Capacity (MWe-net): 538
Commercial Operation: 2/1/75
Plant Age: 1.6 Years

Performance

Net Electrical Energy Generated (MWH):	2,298,183.1
Unit Availability Factor (%):	79.5
Unit Capacity Factor (%) (Using MDC):	50.9
Unit Capacity Factor (%) (Using Design MWE):	48.8

Outages

Total No.	17		
Forced	13		
Scheduled	4		
Total:	1,797	Hours,	20.5%
Forced	665	Hours,	7.6%
Scheduled	1,132	Hours,	12.9%

Cause:

Equipment Failure	12
Maintenance or Testing	4
Regulatory Restriction	1

Method of Shutdown:

Manual	10
Manual Scram	3
Automatic Scram	4

II. Highlights

A. General:

A total of 17 outages occurred during 1975. Of these, 4 were directly related to problems with pumps, 7 were related to problems with valves and 2 concerned pipes and fittings. A voluntary shutdown to inspect in-core monitors and associated fuel channels accounted for the longest, single shutdown — 1062 hours.

B. Outages:

1. Forced: Thirteen forced outages occurred in 1975. The ones of longest duration were: (1) 223 hours, for inspection of core spray piping; (2) 127 hours, for repairs and testing of HPCI valves; and (3) 60 hours, for repairs on failed condenser tubes.

2. Scheduled: Four scheduled outages occurred during the report period. The major one, for 1062 hours, was for inspection of in-core monitors and associated fuel channels.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	32	S	Inspection of circulating water pump.	B	1	Steam & Power (HF)	Pumps
2)	1/10	20	S	Remove one circulating water pump for repair.	B	1	Steam & Power (HF)	Pumps
3)	1/30	25	F	Bypass valve malfunction while performing weekly turbine control valve and bypass valve testing caused high reactor pressure trip.	A	3	Steam & Power (HB)	Valves
4)	2/8	223	F	Core spray pipe inspection per DRO Bulletin 75-01. Install additional valves and drains in off-gas system. Circulating pump replacement.	D	2	Engineered Safety (SF)	Pipes, Fittings
5)	3/1	18	S	Install repaired circulating water pump.	B	1	Steam & Power (HF)	Pumps
6)	3/23	11	F	Recirculation discharge valve leak repairs.	A	1	Reactor Coolant (CB)	Valves
7)	4/19	127	F	HPCI swing check and stop check valve repair and testing.	A	1	Engineered Safety (SF)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/25	12	F	Unable to maintain main condenser vacuum. Blown rupture disk in steam jet air ejectors repaired.	A	1	Steam & Power (HC)	Pipes, Fittings
9)	5/2	34	F	Repacked MOV in drywell.	A	2	Reactor Coolant (CX)	Valves
10)	5/5	60	F	Repaired failed condenser tubes.	A	2	Steam & Power (HC)	Heat Exchangers
11)	6/6	1062	S	Voluntary shutdown for inspection of in-core monitors and associated fuel channels. 119 fuel channels replaced. Also, repair of relief valve and ramshead restraints in pressure suppression chamber.	B	1	Reactor (RC)	Fuel Elements
12)	8/2	17	F	Mechanical failure to condensate pump caused feedwater pump trip. Subsequent reactor scram from low water level.	A	3	Steam & Power (HH)	Pumps
13)	8/14	14	F	Electrical arc in 480-volt emergency service water pump breaker caused instrumentation voltage transient and reactor scram.	A	3	Electric Power (EB)	Circuit Closers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	9/3	23	F	Packing leaks on feed-water stop check valves and main steam isolation valve.	A	1	Reactor Coolant (CD)	Valves
15)	9/29	13	F	Malfunction of turbine EHC system while performing turbine control valve surveillance testing.	A	3	Steam & Power (HA)	Instrumentation & Controls
16)	11/4	57	F	Repair leaks on one reactor feed pump seal line and core spray isolation valve packing	A	1	Reactor Coolant (CH)	Valves
17)	12 11	49	F	MSIV packing leak in drywell repaired.	A	1	Reactor Coolant (CD)	Valves

I. Summary

Description

Location: Scriba, New York
Docket No: 50-333
Reactor Type: BWR
Capacity (MWe-Net): 821
Commercial Operation: 7/28/75
Plant Age: 0.9 Years

Performance

Net Electrical Energy Generated (MWH):	2,154,564
Unit Availability Factor (%):	70.[illegible]
Unit Capacity Factor (%) (Using MDC):	100.[illegible]
Unit Capacity Factor (%) (Using Design MWE):	50.[illegible]

*Data in this summary begins with July 1975. Unit was declared in commercial operation for 50% of rated load on 7/28/75. Total hours considered is 4,416.

Outages

Total No.*	13
Forced	9
Scheduled	4
Total:*	1,187 Hours, 26.9%
Forced	695 Hours, 15.7%
Scheduled	492 Hours, 11.2%

Cause: Equipment Failure	8
Maintenance or Testing	4
Operational Error	1
Method of Shutdown:	
Manual	6
Manual Scram	1
Automatic Scram	5

II. Highlights

A. General: A total of 13 outages occurred during the period July 1 — December 31, 1975. Eight were the result of equipment failures; 4 were for maintenance and/or tests; and 1 was the result of an operational error. Four of the outages were related to problems with valves; 1 was related to problems with instrumentation; 2 were concerned with the turbine; 2 were related to problems with pumps; and 2 were related to problems with piping.

B. Outages:

1. Forced: There were 9 forced outages in 1975. Of these, the ones of longest duration were: (1) 152 hours, to repair pipe leaks in the condensate line; (2) 125 hours, to repair a leak in a feedwater instrumentation line; and (3) 229 hours, to repair cracks in the turbine building closed loop cooling tank and for repairs of condenser tube leaks.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 126 hours, resulting from closure of the main steam line isolation valves as part of a test; (2) 187 hours, due to the loss of the seals on a feedwater pump in conjunction with a broken strainer restricting the flow of a second pump; and (3) 126 hours, resulting from a turbine trip test.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	7/7	36	F	Repair leak on RCIC pump discharge check valve.	A	1	Reactor Coolant (CE)	Valves
2)	7/16	53	S	Turbine trip testing.	B	3	Steam & Power (HA)	Turbines
3)	8/3	36	F	Scrammed while testing turbine control valve.	A	3	Steam & Power (HA)	Valves
4)	8/16	5	F	Indicated incorrect oil level on recirc pump bearing. Problem was discovered to be bad computer card.	A	NA	Reactor Coolant (CB)	Pumps
5)	8/31	152	F	Pipe leak on condensate line.	A	1	Steam & Power (HH)	Pipes, Fittings
6)	9/11	125	F	Leak in feedwater instrumentation line.	A	1	Reactor Coolant (CH)	Pipes, Fittings
7)	9/30	69	F	While looking for a ground on the battery board, a breaker was erroneously opened which served as control power for feedwater booster pump. When the breaker was racked in, the pump tripped leading to a main feedwater pump trip and scram on reactor low water level.	G	3	Electric Power (EC)	Circuit Closers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	10/22	29	F	High pressure sensor trip while doing surveillance test on core spray system.	A	3	Engineered Safety (SF)	Instrumentation & Controls
9)	10/29	126	S	Closure of main steam isolation valves as part of test.	B	2	Reactor Coolant (CD)	Valves
10)	11/11	187	S	Scram due to low reactor water level caused by loss of mechanical seals on feedwater pump in conjunction with suction strainer breaking loose and causing a decrease in flow on another feedwater pump.	B	1	Reactor Coolant (CH)	Pumps
11)	12/7	126	S	Turbine trip as part of test resulting in reactor scram.	B	3	Steam & Power (HA)	Turbines
12)	12/15	229	F	Crack in turbine building closed loop cooling tank, and high conductivity as a result of condenser tube leaks. Tank repaired and plugs and spacers installed in condenser.	A	1	Steam & Power (HC)	Heat Exchangers
13)	12/25	14	F	Hydraulic oil leak in control system for EHC. CIV repaired.	A	1	Steam & Power (HA)	Valves

FORT CALHOUN

I. Summary

Description

Location: Fort Calhoun, Nebraska
Docket No: 50-285
Reactor Type: PWR
Capacity (MWe-net): 457
Commercial Operation: 8/73
Plant Age: 2.4 Years

Performance

Net Electrical Energy Generated (MWH):	2,080,777
Unit Availability Factor (%):	67.4
Unit Capacity Factor (%) (Using MDC):	52.0
Unit Capacity Factor (%) (Using Design MWE):	52.0

Outages

Total No.	10		
Forced	6		
Scheduled	4		
Total:	2,853	Hours,	6%
Forced	139	Hours,	1.6%
Scheduled	2,714	Hours,	31.0%

Cause: Equipment Failure 6
Maintenance or Testing 4
Refueling 1

Method of Shutdown:
Manual 9
Manual Scram 2
Automatic Scram 1

II. Highlights

. General:

A total of 10 outages occurred in 1975. Six were the result of equipment failures; 4 were for maintenance; and 1 was for refueling. Two of the outages were related to problems with control rod drives; 3 were related to problems with instrumentation; 2 were related to problems with a pump.

B. Outages:

1. Forced: There were 6 forced outages in 1975. Of these, the one of longest duration was for 63 hours, for repairs of a leaking bleed-off line from a reactor coolant pump.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 2160 hours, for refueling; (2) 356 hours, for installation of rod block circuitry; and (3) an extended outage of 125 hours, for repairs to the reactor coolant pump seals.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	12/30/74	58	S	Extension of outage started on 12-30-74 for physics testing.	B	1	Reactor (RB)	Control Rod Drive Mechanisms
2)	2/7	2160	S	Refueling	C	1	Reactor (RC)	Fuel Elements
3)	5/9	42	F	CEA clutch coil failure. Replaced.	A	2	Reactor (RB)	Control Rod Drive Mechansims
4)	5/15	16	F	CEA clutch coil failure. Replaced.	A	2	Reactor (RB)	Control Rod Drive Mechanisms
5)	5/30	3	F	Turbine shell penetration leak.	A	1	Steam & Power (HA)	Turbines
6)	6/28	15	S	Perform test/examination on in-core detector connection and wiring.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls
7)	7/27	7	F	Repair EHC pipe fitting leak.	A	1	Steam & Power (HA)	Pipes, Fittings
8)	8/15	8	F	RPS trip resulting from tests.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9a)	9/16	356	S	Complete installation of rod block circuit.	B	1	Instrumentation & Controls (IE)	Instrumentation & Controls
9b)	9/16	125	S	Outage was extended to repair reactor coolant pump seals.	B	—	Reactor Coolant (CB)	Pumps
10)	12/29	63	F	RC pump controlled bleed-off line leakage to containment atmosphere.	A	1	Reactor Coolant (CB)	Pipes, Fittings

GINNA

I. Summary

Description

Location: Ontario, New York
Docket No: 50-244
Reactor Type: PWR
Capacity (MWe-net): 490
Commercial Operation: 3/70
Plant Age: 6.1 Years

Performance

Net Electrical Energy Generated (MWH):	3,041,203
Unit Availability Factor (%):	76.7
Unit Capacity Factor (%) (Using MDC):	73.9
Unit Capacity Factor (%) (Using Design MWE):	73.9

Outages

Total No.	14		
Forced	10		
Scheduled	4		
Total:	2,051	Hours,	23.4%
Forced	324	Hours,	3.7%
Scheduled	1,727	Hours,	19.7%

Cause:

Equipment Failure	9
Maintenance or Testing	3
Refueling	1
Other	1

Method of Shutdown:

Manual	7
Automatic Scram	7

II. Highlights

A. General:

The unit operated at 100% power most of the year. A refueling was performed in March and April consuming 1658 hours. No other outage consumed more than 100 hours. Total outage time for the year was 2051 hours.

B. Outages:

1. Forced: There were 10 forced outages during the period requiring 324 hours. The longest outage was 97 hours to effect repairs to the turbine caused by vibration.

2. Scheduled: Four scheduled outages required 1727 hours; 1658 hours were for refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	3/5	9	F	Rods dropped due to water leakage into rod control cabinets.	A	1	Reactor (RB)	Instrumentation & Controls
2)	3/10	1658	S	Refueling & Maintenance	C	1	Reactor (RC)	Fuel Elements
3)	5/19	9	F	Hotwell level control failed causing low steam generator level.	A	3	Steam & Power (HC)	Instrumentation & Controls
4)	5/21	44	F	E.H. control failure.	A	1	Steam & Power (HA)	Circuit Closers
5)	5/24	1	S	Turbine overspeed trip test.	B	3	Steam & Power (HA)	Turbines
6)	5/26	6	F	E.H. control valve position limiter malfunction.	A	3	Steam & Power (HA)	Circuit Closers
7)	5/31	24	S	E.H. unit repair & main feedwater pump -- MOV repair.	B	1	Steam & Power (HA)	Circuit Closers
8)	6/6	12	F	Main steam isolation valve malfunction.	A	3	Steam & Power (HB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	6/17	4	F	Loss of instrument bus inverter.	A	3	Electric Power (ED)	Generators
10)	6/17	97	F	Manual turbine trip — vibration.	A	1	Steam & Power (HA)	Turbines
11)	6/23	87	F	Main steam line isolation valve malfunction.	A	3	Steam & Power (HB)	Valves
12)	7/24	12	F	Lightning strike in switchyard caused turbine trip.	H	3	Electric Power (EA)	None
13)	10/10	44	S	To replace power cables for lake intake heaters.	B	1	Electric Power (EB)	Electrical Conductors
14)	12/30	44	F	Steam generator tube leak.	A	1	Steam & Power (HB)	Heat Exchangers

I. Summary

Description	Performance		Outages		
Location: Baxley, Georgia	Net Electrical Energy Generated (MWH):	3,102,479	Total No.	45	
Docket No: 50-321			Forced	35	
Reactor Type: BWR	Unit Availability Factor (%):	70.3	Scheduled	10	
Capacity (MWe-net): 786			Total:	2,588	Hours, 29.5%
Commercial Operation: 12/31/75	Unit Capacity Factor (%) (Using MDC):	47.1	Forced	1,062	Hours, 12.1%
Plant Age: 1.1 Years			Scheduled	1,526	Hours, 17.4%
	Unit Capacity Factor (%) (Using Design MWE):	45.1			

Cause:	
Equipment Failure	24
Maintenance or Testing	10
Regulatory Restriction	1
Operational Error	9
Other	1

Method of Shutdown:	
Manual	10
Manual Scram	6
Automatic Scram	29

II. Highlights

A. General:

The plant experienced 45 outages consuming 2588 hours during the year. The unit had an availability factor of 70.3%.

B. Outages:

1. Forced: There were 35 forced outages during the year accounting for 1062 hours. The longest forced outage was for 194 hours to radiograph welds on the recirculation bypass line in the drywell.

2. Scheduled: Ten scheduled outages during the year accounted for 1526 hours; of which, the longest outage was for 949 hours to implement an LPRM vibration fix.

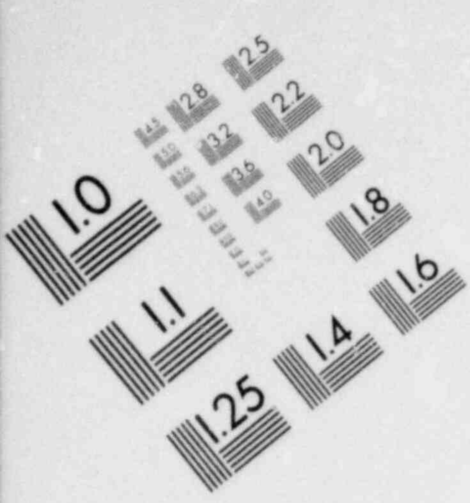
DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	24	S	Continued from 12/28/74 scram-startup testing.	B	3	Instrumentation & Controls (IA)	None
2)	1/2	4	S	Shutdown from outside the control room test.	B	1	Instrumentation & Controls (IC)	None
3)	1/2	33	S	Loss of offsite power test.	B	3	Electric Power (EA)	None
4)	1/8	10	F	Technician tripped main steam line monitor.	G	3	Instrumentation & Controls (IA)	None
5)	1/10	13	F	Control valve partially closed causing low pressure in main steam line.	A	3	Steam & Power (HA)	Valves
6)	1/11	22	F	High H_2 in off gas hold up line.	A	3	Radioactive Waste (MB)	None
7)	1/13	14	F	Condensate booster pumps tripped.	A	3	Steam & Power (HH)	Pumps

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART



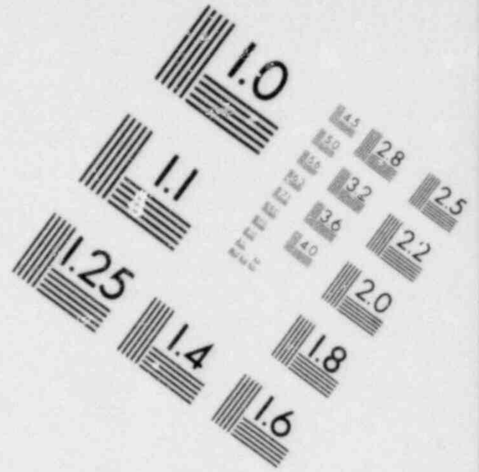
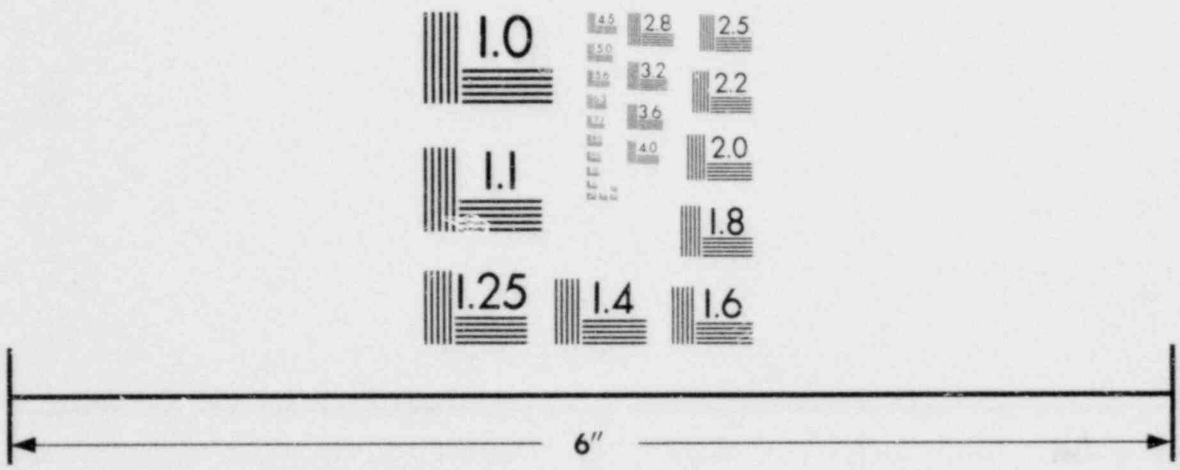
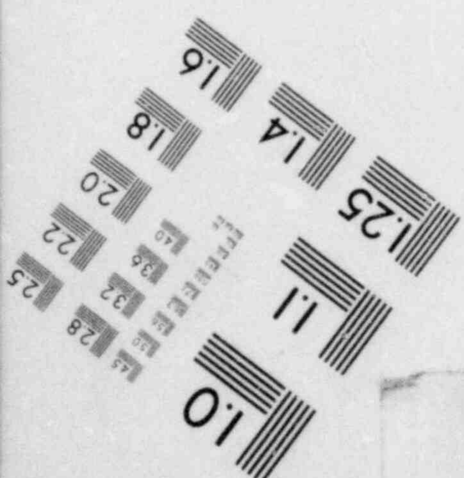
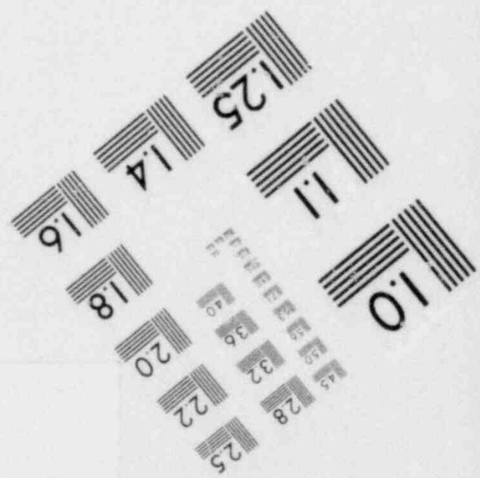


IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART





DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	1/17	29	S	Test — turbine trip.	B	2	Steam & Power (HA)	Turbines
9)	1/19	29	F	Technician error caused pressure spike that scrammed reactor.	G	3	Instrumentation & Controls (IA)	None
10)	1/24	64	S	Leakage in drywell including a recirc. discharge valve.	B	2	Reactor Coolant (CB)	Pipes, Fittings
11)	2/2	194	F	Shut down to radiograph welds in drywell on recirc bypass line.	D	2	Reactor Coolant (CB)	Pipes, Fittings
12)	2/11	40	F	RPS trip due to press. setpoint for turbine stop valve closure trip.	G	3	Steam & Power (HA)	Instrumentation & Controls
13)	2/14	10	F	Closure of MSIV.	A	3	Reactor Coolant (CD)	Valves
14)	2/19	23	F	Hookup error caused feedwater pump to overspeed.	G	3	Reactor Coolant (CH)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	2/21	42	F	Water level instrument spike caused turbine trip.	A	3	Steam & Power (HA)	Instrumentation & Controls
16)	2/23	8	F	Steam flow signal was grounded causing scram.	A	3	Instrumentation & Controls (IA)	Instrumetation & Controls
17)	3/5	68	F	EHC leak at turbine control valve.	A	[illegible]	Steam & Power (HA)	Pipes, Fittings
18)	3/7	4	F	Hi SRM trip due to cold water slug.	G	3	Instrumentation & Controls (IA)	None
19)	3/8	53	F	Manual turbine trip-relief valve stuck open.	A	1	Reactor Coolant (CC)	Valves
20)	3/12	20	F	Reactor pressure switch spike on surveillance testing.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
21)	3/13	13	F	Loss of feedwater due to RFP trip on condensate filter/demin. isolation.	G	3	Steam & Power (HC)	Demineralizers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
22)	3/26	13	F	Recirc. pump slowed down — speed mismatch. Trip on hi APRM flux.	G	3	Reactor Coolant (CB)	Pumps
23)	4/14	11	F	Cooling water leak in generator rectifier panel.	A	2	Steam & Power (HA)	Generators
24)	4/17	11	F	Spurious high reactor pressure.	A	3	Instrumentation & Controls (IA)	None
25)	4/19	382	S	Repair — EHC oil leak & turbine stop valve and control intercept valve screen removal.	B	2	Steam & Power (HA)	Pipes, Fittings
26)	5/5	26	F	Feedwater pump trip causing low water level.	A	3	Reactor Coolant (CH)	Pumps
27)	5/8	15	F	EHC line blow off on control intercept valve.	A	3	Steam & Power (HA)	Pipes, Fittings
28)	5/13	16	F	Leaks on moisture separator reheater drain tanks.	A	1	Steam & Power (HB)	Vessels, Pressures
29)	5/18	20	F	Loss of DC power to EHC panel causing turbine trip.	A	3	Electric Power (EC)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
30)	5/25	9	F	Feedwater pump tripped causing turbine trip.	A	3	Reactor Coolant (CH)	Pumps
31)	6/15	18	F	High press. trip caused by vibration of instrument rack causing drift.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
32)	7/5	34	F	Leaking moisture separator reheaters needed repair.	A	1	Steam & Power (HB)	Heat Exchangers
33)	7/8	33	F	To comply with Tech Spec on ADS valve operability.	A	1	Reactor Coolant (CC)	Valves
34)	7/13	24	F	RFP tripped on low vacuum.	A	1	Reactor Coolant (CH)	Pumps
35)	7/28	76	F	ADS relief valves opened during testing — tripped on low press.	G	3	Reactor Coolant (CC)	Valves
36)	8/8	91	F	Loss of condenser vacuum — steam jet air ejector problems.	A	2	Steam & Power (HC)	Heat Exchangers
37)	9/3	14	S	Moistu e separator reheater hi level during test.	B	3	Steam & Power (HB)	None

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
38)	9/7	15	S	Generator load rejection test.	B	3	Steam & Power (HA)	Generators
39)	9/13	12	S	MSIV closure test.	B	3	Reactor Coolant (CD)	Valves
40)	9/21	23	F	Repair steam leaks — turbine bldg.	A	1	Steam & Power (HB)	Pipes, Fittings
41)	9/30	33	F	Repair steam leaks.	A	1	Reactor Coolant (CC)	Pipes, Fittings
42)	10/5	18	F	Power/load unbalance test.	B	3	Steam & Power (HA)	Generators
43)	11/1	10	F	DW equipment sump leak.	A	1	Engineered Safety (SA)	Vessels, Pressures
44)	11/16	949	S	Shut down to implement LPRM vibration fix.	H	1	Instrumentation & Controls (ID)	Instrumentation & Controls
45)	12/28	18	F	Malfunction in MSIV limit switch during closure testing.	A	3	Reactor Coolant (CD)	Instrumentation & Controls

HUMBOLDT BAY

I. Summary

Description

Location: Humboldt, California
Docket No: 50-133
Reactor Type: BWR
Capacity (MWe-net): 65
Commercial Operation: 8/63
Plant Age: 12.7 Years

Performance

Net Electrical Energy Generated (MWH):	382,938
Unit Availability Factor (%):	83.9
Unit Capacity Factor (%) (Using MDC):	69.4
Unit Capacity Factor (%) (Using Design MWE):	69.4

Outages

Total No.	11	
Forced	6	
Scheduled	5	
Total:	1,413	Hours, 16.1%
Forced	145	Hours, 1.6%
Scheduled	1,268	Hours, 14.5%

Cause:

Equipment Failure	4
Maintenance or Testing	3
Refueling	1
Regulatory Restriction	1
Operational Error	2

Method of Shutdown:

Manual	5
Automatic Scram	6

II. Highlights

A. General:

A total of 11 outages occurred during 1975; 4 were directly related to equipment failures; 3 for maintenance; 1 for refueling; 1 for regulatory restriction; and 2 for operating errors.

B. Outages:

1. Forced: There were six forced outages during 1975; the one of longest duration (63 hrs.) was caused by logic problems with the steam bypass valves.

2. Scheduled: Of the five scheduled shutdowns, the ones of longest duration were: (1) 938 hours, for refueling and; (2) 192 hours, for repairs of a reactor head o-ring leak.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	56	S	Repaired drywell cooling fan and cooler outlet dampers.	B	1	Engineered Safety (SA)	Blowers
2)	1/20	1	S	Repaired reactor feed-pump oil line.	B	1	Reactor Coolant (CH)	Pipes, Fittings
3)	2/7	81	S	Completed NDT inspection of core spray piping and 3 accessible welds on feed-water line as required by IE Bulletin 75-01.	D	1	Engineered Safety (SF)	Pipes, Fittings
4)	5/30	938	S	Refueling	C	1	Reactor (RC)	Fuel Elements
5)	7/14	192	S	Repair reactor head o-ring leak and drywell cooling fan.	B	1	Reactor Coolant (CA)	Vessels, Pressures
6)	8/21	38	F	Generator trip — reactor trip on low condenser vacuum. (No cause of generator trip found.)	A	3	Steam & Power (HC)	Instrumentation & Controls
7)	9/6	10	F	Generator trip — reactor trip on low condenser vacuum.	A	3	Steam & Power (HC)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	9/9	63	F	Generator trip — reactor trip on low condenser vacuum. Vacuum trip due to bypass valves — logic problems.	A	3	Steam & Power (HC)	Instrumentation & Controls
9)	10/3	8	F	Technical maintenance error. (During replacement of range switch, contacts on another range switch inadvertently shorted.)	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
10)	10/13	7	F	Turbine control problems. Frequency controller caused a load rejection that upset the feedwater condensate system. During recovery, an operator error caused the reactor trip.	G	3	Steam & Power (HA)	Instrumentation & Controls
11)	11/14	19	F	Generator tripped when control power fuses blew. Loss of generator caused pressure spike that tripped reactor on high flux.	A	3	Steam & Power (HA)	Circuit Closers

I. Summary

Description

Location: Indian Point, New York
Docket No: 50-247
Reactor Type: PWR
Capacity (MWe-net): 873
Commercial Operation: 8/73
Plant Age: 2.5 Years

Performance

Net Electrical Energy Generated (MWH):	4,885,079
Unit Availability Factor (%):	74.8
Unit Capacity Factor (%) (Using MDC):	64.5
Unit Capacity Factor (%) (Using Design MWE):	63.8

Outages

Total No.	53	
Forced	41	
Scheduled	12	
Total:	2,212	Hours, 25.2%
Forced	652	Hours, 7.4%
Scheduled	1,560	Hours, 17.8%

Cause: Equipment Failure	38
Maintenance or Testing	12
Operational Error	1
Other	2
Method of Shutdown:	
Manual	12
Automatic Scram	38

II. Highlights

A. General:

A total of 53 outages occurred in 1975. Thirty-eight were the result of equipment failure; 12 were for maintenance and/or testing; two were due to external causes; and one was the result of an operational error. Ten of the outages were related to problems with valves; 17 were the result of problems with the steam generator; 8 were the result of problems with instrumentation; and 8 were the result of problems with pumps. Numerous feedwater trips occurred on startup due to poor feedwater flow control at low loads.

B. Outages:

1. Forced: There were 41 forced outages which occurred in 1975. Of these, the ones of longest duration were: (1) [illegible]50 hours, due to a generator fault; and (2) 43 hours due to a loss of a heater drain pump.

2. Scheduled: There were 12 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 825 hours, for a changeover to all volatile treatment of the steam generator water chemistry; (2) 318 hours, for an inspection of seismic restraints and repair of seals on reactor coolant pumps; and (3) 55 hours; to replace two control rod drive cooling fans.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	94	S	Inspection of seismic pipe restraints.	B	1	Engineered Safety (SF)	Shock Suppressors
2)	1/5	43	F	Low level on steam generator caused by loss of heater drain pump.	A	3	Steam & Power (HH)	Pumps
3)	1/31	73	S	Inspection of seismic pipe restraints.	B	1	Engineered Safety (SF)	Shock Suppressors
4)	2/11	2	F	Electrical problem at sub-station.	H	3	Electric Power (EA)	Circuit Closers
5)	2/28	825	S	Changeover to all volatile treatment of steam generator water chemistry (secondary side).	B	1	Steam & Power (HB)	Heat Exchangers
6)	4/5	26	S	Automatic shutdown to perform turbine overspeed trip.	B	3	Steam & Power (HA)	Turbines
7)	4/6	12	F	Automatic trip due to turbine governor control valve not responding.	A	3	Steam & Power (HA)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/7	9	F	Low vacuum trouble.	A	1	Steam & Power (HC)	Heat Exchangers
9)	4/15	20	S	Repair to vent valve connection on discharge line from the main boiler feed pump.	B	-	Steam & Power (HH)	Valves
10)	4/20	2	F	Steam generator low level and steam flow/feedwater flow mismatch due to loss of heater drain pump.	A	1	Steam & Power (HH)	Pumps
11)	5/2	48	S	Scheduled outage to repair leaking nipple on main boiler feed pump discharge piping.	B	1	Steam & Power (HH)	Pipes, Fittings
12)	5/18	4	F	Spurious trip while performing overspeed trip test.	A	3	Steam & Power (HA)	Instrumentation & Controls
13)	5/18	9	F	Trip due to high level on steam generator.	A	3	Steam & Power (HB)	Heat Exchangers
14)	6/2	21	F	Unit trip due to loss of electrical feeder.	A	3	Electric Power (EB)	Electrical Conductors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	6/19	7	F	Unit tripped due to steam generator low level and steam flow/feedwater flow mismatch as a result of failure of static inverter which feeds instrument bus.	A	3	Electric Power (ED)	Generators
16)	6/27	5	F	Unit tripped due to steam generator low level and steam flow/feedwater flow mismatch caused by switching of level control channel during periodic testing.	A	3	Steam & Power (HB)	Instrumentation & Controls
17)	6/30	4	F	Unit tripped due to inadvertent closing of one turbine stop valve.	G	3	Steam & Power (HA)	Valves
18)	7/13	12	F	Unit trip due to steam generator mismatch caused by overspeed trip of main boiler feed pump.	A	3	Steam & Power (HH)	Pumps
19)	7/14	23	S	Repair leak on heater drain tank pump discharge header drain valve.	B	1	Steam & Power (HH)	Valves
20)	7/20	9	F	Unit tripped due to system disturbance which affected the instrument busses.	H	3	Electric Power (EA)	Electrical Conductors
21)	7/28	318	S	Inspection of seismic restraints and repair seals on reactor coolant pumps.	B	3	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
22)	8/12	19	F	Reactor trip due to low level on steam generator caused by main steam valve closing.	A	3	Steam & Power (HB)	Valves
23)	8/13	7	F	Reactor trip due to low level on steam generator caused by main steam valve closing.	A	3	Steam & Power (HB)	Valves
24)	8/14	19	F	Reactor trip due to spurious over-power ΔT protection signal.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
25)	8/17	2	S	Turbine taken off to repair leaking valve on heater drain tank pump discharge header.	B	NA	Steam & Power (HH)	Valves
26)	8/17	1	F	Turbine trip due to high level on steam generator.	A	3	Steam & Power (HB)	Heat Exchangers
27)	8/17	10	F	Reactor trip due to high level on steam generator.	A	3	Steam & Power (HB)	Heat Exchangers
28)	8/22	4	F	Reactor trip due to steam generator mismatch.	A	3	Steam & Power (HH)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
29)	8/22	2	F	Reactor trip due to safety injection signal caused by low T avg. and spurious high steam flow signal.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
30)	8/23	30	S	Unit taken off to balance reactor coolant pump.	B	1	Reactor Coolant (CB)	Pumps
31)	8/29	2	F	Reactor trip due to steam generator low level with steam flow/feedwater flow mismatch.	A	3	Steam & Power (HH)	Heat Exchangers
32)	9/7	3	F	Reactor tripped due to steam generator low level mismatch caused by trip of main boiler fuel pump.	A	3	Steam & Power (HH)	Pumps
33)	9/12	46	S	Repair of heater drain tank level control valves and leaking nipple in downstream piping.	B	1	Steam & Power (HH)	Valves
34)	9/28	18	F	Unit taken off to repair weld leak on main boiler feed pump discharger header drain valve.	A	1	Steam & Power (HH)	Valves
35)	9/29	2	F	Reactor tripped due to high level on steam generator.	A	3	Steam & Power (HB)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
36)	10/4	4	F	Turbine trip — boiler feed pump control trouble.	A	3	Steam & Power (HH)	Pumps
37)	10/5	6	F	Turbine trip — steam generator high level.	A	3	Steam & Power (HB)	Heat Exchangers
38)	10/11	9	F	Turbine trip — steam generator low level.	A	3	Steam & Power (HB)	Heat Exchangers
39)	10/16	350	F	Generator trip — generator fault.	A	3	Steam & Power (HA)	Generators
40)	10/31	3	F	Unit trip — steam generator high level.	A	3	Steam & Power (HB)	Heat Exchangers
41)	11/9	3	F	Unit tripped due to steam generator lo level mismatch.	A	3	Steam & Power (HB)	Heat Exchangers
42)	11/14	55	S	Replace two defective control rod drive mechanisms cooling fans.	B	1	Reactor (RB)	Blowers
43)	11/17	1	F	Unit tripped due to high level in steam generator.	A	3	Steam & Power (HB)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
44)	11/17	1	F	Turbine trip due to high level in the steam generator.	A	NA	Steam & Power (HB)	Heat Exchangers
45)	11/17	1	F	Turbine trip due to high level in the steam generator.	A	NA	Steam & Power (HB)	Heat Exchangers
46)	11/27	10	F	Unit tripped due to steam generator mismatch caused by control failure of heater drain tank pump level control valve.	A	3	Steam & Power (HH)	Instrumentation & Controls
47)	11/28	1	F	Steam generator mismatch. (Poor feedwater control at low levels.)	A	3	Steam & Power (HH)	Instrumentation & Controls
48)	11/28	5	F	Steam generator mismatch. (Poor feedwater control at low levels.)	A	3	Steam & Power (HH)	Instrumentation & Controls
49)	12/13	5	F	Unit trip due to steam generator low level mismatch.	A	3	Steam & Power (HH)	Heat Exchangers
50)	12/23	20	F	Steam generator low level due to main steam line isolation valve closing.	A	3	Steam & Power (HH)	Valves
51)	12/23	1	F	Unit tripped due to steam generator high level.	A	3	Steam & Power (HB)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
52)	12/23	3	F	Unit tripped due to steam generator high level.	A	3	Steam & Power (HB)	Instrumentation & Controls
53)	12/27	3	F	Unit trip due to steam generator low level caused by condensate pump deterioration.	A	3	Steam & Power (HH)	Pumps

I. Summary

Description

Location: Carlton, Wisconsin
Docket No: 50–305
Reactor Type: PWR
Capacity (MWe-net): 560
Commercial Operation: 6/16/74
Plant Age: 1.7 Years

Performance

Net Electrical Energy Generated (MWH):	3,341,153
Unit Availability Factor (%):	88.2
Unit Capacity Factor (%) (Using MDC):	71.3
Unit Capacity Factor (%) (Using Design MWE):	68.1

Outages

Total No.	32	
Forced	26	
Scheduled	6	
Total:	1,032	Hours, 11.8%
Forced	743	Hours, 8.5%
Scheduled	289	Hours, 3.3%

Cause:	
Equipment Failure	24
Maintenance or Testing	6
Operational Error	2
Other	1

Method of Shutdown:	
Manual	11
Manual Scram	2
Automatic Scram	18

II. Highlights

A. General:

A total of [illegible] outages occurred in 1975. Twenty-four were the result of equipment failure; 6 were for maintenance; and 2 were the result of operational errors. Seven of the outages were related to problems with valves; 5 were related to problems with instrumentation; 10 were related to problems with electrical systems; 2 were related to problems with pumps; and 4 were related to problems with piping.

B. Outages:

1. **Forced:** There were 26 forced outages in 1975. Of these, the ones of longest duration were: (1) 104 hours for repairs of an auxiliary bus; (2) 125 hours for repairs of leaks in the steam generator; (3) 205 hours to overhaul the main steam isolation valves; and (4) 44 hours for repair of leaking pressurizer relief valve.

2. Scheduled: There were 6 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 94 hours for repair of feedwater heater leaks; (2) 95 hours for general maintenance; and (3) 39 hours for inspection of the feedwater heaters and repair of cracked welds.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/4	25	F	Feedwater regulating valve sticky during load reduction.	A	3	Steam & Power (HH)	Valves
2)	1/18	39	S	Inspect feedwater heater and repair cracked weld on charging pump.	B	1	Steam & Power (HH)	HEAT EXCHANGERS
3)	1/24	2	F	Failed power supply in turbine overspeed system.	A	3	Steam & Power (HA)	Instrumentation & Controls
4)	2/7	28	S	Unit tripped while decreasing load to repair turbine EH controller.	B	3	Steam & Power (HA)	Circuit Closers
5)	2/21	7	S	Repair leak on instrument line on steam generator.	B	NA	Steam & Power (HB)	Pipes, Fittings
6)	3/28	31	F	Opening of 345-KV breaker caused load swing which lifted relief valves on the moisture separator. Repair of EH controller and relief valves.	A	1	Electric Power (ED)	Circuit Closers
7)	4/2	14	F	Bus fault in main auxiliary bus resulting in loss of main auxiliary transformer.	A	3	Electric Power (EB)	Electrical Conductors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/5	26	S	Unit off line to repair turbine EH control system.	B	1	Steam & Power (HA)	Circuit Closers
9)	4/26	104	F	Repair of overheating reserve auxiliary bus work and gasket replacement on Main Steam Isolation Valve Bypass.	A	1	Electric Power (EB)	Electrical Conductors
10)	5/2	3	F	Operator mistakenly opened output relay during load adjustment following recovery from previous shutdown.	G	3	Electric Power (ED)	Circuit Closers
11)	5/7	13	F	Repair crack in main feed pump recirc line.	A	1	Steam & Power (HH)	Pipes, Fittings
12)	5/8	8	F	During rapid load reduction required to remove a feed pump which had lost pressure and flow indication, a low steam generator water level trip occurred. Discovered casing crack in feedwater pump.	A	3	Steam & Power (HH)	Pumps
13)	6/19	94	S	Repair of feedwater heater extraction line leak.	B	1	Steam & Power (HH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	6/27	8	F	Developed ground fault in 1A feedwater pump.	A	3	Steam & Power (HH)	Pumps
15)	6/28	11	F	Feedwater regulating valve failed to seat properly resulting in loss of steam generator water level control.	A	3	Steam & Power (HH)	Valves
16)	7/12	6	F	Malfunction of turbine control system caused load rejection which resulted in loss of steam generator water level control and unit trip.	A	3	Steam & Power (HA)	Instrumentation & Controls
17)	7/18	44	F	Excessive primary leakage from pressurizer power operated relief valve.	A	1	Reactor Coolant (CB)	Valves
18)	8/28	3	F	A lightning strike on 345-KV output lines caused a turbine and reactor trip.	H	3	Electric Power (EA)	Electrical Conductors
19)	8/28	1	F	Steam generator water level control system failed to limit steam generator swell resulting in a high water level trip during load increase from previous shutdown.	A	3	Steam & Power (HB)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
20)	9/1	9	F	Malfunction of overpower delta T protection caused false shutdown signal.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
21)	9/2	6	F	Blown fuse to control rod drive mechanism caused rod drop followed by high flux rate trip.	A	3	Reactor (RB)	CRDRVE
22a)	9/11	95	S	Maintenance shutdown to repair leaking pressurizer manway cover.	B	1	Reactor Coolant (CB)	Vessels, Pressures
22b)	9/11	125	F	Forced to extend maintenance shutdown to repair leaks in the steam generator manways.	A	—	Steam & Power (HB)	Heat Exchangers
23)	10/15	12	F	Worn bushing in main steam isolation valve resulted in a valve dip signal due to excessive clearance to microswitch. Repair on first trip proved inadequate and unit was shut down to perform overhaul of MSIVs.	A	3	Steam & Power (HB)	Valves
24)	10/25	13	F	Repair MSIV	A	3	Steam & Power (HB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
25)	10/26	205	F	Repair MSIVs.	A	2	Steam & Power (HB)	Valves
26)	11/4	28	F	Repair overheating bus work.	A	1	Electric Power (ED)	Electrical Conductors
27)	11/8	38	F	Repair overheating bus work.	A	1	Electric Power (ED)	Electrical Conductors
28)	11/13	12	F	Loss of EH pressure caused load rejection from full load; resulting system upset caused steam generator water levels to get out of control and subsequently tripped reactor.	A	3	Steam & Power (HA)	Turbines
29)	11/14	5	F	Repair cracked warm up line on feedwater pump.	A	1	Steam & Power (HH)	Pipes, Fittings
30)	11/27	5	F	Spurious closure to turbine stop valve caused unit to trip.	A	3	Steam & Power (HA)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
31)	12/4	6	F	Loss of power to non-interruptable bus resulted in loss of turbine overspeed control system.	G	3	Steam & Power (HA)	Electrical Conductors
32)	12/19	6	F	Apparent malfunction of EH control system resulted in unit load swings until manually tripped.	A	2	Steam & Power (HA)	Instrumentation & Controls

I. Summary

Description

Location: Genoa, Wisconsin
Docket No: 50–409
Reactor Type: BWR
Capacity (MWe-net): 50
Commercial Operation: 9/69
Plant Age: 7.7 Years

Performance

Net Electrical Energy Generated (MWH):	263,368
Unit Availability Factor (%):	69.6
Unit Capacity Factor (%) (Using MDC):	62.6
Unit Capacity Factor (%) (Using Design MWE):	60.1

Outages

Total No.	20		
Forced	13		
Scheduled	7		
Total:	2,702	Hours,	30.8%
Forced	318	Hours,	3.6%
Scheduled	2,384	Hours,	27.2%

Cause:	
Equipment Failure	8
Maintenance or Testing	5
Refueling	1
Regulatory Restriction	1
Operational Error	5

Method of Shutdown:	
Manual	8
Manual Scram	1
Automatic Scram	11

II. Highlights

A. General:

Operations were fairly normal for the year. An annual refueling was performed during May — June. There were a total of 20 outages during the year consuming 2702 hours; the refueling outage consumed 2153 hours.

B. Outages:

1. Forced: There were 13 forced outages during the year consuming 318 hours. The longest forced outage was 190 hours as a result of the loss of seal injection supply to a forced circulation pump.

2. Scheduled: There were 7 scheduled outages during the year consuming 2384 hours. A refueling outage starting in May required 2153 hours. The only other outage which consumed more than 100 hours was a 103 hour outage to locate a DC ground and repairs to a transfer switch.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/13	32	F	Improper switching of safety system channel.	G	3	Instrumentation & Controls (IA)	None
2)	2/15	62	S	Shut down in response to NRC Bulletin to inspect core spray piping.	D	1	Engineered Safety (SF)	Pipes, Fittings
3)	2/21	4	F	Incorrect switching of safety system instrumentation.	G	3	Instrumentation & Controls (IA)	None
4)	3/10	19	F	Spurious hi flux spike.	A	3	Instrumentation & Controls (IA)	None
5)	3/14	7	F	Operator turned wrong nuclear channel for calibration.	G	3	Instrumentation & Controls (IA)	None
6)	3/21	20	F	Suction bellows to condensate pump partially collapsed.	A	3	Steam & Power (HH)	Pipes, Fittings
7)	4/17	2	F	Operator turned wrong switch on bench board — steam isolation.	G	3	Reactor Coolant (CC)	None

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	4/17	103	S	DC ground & transfer switch damaged.	B	1	Electric Power (EC)	Circuit Closers
9)	5/9	2153	S	Refueling	C	3	Reactor (RC)	Fuel Elements
10)	8/13	8	F	Loss of seal injection supply to forced circulation pump (FCP).	A	3	Reactor Coolant (CB)	Pumps
11)	8/13	5	F	Loss of relay in scram circuit.	A	3	Instrumentation & Controls (IA)	Relays
12)	8/14	190	F	Loss of seal injection supply to FCP.	A	3	Reactor Coolant (CB)	Pumps
13)	8/18	5	S	Oil leak on control rod drive accumulator repaired.	B	1	Reactor (RB)	Control Rod Drive Mechanisms
14)	8/28	4	S	Replace leather seals on FCP suction and discharge valves.	B	1	Reactor Coolant (CB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	9/14	16	S	Maintenance on FCP roto-valve operators.	B	1	Reactor Coolant (CB)	Valve Operators
16)	9/17	3	F	Replaced velocity limiter in channel 7 of P/F circuit.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
17)	12/11	41	S	Quarterly Tech Spec test.	B	1	Reactor Coolant (CX)	None
18)	12/15	6	F	Repairs to sampling line connection on feedwater system.	A	1	Reactor Coolant (CH)	Pipes, Fittings
19)	12/15	7	F	Repairs to sampling line connection on feedwater system.	A	1	Reactor Coolant (CH)	Pipes, Fittings
20)	12/31	15	F	Mechanic tripped electrical breaker for coupling fluid pump motor.	G	2	Electric Power (ED)	None

I. Summary

Description

Location: Wiscasset, Maine
Docket No: 50-309
Reactor Type: PWR
Capacity (MWe-Net): 790
Commercial Operation: 12/28/72
Plant Age: 3.1 Years

Performance

Net Electrical Energy Generated (MWH):	4,502,452
Unit Availability Factor (%):	79.9
Unit Capacity Factor (%) (Using MDC):	67.6
Unit Capacity Factor (%) (Using Design MWE):	65.1

Outages

Total No.	14
Forced	10
Scheduled	4
Total:	1,757 Hours, 20.1%
Forced	265 Hours, 3.0%
Scheduled	1,492 Hours, 17.1%

Cause:	
Equipment Failure	9
Maintenance or Testing	2
License Exam	1
Refueling	1
Operational Error	1

Method of Shutdown	
Manual	5
Manual Scram	4
Automatic Scram	5

II. Highlights

A. General:

A total of 14 outages occurred in 1975. Nine were the result of equipment failures; 2 were for maintenance and/or test; 1 was for a license exam; 1 was for refueling; and 1 was the result of an operational error. Two of the shutdowns were related to problems with valves; 2 were related to problems with piping; 2 were related to problems with instrumentation; 5 were related to problems with electrical equipment; and 1 was related to a problem with control rods.

B. Outages:

1. Forced: There were 10 forced outages in 1975. Of these, the ones of longest duration were: (1) 145 hours, for an examination of feedwater piping following a transient; and (2) 25 hours, due to failure of an M-G set output breaker.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 1356 hours, for refueling and maintenance; and (2) 87 hours, for replacement of cables on a motor control valve in the reactor coolant system.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/13	9	F	Failure of control air supply to feedwater valve.	A	2	Auxiliary Process (PA)	Valves
2)	1/16	145	F	Non-destructive examination of feedwater piping as a result of transient induced by failure of feedwater valve.	A	1	Reactor Coolant (CH)	Pipes, Fittings
3)	2/21	30	S	Operator licensing exams.	E	1	Reactor (RB)	Control Rod Drive Mechanisms
4)	3/1	8	F	Inadvertent boration followed by heater drain tank level malfunction.	G	2	Auxiliary Process (PC)	Instrumentation & Controls
5)	4/26	11	F	Failure of a 15 V power supply caused a total of 4 CEA's to drop into the core.	A	2	Electric Power (EC)	Electrical Conductors
6)	5/2	1356	S	Refueling & maintenance.	C	1	Reactor (RC)	Fuel Elements
7)	6/30	25	F	Failure of M-G set output breakers.	A	3	Electric Power (ED)	Circuit Closers
8)	7/16	19	S	Tie in of south leg of the cooling water discharge diffuser.	B	1	Auxiliary Water (WE)	Pipes, Fittings
9)	7/18	14	F	Condenser tube leak.	A	2	Steam & Power (HC)	Heat Fxchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	8/10	20	F	Closure of 1 steam line excess flow check valve due to low air pressure resulted in low steam generator level trip.	A	3	Auxiliary Process (PA)	Valves
11)	11/5	14	F	Rod drive M-G set output breakers failed open.	A	3	Reactor (RB)	Circuit Closers
12)	11/6	12	F	Turbine governor valve control system failure.	A	3	Steam & Power (HA)	Instrumentation & Controls
13)	11/14	87	S	Replaced RCS MCV cables & cleaned condenser tubes.	B	1	Reactor Coolant (CB)	Electrical Conductors
14)	12/26	7	F	Spike in delta T caused by an apparent disturbance in the ground system resulted in a high power trip.	A	3	Instrumentation & Controls (IA)	Electrical Conductors

MILLSTONE POINT 1

I. Summary

Description	Performance		Outages		
Location: Waterford, Connecticut	Net Electrical Energy Generated (MWH):	3,896,991	Total No.	17	
Docket No: 50-245			Forced	14	
Reactor Type: BWR	Unit Availability Factor (%):	75.6	Scheduled	3	
Capacity (MWe-net): 690			Total:	2,135	Hours, 24.4%
Commercial Operation: 3/71	Unit Capacity Factor (%) (Using MDC):	68.4	Forced	1,516	Hours, 17.3%
Plant Age: 5.1 Years			Scheduled	619	Hours, 7.1%
	Unit Capacity Factor (%) (Using Design MWE):	68.4			

Cause: Equipment Failure 14
Maintenance or Testing 3

Method of Shutdown:
Manual 13
Manual Scram 1
Automatic Scram 3

II. Highlights

A. General:

A total of 17 outages occurred in 1975. Fourteen were caused by equipment failure and 3 were for maintenance and/or tests. Three of the outages were related to problems with the main transformer; 3 were related to problems with piping; 4 were related to problems with instrumentation; and 5 were related to problems with valves.

B. Outages:

1. Forced: There were 14 forced outages in 1975. Of these, the ones of longest duration were: (1) 961 hours, due to a breakdown of the insulation in the main transformer; (2) 223 hours, due to a crack in the jet pump break detection system; and (3) 87 hours, for repairs of a leaking valve.

2. Scheduled: There were 3 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 176 hours, for compliance with Bulletin 75-01 (to examine core spray lines); and (2) 399 hours, for repairs of the low pressure sensing lines on flow restrictors.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/8	176	S	Shutdown in compliance with actions required in Bulletin 75-01, for examination of core spray lines. Repair to M.S. flow restrictor sensing line extended this outage.	B	1	Engineered Safety (SF)	Pipes, Fittings
2)	2/15	9	F	Drywell entry made to perform maintenance on T.I.P. index.	A	1	Instrumentation & Controls (ID)	Instrumentation & Controls
3)	3/11	87	F	During routine surveillance a leak was noted in a vent/test line, outside the drywell, connected to LPCI system. Investigation revealed a blown valve stuffing box.	A	1	Engineered Safety (SF)	Valves
4)	4/26	44	S	Plant was shut down due to increasing trend of unidentified leakage in the drywell. Found valve packing leak.	B	1	Reactor Coolant (CB)	Valves
5)	5/20	55	F	While lift testing Target Rock APR's, one remained unseated which resulted in an uncontrolled blowdown to 160 psig. A new pilot section was later installed in the valve.	A	2	Engineered Safety (SF)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
6)	6/20	35	F	Unit was shut down for service water pump repairs.	A	1	Auxiliary Water (WA)	Pumps
7)	7/27	399	S	Unit was shut down to repair low pressure sensing lines on main steam line flow restrictors.	B	1	Reactor Coolant (CC)	Pipes, Fittings
8)	8/13	16	F	Unit taken off line because of excessive arcing observed on B phase of main disconnects.	A	1	Electric Power (EA)	Circuit Closers
9)	8/18	5	F	Two automatic pressure-relief valves failed to meet tech. specs.	A	1	Engineered Safety (SF)	Valves
10)	8/30	8	F	An electric pressure regulator pressure transient caused APRM scram. Problem attributed to clogged valve filter.	A	3	Reactor Coolant (CC)	Valves
11)	9/12	961	F	An alarm in the control room led to discovery of an explosive gas mixture in the main transformer. An investigation revealed a breakdown of insulation.	A	1	Electric Power (EA)	Transformers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
12)	10/25	15	F	Removed unit from service for maintenance on main transformer.	A	1	Electric Power (EA)	Transformers
13)	10/27	19	F	Shut down for maintenance on transverse incore probe system.	A	1	Instrumentation & Controls (ID)	Instrumentation & Controls
14)	11/13	223	F	Unit removed from service to investigate source of increasing unidentified drywell leakage. Found crack in jet pump break detection sensing line.	A	1	Reactor Coolant (CI)	Pipes, Fittings
15)	11/23	9	F	Pressure oscillations caused by faulty electric pressure regulator caused operator to shift control to the mechanical regulator. Oscillations become more severe, resulting in APRM high level scram.	A	3	Reactor Coolant (CC)	Instrumentation & Controls
16)	11/26	8	F	While switching to the mechanical pressure regulator, a pressure spike occurred which resulted in an APRM high level scram.	A	3	Reactor Coolant (CC)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	12/11	66	F	Unit was removed from service to install missing part on high side tap changer of main transformer.	A	1	Electric Power (EA)	Transformers

MONTICELLO

I. Summary

Description

Location: Monticello, Minnesota
Docket No: 50-263
Reactor Type: BWR
Capacity (MWe-net): 545
Commercial Operation: 7/4/71
Plant Age: 4.8 Years

Performance

Net Electrical Energy Generated (MWH):	2,879,458
Unit Availability Factor (%):	72.2
Unit Capacity Factor (%) (Using MDC):	61.1
Unit Capacity Factor (%) (Using Design MWE):	60.3

Outages

Total No.	6		
Forced	2		
Scheduled	4		
Total:	2,436	Hours,	27.8%
Forced	35	Hours,	0.4%
Scheduled	2,401	Hours,	27.4%

Cause:

Equipment Failure	2
Maintenance or Testing	2
Refueling	2

Method of Shutdown:

Manual	3
Automatic Scram	2

II. Highlights

A. General:

A total of 6 outages occurred in 1975. Two were the result of equipment failures; 2 were for maintenance and/or tests; and 2 were for refueling. One of the outages was related to a problem with a demineralizer; 1 was related to a problem with the pressure control instrumentation; 1 was related to a problem with valves; and 2 were for refueling.

B. Outages:

1. Forced: There were 2 forced outages in 1975. The one of longest duration was for 23 hours, due to a failure of the pressure control system.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 688 hours, for refueling; and (2) 1655 hours, for a second refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/10	688	S	Refueling. Replaced cracked recirculation valve bypass line and completed inspection requirements of IE Bulletin No. 75-1.	C	1	Reactor (RC)	Fuel Elements
2)	5/15	51	S	Reactor relief valve inspection.	B	1	Engineered Safety (SF)	Valves
3)	5/28	23	F	Pressure control system failure resulted in neutron monitoring system high flux scram.	A	3	Reactor Coolant (CC)	Instrumentation & Controls
4)	8/31	12	F	Condensate demineralizer component malfunction resulted in interruption of normal feedwater flow which caused both reactor feedwater pumps to trip on suction pressure. A reactor low water level scram was the result.	A	3	Steam & Power (HH)	Demineralizers
5)	9/11	1655	S	Refueling	C	1	Reactor (RC)	Fuel Elements
6)	11/20	7	S	Semi-annual turbine test.	B	NA	Steam & Power (HA)	Turbines

I. Summary

Description

Location: Oswego, New York
Docket No: 50–220
Reactor Type: BWR
Capacity (MWe-net): 610
Commercial Operation: 12/69
Plant Age: 6.2 Years

Performance

Net Electrical Energy Generated (MWH):	3,044,948
Unit Availability Factor (%):	72.1
Unit Capacity Factor (%) (Using MDC):	56.9
Unit Capacity Factor (%) (Using Design MWE):	56.9

Outages

Total No.	16	
Forced	13	
Scheduled	3	
Total:	2,522	Hours, 28.8%
Forced	322	Hours, 3.7%
Scheduled	2,200	Hours, 25.1%

Cause:	
Equipment Failure	12
Maintenance or Testing	1
Refueling	1
Regulatory Restriction	1
Operational Error	1

Method of Shutdown:	
Manual	10
Automatic Scram	6

II. Highlights

A. General:

A total of 16 outages occurred in 1975. Twelve were the results of equipment failure; 1 was for maintenance and/or tests; 1 was for refueling; 1 was for an NRC mandate concerning piping; and 1 was the result of an operational error. Three of the outages were related to problems with instrumentation; 4 were related to problems with valves; 2 were related to problems with the turbine governor; 1 was for refueling; 2 were related to problems with transformers; 2 were related to problems with piping; 1 was related to problems with chloride concentration; and 1 was related to a problem with a pump.

B. Outages:

1. Forced: There were 13 forced outages in 1975. Of these, the ones of longest duration were: (1) 9[illegible] hours, to correct an above normal floor drain flow in the drywell; and (2) 43 hours, for repair of leaking valve in the recirculation system.

2. <u>Scheduled</u>: There were 3 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 212 hrs., for an NRC mandate regarding piping inspections; and (2) 1577 hrs., for annual refueling and maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	98	F	Contd. from 1974. Replaced gaskets on electromatic relief valves.	A	3	Reactor Coolant (CC)	Valves
2)	1/12	14	F	Reactor trip on high water level caused by electronic failure in feedwater control system. Replaced transformer in steam flow-feedwater flow comparator.	A	3	Reactor Coolant (CH)	Instrumentation & Controls
3)	1/18	20	F	Reactor tripped on high water level due to feedwater control failure. Replaced steam flow-feedwater flow comparitor.	A	3	Reactor Coolant (CH)	Instrumentation & Controls
4)	2/3	9	F	Failure of generator's voltage regulator.	A	3	Steam & Power (HA)	Instrumentation & Controls
5)	2/3	212	S	Shut down for NRC mandated inspections of the core spray piping.	D	1	Engineered Safety (SF)	Pipes, Fittings
6)	3/8	27	F	Repair of turbine control valve.	A	1	Steam & Power (HA)	Valves
7)	3/18	4	F	Removed a loose transformer bus duct rod.	A	NA	Electric Power (ED)	Transformers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	3/20	37	F	Crack in turbine oil-to-clutch piping.	A	3	Steam & Power (HA)	Pipes, Fittings
9)	4/11	43	F	Repair packing leak in bypass valve in recirculation system.	A	1	Reactor Coolant (CB)	Valves
10)	7/25	24	F	Test switch not in TEST position while performing main steam line isolation valve test.	G	3	Reactor Coolant (CD)	Instrumentation & Controls
11)	9/13	1977	S	Annual refueling and overhaul.	C	1	Reactor (RC)	Fuel Elements
12)	12/4	11	S	Turbine overspeed governor adjustment and valve packing leaks.	B	1	Steam & Power (HA)	Mechanical Function Units
13)	12/6	8	F	Turbine overspeed governor adjustment.	A	1	Steam & Power (HA)	Mechanical Function Units
14)	12/6	10	F	Leak on valve to feedwater flow transmitter.	A	1	Reactor Coolant (CH)	Valves
15)	12/7	8	F	High chloride concentration in reactor water.	A	1	Reactor Coolant (CG)	Demineralizers
16)	12/27	20	F	Pump bearing oil leak.	A	1	Reactor Coolant (CB)	Pumps

OCONEE 1

I. Summary

Description

Location: Seneca, South Carolina
Docket No: 50-269
Reactor Type: PWR
Capacity (MWe-net): 886
Commercial Operation: 7/15/73
Plant Age: 2.7 Years

Performance

Net Electrical Energy Generated (MWH):	5,285,630
Unit Availability Factor (%):	76.2
Unit Capacity Factor (%) (Using MDC):	69.3
Unit Capacity Factor (%) (Using Design MWE)	68.9

Outages

Total No.	19		
Forced	13		
Scheduled	6		
Total:	2,085	Hours,	23.8%
Forced	171	Hours,	2.0%
Scheduled	1,914	Hours,	21.8%

Cause:

Equipment Failure	9
Maintenance or Testing	5
Refueling	1
Administrative	1
Operational Error	3
Other	1

Method of Shutdown:

Manual	8
Automatic Scram	11

II. Highlights

A. General:

A total of 19 outages occurred in 1975. One was apportioned to both refueling and maintenance. Nine were the result of equipment failure; 5 were the result of maintenance and/or tests; 1 was for refueling; 1 was for administrative reasons; 3 were the result of operational errors and one was designated as other. Five of the outages were related to problems with pumps; 1 was related to problems with control rods; 7 were related to problems with instrumentation; 4 were related to problems with the turbine; and 2 were related to problems with valves.

B. Outages:

1. **Forced:** There were 13 forced outages in 1975. The one of longest duration was for 31 hours, because of a spurious pressure/temperature trip.

2. Scheduled: There were 6 scheduled outages during the report period. The ones of longest duration were: 1676 hours, for refueling and pump maintenance; and 163 hours, for reactor coolant pump lubrication change.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1a)	1/1	744	S	Continuation of 1974 outage. Refueling and pump maintenance.	C	1	Reactor (RC)	Fuel Elements
1b)	2/1	932	S	Continuation of 1974 outage. Reactor coolant pump seal maintenance.	B	——	Reactor Coolant (CB)	Pumps
2)	3/12	13	F	Steam leak on turbine instrumentation valve.	A	3	Steam & Power (HA)	Valves
3)	3/13	6	F	Fault on ICS instrumentation.	A	3	Instrumentation & Controls (IC)	Instrumentation & Controls
4)	3/15	3	F	Turbine control oil leak.	A	1	Steam & Power (HA)	Turbines
5)	3/15	4	S	Turbine overspeed trip test.	B	3	Steam & Power (HA)	Turbines
6)	3/21	31	F	Unit trip due to spurious pressure/temperature trip. Unit off to inspect RC pump motors.	G	3	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	4/22	7	F	Integrated control system malfunction.	A	3	Instrumentation & Controls (IC)	Instrumentation & Controls
8)	4/23	9	F	Unit tripped during transient.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
9)	5/17	30	S	Control rod repatch.	F	1	Reactor (RB)	Control Rods
10)	6/8	17	F	Turbine trip due to low turbine control oil pressure.	A	3	Steam & Power (HA)	Turbines
11)	6/9	10	F	Unit tripped during restart due to high RC pressure while in manual control.	G	3	Reactor Coolant (CB)	Instrumentation & Controls
12)	7/19	11	S	Reactor coolant pump lubrication test.	B	1	Reactor Coolant (CB)	Pumps
13)	7/26	163	S	Reactor coolant pump lubrication change.	B	1	Reactor Coolant (CB)	Pumps
14)	8/2	7	F	Failure of stator cooling pressure switch.	A	3	Steam & Power (HA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	8/8	8	F	Unit trip while testing turbine valves.	A	3	Steam & Power (HA)	Turbines
16)	8/9	13	F	Trip due to flux imbalance on restart.	H	3	Instrumentation & Controls (IA)	Instrumentation & Controls
17)	11/7	17	F	Repaired packing leaks on steam generator instrumentation.	A	1	Steam & Power (HA)	Instrumentation & Controls
18)	12/5	30	S	Changed oil in reactor coolant pumps.	B	1	Reactor Coolant (CB)	Pumps
19)	12/10	30	F	Replaced failed control rod drive stator.	A	1	Reactor (RB)	Control Rod Drive Mechanisms

I. Summary

Description

Location: Seneca, South Carolina
Docket No: 50-270
Reactor Type: PWR
Capacity (MWe-net): 886
Commercial Operation: 9/9/74
Plant Age: 2.1 Years

Performance

Net Electrical Energy Generated (MWH):	4,967,625
Unit Availability Factor (%):	73.1
Unit Capacity Factor (%) (Using MDC):	65.1
Unit Capacity Factor (%) (Using Design MWe):	63.9

Outages

Total No.	18	
Forced	13	
Scheduled	5	
Total:	2,356	Hours, 26.9%
Forced	1,914	Hours, 21.8%
Scheduled	442	Hours, 5.1%

Cause:	Equipment Failure	13
	Maintenance or Testing	4
	Administrative	1
Method of Shutdown:		
	Manual	11
	Manual Scram	1
	Automatic Scram	5

II. Highlights

A. General:

A total of 18 outages occurred in 1975. Thirteen were related to equipment failure; 4 were for maintenance purposes; and one was administrative. One of the outages was caused by problems with control rod drives; and 4 were caused by problems with pumps.

B. Outages:

1. Forced: There were 13 forced outages during 1975. The ones of longest duration were: (1) 1142 hours because of leaking pressurizer relief valves; (2) 128 hours, because of excessive leakage from two reactor coolant valves; (3) 263 hours, to investigate low RC pump oil level and to perform surveillance tests; and (4) 236 hours, to repair electrical penetrations in the reactor building.

2. Scheduled: There were 5 scheduled outages during the report period. The one of longest duration was 389 hours, for general maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/3	4	F	Crack in main steam instrument line weld.	A	1	Steam & Power (HB)	Pipes, Fittings
2)	1/19	1142	F	Leaking pressurizer relief valves.	A	1	Reactor Coolant (CB)	Valves
3)	3/7	1	S	Turbine overspeed trip test.	B		Steam & Power (HA)	Turbines
4)	3/20	14	F	Failure of pressurizer spray valve motor.	A	1	Reactor Coolant (CB)	Motors
5)	3/27	16	S	Unit loss of load test and replacement of pressurizer spray valve motors.	B	2	Reactor Coolant (CB)	Valve Operators
6)	3/29	10	F	Excessive leakage on RC-1 and RC-3.	A	1	Reactor Coolant (CB)	Valves
7)	4/1	128	F	Excessive packing leakage on valves RC-1 and RC-3.	A	1	Reactor Coolant (CB)	Valves
8)	5/1	4	F	Loss of condenser vacuum.	A	3	Steam & Power (HC)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	5/2	15	S	Isothermal Reactor Coolant System temperature measurements.	B	1	Reactor Coolant (CB)	NA
10)	6/28	263	F	Shut down to investigate low RC pump oil level and to perform surveillance tests.	A	1	Reactor Coolant (CB)	Pumps
11)	8/2	21	S	Control rod repatch.	F	1	Reactor (RB)	Control Rods
12)	8/5	7	F	Faulty control rod drive power supply.	A	3	Reactor (RB)	Control Rod Drive Mechanisms
13)	8/5	22	F	Unit tripped on loss of main feedwater pump trip.	A	3	Steam & Power (HH)	Pumps
14)	8/7	41	F	Shut down to repair RC pump seal injection.	A	1	Reactor Coolant (CB)	Pumps
15)	8/23	14	F	Unit tripped during testing.	A	3	Instrumentation & Controls (IA)	NA
16)	8/29	389	S	Maintenance shutdown (pumps, etc.).	B	1	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	9/19	29	F	Spurious generator relay operation.	A	3	Steam & Power (HA)	Relays
18)	10/24	236	F	Repaired reactor building electrical penetration.	A	1	Engineered Safety (SA)	Penetrations

I. Summary

Description

Location: Oconee Co., South Carolina
Docket No: 50-287
Reactor Type: PWR
Capacity (MWe-net): 986
Commercial Operation: 12/16/74
Plant Age: 1.3 Years

Performance

Net Electrical Energy Generated (MWH):	5,037,298
Unit Availability Factor (%):	77.2
Unit Capacity Factor (%) (Using MDC):	66.0
Unit Capacity Factor (%) (Using Design MWE):	64.8

Outages

Total No.	23	
Forced	17	
Scheduled	6	
Total:	2,001 Hours,	22.8%
Forced	891 Hours,	10.2%
Scheduled	1,110 Hours,	12.6%

Cause: Equipment Failure	12
Maintenance or Testing	6
Administrative	1
Operational Error	3
Other	1
Method of Shutdown:	
Manual	15
Automatic Scram	8

II. Highlights

A. General:

The plant experienced 23 outages during the year accounting for 2001 hours of outage time. The plant availability factor was 77.2%.

B. Outages:

1. Forced: There were 17 forced outages accounting for 891 hours during the year. The longest forced outage required 342 hours to replace the seals on a reactor coolant pump.

2. Scheduled: There were 6 scheduled outages during the year accounting for 1110 hours. The longest outage accounted for 474 hours to perform maintenance on the reactor coolant pump seals; another outage for the same reason required 309 hours.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	283	S	Continuation of Dec. -74 shutdown due to condenser air leakage.	B	1	Steam & Power (HC)	Heat Exchangers
2)	2/4	107	F	Feedwater leak in check valve.	A	1	Steam & Power (HH)	Valves
3)	3/9	42	F	Excessive reactor coolant leakage.	A	1	Reactor Coolant (CB)	Pipes, Fittings
4)	3/11	1	F	Turbine tripped while shifting auxiliary transformers.	G	3	Steam & Power (HA)	Turbines
5)	4/7	7	F	Trip while aligning demineralizer valves.	G	3	Reactor Coolant (CG)	Demineralizers
6)	4/7	474	S	Maintenance on reactor coolant pump seals.	B	1	Reactor Coolant (CB)	Pumps
7)	4/27	5	F	Unit tripped while operating switch gear.	A	3	Electric Power (EA)	Circuit Closers
8)	4/28	10	F	Shut down to identify RC leakage.	A	1	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	4/30	2	F	Shut down to identify RC leakage.	A	1	Reactor Coolant (CB)	Pumps
10)	5/1	342	F	Reactor coolant pump seal replaced.	A	1	Reactor Coolant (CB)	Pumps
11)	5/25	7	F	Trip due to turbine bypass circuitry.	A	3	Steam & Power (HA)	Instrumentation & Controls
12)	6/13	309	S	Reactor coolant pump seal repair.	B	1	Reactor Coolant (CB)	Pumps
13)	6/27	5	F	Shut down to balance reactor coolant pump.	A	1	Reactor Coolant (CB)	Pumps
14)	7/11	9	S	100% turbine trip test.	B	1	Steam & Power (HA)	Turbines
15)	7/20	6	F	Turbine tripped on momentary loss of DC input power on EHC.	A	1	Electric Power (EC)	Relays
16)	8/21	28	F	Shut down to identify leakage in reactor building.	F	1	Engineered Safety (SA)	Penetrations

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
17)	8/30	8	F	Indicated water leakage into RC pump air cooler.	A	1	Reactor Coolant (CB)	Heat Exchangers
18)	9/3	7	F	Turbine oil system malfunction.	A	3	Steam & Power (HA)	Turbines
19)	9/12	5	F	Turbine oil system malfunction.	A	3	Steam & Power (HA)	Turbines
20)	9/30	281	F	Reactor trip during load transient.	H	3	Electric Power (EA)	NA
21)	10/14	28	F	Reactor trip on flux imbalance.	G	3	Reactor (RB)	Control Rod Drive Mechanisms
22)	11/26	18	S	Control rod repatch.	B	1	Reactor (RB)	Control Rods
23)	12/19	17	S	Repaired feedwat. r valve leaks.	B	1	Steam & Power (HH)	Valves

I. Summary

Description

Location: New Jersey, Ocean, Toms River
Docket No: 50-219
Reactor Type: BWR
Capacity (MWe-Net): 650
Commercial Operation: 12/69
Plant Age: 6.3 Years

Performance

Net Electrical Energy Generated (MWH):	3,145,826
Unit Availability Factor (%):	73.3
Unit Capacity Factor (%) (Using MDC):	64.6
Unit Capacity Factor (%) (Using Design MWE):	61.6

Outages

Total No.	13
Forced	10
Scheduled	3
Total:	2,338 Hours, 26.7%
Forced	944 Hours, 10.8%
Scheduled	1,394 Hours, 15.9%

Cause:	
Equipment Failure	9
Refueling	2
Administrative	1
Operational Error	1

Method of Shutdown:	
Manual	6
Manual Scram	2
Automatic Scram	5

II. Highlights

A. General: A total of 13 outages occurred in 1975. Nine were related to problems with equipment; 2 were for refueling; 1 was for administrative reasons; and 1 was an operational error. Three of the outages were caused by problems with valves; and 2 were caused by problems with the condenser.

B. Outages:

1. Forced: There were 10 forced outages in 1975. Of these, the ones of longest duration were: (1) 133 hours, due to a scram when all 3 feedwater pumps tripped; (2) 174 hours, while waiting issuance of a reloading license; (3) 105 hours, due to a scram caused by a turbine vacuum trip; (4) 155 hours, because of condenser tube leaks; and (5) 161 hours, also because of condenser tube leaks.

2. <u>Scheduled</u>: There were 3 scheduled outages during the report period. The two major ones were for refueling; one for 1246 hours and one for 116 hours which started on the 27th of December.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/4	133	F	Scram caused ', low water level after trip of all 3 feedwater pumps. Procedure changes made.	G	3	Reactor Coolant (CH)	Pumps
2)	3/29	1246	S	Refueling	C	1	Reactor (RC)	Fuel Elements
3)	5/20	174	F	Awaiting issuance of reload license and Technical Specification changes.	F	1	Reactor (RC)	NA
4)	6/13	49	F	The plant was shut down to investigate increasing unidentified leak rate within the drywell. Feedwater check valve leak found & repaired.	A	1	Reactor Coolant (CH)	Valves
5)	7/25	15	F	Unit shut down when blown fuse caused loss of DC power to two pressure regulating valves.	A	3	Electric Power (EC)	Circuit Closers
6)	8/27	105	F	Unit shut down following a reactor scram caused by a turbine vacuum trip.	A	3	Steam & Power (HA)	Turbines
7)	9/1	58	F	Reactor scram caused by a turbine vacuum trip.	A	3	Steam & Power (HA)	Turbines
8)	9/24	155	F	High conductivity from condenser leaks.	A	2	Steam & Power (HC)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	10/1	64	F	The unit was shut down because of high reactor water conductivity.	A	2	Reactor Coolant (CG)	Demineral-izers
10)	10/5	30	F	Reactor scram during turbine stop valve testing.	A	3	Steam & Power (HA)	Valves
11)	11/25	161	F	Unit shut down because of condenser tube leakge.	A	1	Steam & Power (HC)	Heat Exchangers
12)	12/19	32	S	Unit was shut down to investigate an increasing unidentified leak rate in the drywell. Repaired packing leak in isolation condenser valve.	B	1	Reactor Coolant (CE)	Valves
13)	12/27	116	S	Refueling	C	1	Reactor (RC)	Fuel Elements

I. Summary

Description

Location: South Haven, Michigan
Docket No: 50-255
Reactor Type: PWR
Capacity (MWe-net): 821
Commercial Operation: 12/71
Plant Age: 4.0 Years

Performance

Net Electrical Energy Generated (MWH):	2,427,933
Unit Availability Factor (%):	64.5
Unit Capacity Factor (%) (Using MDC):	40.5
Unit Capacity Factor (%) (Using Design MWE):	33.7

Outages

Total No.	12	
Forced	10	
Scheduled	2	
Total:	3,108 Hours,	35.5%
Forced	2,757 Hours,	31.5%
Scheduled	351 Hours,	4.0%

Cause:	Equipment Failure	10
	Maintenance or Testing	1
	Refueling	1
Method of Shutdown:		
	Manual	9
	Manual Scram	1
	Automatic Scram	2

II. Highlights

A. General:

The unit operated at a nominal 80% of power after the outage which began in November 1974 to repair condenser tube leakage. The outage was extended for repairs to the steam generator and finally ended in April.

B. Outages:

1. Forced: There were 10 forced outages during the year requiring 2757 hours; 2205 hours were the continuation of the November 1974 outage for condenser repairs. Another outage required 227 hours to repair control rod drive seal leakage; this was also followed by a 135 hour outage for the same reason.

2. Scheduled: There were two scheduled outages requiring 351 hours. A refueling outage which began December 20 consumed 280 hours during the year.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	2205	F	Continuation of shutdown started in 1974 to repair condenser leakage.	A	1	Steam & Power (HC)	Heat Exchangers
2)	4/6	3	F	To repair leak in F.W. heater valve.	A	1	Steam & Power (HH)	Heat Exchangers
3)	4/22	14	F	E.H. oil line failure on turbine control system.	A	3	Steam & Power (HA)	Pipes, Fittings
4)	6/20	227	F	Repair CRDM seal leak.	A	1	Reactor (RB)	Control Rod Drive Mechanisms
5)	6/30	8	F	Feedwater pump trip.	A	2	Steam & Power (HH)	Pumps
6)	7/25	46	F	To repair control rod drive motor.	A	1	Reactor (RB)	Control Rod Drive Mechanisms
7)	8/12	15	F	To repair CRDM seal leak-off.	A	1	Reactor (RB)	Control Rod Drive Mechanisms
8)	8/17	135	F	To repair control rod drive mechanism (CRDM).	A	1	Reactor (RB)	Control Rod Drive Mechanisms
9)	8/30	9	F	To repair CRDM.	A	1	Reactor (RB)	Control Rod Drive Mechanisms

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	9/6	95	F	To repair CRDM.	A	1	Reactor (RB)	Control Rod Drive Mechanisms
11)	10/28	71	S	To repair main electrical generator hydrogen coolers.	B	1	Steam & Power (HA)	Heat Exchangers
12)	12/20	280	S	While shutting down for scheduled refueling, there was a reactor protection system flow trip.	C	3	Reactor (RC)	Fuel Elements

PEACH BOTTOM 2

I. Summary

Description

Location: Peach Bottom, Pennsylvania
Docket No: 50-277
Reactor Type: BWR
Capacity (MWe-net): 1065
Commercial Operation: 7/5/74
Plant Age: 1.9 Years

Performance

Net Electrical Energy Generated (MWH):	5,082,479
Unit Availability Factor (%):	75.8
Unit Capacity Factor (%) (Using MDC):	55.2
Unit Capacity Factor (%) (Using Design MWE):	54.5

Outages

Total No.	15	
Forced	10	
Scheduled	5	
Total:	2,123	Hours, 24.2%
Forced	905	Hours, 10.3%
Scheduled	1,218	Hours, 13.9%

Cause:

Equipment Failure	6
Maintenance or Testing	3
Regulatory Restriction	2
Operational Error	3
Other	1

Method of Shutdown:

Manual	7
Automatic Scram	8

II. Highlights

A. General:

A total of 15 outages occurred in 1975. Six were the result of equipment failure; 3 were for maintenance and/or tests; 2 were for regulatory restrictions; 3 were the result of operational error; and 1 was due to lightning. Five of the outages were related to problems with instrumentation; 2 were related to problems with piping; 4 were related to problems with valves; 3 were related to problems with electrical systems or components; and 1 was related to a problem with the feedwater heater.

B. Outages:

1. Forced: There were 10 forced outages in 1975. Of these, the ones of longest duration were: (1) 522 hours, for leak rate testing of primary containment penetration valves; (2) 128 hours, due to turbine trip resulting from lightning striking a transformer; and (3) 84 hours because of excessive leakage in the feedwater heater.

2. Scheduled: There were 5 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 220 hours, for compliance with DPO Bulletin No. 74-10A (examination of recirc piping); (2) 230 hours, for I.E. Bulletin No. 75-01 (ultrasonic testing of welds in ECCS piping); and (3) 710 hours, for correction of LPRM vibration problems.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/14	220	S	Compliance with DRO Bulletin No. 74-10A, recirculation piping examination, plus condenser baffle work and other maintenance.	B	1	Reactor Coolant (CB)	Pipes, Fittings
2)	2/11	230	S	Unit was shut down in accordance with IE Bulletin No. 75-01 to perform ultrasonic testing on various welds and piping of the ECCS system.	D	1	Engineered Safety (SF)	Pipes, Fittings
3)	3/18	15	F	Reactor scram caused by jarring one of the reactor high pressure scram switches during a period when a similar switch was being surveillance tested.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
4)	4/28	12	F	Reactor scram was caused by a false APRM flow bias signal. The false signal was momentarily created when a recirculation pump flow transmitter was being returned to service.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
5)	5/17	522	F	Shut down to perform local leak rate testing of certain Primary Containment Penetration valves in accordance with Tech. Specs. Maintenance to repair valves not	A	1	Engineered Safety (SA)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
6)	6/10	20	F	Turbine tripped due to an EHC ground.	A	3	Steam & Power (HA)	Electrical Conductors
7)	6/12	67	F	Reactor shut down due to a leaking check valve in the main steam line.	A	1	Reactor Coolant (CC)	Valves
8)	6/14	128	F	Turbine tripped as a result of lightning hitting a transformer and a resulting faulty breaker.	H	3	Electric Power (ED)	Transformers
9)	8/5	24	F	Operator error caused electrical switching transient resulting in loss of plant protection system power supply.	G	3	Electric Power (ED)	Circuit Closers
10)	8/15	22	S	Unit removed from service to install acoustic sensors to the LPRMs underneath the reactor vessel.	D	3	Instrumentation & Controls (IA)	Instrumentation & Controls
11)	9/5	28	S	Replaced reactor water cleanup inlet isolation valve.	B	1	Reactor Coolant (CG)	Valves
12)	10/31	718	S	Unit removed from service to accommodate reactor modifications for correction of LPRM vibration.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
13)	12/11	84	F	Unit had to be manually shut down due to excessive leakage in three feedwater heaters.	A	1	Reactor Coolant (CH)	Heat Exchangers
14)	12/26	25	F	Reactor scrammed because of low oil pressure on the turbine EHC system.	A	3	Steam & Power (HA)	Instrumentation & Controls
15)	12/29	8	F	Reactor scram from APRM high flux during testing of the main steam line isolation valves.	G	3	Instrumentation & Controls (IA)	Valves

I. Summary

Description	Performance		Outages		
Location: York, Pennsylvania	Net Electrical Energy Generated (MWH):	5,282,316	Total No.	13	
Docket No: 50-278			Forced	12	
Reactor Type: BWR	Unit Availability Factor (%):	86.0	Scheduled	1	
Capacity (MWe-net): 1065			Total:	1,228	Hours, 14.0%
Commercial Operation: 12/23/74	Unit Capacity Factor(%) (Using MDC):	58.3	Forced	831	Hours, 9.5%
Plant Age: 1.3 Years			Scheduled	397	Hours, 4.5%
	Unit Capacity Factor (%) (Using Design MWE):	56.7			

Cause:

Equipment Failure	11
Maintenance or Testing	2
Regulatory Restriction	1
Operational Error	1

Method of Shutdown:

Manual	6
Automatic Scram	6

II. Highlights

A. General:

A total of 13 outages occurred in 1975. Eleven were the result of equipment failure; 1 was for maintenance and/or tests; and 1 was the result of an operational error. Three of the outages were caused by problems related to pumps; 5 were related to problems with valves; 1 was related to problems with the turbine; 2 were related to problems with heat exchangers; 1 was related to a problem with the generator; and 1 was related to a problem with electrical conductors.

B. Outages:

1. Forced: There were 12 forced outages in 1975. Of these, the ones of longest duration were: (1) 334 hours, for repairs of a recirc pump seal leak; (2) 164 hours, for repairs of a RHR heat exchanger leak; and (3) 96 hours, for repair of a recirc pump seal leak.

2. Scheduled: There were 3 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 170 hours, for condenser maintenance and for compliance with RO Bulletin No. 74-10A (examination of recirc. pump); and (2) 145 hours, for repair of a recirc pump seal.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1a)	1/3	24	F	Steam leak shorted wires and indicated trouble with the thrust bearing wear detector which tripped the turbine.	A	3	Steam & Power (HA)	Turbines
1b)	1/3	170	S	Maintenance work on condenser plus compliance with DRO Bulletin No. 74-10A, recirc. piping examination.	B	NA	Steam & Power (HC)	Heat Exchangers
2a)	1/17	334	F	Repair of recirculation pump seal.	A	1	Reactor Coolant (CB)	Pumps
2b)	1/17	82	S	Compliance with IE Bulletin No. 75-01, examination of core spray and other ECCS.	D	NA	Engineered Safety (SF)	Pipes, Fittings
3)	2/4	96	F	Recirc pump seal failed during startup; replaced seal.	A	1	Reactor Coolant (CB)	Pumps
4)	2/17	164	F	Shutdown operation disclosed leak in RHR heat exchanger.	A	NA	Reactor Coolant (CF)	Heat Exchangers
5)	5/18	56	F	Unit shut down to repair packing leak in the recirculation pump loop equalizer valve.	A	1	Reactor Coolant (CB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
6)	8/5	21	F	Inability to recover main condenser vacuum after off-gas system operational transient caused by the overloading of a 4 KV electrical bus.	A	3	Electric Power (EB)	Electrical Conductors
7)	8/9	49	F	Operator valving error caused turbine trip on high level from moisture separator drain tank.	G	3	Steam & Power (HA)	Valves
8)	8/10	7	F	A dump valve control failure caused a turbine trip on moisture separator drain tank high level.	A	3	Steam & Power (HA)	Valve Operators
9)	9/3	15	F	Scram due to leaky valve on an RPS instrument.	A	3	Instrumentation & Controls (IA)	Valves
10)	9/17	145	S	Recirculation pump seal leak.	B	1	Reactor Coolant (CB)	Pumps
11)	10/28	7	F	Shut down to correct a ground on the brush rigging of the exciter.	A	1	Steam & Power (HA)	Generators
12)	12/25	24	F	Reactor shut down because of packing leaks in the drywell.	A	1	Engineered Safety (SA)	Valves
13)	12/30	34	F	Scram as a result of condenser vacuum leaks.	A	3	Steam & Power (HC)	Heat Exchangers

I. Summary

Description

Location: Plymouth, Massachusetts
Docket No: 50-293
Reactor Type: BWR
Capacity (MWe-net): 670
Commercial Operation: 12/72
Plant Age: 3.5 Years

Performance

Net electrical energy Generated (MWH):	2,587,248
Unit Availability Factor (%):	71.3
Unit Capacity Factor (%) (Using MDC):	44.1
Unit Capacity Factor (%) (Using Design MWE):	42.9

Outages

Total No.	15	
Forced	13	
Scheduled	2	
Total:	2,517 Hours,	28.7%
Forced	1,900 Hours,	21.7%
Scheduled	617 Hours,	7.0%

Cause:	
Equipment Failure	9
Maintenance or Testing	2
Regulatory Restriction	1
Operational Error	2
Other	1

Method of Shutdown:	
Manual	6
Manual Scram	3
Automatic Scram	6

II. Highlights

A. General:

A total of 15 outages occurred in 1975. Nine were the result of equipment failures; 2 were for maintenance and/or tests; 1 was for a regulatory restriction; and 2 were the result of operational errors. Six of the outages were related to problems with valves; 2 were related to problems with pumps; 2 were related to problems with instrumentation; 2 were related to problems
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with the main condenser; 1 was related to problems with electrical systems; and 2 were related to a problem with piping.

B. Outages:

1. Forced: There were 13 forced outages in 1975. Of these, the ones of longest duration were: (1) 324 hours, for the replacement of a seal on a recirc pump; (2) 294 hours, for the examinations of welds, in compliance with Bulletin No. 75-01; (3) 96 hours, for repairs of leaks in

the drywell; and [illegible] 751 hours for repair of the relief valve downcomer and other maintenance.

2. <u>Scheduled</u>: There were 2 scheduled outages during the report period. One was for 128 hours, for repair of leaks in the main condenser, and the other was for 153 hours, for repair of a feedwater regulating valve. The outage on September 13 was allocated 336 hours scheduled because work which took place during the outage had previously been scheduled.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/12	324	F	Replaced a seal on recirculation pump and performed scheduled inspection of condenser.	A	2	Reactor Coolant (CB)	Pumps
2)	1/30	294	F	Performed examinations of specified welds	D	1	Engineered Safety (SF)	Pipes, Fittings
3)	2/22	9	F	Unplanned reactor scram during performance of APRM functional surveillance test.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
4)	4/16	128	S	Removed unit from service to repair miscellaneous steam leaks at valves in condenser compartment.	B	1	Steam & Power (HC)	Valves
5)	4/23	168	F	Replaced two main steam relief valves.	A	1	Reactor Coolant (CC)	Valves
6)	5/3	8	F	Unplanned reactor scram due to false indication of high water level while performing routine maintenance on the feedwater control system.	A	3	Reactor Coolant (CH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	5/22	96	F	Repaired miscellaneous valve leaks in the drywell.	A	2	Engineered Safety (SA)	Valves
8)	6/30	153	S	Repaired a steam leak on a feedwater regulating valve.	B	1	Reactor Coolant (CH)	Valves
9)	7/18	46	F	Repair of a leaking reactor water clean-up pump inside the drywell.	A	1	Reactor Coolant (CG)	Pumps
10)	7/20	49	F	Removed unit from service to replace a main steam relief/safety valve.	A	2	Reactor Coolant (CC)	Valves
11)	8/5	16	F	Scram during turbine condenser low vacuum functional surveillance test.	G	3	Steam & Power (HC)	Heat Exchangers
12)	8/9	29	F	Repair of two leaking valves in the AOG system and inspect main condenser for tube leaks.	A	1	Radioactive Waste (MB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
13)	8/18	48	F	Condenser low vacuum scram during condenser backwash activities due to excessive debris restricting the cooling water flow.	H	3	Steam & Power (HC)	Heat Exchangers
14)	9/10	62	F	Unplanned reactor scram occurred following instrument bus power transfer.	A	3	Electric Power (ED)	Electrical Conductors
15a)	9/13	751	F	An unplanned reactor scram occurred upon loss of normal 345 KV power to the station. The outage was extended to include previously scheduled work. See 15b. Inspected relief valve downcomer & repaired. Repaired circulating water pumps. Replaced relief valve.	A	3	Reactor Coolant (CB)	Pipes, Fittings
15b)	9/13	336	S	Plugged condenser tube leaks and other scheduled maintenance.	B	NA	Steam & Power (HC)	Heat Exchangers

POINT BEACH I

I. Summary

Description

Location: Two Creeks, Wisconsin
Docket No: 50—266
Reactor Type: PWR
Capacity (MWe-net): 497
Commercial Operation: 12/21/70
Plant Age: 5.2 Years

Performance

Net Electrical Energy Generated (MWH):	2,921,849
Unit Availability Factor (%):	71.9
Unit Capacity Factor (%) (Using MDC):	69.3
Unit Capacity Factor (%) (Using Design MWE):	67.6

Outages

Total No.	5		
Forced	2		
Scheduled	3		
Total:	2,461	Hours,	28.1%
Forced	1,262	Hours,	14.4%
Scheduled	1,199	Hours,	13.7%

Cause:

Equipment Failure	2
Maintenance or Testing	1
Refueling	1
Op. Tng. and License Exam	1

Method of Shutdown:

Manual	3
Automatic Scram	2

II. Highlights

A. General:

A total of 5 outages occurred in 1975. Two were the results of equipment failures; 1 was for maintenance; 1 was for refueling; and 1 was for operator licensing exams. Two of the outages were related to problems with the steam generator and 1 was related to problems with the turbine.

B. Outages:

1. Forced: There were 2 forced outages in 1975. One was for 910 hours, because of a steam generator tube failure and control rod problems; and one was for 352 hours, because of a failure of an electrical breaker resulting in damage to the turbine bearings.

2. **Scheduled: There were 3 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 430 hours for steam generator sludge lancing; and (2) 744 hours for refueling.**

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/15	430	S	Steam generator sludge lancing outage. Extended due to reactor coolant pump seal replacement.	B	1	Steam & Power (HB)	Heat Exchangers
2)	2/27	910	F	Steam generator tube failure and CRD problems.	A	1	Steam & Power (HB)	Heat Exchangers
3)	6/28	25	S	Operators' licensing exams plus maintenance.	E	1	Reactor (RB)	Control Rod Drive Mechanisms
4)	11/16	352	F	Reactor and turbine trip caused by loss of non-safeguards bus due to an electrical breaker failure; turbine bearing damage occurred.	A	3	Steam & Power (HA)	Turbines
5)	11/16	744	S	Refueling	C	3	Reactor (RC)	Fuel Elements

POINT BEACH 2

I. Summary

Description

Location: Two Creeks, Wisconsin
Docket No: 50–301
Reactor Type: PWR
Capacity (MWe-net): 497
Commercial Operation: 4/20/73
Plant Age: 3.4 Years

Performance

Net Electrical Energy Generated (MWH):	3,741,304
Unit Availability Factor (%):	93.9
Unit Capacity Factor (%) (Using MDC):	87.9
Unit Capacity Factor (%) (Using Design MWE):	85.8

Outages

Total No.	10	
Forced	2	
Scheduled	8	
Total:	535	Hours, 6.1%
Forced	209	Hours, 2.4%
Scheduled	326	Hours, 3.7%

Cause:

Equipment Failure	1
Maintenance or Testing	8
Operational Error	1

Method of Shutdown:

Manual	9
Automatic Scram	1

II. Highlights

A. General:

A total of 10 outages occurred in 1975. One was the result of equipment failure; 8 were for maintenance; and 1 was the result of an operational error. Four of the outages were related to problems with steam generators or condensers; 3 were related to problems with the turbines; 2 were related to problems with valves; and 1 was related to a problem with a pump.

B. Outages:

1. Forced: There were 2 forced outages in 1975. Of these, the one of longest duration was for 203 hours, due to steam generator tube leakage.

2. **Scheduled: There were 8 scheduled outages during the report period. Of these, the one of longest duration was: (1) 76 hours for repair of a pressurizer spray valve and turbine stop valve.**

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/11	6	F	Inadvertent trip on low condenser vacuum from improper lineup of air ejectors to circulating water discharge.	G	3	Steam & Power (HC)	Heat Exchangers
2)	2/20	76	S	Repair pressurizer spray valve and turbine stop valve leaks.	B	1	Reactor Coolant (CB)	Valves
3)	4/18	50	S	Secondary water chemistry conditioning and moisture separator reheater tube plugging.	B	1	Steam & Power (HB)	Heat Exchangers
4)	5/10	29	S	Repair high pressure turbine casing leak. Outage extended approximately six hours due to low system load requirement.	B	1	Steam & Power (HA)	Turbines
5)	5/30	45	S	Repair high pressure turbine causing leak and moisture separator reheater tube plugging.	B	1	Steam & Power (HA)	Turbines
6)	6/21	38	S	Repair turbine control valve leak. Briefly extended for turbine vibration test.	B	1	Steam & Power (HA)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	7/26	36	S	Repair reactor coolant pump motor oil leak.	B	1	Reactor Coolant (CB)	Pumps
8)	8/11	203	F	Steam generator tube leakage. Outage extended ~40 hrs. for reactor coolant pump motor oil leak repair. Extension also for RTD bypass loop valve repair.	A	1	Steam & Power (HB)	Heat Exchangers
9)	10/18	25	S	Repair steam generator secondary side manhole gasket steam leakage.	B	1	Steam & Power (HB)	Heat Exchangers
10)	11/1	27	S	Repair high pressure turbine steam leak at cylinder heating steam inlet.	B	1	Steam & Power (HA)	Turbines

I. Summary

Description

Location: Goodhue, Minnesota
Docket No: 50-282
Reactor Type: PWR
Capacity (MWe-Net): 530
Commercial Operation: 12/5/73
Plant Age: 2.1 Years

Performance

Net Electrical Energy Generated (MWH):	3,694,168
Unit Availability Factor (%):	86.3
Unit Capacity Factor (%) (Using MDC):	81.1
Unit Capacity Factor (%) (Using Design MWE):	80.0

Outages

Total No.	20	
Forced	12	
Scheduled	8	
Total:	1,202 Hours,	13.7%
Forced	391 Hours,	4.5%
Scheduled	811 Hours,	9.2%

Cause:	Equipment Failure	9
	Maintenance or Testing	8
	Op. Tng. & License Exam	1
	Operational Error	3
Method of Shutdown:		
	Manual	5
	Manual Scram	2
	Automatic Scram	8

II. Highlights

A. General: A total of 20 outages occurred in 1975. Nine were the result of equipment failure; 7 were for maintenance and/or testing; 1 was for an operator licensing exam; and 3 were the result of operational errors. Five of the outages were related to problems with the condenser; 4 were related to problems with valves; 2 were related to problems with instrumentation; 3 were related to problems with piping; 2 were related to problems with the turbine; 1 was related to a problem with the generator; and 1 was related to a problem with an electrical power supply.

B. Outages:

1. Forced: There were 12 forced outages in 1975. Of these, the ones of longest duration were: (1) 266 hours, for repairs of leaks in a reactor coolant pump seal injection line and (2) 69 hours, due to a failure of a feedwater pump.

2. Scheduled: There were 8 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 461 hours, for an inspection of the steam generator tubes; and (2) 199 hours, to perform a refueling surveillance check and a containment penetrations check.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1a)	1/2	8	F	Solenoid valve failed which caused feedwater regulating valve to close.	A	3	Steam & Power (HH)	Valve Operators
1b)	1/2	70	S	Remained shutdown for various maintenance items.	B	—	Steam & Power (HH)	Pipes, Fittings
2)	1/8	6	F	Shut down to plug condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
3)	1/21	6	F	Shut down to look for condenser tube leak.	A	1	Steam & Power (HC)	Heat Exchangers
4)	1/21	8	F	Shut down to look for condenser tube leak.	A	1	Steam & Power (HA)	Heat Exchangers
5)	1/22	4	F	Tripped when generator accidentally locked out.	G	3	Steam & Power (HA)	Generators
6)	2/11	9	S	Shut down to fix miscellaneous leaks.	B	—	Auxiliary Water (WB)	Pipes, Fittings
7)	2/22	24	S	Licensing exams and maintenance.	E	2	Reactor (RB)	Control Rod Drive Mechanisms

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	3/8	3	F	Shut down to plug condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
9)	3/27	3	F	Operator accidentally closed one MSIV.	G	3	Steam & Power (HB)	Valves
10)	4/24	461	S	Shut down to inspect steam generator tubes.	B	2	Steam & Power (HI)	Heat Exchangers
11)	6/12	3	F	Shut down to inspect leaking manual valve in containment.	A	NA	Engineered Safety (SA)	Valves
12)	6/18	10	S	Shut down to repair leaking manual valve in containment.	B	NA	Engineered Safety (SA)	Valves
13)	6/26	266	F	Shut down to repair leak in RCP seal injection line.	A	3	Reactor Coolant (CB)	Pipes, Fittings
14)	8/1	10	F	Trip when loss of power supply to one protection channel caused a FW regulating valve to close.	A	3	Electric Power (ED)	Electrical Conductors
15)	8/9	199	S	Shut down to perform refueling surveillance and containment penetrations testing.	B	1	Engineered Safety (SA)	Penetrations

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
16)	10/3	12	S	Shut down for turbine overspeed test; unplanned reactor trip aborted the test.	B	3	Steam & Power (HA)	Turbines
17)	10/13	69	F	Reactor trip caused by failure of feedwater pump.	A	3	Steam & Power (HH)	Pumps
18)	10/30	5	F	Reactor trip caused by technician error during NIS calibration.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
19)	11/23	15	S	Shut down to perform turbine overspeed test and maintenance.	B	NA	Steam & Power (HA)	Turbines
20)	12/7	11	S	Shut down to replace excore neutron detector.	B	NA	Instrumentation & Controls (IA)	Instrumentation & Controls

I. Summary

Description

Location: Goodhue, Minnesota
Docket No: 50-306
Reactor Type: PWR
Capacity (MWe-net): 530
Commercial Operation: 12/21/74
Plant Age: 1.0 Years

Performance

Net Electrical Energy Generated (MWH):	3,176,256
Unit Availability Factor (%):	80.3
Unit Capacity Factor (%) (Using MDC):	69.7
Unit Capacity Factor (%) (Using Design MWE):	68.4

Outages

Total No.	27		
Forced	20		
Scheduled	7		
Total:	1,726	Hours,	19.7%
Forced	1,195	Hours	13.6%
Scheduled	531	Hours	6.1%

Cause:
- Equipment Failure 18
- Maintenance or Testing 7
- Operational Error 2

Method of Shutdown:
- Manual 4
- Manual Scram 5
- Automatic Scram 18

II. Highlights

A. General:

A total of 27 outages occurred in 1975. Eighteen were related to problems with equipment failure; 7 were for maintenance and/or tests; and 2 were caused by operational errors; 6 of the outages were related to problems with the turbine; 3 were related to the generator; 3 were related to problems with control rod drives; 4 were related to problems with the condenser; 5 were related to problems with valves; and 2 were related to problems with pumps.

B. Outages:

1. Forced: There were 20 forced outages in 1975. Of these, the ones of longest duration were: (1) 176 hours, because of generator ground; (2) 145 hours, for repairs of the EHC system; (3) 217 hours, because of condenser problems; (4) 158 hours, because of reactor coolant pump seal leaks; and (5) 364 hours, because of turbine blade failure.

2. Scheduled: There were 7 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 373 hours, for inspection of turbine bearings, and (2) 76 hours, for condenser modifications.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/21	10	F	Negative flux rate trip caused by dropped rod. Fuse in gripper coil circuit opened.	A	3	Reactor (RB)	Control Rod Drive Mechanisms
2)	1/30	67	S	Planned trip from 50% power.	B	3	Steam & Power (HA)	Turbines
3)	2/15	4	S	Turbine trip test from 100% power.	B	3	Steam & Power (HA)	Turbines
4)	2/15	4	S	Loss of off-site power test.	B	3	Electric Power (EA)	Circuit Closers
5)	2/15	3	S	Loss of RC flow test.	B	3	Reactor Coolant (CB)	Pumps
6)	2/20	4	S	Generator trip test from 100% power	B	3	Steam & Power (HA)	Generators
7)	3/5	6	F	Trip caused by feed-water regulating valve malfunction.	A	3	Steam & Power (HH)	Valves
8)	3/8	176	F	Trip caused by generator ground. Hydrogen cooler found leaking. Also repaired MSIV shaft seal leakage.	A	3	Steam & Power (HA)	Generators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	3/16	10	F	Negative flux rate trip when one rod dropped.	A	3	Reactor (RB)	Control Rod Drive Mechanisms
10)	3/18	6	F	Negative flux rate trip when one rod dropped; error in maintenance procedures.	G	3	Reactor (RB)	Control Rod Drive Mechanisms
11)	3/21	76	S	Shutdown for condenser modification and other maintenance.	B	1	Steam & Power (HC)	Heat Exchangers
12)	4/3	5	F	Trip caused by spurious relay action during surveillance test.	A	3	Instrumentation & Controls (IA)	Relays
13)	4/5	7	F	One MSIV closed.	G	3	Steam & Power (HB)	Valves
14)	4/29	18	F	Unexplained MSIV closure.	A	3	Steam & Power (HB)	Valves
15)	5/4	16	F	Unit taken off line due to high turbine rotor stress.	A	1	Steam & Power (HA)	Turbines

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
16)	6/1	373	S	Shutdown to inspect turbine bearing.	B	3	Steam & Power (HA)	Turbines
17)	6/22	145	F	Shutdown to repair fluid leak in EH control system. Damage to turbine bearing on restart.	A	1	Steam & Power (HA)	Pipes, Fittings
18)	6/28	8	F	Unable to hold condenser vacuum on restart.	A	2	Steam & Power (HC)	Heat Exchangers
19)	6/28	217	F	Unable to hold condenser vacuum on restart. No. 6 bearing again damaged. Shutdown to repair turbine bearing.	A	2	Steam & Power (HA)	Turbines
20)	8/18	6	F	Trip when solenoid valve which controls air to FW regulating valve failed.	A	3	Steam & Power (HH)	Valves
21)	8/22	5	F	Trip when FW regulating valve malfunctioned.	A	3	Steam & Power (HH)	Valves
22)	9/13	23	F	Unit trip due to FWP trip when tubing to pressure switch broke.	A	3	Steam & Power (HH)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
23)	9/14	8	F	Manual trip due to problems maintaining condenser vacuum.	A	2	Steam & Power (HC)	Heat Exchangers
24)	9/15	6	F	Manual trip due to problems maintaining condenser vacuum.	A	2	Steam & Power (HC)	Heat Exchangers
25)	9/15	1	F	Trip from high steam generator level.	A	3	Steam & Power (HB)	Heat Exchangers
26)	10/15	158	F	Failure of No. 22 reactor coolant pump seals.	A	1	Reactor Coolant (CB)	Pumps
27)	12/16	364	F	Turbine blading failure, steam generator modifications, eddy current testing, etc.	A	2	Steam & Power (HA)	Turbines

QUAD CITIES 1

I. Summary

Description

Location: Cordova, Illinois
Docket No: 50-254
Reactor Type: BWR
Capacity (MWe-Net): 809
Commercial Operation: 2/18/73
Plant Age: 3.7 Years

Performance

Net Electrical Energy Generated (MWH):	4,270,882
Unit Availability Factor (%):	85.1
Unit Capacity Factor (%) (Using MDC):	216.7
Unit Capacity Factor (%) (Using Design MWE):	62.3

Outages

Total No.	9
Forced	7
Scheduled	2
Total:	1,309 Hours, 14.9%
Forced	1,067 Hours, 12.2%
Scheduled	242 Hours, 2.7 %

Cause:

Equipment Failure	6
Maintenance or Testing	3
Operational Error	1
Other	1

Method of Shutdown:

Manual	3
Automatic Scram	6

II. Highlights

A. General:

A total of 9 outages occurred in 1975. Five were the result of equipment failure; 2 were for maintenance; 1 was the result of an operational error; and 1 was due to a load rejection. Two of the outages were related to problems with piping; 3 were related to problems with instrumentation; 2 were related to problems with the electrical system; 1 was related to problems with valves; and 1 was related to problems with the condenser.

B. Outages:

1. Forced: There were 7 forced outages in 1975. Of these, the ones of longest duration were: (1) 948 hours, for repair of cracks found in the recirculation line; and (2) 45 hours, because of problems with an MSIV-to-condenser bypass line relief valve.

2. Scheduled: There were 2 scheduled outages during the report period: (1) 80 hours, for a control rod pattern change and CRD replacement; and (2) 65 hours, for condenser maintenance. An extension of one outage required 97 hours for the inspection of piping.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1a)	1/9	7	F	Reactor scram due to the loss of the Essential Service Bus.	A	3	Electric Power (ED)	Electrical Conductors
1b)	1/9	97	S	Inspection of the recirculation pipe for cracks.	B	NA	Reactor Coolant (CB)	Pipes, Fittings
1c)	1/13	948	F	After recirc pipe cracks were discovered, the outage was reclassified as a forced outage.	A	NA	Reactor Coolant (CB)	Pipes, Fittings
2)	2/24	24	F	Repair a heater tracer test tap leak on feed water pump discharge line.	A	1	Reactor Coolant (CH)	Pipes, Fittings
3)	2/27	7	F	Reactor scram on low water lev[illegible]hen a feedwater [illegible]lating valve close[illegible] when the milli[illegible]t t[illegible] cur[illegible]ent conv[illegible]rter o[illegible] [illegible]actor feed pump was removed by mistake.	C	3	Reactor Coolant (CH)	Instrumentation & Controls
4)	5/1	80	S	Control rod pattern changed. Minor maintenance outage. Three control rod drives were replaced.	B	1	Reactor (RB)	Control Rod Drive Mechanisms

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
5)	5/8	10	F	Reactor scram due to erroneous calibration signal. Turbine trip at greater than 40% power	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
6)	5/13	8	F	Reactor scram due to erroneous test signal resulting in stop valve closure and turbine trip at greater than 40% power.	A	3	Steam & Power (HA)	Instrumentation & Controls
7)	7/24	65	S	Reactor shutdown for condenser maintenance.	B	1	Steam & Power (HC)	Heat Exchangers
8)	8/19	45	F	Scram from low condenser vacuum due to actuation of a MSIV-to-condenser bypass line relief valve.	A	3	Reactor Coolant (CD)	Valves
9)	8/26	18	F	Forced scram due to load rejection on system.	H	3	Electric Power (EA)	Generators

QUAD CITIES 2

I. Summary

Description

Location: Cordova, Illinois
Docket No: 50-[illegible]
Reactor Type: BWR
Capacity (MWe-net): 809
Commercial Operation: 3/10/73
Plant Age: 3.6 Years

Performance

Net Electrical Energy Generated (MWH):	2,475,331
Unit Availability Factor (%):	51.7
Unit Capacity Factor (%) (Using MDC):	125.6
Unit Capacity Factor (%) (Using Design MWE):	36.2

Outages

Total No.	13		
Forced	9		
Scheduled	4		
Total:	4,230	Hours,	48.3%
Forced	539	Hours,	6.2%
Scheduled	3,691	Hours,	42.1%

Cause:

Equipment Failure	8
Maintenance or Testing	2
Refueling	1
Administrative	1
Operational Error	1

Method of Shutdown:

Manual	6
Manual Scram	2
Automatic Scram	5

II. Highlights

A. General

There were [illegible] outages in 1975. Eight were the result of equipment failure; 2 were for maintenance; 1 was for refueling; one was administrative; and one was an operational error. Two of the outages were related to problems with pumps; 3 were related to problems with piping; one was related to problems with the turbine; and the major outage was for refueling.

B. Outages:

1. Forced: There were 9 forced outages in 1975. Of these, the ones of longest duration were: (1) 21[illegible] hours, for repair of a feedwater line; (2) 137 hours, because of a malfunctioning feedwater regulating valve; and (3) 87 hours, for repairs of a feedwater header flush line.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 2762 hours, for refueling (which had begun at the latter part of 1974); (2) 833 hours, for core maintenance; and (3) 65 hours, for an investigation into the high activity level in the drywell.

QUAD CITIES 2

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	2762	S	Continuation of refueling outage.	C	1	Reactor (RC)	Fuel Elements
2)	4/26	2	F	Turbine trip on high moisture separator level.	A	3	Steam & Power (HA)	Turbines
3)	4/26	31	S	Unit brought down for control rod sequence change.	F	1	Reactor (RB)	Control Rods
4)	4/28	47	F	Unit brought down because of high primary system conductivity due to dislodgement of a condenser tube plug.	A	1	Steam & Power (HC)	Heat Exchangers
5)	5/5	8	F	Accidental trip of reactor feed pump caused low vessel level scram.	G	3	Reactor Coolant (CH)	Pumps
6)	5/20	19	F	Feedwater system repairs.	A	2	Reactor Coolant (CH)	Pumps
7)	7/21	87	F	Unit shut down to repair a leak on a feedwater header flush line.	A	1	Reactor Coolant (CH)	Pipes, Fittings
8)	8/15	65	S	Unit shut down to check high activity in the drywell. CAM sample point improperly positioned.	B	1	Radioactive Waste (MB)	Pipes, Fittings

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	8/17	216	F	Unit manually scrammed due to low flow feedwater line break.	A	2	Reactor Coolant (CH)	Pipes, Fittings
10)	8/31	137	F	Scram due to high water level in the vessel when a feedwater regulating valve failed open.	A	2	Reactor Coolant (CH)	Valves
11)	10/3	833	S	Unit shut down for core maintenance. Replaced damaged fuel.	B	1	Reactor (RC)	Fuel Elements
12)	11/15	14	F	Unit scrammed due to high water level trip in moisture separator.	A	3	Steam & Power (HB)	Vessels, Pressures
13)	12/24	9	F	Scram from low reactor vessel water level caused by failure of feedwater regulating valve.	A	3	Reactor Coolant (CH)	Valves

I. Summary

Description

Location: Sacramento, California
Docket No: 50-312
Reactor Type: PWR
Capacity (MWe-Net): 913
Commercial Operation: 4/18/75
Plant Age: 1.2 Years

Performance

Net electrical energy generated (MWH):	1,326,506
Unit availability factor (%):	27.5
Unit capacity factor (%) (using MDC):	26.2
Unit capacity factor (%) (using Design MWe):	25.3

*Data covers the period from date of commercial operation (4/18/75) to the end of the year. Total hours for the period is 6193 hours.

Outages

Total No.	8	
Forced	8	
Scheduled	0	
Total*:	4,491	Hours, 72.5%
Forced	4,491	Hours, 72.5%
Scheduled	-	

Cause: Equipment Failure 7
Operational Error 1

Method of Shutdown:
Manual 6
Automatic Scram 2

II. Highlights

A. General:

A total of 8 outages occurred in 1975. Seven were the result of equipment failures; and 1 was the result of an operational error. Four of the outages were related to problems with instrumentation; 3 were related to problems with valves; and 1 was related to problems with the turbine.

B. Outages:

1. Forced: There were 8 forced outages in 1975. The one of longest duration was 4438 hours for turbine reblading.

2. Scheduled: There were no scheduled outages during the report period.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	4/19	9	F	Tripped generator to repair valve.	A	1	Steam & Power (HB)	Valves
2)	4/22	9	F	Reactor tripped on power imbalance.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
3)	5/31	6	F	R.C.P. — air cooler leakage alarm; found to be caused by condensation.	A	1	Reactor Coolant (CB)	Instrumentation & Controls
4)	6/2	5	F	Modified RCP motor cooler moisture detector.	A	1	Reactor Coolant (CB)	Instrumentation & Controls
5)	6/8	10	F	Modified remaining RCP's motor cooler moisture detectors.	A	1	Reactor Coolant (CB)	Instrumentation & Controls
6)	6/8	1	F	Turbine trip. I&C technician adjusting governor valve made inadvertent trip.	G	3	Steam & Power (HA)	Valves
7)	6/15	13	F	Repair of OTSG. Feedwater line blowdown line valve leakage.	A	1	Steam & Power (HH)	Valves
8)	6/30	4438	F	Excessive vibration detected in turbine bearing. Found thrown blades in No. 2 L.P. turbine. Cracks found in No. 1 L.P. turbine blades. Unit down for turbine reblading.	A	1	Steam & Power (HA)	Turbines

I. Summary

Description

Location: Hartsville, South Carolina
Docket No: 50-261
Reactor Type: PWR
Capacity (MWe-Net): 707
Commercial Operation: 3/71
Plant Age: 5.3 Years

Performance

Net electrical energy generated (MWH):	4,170,774
Unit availability factor (%):	72.7
Unit capacity factor (%) (using MDC):	71.6
Unit capacity factor (%) (using Design MWe):	67.3

Outages

Total No.	18	
Forced	15	
Scheduled	3	
Total:	2,389	Hours, 27.3%
Forced	1,097	Hours, 12.6%
Scheduled	1,292	Hours, 14.7%

Cause:

Equipment Failure	15
Maintenance or Testing	2
Refueling	1

Method of Shutdown:

Manual	7
Manual Scram	2
Automatic Scram	9

II. Highlights

A. General:

A total of 18 outages occurred in 1975. Fifteen were the result of equipment failures; 2 were for maintenance and/or testing; and 1 was for refueling. Three of the outages were related to problems with valves; 2 were related to problems with electrical equipment; 2 were related to problems with heat exchangers; 1 was related to problems with the control rod drives; 3 were related to problems with pumps; and 3 were related to problems with instrumentation.

B. Outages

1. **Forced**: There were 15 forced outages in 1975. The ones of longest duration were: (1) 610 hours, due to failure of a reactor coolant pump; (2) there were 2 outages each for 132 hours, to repair reactor coolant pump seals; and (3) 67 hours, due to control rod drive coil failure.

2. <u>Scheduled</u>: There were 3 scheduled outages in 1975: (1) 308 hours, for steam generator eddy current testing and repair of the condenser; (2) 962 hours, for refueling; and (3) 22 hours for maintenance on the feedwater heaters.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/3	59	F	Repair of condenser & instrument bus failure.	A	1	Steam & Power (HC)	Heat Exchangers
2)	3/8	22	S	Maintenance on feedwater heater.	B	1	Steam & Power (HH)	Heat Exchangers
3)	4/1	3	F	Feedwater valve closed due to moisture collecting in controller box.	A	3	Steam & Power (HH)	Instrumentation & Controls
4)	4/1	6	F	Feedwater regulating valve closed due to an electrical fault on solenoid valve.	A	3	Steam & Power (HH)	Valve Operators
5)	4/5	4	F	Feedwater regulating valve closed due to failure of controller.	A	3	Steam & Power (HH)	Instrumentation & Controls
6)	4/12	308	S	Steam generator eddy current testing condenser repairs, NRC startup tests.	B	3	Steam & Power (HB)	Heat Exchangers
7)	4/24	132	F	Failure of reactor coolant pump seals; NRC startup tests.	A	1	Reactor Coolant (CB)	Pumps
8)	5/1	610	F	Turbine trip — high steam generator level while reducing load due to reactor coolant pump failure.	A	3	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	5/30	4	F	Loss of instrument air resulted in trip due to feedwater — steam flow mismatch on steam generator.	A	3	Auxiliary Process (PA)	Blowers
10)	6/1	132	F	Repair of reactor coolant pump seal.	A	1	Reactor Coolant (CB)	Pumps
11)	7/11	67	F	Control rod drive coil failure.	A	1	Reactor (RB)	Control Rod Drive Mechanisms
12)	8/27	2	F	Steam flow — feed flow mismatch due to faulty hotwell level switch which tripped condensate pumps. Condensate pump trip caused steam generator pump trip which resulted in low level in generator.	A	3	Steam & Power (HC)	Instrumentation & Controls
13)	9/4	2	F	Turbine tripped due to high steam generator level caused by loss of both heater drain pumps due to malfunction of drain pump discharge valve.	A	3	Steam & Power (HH)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	9/21	23	F	Rod control failure. Defective fuse in power supply to rod control cabinet. Extension of outage caused by failure of source range detector and preamp.	A	2	Reactor (RB)	Circuit Closers
15)	10/18	21	F	Loss of instrument bus due to failure of inverter transformer.	A	2	Electric Power (ED)	Transformers
16)	10/31	962	S	Refueling	C	1	Reactor (RC)	Fuel Elements
17)	12/17	17	F	Low steam generator level with steam flow greater than feedwater flow caused by low hot-wall level switch failure resulting in tripping condensate and feedwater pumps.	A	3	Steam & Power (HC)	Instrumentation & Controls
18)	12/28	15	F	Turbine stop valve would not open after valve test.	A	1	Steam & Power (HA)	Valves

I. Summary

Description

Location: San Clemente, California
Docket No: 50-206
Reactor Type: PWR
Capacity (MWe-net): 450
Commercial Operation: 1/1/68
Plant Age: 8.5 Years

Performance

Net Electrical Energy Generated (MWH):	3,245,108
Unit Availability Factor (%):	87.4
Unit Capacity Factor (%) (Using MDC):	86.2
Unit Capacity Factor (%) (Using Design MWE):	82.4

Outages

Total No.	4		
Forced	2		
Scheduled	2		
Total:	1,100	Hours,	12.6%
Forced	14	Hours,	0.2%
Scheduled	1,086	Hours,	12.4%

Cause:	
Equipment Failure	1
Maintenance or Testing	1
Refueling	1
Other	1

Method of Shutdown:	
Manual	1
Manual Scram	2
Automatic Scram	1

II. Highlights

A. General:

A total of 4 outages occurred in 1975. One was caused by equipment failure; one was for the purpose of maintenance; one was for refueling; a d one was caused by seaweed fouling the intake structure.

B. Outages:

1. Forced: There were 2 forced outages in 1975. The one of longest duration was for 10 hours, caused by seaweed fouling the intake structure.

2. **Scheduled:** There were 2 scheduled outages during the report period: 959 hours for refueling, and 127 hours for repair of a pressurizer relief valve.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/19	4	F	While testing a pressurizer level channel, a second level channel spiked due to a failure of the No. 2 inverter; trip from pressurizer high level.	A	3	Reactor Coolant (CB)	Instrumentation & Controls
2)	3/14	959	S	Refueling and miscellaneous maintenance.	C	2	Reactor (RC)	Fuel Elements
3)	5/21	10	F	Reactor trip from restricted circulating water flow caused by seaweed fouling intake structure.	H	2	Auxiliary Water (WE)	Filters
4)	6/11	127	S	Shut down to repair pressurizer safety valves. Also plugged leaking steam generator tube.	B	1	Reactor Coolant (CB)	Valves

I. Summary

Description

Location: Surry, Virginia
Docket No: 50-280
Reactor Type: PWR
Capacity (MWe-Net): 823
Commercial Operation: 12/22/72
Plant Age: 3.5 Years

Performance

Net electrical energy generated (MWH):	3,916,527
Unit availability factor (%):	62.0
Unit capacity factor (%) (using MDC):	56.7
Unit capacity factor (%) (using Design MWe):	54.3

Outages

Total No.	26		
Forced	23		
Scheduled	3		
Total:	3,334	Hours,	38.0%
Forced	732	Hours,	8.3%
Scheduled	2,602	Hours,	29.7%

Cause:

Equipment Failure	11
Maintenance or Testing	2
Refueling	2
Operational Error	11

Method of Shutdown:

Manual	8
Manual Scram	1
Automatic Scram	17

II. Highlights

A. General:

A total of 26 outages occurred in 1975. Eleven were the result of equipment failures; 2 were for maintenance and/or tests; 2 were for refueling; and eleven were the result of operational errors. Two of the outages were related to problems with the steam generators; 9 were related to problems with instrumentation; 5 were related to problems with valves; 2 were related to problems with electrical equipment; 1 was related to a problem with a diesel generator; 1 was related to problems with the main generator; and 1 was related to a problem with the turbine.

B. Outages:

1. Forced: There were 23 forced outages in 1975. The ones of longest duration were: (1) 209 hours, for repairs of both containment air compressors; (2) 107 hours, for repairs of tube leaks in the steam generator; and (3) 133 hours, due to steam generator tube leaks.

2. Scheduled: There were 3 scheduled outages during the report period. The ones of longest duration were: (1) 776 hours, a continuation of the 1974 refueling; (2) 174 hours, for repairs of a turbine bearing; and (3) 1652 hours, for refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	776	S	Unit was down for refueling (continued from 10-24-74).	C	3	Reactor (RC)	Fuel Elements
2)	2/2	3	F	Error in wiring caused feedwater regulating valve to fail open.	G	3	Steam & Power (HH)	Electrical Conductors
3)	2/2	3	F	Low steam flow/feedwater mismatch due to feedwater control being sensitive during startup.	G	3	Steam & Power (HH)	Instrumentation & Controls
4)	2/2	2	F	Loss of feedwater pump occurred during bus shifting with a breaker in the test position.	G	3	Steam & Power (HH)	Circuit Closers
5)	2/2	3	F	Feedwater control sensitivity contributed to cause of outage.	G	3	Steam & Power (HH)	Instrumentation & Controls
6)	2/3	4	F	Feedwater control sensitivity contributed to cause of outage.	G	3	Steam & Power (HH)	Instrumentation & Controls
7)	2/9	14	F	100% load reject test.	B	3	Steam & Power (HA)	Generators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	2/25	20	F	High steam generator level and steam flow/feed flow mismatch.	G	3	Steam & Power (HH)	Instrumentation & Controls
9)	3/7	174	S	Repair of turbine bearing.	B	1	Steam & Power (HA)	Turbines
10)	4/2	2	F	Faulty relay on loop stop valves caused trip while performing surveillance test.	A	3	Reactor Coolant (CB)	Relays
11)	4/21	75	F	Repaired leaking valves in residual heat removal system.	A	1	Reactor Coolant (CF)	Valves
12)	4/24	1	F	Feedwater control sensitivity during startup. Unit tripped on high level in steam generator.	G	3	Steam & Power (HH)	Instrumentation & Controls
13)	4/24	1	F	Feedwater control sensitivity during startup. Unit tripped on low level — feedwater mismatch.	G	3	Steam & Power (HH)	Instrumentation & Controls
14)	4/25	12	F	Pressurizer spray valve failed open.	A	3	Reactor Coolant (CB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	4/30	38	F	Failure of No. 3 diesel generator.	A	3	Electric Power (EE)	Engines, Internal Combustion
16)	6/24	59	F	Packing failure on valve resulted in excessive primary coolant leakage.	A	1	Reactor Coolant (CB)	Valves
17)	7/24	209	F	Failure of both containment air compressors. During rampdown had auto scram due to spike on feed control instrumentation.	A	3	Auxiliary Process (PA)	Blowers
18)	8/23	32	F	Repaired leaking primary system valve.	A	1	Reactor Coolant (CB)	Valves
19)	9/26	107	F	Repairs to protection RTD and primary to secondary tube leak in steam generator.	A	1	Steam & Power (HB)	Heat Exchangers
20)	10/1	1652	S	Refueling	C	1	Reactor (RC)	Fuel Elements
21)	12/10	133	F	Steam generator leak.	A	1	Steam & Power (HB)	Heat Exchangers
22)	12/16	1	F	Steam flow/feedwater flow mismatch with low steam generator level. Feed control is sensitive during startup.	G	3	Steam & Power (HH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
23)	12/16	1	F	High steam generator level. Feed control is sensitive during startup.	G	3	Steam & Power (HH)	Instrumentation & Controls
24)	12/27	7	F	Turbine governor valves opened. Repaired EHC system.	A	2	Steam & Power (HA)	Instrumentation & Controls
25)	12/28	2	F	Overborated	G	1	Auxiliary Process (PC)	—
26)	12/28	3	F	Main feed regulating valve stuck closed.	A	3	Steam & Power (HH)	Valves

I. Summary

Description

Location: Gravel Neck, Virginia
Docket No: 50-281
Reactor Type: PWR
Capacity (MWe-net): 823
Commercial Operation: 5/1/73
Plant Age: 2.8 Years

Performance

Net Electrical Energy Generated (MWH):	5,053,082
Unit Availability Factor (%):	79.6
Unit Capacity Factor (%) (Using MDC):	73.2
Unit Capacity Factor (%) (Using Design MWE):	70.1

Outages

Total No.	20	
Forced	17	
Scheduled	3	
Total:	1,790 Hours,	20.4%
Forced	458 Hours,	5.2%
Scheduled	1,332 Hours,	15.2%

Cause: Equipment Failure	10
Maintenance or Testing	2
Refueling	1
Operational Error	7
Method of Shutdown:	
Manual	2
Manual Scram	3
Automatic Scram	14

II. Highlights

A. General:

A total of 20 outages occurred in 1975. Ten were related to problems with equipment failure; 2 were for maintenance; 1 was for refueling; and 7 were the result of operational errors. One of the outages was related to problems with the steam generator; 3 were related to problems with instrumentation; 5 were related to problems with valves; and 3 were related to problems with pipes and fittings.

B. Outages:

1. **Forced:** There were 17 forced outages in 1975. The ones of longest duration were: (1) 121 hours, for problems with the air supply system; (2) 114 hours, for leaks in the secondary system; and (3) 80 hours, due to broken blades in the low pressure turbine.

2. Scheduled: There were 3 scheduled outages during the report period. The ones of longest duration were: (1) 1253 hours, for refueling; and (2) 54 hours, due to a malfunction of the EHC system.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	80	F	Continued from 9/6/74. High turbine vibration due to broken blades in low pressure turbine.	A	2	Steam & Power (HA)	Turbines
2)	1/17	8	F	Repair leak in bypass line at main feedwater pump discharge valve.	A	NA	Steam & Power (HH)	Pipes, Fittings
3)	1/18	15	F	Operator error while manually feeding steam generator.	G	3	Steam & Power (HH)	NA
4)	2/2	3	F	Operator error isolating feed to steam generator.	G	3	Steam & Power (HH)	NA
5)	2/2	16	F	Steam generator low level with steam flow/feed flow mismatch.	G	3	Steam & Power (HH)	NA
6)	3/21		S	Reactor trip due to a malfunction in the turbine EHC control system during scheduled shutdown for maintenance on secondary system.	B	3	Steam & Power (HB)	Pipes, Fittings
7)	4/18	25	S	Startup test — trip from 100% power.	B	2	Steam & Power (HA)	NA
8)	4/26	1253	S	Refueling	C	1	Reactor (RC)	Fuel Elements

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
9)	6/15	4	F	Reactor trip caused by high steam flow with low T-avg resulting from unequal pressure between steam generator and header and rapid header isolation valve opening.	G	3	Steam & Power (HB)	Valves
10)	6/16	4	F	Malfunction of EHC caused turbine to roll at a high rate as the governor valves opened spiking 1st stage pressure which resulted in turbine trip and subsequent reactor trip.	A	3	Steam & Power (HA)	Instrumentation & Controls
11)	6/17	2	F	Turbine control valves opened rapidly spiking 1st stage pressure tripping turbine and reactor EHC malfunction.	A	3	Steam & Power (HA)	Instrumentation & Controls
12)	7/6	121	F	Failure in air system which holds main steam trip valves open. Auto scram occurred when trip valve slammed closed.	A	3	Auxiliary Process (PA)	Valves
13)	7/11	4	F	During start-up a steam generator level trip occurred during manual feedwater control.	G	3	Steam & Power (HH)	NA

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	8/14	3	F	Repair rod control system failure.	A	2	Reactor (RB)	Instrumentation & Controls
15)	8/15	11	F	Repair d operator on main steam trip valve.	A	3	Steam & Power (HB)	Valve Operators
16)	9/16	12	F	Attempted to return breaker to service without removing maintenance ground causing unit trip.	G	3	Electric Power (ED)	Circuit Closers
17)	10/9	114	F	Rod position indicated failure due to secondary steam leak.	A	3	Instrumentation & Controls (ID)	Pipes, Fittings
18)	10/13	4	F	Feed reg. valve failed resulting in steam generator level trip.	A	3	Steam & Power (HA)	Valves
19)	10/15	4	F	Main steam trip valve failed shut when operator secured air to valve operator while conducting containment isolation valve leak rate test on other unit.	G	3	Steam & Power (HB)	Valves
20)	10/19	53	F	Steam leaks on steam generator level taps.	A	1	Steam & Power (HB)	Heat Exchangers

THREE MILE ISLAND 1

I. Summary

Description

Location: Dauphin, Pennsylvania
Docket No: 50-289
Reactor Type: PWR
Capacity (MWe-net): 819
Commercial Operation: 9/2/74
Plant Age: 1.5 Years

Performance

Net Electrical Energy Generated (MWH):	5,541,523
Unit Availability Factor (%):	82.2
Unit Capacity Factor (%) (Using MDC):	79.9
Unit Capacity Factor (%) (Using Design MWE):	77.3

Outages

Total No.	16		
Forced	13		
Scheduled	3		
Total:	1,559	Hours,	17.8%
Forced	922	Hours,	10.5%
Scheduled	637	Hours,	7.3%

Cause:
- Equipment Failure 11
- Maintenance or Testing 3
- Operational Error 2

Method of Shutdown:
- Manual 7
- Automatic Scram 5

II. Highlights

A. General:

A total of 16 outages occurred in 1975. Eleven were related to problems with equipment failure, 3 were for maintenance, and 2 were the result of operational error. There were 2 outages associated with each of the following components: electrical conductors, I & C, transformers, pumps, and valves. Three outages involved turbines, and 1 each for relays, control rods, and control rod drives.

B. Outages:

1. Forced: There were 13 forced outages in 1975. Of these, the ones of longest duration were: (1) 214 hours, for electrical conductors; (2) 144 hours, for replacement of a pump motor shaft; (3) 303 hours, for a turbine control valve; and (4) 104 hours, for repair of a make-up valve.

2. Scheduled: There were 3 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 266 hours, to correct problems with a reactor coolant pump; (2) 277 hours, for control rod interchange; and (3) 94 hours, for control rod drive maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/23	40	F	Personnel error led to turbine trip.	G	3	Steam & Power (HA)	Turbines
2)	3/30	14	F	A faulty relay gave an erroneous signal which indicated loss of DC power to the turbines EHC system. The signal tripped the turbine. The reactor tripped from high RC pressure.	A	3	Steam & Power (HA)	Relays
3)	4/5	214	F	Faulty cable connector on reactor vessel head caused control rod to drop into the core and rod withdraw problems.	A	1	Reactor (RB)	Electrical Conductors
4)	5/9	9	F	Turbine trip due to mechanical failure in moisture separator high level switch.	A	3	Steam & Power (HB)	Instrumentation & Controls
5)	5/22	16	F	Fans and cooling pumps on main transformer tripped necessitating removing turbine off line.	A	NA	Electric Power (EB)	Transformers
6)	5/25	144	F	Motor shaft sheared on DR-P1B. Unit shut down for repairs and a previously scheduled control rod interchange.	A	1	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	5/31	277	S	Unit in scheduled control rod interchange.	B	1	Reactor (RB)	Control Rods
8)	6/18	20	F	A brush recorder monitoring the turbine EHC system caused erroneous voltage spikes resulting in rapid load reduction and reactor trip on high pressure.	A	3	Steam & Power (HA)	Instrumentation & Controls
9)	6/21	4	F	Turbine had to be taken off the line due to main transformer problems.	A	NA	Electric Power (EB)	Transformers
10)	6/25	27	F	Reactor trip due to high positive imbalance created when control rod group 7 Phase Bus bar faulted to neutral causing group 7 rods to drop into the core.	A	3	Reactor (RB)	Electrical Conductors
11)	6/26	24	F	When attempting to put the turbine on line after the reactor was returned to service, the turbine tripped due to high eccentricity.	A	NA	Steam & Power (HA)	Turbines
12)	9/26	266	S	Repair RC-P-1A to reduce vibration.	B	1	Reactor Coolant (CB)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
13)	10/16	94	S	Repair CRD stator.	B	1	Reactor (RB)	Control Rod Drives Mechanisms
14)	11/12	303	F	Repair control rod drive stator; duration increased due to repairs to control valve of turbine.	A	1	Steam & Power (HA)	Valves
15)	12/16	104	F	Repair make-up valve.	A	1	Auxiliary Water (WC)	Valves
16)	12/22	3	F	Turbo-gen. trip while testing the deluge system.	G	NA	Other Auxiliary (AB)	Turbines

I. Summary

Description

Location: Florida City, Florida
Docket No: 50—250
Reactor Type: PWR
Capacity (MWe-net): 745
Commercial Operation: 12/14/72
Plant Age: 3.2 Years

Performance

Net Electrical Energy Generated (MWH):	4,374,597
Unit Availability Factor (%):	79.4
Unit Capacity Factor (%) (Using MDC):	75.0
Unit Capacity Factor (%) (Using Design MWE):	72.9

*Data contains discrepancy of 16 hours.

Outages

Total No.	22		
Forced	18		
Scheduled	4		
Total:*	1,793	Hours,	20.5%
Forced	196	Hours,	2.3%
Scheduled	1,597	Hours,	18.2%

Cause:
- Equipment Failure 13
- Maintenance or Testing 3
- Refueling 1
- Operational Error 5

Method of Shutdown:
- Manual 4
- Manual Scram 1
- Automatic Scram 16

II. Highlights

A. General:

A total of 22 outages occurred in 1975. Thirteen were the result of equipment failures; 3 were for maintenance and/or tests; 5 were the result of operational errors; and 1 was for refueling. Five of the outages were related to problems with instrumentation; 6 were related to problems with valves; 5 were related to problems with electrical equipment; and 2 were related to the generator.

B. Outages:

1. **Forced**: There were 18 forced outages in 1975. The one of longest duration was for 128 hours, to repair the turbine control valves.

2. **Scheduled**: There were 4 scheduled outages during the report period. The ones of longest duration were: (1) 120 hours, for inspection of hydraulic snubbers, and (2) 1443 hours, for refueling.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/20	3	F	Reactor tripped by steam generator low level in coincidence with steam flow greater than feedwater flow caused by transient associated with a moisture separator — reheater drain system test.	A	3	Steam & Power (HB)	Instrumentation & Controls
2)	2/20	2	F	Steam generator high level caused by failure of the feedwater control valve to close.	A	3	Steam & Power (HH)	Valves
3)	2/20	1	F	Operating with feedwater control system in manual during unit start up. Reactor tripped by steam generator low level in coincidence with steam flow greater than feedwater flow.	G	3	Steam & Power (HB)	Instrumentation & Controls
4)	3/1	6	F	Unit was tripped by loss of generator field, caused by failure of generator's voltage regulatory power supply system.	A	3	Steam & Power (HA)	Generators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
5)	3/2	3	F	Load was tripped by 2 out of 3 channels, overpower delta T. Reactor protection system setpoints were being reduced. Overpower comparator failed to reset when the switch was returned to normal position and reactor tripped when next channel was placed in the tripped mode.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
6)	3/3	128	F	Unit was removed from service to repair turbine control valves.	A	3	Steam & Power (HA)	Valves
7)	4/23	19	F	Repaired steam generator feedwater control valves.	A	1	Steam & Power (HH)	Valves
8)	5/25	33	S	Unit taken off line to perform annual engineered safeguards test.	B	1	Engineered Safety (SF)	Instrumentation & Controls
9)	7/5	2	F	Steam generator main steam isolation valve was inadvertently closed while trouble shooting to locate ground on 125 V d.c. system. Reactor tripped on overtemperature delta T.	G	3	Electric Power (EC)	Electrical Conductors

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
10)	7/5	3	F	Reactor trip on steam generator low level coincidence with steam flow greater than feedwater flow with feedwater flow on manual control while returning the unit to service.	G	3	Steam & Power (HB)	Heat Exchangers
11)	7/15	120	S	Unit removed from service to perform hydraulic snubber inspection and secondary maintenance.	B	1	Reactor Coolant (CX)	Shock Suppressors
12)	7/20	1	S	Unit was removed from service to perform test on the turbine generator.	B	NA	Steam & Power (HA)	Generators
13)	7/20	3	F	Turbine was tripped on steam generator high level caused by the inadvertent rapid opening of turbine control valves.	A	3	Steam & Power (HB)	Valves
14)	7/23	1	F	Steam flow greater than feedwater flow mismatch with low level in steam generator.	G	3	Steam & Power (HB)	Instrumentation & Controls
15)	8/26	3	F	Reactor tripped on reactor coolant pump breaker open coincident with reactor power greater than 45% caused by reactor coolant pump relay.	A	3	Reactor Coolant (CB)	Relays

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
16)	8/26	8	F	Reactor coolant pump breaker opened on fault indication.	A	3	Reactor Coolant (CB)	Relays
17)	9/19	3	F	Turbine tripped on steam generator high level caused by failure of feedwater control valve actuator linkage. Turbine trip caused reactor trip.	A	3	Steam & Power (HB)	Valve Operators
18)	10/2	4	F	Reactor trip from open breaker on reactor coolant pump coincident with power greater than 45%. Burned relay.	A	3	Reactor Coolant (CB)	Relays
19)	10/7	1	F	Reactor tripped when the reactor trip breaker was opened in error during reactor protection system test.	G	2	Instrumentation & Controls (IA)	Circuit Closers
20)	10/17	3	F	Turbine trip from steam generator high level caused by failure of linkage on steam generator feedwater control valve actuator.	A	3	Steam & Power (HB)	Valve Operators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
21)	10/25	1443	S	Unit removed from service for refueling, maintenance, and inspection.	C	1	Reactor (RB)	Fuel Elements
22)	12/27	3	F	Generator tripped by actuation of the generator lockout relay caused by the malfunction of a pressure switch in the turbine protection system during test of the thrust bearing trip. Reactor tripped on steam generator low level.	A	3	Steam & Power (HA)	Instrumentation & Controls

I. Summary

Description

Location: Florida City, Florida
Docket No: 50-251
Reactor Type: PWR
Capacity (MWe net): 745
Commercial Operation: 9/7/73
Plant Age: 2.5 Years

Performance

Net Electrical Energy Generated (MWH):	3,989,524
Unit Availability Factor (%):	70.5
Unit Capacity Factor (%) (Using MDC):	68.4
Unit Capacity Factor (%) (Using Design MWE):	65.7

Outages

Total No.	14		
Forced	7		
Scheduled	7		
Total:	2,584	Hours,	29.5%
Forced	25	Hours,	0.3%
Scheduled	2,559	Hours,	29.2%

Cause:

Equipment Failure	7
Maintenance or Testing	6
Refueling	1

Method of Shutdown:

Manual	6
Automatic Scram	8

II. Highlights

A. General:

A total of 14 outages occurred in 1975. Seven were related to problems with equipment failure; 6 were for maintenance and/or tests; and, 1 was for refueling. Two of the outages were related to problems with and/or maintenance on the steam generator; and 6 were related to problems with valves.

B. Outages:

1. Forced: There were 7 forced outages in 1975. The duration of the longest outage was 5 hours. There were 3 fire hour outages.

2. Scheduled: There were 7 scheduled outages during the report period. The ones of longest duration were: (1) 2009 hours, for maintenance and refueling; (2) 251 hours, for repairs of tube leaks in the steam generator; and (3) 154 hours, for repairs of tube leaks in the steam generator.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/5	97	S	Scheduled maintenance. Repair of turbine exhaust expansion joint leak; repair of misc. steam leaks.	B	1	Steam & Power (HA)	Pipes, Fittings
2)	3/29	2009	S	Refueling, maintenance, and inspections.	C	1	Reactor (RC)	Fuel Elements
3)	6/22	3	S	Turbine overspeed test.	B	3	Steam & Power (HA)	Turbines
4)	6/27	2	F	Malfunction of heater drain pump discharge control valve actuation linkage.	A	3	Steam & Power (HH)	Valve Operators
5)	7/22	5	F	Reactor tripped on steam generator low level coincidence with steam flow > feed flow caused by trip of steam generator feedwater pump on low suction pressure.	A	3	Steam & Power (HH)	Pumps
6)	8/3	154	S	Repair of steam generator tube leak.	B	1	Steam & Power (HB)	Heat Exchangers
7)	9/3	2	F	Malfunction of feedwater control valve.	A	3	Steam & Power (HH)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	9/3	5	F	Malfunction of feedwater control valve.	A	3	Steam & Power (HH)	Valves
9)	9/3	2	F	Malfunction of feedwater control valve.	A	3	Steam & Power (HH)	Valves
10)	9/21	251	S	Repaired steam generator tube leak.	B	1	Steam & Power (HB)	Heat Exchangers
11)	10/11	27	S	Investigate and repair moisture-separator reheater and repair turbine governor control system.	B	1	Steam & Power (HA)	Mechanical Functions
12)	11/16	5	F	Reactor tripped on steam generator low level, coincidence with steam flow > feedwater flow caused by transient associated with automatic actuation of turbine runback system while performing turbine valve test.	A	3	Steam & Power (HB)	Valves
13)	11/16	4	F	Turbine was tripped on steam generator high level caused by failure of linkage on steam generator bypass feedwater control valve actuator.	A	3	Steam & Power (HH)	Valve Operators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
14)	12/7	18	S	Unit removed from service to perform required surveillance test on engineered safeguards system.	B	1	Engineered Safety (SX)	NA

VERMONT YANKEE

I. Summary

Description

Location: Vernon, Vermont
Docket No: 50-271
Reactor Type: BWR
Capacity (MWe-net): 514
Commercial Operation: 11/29/72
Plant Age (Years): 3.3 Years

Performance

Net Electrical Energy Generated (MWH):	3,561,206
Unit Availability Factor (%):	87.8
Unit Capacity Factor (%) (Using MDC):	80.7
Unit Capacity Factor (%) (Using Design MWE):	79.1

Outages

Total No.	9		
Forced	6		
Scheduled	3		
Total:	1,073	Hours,	12.2%
Forced	358	Hours,	4.1%
Scheduled	715	Hours,	8.1%

Cause:	Equipment Failure	3
	Maintenance or Testing	1
	Regulatory Restriction	2
	Operational Error	3

Method of Shutdown:

Manual	1
Manual Scram	3
Automatic Scram	4

II. Highlights

A. General:

A total of 9 outages occurred in 1975. Three were the result of equipment failure; 1 was for maintenance and/or testing; 2 were for regulatory restrictions; and 3 were the result of operational error. Two of the outages were related to problems with instrumentation; 2 were related to problems with electrical equipment; 3 were related to problems with valves and 1 was for inspection of pipings.

B. Outages:

1. Forced: There were 6 forced outages in 1975. Of these, the ones of longest duration were: (1) 249 hours, due to the failure of the startup transformer; and (2) 75 hours, due to an operator error which resulted in a high reactor water level scram.

2. Scheduled: There were 3 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 115 hours, for NRC directed weld inspection; (2) 540 hours, to install lower core plate plugs to reduce LPRM vibrations; and (3) 60 hours, for the installation of a new startup transformer.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	2/15	115	S	Shut down for NRC directed weld inspection of core spray piping.	D	2	Engineered Safety (SF)	Pipes, Fittings
2)	3/17	10	F	Operator switching error separated generator from grid. Auto scram from full load.	G	3	Steam & Power (HA)	Generators
3)	3/23	75	F	Operator error resulted in high reactor water level scram; outage extended to repack recirc suction valve.	G	3	Reactor Coolant (CB)	Valves
4)	4/21	8	F	One main steam radiation trip channel out of service for surveillance testing when fuse blew in operational channel causing auto scram.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
5)	5/11	9	F	During turbine control valve exercise, operator inadvertently closed turbine stop valve causing a high pressure scram.	G	3	Steam & Power (HA)	Valves
6)	6/5	249	F	Failure of startup transformer, which is power source for cooling tower fans, required shutdown and repair.	A	1	Electric Power (EB)	Transformers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	8/7	540	S	Shut down to install lower core plate plugs to reduce LPRM vibration.	D	2	Instrumentation & Controls (IA)	Instrumentation & Controls
8)	9/7	7	F	Repair of valve on moisture separator drain line.	A	NA	Steam & Power (HB)	Valves
9)	11/7	60	S	Installed new startup transformer.	B	2	Electric Power (EB)	Transformers

I. Summary

Description	Performance		Outages		
Location: Rowe, Massachusetts	Net Electrical Energy Generated (MWH):	1,193,421	Total No.	10	
Docket No: 50-29			Forced	6	
Reactor Type: PWR	Unit Availability Factor (%):	82.4	Scheduled	4	
Capacity (MWe-net): 175			Total:	1,547	Hours, 17.6%
Commercial Operation: 2/61	Unit Capacity Factor (%) (Using MDC):	77.8	Forced	27	Hours, 0.3%
Plant Age: 15.1 Years			Scheduled	1,520	Hours, 17.3%
	Unit Capacity Factor (%) (Using Design MWE):	77.8			

Cause:		
	Equipment Failure	5
	Maintenance or Testing	2
	Refueling	1
	Op. Tng. and License Exam	1
	Other	1
Method of Shutdown:		
	Manual	7
	Automatic Scram	3

II. Highlights

A. General:

A total of 10 outages occurred in 1975. Five were the result of equipment failure; 2 were for maintenance and/or testing; 1 was for licensing exams; 1 was for refueling; and one was due to external causes. Four of the outages were related to problems with electrical equipment; 2 were for control rod surveillance; 1 was related to problems with instrumentation; and 1 was for refueling.

B. Outages:

1. Forced: There were 6 forced outages in 1975. Of these, the one of longest duration was for 7 hours, for cleaning of electrical breakers.

2. Scheduled: There were 4 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 1458 hours for refueling, and (2) 24 hours for surveillance of control rod drop times.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	3/22	24	S	Shut down for control rod drop time surveillance testing and and operator training.	B	1	Reactor (RB)	Control Rods
2)	5/13	5	F	Electrical fault at hydro station caused relay operation at Yankee and isolation of service transformer. This resulted in loss of condensate pump which subsequently caused a scram from low steam generator level.	H	3	Electric Power (EA)	Electrical Conductors
3)	5/16	6	F	Loss of boiler feed pumps resulted in scram from low level in steam generator. Instrument drift.	A	3	Steam & Power (HH)	Instrumentation & Controls
4)	5/31	22	S	Surveillance testing of control rod drop times and operator training.	B	1	Reactor (RB)	Control Rods
5)	6/21	16	S	NRC reactor operator licensing exams.	E	1	Reactor (RB)	Control Rod Drive Mechanisms
6)	10/18	1458	S	Refueling	C	1	Reactor (RC)	Fuel Elements

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
7)	12/18	7	F	Arcing of airbreak contack in switchyard caused shutdown for cleaning.	A	1	Electric Power (EA)	Circuit Closers
8)	12/25	2	F	Excessive steam leakage from pressurizer spray valve.	A	1	Reactor Coolant (CB)	Valves
9)	12/27	4	F	Improper set on pilot wire relay opened transmission line and caused loss of power to one of reactor coolant pumps.	A	1	Reactor Coolant (CB)	Relays
10)	12/27	3	F	Improper set on pilot where relay opened transmission line and caused loss of power to reactor coolant pump.	A	3	Reactor Coolant (CB)	Relays

I. Summary

Description	Performance		Outages	
Location: Zion, Illinois	Net electrical energy generated (MWH):	4,909,363	Total No.	26
Docket No: 50-295			Forced	20
Reactor Type: PWR	Unit availability factor (%):	70.0	Scheduled	6
Capacity (MWe-Net): 1050			Total*:	2,632 Hours, 30.0%
Commercial Operation: 12/31/73	Unit capacity factor (%) (using MDC):	65.9	Forced	1,276 Hours, 14.6%
Plant Age: 2.5 Years			Scheduled	1,313 Hours, 15.0%
	Unit capacity factor (%) (using Design MWe):	54.1	Unidentified	43 Hours, 0.5%

*This data reflects a discrepancy of 43 hours which could not be resolved. Since it represents about 0.5%, it was allowed.

Cause:	
Equipment Failure	13
Maintenance or Testing	6
Op rational Error	7
Other	1
Method of Shutdown:	
Manual	7
Manual Scram	1
Automatic Scram	18

II. Highlights

A. General:

A total of 26 outages occurred in 1975. Thirteen were the result of equipment failure; 5 were for main enance and/or tests; 7 were the result of operational errors; and 1 was to convert to an all volatile chemistry treatment in the secondary system. Eight of the outages were related to problems with heat exchangers; 5 were related to problems with instrumentation; 4 were related to problems with pumps; 5 were related to problems with valves; 3 were related to problems with electrical equipment; and 1 was for inspection of piping.

B. Outages:

1. Forced: There were 20 forced outages in 1975. Of these, the ones of longest duration were: (1) 839 hours for inspection of coolant pump seals and pump replacement; and (2) 94 hours for repairs of a control rod drive ventilation fan.

2. Scheduled: There were 6 scheduled outages during the report period. Of these, the ones of longest duration were: (1) 760 hours for conversion to all volatile chemistry on the secondary system; (2) 308 hours for the installation of anti-vibration clips on condenser tubes; and (4) 136 hours for inspection of reactor coolant isolation valves.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/17	6	F	Reactor trip due to overpower temperature spike while other channel being tested.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls
2)	1/17	1	F	Reactor trip due to high level on steam generator.	A	3	Steam & Power (HB)	Heat Exchangers
3)	1/18	94	F	Control rod drive ventilation fan tripped which caused reactor trip.	A	1	Reactor (RB)	Blowers
4)	2/3	69	F	While shutting down to repair condenser tube leaks, the unit tripped on high steam generator level.	A	3	Steam & Power (HC)	Heat Exchangers
5)	2/6	30	F	Condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
6)	2/7	4	F	Steam generator steam flow/feed flow mismatch and low level.	A	3	Steam & Power (HB)	Instrumentation & Controls
7)	2/17	22	F	Unable to control steam generator levels due to oscillations of feedwater pump.	A	3	Steam & Power (HH)	Pumps

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	2/18	5	F	Steam generator high level caused turbine trip and reactor trip.	A	3	Steam & Power (HB)	Heat Exchangers
9)	2/25	760	S	Switch over to all volatile chemistry treatment on the secondary system.	H	3	Steam & Power (HB)	Heat Exchangers
10)	4/4	59	F	Condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
11)	4/25	13	F	Reactor trip due to reactor coolant pump trip and power greater than 60% during protection logic testing. Switch malfunction on loop isolation valve.	A	3	Reactor Coolant (CB)	Circuit Closers
12)	4/26	4	F	Turbine trip/reactor trip due to high steam generator level.	A	3	Steam & Power (HB)	Heat Exchangers
13)	5/5	8	F	While attempting to put D.C. power source on charge, operator de-energized the bus which caused reactor trip.	G	3	Electric Power (EC)	Electrical Conductors
14)	5/18	10	F	Main steam line isolation valve would not stroke during testing. The test solenoid was repaired.	A	2	Steam & Power (HB)	Valve Operators

DETAILS OF PLANT OUTAGES continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	5/24	308	S	Anti-vibration clips were installed on the condenser tubes. Maintenance was performed on pump seals.	B	1	Steam & Power (HC)	Heat Exchangers
16)	6/6	4	F	Steam generator high level EHC not adjusted for initial load.	G	3	Steam & Power (HB)	Instrumentation & Controls
17)	6/6	839	F	Trip from steam flow/feed flow mismatch. A drain valve was left open on the excess letdown system and 15,000 gals of coolant were lost. Inspected coolant pump seals. Replaced reactor coolant pump.	G	3	Reactor Coolant (CB)	Pumps
18a)	8/27	3	F	Instrument mechanic was working on the turbine drives feedwater pump and caused a pump overspeed trip.	G	3	Steam & Power (HH)	Pumps
18b)	8/27	39	S	Decision was made to keep the unit down until diesel generator repairs were completed.	B	—	Electric Power (EE)	Generators
19)	9/12	80	F	Erroneously aligned valves caused a containment spray while testing the containment spray pumps.	G	1	Engineered Safety (SB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
20)	9/15	1	F	Problem with turbine relatching valves.	A	3	Steam & Power (HA)	Valves
21)	9/18	20	F	Safety injection caused by workman bumping a main steam pressure transmitter with a ladder while instrument mechanics were working on another channel.	G	3	Steam & Power (HB)	Instrumentation & Controls
22)	9/19	136	S	Down to inspect reactor coolant isolation valves because of failure of similar valve on Unit 2.	B	1	Reactor Coolant (CB)	Valves
23)	10/24	24	S	Down to change oil in reactor coolant pumps. Reactor tripped from 3% power on lo-lo steam generator level while bringing unit down.	B	3	Reactor Coolant (CB)	Pumps
24)	11/4	4	F	While workman were changing a diode in control panel for feedwater regulating valve, the valve closed causing a steam flow-feedwater flow mismatch.	G	3	Steam & Power (HH)	Instrumentation & Controls

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
25)	11/28	5	S	While the unit was coming down for circulating water discharge inspection, it tripped at 8% power because of feedwater flow oscillations.	B	3	Auxiliary Water (WE)	Pipes, Fittings
26)	12/20	41	S	Repair leaking feedwater regulating valve.	B	1	Steam & Power (HH)	Valves

I. Summary

Description

Location: Zion, Illinois
Docket No: 50-304
Reactor Type: PWR
Capacity (MWe-Net): 1050
Commercial Operation: 9/17/74
Plant Age: 2.0 Years

Performance

Net Electrical Energy Generated (MWH):	4,828,978
Unit Availability Factor (%):	72.2
Unit Capacity Factor (%) (Using MDC):	64.9
Unit Capacity Factor (%) (Using Design MWE):	53.3

Outages

Total No.	32	
Forced	31	
Scheduled	1	
Total:*	2,436 Hours,	27.8%
Forced	2,335 Hours,	26.7%
Scheduled	31 Hours,	0.3%
Unidentified	70 Hours,	0.8%

Cause:

Equipment Failure	23
Maintenance or Testing	1
Operational Error	7
Other	1

Method of Shutdown:

Manual	11
Manual Scram	1
Automatic Scram	20

*This data reflects a discrepancy of 70 hours which could not be resolved. Since it represents less than 1%, it was allowed.

II. Highlights

A. General:

A total of 32 outages occurred in 1975. Twenty-three were the result of equipment failure; 1 was for maintenance and/or testing; 7 were the result of operational errors; and 1 was the result of condenser tube leaks on Unit 1. Twelve of the outages were related to problems with heat exchangers (either the steam generator or main condenser); 10 were related to problems with valves; 1 was related to a problem with the diesel generators; 5 were related to problems with instrumentation; 1 was related to a problem with electrical equipment; and 2 were related to shock suppressors.

B. Outages:

1. Forced: There were 31 forced outages in 1975. Of these, the ones of longest duration were: (1) 236 hours, for maintenance on a feedwater regulating valve; (2) 183 hours, due to problems with the secondary water chemistry; (3) 156 hours, due to leaks in the condenser tubes; (4) 122 hours, due to leaks in the condenser tubes; (5) 238 hours, due to leaks in the condenser tubes; (6) 599 hours, to install supports for extraction steam expansion bellows in the condenser; and (7) 565 hours, due to a suspected failure of an isolation valve in the reactor coolant loop.

2. Scheduled: There was only one scheduled outage during the report period. It was for 31 hours, for replacement of a feedwater regulating valve.

DETAILS OF PLANT OUTAGES

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
1)	1/1	236	F	Reactor tripped on 12-30-74 due to steam flow/feed flow mismatch and low level in steam generator. Maintenance performed on feedwater regulating valve.	A	3	Steam & Power (HH)	Valves
2)	1/10	14	F	Shut down because both diesels inoperable.	A	1	Electric Power (EE)	Generators
3)	1/11	3	F	Reactor trip due to high level in steam generator.	A	3	Steam & Power (HB)	Heat Exchangers
4)	2/8	183	F	Secondary water chemistry problems due to condenser tube leaks on Unit 1.	H	2	Steam & Power (HH)	
5)	2/17	156	F	Condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
6)	3/4	21	F	Steam flow/feed flow mismatch when instrument mechanics were performing steam flow calibration on channel with the channel being tested selected for steam generator level control.	G	3	Steam & Power (HB)	Instrumentation & Controls
7)	3/5	12	F	Reactor trip on steam flow/feed flow mismatch.	A	3	Steam & Power (HB)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
8)	3/5	4	F	While increasing valve limit position, governor valves closed and reactor tripped on steam generator low level.	A	3	Steam & Power (HA)	Valves
9)	3/9	5	F	Reactor & turbine trip caused by low flow signal which occurred when instrument mechanics put loop flow transmitter back in service.	G	3	Reactor Coolant (CB)	Instrumentation & Controls
10)	3/13	23	F	Power operated atmospheric-relief valve leaking. Manually tripped turbine with power less than 10%.	A	1	Steam & Power (HB)	Valves
11)	3/14	15	F	Moisture separator-reheater relief valves lifted. Also, intercept valve failed to open during attempt to re-escalate to power.	A	1	Steam & Power (HB)	Valves
12)	3/15	4	F	Turbine trip — reactor trip on high steam generator level.	A	3	Steam & Power (HB)	Heat Exchangers
13)	3/15	3	F	Turbine trip — reactor trip on high steam generator level.	A	3	Steam & Power (HB)	Heat Exchangers
14)	3/15	9	F	Intercept valve cycling. Replaced transmitter on controller.	A	3	Steam & Power (HB)	Valve Operators

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
15)	3/16	3	F	Steam generator high level.	A	3	Steam & Power (HB)	Heat Exchangers
16)	3/18	4	F	Reactor & turbine trip due to feedwater swell in loop 2D. Loop 2B feedwater piping snubber damaged during subsequent water hammer.	A	3	Steam & Power (HH)	Shock Suppressors
17)	3/18	4	F	Steam flow — feed flow mismatch and steam generator low level.	A	3	Steam & Power (HB)	Heat Exchangers
18)	3/19	122	F	Condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers
19)	4/1	8	F	Steam generator high level while attempting to restore auto control of feedwater regulating valve after calibrating steam generator level transmitter.	A	3	Steam & Power (HB)	Instrumentation & Controls
20)	4/1	2	F	Reactor trip due to steam generator low level and steam flow/feed flow mismatch.	A	3	Steam & Power (HB)	Heat Exchangers
21)	4/5	238	F	Condenser tube leaks.	A	1	Steam & Power (HC)	Heat Exchangers

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
22)	6/3	599	F	Installed supports for extraction steam expansion bellows in the condenser. Oil was changed in all reactor coolant pump motors.	A	1	Steam & Power (HC)	Shock Suppressors
23)	7/22	9	F	MSIV hydraulic system not working properly.	A	3	Steam & Power (HB)	Valve Operators
24)	7/23	1	F	Steam/feedwater flow mismatch while trying to bring unit up.	G	1	Steam & Power (HB)	Instrumentation & Controls
25)	8/8	31	S	Repacked feedwater regulating valve.	B	1	Steam & Power (HH)	Valves
26)	8/30	565	F	Failure of reactor coolant loop hot leg isolation valve stem causing disc to drop into flow path.	A	3	Reactor Coolant (CB)	Valves
27)	9/23	8	F	Unit tripped on low steam generator level.	G	3	Steam & Power (HB)	Heat Exchangers
28)	10/9	51	F	Clamp was installed on accumulator check valve to prevent leakage of 0.25 gpm into containment.	A	1	Reactor Coolant (CB)	Valves

DETAILS OF PLANT OUTAGES (continued)

No.	Date (1975)	Duration (Hrs)	Type	Description	Cause	Shutdown Method	System Involved	Component Involved
29)	10/22	3	F	During reactor protection test, the bypass reactor trip breaker was opened prior to closing the master trip breaker.	G	3	Instrumentation & Controls (IA)	Circuit Closers
30)	11/19	5	F	While performing load-follow test, unit tripped on overpower delta T.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls
31)	12/27	1	F	Unit tripped on steam-feedwater flow mismatch during return to power.	G	3	Steam & Power (HB)	Heat Exchangers
32)	12/29	24	F	Unit down to identify the 2.5 gpm reactor coolant system leak. The main part of the leak came from an accumulator check valve.	A	1	Reactor Coolant (CB)	Valves

APPENDIX C

LERS

Appendix C-1

LER SYSTEMS CATEGORIES (1974)

System	Subsystems Included
Reactor Core	pellets, cladding, and structural elements
Control Rod System	drive mechanisms, rods proper, housing, and associated circuitry
Core Support and Internal Structures	core support structures, fuel grid, shroud, and downcomers
Primary Coolant System	pressure vessel, main recirculation pumps, jet pumps, primary piping, valves, pressurizer, boric acid system, associated pressure, volume and temperature control systems, steam separators and driers in BWR's, main steam isolation valves (MSIV's) and primary side of steam generators
Secondary Systems	turbine, piping, valves, pumps, main condenser, and reheaters
Electrical Generation Systems	main generator(s), switch gear and associated controls
Containment Systems	containment structures, closures, pressure control system, and the pressure suppression chamber
Emergency Core Cooling Systems	core spray system, residual heat removal system, containment spray, and associated controls
Process Control Instrumentation	sensors, circuitry, logic modules, recorders and annunciators
Reactor Protection Systems	same as for PCI
Personnel Protection Radiation Monitoring Systems	same as for PCI
Offsite and Onsite Electrical Distribution Systems	switch gear, transformers, transmission lines

System	Subsystems Included
Gaseous Effluent Treatment Systems	standby gas treatment systems, recombiners, associated piping and valves, compressors, ejectors, charcoal absorbers, cryogenic units, sampling and monitoring instrumentation
Liquid Effluent Treatment	tanks, filters, evaporators, solidification systems, piping, pumps, conveyors, loadout stations, and associated instruments and controls
Auxiliary Systems	emergency diesel generators, batteries, battery chargers, service water system, coolant makeup system, fuel storage, vent systems, and heating and air conditioning
Other	

Appendix C-2

LER SYSTEMS CATEGORIES (1975)

System	Subsystems Included
Reactor core	Pellet core, cladding, element structures
Control rod	Drive mechanism, rods proper, housing, associated circuitry
Core support and reactor vessel internals	Core support structures, fuel grid, shroud, downcomers
Primary	Pressure vessel, main recirculation pumps, jet pumps, primary piping, valves, pressurizer, boric acid system, associated pressure, volume and temperature control systems, primary side or steam generators. BWR's also include steam separators, driers, main steam isolation valves (MSIV's) and the feedwater system and associated controls (excluding condensate system and full-flow demineralizers).
Secondary	Main steam lines, turbine, piping, valves, pumps, main turbine, condenser, reheaters, condensate system, full-flow demineralizers. PWR's also include feedwater system, and steam generators (secondary side).
Electrical generation	Main generator, switch gear and associated controls.
Containment	Containment structures, closure system, pressure control system, containment sprays, pressure suppression chamber, ice condenser.
Engineered safety	Low Pressure Coolant Injection (LPCI), High Pressure Coolant Injection (HPCI), core spray, Residual Heat Removal (RHR), core flooding, Reactor Core Isolation Cooling (RCIC), valves, associated controls.
Process control	Sensors, circuitry, logic modules, recorders, annunciator.

System	Subsystems Included
Reactor operation	Reactor protection instrumentation
Radiation monitor	Personnel protection radiation monitoring system
Electrical distribution	Switch gear, transformers, transmission lines
Gaseous effluent	Recombiners, associated piping and valves, compressors, ejectors, charcoal absorbers, cyrogenic units, sampling and monitoring instrumentation
Liquid effluent	Tanks, centrifuges, filters, evaporators, solidification systems, pumps, piping, conveyors, loadout stations, associated instruments and controls
Auxiliary	Emergency diesel generators, batteries, battery chargers, service water system, coolant makeup system, fuel storage, vent systems, heating and air conditioning
Other	Not specifically categorized

Appendix D

INTERIM CRITERIA

The following criteria for abnormal occurrence determinations were selected for interim use and policy determination.

A. Events involving an actual loss of the protection provided for the health or safety of the public. This includes:

-Significant exposure to, or release of, licensed material.

-Substantial loss of control over licensed material.

B. Events involving major reduction in the degree of protection provided for the health or safety of the public. This includes:

-Moderate exposure to, or release of, licensed material.

-Failure of key safety-related equipment when required to perform during operation.

Examples of the types of events that meet these criteria are:

1. Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankle, hands, or forearms of any individual to 75 rems or more of radiation. (10 CFR 20.403(b)(1))

2. An exposure to an individual in an unrestricted area such that a whole body dose received exceeds 0.5 rems. (10 CFR 20.105(a))

3. The effluent release of radioactive material in concentrations which, if averaged over a period of 24 hours, exceeds 500 times the regulatory limit. (Appendix B, Table II, 10 CFR 20)

4. Any substantiated theft, unlawful diversion, or loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas. (10 CFR 20)

5. Any missing discrete items or unaccounted for quantities of special nuclear material in amounts greater than two times the applicable limit of the limit of error for material unaccounted for. (10 CFR 70.51(e)(5))

6. Any substantiated case of actual or attempted sabotage.

7. Exceeding a safety limit of license technical specifications.

8. Failure of malfunction of one or more components that actually prevents, by itself, the fulfillment of the functional requirements of a system required to cope with accidents.

9. An accidental criticality. (10 CFR 70.52(a))

10. Discovery of a major deficiency in design, construction, or operation that has a generic safety implication which requires immediate remedial action.

11. A generic deficiency in the design, manufacture, or test of licensed material or equipment which threatens to cause personnel exposure or release of licensed material in excess of regulatory limits, or loss of facility operation in excess of one working week, or damage to property in excess of $100,000. (10 CFR 20.403(a))

12. Serious deficiency in management or procedural controls as evidenced by repeated items of noncompliance in several major areas.

13. Failure of reactor protective systems, or other system subject to limiting safety system settings, to initiate and complete the required protective function when required to perform.

14. Events that require the operation of an engineered safety system to protect fuel cladding, reactor coolant pressure boundary, or primary containment boundary.

15. Gross degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

16. Major condition not specifically considered in the Safety Analysis Report or technical specifications that required immediate remedial action.

17. Personnel error or procedural inadequacy which caused a system or structure with a safety or consequence limiting function to fail when required to perform that function.

18. Continued plant operation while exceeding a limiting condition for operation requiring an immediate shutdown or hot standby condition within 8 hours.