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Nuclear Power Plant Operating Experience - 1979

Annual Report

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NUCLEAR POWER PLANT OPERATING EXPERIENCE — 1979

EXECUTIVE SUMMARY

1. INTRODUCTION

This report summarizes the operating experience of 67 licensed nuclear power plants during 1979. Operating statistics and data are presented for each plant that was in commercial operation* at the end of the year and had sufficient electrical generation for meaningful analyses. The one exception is Three Mile Island 2. Authority to operate this facility was suspended by the Nuclear Regulatory Commission (NRC) July 20, 1979; therefore, operational data for this unit covers only the period January 1 through July 19.

At the end of 1979, there were 67 plants licensed to operate — 66 in commercial operation and one (Arkansas 2) in power ascension. Three plants were shut down for an indefinite period, with no decision yet made on future operation — Indian Point 1, Humboldt Bay, and Three Mile Island 2.

The commercial operating experience of 67 plants is reviewed. Included are the data for 25 boiling-water-reactor (BWR) plants; 41 pressurized-water-reactor (PWR) plants; and, for the first time, Fort St. Vrain, a plant equipped with a high-temperature gas-cooled reactor (HTGR). In comparison with the 1978 report (NUREG-0618), Humboldt Bay (BWR) has been deleted while Hatch 2 (BWR) and Fort St. Vrain (HTGR) have been added to the list of plants reviewed. Data only through July 19, 1979, were included for Three Mile Island 2 (PWR).

2. POWER GENERATION

Electrical Output for 1979

In 1979 the total net electrical output for 67 nuclear power plants in commercial operation was 251.94 billion kilowatt hours, which is 11.5% of the total electrical energy generated in the United States for the year from all sources. However, the total net electrical energy output generated by nuclear plants represents a 7.3% decrease compared to the output for 1978. This may be due partly to the increased use of coal and natural gas by electric utilities; however, the dominant cause was the Three Mile Island 2 accident and the regulatory restrictions resulting therefrom. Of the total net electrical energy output of nuclear power plants in 1979, 60.32% was produced by PWRs, 39.63% by BWRs, and 0.05% by the single HTGR.

Plant Availability Factor for 1979

The weighted average plant availability factor for all plants in 1979 was 67.3% for the 67 nuclear power plants in commercial operation.

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*See Appendix A for definition.

The average BWR and PWR availability factors for this period were 72.0% and 65.5%, respectively. The HTGR had an availability factor of 22.2%.

Plant Capacity Factors for 1979

Individual plant capacity factors were calculated using maximum dependable capacity (MDC)* and design electrical rating (DER),* both in megawatts electrical net [MW(e) net]. The weighted average capacity factors for the 67 commercial nuclear power plants were 59.7% using MDC and 58.2% using DER. These values reflect the lower capacity factors of the HTGR, which were 8.5% using MDC and 8.5% using DER. The combined weighted average values for the BWR and PWR plants were 60.5% using MDC and 59.0% using DER.

3. PLANT OUTAGES

During 1979, the 25 operating BWRs experienced an average of 2419 h of outage time compared to an average of 3169 h for the 41 operating PWRs. The percentage of forced outage time at BWRs was 37% compared to 46% at PWRs. The primary cause of forced outages at BWRs was equipment failure; at PWRs the primary cause of forced outages was regulatory restrictions.

Refueling was the primary cause of scheduled outages at both BWRs and PWRs. Regulatory restrictions and maintenance or testing accounted for large percentages of the scheduled outage time at both types of plants. The dominance of regulatory restrictions as the cause for large percentages of forced and scheduled outages was the result of action taken with regard to certain aspects of the Three Mile Island 2 accident and with regard to concern for seismic design deficiencies in safety-related piping.

Fort St. Vrain, an HTGR, began commercial operation July 1, 1979. For the remainder of the year, the unit acquired an availability factor of 22.2%, having experienced eight forced outages and three scheduled outages for a total outage time of 3434 h.

4. REPORTABLE OCCURRENCES

Licensee Event Reports

The 67 commercially operating plants covered in this report submitted 2874 Licensee Event Reports (LERs) during 1979, an increase of 193 over the 2681 submitted in 1978. Of these, 1219 were from the 25 BWR plants, 1609 were from the 41 PWR plants, and 46 were from the single HTGR.

Abnormal Occurrences

An abnormal occurrence is an incident or event which the NRC determines is significant from the standpoint of public health or safety. Each

*See Appendix A for definition.

quarter, the NRC submits to the Congress a report listing any abnormal occurrences for that period as required by Sect. 208 of the Energy Reorganization Act of 1974. The report contains the date and place, nature and probable consequences, cause or causes, and any action taken to prevent recurrence of each abnormal occurrence.

During 1979, there were seven abnormal occurrences reported for commercial nuclear power plants. A summary of each of these occurrences is given in this report. The titles and numbers assigned to these seven abnormal occurrences* are as follows:

AO 79-1 Degraded Engineered Safety Features
AO 79-2 Deficiencies in Piping Design
AO 79-3 Nuclear Accident at Three Mile Island
AO 79-5 Indication of Low Water Level in a Boiling-Water Reactor
AO 79-6 Damage to New Fuel Assemblies
AO 79-7 Deficient Procedures
AO 79-8 Major Degradation of Primary Containment Boundary

5. FUEL PERFORMANCE

The NRC does not monitor every fuel failure that occurs in licensed operating nuclear power plants. The approach taken is (1) to set up operating limits for radioactivity in the coolant (from fuel failures) which are stringent enough to ensure that the dose limits specified in the *Code of Federal Regulations* are not exceeded, and (2) to monitor only those fuel failures which are significant from the viewpoint of the number of fuel rods that failed or those in which the failure is due to a new fuel failure mechanism. Periodically, meetings are held with the nuclear fuel vendors to review the operating experience of their fuel. Operating reactors typically have ~40,000 fuel rods, and the average fuel rod failure rate during the last few years has been near or below 0.02% per cycle.[1] [This excludes the Three Mile Island 2 (TMI-2) reactor, which is estimated to have most, if not all, of its fuel damaged as a result of the accident described in Chap. 4.] Fuel performance has continually improved, yet deviations from the normal occur occasionally.

Specific Fuel-Related Incidents

Several events related to fuel performance were reported during calendar year 1979. The events addressed in the NRC's *Report to Congress on Abnormal Occurrences* (NUREG-0090 series) are briefly described below.

On March 28, 1979, a loss-of-coolant accident at Three Mile Island 2 resulted in structural damage to the upper 40% of the core. Most, if not all, of the fuel rods sustained some damage. The zirconium cladding underwent severe oxidation, which left it embrittled. Fuel melting is not suspected because the maximum temperature in the core was estimated to be well below the fuel melting point of 5100°F.[2] (LER 79-012)

*AO 79-4 concerns a fuel fabrication facility.

During a routine inspection of new fuel at Surry 2 on May 7, 1979, it was found that a substance, later identified as sodium hydroxide, had been poured on 62 of 64 new fuel assemblies. There was no apparent indication of damage to the assemblies; however, they were returned to the vendor for examination and refurbishment.[3] (LER 79-012)

There were 23 additional fuel-related incidents reported to the NRC in Licensee Event Reports; all are briefly described in the report.

6. RADIATION EXPOSURE

Occupational Radiation Exposure

Occupational radiation exposure data submitted to the NRC for workers employed at commercial nuclear power plants indicate that 54% of the total collective dose (man-rems) was incurred by contractor personnel at BWRs compared to 60% at PWRs. At PWRs, the largest portion (46%) of the collective dose (19,807 man-rems) was incurred by workers involved in special maintenance, while at BWRs the largest portion (39%) of the collective dose (16,682 man-rems) was incurred by workers involved in routine maintenance activities.

The average annual dose for individuals who received measurable exposures was 0.62 rems, remaining less than 1 rem as it has every year since 1972.

The total collective dose (39,759 man-rems) is considerably higher than last year's value. Part of the increase could be due to the fact that three additional PWRs completed 1 year of commercial operation and were included for the first time in this series of reports. The activities required by the NRC, as set forth in bulletins issued during 1979, also caused an increase in the collective dose received by workers at several plants.

NUCLEAR POWER PLANT OPERATING EXPERIENCE — 1979

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ABSTRACT

This report is the sixth in a series of reports issued annually that summarizes the operating experience of nuclear power plants in commercial operation in the United States. Power generation statistics, plant outages, reportable occurrences, fuel element performance, and occupational radiation exposure for each plant are presented and discussed, and summary highlights are given. The report includes 1979 data from 67 plants — 25 boiling-water reactor plants, 41 pressurized-water reactor plants, and 1 high-temperature gas-cooled reactor plant.

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1. INTRODUCTION

This report summarizes the operating experience of licensed nuclear power plants in the United States during 1979. 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.

At the end of 1979, 67 nuclear power plants were licensed to operate — 66 in commercial operation and 1 in power ascension (Arkansas 2). This excludes Indian Point 1, Humboldt Bay, and Three Mile Island 2, which are shut down indefinitely with no decision yet made on future operation. However, operational data for Three Mile Island 2 is included and reviewed for the period January 1 to July 19, 1979. The license for Three Mile Island 2 was suspended effective July 20, 1979.

The 1979 operating experience of 67 plants is reviewed; this includes the experience of 25 boiling-water-reactor (BWR) plants, 41 pressurized-water-reactor (PWR) plants, and 1 plant (Fort St. Vrain) equipped with a high-temperature gas-cooled reactor (HTGR), which began commercial operation on July 1, 1979. In comparison to the plants reviewed in the 1978 operating experience report (NUREG-0618), Humboldt Bay (BWR) was deleted while Hatch 2 (BWR), Three Mile Island 2 (PWR), and Fort St. Vrain (HTGR) were added. The plants included in this report are presented in Table 1.1 together with the date when each plant began commercial operation and the name of the nuclear steam-supply system (NSSS) manufacturer.

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 (LERs), including abnormal occurrences, fuel performance, and occupational radiation exposures.

Table 1.1 Nuclear Power Plants in Commercial Operation — 12/31/79[a]

Plant name	Utility	Reactor type	NSSS[b]	Began commercial operation
Dresden 1	Commonwealth Edison Co.	BWR	GE	7/60
Yankee-Rowe	Yankee Atomic Electric Co.	PWR	W	7/61
Big Rock Point	Consumers Power Co.	BWR	GE	3/63
San Onofre 1	Southern California Edison and San Diego Gas & Electric Co.	PWR	W	1/68
Haddam Neck	Connecticut Yankee Atomic Power Co.	PWR	W	1/68
La Crosse	Dairyland Power Cooperative	BWR	AC	11/69
Oyster Creek 1	Jersey Central Power & Light Co.	BWR	GE	12/69
Nine Mile Point	Niagara Mohawk Power Corp.	BWR	GE	12/69
Ginna	Rochester Gas & Electric Co.	PWR	W	7/70
Dresden 2	Commonwealth Edison Co.	BWR	GE	7/70
Point Beach 1	Wisconsin Electric Power Co. and Wisconsin-Michigan Power Co.	PWR	W	12/70
Robinson 2	Carolina Power and Light Co.	PWR	W	3/71
Millstone 1	Northeast Nuclear Energy Co.	BWR	GE	3/71
Monticello	Northern States Power Co.	BWR	GE	6/71
Dresden 3	Commonwealth Edison Co.	BWR	GE	11/71
Palisades	Consumers Power Co.	PWR	CE	12/71
Point Beach 2	Wisconsin Electric Power Co. and Wisconsin-Michigan Power Co.	PWR	W	10/72
Vermont Yankee	Vermont Yankee Nuclear Power Corp.	BWR	GE	11/72
Pilgrim 1	Boston Edison Co.	BWR	GE	12/72
Surry 1	Virginia Electric & Power Co.	PWR	W	12/72
Turkey Point 3	Florida Power & Light Co.	PWR	W	12/72
Maine Yankee	Maine Yankee Atomic Power Corp.	PWR	CE	12/72
Quad Cities 1	Commonwealth Edison Co. and Iowa-Illinois Gas & Electric Co.	BWR	GE	2/73
Quad Cities 2	Commonwealth Edison Co. and Iowa-Illinois Gas & Electric Co.	BWR	GE	3/73
Surry 2	Virginia Electric & Power Co.	PWR	W	5/73
Oconee 1	Duke Power Co.	PWR	BW	7/73
Indian Point 2	Consolidated Edison Co.	PWR	W	8/73
Turkey Point 4	Florida Power & Light Co.	PWR	W	9/73
Fort Calhoun 1	Omaha Public Power District	PWR	CE	9/73
Prairie Island 1	Northern States Power Co.	PWR	W	12/73
Zion 1	Commonwealth Edison Co.	PWR	W	12/73
Kewaunee	Wisconsin Public Service Corp.	PWR	W	6/74
Peach Bottom 2	Philadelphia Electric Co.	BWR	GE	7/74
Cooper Station	Nebraska Public Power District	BWR	GE	7/74
Browns Ferry 1	Tennessee Valley Authority	BWR	GE	8/74
Oconee 2	Duke Power Co.	PWR	BW	9/74
Three Mile Island 1	Metropolitan Edison Co.	PWR	BW	9/74
Zion 2	Commonwealth Edison Co.	PWR	W	9/74
Oconee 3	Duke Power Co.	PWR	BW	12/74

Table 1.1 (continued)

Plant name	Utility	Reactor type	NSSS[b]	Began commercial operation
Arkansas 1	Arkansas Power & Light Co.	PWR	W	12/74
Prairie Island 2	Northern States Power Co.	PWR	W	12/74
Peach Bottom 3	Philadelphia Electric Co.	BWR	GE	12/74
Duane Arnold	Iowa Electric Light & Power Co.	BWR	GE	2/75
Browns Ferry 2	Tennessee Valley Authority	BWR	GE	3/75
Rancho Seco	Sacramento Municipal Utility District	PWR	BW	4/75
Calvert Cliffs 1	Baltimore Gas & Electric Co.	PWR	CE	5/75
FitzPatrick	Power Authority of New York	BWR	GE	7/75
Cook	Indiana & Michigan Power Co.	PWR	W	8/75
Brunswick 2	Carolina Power & Light Co.	BWR	GE	11/75
Hatch 1	Georgia Power Co.	BWR	GE	12/75
Millstone 2	Northeast Nuclear Energy Co.	PWR	CE	12/75
Trojan	Portland General Electric Co.	PWR	W	5/76
Indian Point 3	Power Authority of New York	PWR	W	8/76
Beaver Valley 1	Duquesne Light Co.	PWR	W	10/76
St. Lucie 1	Florida Power & Light Co.	PWR	CE	12/76
Browns Ferry 3	Tennessee Valley Authority	BWR	GE	3/77
Crystal River 3	Florida Power Corp.	PWR	BW	3/77
Brunswick 1	Carolina Power & Light Co.	BWR	GE	3/77
Calvert Cliffs 2	Baltimore Gas & Electric Co.	PWR	CE	4/77
Salem 1	Public Service Electric & Gas Co.	PWR	W	6/77
Davis-Besse 1	Toledo Edison Co.	PWR	BW	11/77
Farley 1	Alabama Power Co.	PWR	W	12/77
Cook 2	Indiana & Michigan Power Co.	PWR	W	3/78
North Anna 1	Virginia Electric Power Co.	PWR	W	6/78
Fort St. Vrain	Public Service Co. of Colorado	HTGR	GA	7/79
Hatch 2	Georgia Power Co.	BWR	GE	9/79

[a]Does not include Three Mile Island 2 because its license was suspended effective July 20, 1979 (see Vol. 44, No. 149, p. 45271 of the *Federal Register*). However, the TMI-2 operational experience for 1979 is reviewed through July 19 elsewhere in this report. Humbolt Bay 3 (shut down 7/2/76) and Indian Point 1 (shut down 10/31/74) are not listed because they have been shut down, and no decision has yet been made on future operation.

[b]Abbreviations of nuclear steam-supply system manufacturers:

AC — Allis-Chalmers Mfg. Co.
BW — Babcock & Wilcox Co.
CE — Combustion Engineering, Inc.
GA — General Atomic Co.
GE — General Electric Co.
W — Westinghouse Electric Corp.

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2. POWER GENERATION

2.1 Introduction

Tables 2.1, 2.2, and 2.3 summarize the plant availability* and net electrical capacity factors* for the BWRs, PWRs, and HTGR, respectively, for 1979. Table 2.4 is a composite of the BWR and PWR power generation statistics for 1979. Similar information has been reported for the years 1973 through 1978 for the BWRs and PWRs.[1-5] This is the first report containing information on Fort St. Vrain, the only commercial HTGR plant in operation in the United States, because commercial operation began there on July 1, 1979.

2.2 Electrical Output for 1979

In 1979 the total net electrical output for 67 nuclear power plants in commercial operation was 251.94 billion kilowatt hours, which is 11.5% of the total electrical energy generated in the United States for the year from all sources.[6] However, the total net electrical energy output generated by nuclear power in 1979 represents a 7.3% decrease compared to the output for 1978. This may be due partly to the increased use of coal and natural gas by electric utilities;[6] however, the dominant cause was the Three Mile Island 2 accident and the regulatory restrictions resulting therefrom. Of the total net electrical energy output of nuclear power plants in 1979, 60.32% was produced by PWRs, 39.63% by BWRs, and 0.05% by the HTGR.

2.3 Plant Availability Factor for 1979

The weighted average plant availability factor for all plants in 1979 was 67.3% for the 67 nuclear power plants in commercial operation. The average BWR and PWR availability factors for this period were 72.0% and 65.5%, respectively. The HTGR had an availability factor of 22.2%.

The BWR availability factors ranged from zero for Dresden 1 to 97.6% for Monticello. Two BWR plants had availability factors below 50%, while 16 reported availability factors above 70%. Dresden 1 had an availability factor of zero because of the continuance of the shutdown which began October 31, 1978, for upgrading the emergency core-cooling system (ECCS), chemical cleaning, and refueling. This outage was expected to last a minimum of 18 months. Big Rock Point had a 23.5% availability factor resulting primarily from an extended refueling outage to eliminate the vibration in an inlet diffuser in the reactor vessel.

The PWR availability factors ranged from 9.3% for Surry 2 to 98.9% for Prairie Island 2. Seven PWR plants had availability factors below 50%, while 20 plants had availability factors of 70% or more. Surry 2 had an availability factor of 9.3% due to an extensive outage for steam generator replacement. The accident at Three Mile Island 2 resulted in

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*See Appendix A for definition.

Table 2.1. BWR Power Generation Statistics for 1979 (25 plants)

BWR plants	Design electrical capacity [MW(e) net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age[a] (years)
				Using MDC	Using design MW(e)	
Big Rock Point	72	113,674	23.5	20.6	18.0	17.1
Browns Ferry 1	1,065	7,495,748	90.4	80.3	80.3	6.2
Browns Ferry 2	1,065	7,441,305	86.7	79.8	79.8	5.3
Browns Ferry 3	1,065	5,482,585	65.2	58.8	58.8	3.3
Brunswick 1	821	3,169,212	54.6	45.8	44.1	3.1
Brunswick 2	821	3,652,260	65.6	52.8	50.8	4.7
Cooper Station	778	4,994,938	87.6	74.6	73.3	5.6
Dresden 1	200	−13,047	0.0	0.0	0.0	19.7
Dresden 2	794	4,939,630	81.6	73.0	71.0	9.7
Dresden 3	794	3,475,813	67.7	51.3	50.0	8.4
Duane Arnold	538	2,898,764	78.0	64.3	61.5	5.6
FitzPatrick	821	2,964,590	50.8	42.3	41.2	4.9
Hatch 1	786	3,337,875	54.6	49.9	48.5	5.1
Hatch 2	784	1,757,131	85.2	82.8	79.1	1.3
La Crosse	50	200,932	71.8	47.8	45.9	11.7
Millstone 1	660	4,221,264	77.3	73.7	73.0	9.1
Monticello	545	4,399,560	97.6	93.7	92.2	8.8
Nine Mile Point	620	3,005,389	66.1	56.2	55.3	10.1
Oyster Creek	650	4,563,223	85.9	84.0	80.1	10.3
Peach Bottom 2	1,065	8,574,430	94.7	93.1	91.9	5.9
Peach Bottom 3	1,065	6,101,657	74.2	67.3	65.4	5.3
Pilgrim 1	655	4,844,559	89.4	82.5	84.4	7.5
Quad Cities 1	789	4,782,963	81.3	71.0	69.2	7.7
Quad Cities 2	789	3,981,065	87.8	59.1	57.6	7.6
Vermont Yankee	514	3,448,842	82.1	78.1	76.6	7.3
Total	17,806	99,834,368				
Average	712	3,993,375	72.0	63.3	61.9	7.7

[a]Computed from date of first electrical generation through Dec. 31, 1979.

Table 2.2. PWR Power Generation Statistics for 1979 (41 plants)

PWR plants	Design electrical capacity [MW(e) net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age[a] (years)
				Using MDC	Using design MW(e)	
Arkansas 1	850	3,323,490	55.3	45.4	44.6	5.4
Beaver Valley 1	852	1,778,375	40.0	24.8	23.8	3.6
Calvert Cliffs 1	845	4,194,218	70.3	59.1	56.7	5.0
Calvert Cliffs 2	845	5,488,991	77.6	76.0	74.2	3.1
Cook 1	1,054	5,660,137	64.7	61.9	61.3	4.9
Cook 2	1,100	5,953,413	65.9	62.8	61.8	1.8
Crystal River 3	825	3,761,775	58.9	53.9	52.1	2.9
Davis-Besse 1	906	3,129,118	67.0	39.4	39.4	2.3
Farley 1	829	1,743,590	28.6	24.0	24.0	2.4
Fort Calhoun	457	3,666,112	95.7	91.6	91.6	6.4
Ginna	470	2,960,510	72.8	71.9	71.9	10.1
Haddam Neck	575	4,116,339	87.5	85.4	81.7	12.4
Indian Point 2	873	4,804,928	70.3	64.0	62.8	6.5
Indian Point 3	965	4,794,627	66.5	56.7	56.7	3.7
Kewaunee	535	3,439,289	79.0	75.5	73.4	5.7
Maine Yankee	825	4,539,015	68.4	64.0	62.8	7.1
Millstone 2	870	4,363,567	62.8	59.5	58.6	4.1
North Anna 1	907	4,188,866	61.7	53.2	52.7	1.7
Oconee 1	887	5,000,177	71.0	66.4	64.4	6.7
Oconee 2	887	5,968,288	86.0	79.2	76.8	6.1
Oconee 3	887	3,259,529	46.1	43.3	41.9	5.3
Palisades	805	3,433,264	59.9	61.7	48.7	8.0
Point Beach 1	497	3,055,424	76.2	70.5	70.2	9.2
Point Beach 2	497	3,707,450	88.5	85.5	85.2	7.4
Prairie Island 1	530	2,910,820	73.1	66.1	62.7	6.1
Prairie Island 2	530	4,193,044	98.9	95.7	90.3	5.0
Rancho Seco	918	5,711,999	91.1	74.7	71.0	5.2
Robinson 2	700	4,005,007	70.8	68.8	65.3	9.3
Salem 1	1,090	2,042,610	25.5	21.6	21.4	3.0

Table 2.2 (continued)

PWR plants	Design electrical capacity [MW(e) net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age[a] (years)
				Using MDC	Using design MW(e)	
San Onofre 1	436	3,355,531	90.2	87.9	87.9	12.5
St. Lucie 1	802	4,885,058	74.0	71.8	69.5	3.7
Surry 1	822	2,255,180	75.3	33.2	31.3	7.5
Surry 2	822	611,521	9.3	9.0	8.5	6.8
Three Mile Island 1	819	848,038	12.9	12.5	11.8	5.5
Three Mile Island 2[b]	906	1,318,113	33.6	31.2	30.3	1.7
Trojan	1,130	5,266,720	58.1	55.7	53.2	4.0
Turkey Point 3	693	2,874,917	51.8	49.3	47.4	7.2
Turkey Point 4	693	3,845,291	72.9	65.9	63.3	6.5
Yankee-Rowe	175	1,232,264	81.6	80.4	80.4	19.1
Zion 1	1,040	5,537,168	68.1	60.8	60.8	6.5
Zion 2	1,040	4,759,996	67.2	52.2	52.2	6.0
Total	32,189	151,983,769				
Average	785	3,706,921	65.5	58.8	57.2	6.0

[a]Computed from date of first electrical generation through Dec. 31, 1979.

[b]Data given are for the period Jan. 1 through July 19. The TMI-2 license was suspended effective July 20, as announced in the *Federal Register*, Vol. 44, No. 149, p. 45271.

Table 2.3. HTGR Power Generation Statistics for 1979 (1 plant)

HTGR plant	Design electrical capacity [MW(e) net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age[a] (years)
				Using MDC	Using design MW(e)	
Fort St. Vrain[b]	330	123,584	22.2	8.5	8.5	3.1

[a]Computed from date of first electrical generation through Dec. 31, 1979.

[b]Data given are for the period July 1, 1979 (date when commercial operation began) through Dec. 31, 1979.

Table 2.4. Composite of BWR and PWR Power Generation Statistics for 1979

Plants	Design electrical capacity [MW(e) net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age (years)
				Using MDC	Using design MW(e)	
25 BWRs	17,806	99,834,368	72.0	63.3	61.9	7.7
41 PWRs	32,189	151,983,769	65.5	58.8	57.2	6.0
Total	49,995	251,818,137				
Average	758	3,815,426				
Weighted average			68.0	60.5	59.0	6.6

a plant availability factor of 12.9% for Unit 1, which still has an operating license. The factor for Unit 2 (33.6%) is higher because only data through July 19, 1979, were considered, since the license for this unit was suspended effective July 20, 1979. (A factor of 18.4% is obtained if the entire year is considered.) Salem 1 had a factor of 25.5% due primarily to turbine blade replacement. At Farley 1 (which had a factor of 28.6%) a refueling shutdown was extended for the purpose of testing concrete expansion anchor belts.

2.4 Plant Capacity Factors for 1979

Individual plant capacity factors were calculated using maximum dependable capacity (MDC)* and design electrical rating (DER),* both in megawatts electrical net [MW(e) net]. The weighted average capacity factors for the 67 commercial nuclear power plants were 59.7% using MDC and 58.2% using DER. These values reflect the lower capacity factors of the HTGR, which were 8.5% using MDC and 8.5% using DER. The combined weighted average values for the BWR and PWR plants were 60.5% using MDC and 59.0% using DER.

The average capacity factors for the 25 BWRs were 63.3% and 61.9% using MDC and DER, respectively. The MDC capacity factors varied from zero to 93.7%; the DER capacity factors ranged from zero to 92.2%. Six BWRs had capacity factors below 50% using MDC, while 12 were above 70%. Six BWRs had capacity factors below 50% using DER, while 11 were above 70%.

The average capacity factors for the 41 PWRs were 58.8% and 57.2% using MDC and DER, respectively. The MDC capacity factors varied from 9% to 95.7%; the DER capacity factors ranged from 8.5% to 91.6%. Eleven PWRs had MDC capacity factors below 50% while 13 were above 70%. Using DER, 12 PWRs had capacity factors below 50%, while 12 were above 70%.

Power generation information for 1979 is summarized in Table 2.1 through 2.4. More detailed information on individual plants is presented in Appendix B. Tables 2.5 through 2.8 give the distributions of availability and capacity factors as a function of age. Availability and capacity factor distribution is given in Table 2.9.

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*See Appendix A for definition.

Table 2.5. BWR Plant Availability and Capacity Factors as a Function of Plant Age for 1979[a]

Plant age group (years)	Number of plants in age group	Average availability factor (%)	Average capacity factor (%)
0–0.9	0		
1–1.9	1	85.2	79.1
2–2.9	0		
3–3.9	2	59.9	51.5
4–4.9	2	58.2	46.0
5–5.9	6	79.3	70.1
6–6.9	1	90.4	80.3
7–7.9	4	85.1	72.0
8–8.9	2	82.7	71.1
9–9.9	2	79.5	72.0
10–10.9	2	76.0	67.7
11–11.9	1	71.8	45.9
12–16.9	0		
17–20.0	2[b]	11.8	9.0

[a]Based on design electrical rating (DER), megawatts electrical [MW(e)].

[b]Includes Dresden 1, which was shut down all year (beginning Oct. 31, 1978) to upgrade the ECCS. The other unit in this age group is Big Rock Point, which was shut down for ~202 d to eliminate vibration in an inlet diffuser in the reactor vessel.

Table 2.6. PWR Plant Availability and Capacity Factors as a Function of Plant Age for 1979[a]

Plant age group (years)	Number of plants in age group	Average availability factor (%)	Average capacity factor (%)
0–0.9	0		
1–1.9	3	53.7	48.3
2–2.9	3	51.5	38.5
3–3.9	5	56.7	49.1
4–4.9	3	61.9	57.7
5–5.9	7	64.8	55.8
6–6.9	9	68.2	60.3
7–7.9	4	71.0	56.7
8–8.9	1	59.9	48.7
9–9.9	2	73.5	67.8
10–10.9	1	72.8	71.9
11–11.9	0		
12–16.9	2	88.9	84.8
17–20.0	1	81.6	80.4

[a]Based on design electrical rating (DER), megawatts electrical [MW(e)].

Table 2.7. HTGR Plant Availability and Capacity Factors as a Function of Plant Age for 1979[a]

Plant age group (years)	Number of plants in age group	Average availability factor (%)	Average capacity factor (%)
3.1	1	22.2	8.5

[a]Based on design electrical rating (DER), megawatts electrical [MW(e)].

Table 2.8. Composite of BWR and PWR Plant Availability and Capacity Factors as a Function of Plant Age for 1979[a]

Plant age group (years)	Number of plants in age group	Average availability factor (%)	Average capacity factor (%)
0–0.9	0		
1–1.9	4	61.6	56.0
2–2.9	3	51.5	38.5
3–3.9	7	57.6	49.8
4–4.9	5	60.4	53.0
5–5.9	13	71.5	62.4
6–6.9	10	70.4	62.3
7–7.9	8	78.1	64.4
8–8.9	3	75.1	63.6
9–9.9	4	76.5	69.9
10–10.9	3	74.9	69.1
11–11.9	1	71.8	45.9
12–16.9	2	88.9	84.8
17–20.0	3	35.1	32.8

[a]Based on design electrical rating (DER), megawatts electrical [MW(e)].

Table 2.9. Distribution of BWR and PWR Plant Availability Factors and Plant Capacity Factors for 1979[a]

Availability factor (%)	Number of BWRs	Number of PWRs	Total number of plants
90 and over	3	4	7
80–90	9	4	13
70–80	4	12	16
60–70	3	9	12
50–60	4	5	9
Less than 50	2	7	9
	25	41	66
Average availability factors, %	72.0	65.5	67.9

Capacity factor using MDC (%)	Number of BWRs	Number of PWRs	Total number of plants
90 and over	2	2	4
80–90	4	4	8
70–80	6	7	13
60–70	2	10	12
50–60	5	7	12
Less than 50	6	11	17
	25	41	66
Average capacity factors using MDC, %	63.3	58.8	60.5

Capacity factor using DER (%)	Number of BWRs	Number of PWRs	Total number of plants
90 and over	2	2	4
80–90	3	4	7
70–80	6	6	12
60–70	3	10	13
50–60	5	7	12
Less than 50	6	12	18
	25	41	66
Average capacity factors using DER, %	61.9	57.2	59.0

[a]See Table 2.3 for the data on the one HTGR in the United States.

3. PLANT OUTAGES

3.1 Introduction

A review of the plant outages that occurred during 1979 provides a means of assessing the nature, number, and extent of the operating problems experienced at nuclear power plants during the year, as well as the principal systems and components involved. The data for this review were obtained from the data submitted by the licensees for the NRC's monthly publication, *Operating Units Status Report*.

In a few cases, the outage type was classified differently than reported by the licensee. For example, 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. Also, 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 also classified as a scheduled outage.

The tables appearing in this chapter present plant outage data only for the 66 light-water-reactor (LWR) plants commercially operable in 1979. The outage experience for the single HTGR is summarized in Sect. 3.5.4, and details may be readily obtained from the data sheets in Appendix B. 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 Plant Outage Statistics

There were 698 outages, requiring 193,825 h of shutdown time, reported by the 67 nuclear power plants which were in commercial operation during 1979. The 66 LWR plants accounted for 687 outages, requiring 190,391 h — an average of 33.5% for the year. Forced outage time for the LWRs averaged 14.4%, and scheduled outage time averaged 19.1%. The average total unit availability for the 66 LWRs was 68.0%.†

Table 3.1 presents the 1979 performance data for BWRs and lists the systems and components involved in the major outages, that is, outages lasting 5 days (120 h) or longer. Table 3.2 presents similar information for PWRs. Nine major outages at BWRs and six at PWRs involved pipe supports and snubbers. Fifteen major outages at PWRs involved the feedwater system — a reflection of the impact of the TMI-2 accident.

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*See Appendix A for definition.

†The availability plus the percent of total outage time exceeds 100% because, by definition, the availability factor includes "unit reserve shutdown hours" which are also counted in the "total outage time."

Table 3.1. Summary of BWR Power Plant Outages During 1979

	Big Rock Point 1	Browns Ferry 1	Browns Ferry 2	Browns Ferry 3	Brunswick 1	Brunswick 2	Cooper	Dresden 1	Dresden 2	Dresden 3	Duane Arnold	FitzPatrick	Hatch 1
Summary of performance data													
Percent of year operational	100	100	100	100	100	100	100	100	100	100	100	100	100
Percent of year in commercial operation	100	100	100	100	100	100	100	100	100	100	100	[illegible]	100
Scheduled outages during commercial operation													
Hours	1832	472	955	2614	3063	2363	781	8760	1380	142	0	0	3192
Percent	20.9	5.4	10.9	29.8	35.0	26.9	8.9	100.0	15.7	1.6	0	0	36.4
Forced outages during commercial operation													
Hours	4865	366	208	438	915	654	305	0	234	2684	1930	4309	783
Percent	55.6	4.2	2.4	5.0	10.4	7.5	3.5	0	2.7	30.7	22.0	49.2	9.0
Total outage time during commercial operation													
Hours	6697	838	1163	3052	3978	3017	1086	8760	1614	2826	1930	4309	3975
Percent	76.5	9.6	13.3	34.8	45.4	34.4	12.4	100.0	18.4	32.3	22.0	49.2	45.4
Unit availability in commercial operation													
Percent	23.5	90.4	86.7	65.2	54.6	65.6	87.6	0	81.6	67.7	78.0	50.8	54.6

	Hatch 2	La Crosse	Millstone 1	Monticello	Nine Mile Point	Oyster Creek	Peach Bottom 2	Peach Bottom 3	Pilgrim 1	Quad Cities 1	Quad Cities 2	Vermont Yankee	Totals
Summary of performance data													
Percent of year operational	100	100	100	100	100	100	100	100	100	100	100	100	
Percent of year in commercial operation	32	100	100	100	100	100	100	100	100	100	100	100	
Scheduled outages during commercial operation													
Hours	223	1578	1560	54	2915	0	410	2019	40	1218	1002	1450	38,023
Percent	7.9	18.0	17.9	0.6	33.2	0	4.7	23.1	0.4	13.5	11.4	16.6	17.8
Forced outages during commercial operation													
Hours	196	897	423	156	57	1236	54	238	891	432	70	115	22,456
Percent	7.0	10.2	4.8	1.8	0.7	14.1	0.6	2.7	10.2	4.9	0.8	1.3	10.5
Total outage time during commercial operation													
Hours	419	2475	1983	210	2972	1236	464	2257	931	1650	1372	1565	60,479
Percent	14.8	28.2	22.7	2.4	33.9	14.1	5.3	25.8	10.6	18.8	12.2	17.9	28.3[a]
Unit availability in commercial operation													
Percent	85.2	71.8	77.3	97.6	66.1	85.9	94.7	74.2	89.4	81.3	87.8	82.1	72.0[a]

Table 3.1 (continued)

	Big Rock Point 1	Browns Ferry 1	Browns Ferry 2	Browns Ferry 3	Brunswick 1	Brunswick 2	Cooper	Dresden 1	Dresden 2	Dresden 3	Duane Arnold	FitzPatrick	Hatch 1	Hatch 2	La Crosse	Millstone 1	Monticello	Nine Mile Point	Oyster Creek	Peach Bottom 2	Peach Bottom 3	Pilgrim 1	Quad Cities 1	Quad Cities 2	Vermont Yankee	Totals
Systems and components causing major outages (figures indicate number of outages lasting 5 days or longer)																										
Fuel inspection or replacement	1	1	1	1	1	1	1		1				1		1	1		1			1		1	1	2	17
Engineered safety features			1			2		1				1														5
Reactor coolant system					2						1		1		1	1			1		3		1			11
Radioactive waste system																				1	1					2
Electric power					1		1			2																4
Main generator													1													1
Control rod drives															2							1				3
Reactor internals	1																									1
Recirculation pump and motor				1													1		1							3
Pipe supports and snubbers					3	3			1													1	1			9

[a]There were 19.8 h of reserve shutdown time, which is equal to 0.2% unit availability.

Table 3.2. Summary of PWR Power Plant Outages During 1979

	Arkansas 1	Beaver Valley 1	Calvert Cliffs 1	Calvert Cliffs 2	Cook 1	Cook 2	Crystal River 3	Davis-Besse 1	Farley 1	Fort Calhoun	Ginna	Haddam Neck	Indian Point 2	Indian Point 3	Kewaunee	Maine Yankee	Millstone 2	North Anna 1	Oconee 1	Oconee 2	Oconee 3
Summary of performance data																					
Percent of year operational	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100
Percent of year in commercial operation	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100
Scheduled outages during commercial operation																					
Hours	2141	744	1594	1309	[illegible]68	1785	1395	3704	2175	175	1952	1302	2294	2839	934	709	1846	2344	1145	18	1497
Percent	24.4	8.5	18.2	14.9	16.8	20.4	15.9	42.3	24.8	2.0	22.3	14.9	26.2	32.4	10.7	8.1	21.1	26.7	13.1	0.2	17.1
Forced outages during commercial operation																					
Hours	2362	4513	1012	65[illegible]	1623	1201	2205	914	4081	201	430	30	305	96	902	2057	1525	1014	1392	1208	3224
Percent	27.0	51.5	11.5	7.5	18.5	13.7	25.2	10.4	46.6	2.3	4.9	0.3	3.5	1.1	10.3	23.5	17.4	11.6	15.9	13.8	36.8
Total outage time during commercial operation																					
Hours	4503	5257	2606	1963	3091	2986	3600	4618	6256	376	2382	1332	2599	2935	1836	2766	3371	3358	2537	1226	4721
Percent	51.4	60.0	29.7	22.4	35.3	34.1	41.1	52.7	71.4	4.3	27.2	15.2	29.7	33.5	21.0	31.6	38.5	38.3	29.0	14.0	53.9
Unit availability in commercial operation																					
Percent	55.3	40.0	70.3	77.6	64.7	65.9	58.9	67.0	28.6	95.7	72.8	87.5	70.3	66.5	79.0	68.4	62.8	61.7	71.0	86.0	46.1
Systems and components causing major outages (figures indicate number of outages lasting 5 days or longer)																					
Fuel inspection or replacement	1	1	1	1	1	1	1		1		1	1	1	1	1		1	1	1		2
Main turbine	2		2					1													
Condenser				1			1													1	
Feedwater system	3				1	1		1											1	1	1
Steam generators							1			1	2	1		1	1	1	1		1		
Reactor coolant system													1		1		1	1	1	1	
Reactor coolant pumps				3			1	3	1						1			1			
Engineered safety features																			1		
Main steam system		1						1									1				
Main generator					1																
Electric power																					1
Pipe supports and snubbers					1				1												1
Instrumentation and controls																			1		
Safety related piping		1														1					

Table 3.2 (continued)

	Palisades	Point Beach 1	Point Beach 2	Prairie Island 1	Prairie Island 2	Rancho Seco	Robinson 2	Salem 1	San Onofre 1	St. Lucie 1	Surry 1	Surry 2	Three Mile Island 1	Three Mile Island 2[a]	Trojan	Turkey Point 3	Turkey Point 4	Yankee-Rowe	Zion 1	Zion 2	Totals
Summary of performance data																					
Percent of year operational	100	100	100	100	100	100	100	100	100	100	100	100	100	55	100	100	100	100	100	100	
Percent of year in commercial operation	100	100	100	100	100	100	100	100	100	100	100	100	100	55	100	100	100	100	100	100	
Scheduled outages during commercial operation																					
Hours	2755	2033	1022	834	44	1581	2312	2115	402	2078	0	7941	940	13	3569	4102	2012	1436	1101	954	70,614
Percent	31.4	23.2	11.6	9.5	0.5	18.0	26.4	24.1	4.6	23.7	0	90.7	10.[illegible]	0.3	40.7	46.8	22.9	16.4	12.6	10.9	19.9
Forced outages during commercial operation																					
Hours	760	269	5	1520	50	401	272	4413	453	212	5714	0	6692	3172	102	146	384	175	1689	1920	59,298
Percent	8.7	3.1	0.1	17.4	0.6	4.6	3.1	50.4	5.2	2.4	65.2	0	76.4	66.1	1.2	1.7	4.4	2.0	19.3	21.9	16.7
Total outage time during commercial operation																					
Hours	3515	2302	1027	2354	94	1982	2584	6528	855	2290	5714	7941	7632	3185	3671	4248	2396	1611	2790	2874	129,912
Percent	40.1	26.3	11.7	26.9	1.1	22.6	29.5	74.5	9.8	26.1	65.2	90.7	87.1	66.4	41.9	48.5	27.3	18.4	31.9	32.8	36.6[b]
Unit availability in commercial operation																					
Percent	59.9	76.2	88.3	73.1	98.9	91.1	70.8	25.5	90.2	74.0	75.3	9.3	12.9	33.6	58.1	51.8	72.9	81.6	68.1	67.2	65.5%[b]
Systems and components causing major outages (figures indicate number of outages lasting 6 days or longer)																					
Fuel inspection or replacement	1	1	1	1			1	1		1		1	1			2	1		1	1	32
Main turbine				1				1													7
Condenser																					3
Feedwater system			1			1		1		1			1	1				1	1	1	15
Steam generators		2		1			1		1			1									16
Reactor coolant system										1				1		1			1		10
Reactor coolant pumps										1	1					2	2				16
Engineered safety features								1							1	1		1	1		6
Main steam system															1	1					5
Main generator																					1
Electric power				1					1												3
Pipe supports and snubbers	1														1			1			6
Instrumentation and controls																					1
Safety related piping								1	1		1										5

[a]Commercially operable for 3799 h, based on data through July 19. License was suspended effective July 20, 1979.

[b]There were 7401 h of reserve shutdown time, which is equal to an average of 2.[illegible]% unit availability.

3.3 Types of Outages at LWRs

The data on forced and scheduled outages at BWRs and PWRs for plants in commercial operation in 1979 are summarized in Table 3.3. The average number of forced outages was eight per plant, with each outage averaging 158 h. The average number of scheduled outages was three per plant, with each one averaging 635 h compared to 466 h in 1978 — an increase of 36%. On the average, each plant experienced 11 outages, totaling 2885 h.

3.4 Approximate Cause of Plant Outages at LWRs

Plant outages at LWRs and their approximate causes are summarized in Table 3.4. Each outage cause was determined by the NRC staff to be in one of the following eight categories: (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 — 70,089 h (37%). Equipment failures (forced) accounted for 44,471 h or 23% of all outage time. Regulatory restrictions (forced and scheduled) accounted for 53,989 h — 28% of the total outage time. This is a significant increase over that accumulated in 1978 when only 11% (15,090 h) of the total outage time was for regulatory restrictions. The increase was due primarily, but not completely, to restrictions concerning seismic design deficiencies in safety-related piping (~16,000 h — see Sect. 4.3.2) and restrictions concerning certain aspects of operation related to the Three Mile Island 2 accident (~12,000 h — see Sect. 4.3.3).

Although the number of LWR plants considered in this review increased by only 1 (1.5%) from 1978 to 1979, the total outage time increased by 48,323 h (34%) — a reverse of the decreasing trend that had been occurring since 1976 and due primarily to an increase in 1979, over 1978, of 38,899 h of outage time as a result of regulatory restrictions.

Table 3.5 lists the ratio of outage hours for various causes to 100 h of commercial operation. These numbers may also be considered as the percent of time expended for each cause. In 1979, there were 24 BWRs commercially operable 100% of the year (8760 h) and one commercially operable 32% of the year (2833 h); therefore, the total number of operating hours considered for BWRs was 213,073 h. For the PWRs, there were 40 units commercially operable all year and one unit commercially operable 55% of the year (4799 h), giving a total of 355,199 h of operation for the PWRs.

The table indicates that PWRs (as a class) accumulated a larger percentage of outage time than did BWRs for all causes except operational error. The effect of regulatory restrictions at PWRs is apparent but more significant than it appears from the table, because 4.1 h of the 7.1 h shown for BWRs (as a class) was due to the shutdown of Dresden 1 all year for upgrading the ECCS.

Table 3.3. Summary of BWR and PWR Nuclear Power Plant Outages by Type for 1979[a]

Plant type and number	Forced outages		Scheduled outages		Total outages	
	Number of events	Outage duration (h)	Number of events	Outage duration (h)	Number of events	Outage duration (h)
BWR plants (25)	180	22,456	69	38,023	249	60,479
Average per BWR plant	7	898	3	1,521	10	2,419
PWR plants (41)	336	59,298	102	70,614	438	129,912
Average per PWR plant	8	1,446	2	1,722	11	3,169
All plants (66)	516	81,754	171	108,637	687	190,391
Average per plant	8	1,239	3	1,646	11	2,885
Average outage duration per plant		158		635		277

[a]Includes data for Three Mile Island 2 through July 19. (The license for Three Mile Island 2 was suspended July 20, 1979.)

Table 3.4. Approximate Cause of Outages at Light-Water Reactor Units During 197[a]

Events	Forced outages						Scheduled outages					
	Equipment failure	Maintenance or test	Regulatory restrictions	Operational error	Administrative	Other	Maintenance or test	Refueling	Regulatory restrictions	Administrative	Other	Totals
BWRs												
Number of events	147		4	28	1	4	45	17	10			256
Hours of outage	16,083		4,828	1,150	5	390	3,106	24,659	10,258			60,479
PWRs												
Number of events	294	2	24	35	1	10	65	31	15	2	5	484
Hours of outage	28,388	8	30,046	567	70	219	15,090	45,430	8,857	56	1,181	129,912
All plants												
Number of events	441	2	28	63	2	14	110	48	25	2	5	740
Percent of total	60	<1	4	9	<1	2	15	6	3	<1	<1	100
All plants												
Total outage hours	44,471	8	34,874	1,717	75	609	18,196	70,089	19,115	56	1,181	190,391
Percent of total	23	<1	18	1	<1	<1	10	37	10	<1	<1	100

[a]The number of events includes those portions of or continuation of an outage attributable to causes other than the initial cause of the outage. Therefore, the number of events in this table exceeds the number of outages.

Table 3.5. BWR and PWR Outage Ratios (Outage Hours per 100 Hours of Commercial Operation)

Plant type	Refueling	Equipment failure	Maintenance or test	Regulatory restrictions	Operational error	Administrative	Training and licensing	Other	Totals
BWR	11.6	7.5	1.5	7.1	0.5			0.2	28.4[a]
PWR	12.8	8.0	4.3	11.0	0.2			0.4	36.7[a]

[a]Differs from total outage values given in Table 3.1 due to rounding off of numbers.

3.5 Systems and Components Associated with Plant Outages at LWRs

A graphic representation of plant outages is shown in Tables 3.6 and 3.7. These tables classify outages by type and identify the 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 these tables are listed in Appendix B.

The first four columns in each table are interrelated; e.g., Table 3.6 shows that the Duane Arnold plant accounted for 1733 h (3%) of the forced outage time associated with pipes and fittings in the reactor coolant system. The last column in Table 3.6, "Outage Cause," relates only to the first column, "Outage Type," and indicates, for example, that equipment failures accounted for 16,083 h of forced outage time experienced by all BWRs, and this also represents 27% of the total outage time experienced by all BWRs.

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

3.5.1 Boiling-water reactors

Forced outages. Forced outages accounted for 37% of the total outage time at BWRs in 1979. Equipment failures accounted for 27% of the time, while regulatory restrictions accounted for 8% and operator errors accounted for 2%. The major systems involved, each accounting for 9% of the time, were the reactor coolant system, the reactor system, and engineered safety features.

The components requiring the more significant amounts of time were pipes and pipe fittings — 6833 h; an inlet diffuser (at Big Rock Point) — 4847 h; transformers — 2600 h; and valves — 2085 h.

Scheduled outages. Scheduled outages at BWRs totaled 38,023 h, or 63% of the total BWR outage time. Refuelings accounted for 24,659 h, or 41%. Other activities such as maintenance were often 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. Regulatory restrictions accounted for 10,258 h (17%); however, of that amount, 8760 h were accumulated at one unit (Dresden 1) to upgrade the ECCS.

3.5.2 Pressurized-water reactors

Forced outages. Forced outages accounted for 46% of the total PWR outage time in 1979 (i.e., 59,298 of 129,912 h). Most of the forced outage time was devoted to the steam and power conversion system (30,177 h) and the engineered safety features (21,003 h). The dominant components were pipes and pipe fittings and instrumentation and controls. The significant amount of time devoted to these systems and components is a reflection of the restrictions imposed by the NRC as a result of the Three

Table 3.6. Boiling-Water-Reactor Plant Outages in 1979[a,b]

OUTAGE TYPE	ASSOCIATED SYSTEM	ASSOCIATED COMPONENT	PLANTS AFFECTED	OUTAGE CAUSE
FORCED OUTAGES 22,456h 3/%	REACTOR COOLANT 5671h 9%	PIPES, FITTINGS 2269h 4%	DUANE ARNOLD 1733h 3%	EQUIPMENT FAILURE 16,083h 27%
			536h VARIOUS 1%	
		VALVES 1710h 3%	785h OYSTER CREEK 1%	
			435h BRUNSWICK 1-220/HATCH 1-215 1%	
			490h LACROSSE-137/OTHERS-353 1%	
		792h PUMPS 1%	792h BR. FERRY 3-323/OTHERS-469 1%	
		900h I&C-702/OTHERS-198 1%	900h BRKSWK. 1 1-366/OTHERS-534 1%	
	REACTOR 5554h 9%	OTHER (INLET DIFFUSER) 4847h 8%	BIG ROCK POINT 4847h 8%	
		707h CONTROL ROD DRIVES 1%	707h LACROSSE-556/PILGRIM-154 1%	
	ENGINEERED SAFETY FEATURES 5430h 9%	PIPES, FITTINGS 4564h 8%	FITZPATRICK-4228 BRUNSWICK 2-336 4564h 8%	
		600h SHOCK SUPPRESSORS 1%	600h PILG. 1-437/Q. CITIES 1-163 1%	
		266h VALVES-175/OTHERS-91 <1%	266h VARIOUS <1%	
	ELECTRIC POWER 3184h 5%	TRANSFORMERS 2600h 4%	DRESDEN 3 2600h 4%	REGULATORY RESTRICTION 4828h 8%
		584h CONDUCTORS-343/OTHERS-241 1%	584h VARIOUS 1%	
	STEAM AND POWER 1908h 3%	534h MAIN GENERATORS 1%	534h HATCH 1-455/OTHERS-79 1%	
		568h TURBINES-345/I&C-223 1%	568h VARIOUS-345/VARIOUS-223 1%	
		806h VALVES-200/OTHERS-606 1%	806h VARIOUS 1%	OPERATOR ERROR 1150h 2%
	502h I&C 1%	502h I&C-414/OTHERS-88 1%	502h VARIOUS 1%	
	207h VARIOUS <1%	207h VARIOUS <1%	207h VARIOUS <1%	395h OTHER-390/ADMIN.-5 <1%
SCHEDULED OUTAGES 38,023h 63%	REACTOR 24,659h 41%	FUEL ELEMENTS 24,659h 41%	HATCH 1 3101h 5%	REFUELING 24,659h 41%
			NINE MILE POINT 1 2673h 4%	
			BROWNS FERRY 3 2554h 4%	
			BRUNSWICK 1 2256h 4%	
			BIG ROCK POINT 1773h 3%	
			LACROSSE 1578h 3%	
			BRUNSWICK 2 1530h 3%	
			MILLSTONE 1 1464h 2%	
			VERMONT YANKEE 1450h 2%	
			PEACH BOTTOM 3 1266h 2%	
			DRESDEN 2 1143h 2%	
			QUAD-CITIES 1 988h 2%	
			QUAD-CITIES 2 886h <2%	
			817h BROWNS FERRY 2 1%	
			707h COOPER 1%	
			471h BROWNS FERRY 1 <1%	
	ENGINEERED SAFETY FEATURES 10970h 18%	OTHER (UPGRADING OF ECCS AT DRESDEN 1) 8760h >14%	DRESDEN 1 8760h >14%	REGULATORY RESTRICTIONS 10,258h 17%
		SHOCK SUPPRESSORS 1264h 2%	BRUNSWICK 1-588 BRUNSWICK 2-676 1264h 2%	
		784h VALVES 1%	748h VARIOUS 1%	
		198h I&C <1%	198h VARIOUS <1%	
	REACTOR COOLANT 1264h 2%	748h HT EXCH-385/PUMPS-363 1%	748h P. BOT. 3-385/OTHERS-363 1%	MAINTENANCE OR TEST 3106h 5%
		633h VALVES-329/OTHERS-304 1%	633h VARIOUS 1%	
	4826h RADWASTE 1%	482h RECOMBINERS 1%	482h P. BOT. 2-251/P. BOT. 3-231 1%	
	531h VARIOUS 1%	531h VARIOUS 1%	531h VARIOUS 1%	

[a] BWR plant outages totaled 60,479 h (100%).

[b] Abbreviations used in Table 3.6:

admin. = administrative
Br. Ferry 3 = Browns Ferry 3
Brnswk. 1 = Brunswick 1
ht. exch. = heat exchanger
I&C = instrumentation and controls
P. Bot. = Peach Bottom
Pilg. 1 = Pilgrim 1
Q. Cities 1 = Quad Cities 1

Table 3.7. Pressurized-Water-Reactor Plant Outages in 1979[a,b]

OUTAGE TYPE	ASSOCIATED SYSTEM	ASSOCIATED COMPONENT	PLANTS AFFECTED	OUTAGE CAUSE
FORCED OUTAGE	STEAM AND POWER CONVERSION	PIPES, FITTINGS	1585h ZION 2 1%	REGULATORY RESTRICTIONS
			1350h COOK 1 1%	
			1290h MILLSTONE 2 1%	
			1077h COOK 2 1%	
			980h ZION 1 1%	
		8155h 6%	1873h VARIOUS 1%	
		INSTRUMENTATION AND CONTROLS	THREE MILE ISLAND 1 6692h 5%	
		7642h 6%	950h VARIOUS 1%	
		TURBINES	SALEM 1 2389h 2%	
			1670h ARKANSAS 1 1%	
		5722h 4%	1663h CALV. CLIFFS 1-593/OTHERS-770 1%	
		PUMPS	THREE MILE ISLAND 2 2730h 2%	
		3482h 3%	752h OCONEE 2-364/OTHERS-388 1%	
		HEAT EXCHANGERS	1177h CRYST. RIVER-3/PR. ISL. 1-497 1%	
		2866h 2%	1689h GINNA-430/OTHERS-1259 1%	
		1435h VALVES-750/NA-685 1%	1435h VARIOUS 1%	
	30,177h 23%	875h GENR.-403/OTHERS-472 1%	875h VARIOUS 1%	30,046h 23%
	ENGINEERED SAFETY FEATURES	PIPES, FITTINGS	SURRY 1 5373h 4%	EQUIPMENT FAILURE
			BEAVER VALLEY 1 3857h 3%	
			1981h MAINE YANKEE 2%	
		11,501h 9%	290h SAN ONFORE-234/RANCHO SECO-56 <1%	
		SHOCK SUPPRESSORS	FARLEY 1 3833h 3%	
			OCONEE 3 2771h 2%	
		8062h 6%	1458h SALEM 1-720/OTHERS-738 1%	
	21,003h 16%	1440h VARIOUS 1%	1440h SALEM 1-664/OTHERS-776 1%	
	REACTOR COOLANT	1703h PUMPS 1%	1703h CRYST. RIVER 3-1007/OTHERS-696 1%	
		1050h VALVES 1%	1050h TMI-2-413/OTHERS-481 1%	
		1134h CKT. CLSR.-630/PIPE,FTG.-504 1%	134h DAVIS-BESSE 1-630/OTHERS-504 1%	
	5091h 4%	1204h VARIOUS 1%	1204h VARIOUS 1%	
	1037h ELECTRIC POWER 1%	1037h VARIOUS 1%	1037h VARIOUS 1%	
	944h AUX. WATER 1%	944h VALVE OPR.-783/OTHERS-161 1%	944h NORTH ANNA 1-783/OTHERS-161 1%	28,388h 22%
59,298h 46%	1046h I&C-529/OTHERS-517 1%	1046h VARIOUS 1%	1046h VARIOUS 1%	864h OPR. ERROR-567/OTHERS-297 1%
SCHEDULED OUTAGE	REACTOR	FUEL ELEMENTS	PALISADES 2755h 2%	REFUELING
			TURKEY POINT 3 2742h 2%	
			INDIAN POINT 3 2594h 2%	
			NORTH ANNA 1 2344h 2%	
			ROBINSON 2 2277h 2%	
			2185h INDIAN POINT 2 2%	
			2115h SALEM 1 2%	
			1866h TURKEY POINT 4 1%	
			1856h FARLEY 1 1%	
			1763h MILLSTONE 2 1%	
			1754h COOK 2 1%	
			1644h ST. LUCIE 1 1%	
			1570h CALVERT CLIFFS 1 1%	
			1492h OCONEE 3 1%	
			1466h COOK 1 1%	
			1395h CRYSTAL RIVER 3 1%	
			1393h ARKANSAS 1 1%	
			1376h POINT BEACH 1 1%	
			1262h GINNA 1%	
			1101h ZION 1 1%	
			1068h CALVERT CLIFFS 2 1%	
			1064h HADDAM NECK 1%	
			1000h SURRY 2 1%	
			962h OCONEE 1 1%	
			954h ZION 2 1%	
			940h THREE MILE ISLAND 1 1%	
			769h KEWAUNEE <1%	
			744h BEAVER VALLEY 1 <1%	
			724h PRAIRIE ISLAND 1 <1%	
	45,681h 35%	45,681h 35%	506h POINT BEACH 2 <1%	45,430h 35%
	STEAM AND POWER CONVERSION	HEAT EXCHANGERS	SURRY 2 6941h 5%	MAINTENANCE OR TEST
			1321h GINNA-690/PT. BEACH 1-631 1%	
		9138h 7%	876h SAN ONOFRE-402/OTHERS-474 1%	
		INSTRUMENTATION AND CONTROLS	1728h DAVIS-BESSE 1 1%	
		3266h >2%	1538h RANCHO SECO-1536/ARKANSAS 1-2 1%	
		PIPES, FITTINGS	1199h MAINE YANKEE-709/PT. BEACH 2-490 1%	
		2510h 2%	1311h YANKEE-ROWE-468/OTHERS-843 1%	15,090h <12%
		1820h PUMPS 1%	1820h TURKEY PT. 3-872/OTHERS-1128 1%	REGULATORY RESTRICTIONS
		1000h NA 1%	1000h TROJAN (power not needed) 1%	
	18,849h >14%	1115h VALVES-912/OTHERS-203 1%	1115h DAVIS-BESSE 1-745/OTHERS-370 1%	
	ENGINEERED SAFETY FEATURES	SHOCK SUPPRESSORS	1928h TROJAN >1%	
		2437h 2%	509h YANKEE-ROWE-468/OTHERS-41 <1%	
	3787h 3%	1350h PRES.VES.-608/OTHERS-742 1%	1350h TROJAN-608/OTHERS-742 1%	
	REACTOR COOLANT-2080	1704h PUMPS >1%	1704h DAVIS-BESSE 1-749/OTHERS-955 >1%	8857h <7%
70,614h 54%	2297h OTHERS-217 2%	593h VALVES-336/OTHERS-257 <1%	593h VARIOUS <1%	1237h OTHER-1181/ADMIN-56 <1%

[a]PWR plant outages totaled 129,912 h (100%).

[b]Abbreviations used in Table 3.7:

admin. = administrative
aux. = auxiliary
Calv. Cliffs = Calvert Cliffs
ckt. clsr. = circuit closure
Cryst. River = Crystal River
ftg. = fittings
genr. = generator
NA = not applicable
opr. = operator
Pt. Beach = Point Beach
Pr. Isl. = Prairie Island
pres. ves. = pressure vessel

Mile Island 2 accident and the concern about seismic design deficiencies in safety-related piping.

Regulatory restrictions accounted for 30,046 h, which is an increase of 24,909 h over 1978. Equipment failures accounted for 28,388 h in 1979, an increase of 2506 h over 1978.

Scheduled outages. Scheduled outages in PWRs totaled 70,614 h, or 54% of the total PWR outage time. The reactor system accounted for 45,681 h, of which 45,430 h was for refueling. Maintenance or testing, accounting for 15,090 h, decreased slightly from the 1978 total of 15,694 h. In 1978 there was no scheduled outage time for regulatory restrictions, but in 1979 such restrictions accounted for 8857 h of scheduled outage time.

3.5.3 Comments on BWR and PWR outages

Forced outages. Twenty-five BWR plants experienced 22,456 h of forced outage — an overall average of 898 h per plant. Forty-one PWR plants experienced 59,298 h of forced outage — an overall average of 1446 h per plant.

Additional insight into the outages at BWRs and PWRs may be obtained by reviewing the data in Table 3.8 which compares the percentages of forced outage time and the average number of hours per plant for the listed components, which contributed 1% or more of the total outage time.

Excluding the inlet diffuser at Big Rock Point, the component that contributed the most to forced outage time at BWRs was piping, accounting for 274 h per plant. At PWRs piping accounted for 480 h per plant, and shock suppressors accounted for 197 h per plant.

Scheduled outages. The 25 BWRs had 38,023 h of scheduled outage time for an average of 1521 h per plant. The 41 PWRs accumulated 70,614 h for an average of 1722 h per plant. The scheduled outages in the two types of reactors are compared in Table 3.9 on the basis of percentage of outage time and average number of hours per plant for the listed components of either reactor type, which contributed 1% or more of the total outage time.

Fuel elements, the components involved in refueling, accounted for more outage time than the other components at both types of reactors. The average outage time due to fuel elements at PWRs was slightly greater than that at BWRs, averaging ~128 h longer. Aside from fuel elements, shock suppressors were the components commanding the most scheduled outage time at BWRs, excluding the outage for upgrading the ECCS at Dresden 1. At PWRs heat exchangers ranked second behind fuel elements, requiring 223 h per plant.

3.5.4 HTGR outage experience summary

Fort St. Vrain began commercial operation on July 1, 1979, and therefore accumulated enough operating experience to be included in this report. The total time the unit was in commercial operation during 1979 was 4417 h. The unit generated 123,584 MWh net. At the end of the year,

Table 3.8. Components Involved in Forced Outages

System	Component	BWR		PWR	
		%	Av. hours per plant	%	Av. hours per plant
Steam and power	Pipes and/or fittings			6	199
	Instrumentation			6	186
	Turbines			4	140
	Pumps			3	85
	Heat exchanger			2	70
	Main generator	1	21		
Engineered safety features	Pipes, fittings	8	183	9	281
	Shock suppressors	1	24	6	197
Reactor coolant	Pumps	1	32	1	42
	Valves	3	68	1	26
	Pipes and/or fittings	4	91		
Reactor	Inlet diffuser	8	194		
	Control rod drives	1	28		
Electric power	Transformers	4	104		

Table 3.9. Components Involved in Scheduled Outages

System	Component	BWR		PWR	
		%	Av. hours per plant	%	Av. hours per plant
Reactor	Fuel elements	41	986	35	1114
Steam and power	Heat exchangers			7	223
	Instrumentation			2	80
	Pipes and/or fittings			2	61
	Pumps			1	44
Engineered safety features	Shock suppressors	2	51	2	59
	Other (Dresden 1 upgrade)	14	350		
	Valves	1	30		
Reactor coolant	Pumps			1	42
Radioactive waste management	Recombiners	1	19		

it had an availability factor of 22.2% and a unit capacity factor of 8.5% for both MDC and DER.

The unit experienced eight forced outages, accounting for 29.2% of the commercial operating period, and three scheduled outages, accounting for 48.6% of the commercial operating period. (Further details of Fort St. Vrain's outage experience are contained in the individual plant data sheets in Appendix B.)

3.5.5 Summary

During 1979, the 25 operating BWRs experienced an average of 2419 h of outage time compared to an average of 3169 h for the 41 operating PWRs. The percentage of forced outage time at BWRs was 37% compared to 46% at PWRs. The primary cause of forced outages at BWRs was equipment failure. At PWRs the primary cause of forced outages was regulatory restrictions. Refueling was the primary reason for scheduled outages at both BWRs and PWRs. Regulatory restrictions and maintenance or testing accounted for large percentages of the scheduled outage time at both types of plants.

The dominance of regulatory restrictions as the cause of large percentages of forced and scheduled outages was the result of action taken by the NRC with regard to certain aspects of the Three Mile Island 2 accident and with regard to concern for seismic design deficiencies in safety-related piping.

Fort St. Vrain, an HTGR, began commercial operation July 1, 1979. For the remainder of the year, the unit acquired an availability factor of 22.2%, having experienced eight forced outages and three scheduled outages for a total outage time of 3434 h.

4. REPORTABLE OCCURRENCES

4.1 Introduction

The NRC collects and evaluates operational and environmental information concerning licensed nuclear facilities. Incidents or events that occur 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 if necessary. The technical specifications for each plant include a section on reporting requirements, detailing the types of operational and environmental events that must be reported. The NRC Regulatory Guides are used as guidelines for an acceptable reporting program, but they are not substitutes for the plant's technical specifications with which compliance is mandatory. The NRC is undergoing a program to standardize technical specifications, including reporting requirements. Standardization was not completed during the period covered by this report; thus, the plants reviewed herein operated under reporting requirements that varied from plant to plant. It would be inappropriate, therefore, to compare the performance of plants only on the basis of the number of reports submitted.

Data from these reports are stored in the NRC's Licensee Event Report (LER) file for further analysis and evaluation, and for public dissemination. The information reported in the LERs conveys, primarily, negative aspects of plant operations. Extensive knowledge of normal operations, which is the situation most of the time, is needed to put these events in proper perspective. A large number of events of one type may not be significant to safety, whereas a single event of another type 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.

The LERs from which the data are taken may be reviewed at the NRC's Public Document Room. (All reports required by the NRC are filed in the NRC's Public Document Room located at 1717 H Street, N.W., Washington, D.C. Documents relevant to individual power plants are also available at local Public Document Rooms located in the vicinity of each plant.) Computer printouts summarizing reportable occurrences are filed in the NRC's Public Document Room in Washington, D.C., and in all local Public Document Rooms on a biweekly schedule. In addition, the Nuclear Safety Information Center (NSIC), located at Oak Ridge National Laboratory, also maintains a computerized data base of LERs. Although the structure and application of NSIC's data base differ from the NRC's, it is also used for analysis and evaluations conducted for the purpose of enhancing nuclear power plant performance and safety.

4.2 Licensee Event Reports

4.2.1 Introduction

LERs are used to form the basis for comparing performance with design intent and to assess the safety aspect of operation. They include reports of incidents or events that involve system, component, or structural failure; malfunctions; personnel errors; design deficiencies; management deficiencies; and other matters that are related to plant operational safety.

Because nuclear power plant designs employ multiple levels of protection, or defense-in-depth, including the provision of redundant safety systems and components, LER events do not, in general, affect safety directly, nor do they have an actual impact or consequence on the health and safety of the public. However, the information reported in LERs is useful for enhancing the safe operation of the plants.

4.2.2 Reporting requirements

Plant technical specifications include a section on reporting requirements detailing the types of events that should be reported (1) as promptly as possible (within 24 h, with written follow-up within 14 d) or (2) within 30 d. Reporting requirements may be 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 set point 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 calculated 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 s 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 Safety Analysis Report.
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 Safety Analysis Report.
7. Conditions arising from natural or man-made 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 redundance 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.

As a result of action taken by the NRC staff following the accident at Three Mile Island on March 28, 1979, a new rule was published requiring the immediate reporting of significant events by telephone. The purpose of the new rule is to ensure the timely and accurate flow of information from licensees of operating nuclear power reactors following a significant event.[1]

The rule was published in Title 10 of the *Code of Federal Regulations*, Part 50 as Sect. 50.72 and became effective Feb. 29, 1980. Section 50.72 requires licensees to notify the NRC Operations Center as soon as possible and in all cases within 1 h by telephone of the occurrence of any significant event listed in the section. The 12 significant events requiring immediate reporting are:

1. Any event requiring initiation of the licensee's emergency plan or any section of that plan.

2. The exceeding of any technical specification safety limit.

3. Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shut down.

4. Any act that threatens the safety of the nuclear power plant or site personnel, or the security of special nuclear material, including instances of sabotage or attempted sabotage.

5. Any event requiring initiation of shutdown of the nuclear power plant in accordance with technical specification limiting conditions for operation.

6. Personnel error or procedural inadequacy which, during normal operations, anticipated operational occurrences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety function of those structures, systems, and components important to safety that are needed to (a) shut down the reactor safely and maintain it in a safe shutdown condition, or (b) remove residual heat following reactor shutdown, or (c) limit the release of radioactive material to acceptable levels or reduce the potential for such release.

7. Any event resulting in manual or automatic actuation of engineered safety features, including the reactor protection system.

8. Any accidental, unplanned, or uncontrolled radioactive release. (Normal or expected releases from maintenance or other operational activities are not included.)

9. Any fatality or serious injury occurring on the site and requiring transport to an offsite medical facility for treatment.

10. Any serious radioactive contamination of personnel requiring extensive onsite decontamination or outside assistance.

11. Any event meeting the criteria of 10 CFR 20.403 for notification.

12. Strikes of operating employees or security guards, or honoring of picket lines by these employees.

4.2.3 Licensee Event Reports submitted to the NRC in 1979

Introduction. Data taken from the LER file maintained by the NRC have been tabulated (1) to relate the number of LERs submitted during the year to (a) the nuclear plant and system in which the event occurred, (b) the component involved in the event, (c) the cause of the event, (d) the method of discovery of the event, and (e) the status of the reactor at the time the event occurred; and (2) to relate the number of LERs involving personnel errors to the system affected or involved. Tables 4.1 through 4.8 present the data for BWR and PWR plants only. The data for the single HTGR (Fort St. Vrain) are presented separately in Sect. 4.2.4.

The systems, subsystems, and component types used to categorize the LERs are listed in Appendix B.

The 66 LWR plants considered for review in this report submitted 2828 LERs during 1979, an increase of 147 from the 2681 submitted in 1978. The 25 BWRs submitted 1219, while the 41 PWRs submitted 1609. (Fort St. Vrain, the HTGR unit, submitted 46 LERs during the year.)

Systems involved in the reportable occurrences. In Table 4.1, the number of LERs submitted by individual BWR plants is related to the systems involved. Table 4.2 presents the same data for PWR plants. Table 4.3 summarizes the data from Tables 4.1 and 4.2 to show the relative involvement of the various systems in reportable occurrences. Note that engineered safety features were involved in more reportable occurrences than any other system at both BWRs and PWRs; instrumentation and controls and the reactor coolant system were also involved in a large number of reportable occurrences. This is not unusual, since these two systems

Table 4.1. BWR Plant LERs vs System

	Reactor	Reactor coolant and connected systems	Engineered safety features	Instrumentation and controls	Electric power systems	Fuel storage and handling	Auxiliary water systems	Auxiliary process systems	Other auxiliary systems	Steam and power conversion systems	Radioactive waste management systems	Radiation protection systems	Other systems	System code not applicable[a]	Totals	Percent of total number of LERs (1219)
Big Rock Point	2	2	19	1	3	0	0	0	1	0	0	0	0	0	28	2.3
Browns Ferry 1	0	6	9	4	2	0	1	0	3	0	2	2	0	1	30	2.5
Browns Ferry 2	1	5	12	6	1	0	0	0	1	0	0	1	0	0	27	2.2
Browns Ferry 3	1	6	7	5	2	0	0	0	4	0	1	0	0	0	26	2.1
Brunswick 1	10	19	54	21	2	0	2	1	1	2	1	0	1	1	115	9.4
Brunswick 2	10	23	44	11	3	0	2	2	1	7	1	0	0	0	104	8.5
Cooper Station	0	7	15	4	6	0	1	0	5	0	0	1	0	0	39	3.2
Dresden 1	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1	0.1
Dresden 2	1	8	23	5	13	1	0	0	3	0	5	0	0	0	59	4.8
Dresden 3	3	8	7	6	1	0	0	0	0	1	4	0	0	0	30	2.5
Duane Arnold	3	9	13	6	2	0	0	0	0	0	1	1	0	1	36	3.0
FitzPatrick 1	10	15	27	11	5	1	5	2	13	0	2	2	5	11	109	8.9
Hatch 1	5	27	23	11	5	1	5	4	6	0	6	0	0	8	101	8.3
Hatch 2	8	30	34	23	7	0	8	0	4	0	3	2	1	4	124	10.2
Humboldt Bay	0	0	1	0	0	0	0	0	1	0	0	1	0	0	3	0.2
La Crosse	3	0	8	2	1	0	0	0	0	2	1	0	0	0	17	1.4
Millstone 1	4	6	16	6	2	0	0	1	1	0	0	0	0	0	36	3.0
Monticello	1	11	3	3	1	0	0	0	0	2	0	0	0	0	21	1.7
Nine Mile Point 1	0	2	9	2	3	0	0	0	0	0	0	0	0	6	22	1.8
Oyster Creek	3	7	25	3	1	0	0	0	0	0	3	0	0	0	42	3.4
Peach Bottom 2	5	4	24	2	2	0	1	0	9	0	4	1	1	0	53	4.3
Peach Bottom 3	1	9	20	6	1	0	0	1	0	0	1	0	0	0	39	3.2
Pilgrim	4	12	13	2	6	0	3	0	1	0	5	0	1	1	48	3.9
Quad Cities 1	2	10	13	4	4	0	0	1	1	0	2	1	0	1	39	3.2
Quad Cities 2	0	10	16	5	3	0	0	0	0	0	1	1	0	0	36	3.0
Vermont Yankee	3	10	8	5	1	0	1	0	0	0	2	0	1	3	34	2.8
Totals	80	245	443	154	77	3	29	12	55	14	46	13	10	37	1219	99.9[b]
Percent of 1219	6.6	20.2	36.3	12.6	6.3	0.2	2.4	1.0	4.5	1.1	3.8	1.1	0.8	3.0	99.9[b]	

[a] Indicates an operational error or procedural deficiency rather than a failure of a system.

[b] Totals do not equal 100% because of rounding numbers to the nearest tenth of a percent.

Table 4.2. PWR Plant LERs vs System

	Reactor	Reactor coolant and connected systems	Engineered safety features	Instrumentation and controls	Electric power systems	Fuel storage and handling	Auxiliary water systems	Auxiliary process systems	Other auxiliary systems	Steam and power conversion systems	Radioactive waste management systems	Radiation protection systems	Other systems	System code not applicable[a]	Totals	Percent of total number of LERs (1609)
Arkansas 1	0	8	5	0	3	0	1	2	1	0	0	0	0	0	20	1.2
Arkansas 2	4	2	9	28	6	0	12	3	5	1	0	1	2	0	73	4.5
Beaver Valley 1	3	5	10	6	9	0	0	0	0	1	0	0	1	1	36	2.2
Calvert Cliffs 1	4	3	9	13	13	0	9	7	4	1	1	3	0	2	69	4.3
Calvert Cliffs 2	11	4	5	14	5	0	1	1	1	3	2	2	0	0	49	3.0
Cook 1	3	5	14	23	6	0	2	0	3	0	1	2	2	4	65	4.0
Cook 2	0	7	13	15	4	1	3	2	0	1	0	3	2	4	55	3.4
Crystal River 3	5	19	20	12	5	0	0	11	11	3	1	6	1	8	102	6.3
Davis-Besse 1	11	28	21	22	11	0	3	6	17	0	1	6	3	1	130	8.1
Fort Calhoun 1	0	2	9	7	1	0	1	0	1	0	0	0	0	0	21	1.3
Ginna	0	6	5	2	2	0	0	7	0	2	0	0	0	1	25	1.6
Haddam Neck	0	3	1	1	2	0	1	1	1	0	0	0	0	0	10	0.6
Indian Point 2	0	3	4	2	0	0	1	4	0	3	1	0	0	0	18	1.1
Indian Point 3	1	3	1	2	1	0	7	2	0	0	0	0	1	1	19	1.2
Farley 1	5	4	17	8	9	0	3	2	3	0	0	5	0	7	63	3.9
Kewaunee	0	3	7	6	6	0	1	0	1	1	0	0	1	0	26	1.6
Maine Yankee	0	3	6	12	1	0	0	0	0	0	3	0	1	1	27	1.7
Millstone 2	1	4	7	14	4	0	2	1	0	1	3	0	0	0	37	2.3
North Anna 1	21	12	32	28	10	1	5	4	15	13	5	3	1	0	150	9.3
Oconee 1	1	2	6	7	4	0	4	0	3	5	1	3	0	0	36	2.2
Oconee 2	0	1	5	2	0	0	0	0	0	0	0	1	0	0	9	0.6
Oconee 3	1	2	3	3	1	0	1	0	0	1	0	1	0	0	13	0.8
Palisades	2	1	12	5	3	0	2	3	1	11	2	0	0	1	43	2.7
Point Beach 1	0	8	4	3	3	0	0	2	1	0	1	0	0	0	22	1.4
Point Beach 2	0	4	1	1	1	0	0	1	0	0	0	0	0	0	8	0.5
Prairie Island 1	2	3	5	3	2	0	3	1	2	1	0	0	0	2	24	1.5
Prairie Island 2	1	2	0	3	1	0	1	0	0	0	0	0	1	0	9	0.6
Rancho Seco	0	9	3	0	5	0	1	1	0	2	0	0	0	1	22	1.4
Robinson 2	1	7	9	1	2	0	2	2	6	1	0	0	0	2	33	2.1
Salem 1	5	12	14	12	4	0	3	1	7	0	2	0	3	9	72	4.5
San Onofre 1	0	6	4	2	3	0	1	0	0	0	1	1	0	2	20	1.2
St. Lucie 1	12	0	3	6	7	0	1	2	1	4	0	0	0	0	36	2.2
Surry 1	1	3	12	1	4	0	4	0	3	3	6	2	0	5	44	2.7
Surry 2	0	0	4	1	0	0	3	0	0	1	0	0	0	2	11	0.7
Three Mile Island 1	0	5	6	0	1	1	1	0	0	0	0	0	0	2	16	1.0
Three Mile Island 2	3	3	2	1	2	0	2	0	1	0	0	0	0	0	14	0.9
Trojan	0	4	5	3	2	0	0	0	0	0	0	0	0	0	14	0.9
Turkey Point 3	1	1	6	1	3	0	3	3	3	11	1	1	0	1	35	2.2
Turkey Point 4	0	2	0	3	0	1	1	8	0	0	0	1	0	0	16	1.0
Yankee Rowe	6	3	9	1	2	0	0	0	0	1	1	5	0	0	28	1.7
Zion 1	1	4	7	14	4	0	0	1	0	1	13	7	0	5	57	3.5
Zion 2	0	2	5	13	5	0	0	1	0	1	3	1	0	1	32	2.0
Totals	106	208	320	301	157	4	85	79	91	73	49	54	19	63	1609	100
Percent of 1609	6.6	12.9	19.9	18.7	9.8	0.2	5.3	4.9	5.7	4.5	3.0	3.4	1.2	3.9	100	

[a] Indicates an operational error or procedural deficiency rather than a failure of a system.

Table 4.3. LWR Systems Reported in LERs for 1979[a]

System	BWRs		PWRs	
	No. of reports	% of total reports	No. of reports	% of total reports
Reactor	80	6.6	106	6.6
Reactor coolant and connected systems	246	20.2	208	12.9
Engineered safety features	443	36.3	320	19.9
Instrumentation and controls	154	12.6	301	18.7
Electric power systems	77	6.3	157	9.8
Fuel storage and handling	3	0.2	4	0.2
Auxiliary water systems	29	2.4	85	5.3
Auxiliary process systems	12	1.0	79	4.9
Other auxiliary systems	55	4.5	91	5.7
Steam and power conversion systems	14	1.1	73	4.5
Radioactive waste management systems	46	3.8	49	3.0
Radiation protection systems	13	1.1	54	3.4
Other systems	10	0.8	19	1.2
System code not applicable[b]	37	3.0	63	3.9
Totals	1219	99.9	1609	100.0

[a]Small numerical deviations are due to rounding off of numbers.

[b]Indicates an operational error or procedural deficiency rather than a failure of a system.

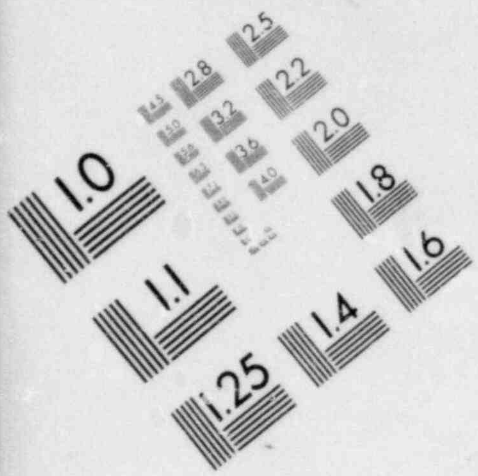
Table 4.4. Systems and Subsystems Involved in Light-Water-Reactor LERs for 1979[a]

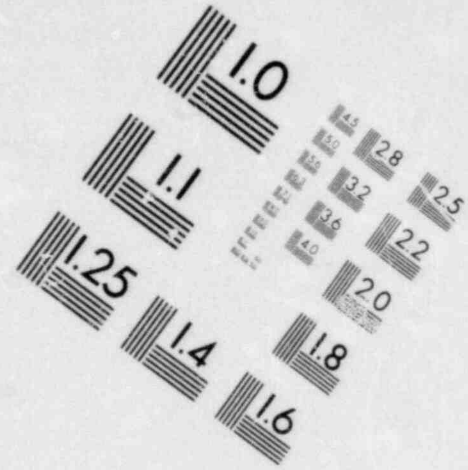
System and subsystem	BWRs		PWRs		Total	
	No. of reports	% of total reports	No. of reports	% of total reports	No. of reports	% of total reports
Reactor	80	6.6	106	6.6	186	6.6
Reactor vessel internals	2	0.2	0	0.0	2	0.1
Reactivity control systems	48	3.9	78	4.8	126	4.5
Reactor core	30	2.5	28	1.7	58	2.1
Reactor coolant system & connected systems	246	20.2	208	12.9	454	16.1
Reactor vessels & appurtenances	8	0.7	14	0.9	22	0.8
Coolant recirculation systems & controls	33	2.7	37	2.3	70	2.5
Main steam systems & controls	20	1.6	17	1.0	37	1.3
Main steam isolation systems & controls	42	3.4	8	0.5	50	1.8
Reactor core isolation cooling systems & controls	48	3.9	1	0.1	49	1.7
Residual heat removal systems & controls	49	4.0	40	2.5	89	3.1
Reactor coolant cleanup systems & controls	20	1.6	12	0.7	32	1.1
Feedwater systems & controls	12	1.0	60	3.7	72	2.5
Reactor coolant pressure boundary leakage detection systems	12	1.0	5	0.3	17	0.6
Other coolant subsystems & their controls	12	0.2	14	0.9	16	0.6
Engineered safety features	443	36.3	320	19.9	763	27.0
Reactor containment systems	78	6.4	39	2.4	117	4.1
Containment heat removal systems & controls	24	2.0	33	2.1	57	2.0
Containment air purification & cleanup systems & controls	19	1.6	17	1.1	36	1.3
Containment isolation systems & controls	81	6.6	73	4.5	154	5.4
Containment combustible control systems & controls	49	4.0	18	1.1	67	2.4
Emergency core-cooling systems & controls	151	12.4	100	6.2	251	8.9
Control room habitability systems & controls	4	0.3	14	0.9	18	0.6
Other engineered safety feature systems & their controls	37	3.0	26	1.6	63	2.2
Instrumentation and controls	154	12.6	301	18.7	455	16.1
Reactor trip systems	55	4.5	155	9.6	210	7.4
Engineered safety feature instrument systems	37	3.0	69	4.3	106	3.7
Systems required for safe shutdown	4	0.3	2	0.1	6	0.2
Safety-related display instrumentation	28	2.3	19	1.2	47	1.7
Other instrument systems required for safety	15	1.2	37	2.3	52	1.8
Other instrument systems not required for safety	15	1.2	19	1.2	34	1.2
Electric power systems	77	6.3	157	9.8	234	8.2
Offsite power systems & controls	3	0.2	11	0.7	14	0.5
AC onsite power systems & controls	7	0.6	24	1.5	31	1.1
DC onsite power systems & controls	9	0.7	10	0.6	19	0.7
Onsite power systems & controls (composite AC & DC)	4	0.3	11	0.7	15	0.5
Emergency generator systems & controls	52	4.3	99	6.2	151	5.3
Emergency lighting systems & controls	0	0.0	0	0.0	0	0.0
Other electric power systems & controls	2	0.2	2	0.1	4	0.1
Fuel storage and handling systems	3	0.2	4	0.2	7	0.2
New-fuel storage facilities	0	0.0	0	0.0	0	0.0
Spent-fuel storage facilities	0	0.0	1	0.1	1	<0.1
Spent-fuel-pool cooling & cleanup systems & controls	1	0.1	3	0.2	4	0.1
Fuel handling systems	2	0.1	0	0.0	2	0.1
Auxiliary water systems	29	2.4	85	5.3	114	4.0
Station service water systems & controls	20	1.6	32	2.0	52	1.8
Cooling systems for reactor auxiliaries & controls	4	0.3	33	2.0	37	1.3
Demineralized water makeup systems & controls	2	0.2	1	0.1	3	0.1
Potable & sanitary water systems & controls	0	0.0	0	0.0	0	0.0
Ultimate heat sink facilities	1	0.1	6	0.4	7	0.2
Condensate storage facilities	1	0.1	10	0.6	11	0.4
Other auxiliary water systems & their controls	1	0.1	3	0.2	4	0.1
Auxiliary process systems	12	1.0	79	4.9	91	3.2
Compressed air systems & controls	1	0.1	0	0.0	1	<0.1
Process sampling systems	3	0.2	2	0.1	5	0.2
Chemical, volume control, & liquid poison systems & controls	8	0.7	76	4.7	84	3.0
Failed-fuel detection systems	0	0.0	1	0.1	1	<0.1
Other auxiliary process systems & their controls	0	0.0	0	0.0	0	0.0
Other auxiliary systems	55	4.5	91	5.7	146	5.2
Air conditioning, heating, cooling & ventilation systems & controls	3	0.2	[illegible]	1.4	26	0.9
Fire protection systems & controls	52	4.3	63	3.9	115	4.1
Communication systems	0	0.0	1	0.1	1	<0.1
Other auxiliary systems & their controls	0	0.0	4	0.2	4	0.1
Steam and power conversion systems	14	1.1	73	4.6	87	3.1
Turbine-generators & controls	1	[illegible]	5	0.3	6	0.2
Main steam-supply system & controls	3	[illegible]	35	2.2	38	1.3
Main condenser systems & controls	0	0.0	0	0.0	0	0.0
Turbine-gland-sealing systems & controls	0	0.0	0	0.0	0	0.0
Turbine bypass systems & controls	2	0.2	2	0.1	4	0.1
Circulating water systems & controls	1	0.1	0	0.0	1	<0.1
Condensate cleanup systems & controls	3	0.2	1	0.1	4	0.1
Condensate and feedwater systems & controls	4	0.3	25	1.6	29	1.0
Steam generator blowdown systems & controls	0	0.0	4	0.2	4	0.1
Other features of steam & power conversion systems	0	0.0	1	0.1	1	<0.1
Radioactive waste management systems	46	3.8	49	3.0	95	3.3
Liquid radioactive waste management systems	11	0.9	9	0.5	20	0.7
Gaseous radioactive waste management systems	15	1.2	8	0.5	23	0.8
Process & effluent radiological monitoring systems	20	1.6	32	2.0	52	1.8
Solid radioactive waste management systems	0	0.0	0	0.0	0	0.0
Radiation protection systems	13	1.1	54	3.3	[illegible]	2.3
Area monitoring systems	2	0.2	10	0.6	12	0.4
Airborne radioactivity monitoring systems	11	0.9	44	2.7	55	1.9
Other systems	10	0.8	19	1.2	29	1.0
System code not applicable[b]	37	3.0	63	[illegible]	100	3.5
Totals	1219	99.9	1609	100	2828	99.8

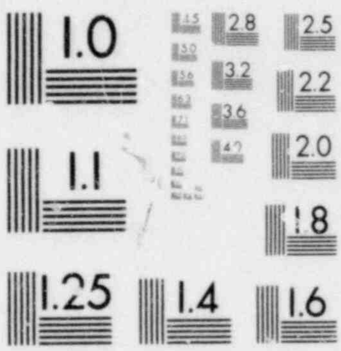
[a] Small numerical deviations are due to rounding off of numbers.

[b] Indicates an operational error or procedural deficiency rather than a failure of a system or subsystem.

IMAGE EVALUATION TEST TARGET (MT-3)







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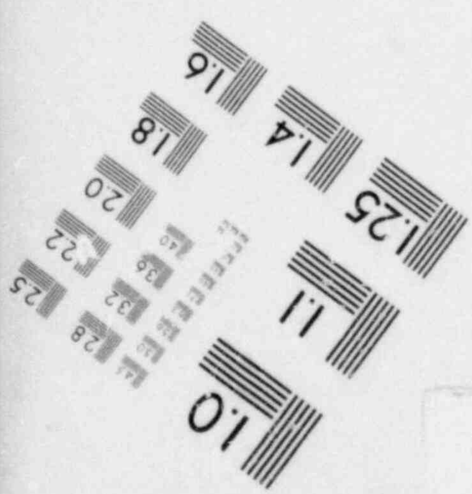

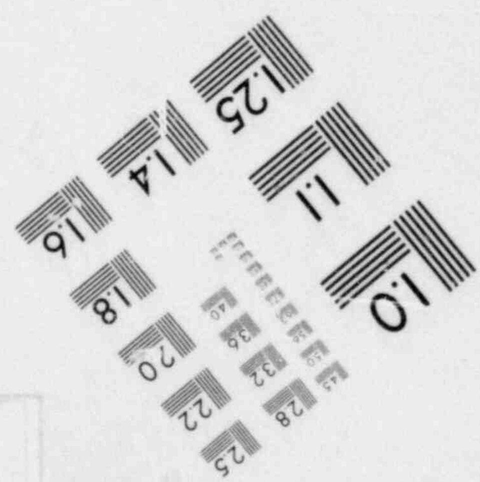

IMAGE EVALUATION
TEST TARGET (MT-3)

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Table 4.5. LWR Components Reported in LERs for 1979

Component	BWRs		PWRs	
	No. of reports	% of total reports	No. of reports	% of total reports
Accumulators	1	0.1	16	1.0
Air dryers	1	0.1	0	0
Annunciator modules	6	0.5	1	0.1
Batteries and chargers	8	0.7	10	0.6
Blowers	1	0.1	23	1.4
Circuit closers/interrupters	36	3.0	52	3.2
Component code not applicable[a]	154	12.6	204	12.7
Control rod drive mechanisms	5	0.4	21	1.3
Control rods	5	0.4	8	0.5
Demineralizers	1	0.1	1	0.1
Electrical conductors	18	1.5	22	1.4
Engines, internal combustion	24	2.0	45	2.8
Filters	7	0.6	8	0.5
Fuel elements	11	0.9	20	1.2
Generators	6	0.5	12	0.7
Hangers, supports, shock suppressors	93	7.6	106	6.6
Heat exchangers	9	0.7	43	2.7
Heaters, electric	0	0	11	0.7
Instrumentation and controls	393	32.2	416	25.9
Mechanical function units	7	0.6	10	0.6
Motors	8	0.7	17	1.1
Other components	31	2.5	33	2.1
Penetrations, primary containment	19	1.6	34	2.1
Pipes and/or fittings	44	3.6	62	3.9
Pumps	43	3.5	85	5.3
Recombiners	5	0.4	3	0.2
Relays	42	3.4	46	2.9
Transformers	2	0.2	5	0.3
Turbines	6	0.5	4	0.2
Valve operators	76	6.2	79	4.9
Valves	148	12.1	206	12.8
Vessels, pressure	9	0.7	6	0.4
Total	1219	100	1609	100.2

[a]Indicates an operational error or procedural deficiency rather than a component failure.

Table 4.6. LERs Submitted by Light-Water-Reactor Plants in 1979 Arranged by Cause, Method of Discovery, and Reactor Status at Time of Occurrence

	BWRs		PWRs		BWRs and PWRs	
	No. of reports	% of BWR reports	No. of reports	% of PWR reports	Total reports	% of total reports
Approximate cause						
Component failure	634	52.0	811	50.4	1445	51.1
Defective procedures	62	5.1	84	5.2	146	5.2
Design/fabrication error	180	14.8	245	15.2	425	15.0
External cause	9	0.7	8	0.5	17	0.6
Other	149	12.2	224	13.9	373	13.2
Personnel error	185	15.2	237	14.7	422	14.9
Totals	1219	100.0	1609	99.9[a]	2828	100.0
Method of discovery						
External source	92	7.5	134	8.3	226	8.0
Item not applicable	27	2.2	21	1.3	48	1.7
Observation/evaluation	2	0.2	0	0.0	2	0.1
Operational event	422	34.6	758	47.1	1180	41.7
Routine test or inspection	593	48.6	578	35.9	1171	41.4
Special dosimeter report	1	0.1	0	0.0	1	0.0
Special test or inspection	82	6.7	118	7.3	200	7.1
Totals	1219	100.0[a]	1609	99.9[a]	2828	100.0[a]
Reactor status at time of occurrence						
Construction	3	0.2	2	0.1	5	0.2
Item not applicable	3	0.2	5	0.3	8	0.3
Load change during power operation	36	3.0	39	2.4	75	2.7
Other	5	0.4	15	0.9	20	0.7
Preoperational startup, power ascension	36	3.0	79	4.9	115	4.1
Refueling	140	11.5	174	10.8	314	11.1
Routine shutdown operations	16	1.3	31	1.9	47	1.7
Routine startup operations	96	7.9	100	6.2	196	6.9
Shut down except for refueling	174	14.3	317	19.7	491	17.4
Steady-state power operation	710	58.2	846	52.6	1556	55.0
Undetermined	0	0.0	1	0.1	1	0.0
Totals	1219	100.0	1609	99.9[a]	2828	100.1[a]

[a] Numbers may not add up to 100.0% because of rounding errors.

Table 4.7. Personnel Errors vs System for Light-Water-Reactor Plants in 1979

System	BWR		PWR		BWRs and PWRs	
	No. of reports	% of BWR reports	No. of reports	% of PWR reports	Total reports	% of total reports
Reactor	24	13.0	16	6.8	40	9.5
Reactor coolant and connected systems	35	18.9	25	10.5	60	14.2
Engineered safety features	50	27.0	49	20.7	99	23.5
Instrumentation and controls	9	4.9	32	13.5	41	9.7
Electric power systems	16	8.6	26	11.0	42	10.0
Fuel storage and handling	1	0.5	3	1.3	4	0.9
Auxiliary water systems	3	1.6	10	4.2	13	3.1
Auxiliary process system	0	0.0	13	5.5	13	3.1
Other auxiliary systems	16	8.6	19	8.0	35	8.3
Steam and power conversion systems	1	0.5	3	1.3	4	0.9
Radioactive waste management systems	5	2.7	10	4.2	15	3.6
Radiation protection systems	3	1.6	5	2.1	8	1.9
Other systems	1	0.5	4	1.7	5	1.2
Not applicable	21	11.4	22	9.3	43	10.2
Totals	185	99.8[a]	237	100.1[a]	422	100.1[a]

[a]Small numerical deviations are due to rounding off of numbers.

Table 4.8. Personnel Errors at Light-Water-Reactor Plants for the Years 1969 through 1979

System	Number of personnel errors											System totals	% of system totals
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979		
Reactor	0	2	2	8	16	27	26	36	31	21	40	209	7.2
Reactor coolant and connected systems	2	4	9	16	34	39	73	61	85	56	60	439	15.2
Engineered safety features	1	5	11	16	42	80	104	96	115	118	99	687	23.8
Instrumentation and controls	0	1	0	6	20	31	28	40	63	60	41	290	10.1
Electric power systems	0	2	6	8	13	30	32	42	48	42	42	265	9.2
Fuel storage and handling	2	0	0	3	6	6	4	5	4	6	4	40	1.4
Auxiliary water systems	0	0	1	3	1	9	15	22	23	11	13	98	3.4
Auxiliary process systems	0	1	2	2	12	19	16	19	19	23	13	126	4.4
Other auxiliary systems	0	0	0	0	0	3	3	5	8	33	35	87	3.0
Steam power and conversion systems	0	0	3	9	13	26	18	11	20	13	4	117	4.1
Radioactive waste management systems	0	2	6	7	17	40	46	28	29	11	15	201	7.0
Radiation protection system	0	0	0	0	1	2	3	7	8	14	8	43	1.5
Other systems	0	0	0	0	3	1	2	6	14	18	5	49	1.7
System code not applicable[a]	1	2	2	2	8	3	27	42	53	51	43	234	8.1
Totals (by year)	6	19	42	80	186	316	397	420	520	477	422	2885	100.1[b]
Percent of 11-year total	0.2	0.7	1.5	2.8	6.4	11.0	13.8	14.6	18.0	16.5	14.6	100.1	

[a] Primarily occurrences in which operating personnel failed to perform surveillance tests within a specified time interval.

[b] Numerical deviation due to rounding off of numbers.

and the electric power system are the dominant systems with respect to their extent and importance to safety. Table 4.4 presents a further breakdown of the data to indicate the subsystems involved in the reportable occurrences. As expected, the emergency core-cooling system was involved in a larger number of occurrences, indicating the importance of this system and the attention it consequently receives.

Components involved in the reportable occurrences. Table 4.5 presents data on the components involved in the reportable occurrences. Instrumentation and valves were reported as being involved in more occurrences than the other components; this is to be expected because of the large number of these components in a plant. There were a large number of reports for "component code not applicable"; this item indicates an operational error or a procedural deficiency rather than a component failure.

Cause, method of discovery, and reactor status. Table 4.6 presents data on the cause, method of discovery, and reactor status at the time of the reportable occurrence. Component failures accounted for more than half of the occurrences. Personnel error was the cause of 14.9% of the occurrences in 1979, dropping from 18% in 1978.

Personnel errors. Table 4.7 gives the personnel errors that occurred and the systems involved. Again, the largest number of errors made involved the most extensive and important systems — that is, engineered safety features, reactor coolant system, electric power system, and instrumentation and controls. Table 4.8 presents an historical accounting of personnel errors vs system. The smaller numbers in the earlier years (1969–1973) merely reflect the fact that there were fewer units reporting occurrences during that period. The steady decline in personnel errors since 1977 probably is a reflection of the greater effort made in training programs. The errors listed for "system code not applicable" (8.1%) are primarily occurrences in which operating personnel failed to perform surveillance tests within a specified time interval.

4.2.4 HTGR (Fort St. Vrain) Licensee Event Reports

The only commercial HTGR in operation (Fort St. Vrain) submitted 46 LERs in 1979. The number of LERs vs the system involved in the reported occurrences was as follows:

System	No. of LERs	% of total
Reactor coolant	20	43.5
Electric power	5	10.9
Steam and power conversion	5	10.9
System code not applicable	5	10.9
Other auxiliary systems	3	6.5
Engineered safety features	2	4.3
Fuel storage and handling	2	4.3
Various	4	8.7
Total	46	100.0

The number of LERs vs the components involved were as follows:

Components	No. of LERs	% of total
Instrumentation and controls	12	26.1
Component code not applicable	9	19.6
Pipe hangers and shock suppressors	8	17.4
Valves	7	15.2
Various	10	21.7
Total	46	100.0

The causes for the reportable occurrences and the associated number of LERs were as follows:

Cause	No. of LERs	% of total
Component failure	20	43.5
Personnel errors	9	19.6
Design or fabrication error	5	10.9
Defective procedure	2	4.3
Other	10	21.7
Total	46	100.0

4.2.5 Operational events acted upon by the NRC

Licensee event reports are assessed by the NRC for their significance relative to safety and performance according to the design intent. Those events considered to be significant from the standpoint of public health and safety are reported to Congress quarterly (see Sect. 4.3). All events of possible significance to safety are reported to the (other) licensees (and other interested parties) for their information, and for corrective action and response if necessary. Three types of reports, distributed by the Office of Inspection and Enforcement of the NRC, are directed specifically to licensees: (1) I&E Information Notices, (2) I&E Circulars, and (3) I&E Bulletins. A fourth type of report, "Power Reactor Events," is directed more to the general public and persons interested in the nuclear industry; these reports are distributed by the NRC's Office of Management and Program Analysis.

4.3 Abnormal Occurrences

An abnormal occurrence is an unscheduled incident or event at, or associated with, any facility that is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or to the Energy Reorganization Act of 1974, which the NRC determines is significant from the standpoint of public health or safety.

The NRC developed the following criteria by which abnormal occurrences are 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 a major reduction in the degree of protection provided for the health and safety of the public.

Each quarter the NRC submits to the Congress a report listing any abnormal occurrences for that period, as required by Sect. 208 of the Energy Reorganization Act of 1974. The report contains the date and place, nature and probable consequences, cause or causes, and any action taken to prevent recurrence of each abnormal occurrence.

During 1979, seven abnormal occurrences took place at commercial nuclear power plants and were reported to Congress. A brief summary of each occurrence is given below. Also included is updated information on previously reported occurrences.

4.3.1 Degraded engineered safety features (AO 79-1)[2,3]

On September 16, 1978, an unusual sequence of events involving the electrical power sources occurred at Arkansas 1 and 2, culminating in the spurious activation and degraded operation of the engineered safety features (ESF) of Unit 2. Analysis of the course of the incident identified serious deficiencies in the operation and design of the electrical distribution system. There were no radiological consequences.

Three safety concerns emerged from the analysis of the incident:

1. The offsite power supply for Unit 1 ESF loads was deficient in that degraded voltage could have resulted in the unavailability of ESF equipment if it were needed.
2. The design of the electrical system that provides offsite power to Arkansas 1 and 2 did not fully meet the NRC's regulations because in certain circumstances a failure of one of the two offsite power circuits would also result in failure of the other circuit.
3. Deficiencies existed in the operation of the Unit 2 inverters that convert battery power to ac power for certain safety-related equipment.

4.3.2 Deficiencies in piping design (AO 79-2)[2-4]

The NRC ordered five plants to shut down on March 13, 1979, until reanalysis and necessary modifications were made to safety-related piping systems to bring them into conformance with requirements for withstanding earthquakes. The plants ordered to shut down were Beaver Valley 1, FitzPatrick, Maine Yankee, and Surry 1 and 2. The deficiencies in piping design at these plants were caused by the use of an algebraic summation method to combine seismic forces in a computer code which resulted in the prediction of stresses significantly lower than would be predicted by NRC-approved techniques.

An additional issue involving the accuracy of the information input for seismic analyses was also identified. The majority of all operating plants have had to modify and/or add supports because of deviations identified between existing "as-built" structures in the field and design documents (drawings/specifications).

4.3.3 Nuclear accident at Three Mile Island (AO 79-3)[2-5]

Information pertaining to the Three Mile Island accident has been published in the *Federal Register* and extensively reported by the news media; therefore, only a brief description is given here.

At approximately 4:00 AM on March 28, 1979, Three Mile Island 2 experienced a loss of feedwater, which led to a turbine trip and later a reactor trip. Subsequently, a series of events took place that resulted in offsite releases of radioactivity and significant damage to portions of the reactor core. The sequence of events that led to core damage involved equipment malfunctions, design-related problems, and operational errors that, to varying degrees, all contributed to the consequences of the accident. Because plant conditions were substantially degraded, improvised operating modes for postaccident recovery were required.

Because low but intermittently changing radiation levels were measured off the plant site and in view of the uncertainty associated with information then available on the evolving events, the Governor of Pennsylvania as a precautionary measure advised that young children and pregnant women within a 5-mile radius of the plant be evacuated from the area.

The collective dose to the total population within a 50-mile radius of the plant due to the accident has been estimated to be 3300 man-rems. The maximum hypothetical individual dose offsite was less than 100 millirems, as compared to the natural background radiation dose of ~100–125 millirems per year for the area.

The details of the accident continue to be extensively investigated. However, based on partial investigations, there were six main factors that appear to have caused or increased the severity of the accident. These factors, which include combinations of personnel error, design deficiencies, and component failures, are discussed below.

1. At the time of the initiating event (loss of feedwater), both of the auxiliary feedwater systems, a total of three separate valves, were valved out of service. This was a violation of the plant's technical specifications.

2. The pressurizer relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level. Over 2 h elapsed before the operators discovered that the valve did not reseat.

3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have led to erroneous inferences of a high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high-pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Gases continued to be evolved from the primary coolant via the letdown system. Leaks in the waste gas system allowed this highly radioactive gas to enter the auxiliary building and fuel handling building atmosphere. Ultimately, the gases were discharged to the environment via the ventilation systems after being filtered. This was the principal source of the offsite release of radioactive noble gases.

5. Subsequently, the high-pressure injection system was intermittently operated in an attempt to control primary coolant inventory losses through the pressurizer relief valve, apparently based on the pressurizer

level indication. Because of the presence of steam and/or noncondensible voids elsewhere in the reactor coolant system, this action led to a further reduction in the primary coolant inventory.

6. Tripping all the reactor coolant pumps during the course of the transient to protect against pump damage from pump vibration led to fuel damage because voids in the reactor coolant system prevented effective cooling of the core by natural circulation.

4.3.4 Indication of low water level in a boiling-water reactor (AO 79-5)[3]

A loss-of-feedwater transient at Oyster Creek on May 2, 1979, resulted in a significant reduction in water inventory in the area above the reactor core, as measured by one set of water-level instruments, while the remaining two sets of water-level instruments in the reactor annulus indicated water levels above any protective set point. The water level measured within the core shroud area fell below the triple-low set point (a safety limit) of 5 ft, 6 in. above the top of the fuel. Subsequent analyses by the licensee have conservatively determined that the minimum water level above the top of the fuel was 1 to 1-1/2 ft. Coolant sample analyses and off-gas release rates support the conclusion that no fuel damage occurred.

4.3.5 Damage to new fuel assemblies (AO 79-6)[3]

On May [illegible], 1979, while routine inspections of new fuel were being conducted at Surry 2 (a PWR), it was discovered that a foreign substance had been poured onto 62 of the 64 new fuel assemblies stored in the Fuel Building, a vital area containing both new and spent fuel. An analysis of the substance revealed it to be sodium hydroxide. As a result of this analysis and because of the uncertainty about the extent of damage, the licensee (Virginia Electric & Power Company) returned all the assemblies to the vendor for inspection and refurbishment. The licensee determined that there were no indications of damage to the spent fuel, nor was there evidence of unauthorized individuals gaining access to the vital area.

This incident was an alleged criminal act; therefore, on May 7, 1979, the licensee notified the FBI. The FBI conducted an investigation, which culminated in two plant workers surrendering to Surry County authorities on June 19, 1979. A grand jury hearing was held in Surry, Virginia, on July 24, 1979; trial was scheduled for October 10–12, 1979. The two workers, under advice from their attorney, have refused to describe the details of the safety issues which reportably motivated them to commit the act.

As a result of the incident and to assist the FBI in its investigation, the licensee considerably reduced the number of people permitted access to the Fuel Building and stationed a security guard inside to verify access authorization.

4.3.6 Deficient procedures (AO 79-7)[3]

On June 2, 1979, while Arkansas 1 (a PWR) was being prepared for startup, an NRC inspector in the control room found the controls of the emergency feedwater system so positioned that the system could not automatically respond if needed. There was no assurance that the system would have been returned to its normal standby status prior to power operation had the inspector not noticed the problem, since there was no procedural requirement to check the system status.

The licensee, returned the plant to cold shutdown and maintained it in cold shutdown until the NRC staff was satisfied with the utility's methods for controlling the development of operating procedures, with the adequacy of existing procedures, and until there was assurance that operators would not deviate from those procedures.

4.3.7 Major degradation of primary containment boundary (AO 79-8)[5]

On September 14, 1979, the Consumers Power Company (licensee) notified the NRC of the discovery of two improperly positioned valves in the containment purge system at their Palisades Nuclear Plant (a PWR). While preparing to perform a Type C (local isolation valve) leak test between two manual valves in a 4-in. bypass line around the main 48-in. containment purge valve, plant personnel discovered that both of these manual isolation valves were locked in the open position. These valves should have been locked in the closed position. Investigation by the licensee indicated that the valves may have been improperly positioned since April 1978 when an efficiency test of the bypass line filters was performed. The plant has operated at power for the major portion of that time period.

The principal cause of this event was a lack of necessary attention to detail in the development of procedures for ensuring containment integrity. The master checklist for ensuring containment integrity which is used to perform a valve lineup prior to each startup from cold shutdown did not include these valves.

The NRC staff determined that the event demonstrated a weakness in the licensee's ability to control testing and maintenance activities, to develop and review procedures, to adhere to approved procedures, and to conduct audit activities and on November 9, 1979, proposed imposition of civil penalties in the amount of $450,000 for the prolonged violation of containment integrity.

On November 16, 1979, the Director of the Office of Inspection and Enforcement sent a letter to chief executives of all utilities with operating licenses and construction permits informing them of the enforcement action against Consumers Power Company and stating the intention to take similar action in any future instances where ineffective management leads to a serious breach of safety.

4.3.8 Updated information on previously reported abnormal occurrences

The NRC, NRC licensees, and other involved parties, such as reactor vendors and architect-engineers, continued the implementation of actions necessary to prevent recurrence of previously reported abnormal occurrences. Updated information on these abnormal occurrences is briefly summarized below. (The numbers and descriptive titles are the same as those used when the occurrences were originally reported to Congress.)

75-5 Cracks in pipes at boiling-water reactors. This occurrence involved the susceptibility of stainless steel piping to stress-corrosion cracking; it was originally reported in NUREG-75/0090, *Report to Congress on Abnormal Occurrences: January–June 1975*, and was updated in subsequent reports in this series [NUREG-0090-1, -2, -3, -9, NUREG-0090, Vol. 1 (No. 3), and Vol. 2 (Nos. 2 and 4)].

75-7 Steam generator feedwater flow instability at pressurized-water reactors. This occurrence involved steam generator water hammer; it was originally reported in NUREG-75/0090, *Report to Congress on Abnormal Occurrences: January–June 1975*, and was updated in subsequent reports in this series [NUREG-0090-1, -6, NUREG-0090, Vol. 1 (No. 4) and Vol. 2 (No. 2)].

Steam generator water hammer has occurred in certain nuclear power plants as a result of rapid condensation of steam in a steam generator feedwater line. The consequent acceleration of a slug of water and its impact ("hammering") within the piping system causes undue stresses in the piping and its support system. The significance of these events varies from plant to plant. Since a total loss of feedwater could affect the ability of the plant to cool down after a reactor shutdown, the NRC is concerned about these events, even though an event of this type with potentially serious consequences is unlikely to occur.

76-1 Deficiencies in the Mark I containment systems of certain boiling-water reactors. This occurrence was originally reported in NUREG-0090-3, *Report to Congress on Abnormal Occurrences: January–March 1976*, and was updated in subsequent reports in this series [NUREG-0090-4, -6, NUREG-0090, Vol. 1 (Nos. 1 and 3), and Vol. 2 (No. 3)].

76-11 Steam generator tube integrity. This item of concern has involved tube rupture due to wearing caused by loose parts in the system, "denting" due to a corrosion-related phenomenon, "deep crevice cracking," and tube leakage due to fatigue cracks caused by flow-induced vibrations. An abnormal occurrence involving steam generator tube integrity was originally reported in NUREG-0090-5, *Report to Congress on Abnormal Occurrences: July–September 1976*, and was updated in subsequent reports in the series [NUREG-0090-8, NUREG-0090, Vol. 1 (No. 4) and Vol. 2 (Nos. 3 and 4)].

76-16 Feedwater nozzle cracking in boiling-water reactors. This concern was originally reported in NUREG-0090-6, *Report to Congress on Abnormal Occurrences: October–December 1976*, and was updated in subsequent reports in this series [NUREG-0090, Vol. 1 (No. 4) and Vol. 2 (No. 2)].

Over the last several years, inspections at 22 BWR plants licensed for operation in the United States have disclosed some degree of cracking in the feedwater nozzles of the reactor vessel at 18 of these facilities. In a closely related area, cracks have been found in control rod drive return line nozzles, the openings in BWR pressure vessels through which the high-pressure water in excess of that needed to operate and cool the control rod drives is returned to the pressure vessel. The cracks resemble those found in feedwater nozzles. Both conditions probably result from cyclic thermal stresses.

77-8 Generic design deficiency. This generic concern involves insufficient net positive suction head for the containment recirculation spray pumps. An occurrence was originally reported in NUREG-0090-10, *Report to Congress on Abnormal Occurrences. October–December 1977*, and was updated in a subsequent report in this series [NUREG-0090, Vol. 1 (No. 4)].

77-9 Environmental qualification of safety-related electrical equipment inside containment. This occurrence was originally reported in NUREG-0090-10, *Report to Congress on Abnormal Occurrences: October–December 1977*, and was updated in subsequent reports in this series [NUREG-0090, Vol. 1 (Nos. 1 and 2) and Vol. 2 (No. 2)].

There have been some 32 separate reports of unqualified equipment (involving 5 different types of equipment) at 29 different plants. The unqualified equipment reported included: (1) limit switches mounted on safety-related valve stems to indicate valve stem position, (2) containment isolation valve motor operators, (3) instrument and control cable insulated terminal lugs, (4) aluminum limit switch housings on containment isolation valves, and (5) ASCO pilot solenoid valves for miscellaneous valve air operators.

78-2 Fuel assembly control rod guide tube integrity (a generic concern). This occurrence was originally reported in NUREG-0090, Vol. 1 (No. 2), *Report to Congress on Abnormal Occurrences: April–June 1978*, and was updated in NUREG-0090, Vol. 1 (No. 4) and Vol. 2 (No. 2).

Examination of fuel assembly control rod guide tubes after service in several operating PWRs disclosed significant amounts of wear. In extreme cases, some tubes were worn through completely, showing sizable holes. The cause was determined to be flow-induced vibration of fully withdrawn control rods. The rod tips, vibrating against the guide tubes, induced degrading wear, probably aided by corrosion.

78-5 Loss of containment integrity. This occurrence was originally reported in NUREG-0090, Vol. 1 (No. 4), *Report to Congress on Abnormal Occurrences: October–December 1978*, and was updated in NUREG-0090, Vol. 2 (Nos. 2 and 4).

The NRC staff was informed that at least three valve vendors reported that their valves may not close against the ascending differential pressure and the resulting dynamic loading of a design-basis loss-of-coolant accident (LOCA). All identified licensees whose plants had questioned the designs are maintaining the valves in the closed position or are restricting the opening of the valves when primary containment integrity is required. Reevaluation of valve performance under the design-basis LOCA condition is being made by affected licensees.

4.3.9 Other events of interest

Descriptions of the following events are included in this report because they may possibly be perceived by the public to be significant with regard to public health. The events did not involve a major reduction in the level of protection provided for public health or safety and therefore are not reportable as an abnormal occurrence.

Cracking in main feedwater system piping (PWR plants).[3] On May 20, 1979, Indiana and Michigan Electric Company informed the NRC of cracking in two feedwater lines at Cook 2. Leaking circumferential cracks were identified in the 16-in. main feedwater lines in the immediate vicinity of the steam generator nozzles.

Construction deficiencies.[5] During NRC inspections conducted in April and May 1979 of construction activities at Marble Hill 1 and 2, various problems were discovered that indicated inadequacies in the licensee's (Public Service Company of Indiana) quality assurance program. On June 12, 1979, NRC received allegations of improper concrete honeycomb repairs, and subsequent inspections and investigations confirmed these allegations. These findings, together with the previously identified quality assurance problems associated with concrete placement activities, led to the cessation of concrete placement work in safety-related structures.

Release of low-level radioactive gas.[5] At 6:09 AM on September 25, 1979, North Anna, Unit 1 experienced a secondary system component failure, which resulted in plant shutdown and the operation of safety equipment to control the transient. During recovery operations, which entail securing the safety equipment and restoring system valve lineups to normal, the volume control tank, which holds 300 ft^3 of radioactive primary coolant water and hydrogen gas under low pressure, was overpressurized. This resulted in the release of a mixture of hydrogen and noble gases from the reactor coolant to the radiological waste tanks and from there to the auxiliary building atmosphere.

Turbine disk cracking.[4] On November 5, 1979, Wisconsin Electric Power Company, in a meeting on another subject, notified the NRC of cracking in the keyway areas of low-pressure steam turbines manufactured by Westinghouse Electric Corporation.

On November 20, 1979, the Westinghouse Steam Turbine Division confirmed the existence of bore cracking, in addition to keyway cracking, after an inspection of the low-pressure turbine at Zion 1.

The primary NRC concern, since the turbines are not safety related, has been the possibility of the generation of missiles, which might cause a breach of the containment. This is a postulated concern, since in the only known disk failure in a nuclear turbine in the United States, the missiles generated did not penetrate the turbine housing and thus there were no external missiles. The NRC is currently evaluating the potential for other problems resulting from a turbine failure.

5. FUEL PERFORMANCE

5.1 Introduction

The NRC does not monitor every fuel failure that occurs in licensed operating nuclear power plants. The approach taken is to set up operating limits for radioactivity in the coolant (due to fuel failures) which are stringent enough to ensure that dose limits specified in the *Code of Federal Regulations* are not exceeded and to monitor only those fuel failures which are significant from the viewpoint of the number of fuel rods that failed or those in which the failure is due to a new fuel failure mechanism. Periodically, meetings are held with the nuclear fuel vendors to review the operating experience of their fuel. Operating reactors typically have about 40,000 fuel rods, and the average fuel rod failure rate during the last few years has been near or below 0.02% per cycle.[1] (This excludes the TMI-2 reactor, which is estimated to have most, if not all, of its fuel damaged as a result of the 1979 accident.) Fuel performance has continually improved, yet deviations from the normal occur occasionally.

5.2 Specific Fuel-Related Incidents

Several events related to fuel performance were reported during calendar year 1979. The events addressed in the NRC's *Report to Congress on Abnormal Occurrences* (NUREG-0090 series) are described in Sects. 5.2.1 and 5.2.2. The events reported as Licensee Event Reports (LERs) are discussed in Sects. 5.2.3 through 5.2.15.

5.2.1 Three Mile Island 2 (PWR)

On March 28, 1979, a loss-of-coolant accident at Three Mile Island 2 resulted in structural damage to the upper 40% of the core. Most, if not all, of the fuel rods sustained some damage. The zirconium cladding underwent severe oxidation, which left it embrittled. Fuel melting is not suspected because the maximum temperature in the core was estimated to be well below the fuel melting point of 5100°F.[2] (LER 79-012)

5.2.2 Surry 2 (PWR)

During a routine inspection of new fuel at Surry 2 on May 7, 1979, it was found that a substance, later identified as sodium hydroxide, had been poured on 62 of 64 new fuel assemblies. There was no apparent damage to the assemblies; however, they were returned to the vendor for examination and refurbishment.[3] (LER 79-012)

5.2.3 Brunswick 2 (BWR)

An indication of possible fuel leakage at Brunswick 2 was reported on July 27, 1979. The probable cause was stated to be pellet-cladding interaction caused by exceeding the fuel preconditioning limits while increasing power. Control rod 30-23 double-notched (moved 12 in. instead of 6 in. as demanded) while the rods were being withdrawn. (LER 79-056)

5.2.4 Brunswick 2 (BWR)

Carolina Power and Light Company reported to the NRC on December 6, 1979, that the amount of radioactive iodine in the reactor coolant at Brunswick 2 exceeded the technical specification limit. The fuel bundles were to be discharged during the next refueling outage and the leaking bundles replaced. (LER 79-099)

5.2.5 La Crosse (BWR)

Two reports of fuel degradation at La Crosse were dated April 19, 1979. During refueling activities on April 5, 1979, inspection of irradiated fuel assembly 2-33 revealed abnormal degradation in the stainless steel cladding of one fuel rod. A small portion of the fuel rod (~8.5 in.) became displaced from the assembly. (A similar occurrence was reported in LER 77-04.) The cause was attributed to pellet-cladding interaction, with oxygen-assisted stress corrosion, resulting in longitudinal and circumferential cracking of the stainless steel cladding. Assembly 2-33 was due for discharge and will not be reused. (LER 79-005)

On April 10, 1979, also during refueling activities, inspection of irradiated fuel assembly 2-13 revealed abnormal degradation in the stainless steel cladding of one fuel rod. A small portion of the fuel rod (~3.5 in.) became displaced from the assembly. (Similar occurrences were reported in LERs 77-04 and 79-05.) The cause was attributed to pellet-cladding interaction, with oxygen-assisted stress corrosion, resulting in longitudinal and circumferential cracking of the stainless steel cladding. Assembly 2-13 was due for discharge and will not be reused. (LER 79-006)

5.2.6 Quad Cities 1, 2 (BWRs)

Reports from Quad Cities 1 and 2 dated January 16 and April 30, 1979, stated that the average release rate of radioiodine and radioactive material in particulate form with half-lives greater than 8 days exceeded the technical specification limit because of leaking fuel. Significant power changes resulted in a corresponding increase in fission product levels in the reactor coolant. This "spiking," which lasts only a short period of time, was caused by some of the fission product inventory being released from the failed fuel into the reactor coolant. (Letters to the NRC)

5.2.7 Vermont Yankee (BWR)

A report from Vermont Yankee dated April 11, 1979, stated that a review of the vent stack radioiodine sample on March 12, 1979, indicated that the ^{131}I release limit had been exceeded. Suspecting leaking fuel, they conducted tests, which resulted in 24 out of 124 fuel bundles being replaced. (LER 79-006)

5.2.8 Vermont Yankee (BWR)

A report on Vermont Yankee to the NRC, dated October 17, 1979, gave evidence of wearing of the lower end plug on the water rods associated with two fuel assemblies due to flow-induced motion. (LER 79-025)

5.2.9 Connecticut Yankee (PWR)

Axial cracks were discovered in 36 of 48 fuel assemblies at Connecticut Yankee and reported to the NRC February 28, 1979, followed by an undate report on July 24. The probable failure mechanism is brittle fracture of the stainless steel cladding caused by a power ramp at the end of cycle 7, followed by reduced power operation. (LER 79-001)

5.2.10 Crystal River 3 (PWR)

Five reports from Florida Power Corporation (1/22/79, 2/9/79, 2/22/79, 3/16/79, and 9/4/79) describe seven events in which the dose equivalent of ^{131}I in the reactor coolant exceeded the limit of 1 µCi/g. (Five similar events had occurred earlier.) All of these events were caused by an expected iodine spike following a reactor trip with known leaking fuel. (LERs 78-075, 79-007, 79-011, 79-020, and 79-077)

5.2.11 Maine Yankee (PWR)

Three reports from Maine Yankee (one dated September 25 and two dated December 7, 1979) describe four events in which the radioiodine concentration in the reactor coolant exceeded the technical specification limit. Fuel sipping was anticipated for the next refueling to determine if cladding failure has occurred. (LERs 79-017, 79-029, and 79-030)

5.2.12 North Anna 1 (PWR)

A report dated October 5, 1979, stated that on September 6, 1979, during a review of chemistry logs at North Anna 1, it was discovered that an operator had failed to report that the specific activity of ^{131}I in the primary coolant had exceeded the technical specification limit on September 23, 1978. A fuel failure had occurred on July 25, 1978. (LER 79-109)

5.2.13 Prairie Islana 1 (PWR)

It was reported on May 2, 1979, that an inspection of region 4 assemblies removed during the cycle 4-5 refueling at Prairie Island 1 revealed rod bowing which was greater than seen on previously discharged assemblies. There was no apparent fuel damage. All region 4 fuel was discharged from the core; inspection of fuel in the other regions showed no abnormalities. The fuel vendor is studying the problem. (LER 79-012)

5.2.14 Prairie Island 1 (PWR)

A report from Prairie Island 1 dated May 11, 1979, stated that Exxon Nuclear Co., Inc. had made an error in the core loading pattern involving improper location of gadolinium-bearing assemblies. The assemblies were repositioned and confirmed to be properly located. (LER 79-014)

5.2.15 Yankee-Rowe (PWR)

A report to the NRC from Yankee-Rowe on August 10, 1979, stated that the fuel pin pressure at Yankee-Rowe exceeds its specified value. Calculations indicate no adverse effect. (LER 79-018)

6. RADIATION EXPOSURE

6.1 Occupational Radiation Exposure

This chapter reviews the data on occupational radiation exposure of personnel at BWR and PWR commercial nuclear power plants. Data from 67 plants are considered based upon their completion of at least 1 year of commercial operation as of December 31, 1979. Indian Point 1, although defueled, is included in the review, while Fort St. Vrain (an HTGR) is not included because it had accumulated only 6 months of commercial operation during the year.

The primary sources of information on occupational radiation exposure are two types of annual reports that are required to be submitted to the NRC in March of each year:

1. A report indicating the number, job description, and cumulative dose of those individuals whose annual whole-body dose exceeded 100 millirems is required by the technical specifications of each plant. The standard format for the report is given in NRC's Regulatory Guide 1.16.
2. A statistical summary report indicating the total number of individuals monitored and the number of individuals whose annual whole-body dose fell into certain dose ranges is required by 10 CFR 20.407.

Tables 6.1 and 6.2, derived primarily from the first type of annual report, indicate that 54.4% of the total collective dose (man-rems) was incurred by contractor personnel at BWRs compared to 60.5% at PWRs. Table 6.3 presents a breakdown of these collective doses by work function for the last 6 years. One can see that workers performing routine and special maintenance activities continue to receive about two-thirds of the total collective dose. At PWRs the largest portion (46%) of the collective dose (19,805 man-rems) was incurred by workers involved in special maintenance, while at BWRs the largest portion (39%) of the collective dose (16,674 man-rems) was incurred by workers involved in routine maintenance activities.

Table 6.4 summarizes the exposure information reported pursuant to 10 CFR 20.407 by commercial BWRs and PWRs during the last 7 years. The average annual dose for individuals receiving measurable exposures is 0.62 rems, remaining less than 1 rem as it has every year since 1972.

The total collective dose for 1979, 39,759 man-rems, is a considerable increase over last year's value. Part of the increase could be due to the fact that three additional PWRs completed 1 year of commercial operation and were included for the first time. The activities required by the NRC, as set forth in bulletins issued during 1979, also caused an increase in the collective dose received by workers at several plants.

For additional information, refer to the NRC report, *Occupational Radiation Exposure at Commercial Nuclear Power Plants — 1979* (NUREG-0713), which can be obtained from the National Technical Information Service.

Table 6.1. Annual Whole-Body Doses at BWRs — 1979[a]

Plant name	Plant and utility personnel		Contractor personnel		Totals	
	No. of workers with doses >0.10 rems	Collective dose (man-rems)	No. of workers with doses >0.10 rems	Collective dose (man-rems)	No. of workers with doses >0.10 rems	Collective dose (man-rems)
Big Rock Point	327	348	192	101	519	449
Browns Ferry 1, 2, 3	1,667	912	245	191	1,912	1,103
Brunswick 1, 2	434	501	1,550	1,962	1,984	2,463
Cooper Station	133	122	104	83	237	205
Dresden 1, 2, 3		1,370		756	1,572[b]	2,126
Duane Arnold	86	61	352	238	438	299
FitzPatrick		300		502	575[b]	802
Hatch 1	387	177	627	316	1,014	493
Humboldt Bay	50	20	1	1	51	21
La Crosse	72	161	14	20	86	181
Millstone 1	446	367	1,301	1,039	1,747	1,406
Monticello	266	93	96	45	362	138
Nine Mile Point		509		860	1,084[b]	1,369
Oyster Creek	370	327	357	133	727	460
Peach Bottom 2, 3	741	605	872	648	1,613	1,253
Pilgrim	219	356	648	368	867	724
Quad Cities 1, 2		862		1,187	1,416[b]	2,049
Vermont Yankee	429	512	647	621	1,076	1,133
Totals	5,627+	7,603	7,006+	9,071	17,280	16,674

[a]Includes only those reactors that had been in commercial operation for at least 1 year as of December 31, 1979.

[b]Data presented is taken from the annual reports submitted in accordance with Regulatory Guide 1.16 except where the reported number of personnel receiving doses greater than 0.100 rems deviates by 15% or more from the number of personnel reported pursuant to 10 CFR 20.407. For these plants, the total number of personnel shown in the table is the number of workers whose doses exceeded 0.100 rems, as determined from the 10 CFR 20.407 reports.

Table 6.2. Annual Whole-Body Doses at PWRs — 1979[a]

Plant name	Plant and utility personnel: No. of workers with doses >0.10 rems	Plant and utility personnel: Collective dose (man-rems)	Contractor personnel: No. of workers with doses >0.10 rems	Contractor personnel: Collective dose (man-rems)	Totals: No. of workers with doses >0.10 rems	Totals: Collective dose (man-rems)
Arkansas 1	278	85	500	182	778	267
Beaver Valley	121	52	181	53	302	105
Calvert Cliffs 1, 2	,75	334	570	370	1,045	704
Cook 1, 2[b]	250	254	704	437	954	691
Crystal River	296	140	569	327	865	467
Davis-Besse	227	24	58	4	285	28
Farley		165		417	858[c]	582
Fort Calhoun	180	72	77	43	253	115
Ginna	310	391	177	209	487	600
Haddam Neck	472	302	886	622	1,358	924
Indian Point 1, 2	508	644	586	591	1,094	1,235
Indian Point 3		185		577	673[c]	762
Kewaunee		43		70	205[c]	113
Maine Yankee		92		19	218[c]	111
Millstone 2	221	131	527	239	748	370
North Anna 1[b]		141		104	662[c]	245
Oconee 1, 2, 3		818		180	1,279[c]	998
Palisades	754	468	747	341	1,501	809
Point Beach 1, 2	158	186	356	428	514	614
Prairie Island 1, 2		111		42	305[c]	153
Rancho Seco		79		83	157[c]	162
Robinson 2	266	376	743	757	1,009	1,133
Salem 1	587	239	800	382	1,387	621
San Onofre	100	60	112	54	212	114
St. Lucie	205	173	265	157	470	330
Surry 1, 2	467	570	2,536	2,789	3,003	3,359
Three Mile Island 1, 2[b]	1,211	524	2,360	981	3,571	1,505
Trojan	253	131	218	103	471	234
Turkey Point 3, 4	687	460	576	670	1,263	1,130
Yankee-Rowe		66		46	221[c]	112
Zion 1, 2		501		711	1,007[c]	1,212
Totals	8,026	7,817	13,548	11,988	27,155	19,805

[a]Includes only those reactors that had been in commercial operation for at least 1 year as of December 31, 1979.

[b]Concluded first year of commercial operation in 1979.

[c]Data presented is taken from the annual reports submitted in accordance with Regulatory Guide 1.16 except where the reported number of personnel receiving doses greater than 0.100 rems deviates by 15% or more from the number of personnel reported pursuant to 10 CFR 20.407. For these plants, the total number of personnel shown in the table is the number of workers whose doses exceeded 0.100 rems, as determined from the 10 CFR 20.407 reports.

Table 6.3. Percentages of Total Collective Doses Incurred by Workers at BWRs and PWRs by Work Function

Work function	Percent of total collective dose					
	1974	1975	1976	1977	1978	1979
Reactor operations and surveillance	14.0	10.8	10.4	10.5	13.2	12.2
Routine maintenance	45.4	52.5	31.7	28.1	31.5	29.2
In-service inspection	2.7	2.9	5.7	6.4	7.7	9.0
Special maintenance	20.4	19.0	39.5	42.5	35.9	39.4
Waste processing	3.5	6.9	4.8	5.8	5.0	3.6
Refueling	14.0	7.7	7.9	6.7	6.5	6.6

Table 6.4. Summary of Annual Exposures Reported by Nuclear Power Facilities, 1973–1979[a]

Year	Reactor type	Number of reactors included	Total collective dose (man-rems)	No. of workers with measurable doses	Total megawatt-years generated	Average annual dose (rems/worker)	Average No. of man-rems per reactor	Average No. of workers per reactor	Man-rems per megawatt-year
1973	PWR	12	9,399	9,440	3,770	1.00	783	787	2.5
	BWR	12	4,564	5,340	3,394	0.85	380	445	1.3
	Total	24	13,963	14,780	7,164	0.94	582	616	1.9
1974	PWR	20	6,627	9,697	6,824	0.68	331	485	1.0
	BWR	14	7,095	8,769	4,059	0.81	507	626	1.7
	Total	34	13,722	18,466	10,883	0.74	404	543	1.3
1975	PWR	26	8,268	10,884	11,983	0.76	318	419	0.7
	BWR	18	12,611	14,607	5,786	0.86	701	812	2.2
	Total	44	20,879	25,491	17,769	0.82	475	579	1.2
1976	PWR	30	13,807	17,588	13,325	0.79	460	586	1.0
	BWR	23	12,626	17,859	8,586	0.71	549	776	1.5
	Total	53	26,433	35,447	21,911	0.75	499	669	1.2
1977	PWR	34	13,469	20,878	17,341	0.65	396	614	0.8
	BWR	23	19,042	21,388	9,103	0.89	828	930	2.1
	Total	57	32,511	42,266	26,444	0.77	570	742	1.2
1978	PWR	39	16,713	25,720	19,840	0.65	429	659	0.8
	BWR	25	15,096	20,278	11,774	0.74	604	811	1.3
	Total	64	31,809	45,998	31,614	0.69	497	719	1.0
1979	PWR	42	21,437	39,060	18,249	0.55	510	930	1.2
	BWR	25	18,322	25,013	11,671	0.73	733	1,001	1.6
	Total	67	39,759	64,073	29,920	0.62	593	956	1.3

[a]The figures in this table are based on the number of nuclear power reactors that had been in commercial operation for at least 1 year as of December 31 of each of the years indicated. Indian Point 1, although defueled, is counted, but Fort St. Vrain is not.

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2. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, January–March 1979*, NUREG-0090, Vol. 2, No. 1 (July 1979).*

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*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and/or the National Technical Information Service, Springfield, VA 22161.

**Available for purchase from the National Technical Information Service.

Appendix A

GLOSSARY

Abnormal occurrence	See Sect. 4.3 and Appendix C
Average daily power level, MW(e)	The net electrical energy generated during the day (measured from 0001 to 2400 h inclusive) in megawatt-hours divided by 24 h.
Licensed thermal power, MW(t)	The maximum thermal power of the reactor authorized by the NRC, expressed in megawatts.
Date of commercial operation	Date unit was declared by utility owner to be available for the regular production of electricity; usually related to satisfactory completion of qualification tests, as specified in the purchase contract, and to accounting policies and practices of utility.
Design electrical rating (DER), net MW(e)	The nominal net electrical output of the unit specified by the utility and used for the purpose of plant design.
Forced outage	An outage required to be initiated no later than the weekend following discovery of an off-normal condition.
Forced outage hours	The clock hours during the report period when a unit is unavailable due to forced outages.
Gross electrical energy generated, MWh	Electrical output of the unit during the report period as measured at the output terminals of the turbine generator, in megawatt-hours.
Gross hours	The clock hours from the beginning of a specified situation until its end. For outage durations, the clock hours during which the unit is not in power production.
Gross thermal energy generated, MWh	The thermal energy produced by the unit during the report period as measured or computed by the licensee, in megawatt-hours.
Hours generator on-line	Also "unit service hours." The total clock hours in the report period during which the unit operated with

	breakers closed to the station bus. These hours added to the total outage hours experienced by the unit during the report period shall equal the hours in the report period.
Hours in reporting period	For units in power ascension at the end of the period, the gross hours from the beginning of the period or the first electrical production, whichever comes last, to the end of the period. For units in commercial operation at the end of the period, the gross hours from the beginning of the period or of commercial operation, whichever comes last, to the end of the period or decommissioning, whichever comes first.
Hours reactor critical	The total clock hours in the report period during which the reactor sustained a controlled chain reaction.
Maximum dependable capacity (gross) (MDC gross), gross MW(e)	Dependable main-unit gross capacity, winter or summer, whichever is smaller. The dependable capacity varies because the unit efficiency varies during the year due to variations in cooling water temperature. It is the gross electrical output as measured at the output terminals of the turbine generator during the most restrictive seasonal conditions (usually summer).
Maximum dependable capacity (net) (MDC net), net MW(e)	Maximum dependable capacity (gross) less the normal station service loads.
Nameplate rating, gross MW(e)	The nameplate power designation of the generator, in megavolt-amperes (MVA), times the nameplate power factor of the generator. Note that the nameplate rating of the generator may not be indicative of the maximum or dependable capacity, since some other item of equipment of a lesser rating (e.g., turbine) may limit unit output.
Net electrical energy generated	Gross electrical output of the unit, measured at the output terminals of the turbine generator during the reporting period, minus the normal station service electrical energy utilization. If this quantity is less

	than zero, a negative number should be recorded.
Outage	A situation in which no electrical production takes place.
Outage duration	The total clock hours of the outage measured from the beginning of the report period or the outage, whichever comes first.
Period hours	See "hours in reporting period."
Power reduction	A reduction in the average daily power level of more than 20% from the previous day. All power reductions are defined as outages of zero hours duration for the purpose of computing unit service and availability factors and forced outage rate.
Regulatory restriction	Special restrictions imposed by the NRC or other state or federal regulatory agencies limiting power level to less than authorized until the restrictive condition is resolved. Does not include self-imposed operating restrictions.
Restricted power level	Maximum net electrical generation to which the unit is restricted during the report period due to the state of equipment, external conditions, administrative reasons, or a directive from the NRC.
Scheduled outage	Planned removal of a unit from service for refueling, inspection, training, or maintenance. Those outages which do not fit the definition of "forced outage" are perforce "scheduled outages."
Startup and power-ascension-test phase	Period following initial criticality during which the unit is tested at successively higher levels, culminating with operation at full power for a sustained period and completion of warranty runs. Following this phase, the utility generally considers the unit to be available for commercial operation.
Unit	The set of equipment uniquely associated with the reactor, including turbine generators, and ancillary

	equipment, considered as a single electrical energy production facility.
Unit age	The elapsed time from the date of first electrical generation through December 31 of the current year.
Unit available hours	The total clock hours in the report period during which the unit operated on-line or was capable of such operation. (Unit reserve shutdown hours + hours generator on-line.)
Unit availability factor	$\frac{\text{Unit available hours} \times 100}{\text{Period hours}}$
Unit capacity factors	
Using licensed thermal power	$\frac{\text{Gross thermal energy generated} \times 100}{\text{Period hours} \times \text{licensed thermal power}}$
Using nameplate rating	$\frac{\text{Gross electrical energy generated} \times 100}{\text{Period hours} \times \text{nameplate rating}}$
Using DER	$\frac{\text{Net electrical energy generated} \times 100}{\text{Period hours} \times \text{DER}}$
Using MDC gross*	$\frac{\text{Gross electrical energy generated} \times 100}{\text{Period hours} \times \text{MDC gross}}$
Using MDC net*	$\frac{\text{Net electrical energy generated} \times 100}{\text{Period hours} \times \text{MDC net}}$
Unit forced outage rate	$\frac{\text{Forced outage hours}}{\text{Unit service hours} + \text{forced outage hours}}$
Unit reserve shutdown	The removal of the unit from on-line operation for economic or other similar reasons when operation could have been continued.
Unit reserve shutdown hours	The total clock hours in the report period during which the unit was in reserve shutdown mode.
Unit service factor	$\frac{\text{Unit service hours} \times 100}{\text{Period hours}}$
Unit service hours	See "hours generator on-line."

*NOTE: If MDC gross and/or MDC net have not been determined, the DER is substituted for this quantity for unit capacity factor calculations.

Appendix B

INDIVIDUAL PLANT SUMMARIES FOR 1979

Summaries of the 1979 operating experience for each plant are presented in this appendix. The information provided includes plant operating and outage statistics, details on each outage, and highlights of operating experience.

Symbols used in the table provided for each summary are as follows: Under "type," F is used for forced and S is used for scheduled. Under "cause," the following symbols are used:

A — equipment failure
B — maintenance or test
C — refueling
D — regulatory restriction
E — operator training and license exams
F — administrative
G — operational error
H — other

Under "shutdown method," the symbols used are: 1 — manual, 2 — manual scram, 3 — automatic scram, 4 — continuations, and 9 — other.

The system descriptions are given in Table B.1, and the component types are defined in Table B.2. The individual plant summaries are arranged alphabetically by plant name.

The daily average power curves for the year, presented with the plant summaries, are based on maximum dependable capacity (MDC) of the plants as of December 31, 1979; under optimum conditions, the average power may exceed 100% of the MDC.

Table B.1. System descriptions

System	Code
Reactor	RX
Reactor vessel internals	RA
Reactivity control systems	RB
Reactor core	RC
Reactor coolant system and connected systems	CX
Reactor vessels and appurtenances	CA
Coolant recirculation systems and controls	CB
Main steam systems and controls	CC
Main steam isolation systems and controls	CD
Reactor core isolation cooling systems and controls	CE
Residual heat removal systems and controls	CF
Reactor coolant cleanup systems and controls	CG
Feedwater systems and controls	CH
Reactor coolant pressure boundary leakage detection systems	CI
Other coolant subsystems and their controls	CJ
Engineered safety features	SX
Reactor containment systems	SA
Containment heat removal systems and controls	SB
Containment air purification and cleanup systems and controls	SC
Containment isolation systems and controls	SD
Containment combustible gas control systems and controls	SE
Emergency core-cooling systems and controls	SF
Control room habitability systems and controls	SG
Other engineered safety feature systems and 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 and controls	EA
AC onsite power systems and controls	EB
DC onsite power systems and controls	EC
Onsite power systems and controls (composite AC and DC)	ED
Emergency generator systems and controls	EE
Emergency lighting systems and controls	EF
Other electric power systems and controls	EG
Fuel storage and handling systems	FX
New fuel storage facilities	FA
Spent-fuel storage facilities	FB
Spent-fuel-pool cooling and cleanup systems and controls	FC
Fuel handling systems	FD

Table B.1 (continued)

System	Code
Auxiliary water systems	WX
Station service water systems and controls	WA
Cooling systems for reactor auxiliaries and controls	WB
Demineralized water makeup systems and controls	WC
Potable and sanitary water systems and controls	WD
Ultimate heat sink facilities	WE
Condensate storage facilities	WF
Other auxiliary water systems and their controls	WG
Auxiliary process systems	PX
Compressed air systems and controls	PA
Process sampling systems	PB
Chemical, volume control, and liquid poison systems and controls	PC
Failed-fuel detection systems	PD
Other auxiliary process systems and their controls	PE
Other auxiliary systems	AX
Air conditioning, heating, cooling, and ventilation systems and controls	AA
Fire protection systems and controls	AB
Communication systems	AC
Other auxiliary systems and their controls	AD
Steam and power conversion systems	HX
Turbine-generators and controls	HA
Main steam-supply system and controls (other than CC)	HB
Main condenser systems and controls	HC
Turbine-gland-sealing systems and controls	HD
Turbine bypass systems and controls	HE
Circulating water systems and controls	HF
Condensate cleanup systems and controls	HG
Condensate and feedwater systems and controls (other than CH)	HH
Steam generator blowdown systems and controls	HI
Other features of steam and power conversion systems (not included elsewhere)	HJ
Radioactive waste management systems	MX
Liquid radioactive waste management systems	MA
Gaseous radioactive waste management systems	MB
Process and effluent radiological monitoring systems	MC
Solid radioactive waste management systems	MD
Radiation protection systems	BX
Area monitoring systems	BA
Airborne radioactivity monitoring systems	BB

Table B.2. Component types

Component type	Component type includes
Accumulators	Scram accumulators, safety injection tanks, surge tanks, holdup/storage tanks
Air dryers	
Annunciator modules	Alarms, bells, buzzers, claxons, horns, gongs, sirens,
Batteries and 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	Buses, cables, wires
Engines, internal combustion	Butane, diesel, gasoline, natural gas, and propane engines
Filters	Strainers, screens
Fuel elements	
Generators	Inverters
Heaters, electric	Heat tracers
Heat exchangers	Condensers, coolers, evaporators, regenerative heat exchangers, steam generators, fan coil units
Instrumentation and controls	Controllers, sensors/detectors/elements, indicators, differentials integrators (totalizers), power supplies, recorders, switches, transmitters, computation modules
Mechanical function units	Mechanical controllers, governors, gear boxes, varidrives, couplings
Motors	Electric motors, hydraulic motors, pneumatic (air) motors, servomotors

Table B.2 (continued)

Component type	Component type includes
Penetrations, primary containment	Air locks, personnel access, fuel handling, equipment access, electrical, instrument line, process piping
Pipes and/or fittings	
Pumps	
Recombiners	
Relays	Switchgear
Shock suppressors and supports	Hangers, supports, sway braces/stabilizers, snubbers, antivibration devices
Transformers	
Turbines	Steam turbines, gas tubines, hydro turbines
Valves	Valves, dampers
Valve operators	Explosive, squib
Vessels, pressure	Containment vessels, dry wells, pressure suppression chambers, pressurizers, reactor vessels

ARKANSAS 1

I. Summary

Description	Performance		Outages			
Location: Russellville, Arkansas	Net Electrical Energy Generated (MWH):	3,323,490	Total No.	7		
Docket No: 50-313			Forced	4		
Reactor Type: PWR	Unit Availability Factor (%):	55.3*	Scheduled	3		
Capacity (MWe-Net): 836			Total:	4,503*	Hours,	51.4%
Commercial Operation: 12/19/74	Unit Capacity Factor (%) (Using MDC):	45.4	Forced	2,362	Hours,	27.0%
Plant Age: 5.4 Years	Unit Capacity Factor (%) (Using Design MWE):	44.6	Scheduled	2,141	Hours,	24.4%

II. Highlights

From January 1 until February 5, the unit was shut down for repairs to the failed low-pressure turbine blades. From March 30 to June 24, a refueling shutdown was in effect for plant modifications and procedure changes required by NRC's IE Bulletin 70-05A relative to the TMI-2 accident and the emergency feedwater system operation. On July 9, a shutdown was required because of turbine bearing vibration, and during this time the seismic pipe supports were repaired in accordance with IE Bulletins 79-02 and 79-14. The unit returned to service August 8 and operated until October 10 when another TMI-2-related shutdown was required to provide vital power to the emergency feedwater pump and to modify the in-core temperature-detection devices. On November 10, the unit returned to service and operated until December 31, whereupon a third TMI-2-related shutdown was required to modify the feedwater instrumentation in accordance with NRC report NUREG-0578 (*TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations*).

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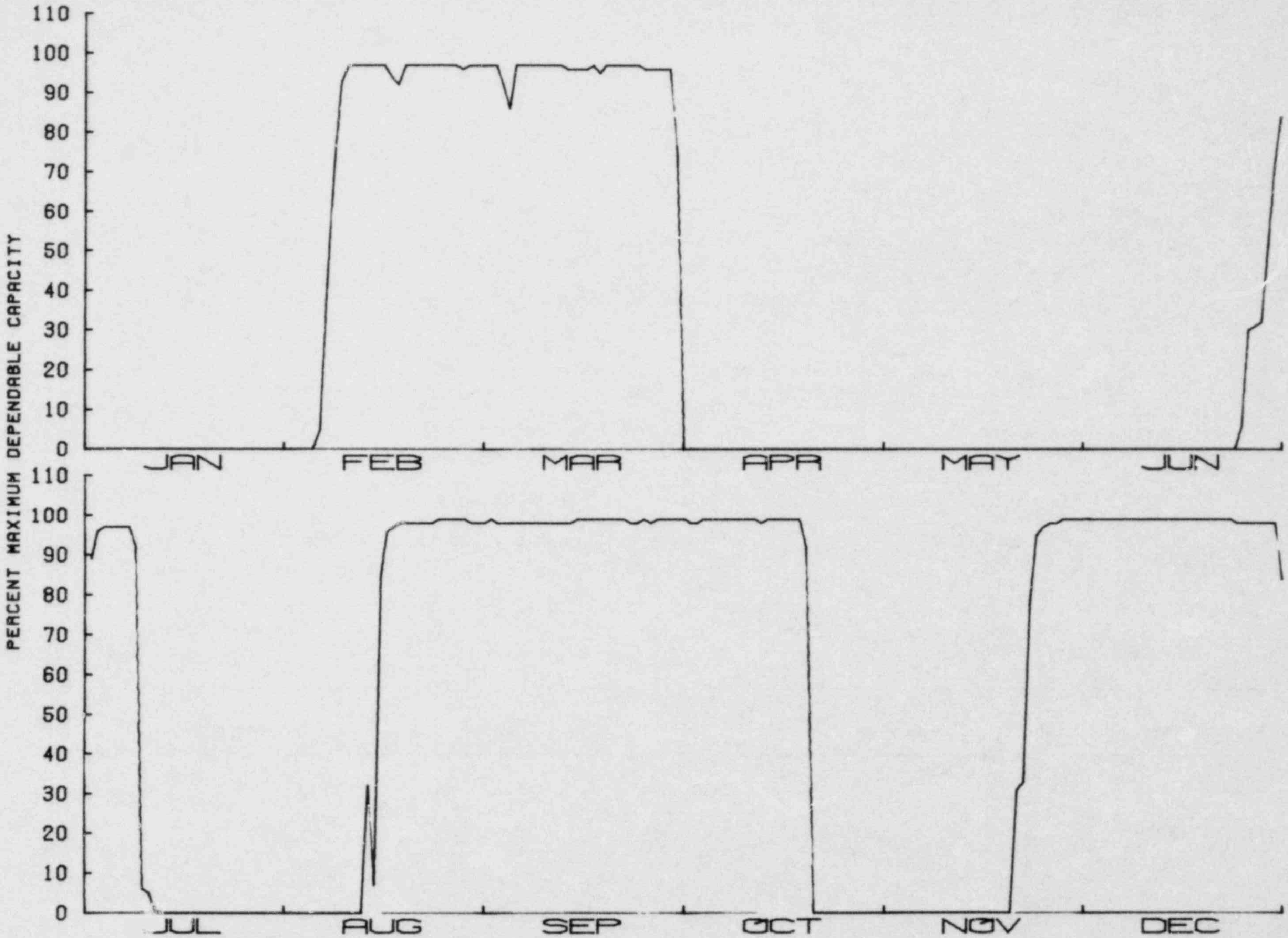
*Includes 591.5 h of reserve shutdown equal to 6.7% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	12/31/78	860	F	Main low pressure turbine failure (cont. from December, 1978)	A	1	Steam and power conversion (HA)	Turbines
2a)	3/30	1290	S	Refueling	C	1	Reactor (RC)	Fuel elements
2b)	3/30 (cont.)	353	F	NRC restrictions on B&W plants concerning procedural, training and design changes related to feedwater transients	D	4	Steam and power conversion (HH)	N/A
2c)	3/30 (cont.)	300	F	NRC hold due to procedural question on the emergency feedwater system operation	D	4	Steam and power conversion (HH)	N/A
2d)	3/30 (cont.)	103	S	Zero power physics testing	B	1	Reactor (RC)	Fuel element
3)	7/8	14	F	Turbine governor valve position indication arm failed	A	3	Steam and power conversion (HA)	Valves
4)	7/9	810	F	Main turbine bearing high vibration	A	1	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
5)	8/13	25	F	Switchyard relay failure	A	3	Electric power (EB)	Relays
6)	10/20	746	S	Provide vital power to emergency feedwater pump and modify in-core temperature detection devices	B	1	Steam and power conversion (HH)	Pumps
7)	12/31	2	S	Commitments to NRC to provide modifications lue to TMI 2	D	1	Steam and power conversion (HH)	Instrumentation and controls



DESIGN ELEC. RATING = 850 MAX. DEPEND. CAP. = 836 (100%) ARKANSAS 1

BEAVER VALLEY 1

I. Summary

Description

Location: Shippingport, Penn.
Docket No: 50-334
Reactor Type: PWR
Capacity (MWe-Net): 817
Commercial Operation: 9/30/76
Plant Age: 3.6 Years

Performance

Net Electrical Energy Generated (MWH):	1,778,375
Unit Availability Factor (%):	40.0
Unit Capacity Factor (%) (Using MDC):	24.8
Unit Capacity Factor (%) (Using Design MWE):	23.8

Outages

Total No.	18	
Forced	17	
Scheduled	1	
Total:	5,257 Hours,	60.0%
Forced	4,513 Hours,	51.5%
Scheduled	744 Hours,	8.5%

II. Highlights

In January, six shutdowns occurred; one resulted in safety injection when a main steam line stop valve closed. On March 9, the unit was shut down for evaluation of seismic design deficiencies in safety-related piping and supports. During the shutdown, circumferential cracks in all three steam generator feedwater nozzle-to-piping welds were identified and repaired. Operation was resumed on August 17. On December 1, the unit was shut down for an extended refueling and modification outage.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/30	348	F	Reactor trip from 99% full power and safety injection due to high steam flow coincident with low steam pressure in the A and B loops when C main steam stop valve tripped and closed	A	3	Steam and power conversion (HB)	Valves
2)	1/18	25	F	Reactor trip due to high steam flow coincident with low steam pressure and a safety injection signal. Heater drain valve (LCV-SD106A) failed to open tripping main feed pumps and main steam dump valves cycled open	A	3	Steam and power conversion (HH)	Valves
3)	1/20	4	F	Loss of No. 3 inverter; failure of No. 3 uninterruptible power supply resulted in loss of 1C reactor coolant pump breaker	A	3	Electric power (ED)	Generators (inverter)
4)	1/26	13	F	Overfeeding the 1B steam generator followed by a feedwater isolation and reactor trip on high steam generator level of 1B steam generator	G	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

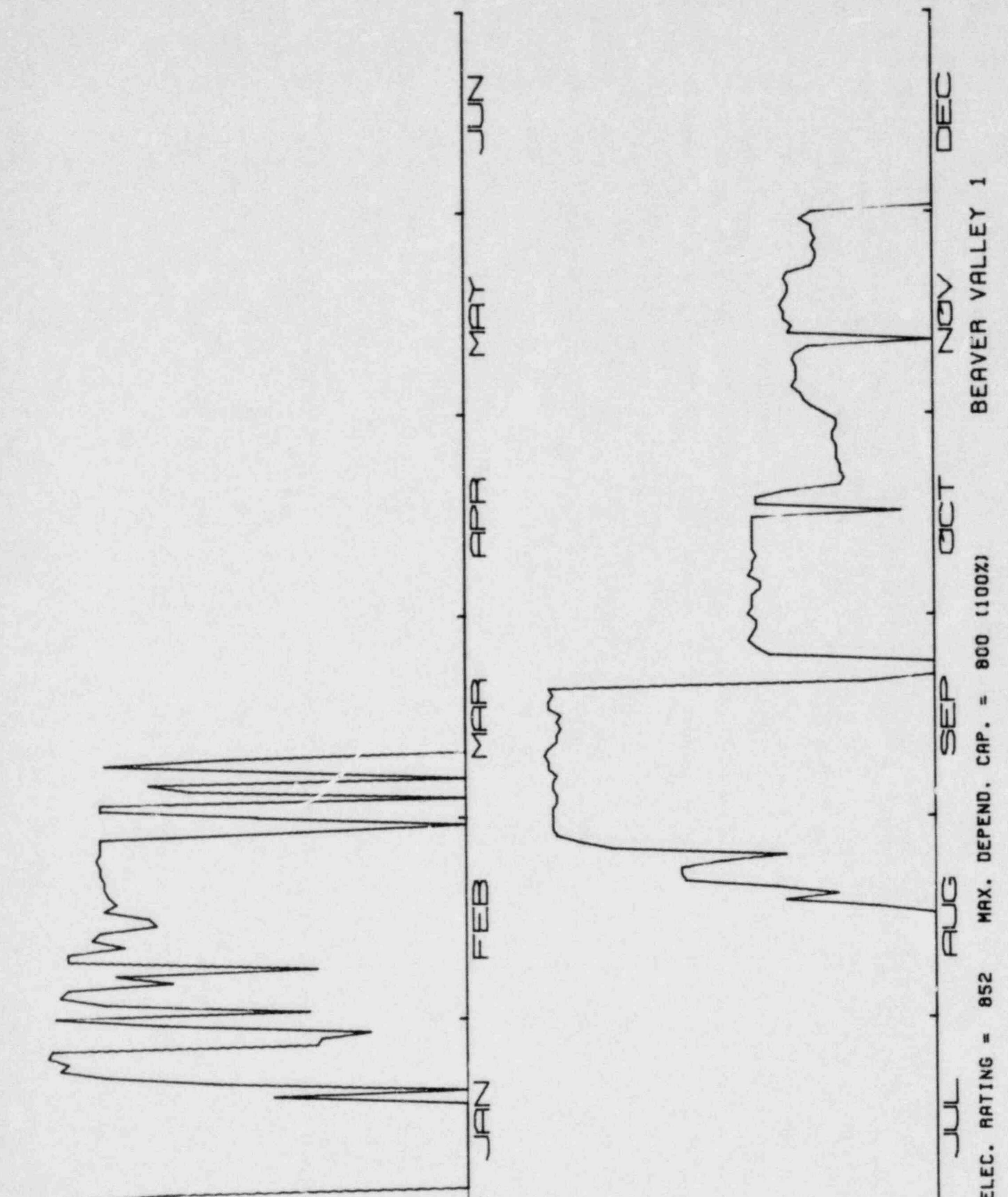
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
5)	1/28	11	F	Reactor trip after reducing power below 10% after losing cooling water to the 1C reactor coolant pump. A blown diaphragm on component cooling water trip valve (TV-CCR103C1) was replaced	A	3	Auxiliary water (WB)	Valves
6)	1/29	4	F	Reactor trip on Lo-Lo steam generator level due to intermittent failure of feed regulation bypass valve to close	A	3	Steam and power conversion (HH)	Valves
7)	2/5	2	F	Coolant pump "C" trip due to voltage surge	A	3	Reactor coolant (CB)	Motors
8)	2/7	4	F	Maintenance unable to pack feedwater regulation valves with unit operating	B	1	Steam and power conversion (HH)	Valves
9)	2/26	44	F	High water level (680 ft) in Ohio River	H	1	Auxiliary water (WE)	N/A

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10)	3/3	26	F	Tripped during turbine thrust bearing trip check	A	3	Steam and power conversion (HA)	Turbines
11)	3/5	36	F	High water level (greater than 680 ft) in Ohio River	H	1	Auxiliary water (WE)	N/A
12)	3/9	3857	F	Design review of safety related piping systems for stress and modification for seismic events	D	1	Engineered safety features (SX)	Pipes, fittings
13)	8/19	10	F	Loss of main feed pump 1A on low suction pressure	A	2	Steam and power conversion (HH)	Pumps
14)	9/20	89	F	Loss of No. 4 Inverter	A	3	Electric power (ED)	Generators (inverter)
15)	10/16	17	F	Rods dropped due to malfunction of rod control cluster assembly	A	1	Reactor (RB)	Control rod drives

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
16)	11/10	21	F	Repair steam leak on feedwater piping	A	1	Steam and power conversion (HH)	Pipes, fittings
17)	11/11	2	F	S/G level problems during power ascension	A	3	Steam and power conversion (HH)	Instrumentation and controls
18)	12/1	744	S	Refueling	C	1	Reactor (RC)	Fuel elements


PERCENT MAXIMUM DEPENDABLE CAPACITY
110
100
90
80
70
60
50
40
30
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10
0
JAN
FEB
MAR
APR
MAY
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AUG
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DEC
DESIGN ELEC. RATING = 852 MAX. DEPEND. CAP. = 800 (100%)
BEAVER VALLEY 1

BIG ROCK POINT

I. Summary

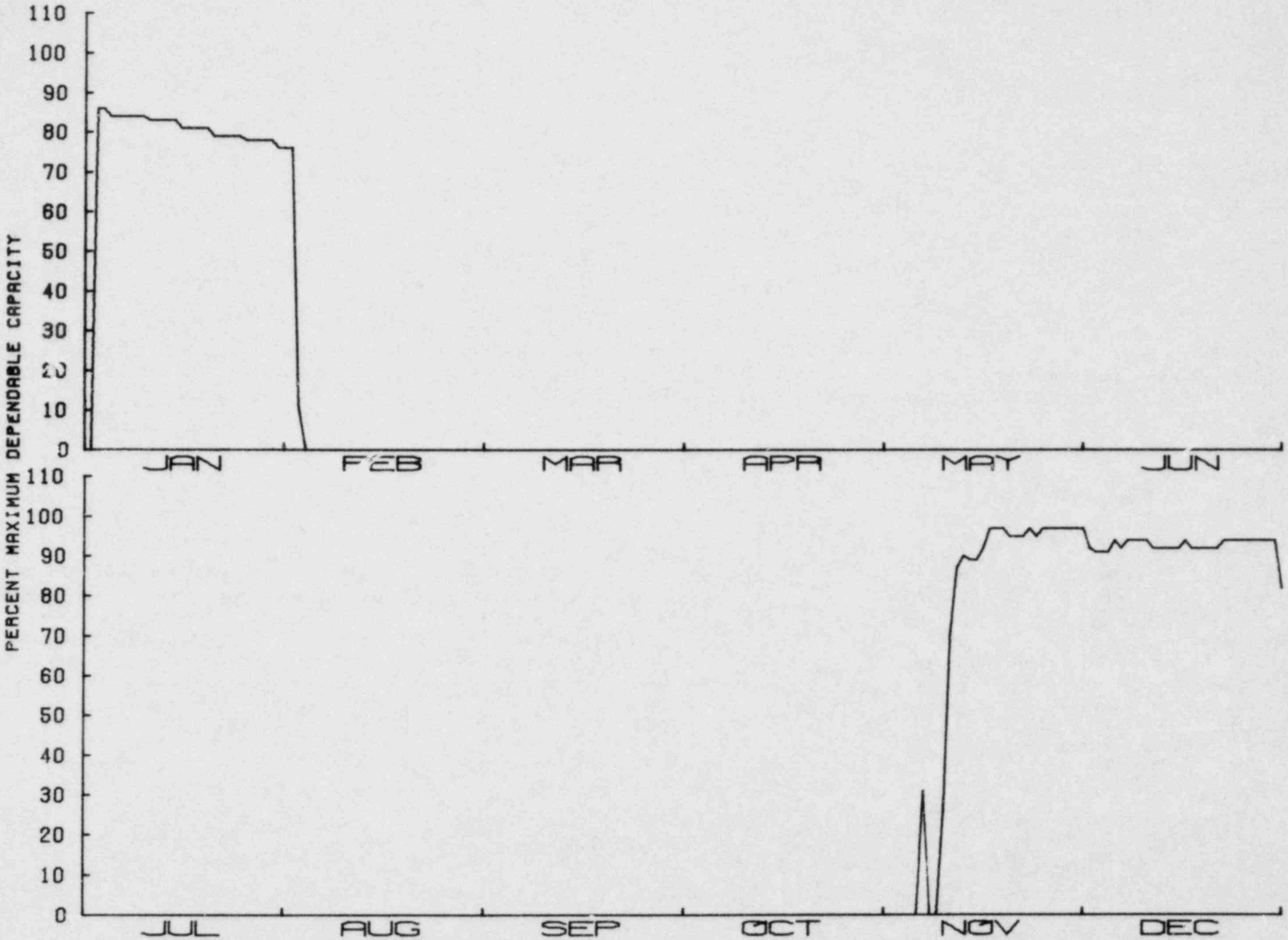
Description	Performance		Outages		
Location: Big Rock Point, Michigan	Net Electrical Energy Generated (MWH):	113,674	Total No.	3	
Docket No: 50-155	Unit Availability Factor (%):	23.5	Forced	1	
Reactor Type: BWR	Unit Capacity Factor (%) (Using MDC):	20.6	Scheduled	2	
Capacity (MWe-Net): 65	Unit Capacity Factor (%) (Using Design MWE):	18	Total:	6,697 Hours,	76.5%
Commercial Operation: 3/29/63			Forced	4,865 Hours,	55.6%
Plant Age: 17.1 Years			Scheduled	1,832 Hours,	20.9%

II. Highlights

At the beginning of the year, the unit was operating under power-level restrictions at 63 MW(e) due to the thermalhydraulic limits of the fuel. On February 3, a refueling outage was started; during the outage, the welds of a new core spray ring were reworked, extending the outage by 3 weeks. On April 17, refueling and core spray ring repair were completed, but during testing a leak in a control rod drive (CRD) housing (thimble F-2) was discovered as well as vibrating hardware in the reactor vessel, resulting from a loose diffuser over the No. 1 recirculation inlet. Eliminating the vibration and repairing the leaking CRD thimble extended the outage to November 4. On November 4, operation was resumed without power restriction. On December 31, a TMI-2-related shutdown was initiated to implement NRC requirements regarding relief valve position indication, manual resetting of containment isolation, and a radiation monitor for assessing core damage, in accordance with NRC report NUREG-0578 (*TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations*).

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a)	2/2	18	F	Replace valve disc with modified design	A	2	Engineered safety features (SA)	Valves
1b)	2/2 (cont.)	1773	S	Refueling	C	4	Reactor (RC)	Fuel elements
1c)	2/2 (cont.)	4847	F	Correct inlet diffuser vibration problem in reactor vessel and repair leak in CRD housing	A	4	Reactor (RA)	Other (inlet diffuser)
2a)	11/6	54	S	Replace recirculating pump seal and repair incore flange leaks	B	1	Reactor coolant (CB)	Pumps
2b)	11/6 (cont.)	3	S	Repair leak in turbine bypass valve drain lin	B	4	Steam and power conversion (HE)	Pipes, fittings
3)	12/31	2	S	Regulatory shutdown for checking relief valve position, manual reset of containment isolation, and radiation monitors	D	1	Instrumentation and controls (IB)	Instrumentation and controls



DESIGN ELEC. RATING = 72 MAX. DEPEND. CAP. = 63 (100%) BIG ROCK POINT 1

BROWNS FERRY 1

I. Summary

Description	Performance		Outages			
Location: Decatur, Alabama	Net Electrical Energy Generated (MWH):	7,495,748	Total No.	17		
Docket No: 50-259			Forced	14		
Reactor Type: BWR	Unit Availability Factor (%):	90.4	Scheduled	3		
Capacity (MWe-Net): 1,065			Total:	838	Hours,	9.6%
Commercial Operation: 8/1/74	Unit Capacity Factor (%) (Using MDC):	80.3	Forced	366	Hours,	4.2%
Plant Age: 6.2 Years			Scheduled	472	Hours,	5.4%
	Unit Capacity Factor (%) (Using Design MWE):	80.3				

II. Highlights

The unit began the year shut down for the reload 2, cycle 3 refueling outage. On January 20, the unit resumed operation, which was routine for the remainder of the year. At the end of the year, the unit was at 90% of full power in coastdown for a scheduled refueling outage to begin January 4, 1980.

DETAILS OF PLANT OUTAGES

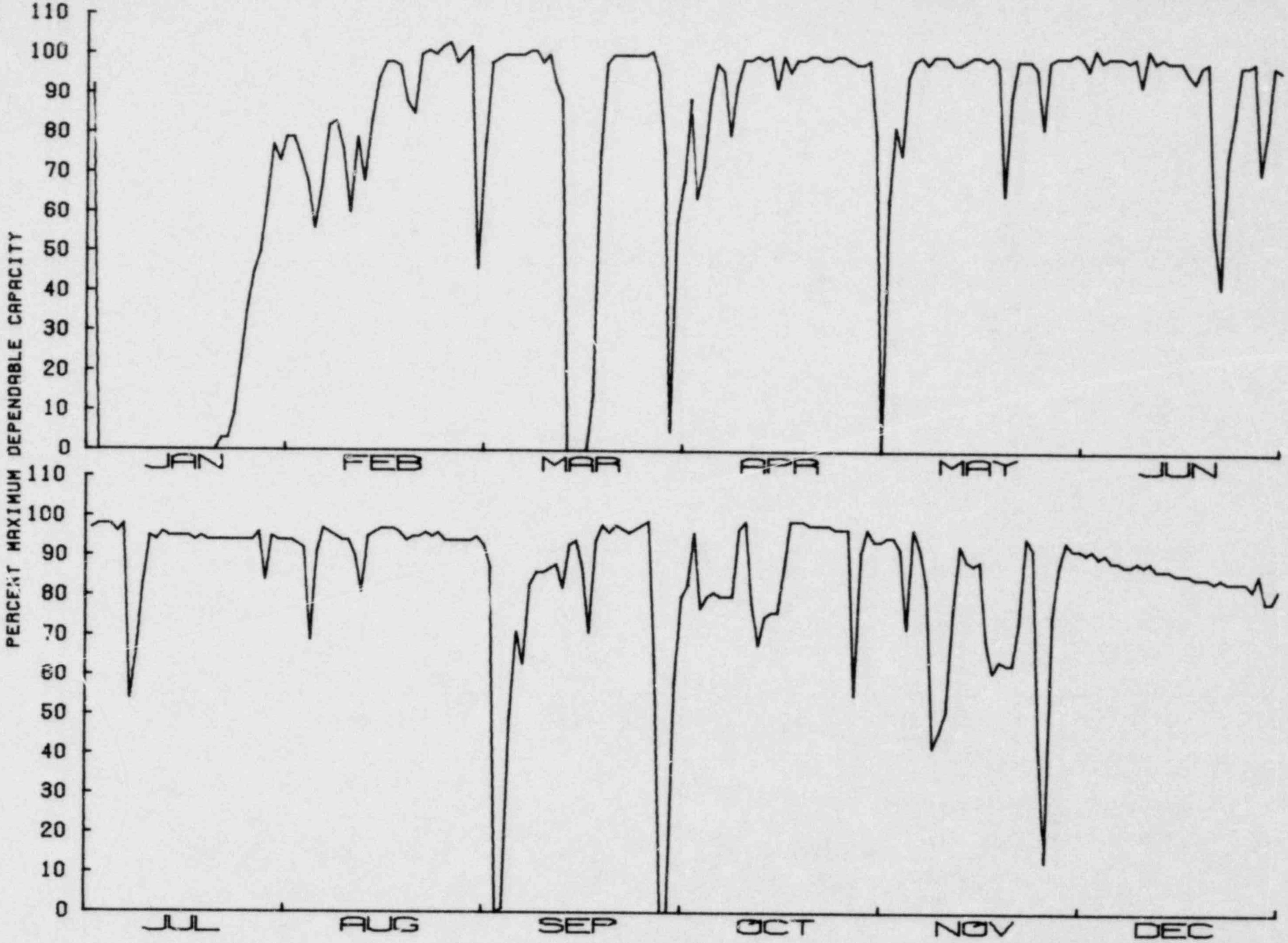
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/1	471	S	Refueling	C	2	Reactor (RC)	Fuel elements
2)	1/20	1	S	Recirculation pump test	B	9	Reactor coolant (CB)	Pumps
3)	1/20	25	F	Balancing main turbine	A	2	Steam and power conversion (HA)	Turbines
4)	1/22	0.3	S	Turbine overspeed trip test	B	9	Steam and power conversion (HA)	Turbines
5)	1/22	12	F	Turbine trip on "sensed" moisture separator high level; when resetting turbine, the Rx tripped on stop valve closure with first stage pressure in excess of 154 psig due to an EHC malfunction	A	3	Steam and power conversion (HB)	Instrumentation and controls
6)	2/27	9	F	Maintenance to number 2 control valve (servo oil leak)	A	2	Steam and power conversion (HB)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	3/12	111	F	Leak in piping on discharge of "B" reactor feedpump	A	2	Reactor coolant (CH)	Pipes, fittings
8)	3/28	19	F	False low reactor water level signal caused by floor drilling operation	H	3	Instrumentation and controls (IA)	Instrumentation and controls
9)	3/29	5	F	Turbine stop valve closure due to moisture separator high water level	A	3	Steam and power conversion (HB)	Instrumentation and controls
10)	4/2	5	F	Stop valve closure due to moisture separator high level	A	3	Steam and power conversion (HB)	Instrumentation and controls
11)	4/29	27	F	Turbine stop valve closure	A	3	Steam and power conversion (HB)	Valves
12)	6/20	17	F	Generator problems ("C" phase arcing)	A	1	Steam and power conversion (HA)	Generators (main generators)

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	9/1	33	F	EHC (turbine control) leak	A	2	Steam and power conversion (HA)	Pipes, fittings
14)	9/3	14	F	Recirculation pump problems	A	1	Reactor coolant (CB)	Pumps
15)	9/26	51	F	Main steam line temperature switch malfunction	A	2	Reactor coolant (CC)	Instrumentation and controls
16)	11/8	9	F	Turbine stop valve closure due to loss of power in "A" level controller	A	3	Steam and power conversion (HB)	Instrumentation and controls
17)	11/24	29	F	Maintenance to drywell control air leak	A	2	Auxiliary process (PA)	Pipes, fittings



DESIGN ELEC. RATING = 1065   MAX. DEPEND. CAP. = 1065 (100%)   BROWNS FERRY 1

BROWNS FERRY 2

I. Summary

Description

Location: Decatur, Alabama
Docket No: 50-260
Reactor Type: BWR
Capacity (MWe-Net): 1,065
Commercial Operation: 3/1/75
Plant Age: 5.3 Years

Performance

Net Electrical Energy Generated (MWH):	7,441,305
Unit Availability Factor (%):	86.7
Unit Capacity Factor (%) (Using MDC):	79.8
Unit Capacity Factor (%) (Using Design MWE):	79.8

Outages

Total No.	16	
Forced	14	
Scheduled	2	
Total:	1,163 Hours,	13.3%
Forced	208 Hours,	2.4%
Scheduled	955 Hours,	10.9%

II. Highlights

Operation during the year was routine and near full power. A refueling outage was conducted in May, and at the end of the year problems with the electrohydraulic control system pressure regulator were being investigated.

DETAILS OF PLANT OUTAGES

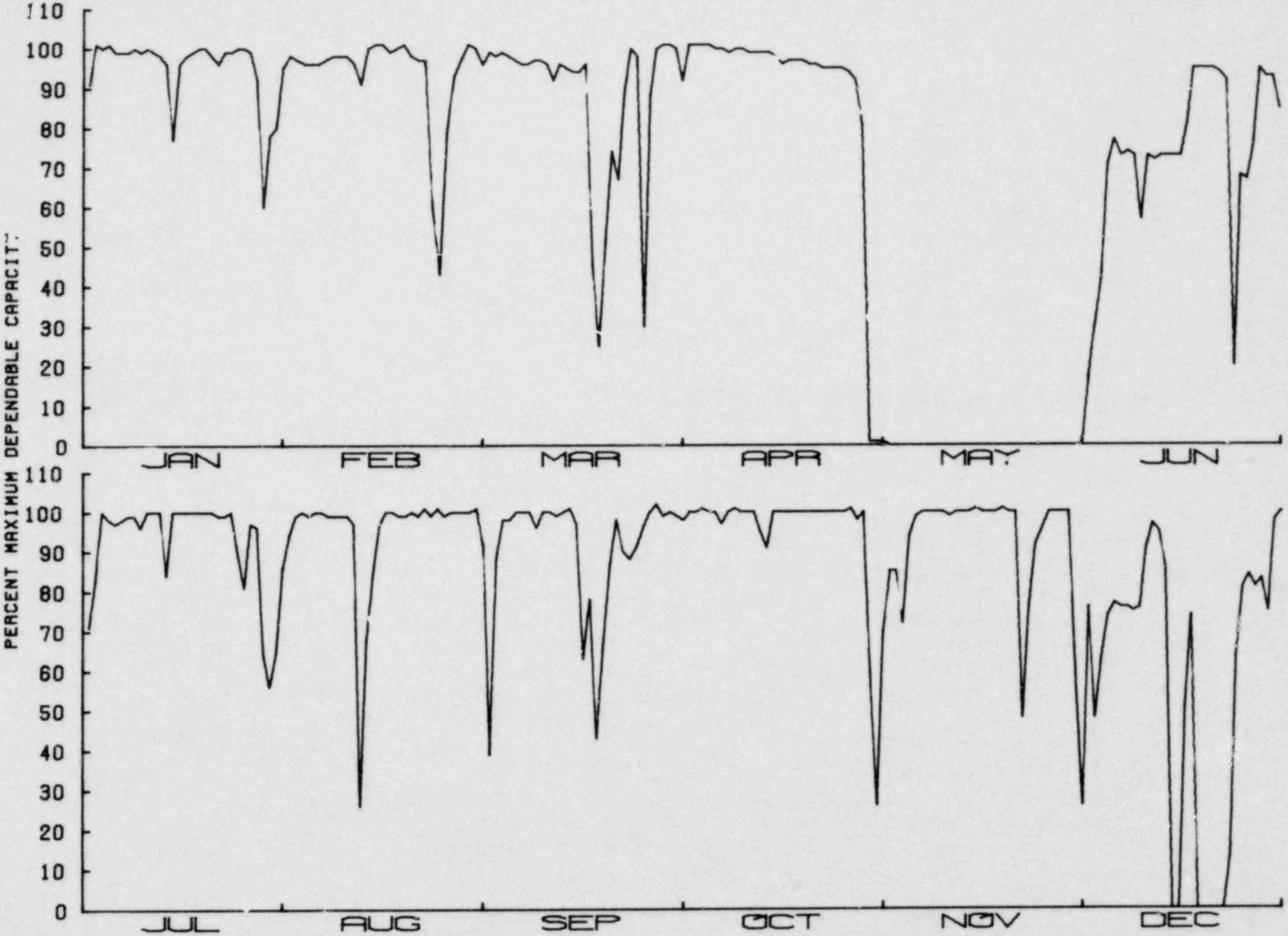
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/21	12	F	Maintenance to combined intermediate valve on L. P. turbine	A	2	Steam and power conversion (HB)	Valves
2)	3/18	12	F	APRM high flux due to pressure regulator problems	A	3	Steam and power conversion (HA)	Instrumentation and controls
3)	3/25	11	F	APRM high flux due to pressure regulator problems	A	3	Steam and power conversion (HA)	Instrumentation and controls
4)	4/27	817	S	Refueling	C	1	Reactor (RC)	Fuel elements
5)	6/23	17	F	Turbine trip while testing master trip solenoid valves	A	1	Steam and power conversion (HA)	Valves
6)	7/29	10	F	EHC oil leak	A	1	Steam and power conversion (HA)	Pipes, fittings

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	8/12	13	F	Maintenance to EHC (oil leak)	A	1	Steam and power conversion (HA)	Pipes, fittings
8)	8/31	1	F	Maintenance to EHC (oil leak)	A	1	Steam and power conversion (HA)	Pipes, fittings
9)	9/1	8	F	Repair EHC oil leak	A	1	Steam and power conversion (HA)	Pipes, fittings
10)	9/17	10	F	Personnel error during performance of Rx low-low water level SI testing	G	3	Instrumentation and controls (IB)	N/A
11)	10/29	21	F	Turbine trip on load rejection from accidental grounding of sudden pressure trip circuit while testing relays in the switch yard	G	3	Electric power (EB)	Relays
12)	11/21	10	F	MSIV closure during testing	G	3	Reactor coolant (CD)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	11/29	21	F	High neutron flux due to main steam pressure regulator malfunction	A	3	Steam and power conversion (HA)	Instrumentation and controls
14)	12/2	9	F	High neutron flux scram due to movement of turbine control valves	A	3	Steam and power conversion (HA)	Valves
15)	12/13	53	F	Maintenance to "A" recirc. pump and FCV-1-5	A	2	Reactor coolant (CB)	Pumps
16)	12/17	138	S	Modifications to primary containment isolation system	B	2	Engineered safety features (SD)	Instrumentation and controls



DESIGN ELEC. RATING = 1065 MAX. DEPEND. CAP. = 1065 (100%) BROWNS FERRY 2

BROWNS FERRY 3

I. Summary

Description		Performance		Outages		
Location: Decatur, Alabama		Net Electrical Energy Generated (MWH):	5,482,585	Total No.	11	
Docket No: 50-296				Forced	10	
Reactor Type: BWR		Unit Availability Factor (%):	65.2	Scheduled	1	
Capacity (MWe-Net): 1,065				Total:	3,052 Hours,	34.8%
Commercial Operation: 3/1/77		Unit Capacity Factor (%) (Using MDC):	58.8	Forced	438 Hours,	5.0%
Plant Age: 3.3 Years				Scheduled	2,614 Hours,	29.8%
		Unit Capacity Factor (%) (Using Design MWE):	58.8			

II. Highlights

Operation throughout the year was routine and near full power except for a refueling outage from August 24 to December 7.

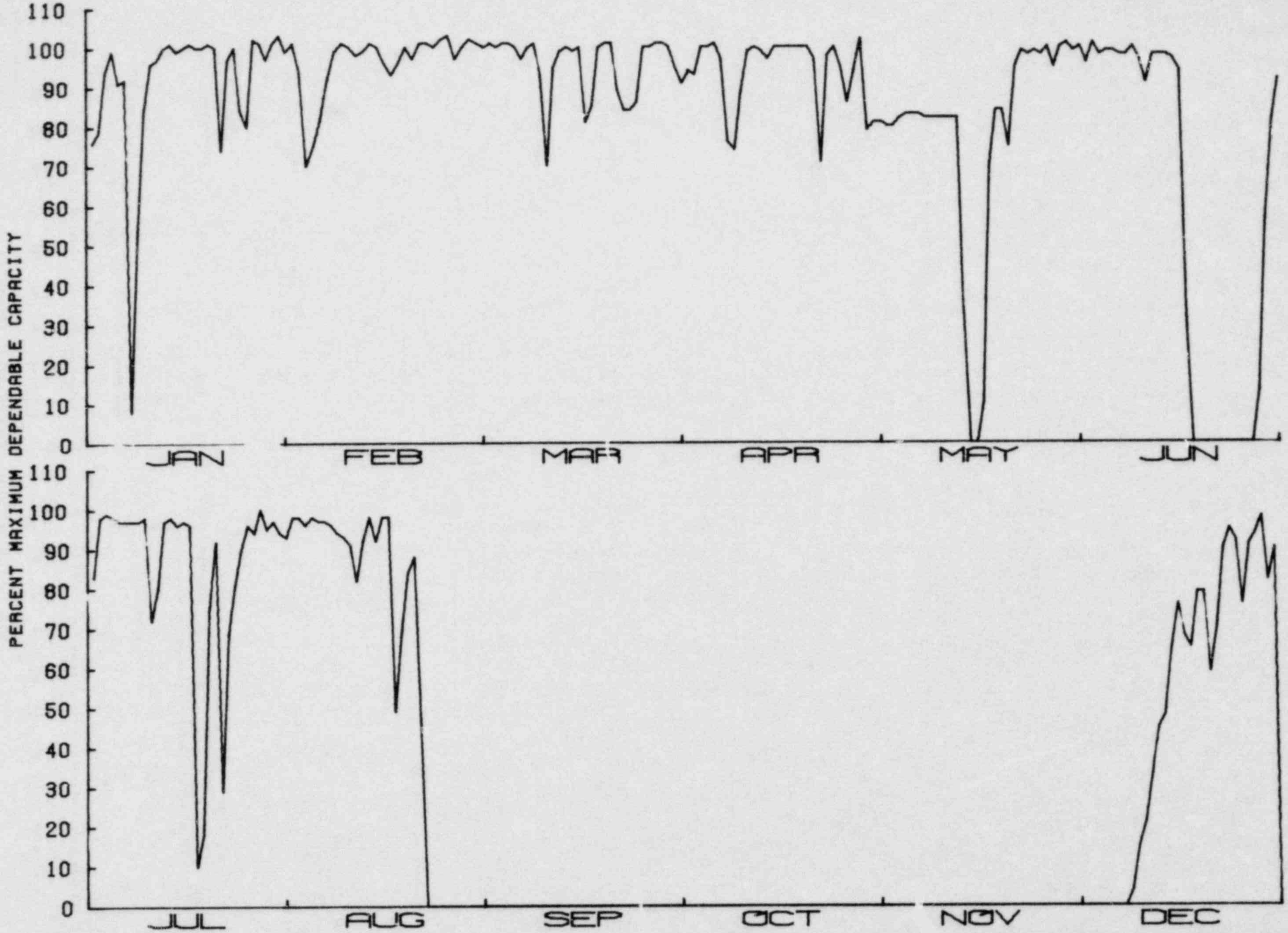
DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/6	9	F	Main steam line high radiation during the performance of testing	A	3	Instrumentation and controls (IB)	Instrumentation and controls
2)	1/6	13	F	EHC vibration problems (Rx was in startup mode)	A	9	Steam and power conversion (HA)	Mechanical function units
3a)	5/13	19*	F	Low Rx water level due to feedwater pump control malfunction	A	3	Reactor coolant (CH)	Instrumentation and controls
3b)	5/13 (cont.)	60*	S	Unit remained down for replacement of torus H_2 sensors	B	4	Engineered safety features (SB)	Instrumentation and controls
4)	6/16	266	F	Upper guide bearings on "A" recirc. pump motor damaged	A	1	Reactor coolant (CB)	Pumps
5)	7/17	9	F	Condenser low vacuum due to a broken weld on a control air valve	A	2	Auxiliary process (PA)	Valves

*Estimated

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	7/17	22	F	High H_2 concentration in the off gas system	A	2	Radioactive waste management (MB)	Other (recombiner)
7)	7/21	10	F	APRM high flux during testing	A	3	Instrumentation and controls (IA)	Instrumentation and controls
8)	8/17	7	F	MSIV closure during performance of testing	G	3	Reactor coolant (CD)	Valves
9)	8/21	57	F	"B" recirculation pump maintenance	A	1	Reactor coolant (CB)	Pumps
10)	8/24	2554	S	Refueling	C	1	Reactor (RC)	Fuel elements
11)	12/30	26	F	Install overhead cables from cooling tower switch gear to bus tie boards	A	2	Electric power (EB)	Electrical conductors



DESIGN ELEC. RATING = 1065   MAX. DEPEND. CAP. = 1065 (100%)   BROWNS FERRY 3

BRUNSWICK 1

I. Summary

Description

Location: Southport, N.C.
Docket No: 50-325
Reactor Type: BWR
Capacity (MWe-Net): 790
Commercial Operation: 3/18/77
Plant Age: 3.1 Years

Performance

Net Electrical Energy Generated (MWH):	3,169,212
Unit Availability Factor (%):	54.6
Unit Capacity Factor (%) (Using MDC):	45.8
Unit Capacity Factor (%) (Using Design MWE):	44.1

Outages

Total No.	21		
Forced	13		
Scheduled	8		
Total:	3,978	Hours,	45.4%
Forced	915	Hours,	10.4%
Scheduled	3,063	Hours,	35.0%

II. Highlights

Refueling was conducted from January 12 to April 16. During a shutdown in May, 41 seismic supports for safety-related piping were modified. In September, another shutdown was effected to inspect hydraulic snubbers. In December, a positive indication system for safety relief valves was added.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/12	2256	S	Refueling	C	1	Reactor (RC)	Fuel elements
2)	4/17	2	S	Turbine overspeed trip test	B	1	Steam and power conversion (HA)	Turbines
3)	4/17	52	F	Turbine runback; a wiring error was found in the stator cooling low flow runback circuit which was installed as a plant modification during the recent refueling outage	A	3	Steam and power conversion (HA)	Electrical conductors
4)	4/19	0.3	S	Turbine electrical overspeed trip circuit recalibration	B	1	Steam and power conversion (HA)	Turbines
5)	4/20	1	S	Turbine electrical backup overspeed test	B	1	Steam and power conversion (HA)	Turbines
6)	4/20	1	S	Turbine electrical backup overspeed test (retested as a result of unsatisfactory test during previous outage)	B	1	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES

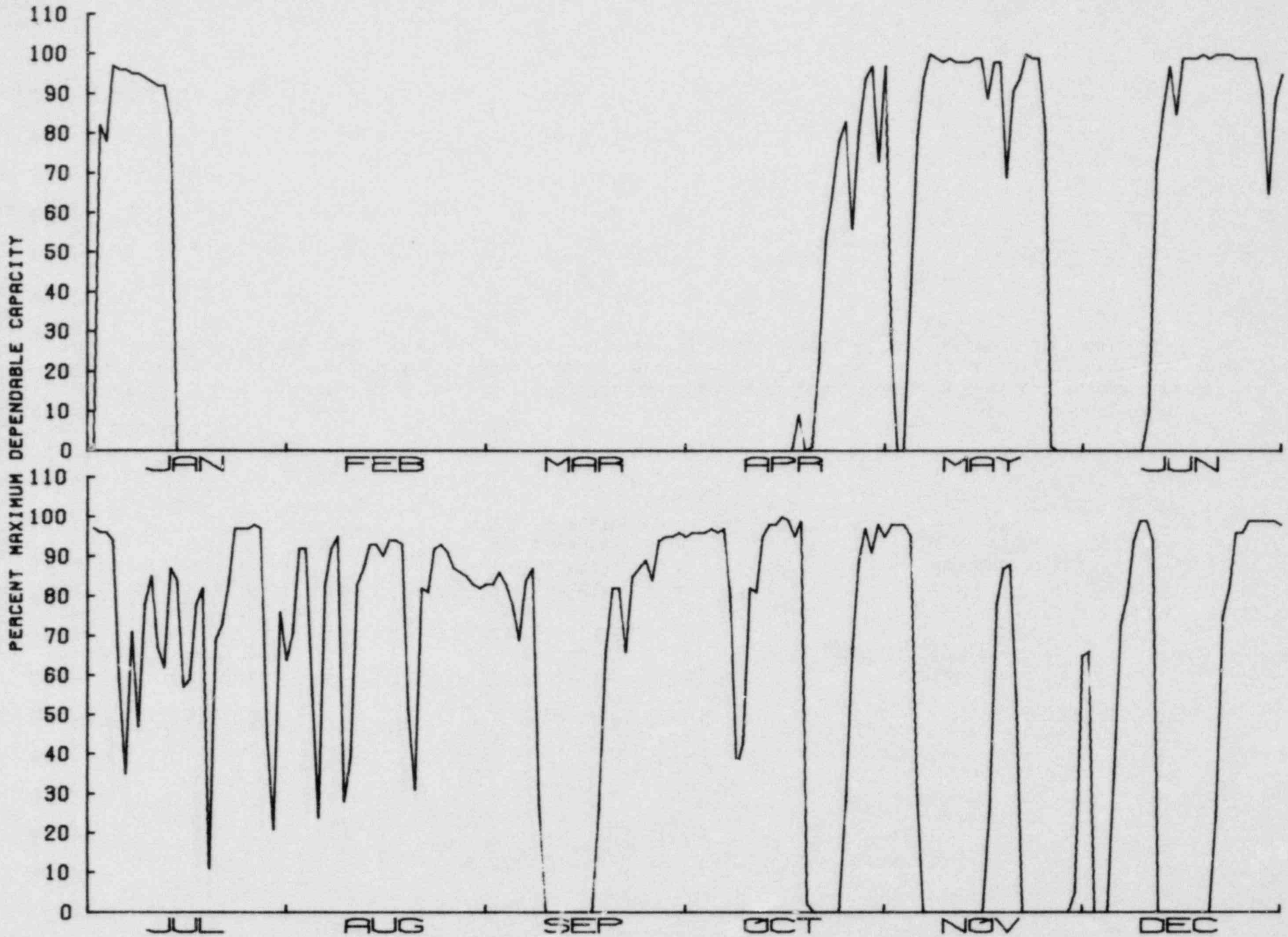
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	5/1	67	F	Flow comparator problems caused by wetting of recirculation flow transmitters in north core spray room	A	3	Reactor coolant (CB)	Instrumentation and controls
8)	5/25	373	S	Pipe support inspections and modifications	D	1	Engineered safety features (SX)	Shock suppressors
9)	7/18	18	F	Reactor scrammed on high APRM flow biased signal. The high signal was caused while placing the B recirculating loop in service following a trip of the B recirculating pump m-g set. The m-g set had tripped on low lube oil pressure following a motor and breaker failure on one of two operating m-g set lube oil pumps. The standby lube oil pump failed to start on loss of the operating pump.	A	3	Reactor coolant (CB)	Motors
10)	7/28	13	F	False low water level signal	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11)	8/4	15	F	Turbine control valve failed shut, causing a pressure spike and average power range monitor increase	A	3	Steam and power conversion (HA)	Valves
12)	8/9	16	F	Turbine control valve failed shut, causing a pressure spike and average power range monitor increase	A	3	Steam and power conversion (HA)	Valves
13)	8/19	12	F	An Rx level instrument low side root valve opened apparently causing pressure perturbations through an instrument diaphragm after testing was performed	A	3	Instrumentation and controls (IA)	Instrumentation and controls
14)	9/8	215	S	Pipe hydraulic snubber inspection in primary containment; generator hydrogen seal repair also necessary due to excessive hydrogen leakage	D	2	Engineered safety feature (SX)	Shock suppressors
15)	10/8	18	F	Reactor scram on low water level due to loss of steam flow signal feedwater controller	A	3	Reactor coolant (CH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
16)	10/19	141	F	Scrammed on main steam line high radiation signal following maintenance due to injection of filter demineralizer resin into vessel	G	3	Reactor coolant (CG)	Valves
17)	11/5	253	F	Runback of master feedwater flow controller during calibration of a steam flow instrument	G	3	Reactor coolant (CH)	Instrumentation and controls
18)	11/20	219	F	Loss of power to emergency buses E1 and E2 due to unstable switchyard voltage condition	A	3	Electric power (EE)	Electrical conductors
19)	12/1	79	F	High drywell leakage from reactor recirculation suction and discharge valves leakoff	A	2	Reactor coolant (CB)	Valves
20)	12/12	215	S	Perform plant modifications to safety relief valves and for a pipe snubber inspection	D	2	Engineered safety features (SH)	Valves
21)	12/12	2	F	Make repairs to the safety relief valve modification	A	1	Engineered safety features (SH)	Valves



DESIGN ELEC. RATING = 821 MAX. DEPENC. CAP. = 790 (100%) BRUNSWICK 1

BRUNSWICK 2

I. Summary

Description	Performance		Outages		
Location: Southport, N.C.	Net Electrical Energy		Total No.	18	
Docket No: 50-324	Generated (MWH):	3,652,260	Forced	11	
Reactor Type: BWR	Unit Availability		Scheduled	7	
Capacity (MWe-Net): 790	Factor (%):	65.6	Total:	3,017 Hours,	34.4%
Commercial Operation: 11/3/75	Unit Capacity Factor (%)		Forced	654 Hours,	7.5%
Plant Age: 4.7 Years	(Using MDC):	52.8	Scheduled	2,363 Hours,	26.9%
	Unit Capacity Factor (%)				
	(Using Design MWE):	50.8			

II. Highlights

Refueling was conducted from March 2 to May 19. During a shutdown in June, 47 seismic supports for safety-related piping were modified. Several other shutdowns were required for inspection of hydraulic snubbers. At the end of the year, the unit was in shutdown for modification of the safety relief valves.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/29	14	F	APRM high flux scram caused by pressure spike which resulted from MSIV "A" failing closed due to an apparent stem-disc separation	A	3	Reactor coolant (CD)	Valves
2)	2/4	7	F	Level and pressure control instability possibly due to turbine control valve movement	A	3	Steam and power conversion (HA)	Valves
3a)	3/2	1530	S	Refueling	C	1	Reactor (RC)	Fuel elements
3b)	3/2 (cont.)	336	F	Core spray pipe replacement material problems	H	4	Engineered safety features (SF)	Pipes, fittings
4)	5/19	1	S	Turbine overspeed trip test	B	1	Steam and power conversion (HA)	Turbines
5)	5/21	32	F	Cause unknown; scram occurred during test of instruments	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

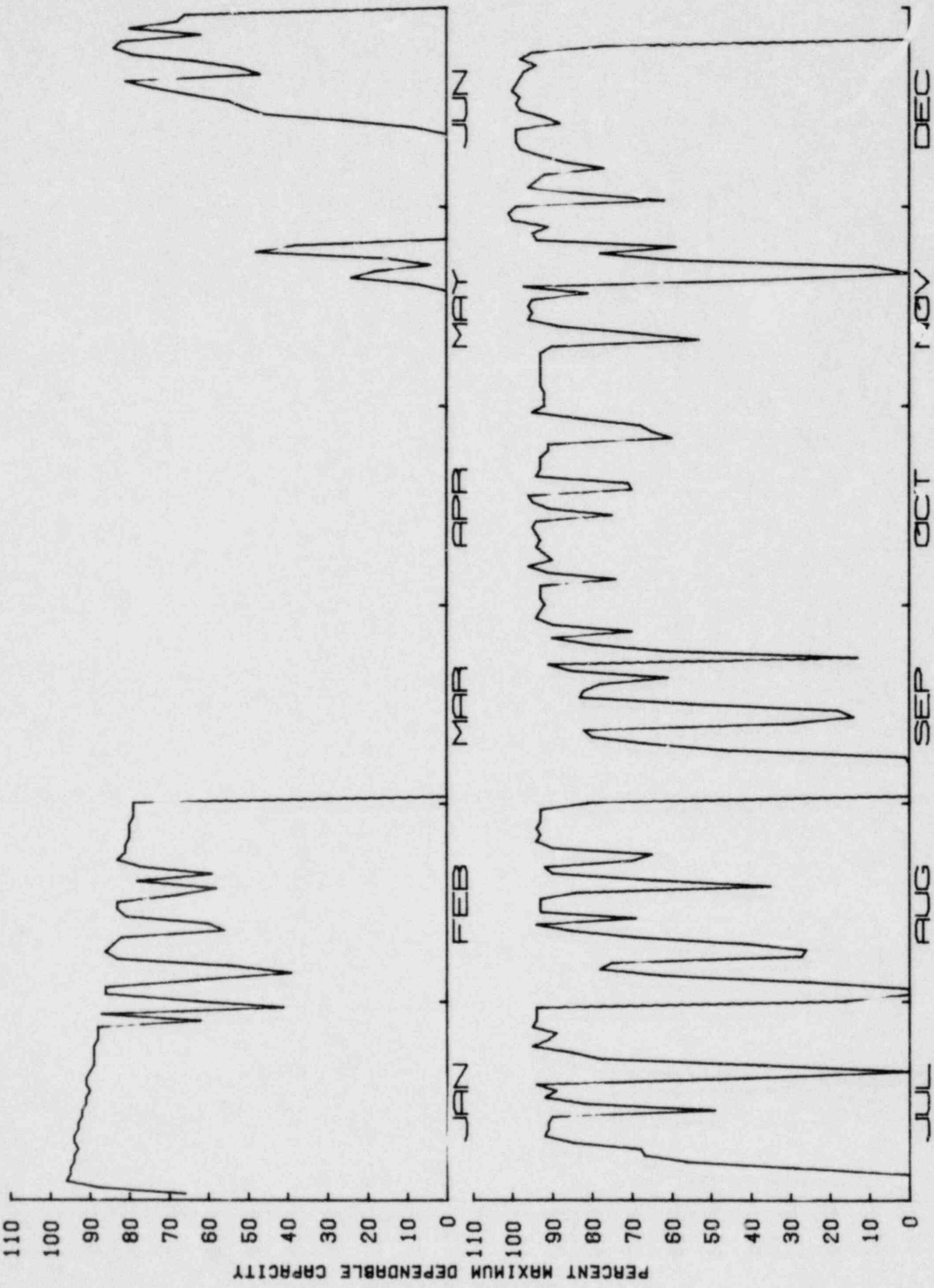
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	5/23	14	F	Low water levels due to operator error with feed pumps	G	3	Reactor coolant (CH)	Instrumentation and controls
7)	5/25	408	S	Pipe support inspections and modification	B	1	Engineered safety features (SF)	Shock suppressors
8)	6/12	17	F	MSIV closure caused by blown fuses on F022D resulting from a wiring error	A	3	Reactor coolant (CD)	Electrical conductors
9)	6/29	126	S	Pipe support inspections and modifications	B	2	Engineered safety features (SX)	Shock suppressors
10)	7/19	42	F	Safety valve malfunction	A	1	Engineered safety features (SF)	Valves
11a)	7/31	20	F	Circulating water intake pump trip followed by a turbine trip, due to a circulation water pipe leak spraying on a relay	A	3	Steam and power conversion (HF)	Pipes, fittings

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11b)	7/31 (cont.)	56	F	Extended shutdown due to HPCI valve motor having burned up 7/21/79 and still not received on site	A	4	Engineered safety features (SF)	Valve operators
12)	8/31	142	S	Pipe support inspections and modifications	D	2	Engineered safety features (SX)	Shock suppressors
13)	9/7	22	F	Steam leak in a steam line drain valve in reactor building, due to valve travel limit switch failure to operate properly	A	1	Reactor coolant (CC)	Valves
14)	9/12	25	F	Nuclear service water leak caused by defective cement lining on inside of pipe	A	1	Auxiliary water (WA)	Pipes, fittings
15)	9/14	17	F	Apparent load rejection	A	3	Steam and power conversion (HA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
16)	9/22	12	S	Stuck detector from channel 7 of transversing in-core probe "D"	B	1	Instrumentation and controls (ID)	Instrumentation and controls
17)	11/19	52	F	High pressure signal during cleaning in the instrument rack area	G	3	Instrumentation and controls (IA)	Instrumentation and controls
18)	1[illegible]/25	144	S	Perform plant modifications to the safety relief valves	D	1	Engineered safety features (SH)	Valves


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 821
MAX. DEPEND. CAP. = 790 (100%)
BRUNSWICK 2

CALVERT CLIFFS 1

I. Summary

Description

Location: Lusby, Maryland
Docket No: 50-317
Reactor Type: PWR
Capacity (MWe-Net): 810
Commercial Operation: 5/8/75
Plant Age: 5.0 Years

Performance

Net Electrical Energy Generated (MWH):	4,194,218
Unit Availability Factor (%):	70.3
Unit Capacity Factor (%) (Using MDC):	59.1
Unit Capacity Factor (%) (Using Design MWE):	56.7

Outages

Total No.	15		
Forced	13		
Scheduled	2		
Total:	2,606	Hours,	29.7%
Forced	1,012	Hours,	11.5%
Scheduled	1,594	Hours,	18.2%

II. Highlights

At the beginning of the year, the unit was still shut down because of damaged blades in the first-stage high-pressure turbine. Routine power operation resumed on January 18 and continued until April 21 when a refueling outage began. During the outage, extensive repairs were made to the turbine. The unit resumed operation on July 14. On November 8 and continuing through the remainder of the year, a forced power reduction to 50–60% of full power was necessary because of unequal power distribution in the core.

DETAILS OF PLANT OUTAGES

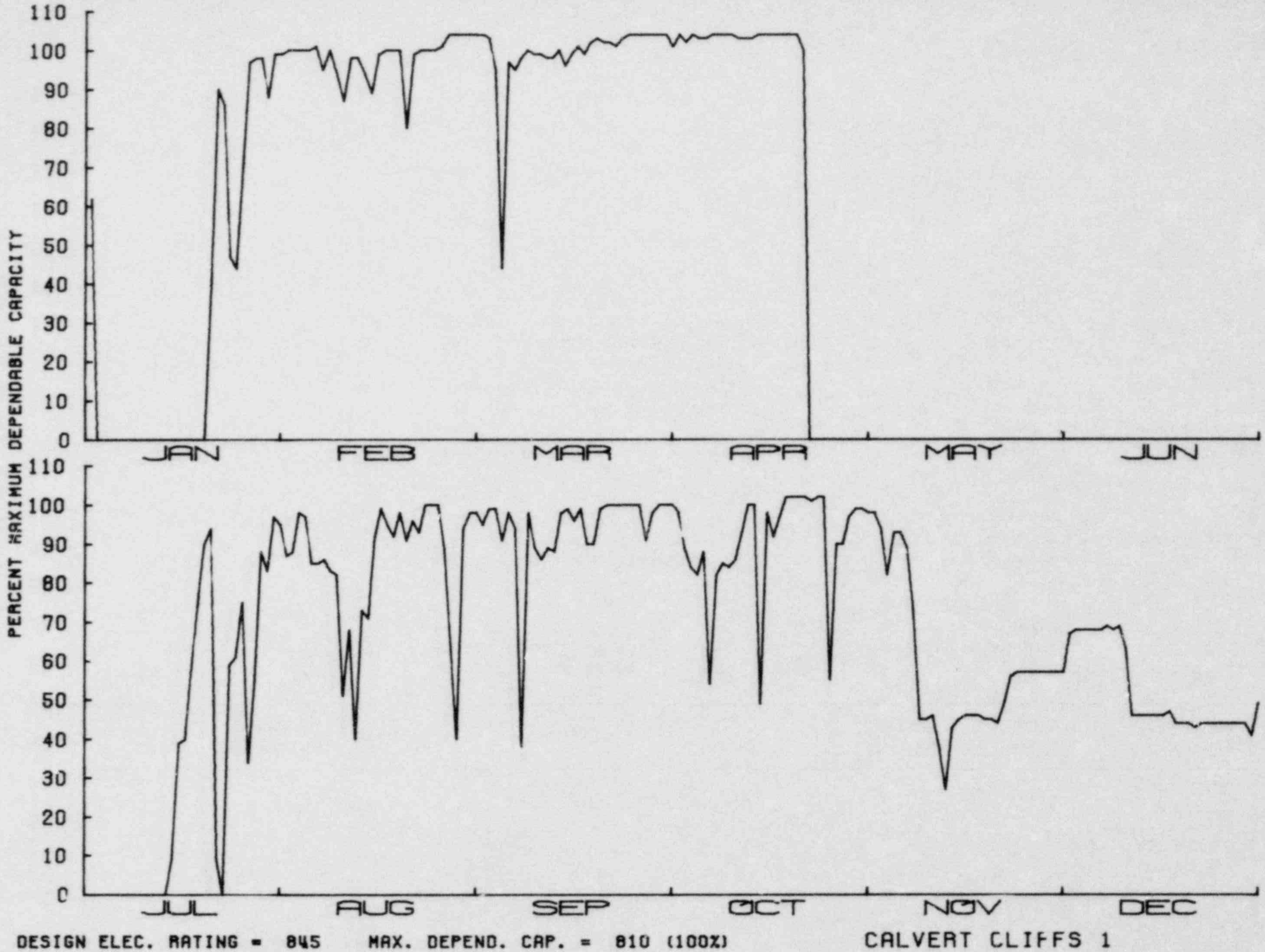
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	12/17/78 (cont.)	430	F	Vibration on high-pressure turbine	A	4	Steam and power conversion (HA)	Turbines
2)	1/22	5	F	High water level in No. 12B feedwater heater	A	3	Steam and power conversion (HH)	Instrumentation and controls
3)	3/4	9	F	CVC-515-CV leak-off plug was leaking	A	1	Auxiliary process (PC)	Valves
4a)	4/21	1570	S	Inspection and refueling	C	1	Reactor (RC)	Fuel elements
4b)	6/25	463	F	Late return from previous scheduled outage (due to turbine repair)	A	4	Steam and power conversion (HA)	Turbines
5)	7/21	24	S	Turbine overspeed test	B	1	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	7/22	15	F	Loss of field to the exciter	A	3	Steam and power conversion (HA)	Generators (exciter)
7)	7/26	17	F	High water levels in feedwater heater	A	3	Steam and power conversion (HH)	Instrumentation and controls
8)	8/10	8	F	Loss of a circulating water pump	A	3	Steam and power conversion (HF)	Pumps
9)	8/12	7	F	Low steam generator level	A	3	Steam and power conversion (HH)	Instrumentation and controls
10)	8/27	16	F	Leak in chemical and volume control system	A	1	Auxiliary process (PC)	Pipes, fittings
11)	9/6	9	F	Failed differential pressure controller on No. 12 feedwater regulating valve	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12)	10/6	6	F	Loss of power to No. 12 generator feed pump speed control circuit	A	3	Steam and power conversion (HH)	Electrical conductors
13)	10/14	8	F	Extraction steam line leak	A	1	Steam and power conversion (HJ)	Pipes, fittings
14)	10/25	8	F	Circulating water pump power loss	A	2	Steam and power conversion (HF)	Electrical conductors
15)	11/11	11	F	Loss of coolant flow due to a faulty breaker relay	A	3	Reactor coolant (CB)	Relays


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 845 MAX. DEPEND. CAP. = 810 (100%)
CALVERT CLIFFS 1

CALVERT CLIFFS 2

I. Summary

Description	Performance		Outages		
Location: Lusby, Maryland	Net Electrical Energy Generated (MWH):	5,488,991	Total No.	15	
Docket No: 50-318			Forced	10	
Reactor Type: PWR	Unit Availability Factor (%):	77.6	Scheduled	5	
Capacity (MWe-Net): 825			Total:	1,963 Hours,	22.4%
Commercial Operation: 4/1/77	Unit Capacity Factor (%) (Using MDC):	76.0	Forced	654 Hours,	7.5%
Plant Age: 3.1 Years			Scheduled	1,309 Hours,	14.9%
	Unit Capacity Factor (%) (Using Design MWE):	74.2			

II. Highlights

Operation during the year was routine. Between October 12 and December 6, refueling and replacement of a reactor coolant pump seal were accomplished.

DETAILS OF PLANT OUTAGES

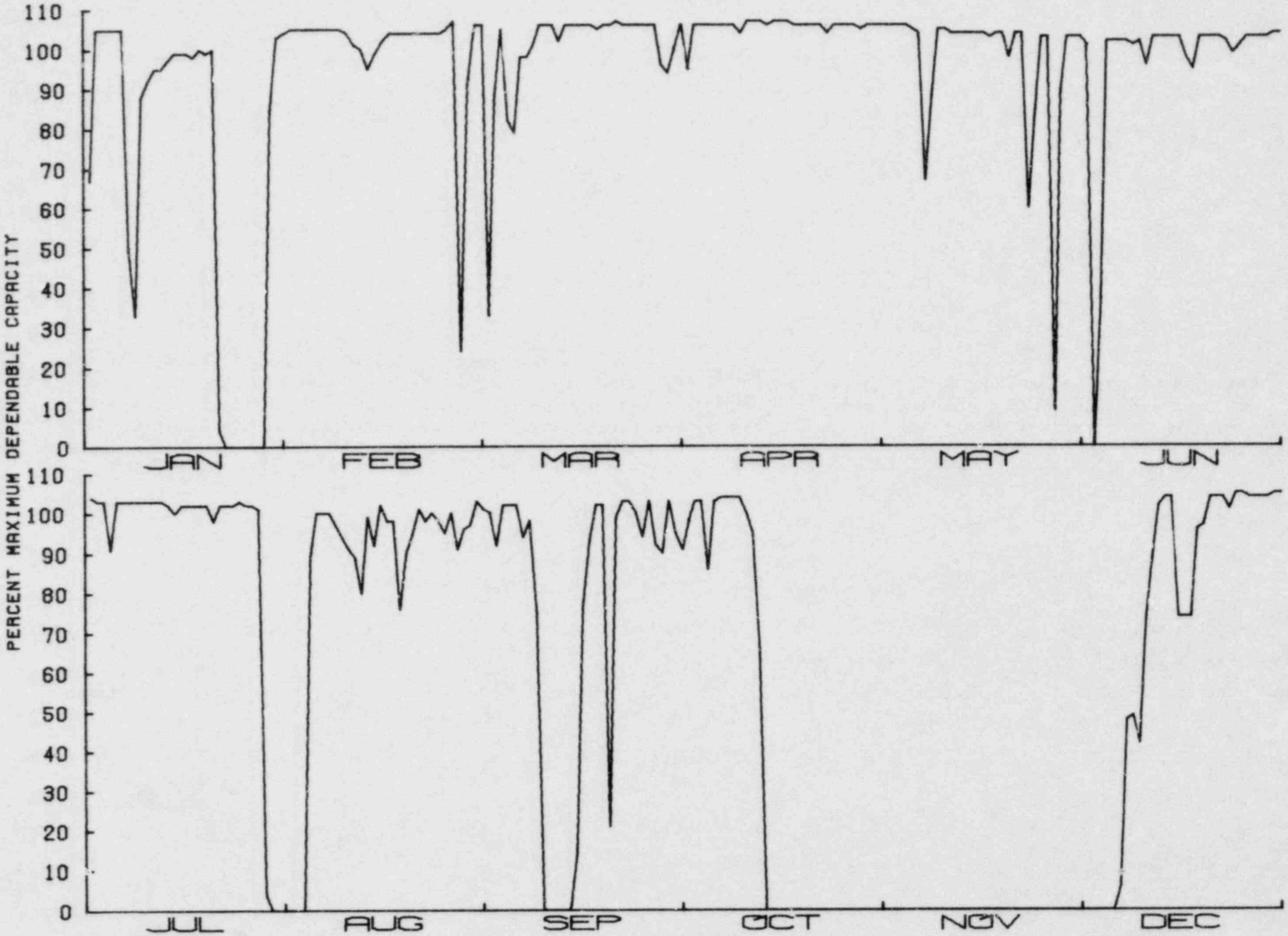
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/6	23	F	Cracked weld on the No. 21A reactor coolant pump middle seal pressure sensing line	A	1	Reactor coolant (CB)	Pipes, fittings
2a)	1/20	24	F	Cracked weld on the No. 22A reactor coolant pump lower seal pressure sensing line	A	2	Reactor coolant (CB)	Pipes, fittings
2b)	1/20 (cont.)	167	S	Testing and for replacement of faulty seals on reactor coolant pumps	B	4	Reactor coolant (CB)	Pumps
3)	2/25	18	S	Repair leaking feedwater check valve	B	1	Steam and power conversion (HH)	Valves
4)	3/1	15	F	Low water level in No. 21 steam generator	A	3	Steam and power conversion (HH)	Instrumentation and controls
5)	3/4	6	S	Repair No. 22 feedwater check valve	B	1	Steam and power conversion (HH)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	5/7	9	F	Blown fuse on DC power to No. 21 inverter	A	3	Electric power (EB)	Generators (inverters)
7)	5/23	11	F	Repair oil leak on unit transformer	A	1	Electric power (EB)	Transformers
8)	5/27	19	F	Repair oil leak on unit transformer	A	1	Electric power (EB)	Transformers
9)	6/2	34	S	Replace governor control valve	B	1	Steam and power conversion (HA)	Valves
10)	7/28	15	S	Furmanite feedwater check valve	B	9	Steam and power conversion (HH)	Valves
11)	7/28	144	F	Condenser tube leaks	A	1	Steam and power conversion (HC)	Heat exchangers
12)	9/8	140	F	Failure of capacitor in No. 21B reactor coolant pump motor	A	3	Reactor coolant (CB)	Motors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	9/19	17	F	Loss of 21 main feedwater pump speed controller	A	3	Steam and power conversion (HH)	Instrumentation and controls
14a)	10/12	4	F	Trip during low vacuum trip test	B	3	Steam and power conversion (HA)	Instrumentation and controls
14b)	10/12 (cont.)	1068	S	Refueling; plant was already shut down due to previous unit trip	C	4	Reactor (RC)	Fuel elements
14c)	10/12 (cont.)	248	F	Replace seal on reactor coolant pump 21B and 22B	A	4	Reactor coolant (CB)	Pumps
15)	12/10	1	S	Overspeed trip test on the turbine	B	2	Steam and power conversion (HA)	Turbines



DESIGN ELEC. RATING = 845   MAX. DEPEND. CAP. = 810 (100%)   CALVERT CLIFFS 2

COOK 1

I. Summary

Description

Location: Bridgman, Michigan
Docket No: 50-315
Reactor Type: PWR
Capacity (MWe-Net): 1,044
Commercial Operation: 8/27/75
Plant Age: 4.9 Years

Performance

Net Electrical Energy Generated (MWH):	5,660,137
Unit Availability Factor (%):	64.7
Unit Capacity Factor (%) (Using MDC):	61.9
Unit Capacity Factor (%) (Using Design MWE):	61.3

Outages

Total No.	8	
Forced	6	
Scheduled	2	
Total:	3,091 Hours,	35.3%
Forced	1,623 Hours,	18.5%
Scheduled	1,468 Hours,	16.8%

II. Highlights

Operation was routine until the refueling outage was started April 6. The outage was extended so that all of the connecting 16-in. elbows from the feedwater lines to the four steam generators could be replaced. On June 18, the unit resumed power operation. The year ended with the unit shut down because of design deficiencies in the containment hydrogen skimmer system.

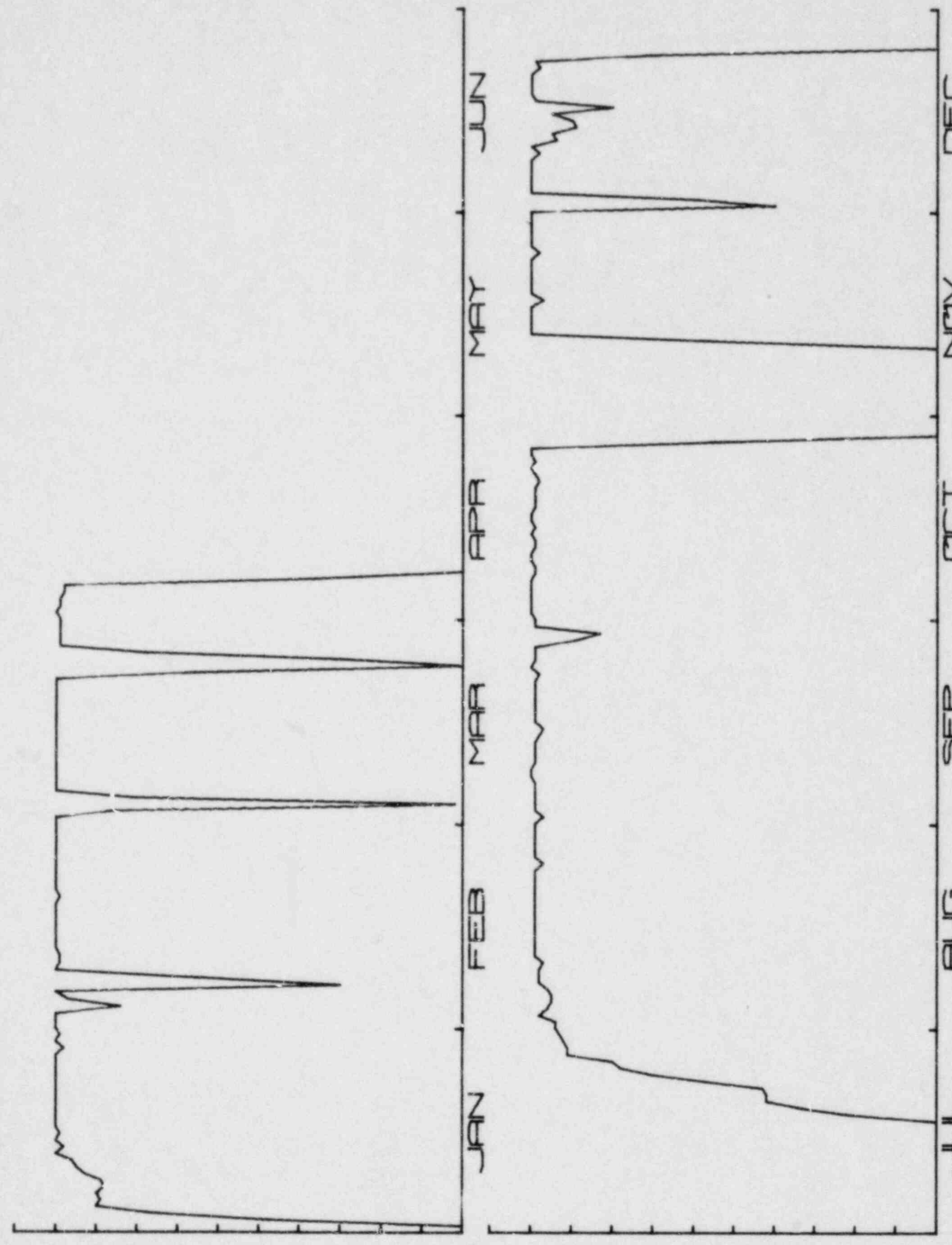
DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/6	19	F	Pressurizer relief tank rupture disc failed during a feed and bleed maneuver to reduce water temperature and caused all ice condenser doors to indicate open	A	1	Auxiliary process (PC)	Valves
2)	3/2	25	F	Failure of the rupture disc on the pressurizer relief tank, causing all ice condenser inlet doors to indicate open	A	1	Auxiliary process (PC)	Valves
3)	3/23	40	F	Failure of two vital instrument bus inverters; the inverter failures also caused inadvertent actuation of the safety injection systems and steam line isolation	A	3	Electric power (ED)	Generators (inverters)
4a)	4/6	1466*	S	Refueling	C	3	Reactor (RC)	Fuel elements
4b)	4/6 (cont.)	1000*	F	Outage was extended to replace all connecting elbows from F/W system to the 4 S/G s	A	4	Steam and power conversion (HH)	Pipes, fittings

*Estimated

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
5)	7/18	2	S	Turbine overspeed trip testing	B	1	Steam and power conversion (HA)	Turbines
6)	10/27	350	F	Repair No. 4 inverter, add oil to No. 4 coolant pump motor upper oil reservoir, investigate high vibration on coolant pump No. 2, and repair leak in stator cooling water system of main generator	A	1	Steam and power conversion (HA)	Pipes, fittings
7)	12/1	14	F	While working on rod control system to clear a "rod control urgent failure" alarm, a wrong card was pulled, dropping the rods in that group, which caused a "negative rate" reactor trip	G	3	Reactor (RB)	Instrumentation and controls
8)	12/24	175	F	Significant non-conformance identified during inspection/evaluation program performed in accordance with IE bulletin. Design deficiencies in the containment Hydrogen Skimmer system	D	1	Engineered safety features (SE)	Shock suppressors



PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 1054 MAX. DEPEND. CAP. = 1044 (100%)
COOK 1

I. Summary

Description

Location: Bridgman, Michigan
Docket No: 50-316
Reactor Type: PWR
Capacity (MWe-Net): 1,082
Commercial Operation: 7/1/78
Plant Age: 1.8 Years

Performance

Net Electrical Energy Generated (MWH):	1,082
Unit Availability Factor (%):	65.9
Unit Capacity Factor (%) (Using MDC):	62.8
Unit Capacity Factor (%) (Using Design MWE):	61.8

Outages

Total No.	11		
Forced	8		
Scheduled	3		
Total:	2,986	Hours,	34.1%
Forced	1,201	Hours,	13.7%
Scheduled	1,785	Hours,	20.4%

II. Highlights

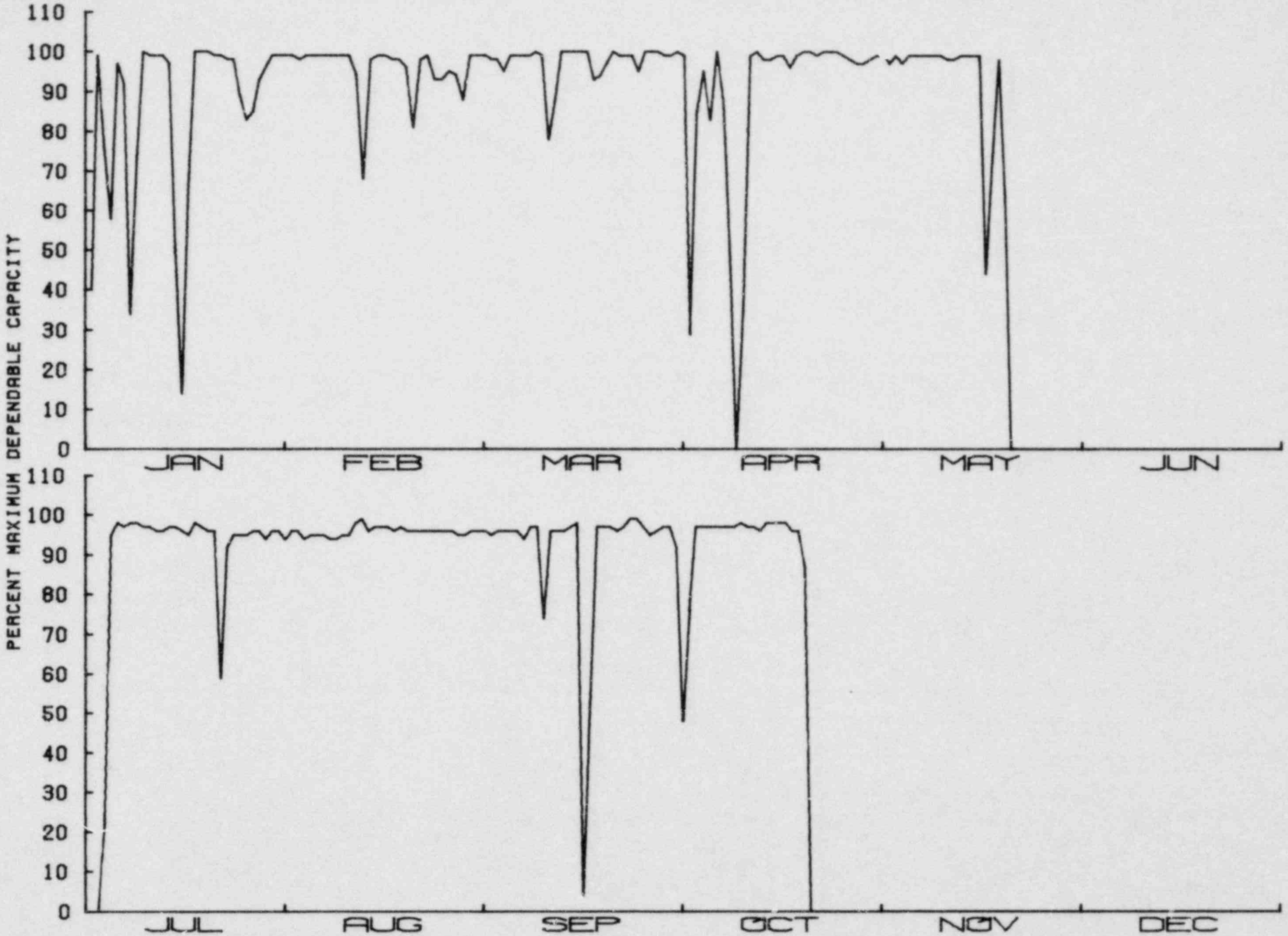
The unit operated routinely until May 19 when a shutdown was effected to replace the 16-in. elbow from the feedwater lines to the four steam generators. The shutdown ended July 3, and routine operation was resumed. On October 19, the first refueling began. At the end of the year, the unit was still shut down, and seismic-related modifications were being made to safety-related piping in accordance with IE Bulletin 79-14.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/3	6	F	Feedwater isolation valves inadvertently closed	G	1	Steam and power conversion (HH)	Instrumentation and controls
2)	1/6	16	F	Safety injection actuation due to indicated "high" steam line differential pressure	A	3	Instrumentation and controls (IB)	Instrumentation and controls
3)	1/13	9	F	Drop in "A" condenser vacuum caused by multiple tube failures	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
4)	1/14	22	F	Main transformer phase 2 ground fault due to ice buildup on bus support insulator	A	3	Electric power (EB)	Transformers
5)	4/1	10	F	High level in No. 1 steam generator	A	3	Steam and power conversion (HH)	Instrumentation and controls
6)	4/7	46	F	Repair oil level alarm device on No. 1 reactor coolant pump upper oil reservoir	A	1	Reactor coolant (CB)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	5/16	15	F	Steam flow/feed mismatch	A	3	Steam and power conversion (HH)	Instrumentation and controls
8)	5/19	1077	F	Repair cracks in 16-in. feedwater elbows	A	1	Steam and power conversion (HH)	Pipes, fittings
9)	7/21	4	S	Collect data on feedwater elbow/ steam generator nozzle test instrumentation	B	2	Steam and power conversion (HH)	Pipes, fittings
10)	9/15	27	S	Low oil level alarms on reactor coolant pump motor bearing oil reservoirs	B	1	Reactor coolant (CB)	Motors
11)	10/19	1754	S	Refueling, maintenance, and design changes	C	1	Reactor (RC)	Fuel elements



DESIGN ELEC. RATING = 1100   MAX. DEPEND. CAP. = 1082 (100%)   COOK 2

COOPER

I. Summary

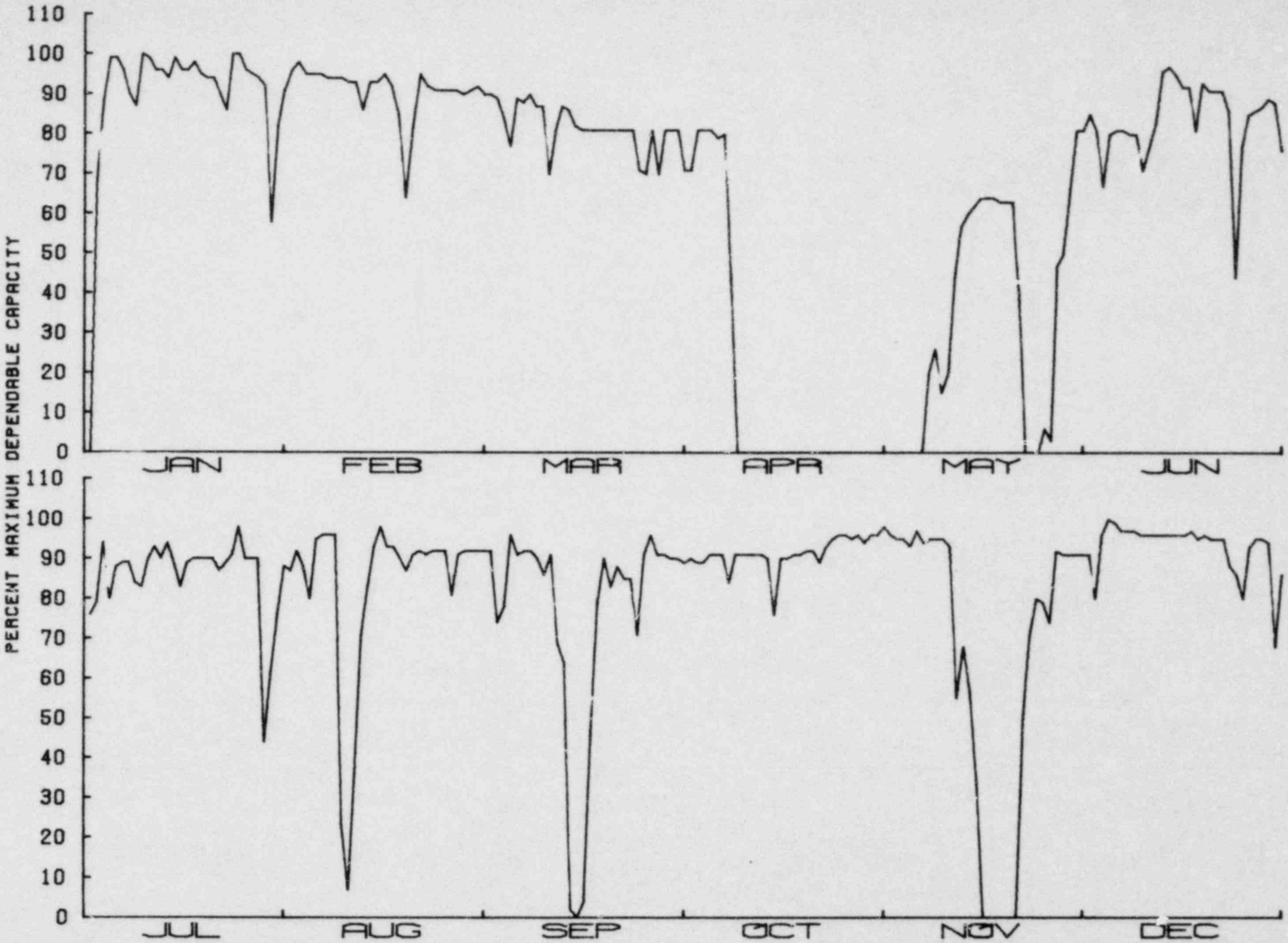
Description	Performance		Outages			
Location: Brownville, Nebraska	Net Electrical Energy Generated (MWH):	4,994,938	Total No.	7		
Docket No: 50-298			Forced	5		
Reactor Type: BWR	Unit Availability Factor (%):	87.6	Scheduled	2		
Capacity (MWe-Net): 764			Total:	1,086	Hours,	12.4%
Commercial Operation: 7/1/74	Unit Capacity Factor (%) (Using MDC):	74.6	Forced	305	Hours,	3.5%
Plant Age: 5.6 Years			Scheduled	781	Hours,	8.9%
	Unit Capacity Factor (%) (Using Design MWE):	73.3				

II. Highlights

Operation was routine throughout the year. Refueling was conducted from April 7 to May 7. There were 7 months in which no outages occurred, 3 months being sequential – January, February, and March.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	4/7	707	S	Refueling	C	2	Reactor (RC)	Fuel elements
2)	5/9	?	F	Main turbine control system malfunction	A	3	Steam and power conversion (HA)	Turbines
3)	5/21	74	S	Repair reactor feed pump suction valve	B	2	Reactor coolant (CH)	Valves
4)	5/25	37	F	Reactor recirculation motor generator set malfunctioned	A	3	Reactor coolant (CB)	Generators (motor generator)
5)	8/9	34	F	Condensate pump expansion bolt failed, causing partial feedwater loss	A	3	Steam and power conversion (HH)	Pumps
6)	9/13	63	F	Replace recirculation pump "B" seal	A	3	Reactor coolant (CB)	Pumps
7)	11/14	149	F	Inspect and repair diesel generators 1 and 2	A	2	Electric power (EE)	Engines (diesel)



DESIGN ELEC. RATING = 778 MAX. DEPEND. CAP. = 764 (100%) COOPER STATION

CRYSTAL RIVER 3

I. Summary

Description		Performance		Outages		
Location: Red Level, Florida		Net Electrical Energy Generated (MWH):	3,761,775	Total No.	15	
Docket No: 50-302				Forced	14	
Reactor Type: PWR		Unit Availability Factor (%):	58.9	Scheduled	1	
Capacity (MWe-Net): 797				Total:	3,600 Hours,	41.1%
Commercial Operation: 3/13/77		Unit Capacity Factor (%) (Using MDC):	53.9	Forced	2,205 Hours,	25.2%
Plant Age: 2.9 Years				Scheduled	1,395 Hours,	15.9%
		Unit Capacity Factor (%) (Using Design MWE):	52.1			

II. Highlights

Operation was routine during the year. A refueling outage began on April 23 and later was extended for repair of reactor coolant pump seals and inspection of pipe base plates using concrete expansion bolts in accordance with IE Bulletin 79-02. In August, another outage was required for repair of a reactor coolant pump seal.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/6	17	F	Momentary high level instrumentation signal (from condensate heat exchanger)	A	2	Steam and power conversion (HH)	Instrumentation and controls
2)	1/17	23	F	Flooding of turbine building basement caused by a circulating water valve failing open	A	2	Steam and power conversion (HF)	Valves
3)	1/30	8	F	Main feed pump FWP-2B failed	A	3	Steam and power conversion (HH)	Pumps
4)	2/28	22	F	Suspect momentary high level instrumentation signal from low pressure heater	A	3	Steam and power conversion (HH)	Instrumentation and controls
5)	3/4	361	F	Repair extraction steam line expansion joints in "B" condenser	A	1	Steam and power conversion (HJ)	Pipes, fittings
6a)	4/23	1395*	S	Refueling	C	1	Reactor (RC)	Fuel elements

*Estimated

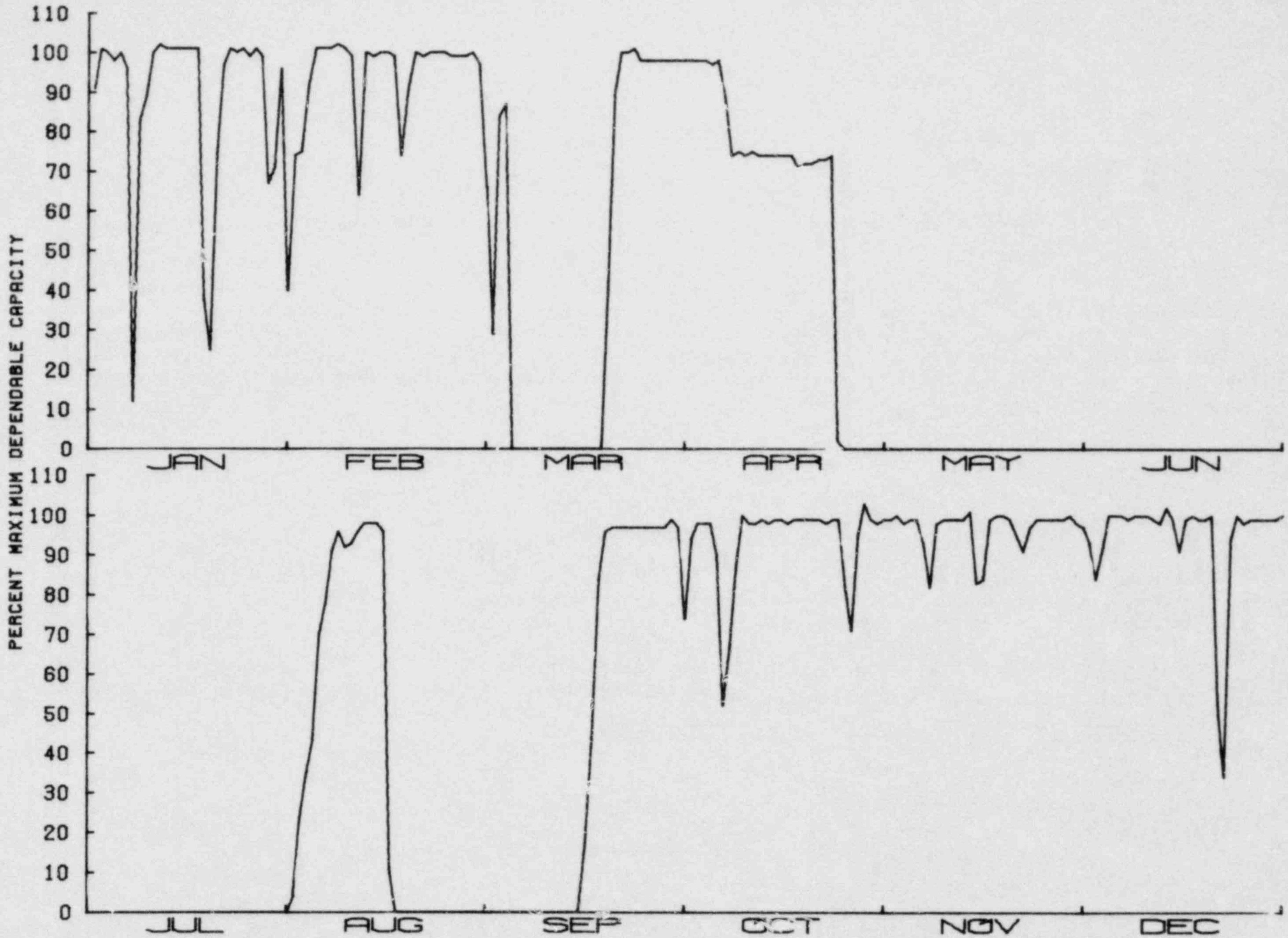
DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6b)	4/23 (cont.)	1000*	F	Repair coolant pump seals	A	4	Reactor coolant (CB)	Pumps
7)	8/1	2	F	Replace failed test valve MSV-409	A	1	Steam and power conversion (HB)	Valves
8)	8/1	18	F	Replace position indication tube for rod 7-4	A	1	Reactor (RB)	Control rod drives
9)	8/16	7	F	Pressure transient during shutdown of reactor coolant pump "C"	A	3	Reactor coolant (CB)	Pumps
10)	8/16	18	F	High RC pressure due to FW oscillation	A	1	Steam and power conversion (HH)	Instrumentation and controls
11)	8/17	11	F	High RC pressure due to FW oscillation	A	3	Steam and power conversion (HH)	Instrumentation and controls

*Estimated

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12)	8/17	680	F	High RC pressure due to FW oscillation; remained off line to repair reactor coolant pump "C" seal and to repair tubes in "B" steam generator	A	3	Steam and power conversion (HB)	Heat exchangers
13)	9/15	4	F	Welded cap failed on 3/4" instrument connection valve on main steam chest cross under line	A	3	Steam and power conversion (HA)	Valves
14)	9/15	10	F	Spurious runback on feedwater pump "B"	A	1	Steam and power conversion (HH)	Instrumentation and controls
15)	12/21	24	F	High pressure trip	A	3	Instrumentation and controls (IA)	Instrumentation and controls



DESIGN ELEC. RATING = 825 MAX. DEPEND. CAP. = 797 (100%) CRYSTAL RIVER 3

DAVIS-BESSE 1

I. Summary

Description	Performance		Outages			
Location: Oak Harbor, Ohio	Net Electrical Energy Generated (MWH):	3,129,118	Total No.	14		
Docket No: 50-346			Forced	8		
Reactor Type: PWR	Unit Availability Factor (%):	67.0*	Scheduled	6		
Capacity (MWe-Net): 906			Total:	4,618	Hours,	52.7%
Commercial Operation: 11/20/77	Unit Capacity Factor (%) (Using MDC):	39.4	Forced	914	Hours,	10.4%
Plant Age: 2.3 Years	Unit Capacity Factor (%) (Using Design MWE):	39.4	Scheduled	3,704	Hours,	42.3%

II. Highlights

On March 30, a maintenance outage was initiated and later was extended to July 12 for modifications required at all Babcock and Wilcox plants as a result of the TMI-2 accident. A loss of offsite power on October 15 adversely affected the reactor coolant pump seals, requiring replacement of the seals on four pumps. The unit resumed power operation on November 20, but on November 30 it was shut down again for the remainder of the year to replace bosses for the resistance temperature detectors (RTDs).

*Includes 1,728 h of unit reserve shutdown hours equal to 19.7% availability.

DETAILS OF PLANT OUTAGES

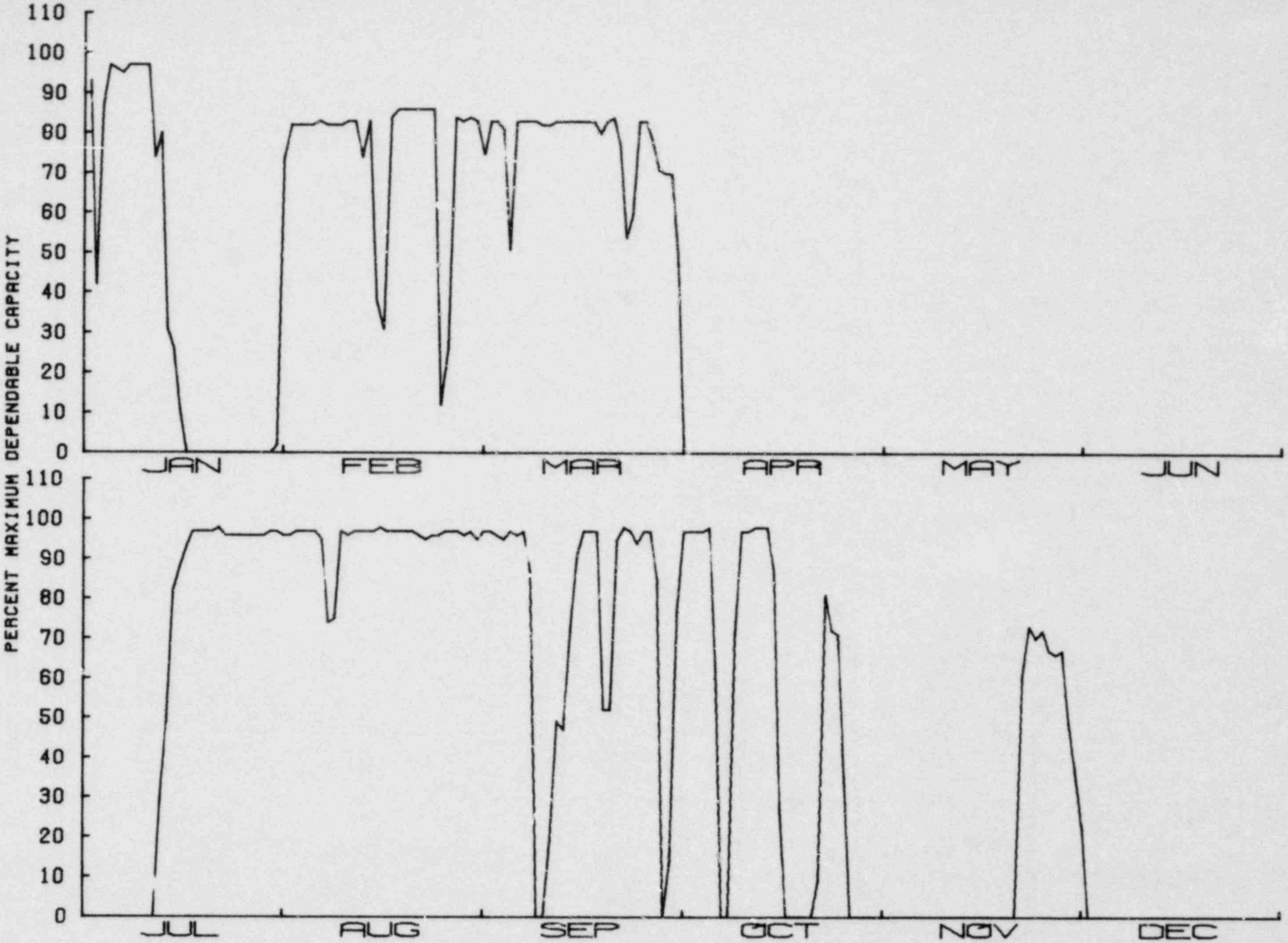
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	12/16/78 (cont.)	4	F	Repair extraction steam line bellows	A	4	Steam and power conversion (HJ)	Pipes, fittings
2)	1/12	25	F	Loss of the reactor coolant system flow indication to the integrated control system due to a ground which tripped power inverter	A	3	Reactor coolant (CB)	Generators (inverters)
3a)	1/14	1	S	Complete unit load rejection test	B	3	Steam and power conversion (HA)	Generators (main generator)
3b)	1/14 (cont.)	368	S	Unit testing; outage continued for replacement of seals on reactor coolant pumps	B	4	Reactor coolant (CB)	Pumps
4)	2/13	23	F	Loss of power to reactor coolant pumps 1-2 and 2-1	A	3	Reactor coolant (CB)	Relays
5)	2/22	32	F	Electrical circuitry in electro-hydraulic control of the turbine failed	A	1	Steam and power conversion (HA)	Electrical conductors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	3/4	6	S	Repairs to turbines electro-hydraulic control system	B	1	Steam and power conversion (HA)	Mechanical function units
7a)	3/30	745	S	Maintenance and repair of main steam safety valves	B	1	Steam and power conversion (HB)	Valves
7b)	3/30 (cont.)	1728	S	Unit remained shut down for modifications required by NRC of B&W plants resulting from TMI-2 accident (i.e. reevaluation of the small break analysis)	D	4	Steam and power conversion (HH)	Instrumentation and controls
8)	9/7	58	S	Isolation of steam leak in containment	B	1	Steam and power conversion (HB)	Pipes, fittings
9)	9/18	17	F	Sticking pump pressure controller on No. 2 electro-hydraulic control pump	A	3	Steam and power conversion (HA)	Mechanical function units

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10)	9/26	41	F	Faulty capacitor on turbine-throttle pressure transmitter power supply	A	3	Steam and power conversion (HA)	Instrumentation and controls
11)	10/5	49	S	Repair pressurizer spray valve RC 2	B	1	Reactor coolant (CB)	Valves
12)	10/15	142	F	Capacitor failure in integrated control system pulser circuit to the turbine electro-hydraulic control system	A	3	Steam and power conversion (HA)	Instrumentation and controls
13)	10/25	630	F	Loss of reactor coolant pump 2-2 from blown fuse in the DC power supply starting a pump two minute time delay trip relay with reactor coolant pump 1-1 already shutdown	A	3	Reactor coolant (CB)	Circuit closers/interrupters
14)	11/30	749	S	Maintenance due to low bearing oil level alarm on RCP 1-2	B	1	Reactor coolant (CB)	Pumps



DESIGN ELEC. RATING = 906 MAX. DEPEND. CAP. = 906 (100%) DAVIS-BESSE 1

DRESDEN 1

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-010
Reactor Type: BWR
Capacity (MWe-Net): 197
Commercial Operation: 7/4/60
Plant Age: 19.7 Years

Performance

Net Electrical Energy Generated (MWH):	-13,047
Unit Availability Factor (%):	0
Unit Capacity Factor (%) (Using MDC):	0
Unit Capacity Factor (%) (Using Design MWE):	0

Outages

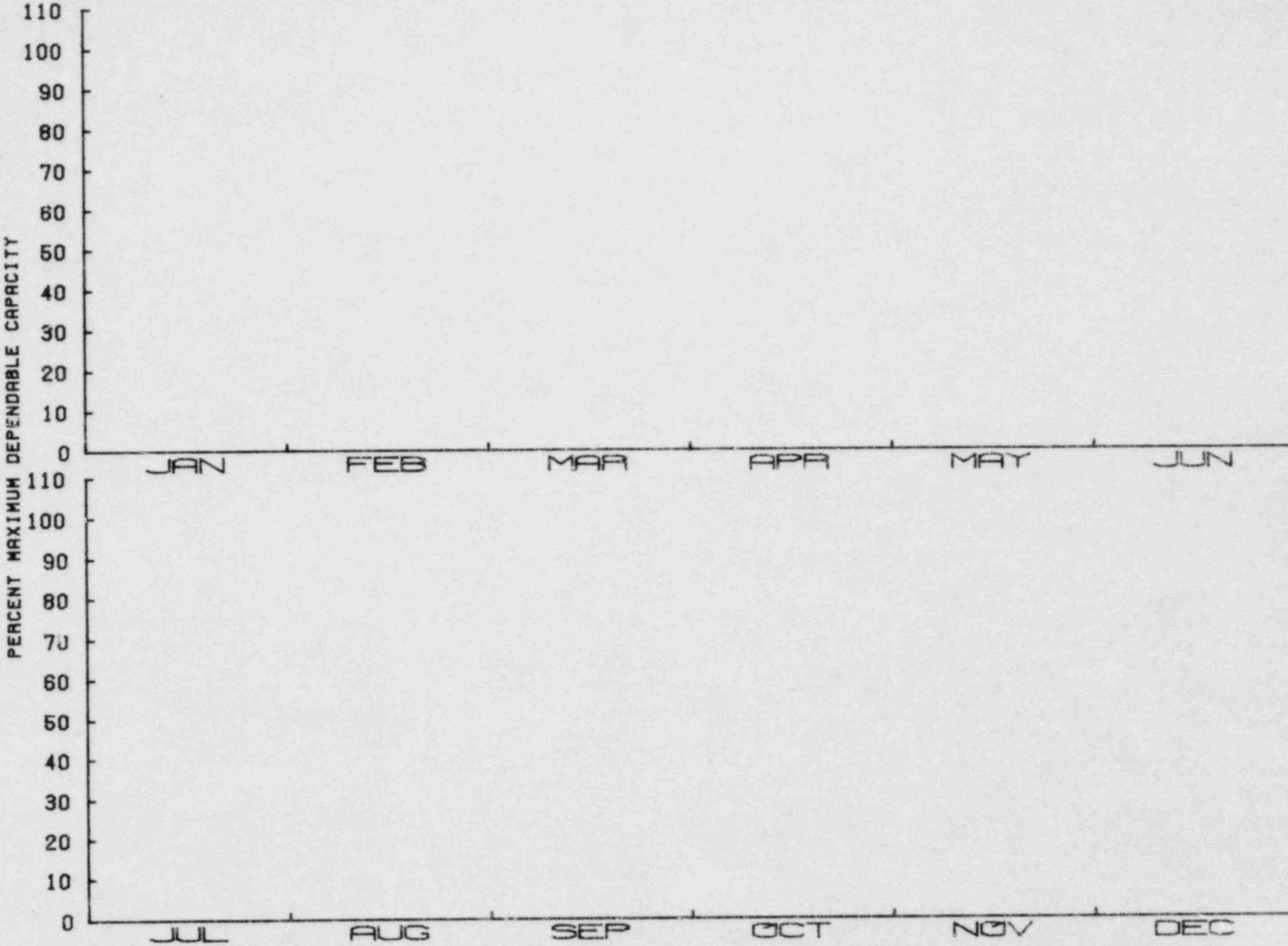
Total No.	1		
Forced	0		
Scheduled	1		
Total:	8,760	Hours,	100.0%
Forced	0	Hours,	0%
Scheduled	8,760	Hours,	100.0%

II. Highlights

The unit was shut down all year for the purpose of upgrading the emergency core-cooling system in accordance with license amendment No. 23, dated January 6, 1978.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	10/31/78 (cont.)	8760	S	Upgrade the ECCS, chemical cleaning, and refueling. Outage is expected to last 18 months. The upgrading of the ECCS is in accordance with license amendment No. 23, date January 6, 1978.	D	4	Engineered safety features (SF)	Other



DESIGN ELEC. RATING = 209 MAX. DEPEND. CAP. = 197 (100%) DRESDEN 1

DRESDEN 2

I. Summary

Description		Performance		Outages			
Location:	Morris, Illinois	Net Electrical Energy Generated (MWH):	4,939,630	Total No.	12		
Docket No:	50-237			Forced	9		
Reactor Type:	BWR	Unit Availability Factor (%):	81.6	Scheduled	3		
Capacity (MWe-Net):	772			Total:	1,614	Hours,	18.4%
Commercial Operation:	6/9/72	Unit Capacity Factor (%) (Using MDC):	73.0	Forced	234	Hours,	2.7%
Plant Age:	9.7 Years			Scheduled	1,380	Hours,	15.7%
		Unit Capacity Factor (%) (Using Design MWE):	71.0				

II. Highlights

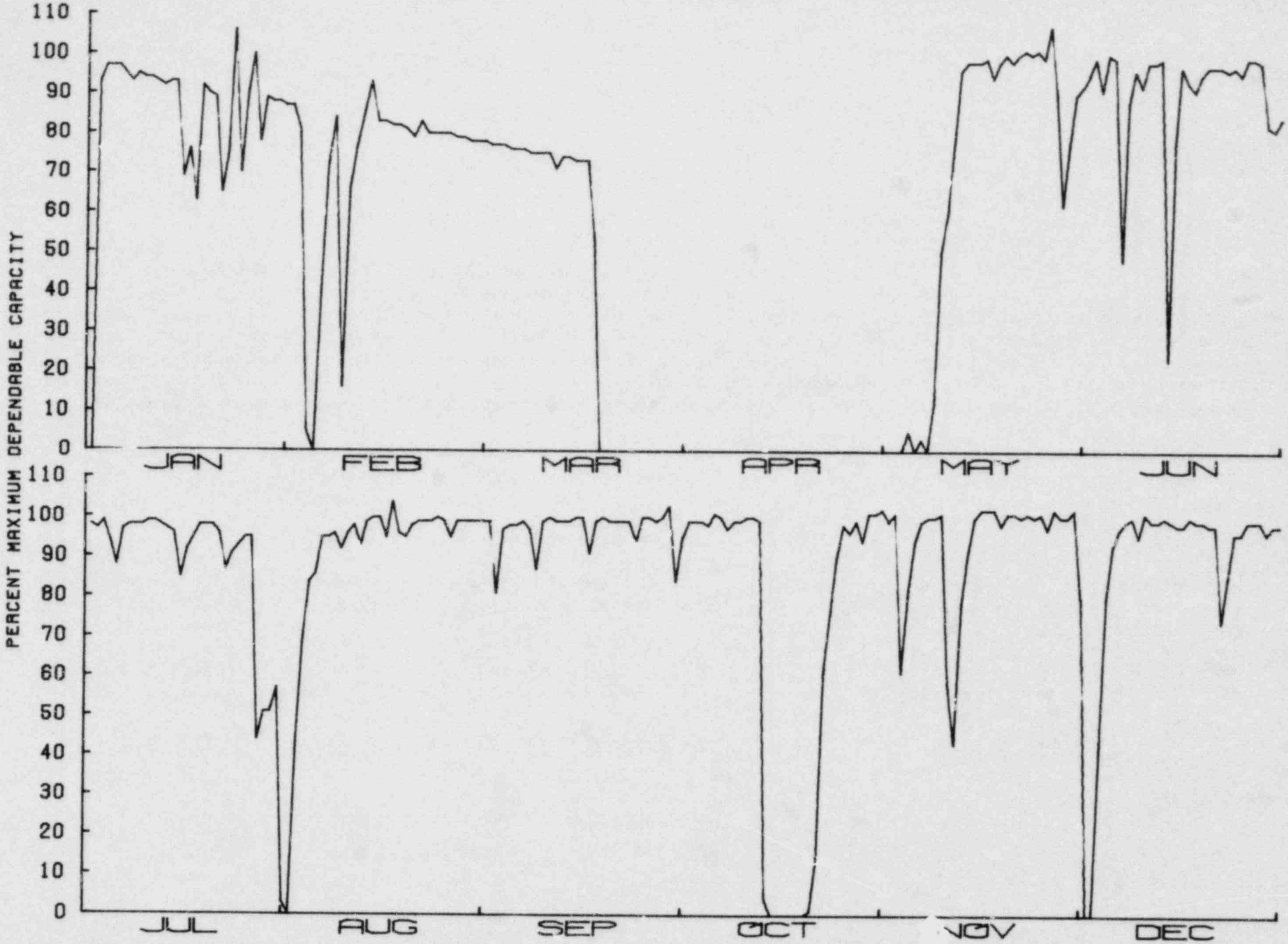
Operation was routine throughout the year. A refueling was accomplished between March 17 and May 4.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/3	49	F	Loss of secondary containment due to overpressurization of the reactor building as a result of the loss of exhaust fans	A	1	Other auxiliary (AA)	Blowers
2)	2/8	16	F	Trip of both scram channels while performing instrumentation surveillance	G	3	Instrumentation and controls (IA)	Instrumentation and controls
3)	3/17	1143	S	Refueling	C	1	Reactor (RC)	Fuel elements
4)	5/5	44	F	"D" TIP machine stuck in index position #2	A	1	Instrumentation and controls (ID)	Instrumentation and controls
5)	5/7	29	F	"D" TIP machine stuck in position #6	A	1	Instrumentation and controls (ID)	Instrumentation and controls
6)	5/8	1	F	Steam leak in the turbine hood	A	9	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	6/12	12	F	Inadvertent closure of main steam isolation valve	G	3	Reactor coolant (CD)	Instrumentation and controls
8)	7/31	49	F	Repair packing leak on core spray valve	A	1	Engineered safety features (SF)	Valves
9)	10/13	183	S	Hanger and anchor bolt inspection	D	1	Engineered safety features (SF)	Shock suppressors
10)	10/20	19	F	Moisture separator drain tank high level	A	3	Steam and power conversion (HB)	Instrumentation and controls
11)	11/10	15	F	Feedwater pump tripped due to low suction pressure trip introduced by instrument mechanics	G	3	Reactor coolant (CH)	Instrumentation and controls
12)	11/30	54	S	Repair leak on moisture separator line	B	1	Steam and power conversion (HB)	Pipes, fittings



DESIGN ELEC. RATING = 794 MAX. DEPEND. CAP. = 772 (100%) DRESDEN 2

DRESDEN 3

I. Summary

Description

Location: Morris, Illinois
Docket No: 50-249
Reactor Type: BWR
Capacity (MWe-Net): 773
Commercial Operation: 11/16/71
Plant Age: 8.4 Years

Performance

Net Electrical Energy Generated (MWH):	3,475,813
Unit Availability Factor (%):	67.7
Unit Capacity Factor (%) (Using MDC):	51.3
Unit Capacity Factor (%) (Using Design MWE):	50.0

Outages

Total No.	12	
Forced	10	
Scheduled	2	
Total:	2,826 Hours,	32.3%
Forced	2,684 Hours,	30.7%
Scheduled	142 Hours,	1.6%

II. Highlights

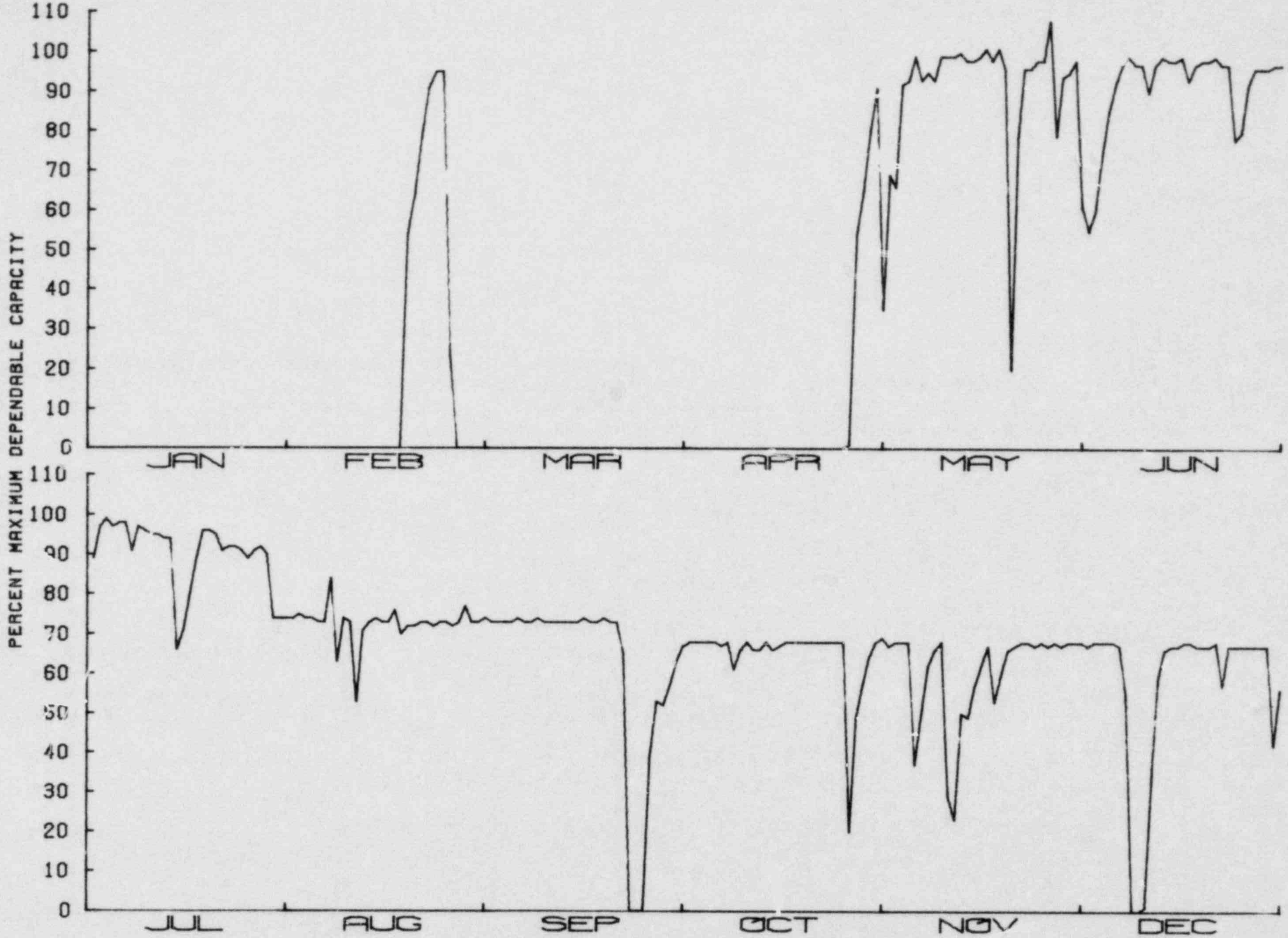
At the beginning of the year, replacement of the main transformer was still in progress. The transformer had been disabled by a fire on December 12, 1978. Replacement was completed on February 16, but another fire in the transformer occurred on February 23, and this second replacement outage lasted until April 24. In July, an administrative derating to 75% of full power was imposed for evaluation of air ejector radioactivity due to an increase in off-gas radioactivity and 7 × 7 fuel assembly degradation. This restriction was maintained the rest of the year. In December, a 4-day outage took place for TMI-2-related modifications. Acoustic monitors were installed on the safety valve discharge lines.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	12/12/78 (cont.)	1116	F	Generator load reject caused by a fire in the main unit transformer	A	4	Electric power (EB)	Transformers
2)	2/23	1484	F	Short in the replacement main transformer and a resultant fire	A	3	Electric power (EB)	Transformers
3)	4/29	5	F	Moisture separator drain tank hi hi level	A	2	Steam and power conversion (HB)	Instrumentation and controls
4)	5/1	4	F	Change stator water cooling filters	A	9	Steam and power conversion (HA)	Generators (main generator)
5)	5/19	14	F	"A" feed reg. valve failure to close below 20%	G	3	Reactor coolant (CH)	Valves
6)	5/31	9	F	Turbine trip (cross under hi press) relief isolated not vented	G	3	Steam and power conversion (HA)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	9/22	71	S	Snubber inspection	B	3	Engineered safety features (SF)	Shock suppressors
8)	10/25	13	F	Foreman slammed door on electro-hydraulic switching system motor generator set relay cabinet and a relay tripped	G	3	Steam and power conversion (HA)	Relays
9)	11/5	7	F	Replaced lockout relay on generator	A	9	Steam and power conversion (HA)	Relays
10)	11/10	25	F	Personnel error during condenser surveillance	G	3	Steam and power conversion (HC)	Instrumentation and controls
11)	12/7	71	S	TMI modifications — acoustic monitors installed on safety valve discharge lines	D	1	Reactor coolant (CC)	Instrumentation and controls
12)	12/10	7	F	Turbine tripped on 3 "B" moisture separator hi-hi	A	3	Steam and power conversion (HB)	Instrumentation and controls



DESIGN ELEC. RATING = 794 MAX. DEPEND. CAP. = 773 (100%) DRESDEN 3

DUANE ARNOLD

I. Summary

Description	Performance		Outages		
Location: Palo, Iowa	Net Electrical Energy		Total No.	9	
Docket No: 50-331	Generated (MWH):	2,898,764	Forced	9	
Reactor Type: BWR	Unit Availability		Scheduled	0	
Capacity (MWe-Net): 515	Factor (%):	78.0	Total:	1,930 Hours,	22.0%
Commercial Operation: 2/1/75	Unit Capacity Factor (%)		Forced	1,930 Hours,	22.0%
Plant Age: 5.6 Years	(Using MDC):	64.3	Scheduled	0 Hours,	0%
	Unit Capacity Factor (%)				
	(Using Design MWE):	61.5			

II. Highlights

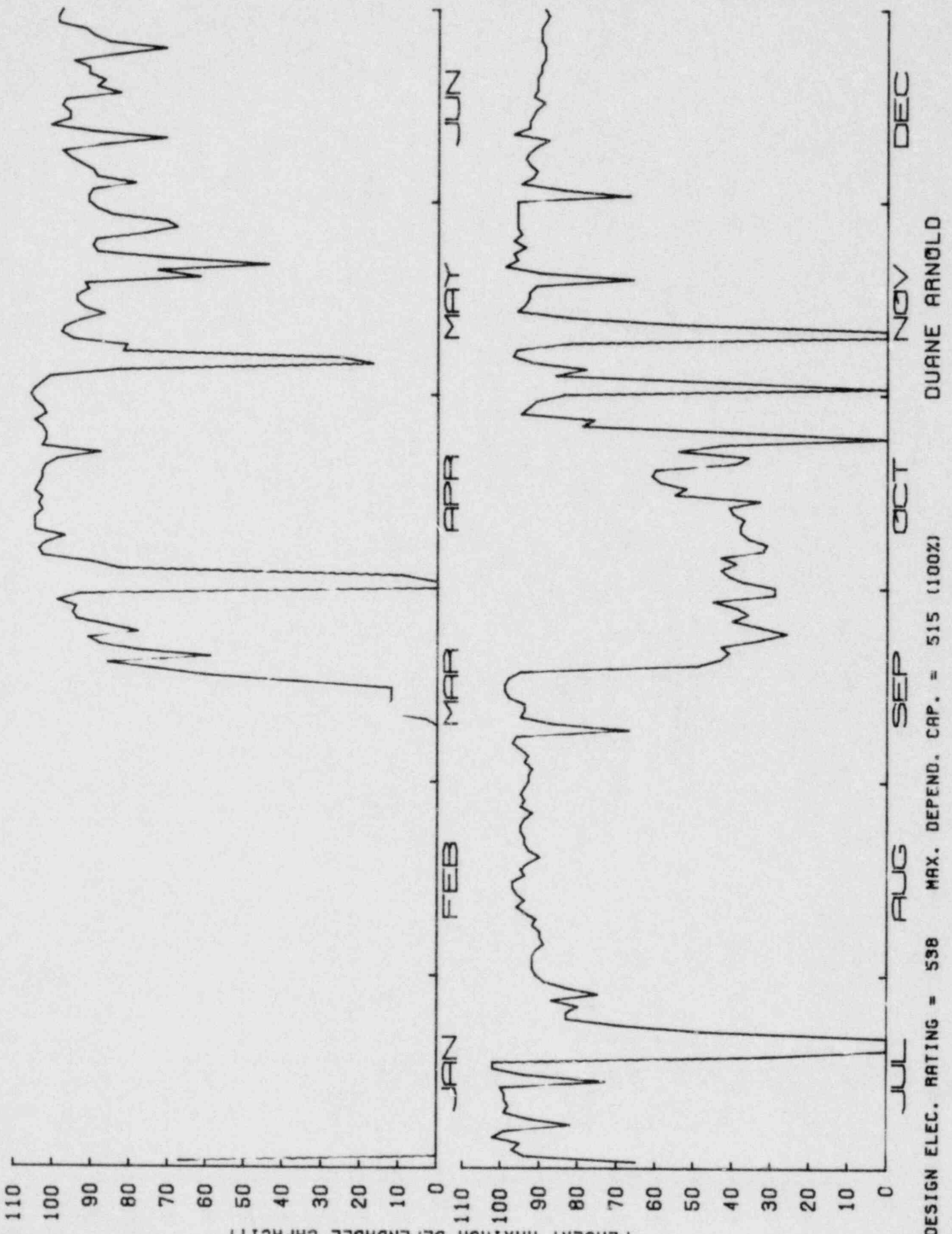
At the beginning of the year, the unit was still shut down for replacement of recirculation system inlet nozzle safe ends because of cracks. This shutdown, which began June 17, 1978, ended March 10, and operation resumed. In September and October, power reductions were effected due to lack of demand for power. At the end of the year, the unit was in an end-of-cycle coastdown.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	6/17/78 (cont.)	1640	F	Unit remained shut down for replacement of recirculation system inlet nozzle safe ends. Startup is being delayed while a flow restriction is being removed from the N_2B riser	A	4	Reactor coolant (CB)	Pipes, fittings
2)	3/31	64	F	Repair RWCU system isolation valve and replace section of RWCU system pipe	A	1	Reactor coolant (CG)	Pipes, fittings
3)	5/5	19	F	Repair HPCI check valve	A	1	Engineered safety features (SF)	Valves
4)	5/21	10	F	Scrammed during testing of recirculation system flow instrumentation	G	3	Reactor coolant (CB)	Instrumentation and controls
5)	7/18	87	F	Turbine trip due to exhaust hood high temperature indication	A	3	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	10/24	29	F	Repair "B" feed pump seal water line	A	1	Reactor coolant (CH)	Pipes, fittings
7)	10/31	32	F	EHC low pressure indication	A	3	Steam and power conversion (HA)	Instrumentation and controls
8)	11/8	43	F	Reactor scram on main steam high radiation caused by N-16 spike due to air in feedwater from condensate demineralizer	A	3	Steam and power conversion (HG)	Demineralizers
9)	11/10	6	F	Turbine trip on EHC low pressure due to mechanical trip valve linkage becoming loose	A	9	Steam and power conversion (HA)	Mechanical function units


PERCENT MAXIMUM DEPENDABLE CAPACITY
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JAN
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APR
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DEC
DESIGN ELEC. RATING = 538 MAX. DEPEND. CAP. = 515 (100%)
DUANE ARNOLD

FARLEY 1

I. Summary

Description

Location: Dothan, Alabama
Docket No: 50-348
Reactor Type: PWR
Capacity (MWe-Net): 829
Commercial Operation: 12/1/77
Plant Age: 2.4 Years

Performance

Net Electrical Energy Generated (MWH):	1,743,590
Unit Availability Factor (%):	28.6
Unit Capacity Factor (%) (Using MDC):	24.0
Unit Capacity Factor (%) (Using Design MWE):	24.0

Outages

Total No.	17		
Forced	14		
Scheduled	3		
Total:	6,256	Hours,	71.4%
Forced	4,081	Hours,	46.6%
Scheduled	2,175	Hours,	24.8%

II. Highlights

A refueling outage was started in March, with expectations of completion in 10 to 12 weeks. However, the outage was extended to November for testing the anchor bolts of pipe support base plates in accordance with IE Bulletin 70-02 and correction of seismic design deficiencies in safety-related piping in accordance with IE Bulletin 79-14. After operation was resumed in November, problems with the feedwater system resulted in five shutdowns during the remainder of the year.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	12/31/78 (cont.)	13	[illegible]	Valve error on flow pump suction header isolation instrument	G	4	Steam and power conversion (HH)	Valves
2)	1/16	29	F	Inverter 1B trip due to grounded choke coil	A	3	Electric power (ED)	Generators (inverters)
3)	1/17	8	F	S/G lo-lo level	G	3	Steam and power conversion (HH)	Instrumentation and controls
4)	1/18	5	F	RCP bus undervoltage	A	3	Electric power (EB)	Circuit closers/ interrputers
5)	1/20	22	F	Turbine trip from loss of condenser vacuum	A	3	Steam and power conversion (HC)	Motors
6)	2/14	3	F	Voltage drop in the Vital AC System induced by a short circuit at the SSPS cabinet while trouble shooting	A	3	Electric power (ED)	Electrical conductors

DETAILS OF PLANT OUTAGES

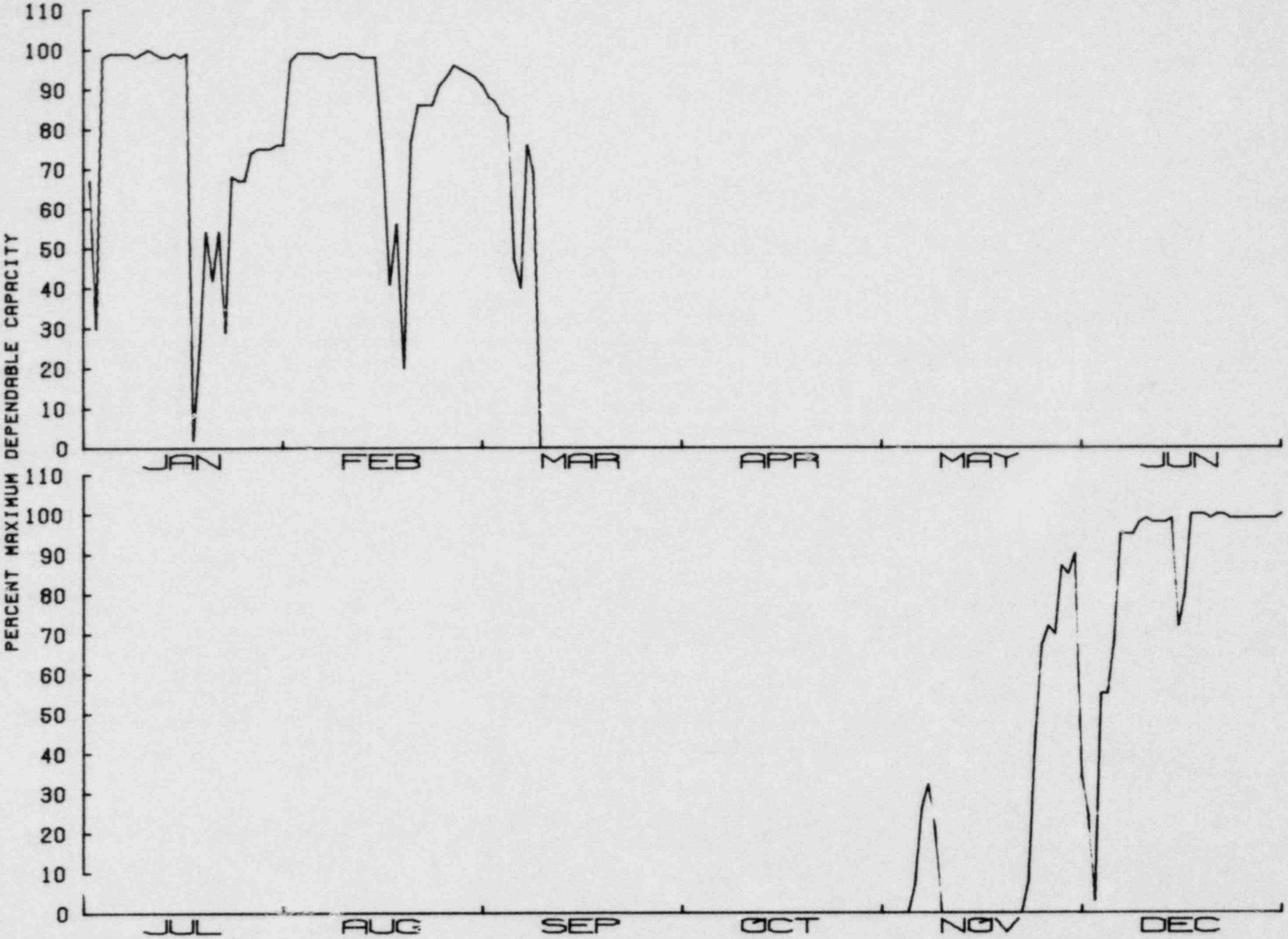
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	2/16	3	F	Loss of main generator excitation	G	3	Steam and power conversion (HA)	Generators (exciter)
8)	2/16	2	F	Low 1A steam generator level	G	3	Steam and power conversion (HH)	Instrumentation and controls
9)	2/17	9	F	1A S/G low-low level with the feed regulating valves in manual	G	3	Steam and power conversion (HH)	Instrumentation and controls
10)	3/5	3	F	Loss of 1 inverter resulting in the feedwater regulating valves closing and steam generator 1C reaching its low-low level setpoint	A	3	Electric power (ED)	Generators (inverter)
11a)	3/8	1856	S	Refueling	C	1	Reactor (RC)	Fuel elements
11b)	3/8 (cont.)	3833	F	Testing of anchor bolts of pipe support base plates per I&E bulletin 79-02	D	4	Engineered safety features (SF)	Shock suppressors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11c)	3/8 (cont.)	84	F	Repair electrical connector on CRD mechanism and core physics testing	A	4	Reactor (RB)	Control rod drives
12)	11/4	10	F	Leakage past "C" steam generator feedwater regulating valve and inexperience in operating the new FW bypass system	G	3	Steam and power conversion (HH)	Valves
13)	11/5	10	F	"A" feedwater regulating valve closed in "auto" causing a S/G lo-lo level trip	A	3	Steam and power conversion (HH)	Valves
14)	11/8	300	S	Repair RCP seals	B	1	Reactor coolant (CB)	Pumps
15)	11/21	29	F	Inexperience in operating new FW bypass system	G	3	Steam and power conversion (HH)	Instrumentation and controls
16)	11/30	18	F	Loss of SGFP suction pressure. Additional unit trip after reaching criticality due to inexperience in transferring from aux feed to FW bypass system	G	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
17)	12/.	19	S	Isolate 1B SGFP for repairs	B	9	Steam and power conversion (HH)	Pumps



DESIGN ELEC. RATING = 829   MAX. DEPEND. CAP. = 829 (100%)   FARLEY 1

FITZPATRICK

I. Summary

Description

Location: Scriba, New York
Docket No: 50-333
Reactor Type: BWR
Capacity (MWe-Net): 800
Commercial Operation: 7/28/75
Plant Age: 4.9 Years

Performance

Net Electrical Energy Generated (MWH):	2,964,590
Unit Availability Factor (%):	50.8
Unit Capacity Factor (%) (Using MDC):	42.3
Unit Capacity Factor (%) (Using Design MWE):	41.2

Outages

Total No.	2	
Forced	2	
Scheduled	0	
Total:	4,309 Hours,	49.2%
Forced	4,309 Hours,	49.2%
Scheduled	0 Hours,	0%

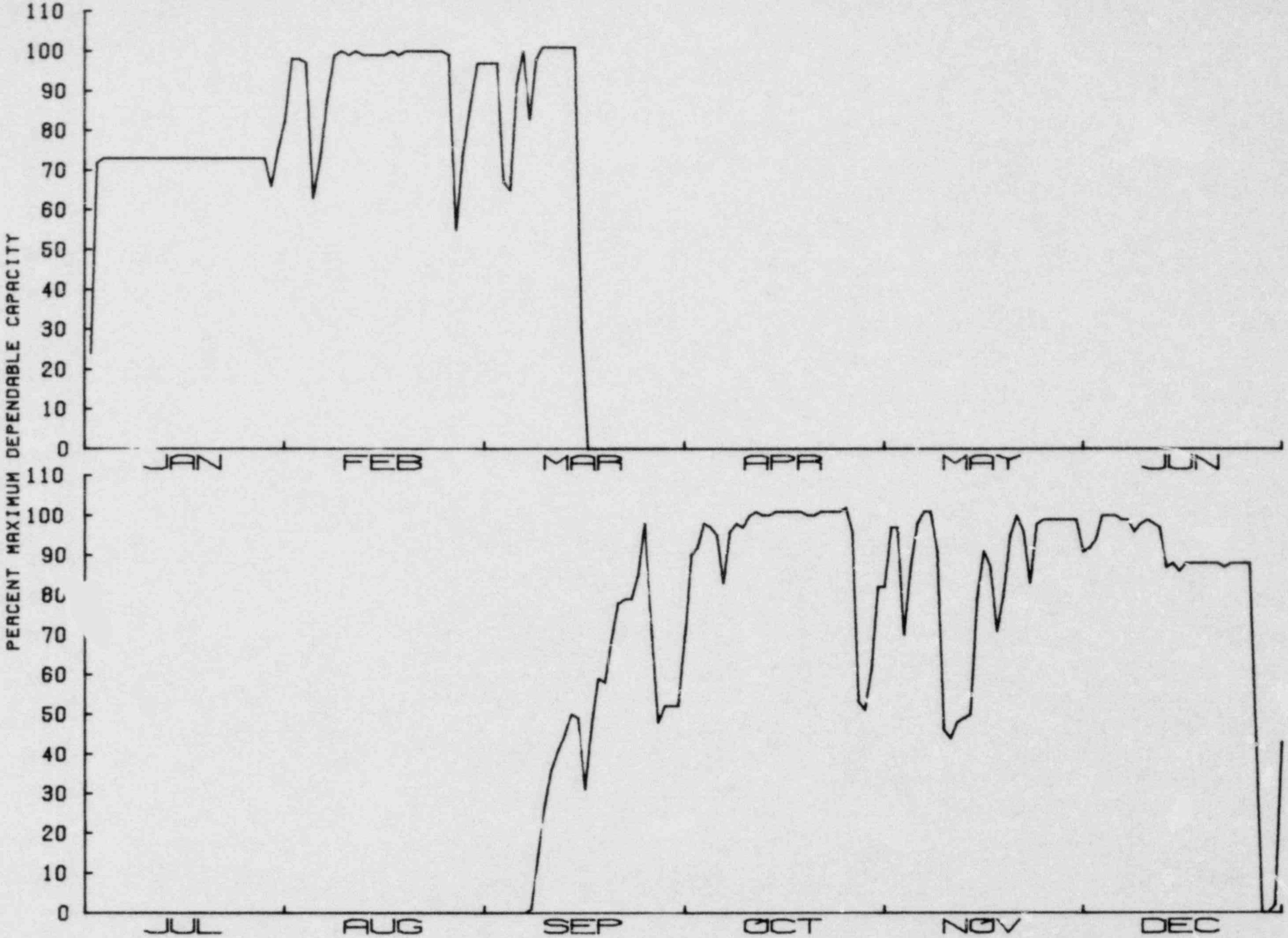
II. Highlights

The unit operated without interruption until March 15 when it was shut down in accordance with an NRC show-cause order to determine if modifications should be made to some safety-related piping systems to bring them into conformance with requirements for withstanding earthquakes per IE Bulletin 79-14.

As a result of the reanalysis, 5 of the 96 safety-related piping systems in the plant required modification to correct overstress under postulated earthquake conditions. The modifications involved installation of new or modified pipe supports (shock absorbers and restraints) and repair of existing supports. Of the approximately 1000 safety-related supports involved, approximately 43 required modification. The majority of the modifications were to make the system as built conform with the intended design. Operation resumed on September 7, and only one other outage occurred during the remainder of the year.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/15	4228	F	Required by NRC show-cause order for reevaluation of stress calculations and modifications for seismic events involving safety related piping systems	D	1	Engineered safety features (SF)	Pipes, fittings
2)	12/27	81	F	Turbine trip on high vibration	A	3	Steam and power conversion (HA)	Turbines



DESIGN ELEC. RATING = 821 MAX. DEPEND. CAP. = 800 (100%) FITZPATRICK

FORT CALHOUN

I. Summary

Description

Location: Fort Calhoun, NE
Docket No: 50-285
Reactor Type: PWR
Capacity (MWe-Net): 457
Commercial Operation: 6/20/74
Plant Age: 6.4 Years

Performance

Net Electrical Energy Generated (MWH):	3,666,112
Unit Availability Factor (%):	95.7
Unit Capacity Factor (%) (Using MDC):	91.6
Unit Capacity Factor (%) (Using Design MWE):	91.6

Outages

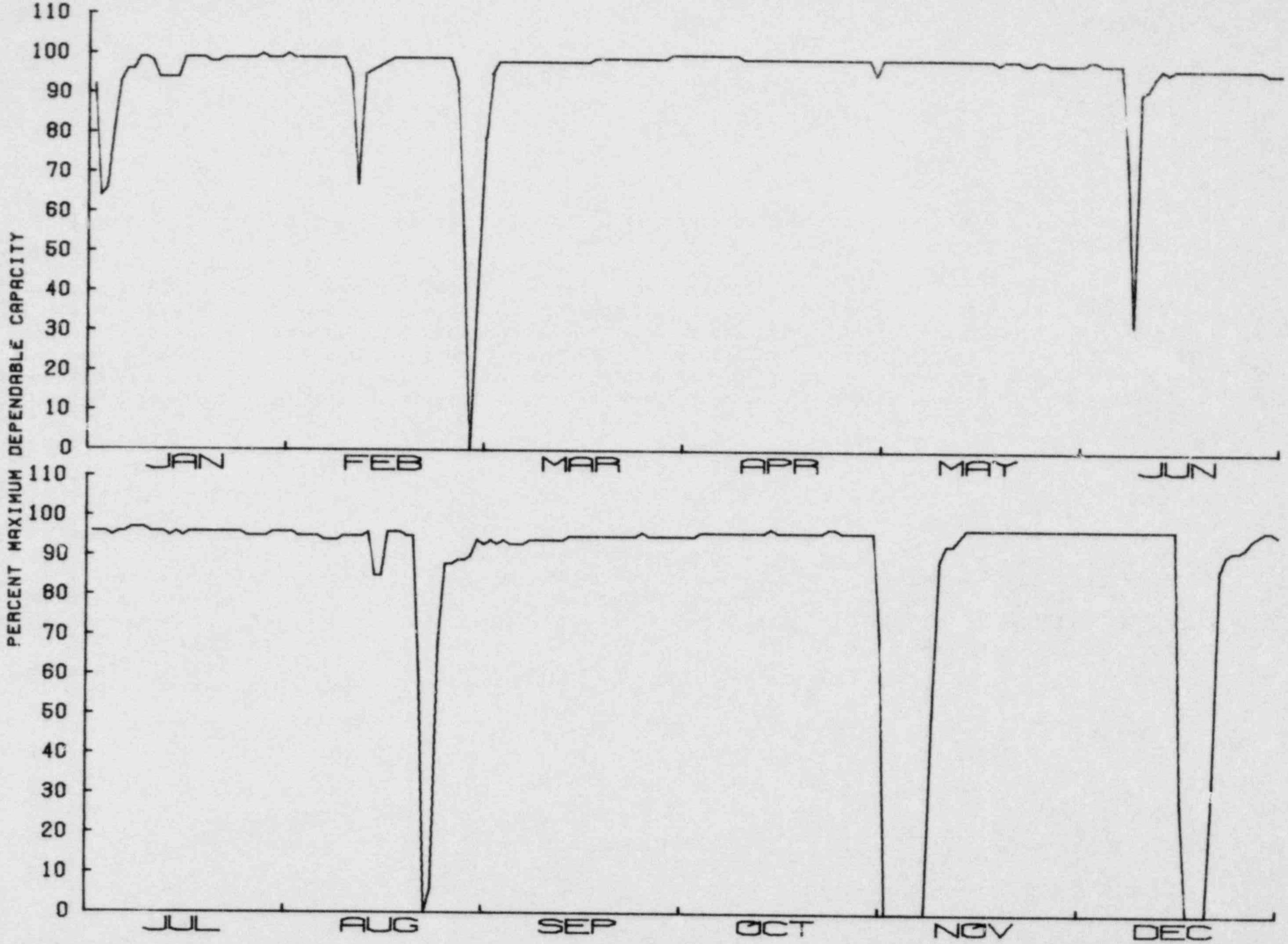
Total No.	6		
Forced	5		
Scheduled	1		
Total:	376	Hours,	4.3%
Forced	201	Hours,	2.3%
Scheduled	175	Hours,	2.0%

II. Highlights

Operation was routine throughout the year. There were 6 months in which operation was uninterrupted, 3 months being in sequence. In November, 8 days of shutdown time were devoted to checking and repairing cracks in the steam generator feedwater line nozzle-to-pipe welds in accordance with IE Bulletin 79-13.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/25	6	F	Closure of turbine control valve due to blown fuse during pressure transmitter replacement	A	2	Instrumentation and controls (IC)	Circuit closures/ interrupters
2)	2/26	15	F	Closure of turbine control valve due to blown fuse during pressure transmitter replacement	A	9	Instrumentation and controls (IC)	Circuit closures/ interrupters
3)	6/7	16	F	Fire protection system deluge valve failure	A	3	Other auxiliary (AB)	Valves
4)	8/21	48	F	Electrical noise spike	A	3	Instrumentation and controls (IA)	Instrumentation and controls
5)	10/31	175	S	Testing of S/G nozzle to piping welds	D	1	Steam and power conversion (HB)	Pipes, fittings
6)	12/16	116	F	RCP seal replacement	A	1	Reactor coolant (CB)	Pumps



DESIGN ELEC. RATING = 457 MAX. DEPEND. CAP. = 457 (100%) FORT CALHOUN 1

FORT ST. VRAIN

I. Summary

Description	Performance*		Outages*			
Location: Platteville, Colorado	Net Electrical Energy Generated (MWH):	123,584	Total No.	11		
Docket No: 50-267			Forced	8		
Reactor Type: HTGR	Unit Availability Factor (%):	22.2	Scheduled	3		
Capacity (MWe-Net): 330			Total:	3,434	Hours,	77.8%
Commercial Operation: 7/1/79	Unit Capacity Factor (%) (Using MDC):	8.5	Forced	1,289	Hours,	29.2%
Plant Age: 3.1 Years			Scheduled	2,145	Hours,	48.6%
	Unit Capacity Factor (%) (Using Design MWE):	8.5				

II. Highlights

The unit began commercial operation on July 1 while still in a refueling shutdown. On July 23 operation was resumed but was restricted to 70% of full power pending resolution by the NRC of discrepancies between the Final Safety Analysis Report and the technical specification bases. During September, the unit was again shut down because of inconsistency in the seismic design of safety-related piping per IE Bulletin 79-14. On October 26, the unit was shut down for an extensive maintenance outage and installation of core region constraint devices; the outage was expected to last until March 1980.

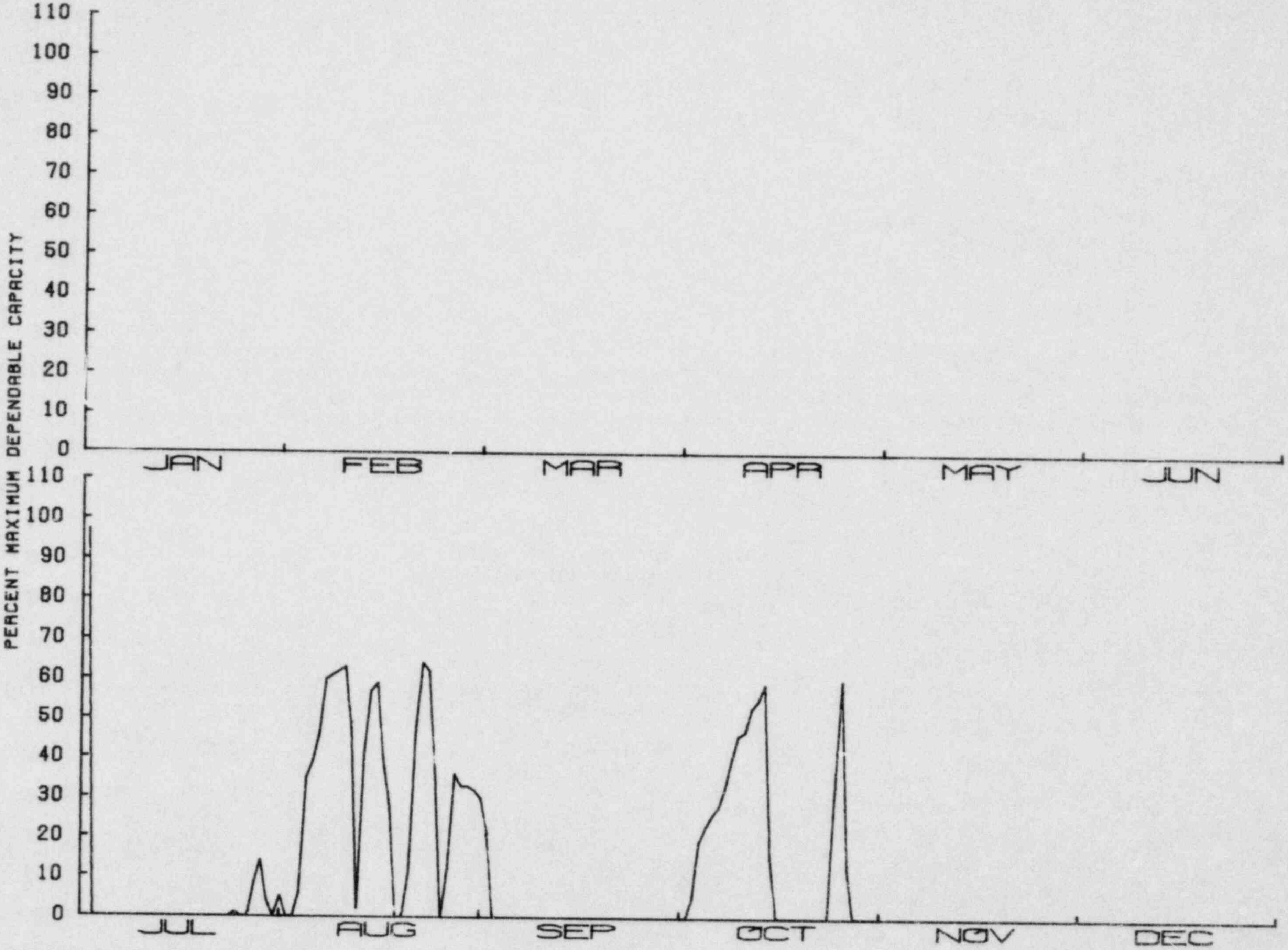
*The number of hours in the reporting period is 4,417 based on the July 1, 1979 date of commercial operation.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/1/79 (cont.)	542	S	Refuel and turbine generator overhaul	C	4	Reactor (RC)	Fuel elements
2)	7/24	47	F	Field ground relay problems	A	2	Steam and power conversion (HA)	Generator (main generator)
3)	7/26	2	S	Turbine overspeed testing	B	2	Steam and power conversion (HA)	Turbines
4)	7/28	53	F	High vibration on turbine	A	3	Steam and power conversion (HA)	Turbines
5)	7/31	77	F	Throttle pressure dropped and load decreased 20 MW; during recovery, three circulators tripped, and turbine tripped	A	3	Reactor coolant (CB)	Blowers
6)	8/11	24	F	Turbine trip while reducing power to recover a tripped circulator	A	3	Reactor coolant (CB)	Blowers

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	8/17	68	F	Instrument panel shorted to ground and tripped	A	3	Instrumentation and controls (IA)	Instrumentation and controls
8)	8/24	46	F	Turbine generator taken off-line in attempt to isolate cause of high primary coolant moisture	A	2	Reactor coolant (CB)	Other
9)	9/1	743	F	Inconsistencies in random sample of safety-related piping for impaired hangers	A	1	Engineered safety features (SF)	Shock suppressors
10)	10/14	231	F	Low steam temperature	A	3	Steam and power conversion (HA)	Other
11)	10/26	1601	S	Maintenance and installation of core region constraint devices	B	2	Reactor (RC)	Other



DESIGN ELEC. RATING = 330 MAX. DEPEND. CAP. = 330 (100%) FORT ST VRAIN

GINNA

I. Summary

Description

Location:	Ontario, New York
Docket No:	50-244
Reactor Type:	PWR
Capacity (MWe-Net):	470
Commercial Operation:	7/70
Plant Age:	10.1 Years

Performance

Net Electrical Energy Generated (MWH):	2,960,510
Unit Availability Factor (%):	72.8
Unit Capacity Factor (%) (Using MDC):	71.9
Unit Capacity Factor (%) (Using Design MWE):	71.9

Outages

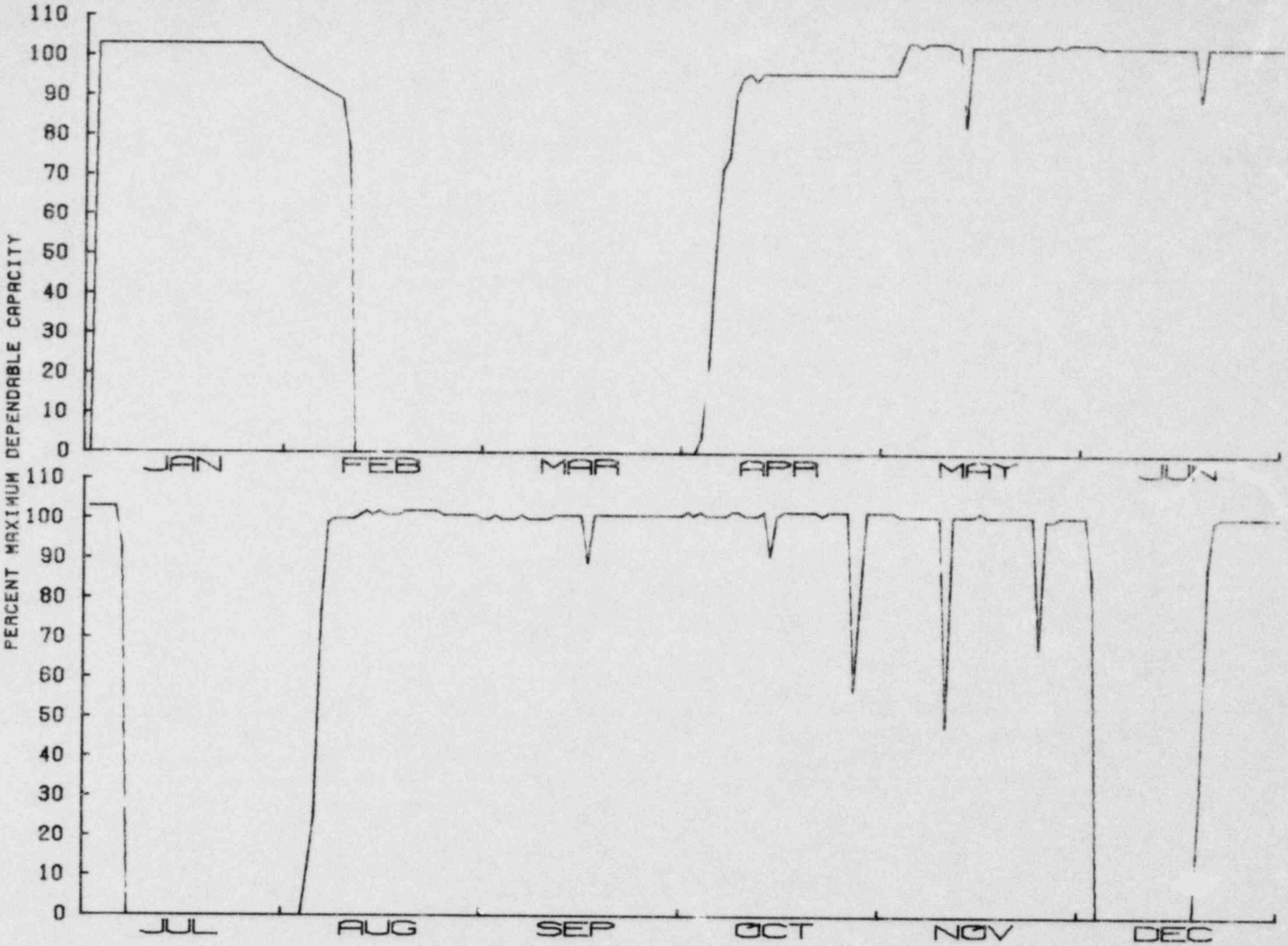
Total No.	5	
Forced	3	
Scheduled	2	
Total:	2,382 Hours,	27.2%
Forced	430 Hours,	4.9%
Scheduled	1,952 Hours,	22.3%

II. Highlights

The unit operated 3 months of the year without interruption. Refueling was accomplished in February. In July, the unit was shut down for inspection of steam generator feedwater nozzle-to-pipe welds in accordance with IE Bulletin 79-13. Circumferential cracks were found and repaired on both steam generators. In December, linear indications were found on the pressurizer power-operated relief valve nozzle near the safe end, and an evaluation was initiated.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/10	1262	S	Refueling; coastdown began 1/26/79	C	1	Reactor (RC)	Fuel elements
2)	7/6	690	S	NRC inspection requirements on feedwater steam generator nozzle weld inspection	D	1	Steam and power conversion (HH)	Heat exchangers (S/G)
3)	8/5	1	F	Loss of condenser vacuum during testing	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
4)	10/27	13	F	"B" steam generator handhole gasket leak	A	9	Steam and power conversion (HB)	Heat exchangers (S/G)
5)	12/2	416	F	Repair tube leak in the "B" steam generator	A	1	Steam and power conversion (HB)	Heat exchangers (S/G)



DESIGN ELEC. RATING = 470 MAX. DEPEND. CAP. = 470 (100%) GINNA

HADDAM NECK

I. Summary

Description	Performance		Outages			
Location: Haddam Neck, Conn.	Net Electrical Energy Generated (MWH):	4,116,339	Total No.	5		
Docket No: 50-213			Forced	1		
Reactor Type: PWR	Unit Availability Factor (%):	87.5*	Scheduled	4		
Capacity (MWe-Net): 550			Total:	1,332	Hours,	15.2%*
Commercial Operation: 1/1/68	Unit Capacity Factor (%) (Using MDC):	85.4	Forced	30	Hours,	0.3%
Plant Age: 12.4 Years			Scheduled	1,302	Hours,	14.9%
	Unit Capacity Factor (%) (Using Design MWE):	81.7				

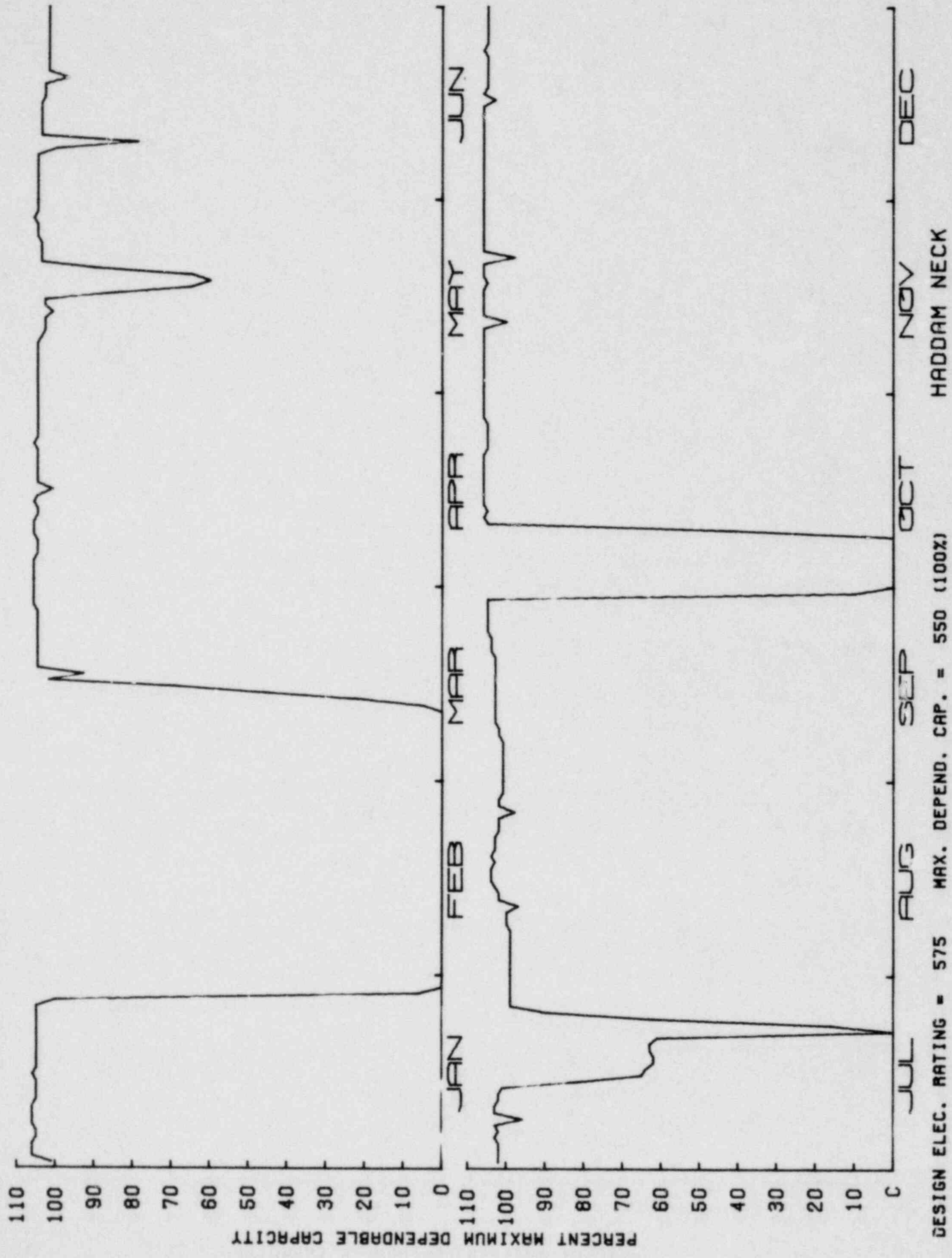
II. Highlights

Operation was routine throughout the year. Refueling was conducted in February, and from September 29 to October 9 the unit was shut down for inspection of the weld area of the steam generator feedwater line nozzle for cracks in accordance with IE Bulletin 79-13. The unit had 6 months of uninterrupted operation, with 4 months in sequence.

*Includes 233.5 reserve shutdown hours equal to 2.7% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/27	1064	S	Normal unloading from full power to zero load for refueling; reactor was inadvertently tripped due to spurious high startup rate signal	C	3	Reactor (RC)	Fuel elements
2)	3/12	0.4	S	Maintenance on electrical equipment	B	9	Electric power (ED)	Electrical conductors
3)	3/12	4	S	Turbine balance	B	9	Steam and power conversion (HA)	Turbines
4)	7/21	30	F	Mismatch on low pressure steam dump system	A	3	Steam and power conversion (HE)	Instrumentation and controls
5)	9/29	234	S	Check weld area of S/G feed line nozzles for cracks	D	1	Steam and power conversion (HH)	Pipes, fittings


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 575 MAX. DEPEND. CAP. = 550 (100%) HADDAM NECK

HATCH 1

I. Summary

Description	Performance		Outages			
Location: Baxley, Georgia	Net Electrical Energy Generated (MWH):	3,337,875	Total No.	12		
Docket No: 50-321			Forced	8		
Reactor Type: BWR	Unit Availability Factor (%):	54.6	Scheduled	4		
Capacity (MWe-Net): 764			Total:	3,975	Hours,	45.4%
Commercial Operation: 12/31/75	Unit Capacity Factor (%) (Using MDC):	49.9	Forced	783	Hours,	9.0%
Plant Age: 5.1 Years			Scheduled	3,192	Hours,	36.4%
	Unit Capacity Factor (%) (Using Design MWE):	48.5				

II. Highlights

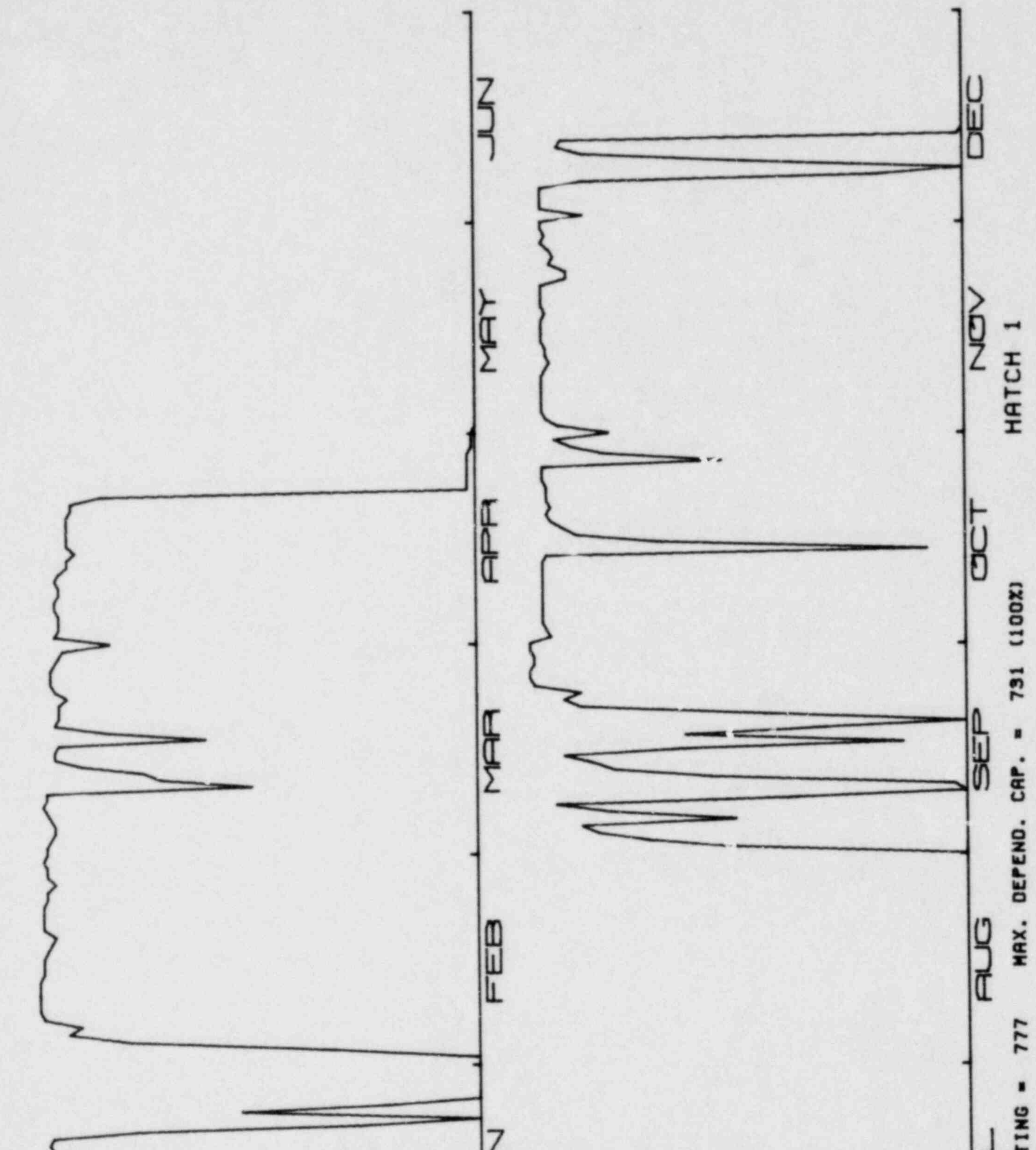
Refueling and maintenance of the unit were conducted between April 21 and August 29. There were 2 months of uninterrupted operation during the year. On December 13, the unit was shut down for the remainder of the year because of a ground fault in the main generator rotor.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/21	38	F	Turbine stop valve fast closure — perator inadvertently tripped power to 125 V dc cabinet	G	3	Steam and power conversion (HA)	Circuit closers/ interrupters
2)	1/24	215	F	High drywell pressure	G	3	Reactor coolant (CB)	Valves
3)	4/21	3101	S	Refueling and maintenance	C	1	Reactor (RC)	Fuel elements
4)	8/29	5	S	Turbine overspeed testing	B	9	Steam and power conversion (HA)	Turbines
5)	8/30	2	F	Alterrex problems	A	9	Steam and power conversion (HA)	Generators (main generator exciter)
6)	9/8	51	S	Repair esidual heat removal service wate umps	B	1	Auxiliary water (WA)	Pumps
7)	9/15	19	F	Scram on turbine control valve fast closure	A	3	Steam and power conversion (HA	Valve operators

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8)	9/18	35	S	Repair Alterrex system	B	1	Steam and power conversion (HA)	Generators (main generator exciter)
9)	10/14	16	F	Stop valve fast closure	A	3	Steam and power conversion (HA)	Valves
10)	12/7	22	F	Low water level signal due to removing "A" RFPT MGU controller	H	3	Reactor coolant (CH)	Instrumentation and controls
11)	12/8	18	F	Loss of motor control center R24-S035	A	1	Radioactive waste management (MB)	Electrical conductors
12)	12/13	453	F	Ground fault in the main generator rotor	A	1	Steam and power conversion (HA)	Generators (main generator)


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 777 MAX. DEPEND. CAP. = 731 (100%)
HATCH 1

HATCH 2

I. Summary

Description

Location: Baxley, Georgia
Docket No: 50-366
Reactor Type: BWR
Capacity (MWe-Net): 749
Commercial Operation: 9/5/79
Plant Age: 1.28 Years

Performance*

Net Electrical Energy Generated (MWH):	1,835,960
Unit Availability Factor (%):	85.2
Unit Capacity Factor (%) (Using MDC):	82.8
Unit Capacity Factor (%) (Using Design MWE):	79.1

Outages*

Total No.	11		
Forced	5		
Scheduled	6		
Total:	419	Hours,	14.8%
Forced	196	Hours,	7.0%
Scheduled	223	Hours,	7.9%

II. Highlights

Commercial operation began September 5, and the unit accumulated an availability of 85.6% for the remainder of the year, having operated in December without interruption.

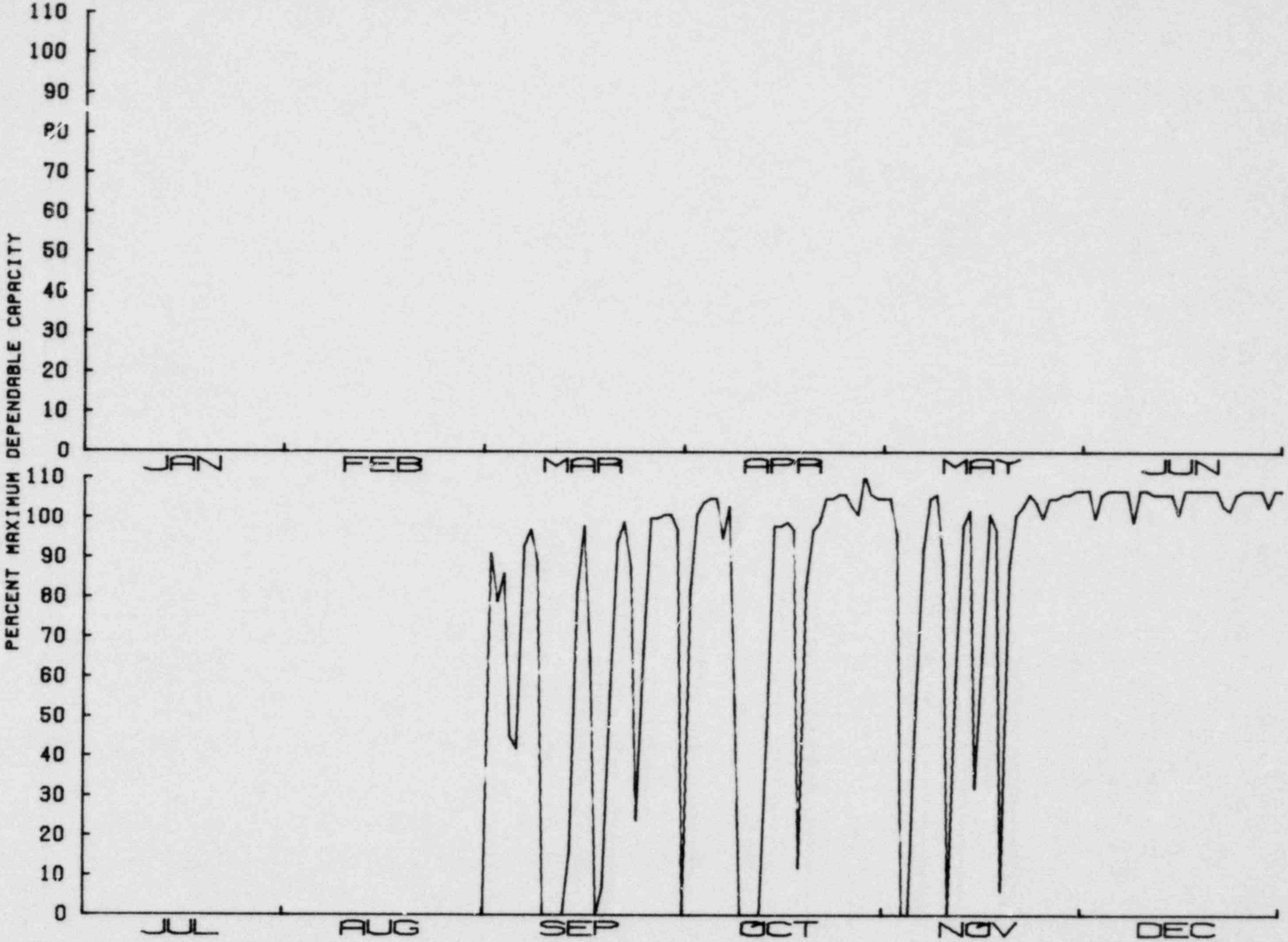
*There are 2,833 hours in the reporting period which began with declaration of commercial operation on September 5, 1979.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	9/8	107	S	Repair MSR steam leaks (note: commercial operation began 9/5/79)	B	1	Steam and power conversion (HB)	Heat exchangers
2)	9/15	41	F	Turbine control valve fast closure	A	3	Steam and power conversion (HA)	Valve operators
3)	9/22	13	S	Repair Alterrex system	B	1	Steam and power conversion (HA)	Generators (main generator exciter)
4)	9/29	28	S	Repair EHC oil leak	B	1	Steam and power conversion (HA)	Pipes, fittings
5)	10/8	109	F	Containment high pressure instrumentation (false alarm)	A	3	Instrumentation and controls (IB)	Instrumentation and controls
6)	10/18	8	S	Repair feedwater check valve	B	1	Reactor coolant (CH)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	11/3	48	S	Inoperable HPCI inboard isolation valve	B	1	Engineered safety feature (SF)	Valves
8)	11/9	13	F	MSR high level	A	3	Steam and power conversion (HB)	Instrumentation and controls
9)	11/10	14	F	MSIV closed due to momentary loss of power	A	3	Reactor coolant (CD)	Electrical conductors
10)	11/14	19	F	Loss of DC to EHC when LPCI inverter taken out of service	A	3	Electric power (ED)	Generators (inverters)
11)	11/18	19	S	Repair LPCI inverter	B	1	Electric power (ED)	Generators (inverters)



DESIGN ELEC. RATING = 784 MAX. DEPEND. CAP. = 784 (100%) HATCH 2

INDIAN POINT 2

I. Summary

Description	Performance		Outages			
Location: Indian Point, New York	Net Electrical Energy Generated (MWH):	4,804,928	Total No.	21		
Docket No: 50-247			Forced	18		
Reactor Type: PWR	Unit Availability Factor (%):	70.3	Scheduled	3		
Capacity (MWe-Net): 864			Total:	2,599	Hours,	29.7%
Commercial Operation: 8/73	Unit Capacity Factor (%) (Using MDC):	64.0	Forced	305	Hours,	3.5%
Plant Age: 6.5 Years			Scheduled	2,294	Hours,	26.2%
	Unit Capacity Factor (%) (Using Design MWE):	62.8				

II. Highlights

In January, the unit had the highest gross and net electrical generation in the plant's history. Refueling and maintenance were accomplished between June 16 and September 15. During 3 months of the year, operation was uninterrupted.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/15	7	F	Spurious signal to trip breaker	A	3	Instrumentation and controls (IA)	Instrumentation and controls
2)	2/15	8	F	BFP header nipple leak	A	3	Steam and power conversion (HH)	Pipes, fittings
3)	2/16	4	F	MBFP oil pump	A	3	Steam and power conversion (HH)	Pumps
4)	2/26	3	F	No. 24 steam generator feedwater regulator closed	A	3	Steam and power conversion (HH)	Valves
5)	2/27	12	F	No. 24 steam generator feedwater regulator closed	A	3	Steam and power conversion (HH)	Valves
6)	2/28	2	F	No. 22 S/G high level	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

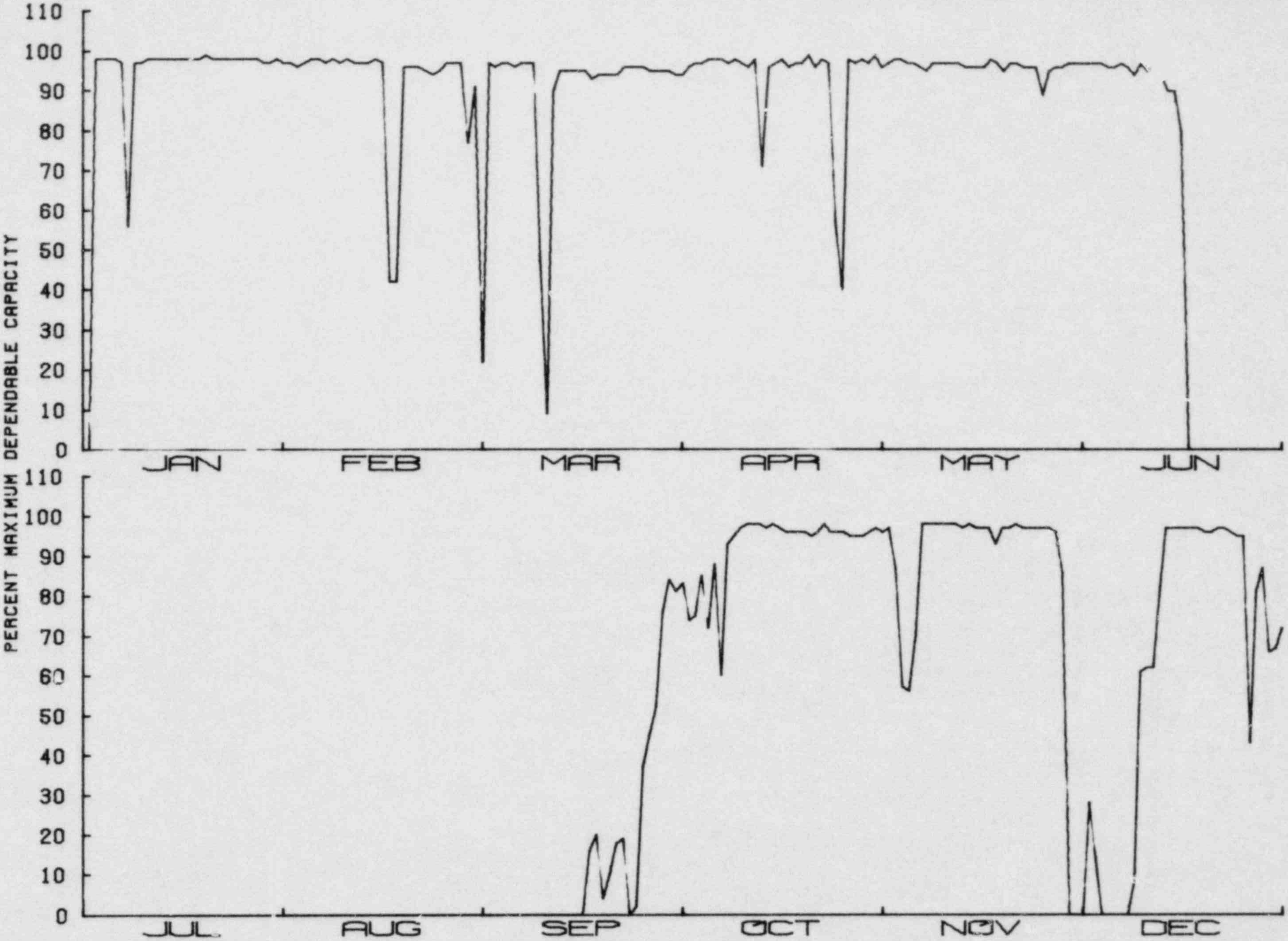
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	3/9	18	F	Trip due to loss of No. 23 instrument bus	A	3	Electric power (IA)	Electrical conductors
8)	3/10	9	F	BFP header nipple leak	A	3	Steam and power conversion (HH)	Pipes, fittings
9)	4/12	3	F	Spurious MSIV closure signal	A	3	Engineered safety features (SD)	Instrumentation and controls
10)	4/23	20	F	Malfunction of condenser steam dump system	A	3	Steam and power conversion (HE)	Valves
11)	6/16	2185	S	Refueling	C	1	Reactor (RC)	Fuel elements
12)	9/15	5	F	No. 24 steam generator high level	A	3	Steam and power conversion (HH)	Instrumentation and controls
13)	9/15	12	F	No. 23 steam generator high level	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
14)	9/18	21	F	Loss of No. 21 MBFP	A	3	Steam and power conversion (HH)	Pumps
15)	9/19	4	F	No. 21 steam generator high level	A	3	Steam and power conversion (HH)	Instrumentation and controls
16)	9/21	27	S	Turbine overspeed test	B	2	Steam and power conversion (HA)	Turbines
17)	9/23	16	F	No. 22 MBFP recirculation drain valves	A	3	Steam and power conversion (HH)	Valves
18)	9/25	3	F	No. 22 steam generator FW regulator valve	A	3	Steam and power conversion (HH)	Valves
19)	11/27	82	S	Repair steam leaks and replace motors on CRDM cooling fans	B	1	Steam and power conversion (HB)	Pipes, fittings

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
20)	12/1	2	F	Feedwater regulators — low level in No. 23 S/G	A	3	Steam and power conversion (HH)	Valves
21)	12/2	156	F	Inspect relief valve leakage into pressurizer relief tank	A	1	Reactor coolant (CJ)	Valves



DESIGN ELEC. RATING = 873 MAX. DEPEND. CAP. = 864 (100%) INDIAN POINT 2

INDIAN POINT 3

I. Summary

Description

Location: Indian Point, New York
Docket No: 50-286
Reactor Type: PWR
Capacity (MWe-Net): 965
Commercial Operation: 8/30/76
Plant Age: 3.7 Years

Performance

Net Electrical Energy Generated (MWH):	4,794,627
Unit Availability Factor (%):	66.5
Unit Capacity Factor (%) (Using MDC):	56.7
Unit Capacity Factor (%) (Using Design MWE):	56.7

Outages

Total No.	9	
Forced	6	
Scheduled	3	
Total:	2,935 Hours,	33.5%
Forced	96 Hours,	1.1%
Scheduled	2,839 Hours,	32.4%

II. Highlights

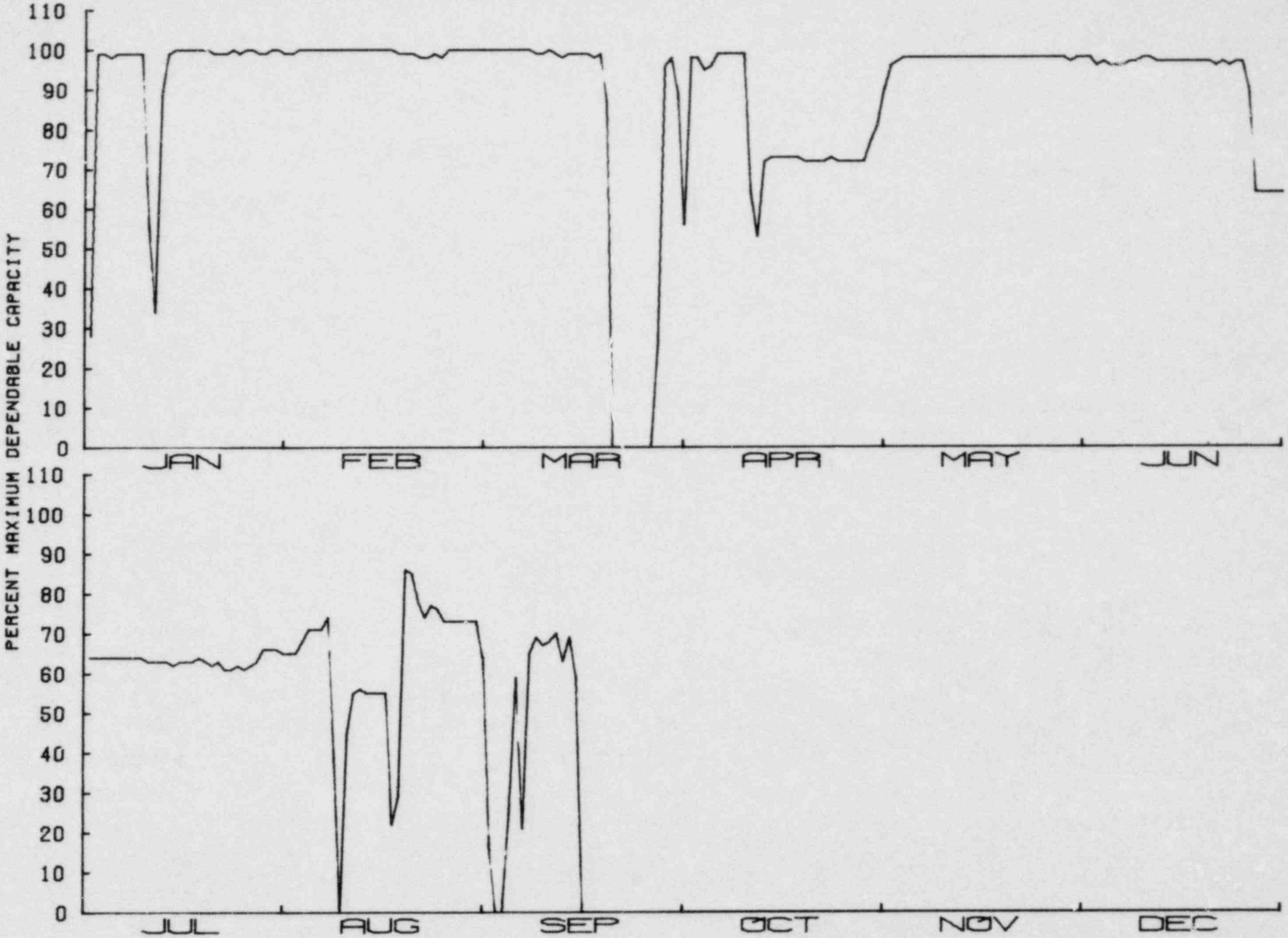
Operation was routine until June, at which time power was reduced to extend the core life to the scheduled date for a refueling and maintenance shutdown. The refueling and turbine maintenance shutdown began on September 14 and continued through the remainder of the year. There were 4 months of uninterrupted operation, with 3 months in sequence.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/9	7	F	Intermittent open circuit due to loose wires	A	3	Instrumentation and controls (IA)	Electrical conductors
2)	1/10	13	F	Intermittent open circuit due to loose wires	A	3	Instrumentation and controls (IA)	Electrical conductors
3)	3/19	181	S	Steam generator tube leak	B	2	Steam and power conversion (HB)	Heat exchangers (steam generator)
4)	4/10	8	F	Failure of control air tubing on FW reg valve	A	3	Auxiliary process (PA)	Pipes, fittings
5)	8/8	33	F	Main transformer fault	A	3	Electric power (EB)	Transformers
6)	8/17	24	F	Ground in turbine trip circuitry	A	3	Steam and power conversion (HA)	Relays

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	9/1	64	S	High amount of chlorides in steam generators due to condenser tube leaks	B	1	Steam and power conversion (HC)	Heat exchangers (condenser)
8)	9/6	11	F	Loss of #32 MBFW pump	A	3	Steam and power conversion (HH)	Pumps
9)	9/14	2594	S	Refueling	C	1	Reactor (RC)	Fuel elements



DESIGN ELEC. RATING = 965 MAX. DEPEND. CAP. = 965 (100%) INDIAN POINT 3

KEWAUNEE

I. Summary

Description

Location: Carlton, Wisconsin
Docket No: 50-305
Reactor Type: PWR
Capacity (MWe-Net): 526
Commercial Operation: 6/74
Plant Age: 5.7 Years

Performance

Net Electrical Energy Generated (MWH):	3,613,500
Unit Availability Factor (%):	79.0
Unit Capacity Factor (%) (Using MDC):	75.5
Unit Capacity Factor (%) (Using Design MWE):	73.4

Outages

Total No.	9	
Forced	6	
Scheduled	3	
Total:	1,836 Hours,	21.0%
Forced	902 Hours,	10.3%
Scheduled	934 Hours,	10.7%

II. Highlights

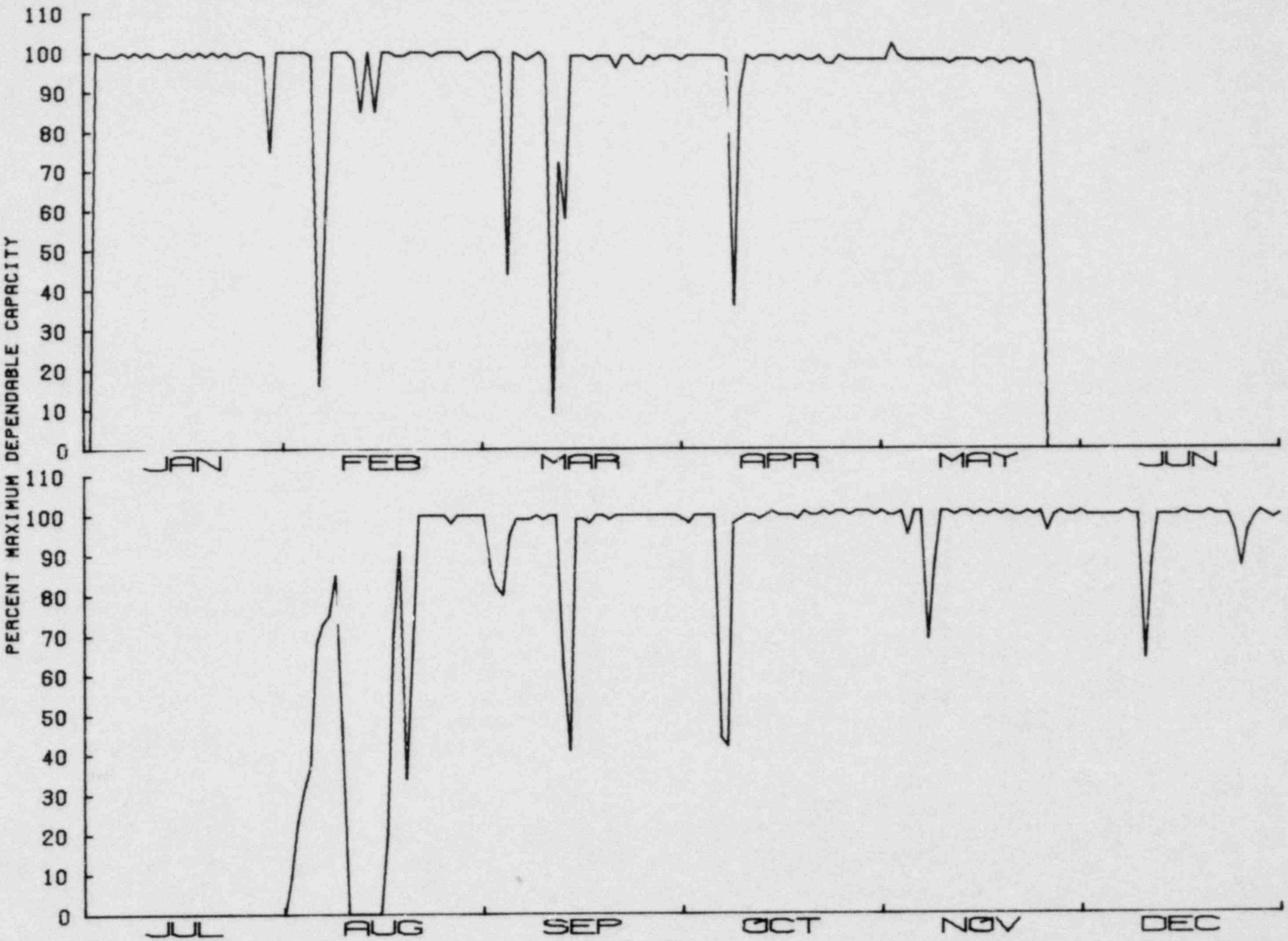
Operation was routine throughout the year. A refueling was conducted in June and July, and the outage was extended about 2 weeks for repair of nozzle-to-pipe welds on the main feedwater line to the steam generator.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/5	15	F	Low forebay level due to heavy lake ice; level below minimum required for circulation water pump	H	2	Steam and power conversion (HF)	N/A
2)	3/11	17	F	Ice blockage of circulating water inlet structure	H	2	Steam and power conversion (HF)	N/A
3)	3/12	3	F	Removal of one of the NI power range control power fuses	A	3	Instrumentation and controls (IA)	Instrumentation and controls
4a)	5/26	769	S	Refueling	C	1	Reactor (RC)	Fuel elements
4b)	5/26 (cont.)	638	F	Refueling outage extended due to additional repairs to turbine, generator, and S/G feedwater lines	A	4	Steam and power conversion (HH)	Pipes, fittings
4c)	5/26 (cont.)	46	F	Inspection of inaccessible safety-related pipe supports	D	4	Engineered safety features (SX)	Shock suppressors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
4d)	5/26 (cont.)	167	F	Correction of RCP No. 2 seal leak-off problems	A	4	Reactor coolant (CB)	Pumps
5)	8/2	6	S	Adjustment of balance weights on the turbine	B	1	Steam and power conversion (HA)	Turbines
6)	8/9	159	S	Repair of pressurizer safety valve leaks	B	1	Reactor coolant (CJ)	Valves
7)	8/19	8	F	Repair of miscellaneous drain valve leaks	A	9	Reactor coolant (CB)	Valves
8)	9/12	5	F	Occurrence of spike on one channel of over power delta T while another was in a tripped condition for repair	A	3	Instrumentation and controls (IA)	Instrumentation and controls
9)	12/10	3	F	Personnel error during testing	G	3	Instrumentation and controls (IA)	Instrumentation and controls



DESIGN ELEC. RATING = 535 MAX. DEPEND. CAP. = 517 (100%) KEWAUNEE

LACROSSE

I. Summary

Description

Location: Genoa, Wisconsin
Docket No: 50-409
Reactor Type: BWR
Capacity (MWe-Net): 48
Commercial Operation: 11/1/69
Plant Age: 11.7 Years

Performance

Net Electrical Energy Generated (MWH):	200,932
Unit Availability Factor (%):	71.8
Unit Capacity Factor (%) (Using MDC):	47.8
Unit Capacity Factor (%) (Using Design MWE):	45.9

Outages

Total No.	14	
Forced	13	
Scheduled	1	
Total:	2,475 Hours,	28.2%
Forced	897 Hours,	10.2%
Scheduled	1,578 Hours,	18.0%

II. Highlights

The year began with a self-imposed power level restriction of 48 MW(e) because of nuclear instrumentation noise at higher levels. This restriction was maintained until power was reduced in November to 40 MW(e) to extend core life to the scheduled refueling in February 1980. April and May were devoted to refueling and maintenance. Operation during the remainder of the year was routine.

DETAILS OF PLANT OUTAGES

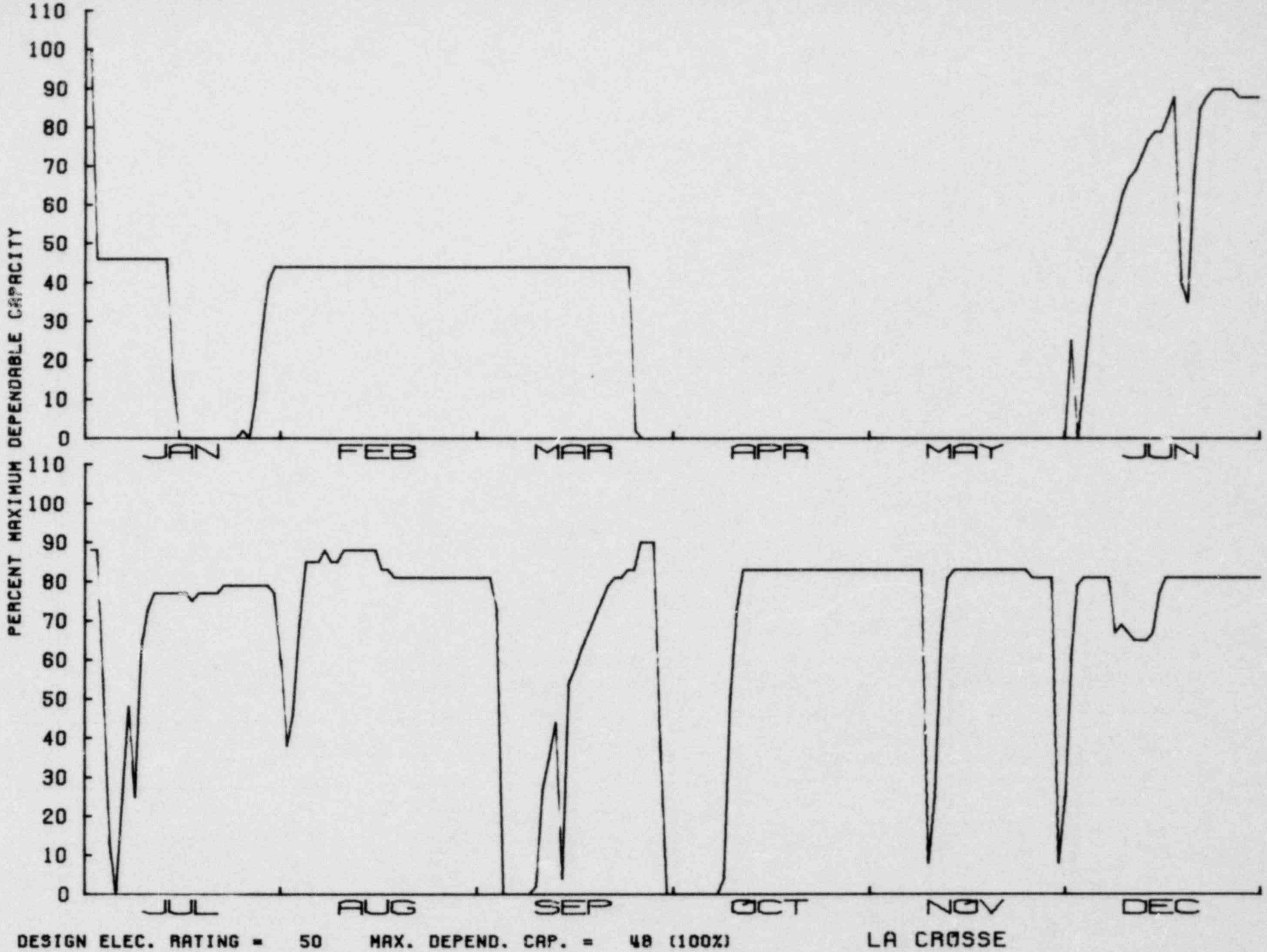
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/13	269	F	Failure of a control rod scram solenoid on CRD 13	A	3	Reactor (RB)	Control rod drives
2)	1/24	37	F	Partial scram of 13 preselected control rods due to failure of a control rod scram solenoid	A	3	Reactor (RB)	Control rod drives
3)	3/25	1578	S	Refueling	C	1	Reactor (RC)	Fuel elements
4)	5/29	9	F	Turbine inlet valve governor problems	A	9	Steam and power conversion (HA)	Mechanical function units
5)	5/30	26	F	Loss of control power due to failure of control rod scram relay	A	3	Instrumentation and controls (IA)	Relays
6)	6/1	34	F	MSIV closure due to loose wire on relay of valve control circuitry	A	3	Reactor coolant (CD)	Electrical conductors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	6/18	15	F	Loss of power to pressure transmitter closed the MSIV, causing a scram	G	3	Reactor coolant (CD)	Circuit closers/ interrupters
8)	7/3	20	F	MSIV closure when a circuit fuse was removed	G	3	Reactor coolant (CD)	Circuit closers/ interrupters
9)	7/4	34	F	Repair main steam bypass valve operating cylinder which had developed an oil leak	A	1	Steam and power conversion (HE)	Valve operators
10)	7/7	12	F	Failure of a seal injection differential pressure transmitter caused both FCPs to trip, which prompted the safey system to scram the reactor	A	3	Reactor coolant (CB)	Instrumentation and controls
11)	9/4	137	F	Repair packing on the 1A forced circulation pump discharge bypass valve and perform maintenance on turbine governor control system	A	1	Reactor coolant (CB)	Valves
12)	9/28	247	F	Mechanical seal leakage in an upper control rod drive mechanism led to seal leak-off water accumulating on a scram solenoid	A	3	Reactor (RB)	Control rod drives

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	11/9	30	F	Response problems in turbine governor control system	A	3	Steam and power conversion (HA)	Instrumentation and controls
14)	11/29	27	F	Turbine building steam isolation valve position limit switch actuation due to vibration	A	3	Reactor coolant (CD)	Instrumentation and controls


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 50 MAX. DEPEND. CAP. = 48 (100%)
LA CROSSE

MAINE YANKEE

I. Summary

Description

Location: Wincasset, Maine
Docket No: 50-309
Reactor Type: PWR
Capacity (MWe-Net): 810
Commercial Operation: 12/28/72
Plant Age: 7.1 Years

Performance

Net Electrical Energy Generated (MWH):	4,539,015
Unit Availability Factor (%):	68.4
Unit Capacity Factor (%) (Using MDC):	64.0
Unit Capacity Factor (%) (Using Design MWE):	62.8

Outages

Total No.	7	
Forced	6	
Scheduled	1	
Total:	2,766 Hours,	31.6%
Forced	2,057 Hours,	23.5%
Scheduled	709 Hours,	8.1%

II. Highlights

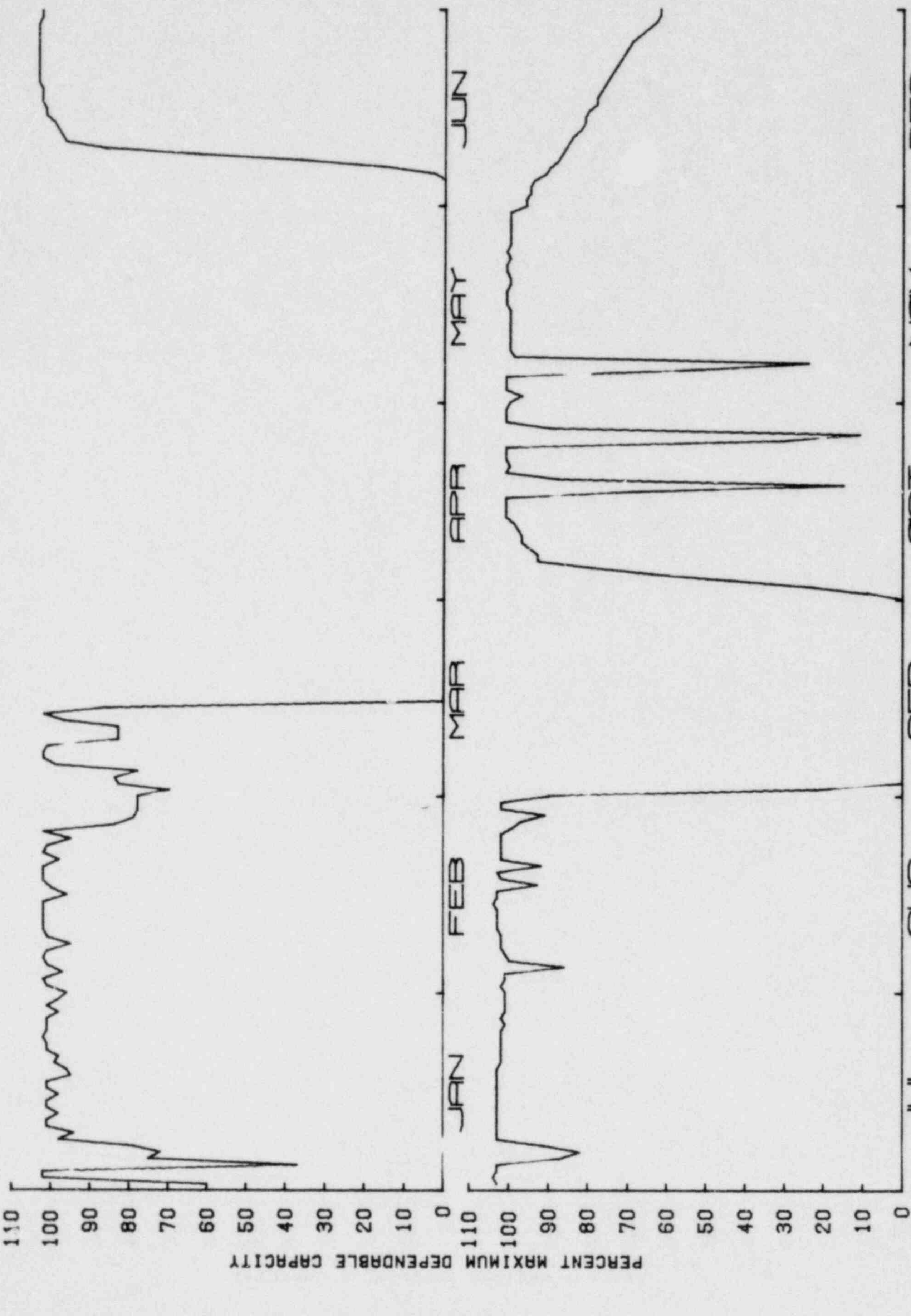
The power level was held at 97% during the year, limited by the maximum load permitted on the low-pressure turbine blading. Numerous load reductions were required during the year to permit location and plugging of leaking condenser tubes. Between March 15 and June 5, the unit was shut down to determine if modifications to safety-related piping systems were required to bring them into conformance with requirements for withstanding earthquakes (IE Bulletin 79-14). Stiffeners were added to reduce flexibility in the base plates of two pipe supports. During September, the unit was in shutdown to inspect for cracking in feedwater system piping per IE Bulletin 79-13. Coastdown began on November 24 and was continued for the remainder of the year in preparation for a refueling outage in January 1980.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/3	11	F	Electrical spike on RPS temperature channels	A	3	Instrumentation and controls (IA)	Instrumentation and controls
2)	1/5	4	F	Electrical spike on RPS temperature channels	A	3	Instrumentation and controls (IA)	Instrumentation ani controls
[illegible]	[illegible]15	1981	F	NRC show-cause order for re-evaluation of stress calculations for seismic loading of safety r[illegible]ated piping systems	D	1	Engineered safety features (SX)	Pipes, fittings
4)	9/1	709	S	Feedwater piping inspections	D	1	Steam and power conversion (HH)	Pipes, fittings
5)	10/17	13	F	Electrical spike on temperature sensing circuits	A	3	Instrumentation and controls (IA)	Instrumentation and controls
6)	10/25	30	F	Repair O-ring leak on SI[illegible] check valve hinge pin	A	1	Engineered safety features (SF)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	11/5	18	F	Electrical spike while trouble-shooting and electrical noise in RPS	A	3	Instrumentation and controls (IA)	Instrumentation and controls



DESIGN ELEC. RATING = 825 MAX. DEPEND. CAP. = 810 (100%) MAINE YANKEE

MILLSTONE 1

I. Summary

Description		Performance		Outages			
Location: Waterford, Conn.		Net Electrical Energy Generated (MWH):	4,221,264	Total No.	9		
Docket No: 50-245				Forced	6		
Reactor Type: BWR		Unit Availability Factor (%):	77.3	Scheduled	3		
Capacity (MWe-Net): 654				Total:	1,983	Hours,	22.7%
Commercial Operation: 3/71		Unit Capacity Factor (%) (Using MDC):	73.7	Forced	423	Hours,	4.8%
Plant Age: 9.1 Years				Scheduled	1,560	Hours,	17.9%
		Unit Capacity Factor (%) (Using Design MWE):	73.0				

II. Highlights

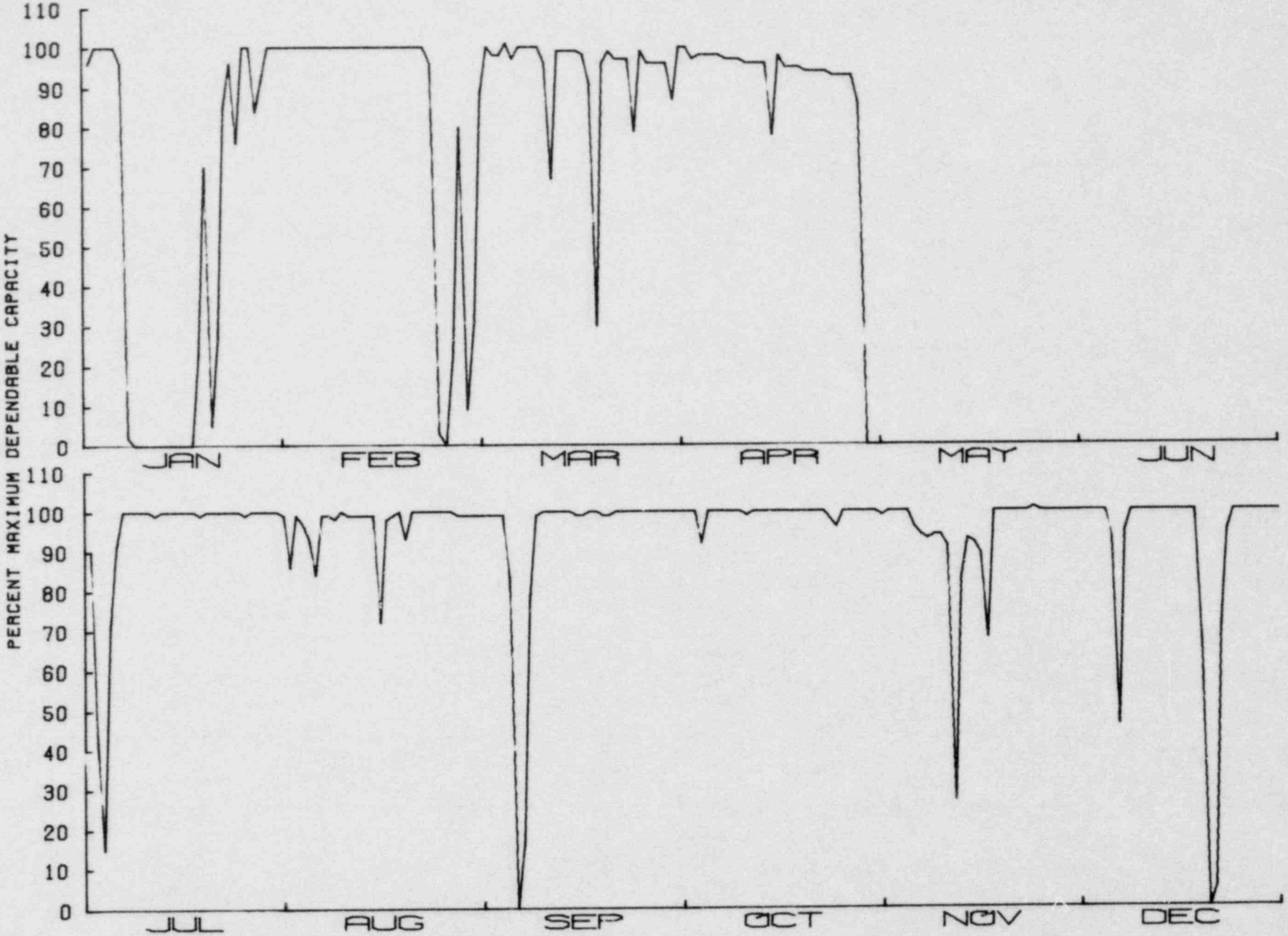
In February, while the unit was at 96% power level and 1032 psig, a blowdown occurred, which terminated at 305 psig when the Target Rock safety/relief valve reseated. The valve was replaced with one of improved design. In May and June, refueling was accomplished. During the year, there were several power reductions for maintenance, of which at least eight were for condenser maintenance.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/6	269	F	Stress corrosion cracking of clean up return line	A	1	Reactor coolant (CG)	Pipes, fittings
2)	1/19	30	F	Change "A" automatic pressure relief valve topworks	A	1	Engineered safety features (SF)	Valve operators
3)	2/22	56	S	1-IC-1 declared inoperable due to loss of position indication while returning to normal valve lineup following routine surveillance	B	1	Reactor coolant (CE)	Valve operator
4)	2/26	32	F	"F" target rock safety relief valve lifted prematurely and failed to reseat at the reset pressure during power ascension	A	2	Reactor coolant (CC)	Valves
5)	3/17	10	F	MSIV position indicating problems caused unit to be taken to hot standby for repairs	A	9	Reactor coolant (CD)	Instrumentation and controls
6)	4/28	1464	S	Refueling	C	1	Reactor (RC)	Fuel elements

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	7/2	30	F	Low water level from feedwater regulator valve lockup on loss of both plant air compressors	A	3	Auxiliary process (PA)	Blowers (compressor)
8)	9/4	40	S	Repair faulty micro switch on the isolation condenser inboard steam supply valve	B	1	Reactor coolant (CE)	Instrumentation and controls
9)	12/19	52	F	Main generator loss of excitation	A	3	Steam and power conversion (HA)	Generators (main generator exciter)



DESIGN ELEC. RATING = 660   MAX. DEPEND. CAP. = 654 (100%)   MILLSTONE 1

MILLSTONE 2

I. Summary

Description

Location: Waterford, Conn.
Docket No: 50-336
Reactor Type: PWR
Capacity (MWe-Net): 864
Commercial Operation: 12/26/75
Plant Age: 4.1 Years

Performance

Net Electrical Energy Generated (MWH):	4,363,567
Unit Availability Factor (%):	62.8*
Unit Capacity Factor (%) (Using MDC):	59.5
Unit Capacity Factor (%) (Using Design MWF):	58.6

Outages

Total No.	6	
Forced	3	
Scheduled	3	
Total:	3,371 Hours,	38.5%*
Forced	1,525 Hours,	17.4%
Scheduled	1,846 Hours,	21.1%

II. Highlights

Refueling was accomplished between March 10 and May 22. In June, a license amendment was received permitting the power level to be raised from 2560 to 2700 MW(t). In August, the unit was shut down for repair of main steam line leaks and inspection of the steam generator feedwater nozzles. Repair of the steam generator feedwater safe ends was accomplished between November 31 and December 4.

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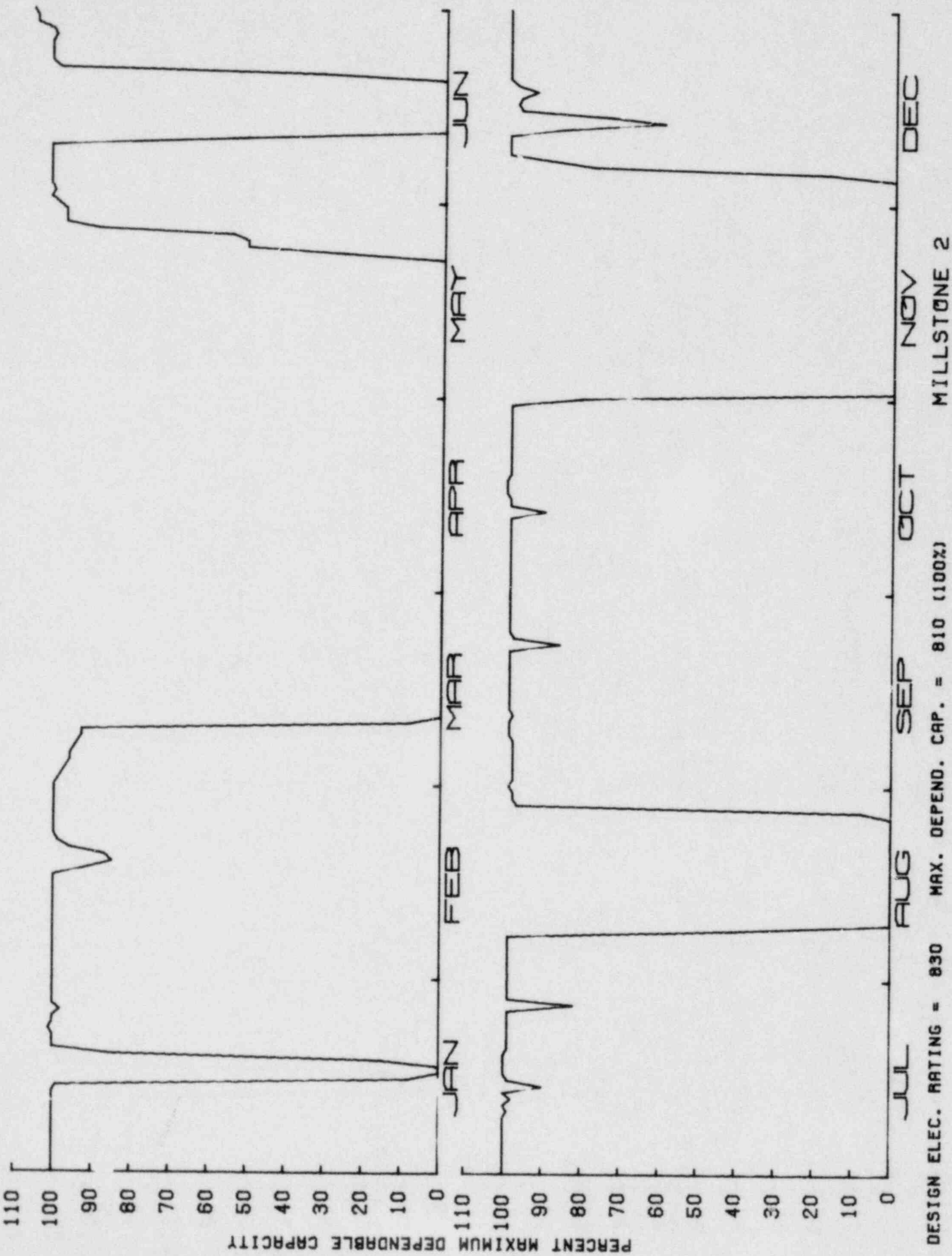
*Includes 109.4 h reserve shutdown equal to 1.2% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/4	81	S	Repair heater drain pumps, feed water heaters, and steam generator level transmitters	B	1	Steam and power conversion (HH)	Heat exchangers (FW heaters)
2)	3/10	1763	S	Incore detector response test; at this time cycle 3 refueling commenced	C	1	Reactor (RC)	Fuel elements
3)	5/22	2	S	Turbine overspeed trip test	B	3	Steam and power conversion (HA)	Turbines
4)	6/10	235	F	Body to bonnet leak on 2-RC-405	A	1	Reactor coolant (CB)	Valves
5)	8/8	461	F	Main steam line leak repairs	A	1	Steam and power conversion (HB)	Pipes, fittings
6)	10/31	829	F	Repair of indications in feedwater piping safe ends	A	1	Steam and power conversion (HH)	Pipes, fittings


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 830 MAX. DEPEND. CAP. = 810 (100%)
MILLSTONE 2

MONTICELLO

I. Summary

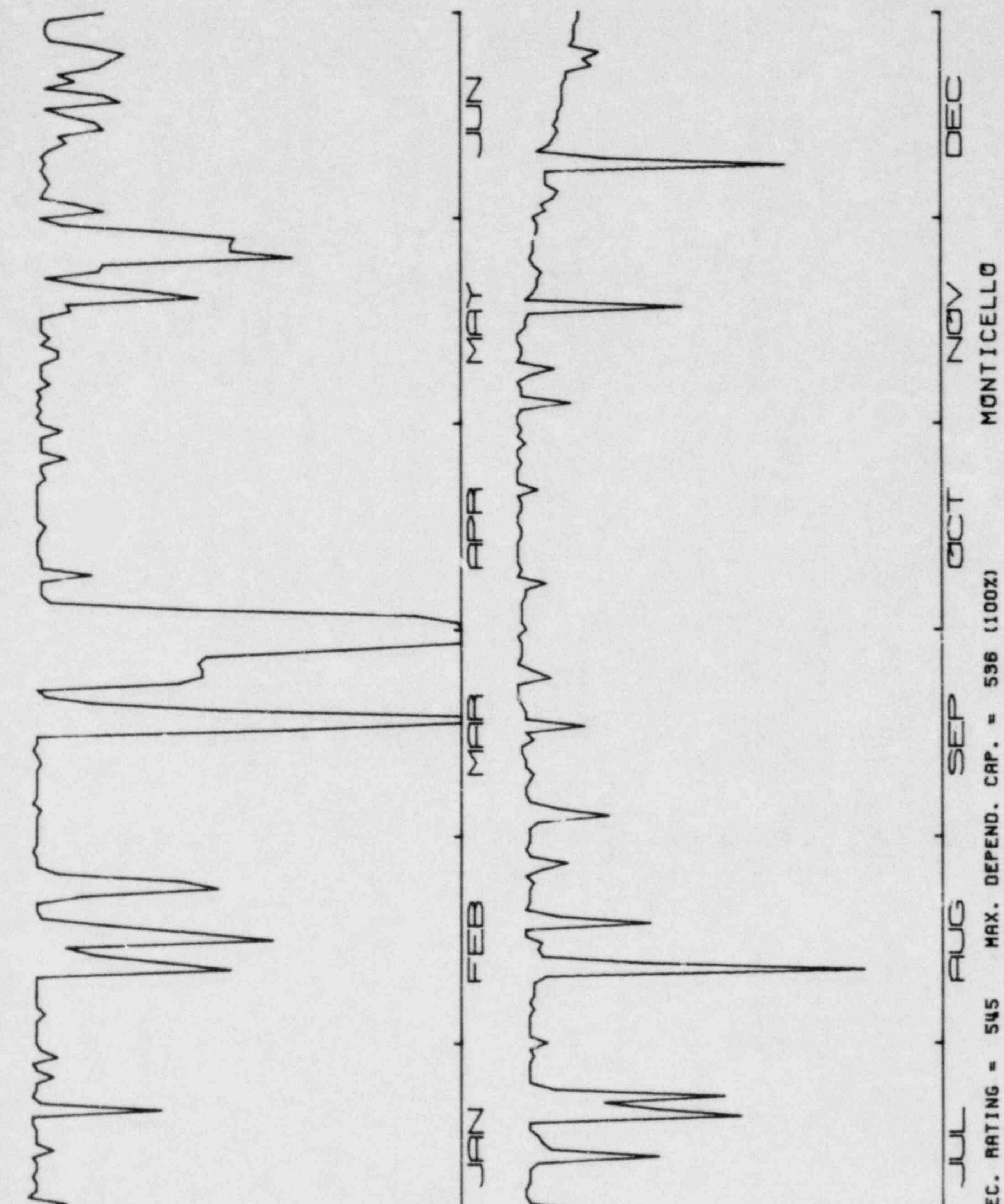
Description	Performance		Outages			
Location: Monticello, Minn.	Net Electrical Energy Generated (MWH):	4,399,560	Total No.	6		
Docket No: 50-263	Unit Availability Factor (%):	97.6	Forced	5		
Reactor Type: BWR	Unit Capacity Factor (%) (Using MDC):	93.7	Scheduled	1		
Capacity (MWe-Net): 536	Unit Capacity Factor (%) (Using Design MWE):	92.2	Total:	210	Hours,	2.4%
Commercial Operation: 6/30/71			Forced	156	Hours,	1.8%
Plant Age: 8.8 Years			Scheduled	54	Hours,	0.6%

II. Highlights

Operation was routine all year, with an accumulated availability of 97.6%. There were 8 months of uninterrupted operation, with 5 months in sequence from August through the end of the year. On November 19, with rods fully withdrawn and full recirculation flow, the unit began coastdown for refueling in February 1980. At the end of the year, the power level was 87%.

DETAILS OF PLANT OUTAGES

No.	Date (1973)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/14	12	F	Scrammed during surveillance test on APRM bias instrumentation due to instrument valving problem	A	3	Instrumentation and controls (IA)	Valves
2)	3/16	54	S	Repair 2 relief valves and miscellaneous maintenance	B	1	Reactor coolant (CC)	Valves
3)	3/28	19	F	Scram on low level following trip of reactor feedwater pump	A	3	Reactor coolant (CH)	Pumps
4)	3/29	107	F	Turbine lockout caused by the thrust bearing wear indicator	A	3	Steam and power conversion (HA)	Turbines
5)	7/20	11	F	Malfunction of master level controller	A	3	Reactor coolant (CH)	Instrumentation and controls
6)	7/23	7	F	Malfunction of master level controller	A	3	Reactor coolant (CH)	Instrumentation and controls


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 545 MAX. DEPEND. CAP. = 536 (100%)
MONTICELLO

NINE MILE POINT 1

I. Summary

Description		Performance		Outages			
Location: Scriba, New York		Net Electrical Energy Generated (MWH):	3,005,389	Total No.	7		
Docket No: 50-220				Forced	2		
Reactor Type: BWR		Unit Availability Factor (%):	66.1	Scheduled	5		
Capacity (MWe-Net): 610				Total:	2,972	Hours,	33.9%
Commercial Operation: 12/69		Unit Capacity Factor (%) (Using MDC)	56.2	Forced	57	Hours,	0.7%
Plant Age: 10.2 Years				Scheduled	2,915	Hours,	33.2%
		Unit Capacity Factor (%) (Using Design MWE):	55.3				

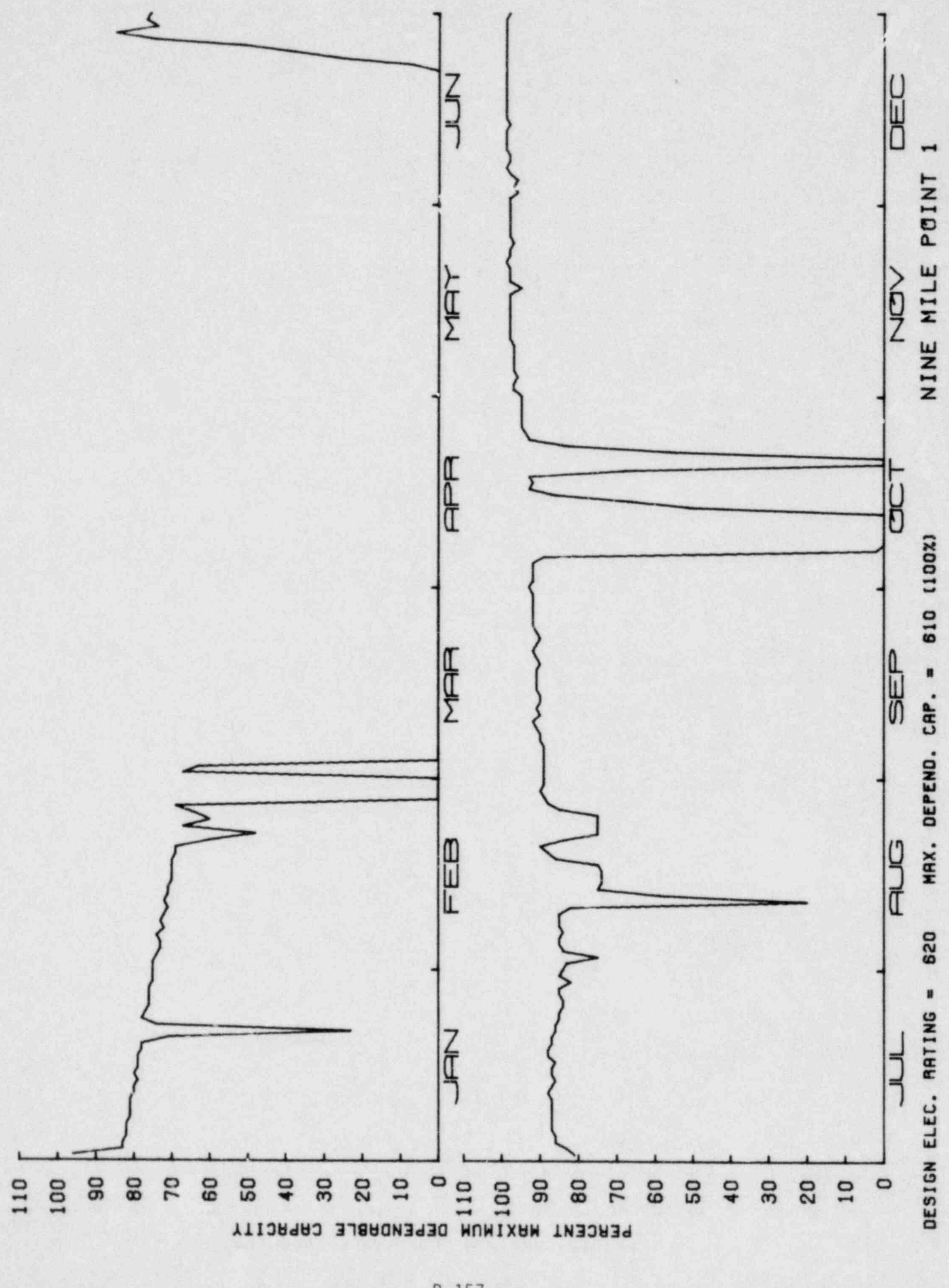
B-155

II. Highlights

At the beginning of the year, the unit was at 75% power in an end-of-cycle coastdown in preparation for the refueling and overhaul outage which was started March 3 and completed June 21. During the refueling outage, field verification was made of seismic restraints pursuant to IE Bulletin 79-02; it was found that 22 restraints were not installed in the containment spray piping inside the reactor building. Upon returning to operation on June 21, power was restricted to 90—95% because one of five loops was inoperative due to damaged recirculation pump internals. In October, adjustments were made to the motor-generator sets of the other recirculation pumps, and full-power operation was attained. Operation was continuous in September and December.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/19	11	S	Quarterly testing on core spray isolation valves	B	1	Engineered safety features (SF)	Valves
2)	2/19	53	F	Drain valve packing leak	A	1	Reactor coolant (CB)	Valves
3)	3/3	2673	S	Refueling and overhaul	C	1	Reactor (RC)	Fuel elements
4)	8/11	6	S	Balance turbine	B	1	Steam and power conversion (HA)	Turbine
5)	8/11	4	F	No. 13 FW pump had to be clutched manually	A	1	Reactor coolant (CH)	Pumps
6)	10/6	171	S	Repair No. 12 reactor recirculation pump	B	1	Reactor coolant (CB)	Pumps
7)	10/19	54	S	Repair No. 12 reactor recirculation pump	B	1	Reactor coolant (CB)	Pumps


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 620 MAX. DEPEND. CAP. = 610 (100%)
NINE MILE POINT 1

NORTH ANNA 1

I. Summary

Description	Performance		Outages		
Location: Mineral, Virginia	Net Electrical Energy		Total No.	8	
Docket No: 50-338	Generated (MWH):	4,188,866	Forced	8	
Reactor Type: PWR	Unit Availability		Scheduled	0	
Capacity (MWe-Net): 898	Factor (%):	61.7	Total:	3,358 Hours,	38.3%
Commercial Opération: 6/6/78	Unit Capacity Factor (%)		Forced	1,014 Hours,	11.6%
Plant Age: 1.7 Years	(Using MDC):	53.2	Scheduled	2,344 Hours,	26.7%
	Unit Capacity Factor (%)				
	(Using Design MWE):	52.7			

II. Highlights

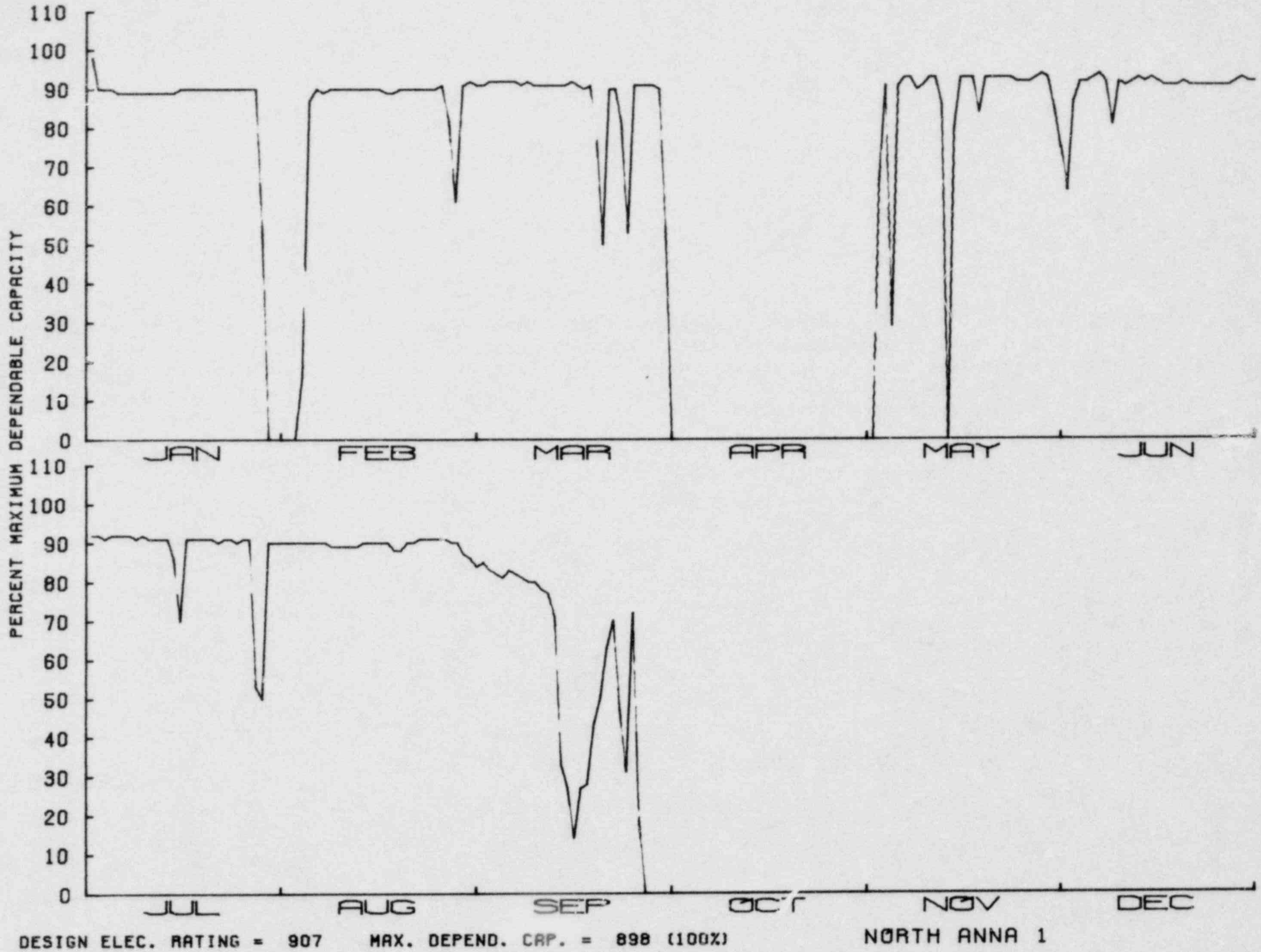
Operation was routine during the year, with operation during the months of June, July, and August being continuous. A refueling and maintenance outage started on September 25 was still in progress at the end of the year.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/27	167	F	Excessive unidentified primary plant leakage	A	1	Reactor coolant (CB)	Pipes, fittings
2)	2/24	7	F	Repair leaking pump discharge header	A	9	Reactor coolant (CB)	Pipes, fittings
3)	3/19	11	F	Leak of electrohydraulic fluid from pump discharge header	A	9	Steam and power conversion (HA)	Pipes, fittings
4)	3/30	783	F	Loss of cooling water from "A" reactor cooling pump; reactor remained shutdown for inspection and maintenance	A	1	Auxiliary water (WB)	Valve operators
5)	5/3	15	F	S/G low level with feed flow/steam flow mismatch; insufficient feedwater flow when feedpump was tripped due to loss of oil flow	A	3	Steam and power conversion (HH)	Pumps
6)	5/12	27	F	Perform periodic testing on SI system and maintenance on "C" steam generator	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	5/18	1	F	Feed flow/steam mismatches and low S/G water level	G	3	Steam and power conversion (HH)	Instrumentation and controls
8a)	9/25	3	F	Turbine generator trip due to high level in a feedwater heater	A	3	Steam and power conversion (HH)	Instrumentation and controls
8b)	9/25	2344	S	Refueling	C	4	Reactor (RC)	Fuel elements


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 907 MAX. DEPEND. CAP. = 898 (100%)
NORTH ANNA 1

OCONEE 1

I. Summary

Description

Location: Seneca, South Carolina
Docket No: 50-269
Reactor Type: PWR
Capacity (MWe-Net): 860
Commercial Operation: 7/15/73
Plant Age: 6.7 Years

Performance

Net Electrical Energy Generated (MWH):	5,000,177
Unit Availability Factor (%):	71.0
Unit Capacity Factor (%) (Using MDC):	66.4
Unit Capacity Factor (%) (Using Design MWE):	64.4

Outages

Total No.	16	
Forced	15	
Scheduled	1	
Total:	2,537 Hours,	29.0%
Forced	1,392 Hours,	15.9%
Scheduled	1,145 Hours,	13.1%

II. Highlights

Operation was routine except for a shutdown required by the NRC relative to TMI-2-related modifications and pipe hangers. In May, a short outage (11.6 h) was required to test the emergency feedwater system, and in June a longer outage (10 days) was required for testing the availability of the emergency feedwater pump and to perform necessary modifications. Refueling was in progress from November 21 through the remainder of the year; during this time, pipe hangers/supports were inspected in accordance with IE Bulletins 79.02 and 79-14.

DETAILS OF PLANT OUTAGES

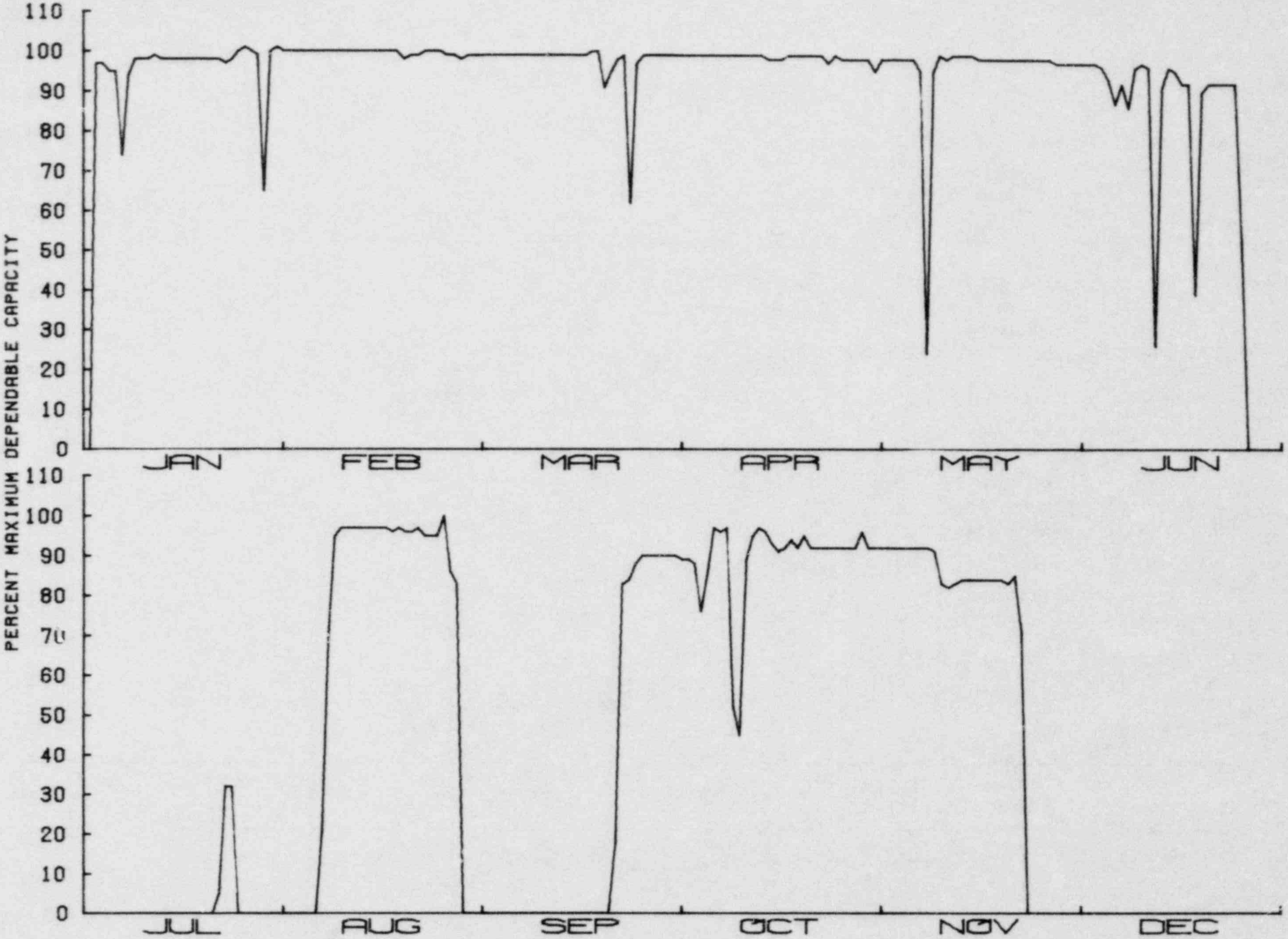
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/23	4	F	Instrumentation in loop "B" feedwater flow failed causing a high RCS pressure trip	A	3	Steam and power conversion (HH)	Instrumentation and controls
2)	5/7	12	F	NRC order to test emergency feedwater pumps	D	3	Steam and power conversion (HH)	Pumps
3)	6/11	12	F	Closure of all intercept valves due to faulty EHC system computer card	A	3	Steam and power conversion (HA)	Instrumentation and controls
4)	6/17	6	F	Switch gear problem	A	3	Electric power (EB)	Relays
5a)	6/24	57	F	NRC order to test emergency feedwater pump availability	D	1	Steam and power conversion (HH)	Pumps
5b)	6/24 (cont.)	183	S	NRC-required modifications and testing of the emergency feedwater system resulted in the unit being out the remainder of June	D	4	Steam and power conversion (HH)	Pumps

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	7/4	355	F	Low pressure injection cooler tube leak	A	3	Engineered safety features (SF)	Heat exchangers
7)	7/19	1	F	Penetration room humidity level high due to operation of relief valve FDW-295	A	2	Steam and power conversion (HB)	Valves
3)	7/19	28	F	Water chemistry out of spec in steam generators	A	1	Steam and power conversion (HG)	Demineralizers
9)	7/20	24	F	High RC pressure while performing RCS leak test	A	3	Instrumentation and controls (IA)	Instrumentation and controls
10)	7/24	309	F	Tube leak in "B" steam generator	A	1	Steam and power conversion (HB)	Heat exchangers (steam generators)
11)	8/6	5	F	Pressure transmitter problem on the feedwater pumps	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12)	8/27	157	F	Excessive packing leakage on RCS instrument valves	A	1	Instrumentation and controls (IA)	Valves
13a)	9/3	353	F	Leak in reactor "O" ring seal	A	1	Reactor coolant (CA)	Pressure vessels
13b)	9/18	11	F	Failure of PI tube	A	4	Instrumentation and controls (ID)	Pipes, fittings
13c)	9/18	39	F	Hold in heatup to repair a CRD stator connector on group 1	A	4	Reactor (RB)	Control rod drives
14)	9/20	6	F	Turbine control problem prevented warmup of shell	A	1	Steam and power conversion (HA)	Turbines
15)	10/8	13	F	Reactor tripped during routine power supply test for CRD system due to personnel error during test	G	3	Reactor (RB)	Control rod drives
16)	11/21	962	S	Refueling	C	1	Reactor (RC)	Fuel elements



DESIGN ELEC. RATING = 887 MAX. DEPEND. CAP. = 860 (100%) OCONEE 1

OCONEE 2

I. Summary

Description

Location: Seneca, South Carolina
Docket No: 50-270
Reactor Type: PWR
Capacity (MWe-Net): 860
Commercial Operation: 9/9/74
Plant Age: 6.1 Years

Performance

Net Electrical Energy Generated (MWH):	5,968,288
Unit Availability Factor (%):	86.0
Unit Capacity Factor (%) (Using MDC):	79.2
Unit Capacity Factor (%) (Using Design MWE):	76.8

Outages

Total No.	14		
Forced	13		
Scheduled	1		
Total:	1,226	Hours,	14.0%
Forced	1,208	Hours,	13.8%
Scheduled	18	Hours,	0.2%

II. Highlights

During the summer months, several power reductions and maintenance were required as a result of condenser tube leaks. In September, an outage was required to tie in the motor-driven emergency feedwater pumps. There were 5 months in which operation was uninterrupted, November and December being the only months in sequence.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/4	18	S	Periodic inaccessible shock suppressor inspection	D	1	Engineered safety features (SF)	Shock suppressors
2)	3/11	3	F	Repair turbine control valve problem	A	9	Steam and power conversion (HA)	Valves
3)	5/7	11	F	NRC order to test emergency feedwater system	D	9	Steam and power conversion (HH)	Pumps
4a)	5/11	139	F	Condenser tube leak	A	1	Steam and power conversion (HC)	Heat exchangers (condenser)
4b)	5/17	94	F	Emergency hatch leak rate test	A	4	Engineered safety features (SD)	Primary containment penetrations
4c)	5/21	9	F	Hold in heatup due to chemistry problem	A	4	Steam and power conversion (HG)	Demineral-izers

DETAILS OF PLANT OUTAGES

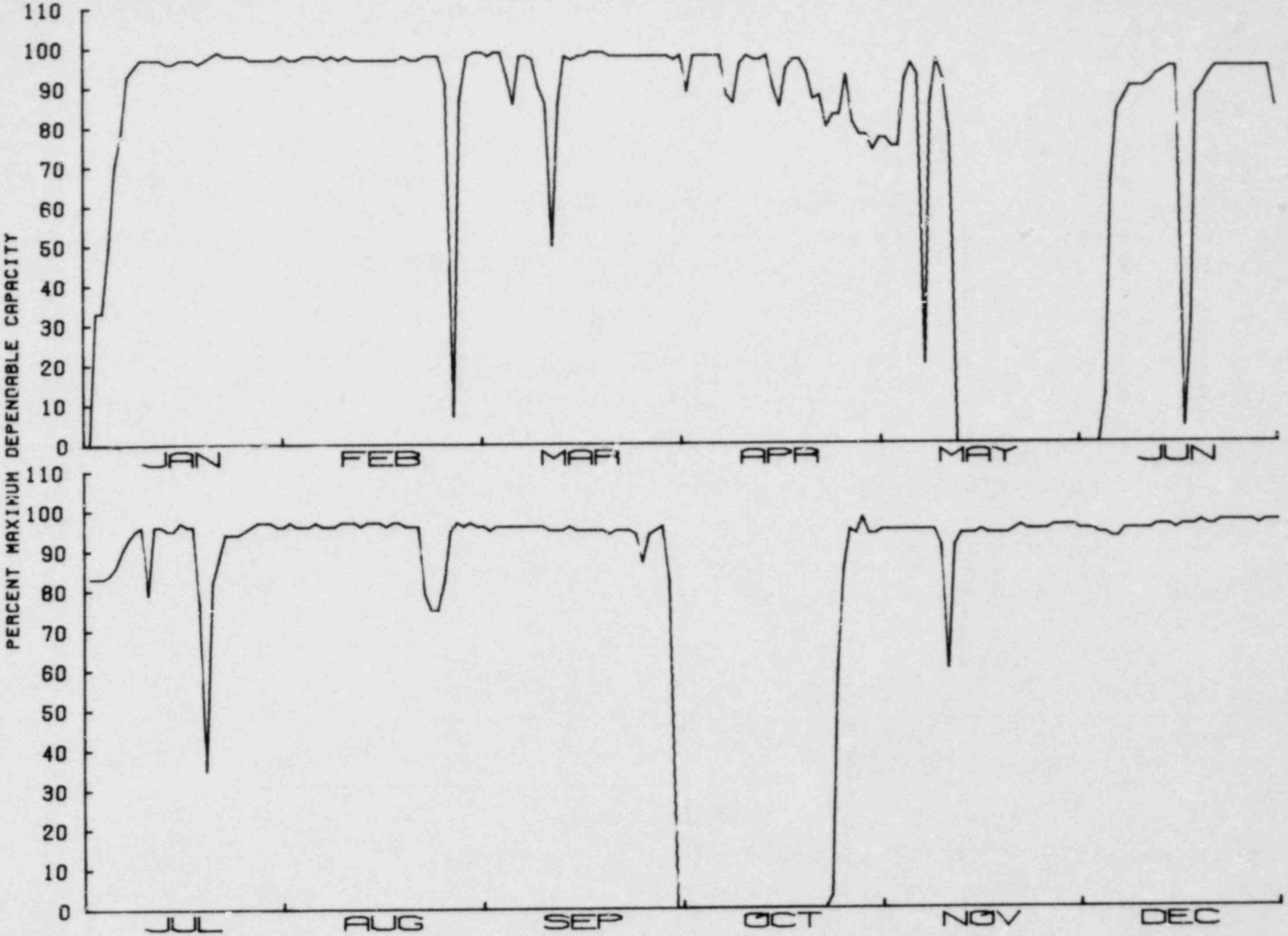
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
4d)	5/22	17	F	Emergency feedwater pump out of service	A	4	Steam and power conversion (HH)	Pumps
4e)	5/22	45	F	Replace nonqualified valve operator	H	4	Engineered safety features (SF)	Valve operators
4f)	5/24	56	F	Cooldown to repair valve	A	4	Reactor coolant (CB)	Valves
4g)	5/27	37	F	Hold in heatup to replace gasket on feedwater valve	A	4	Steam and power conversion (HH)	Valves
4h)	5/28	45	F	Cooldown to repair leak between valve 2N-233 and main steam line B	A	4	Steam and power conversion (HB)	Pipes, fittings
4i)	5/30	38	F	OTSG chemistry out of spec	A	4	Steam and power conversion (HG)	Demineralizers

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
4j)	6/1	68	F	OTSG chemistry out of spec	A	4	Steam and power conversion (HG)	Demineralizers
5)	6/3	9	F	Feedwater system oscillations and CRD problems	A	3	Steam and power conversion (HH)	Instrumentation and controls
6)	6/16	30	F	Valve 2RC-2 packing leak	A	1	Reactor coolant (CB)	Valves
7)	7/18	10	F	Unit tripped by relay operacion after a line fault due to lightning	H	3	Electric power (EA)	Electrical conductors
8)	9/28	364	F	Modification required by NRC for tie-in of motor driven emergency feedwater pumps	D	1	Steam and power conversion (HH)	Pumps
9)	10/14	26	F	Water chemistry out of spec in steam generators	A	2	Steam and power conversion (HG)	Demineralizers

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10)	10/15	127	F	Flange leak between 2RC-68 and pressurizer	A	2	Reactor coolant (CB)	Pipes, fittings
11)	10/20	1	F	Water chemistry out of spec in steam generators	A	2	Steam and power conversion (HG)	Demineralizers
12)	10/20	27	F	Valve 2HP-306 (RCP seal return line drain valve) failure	A	2	Reactor coolant (CB)	Valves
13)	10/21	51	F	Water chemistry out of spec in steam generators	A	2	Steam and power conversion (HG)	Demineralizers
14)	10/23	1	F	False high level indication on MSRH A-1 drain tank	A	3	Steam and power conversion (HB)	Instrumentation and controls



DESIGN ELEC. RATING = 887 MAX. DEPEND. CAP. = 860 (100%)

OCONEE 2

OCONEE 3

I. Summary

Description	Performance		Outages		
Location: Seneca, South Carolina	Net Electrical Energy Generated (MWH):	3,259,529	Total No.	4	
Docket No: 50-287			Forced	4	
Reactor Type: PWR	Unit Availability Factor (%):	46.1	Scheduled	0	
Capacity (MWe-Net): 860			Total:	4,721 Hours,	53.9%
Commercial Operation: 12/16/74	Unit Capacity Factor (%) (Using MDC):	43.3	Forced	3,224 Hours,	36.8%
Plant Age: 5.3 Years			Scheduled	1,497 Hours,	17.1%
	Unit Capacity Factor (%) (Using Design MWE):	41.9			

II. Highlights

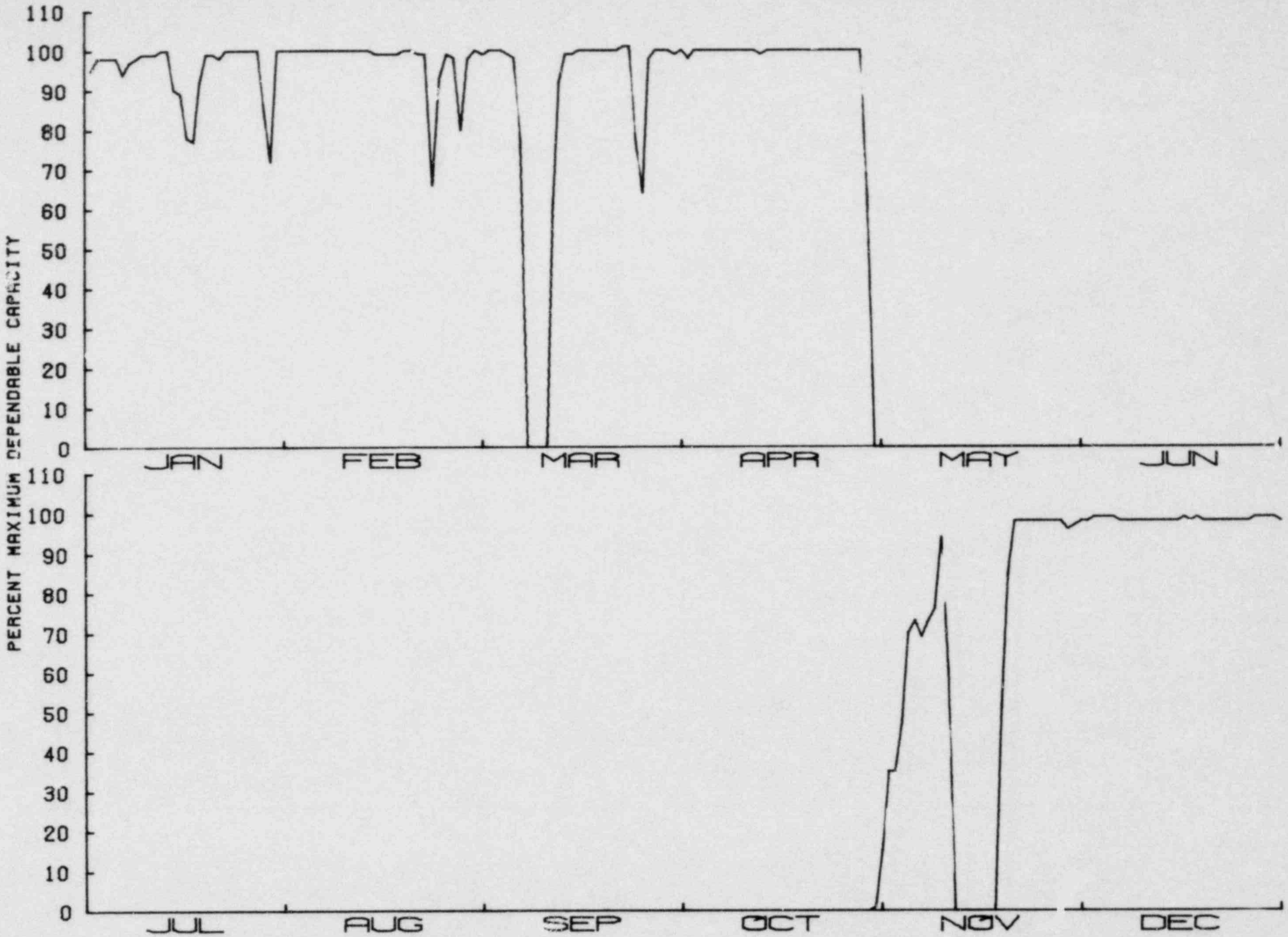
A major extended outage was in effect from April 28 to October 30. During the outage, the following tasks were accomplished: (1) refueling, (2) investigation of possible TMI-2-related safety problems, (3) examination of pipe support base plates using concrete expansion anchor bolts per IE Bulletin 79-02, and (4) seismic analysis of safety-related piping per IE Bulletin 79-14. Except for a few power reductions, operation for the remainder of the year was routine, with operation in January and December being uninterrupted.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/21	3	F	High RCS pressure while returning ICS to auto due to a FDW demand signal problem	A	3	Steam and power conversion (HH)	Instrumentation and controls
2)	3/6	101	F	Flow transmitter isolation valve on RCS "B" hot leg leaking	A	1	Reactor coolant (CB)	Valves
3a)	4/28	169	F	Investigation and modification of possible safety problems related to the TMI-2 accident	D	1	Steam and power conversion (HH)	Instrumentation and controls
3b)	4/28	1344	S	Refueling	C	4	Reactor (RC)	Fuel elements
3c)	7/1	2771	F	Refueling has been completed but unit remains shutdown for pipe support inspections and modifications	D	4	Engineered safety features (SF)	Shock suppressors
3d)	10/24	5	S	HPI flow test	B	4	Engineered safety features (SF)	Pumps

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
3e)	10/24	148	S	Power physics testing	B	4	Reactor (RC)	Fuel elements
3f)	10/31	8	F	Trip due to high level in MSRH drain tank	A	4	Steam and power conversion (HB)	Instrumentation and controls
4)	11/10	172	F	ICS inverter problem	A	3	Electric power (ED)	Generators (Inverters)



DESIGN ELEC. RATING = 887 MAX. DEPEND. CAP. = 860 (100%) OCONEE 3

OYSTER CREEK

I. Summary

Description

Location: Toms River, New Jersey
Docket No: 50-219
Reactor Type: BWR
Capacity (MWe-Net): 620
Commercial Operation: 12/69
Plant Age: 10.3 Years

Performance

Net Electrical Energy Generated (MWH):	4,563,223
Unit Availability Factor (%):	85.9
Unit Capacity Factor (%) (Using MDC):	84.0
Unit Capacity Factor (%) (Using Design MWE):	80.1

Outages

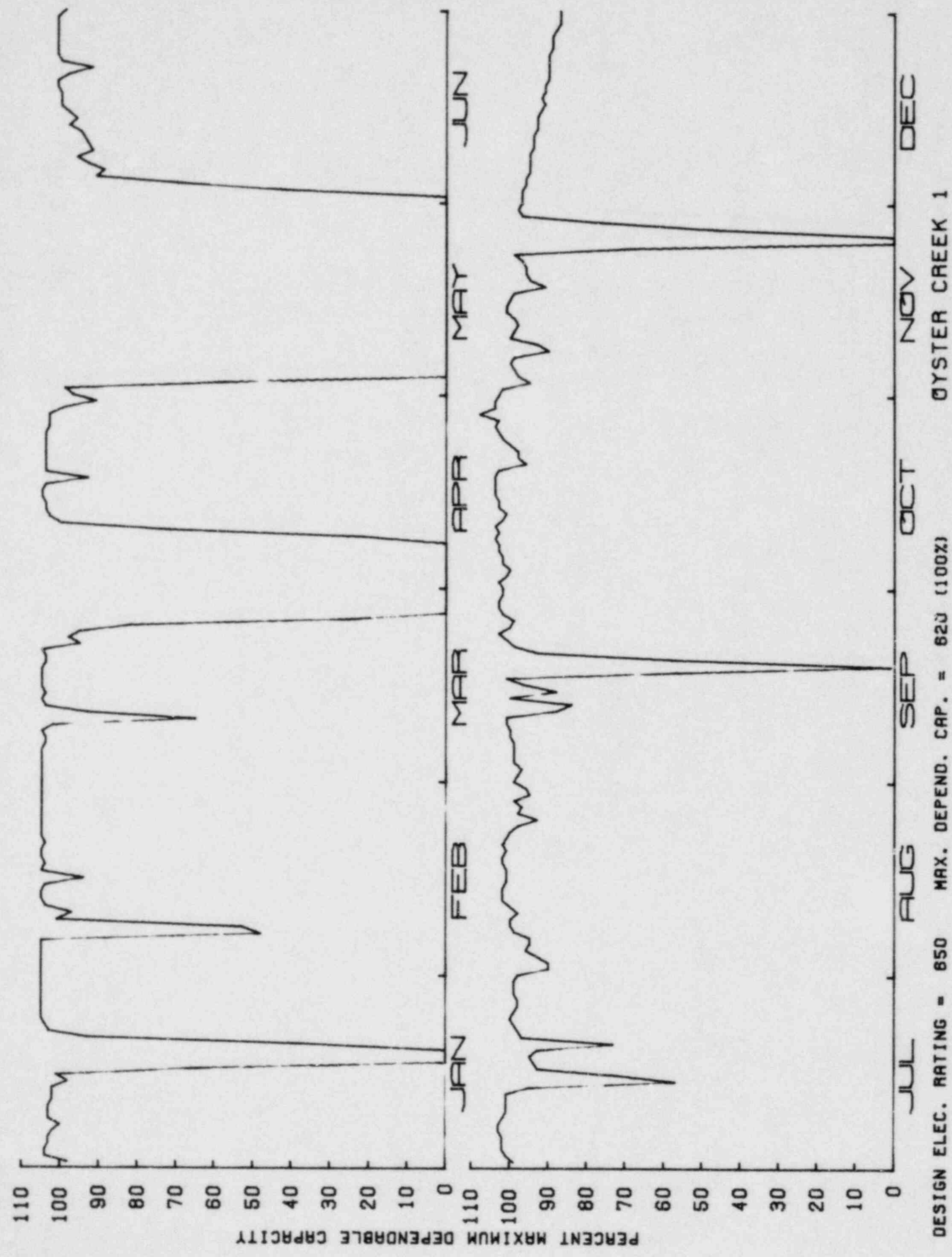
Total No.	6	
Forced	6	
Scheduled	0	
Total:	1,236 Hours,	14.1%
Forced	1,236 Hours,	14.1%
Scheduled	0 Hours,	0%

II. Highlights

Operation was routine except for an outage in May. A scram occurred during testing, and all recirculation pump discharge valves closed and feedwater was lost, resulting in triple-low water level for 36 min. The NRC approved restarting to test for core damage on May 24, and the unit was returned to service on June 2. During the last half of the year, there were 4 months of uninterrupted operation, with 2 months in sequence. At the end of the year, the unit was in end-of-cycle coastdown for refueling in January 1980.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/15	86	F	Cleanup system pipe high vibration	G	3	Reactor coolant (CG)	Pipes, fittings
2)	2/6	17	F	"C" feedwater pump tripped when breaker cubicle door was closed, shaking the "C" differential relay; scram occurred from a low water level after the pump trip	G	3	Reactor coolant (CH)	Relays
3)	3/26	316	F	Repair "D" recirculation pump seal	A	1	Reactor coolant (CB)	Pumps
4)	5/2	728	F	A reactor high pressure scram during testing caused all the recirculation pump discharge valves to close resulting in a triple low water level above the core for 36 min	A	3	Reactor coolant (CB)	Valves
5)	9/17	32	F	A worker struck a cable tray attached to a reactor protection system instrument rack	G	3	Instrumentation and controls (IA)	Electrical conductors
6)	11/23	57	F	Inadvertent opening of an isolation condenser return valve during backseating	G	3	Reactor coolant (CE)	Valves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DESIGN ELEC. RATING = 650 MAX. DEPEND. CAP. = 620 (100%)
OYSTER CREEK 1

PALISADES

I. Summary

Description

Location: South Haven, Michigan
Docket No: 50-255
Reactor Type: PWR
Capacity (MWe-Net): 635
Commercial Operation: 12/71
Plant Age: 8.0 Years

Performance

Net Electrical Energy Generated (MWH):	3,433,264
Unit Availability Factor (%):	59.9
Unit Capacity Factor (%) (Using MDC):	61.7
Unit Capacity Factor (%) (Using Design MWE):	48.7

Outages

Total No.	11		
Forced	10		
Scheduled	1		
Total:	3,515	Hours,	40.1%
Forced	760	Hours,	8.7%
Scheduled	2,755	Hours,	31.4%

II. Highlights

Operation was routine during the year except for an outage in May to add seismic snubbers to two safety injection lines. At the end of the year, the unit was still in a refueling and maintenance outage that began in September.

DETAILS OF PLANT OUTAGES

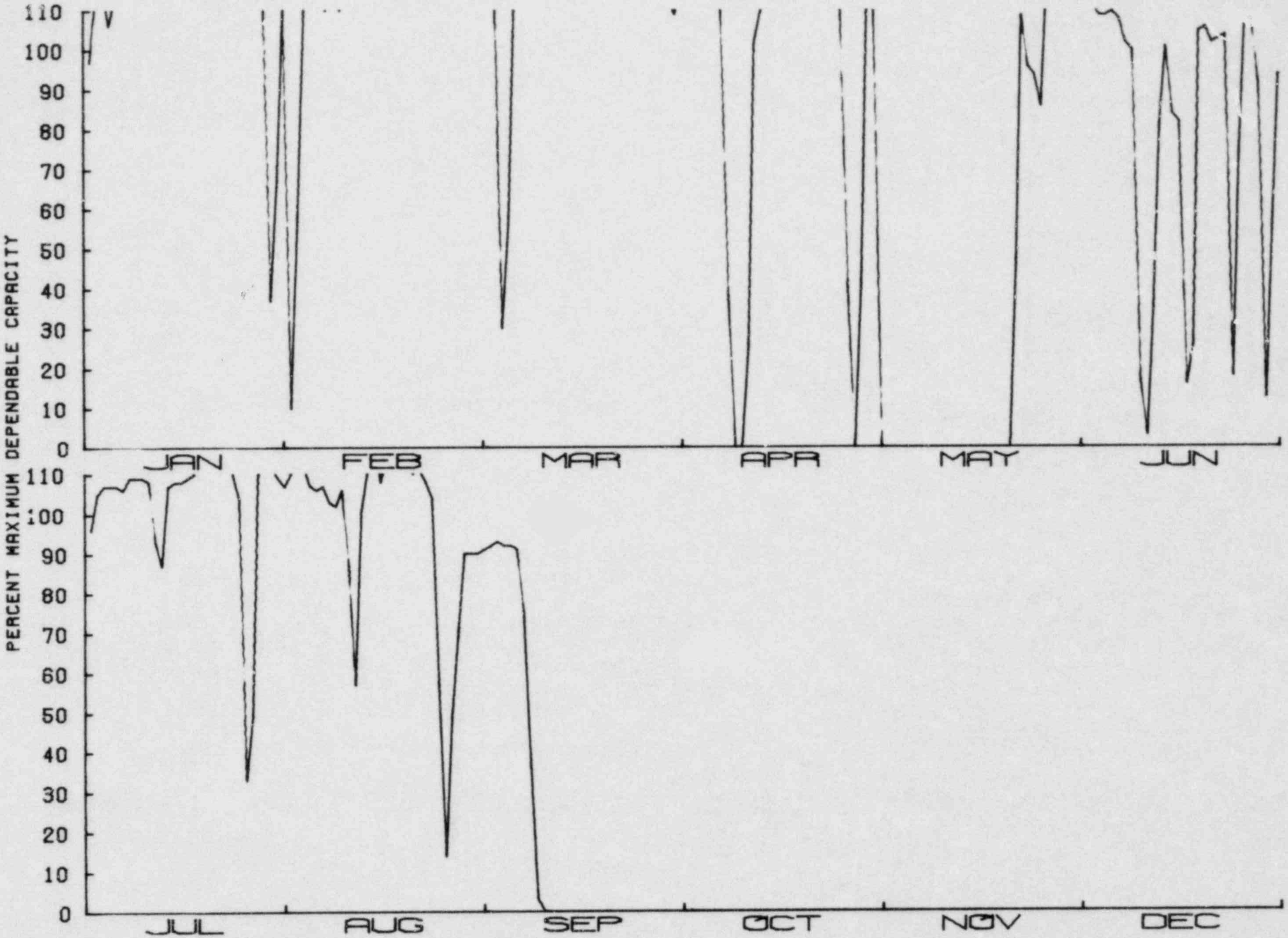
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/28	19	F	Feedwater regulating valve failed open	A	2	Steam and power conversion (HH)	Valves
2)	2/1	25	F	Operator inadvertently tripped a primary coolant pump	G	3	Reactor coolant (CB)	Instrumentation and controls
3)	3/3	22	F	Main feedpump trip due to control valve failure	A	3	Steam and power conversion (HH)	Valves
4)	4/7	67	F	Feedwater pump trip caused the reactor to trip on low s/g level	A	3	Steam and power conversion (HH)	Pumps
5)	4/25	42	F	Loss of generator load due to malfunction of voltage regulator	A	3	Steam and power conversion (HA)	Generator (main generator)
6a)	4/30	40*	F	Loss load condition due to voltage regulator malfunction	A	3	Steam and power conversion (HA)	Generator (main generator)

*Estimated.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6b)	4/30	463*	F	Outage was extended to resolve the inadequate piping restraints for two safety injection lines	A	4	Engineered safety features (SF)	Shock suppressors
7)	6/9	23	F	Condenser tube leak repairs	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
8)	6/16	15	F	Condenser tube leak repairs	A	2	Steam and power conversion (HC)	Heat exchangers (condenser)
9)	8/10	21	F	Feedwater pump tripped during turbine valve testing	A	2	Steam and power conversion (HH)	Pumps
10)	8/24	23	F	Loss of feedwater flow while cutting in the condensate demineralizers	A	2	Steam and power conversion (HH)	Pumps
11)	9/8	2755	S	Refueling	C	1	Reactor (RC)	Fuel elements

*Estimated.



DESIGN ELEC. RATING = 805 MAX. DEPEND. CAP. = 635 (100%) PALISADES

PEACH BOTTOM 2

I. Summary

Description

Location: Peach Bottom, Penn.
Docket No: 50-277
Reactor Type: BWR
Capacity (MWe-Net): 1,051
Commercial Operation: 7/5/74
Plant Age: 5.9 Years

Performance

Net Electrical Energy Generated (MWH):	8,574,430
Unit Availability Factor (%):	94.7
Unit Capacity Factor (%) (Using MDC):	93.1
Unit Capacity Factor (%) (Using Design MWE):	91.9

Outages

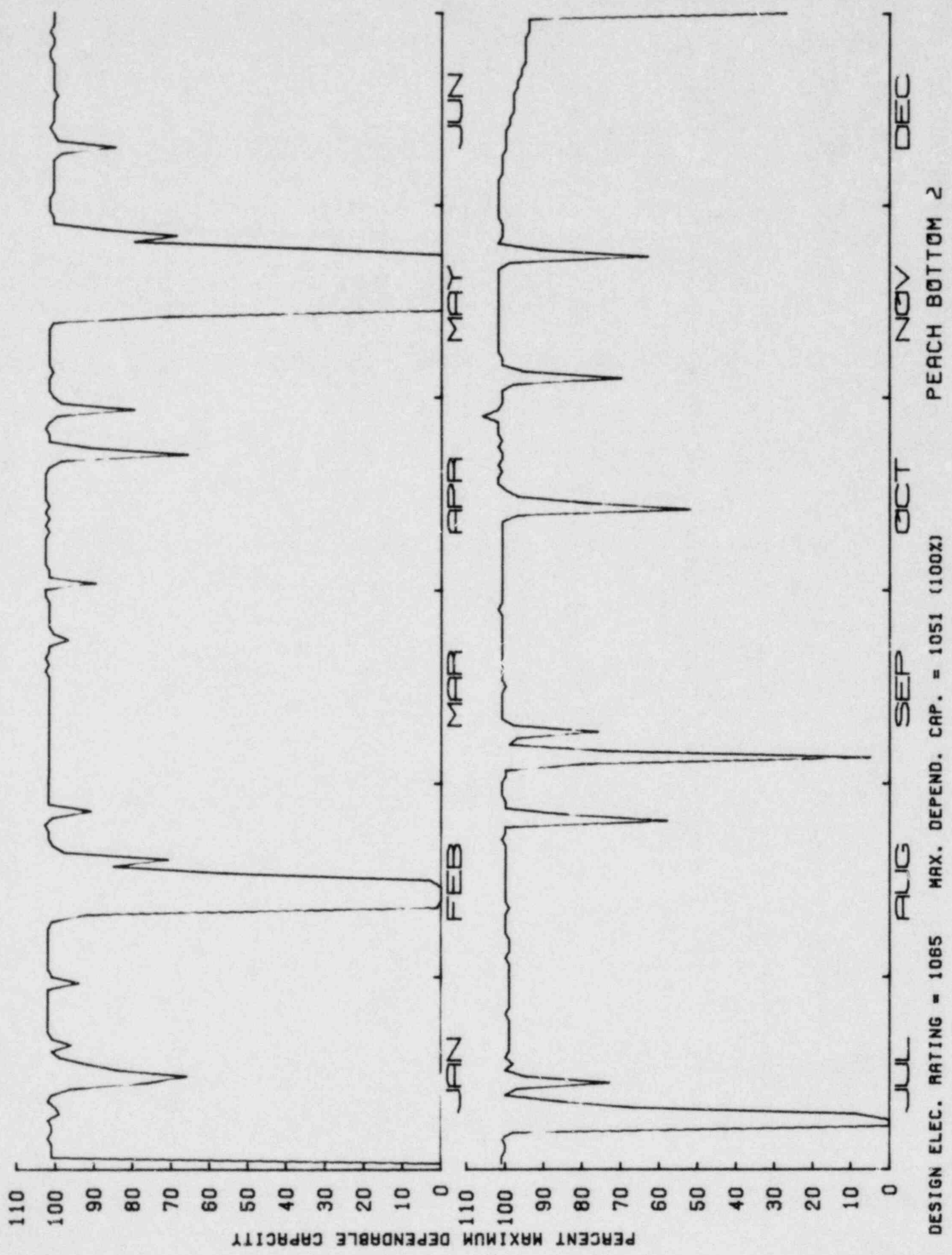
Total No.	6		
Forced	2		
Scheduled	4		
Total:	464	Hours,	5.3%
Forced	54	Hours,	0.6%
Scheduled	410	Hours,	4.7%

II. Highlights

Operation was routine throughout the year, as indicated by the 94.7% availability factor. There were 7 months in which power generation was uninterrupted.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/10	32	F	External pipe leak on reactor feedpump bypass line	A	1	Reactor coolant (CH)	Pipes, fittings
2)	2/10	83	S	Feedwater pump repair	B	1	Reactor coolant (CH)	Pumps
3)	5/13	251	S	Recombiner maintenance and feedwater repair	B	1	Radioactive waste management (MB)	Recombiners
4)	7/7	60	S	Repair full flow test valve on core spray "A" loop	B	1	Engineered safety features (SF)	Valves
5)	9/3	22	F	Mechanical turbine trip valve lock device failed to function properly during testing	A	3	Steam and power conversion (HA)	Turbines
6)	12/31	16	S	Repair core spray full-flow test valve	B	1	Engineered safety features (SF)	Valves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DESIGN ELEC. RATING = 1065 MAX. DEPEND. CAP. = 1051 (100%)
PEACH BOTTOM 2

PEACH BOTTOM 3

I. Summary

Description	Performance		Outages			
Location: Peach Bottom, Penn.	Net Electrical Energy Generated (MWH):	6,101,657	Total No.	13		
Docket No: 50-278	Unit Availability Factor (%):	74.2	Forced	8		
Reactor Type: BWR	Unit Capacity Factor (%) (Using MDC):	67.3	Scheduled	5		
Capacity (MWe-Net): 1,035	Unit Capacity Factor (%) (Using Design MWE):	65.4	Total:	2,257	Hours,	25.8%
Commercial Operation: 12/23/74			Forced	238	Hours,	2.7%
Plant Age: 5.3 Years			Scheduled	2,019	Hours,	23.1%

II. Highlights

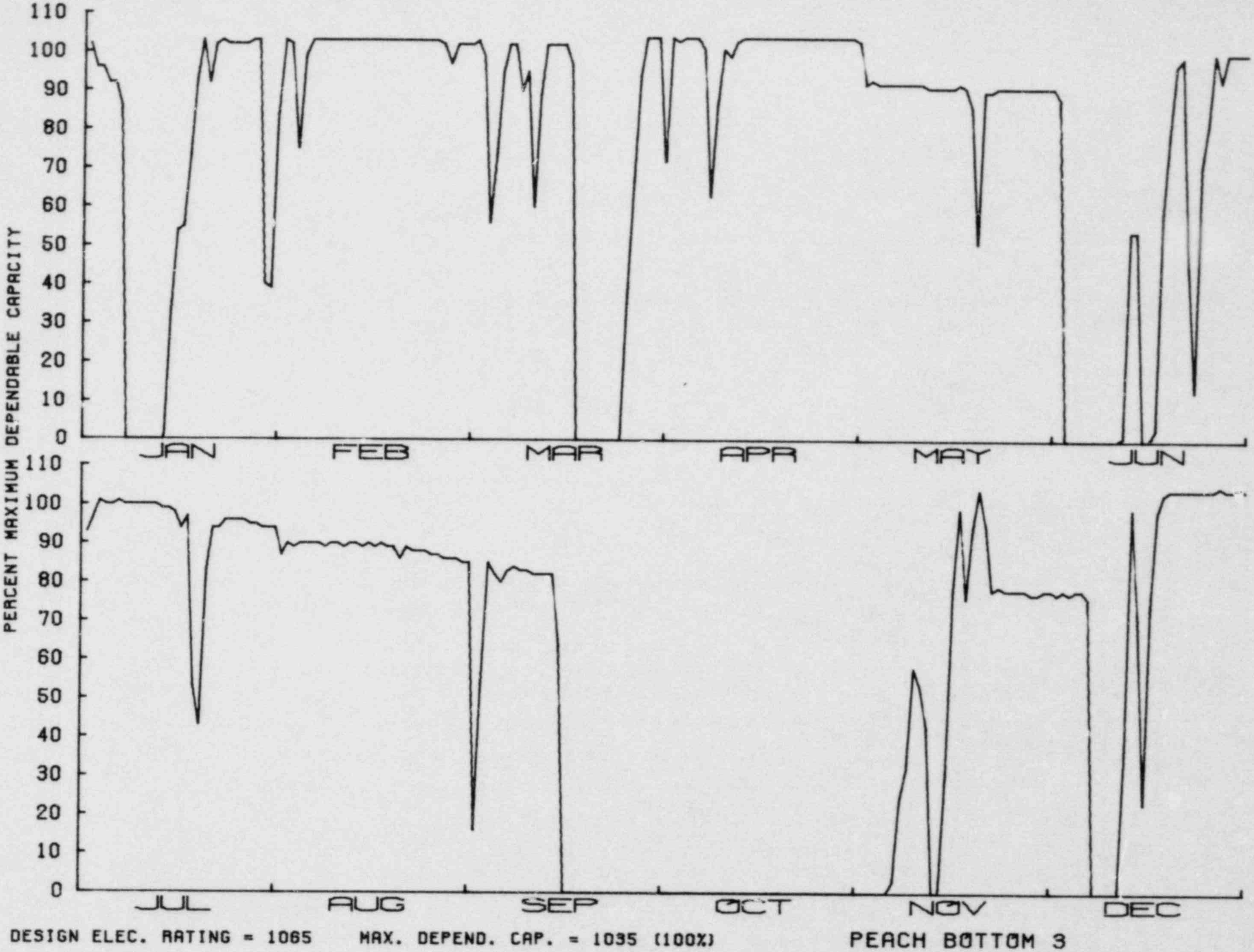
Operation was routine throughout the year, with 4 months in which power generation was uninterrupted. Refueling was accomplished between September 14 and November 6. Some problems were experienced with seismic supports on the control rod drive system 1-in. piping, and there were problems with the transverse in-core probe (TIP) shear valves.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/6	169	S	Feedwater heater repair	B	2	Reactor coolant (CH)	Heat exchangers
2)	1/26	20	F	Feedwater control system failure	A	3	Reactor coolant (CH)	Instrumentation and controls
3)	3/17	216	S	Feedwater heater repair	B	1	Reactor coolant (CH)	Heat exchangers
4)	3/17	6	F	Main generator voltage regulator malfunction	A	1	Steam and power conversion (HA)	Generators (main generator)
5)	6/2	231	S	Recombiner condenser maintenance	B	1	Radioactive waste management (MB)	Recombiners
6)	6/13	75	F	"L" relief valve failed open	A	2	Reactor coolant (CC)	Valves
7)	6/21	25	F	3A recombiner mechanical compressor failure	A	2	Radioactive waste management (MB)	Blowers

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8)	7/18	12	F	Circuit breaker protective relay at 500 kV distribution system caused loss of load	A	3	Electric power (EB)	Relays
9)	9/1	19	F	Loss of vacuum due to air ejector discharge valve relay failure	A	3	Steam and power conversion (HC)	Relays
10)	9/14	1266	S	Refueling	C	1	Reactor (RC)	Fuel elements
11)	11/11	63	F	During testing, turbine stop valve closed momentarily	A	3	Steam and power conversion (HA)	Valves
12)	12/7	137	S	Repair "A" main steam line isolation valve	B	1	Reactor coolant (CD)	Valve operators
13)	12/14	18	F	Temporary loss of power to instrumentation	A	2	Electric power (EB)	Instrumentation and controls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DESIGN ELEC. RATING = 1065
MAX. DEPEND. CAP. = 1035 (100%)
PEACH BOTTOM 3

PILGRIM 1

I. Summary

Description	Performance		Outages			
Location: Plymouth, Mass.	Net Electrical Energy Generated (MWH):	4,844,559	Total No.	11		
Docket No: 50-293			Forced	10		
Reactor Type: BWR	Unit Availability Factor (%):	89.4	Scheduled	1		
Capacity (MWe-Net): 670			Total:	931	Hours,	10.6%
Commercial Operation: 12/72	Unit Capacity Factor (%) (Using MDC):	82.5	Forced	891	Hours,	10.2%
Plant Age: 7.5 Years			Scheduled	40	Hours,	0.4%
	Unit Capacity Factor (%) (Using Design MWE):	84.4				

II. Highlights

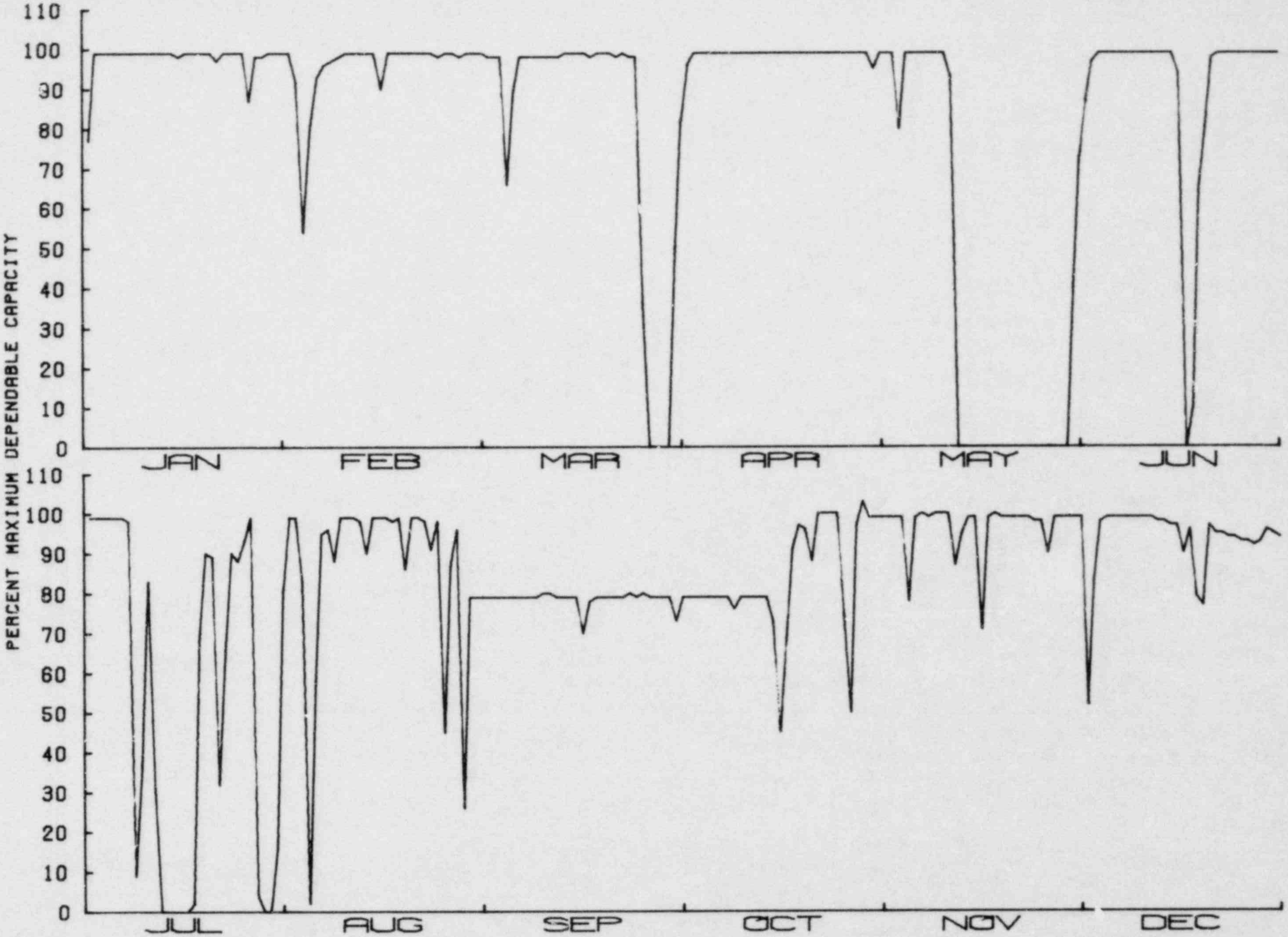
Power generation was uninterrupted for 6 months of the year. In May, the unit was shut down to inspect and modify snubbers on safety-related piping. At the end of the year, the unit was in coastdown for refueling, which was to begin in January.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/25	111	F	Water level control fluctuation	A	3	Reactor coolant (CH)	Instrumentation and controls
2)	5/12	437	F	Seismic inspection and modification of scrubbers	D	2	Engineered safety features (SF)	Shock suppressors
3)	6/16	40	S	Replace weeping safety relief valve	B	2	Reactor coolant (CC)	Valves
4)	7/8	28	F	Repair leak in hydraulic system on MSIV	A	2	Reactor coolant (CD)	Valves
5)	7/11	154	F	Leak in CRD return line weld	A	2	Reactor (RB)	Control rod drives
6)	7/21	12	F	Loss of vacuum during condenser backwash	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
7)	7/27	85	F	Loss of off-site power	A	3	Electric power (EA)	Electrical conductors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8)	8/4	26	F	Repair leak at 4th point heater	A	1	Reactor coolant (CH)	Heat exchangers
9)	8/28	13	F	Loss of off-site power due to lightning	H	9	Electric power (EA)	Electrical conductors
10)	10/15	13	F	Low suction pressure to feed pumps	A	3	Steam and power conversion (HH)	Pumps
11)	10/25	12	F	Mechanical shock to pressure switch on rack in reactor building during surveillance	G	3	Instrumentation and controls (IA)	Instrumentation and controls



DESIGN ELEC. RATING = 655 MAX. DEPEND. CAP. = 670 (100%) PILGRIM 1

POINT BEACH 1

I. Summary

Description		Performance		Outages		
Location: Two Creeks, Wisconsin		Net Electrical Energy Generated (MWH):	3,055,424	Total No.	7	
Docket No: 50-266		Unit Availability Factor (%):	76.2*	Forced	1	
Reactor Type: PWR		Unit Capacity Factor (%) (Using MDC):	70.5	Scheduled	6	
Capacity (MWe-Net): 495		Unit Capacity Factor (%) (Using Design MWE):	70.2	Total:	2,302 Hours,	26.3%*
Commercial Operation: 12/21/70				Forced	269 Hours,	3.1%
Plant Age: 9.2 Years				Scheduled	2,033 Hours,	23.2%

II. Highlights

Operation was routine all year, with refueling accomplished in October and November. Operation was uninterrupted for 6 months, with 4 months (April through July) being sequential.

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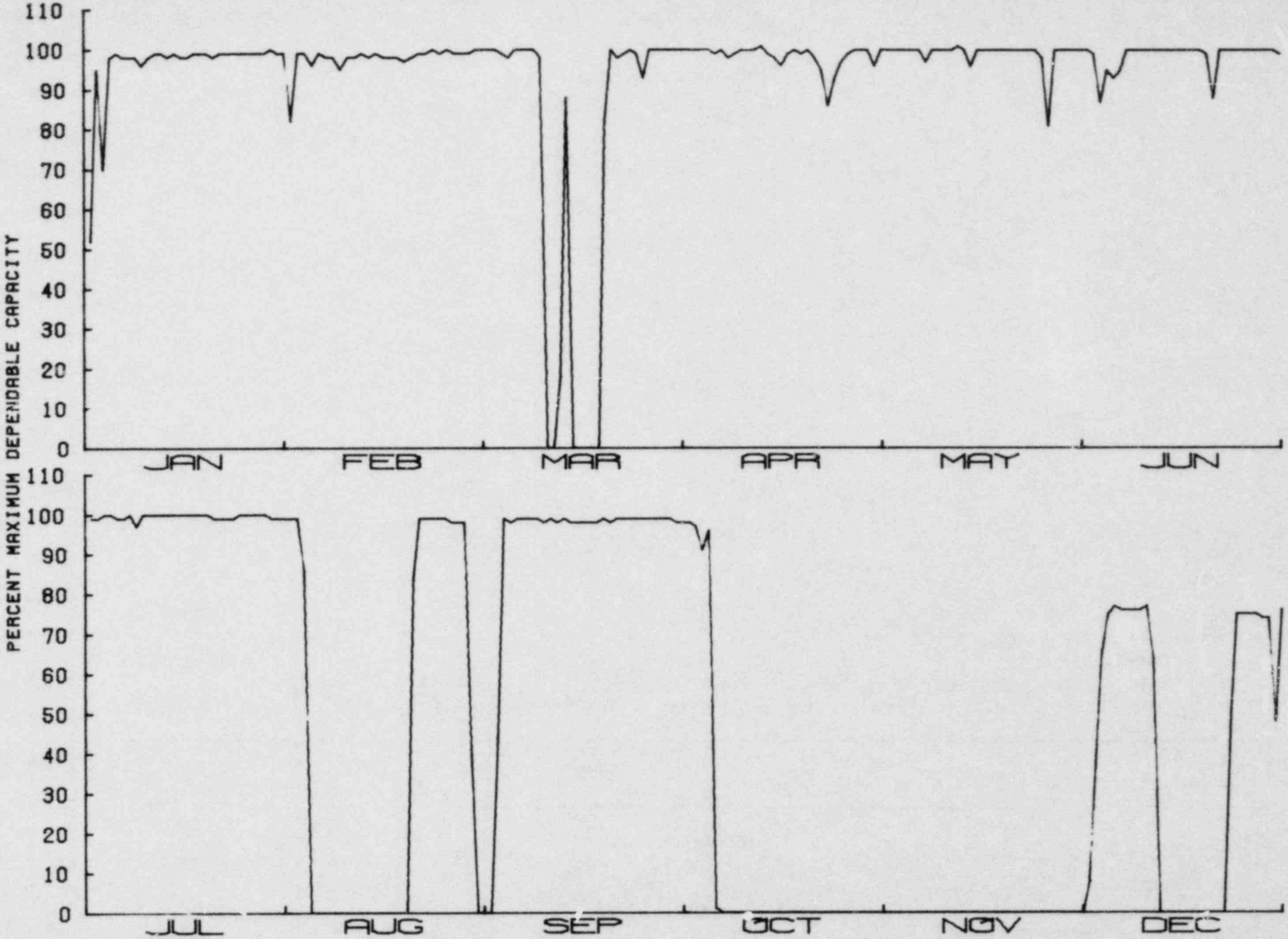
*Includes 219 h of reserve shutdown equal to 2.5% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/10	63	S	Repair moisture separator reheater tube leakage	B	1	Steam and power conversion (HB)	Heat exchangers (MSR)
2)	3/14	119	S	Repair of steam generator primary to secondary leakage	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
3)	8/3	26	S	The 3-in. auxiliary feedline-to-main-feedline branch connections were reinforced to meet code requirements	B	1	Steam and power conversion (HH)	Pipes, fittings
4)	8/5	355	S	Primary-to-secondary leak was detected in "A" steam generator due to deep crevice defects	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
5)	8/29	94	S	Leaking tube in "A" steam generator	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
6)	10/5	1376	S	Refueling	C	1	Reactor (RC)	Fuel elements

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	12/11	269	F	Correct primary-to-secondary leakage	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)



DESIGN ELEC. RATING = 497 MAX. DEPEND. CAP. = 495 (100%) POINT BEACH 1

POINT BEACH 2

I. Summary

Description

Location: Two Creeks, Wisconsin
Docket No: 50-301
Reactor Type: PWR
Capacity (MWe-Net): 495
Commercial Operation: 10/1/72
Plant Age: 7.4 Years

Performance

Net Electrical Energy Generated (MWH):	3,707,450
Unit Availability Factor (%):	88.5*
Unit Capacity Factor (%) (Using MDC):	85.5
Unit Capacity Factor (%) (Using Design MWE):	85.2

Outages

Total No.	6	
Forced	1	
Scheduled	5	
Total:	1,027 Hours,	11.7%*
Forced	5 Hours,	0.1%
Scheduled	1,022 Hours,	11.6%

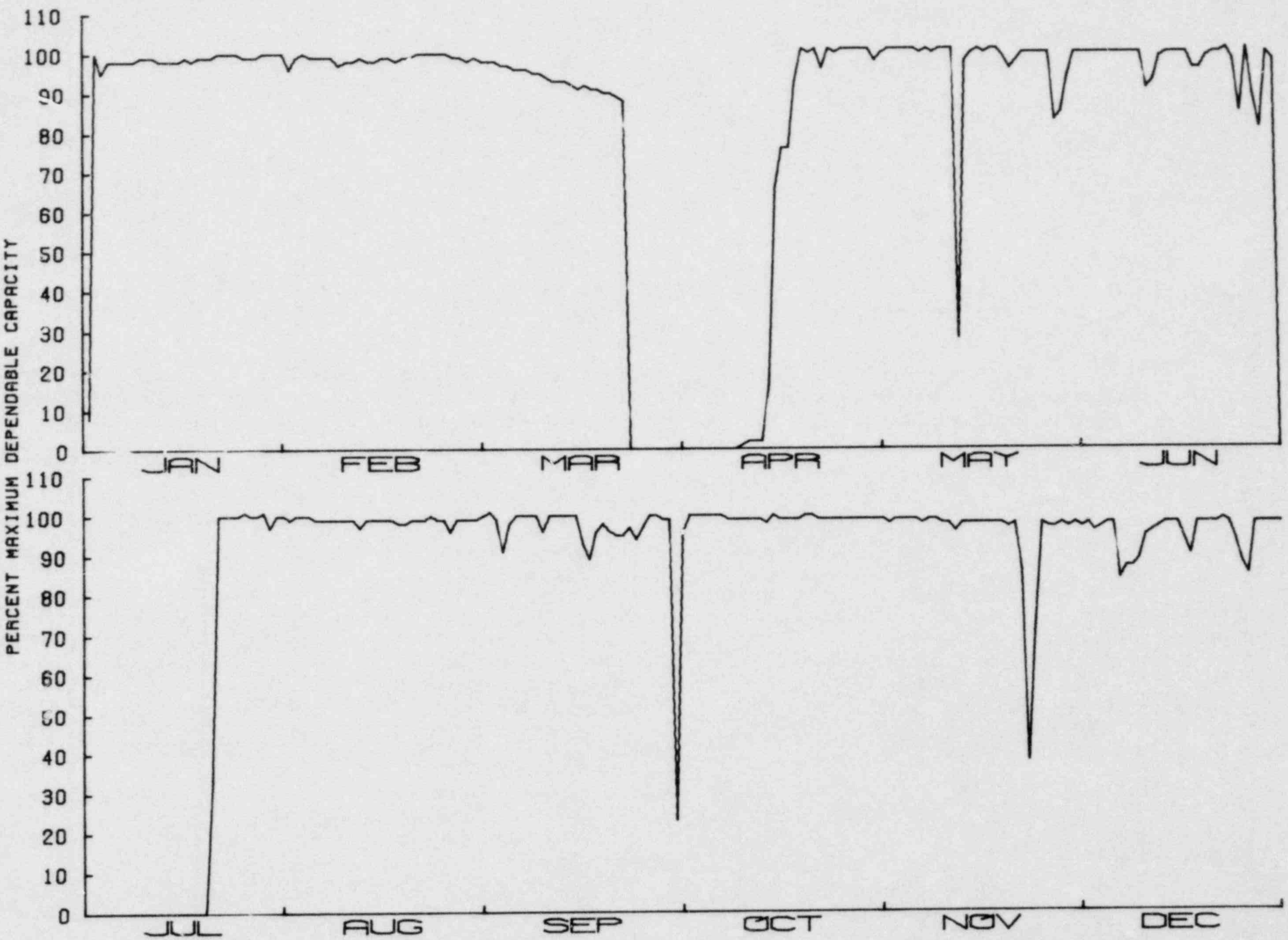
II. Highlights

Refueling was accomplished between March 23 and April 13. The unit was shut down for 20 days for inspection and repair of auxiliary feedwater piping in response to IE Bulletin 79-13. There were 5 months of uninterrupted operation.

*Includes 18.6 h of reserve shutdown equal to 0.2% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/23	506	S	Refueling	C	1	Reactor (RC)	Fuel elements
2)	4/13	1	S	Routine turbine overspeed testing	B	1	Steam and power conversion (HA)	Turbines
3)	5/12	11	S	Safeguards logic modification and primary system circulation test	F	9	Engineered safety features (SF)	Instrumentation and controls
4)	6/30	490	S	Feedwater nozzle volumetric examinations	B	1	Steam and power conversion (HH)	Pipes, fittings
5)	9/29	14	S	Repair "D" moisture separator drain valve and loosen packing on "B" MSIV	B	1	Steam and power conversion (HB)	Valves
6)	11/21	5	F	Failed capacitor in the "A" battery inverter	A	3	Electric power (ED)	Generators (Inverter)



DESIGN ELEC. RATING = 497 MAX. DEPEND. CAP. = 495 (100%) POINT BEACH 2

PRAIRIE ISLAND 1

I. Summary

Description

Location: Goodhue, Minnesota
Docket No: 50-282
Reactor Type: PWR
Capacity (MWe-Net): 503
Commercial Operation: 12/16/73
Plant Age: 6.1 Years

Performance

Net Electrical Energy Generated (MWH):	2,910,820
Unit Availability Factor (%):	73.1
Unit Capacity Factor (%) (Using MDC):	66.1
Unit Capacity Factor (%) (Using Design MWE):	62.7

Outages

Total No.	13	
Forced	7	
Scheduled	6	
Total:	2,354 Hours,	26.9%
Forced	1,520 Hours,	17.4%
Scheduled	834 Hours,	9.5%

II. Highlights

Operation during the first 3 months of the year was uninterrupted. Refueling was accomplished in April. Damage to the high-pressure turbine necessitated a shutdown in July for repairs, and a tube rupture occurred in a steam generator in October, requiring another shutdown.

DETAILS OF PLANT OUTAGES

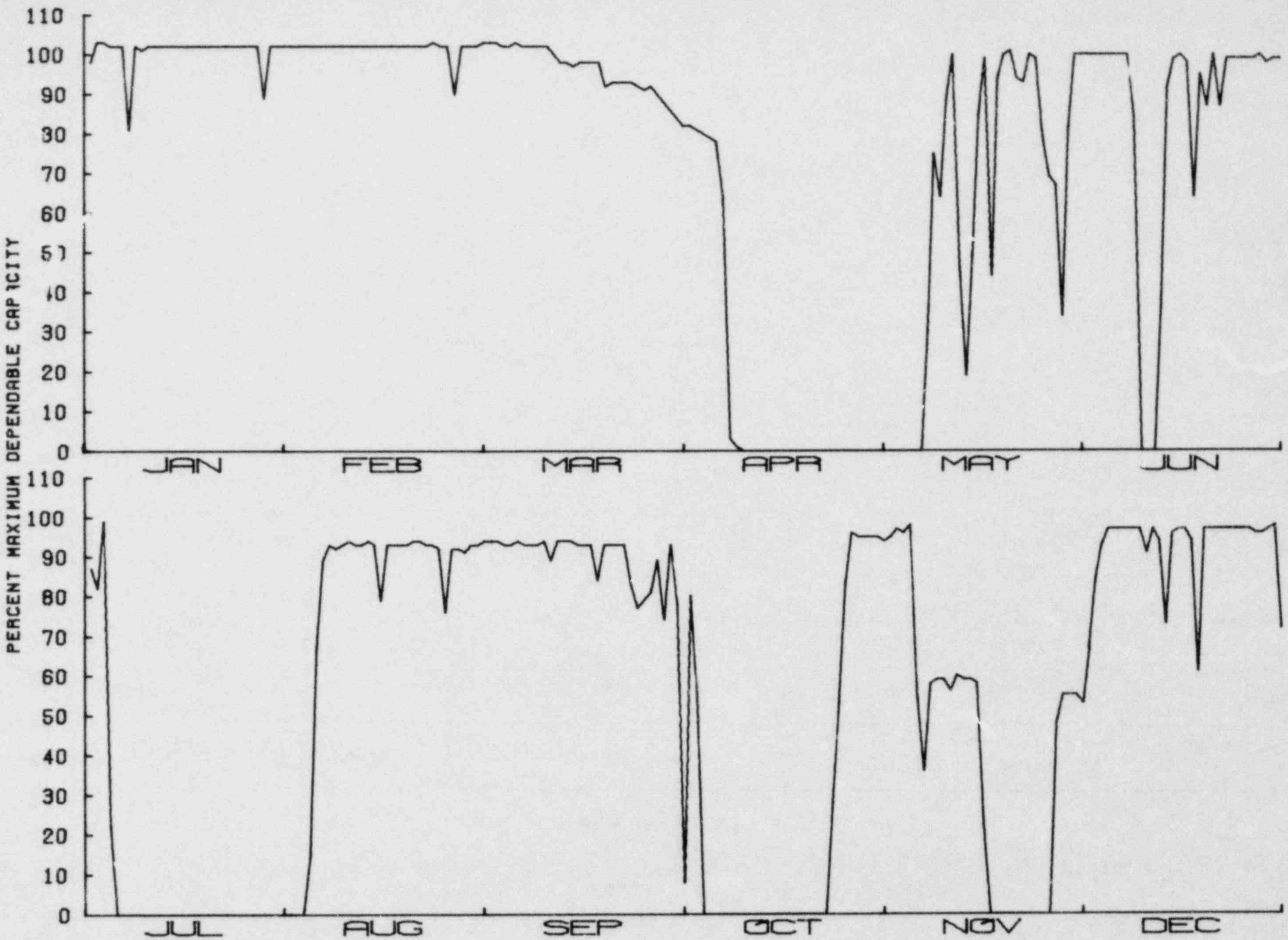
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	4/6	724	S	Refueling	C	1	Reactor (RC)	Fuel elements
2)	5/12	6	F	Malfunction of loop "A" feedwater regulating valve	A	3	Steam and power conversion (HH)	Valves
3)	5/13	12	F	Repair loop "A" feedwater regulating valve	A	1	Steam and power conversion (HH)	Valves
4)	5/17	10	F	Repair loop "A" feedwater regulating valve	A	1	Steam and power conversion (HH)	Valves
5)	6/8	79	S	Feedwater nozzle inspection and repairs to turbo-generator hydrogen seal oil system	B	2	Steam and power conversion (HA)	Generators (main generator)
6)	7/4	731	F	High turbine vibration due to pressure turbine damage	A	1	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	8/4	5	S	Turbine overspeed test	B	2	Steam and power conversion (HA)	Turbines
8)	9/30	16	S	Repair steam leaks and replacement of source range channel detector	B	3	Steam and power conversion (HB)	Pipes, fitting
9)	10/2	497	F	Rupture of tube in No. 11 steam generator	A	3	Steam and power conversion (HB)	Heat exchanger (steam generator)
10)	11/5	16	F	Undervoltage on Bus 11	A	3	Electric power (ED)	Electrical conductors
11)	11/15	248	F	Undervoltage on Bus 12	A	3	Electric power (ED)	Electrical conductors
12)	12/18	7	S	Repair leak in a feedwater sample tap	B	1	Steam and power conversion (HH)	Pipes, fittings

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	12/31	3	S	Repair packing leak on an RID bypass vent valve	B	1	Instrumentation and controls (IA)	Valves



DESIGN ELEC. RATING = 530 MAX. DEPEND. CAP. = 503 (100%) PRAIRIE ISLAND 1

PRAIRIE ISLAND 2

I. Summary

Description

Location: Goodhue, Minnesota
Docket No: 50-306
Reactor Type: PWR
Capacity (MWe-Net): 500
Commercial Operation: 12/21/74
Plant Age: 5.0 Years

Performance

Net Electrical Energy Generated (MWH):	4,193,044
Unit Availability Factor (%):	98.9
Unit Capacity Factor (%) (Using MDC):	95.7
Unit Capacity Factor (%) (Using Design MWE):	90.3

Outages

Total No.	7		
Forced	6		
Scheduled	1		
Total:	94	Hours,	1.1%
Forced	50	Hours,	0.6%
Scheduled	44	Hours,	0.5%

II. Highlights

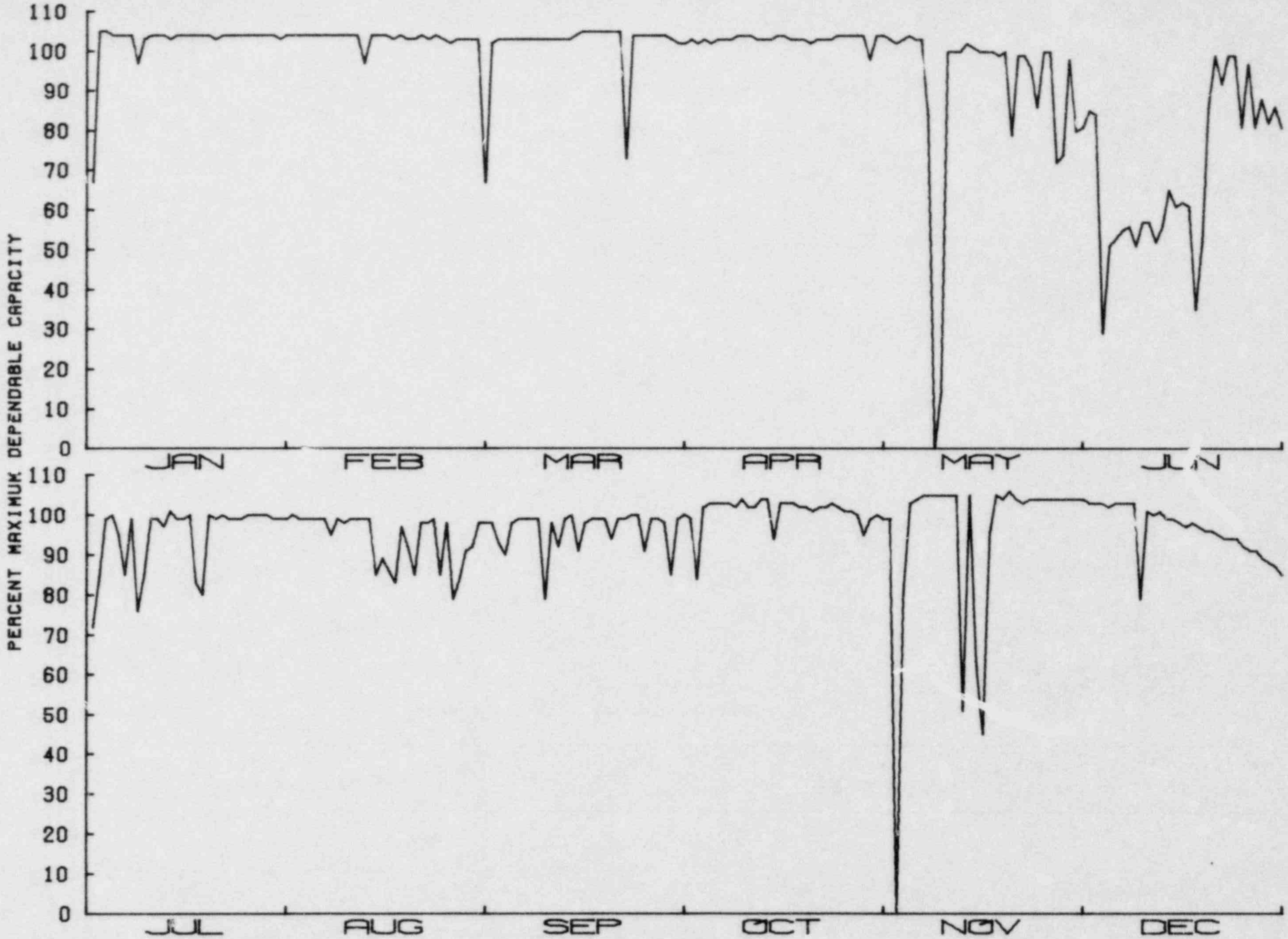
Although there were seven outages during the year, the longest lasted only 44 h, and an availability factor of 98.9% was obtained. There were 6 months of uninterrupted operation; in August, September, and October, operation was continuous.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/28	6	F	Procedural error during a safe-guards surveillance test	G	3	Engineered safety features (SF)	Instrumentation and controls
2)	3/22	4	F	Turbine control valves drifted closed on loss of E-H pressure	G	2	Steam and power conversion (HA)	Mechnical function units
3)	5/7	44	S	Modify SI actuation logic	H	1	Engineered safety features (SF)	Instrumentation and controls
4)	6/3	5	F	Bus undervoltage due to failure of No. 22 feedwater pump motor	A	3	Steam and power conversion (HH)	Motor
5)	7/17	6	F	Trip on lo-lo steam generator level when I&C technician erred while doing maintenance on level control channel	G	3	Instrumentation and controls (IA)	Instrumentation and controls
6)	11/1	26	F	Steam generator high level	A	3	Steam and power conversion (HH)	Valves

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	11/14	3	F	Personnel error in valving out a pressurizer level transmitter	G	3	Instrumentation and controls (IA)	Instrumentation and controls



DESIGN ELEC. RATING = 530 MAX. DEPEND. CAP. = 500 (100%) PRAIRIE ISLAND 2

QUAD CITIES 1

I. Summary

Description

Location: Cordova, Illinois
Docket No: 50-254
Reactor Type: BWR
Capacity (MWe-Net): 769
Commercial Operation: 2/18/73
Plant Age: 7.7 Years

Performance

Net Electrical Energy Generated (MWH):	4,782,963
Unit Availability Factor (%):	81.3*
Unit Capacity Factor (%) (Using MDC):	71.0
Unit Capacity Factor (%) (Using Design MWE):	69.2

Outages

Total No.	12		
Forced	9		
Scheduled	3		
Total:	1,650	Hours,	18.8%*
Forced	432	Hours,	4.9%
Scheduled	1,218	Hours,	13.9%

II. Highlights

Refueling was accomplished between January 18 and February 28. A shutdown in September was devoted to anchor bolt and pipe restraint inspection in accordance with IE Bulletins 79-02 and 79-14. There were 4 months of uninterrupted operation, with June and July being the only sequential months in which power generation was continuous.

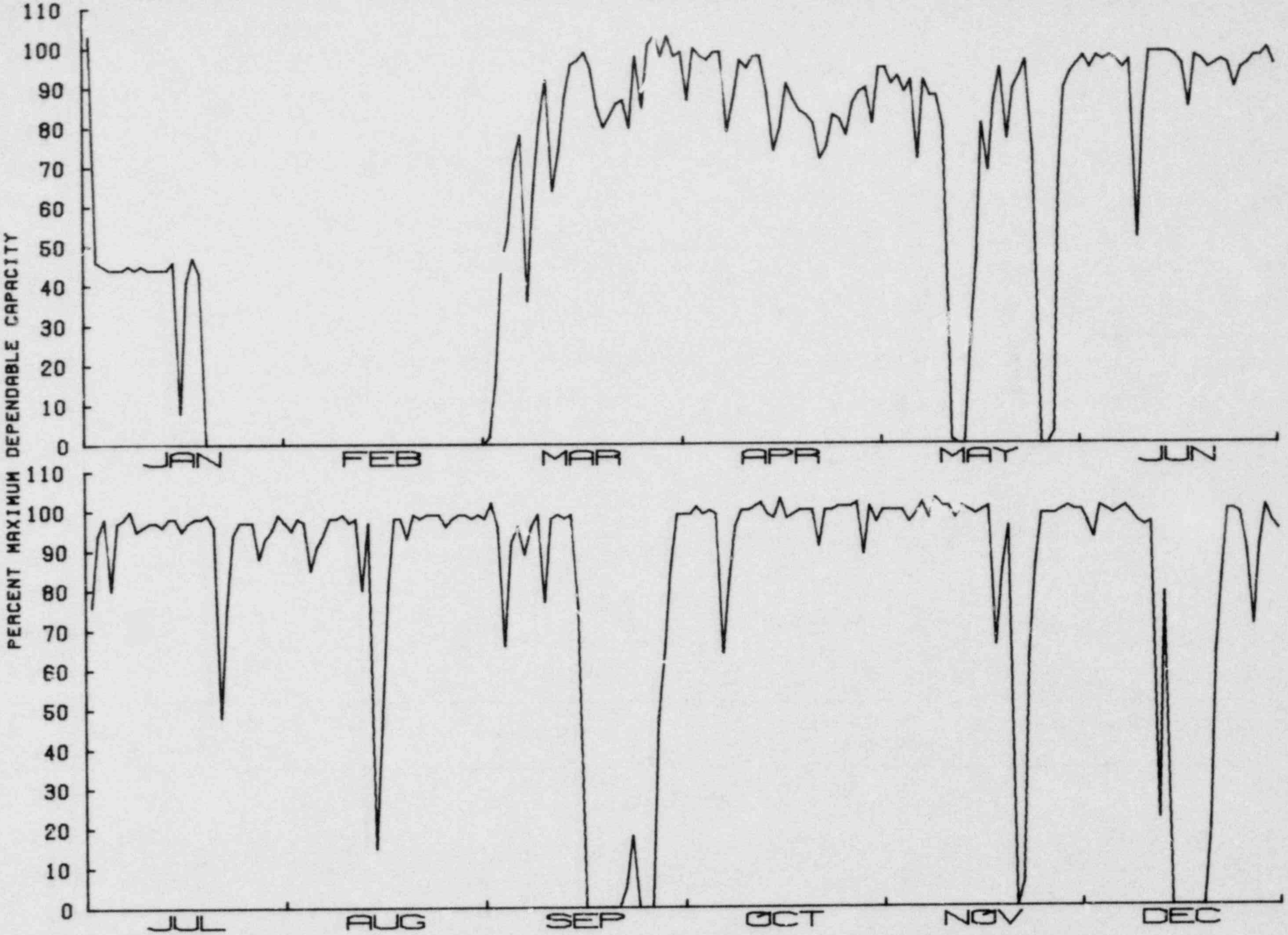
*Includes 19.8 h of reserve shutdown equal to 0.2% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/14	5	F	Lack of nitrogen to inert the drywell	F	1	Engineered safety features (SE)	Pressure vessels (containment)
2)	1/18	988	S	Refueling	C	2	Reactor (RC)	Fuel elements
3)	2/28	36	F	1-203-3E electromatic valve replacement	A	1	Engineered safety features (SF)	Valves
4)	3/7	9	F	A workman closed a door on the unit 1 main transformer causing a transformer over pressure relay to trip	G	3	Electric power (EB)	Relays
5)	5/11	77	S	Maintenance	B	1	(not given)	(not given)
6)	5/24	67	F	Plug condenser tubes	A	1	Steam and power conversion (HC)	Heat exchanger (condenser)
7)	8/14	23	F	Loss of main condenser vacuum	A	3	Steam and power conversion (HC)	Heat exchanger (condenser)

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8)	9/14	163	F	Anchor bolt and piping restraint inspections per NRC bulletins	D	1	Engineered safety features (SF)	Shock suppressors
9)	9/22	78	F	Repair of 1A moisture separator drain tank vent line	A	1	Steam and power conversion (HB)	Pipes, fittings
10)	11/20	38	F	Make repairs to the drain line of the reactor feedpump common discharge header	A	1	Reactor coolant (CH)	Pipes, fittings
11)	12/11	13	F	Blown potential transformer fuses	A	3	Electric power (EB)	Circuit closers/ interruptors
12)	12/14	153	S	Check relief valve position indication manual reset and core damage assessment	D	1	Reactor coolant (CC)	Valves



DESIGN ELEC. RATING = 789 MAX. DEPEND. CAP. = 769 (100%) QUAD CITIES 1

QUAD CITIES 2

I. Summary

Description

Location: Cordova, Illinois
Docket No: 50-265
Reactor Type: BWR
Capacity (MWe-Net): 769
Commercial Operation: 3/10/73
Plant Age: 7.6 Years

Performance

Net Electrical Energy Generated (MWH):	3,981,065
Unit Availability Factor (%):	87.8
Unit Capacity Factor (%) (Using MDC):	59.1
Unit Capacity Factor (%) (Using Design MWE):	57.6

Outages

Total No.	8		
Forced	5		
Scheduled	3		
Total:	1,072	Hours,	12.2%
Forced	70	Hours,	0.8%
Scheduled	1,002	Hours,	11.4%

II. Highlights

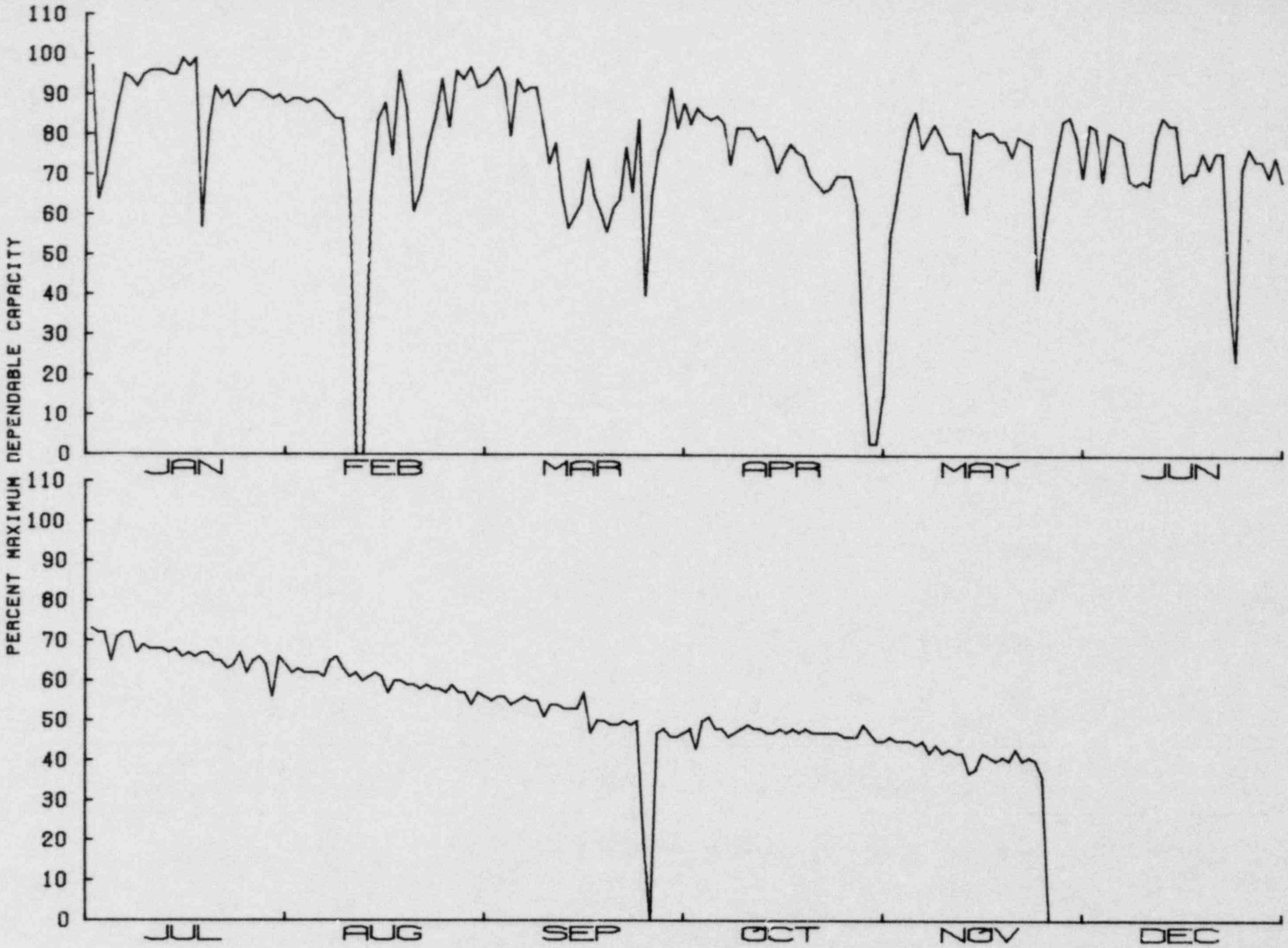
There were 4 months in which power generation was uninterrupted. At the end of the year, the unit was in a refueling outage that began November 25.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/19	9	F	Personnel error during blowdown calibration test	G	3	Engineered safety features (SF)	Valves
2)	2/9	46	S	Battery testing	B	1	Electric power (EC)	Batteries and chargers
3)	4/27	68	S	Maintenance	B	1	(not given)	(not given)
4)	5/24	8	F	False low reactor water level signal	G	3	Instrumentation and controls (IA)	Instrumentation and controls
5)	6/22	8	F	Spurious condenser low vacuum signal	A	3	Instrumentation and controls (IA)	Relays
6)	6/23	10	F	Reactor low water level	A	3	Reactor coolant (CH)	Valves
7)	9/24	35	F	False high reactor pressure signal during instrument calibration	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8)	11/25	888	S	Refueling	C	1	Reactor (RC)	Fuel elements



DESIGN ELEC. RATING = 789 MAX. DEPEND. CAP. = 769 (100%) QUAD CITIES 2

RANCHO SECO

I. Summary

Description

Location: Sacramento, California
Docket No: 50-312
Reactor Type: PWR
Capacity (MWe-Net): 873
Commercial Operation: 4/17/75
Plant Age: 5.2 Years

Performance

Net Electrical Energy Generated (MWH):	5,711,999
Unit Availability Factor (%):	91.1*
Unit Capacity Factor (%) (Using MDC):	74.7
Unit Capacity Factor (%) (Using Design MWE):	71.0

Outages

Total No.	18	
Forced	14	
Scheduled	4	
Total:	1,982 Hours,	22.6%*
Forced	401 Hours,	4.6%
Scheduled	1,581 Hours,	18.0%

II. Highlights

An extended outage was in effect from April 28 to July 5 for making modifications (TMI-2 related) to increase capability and reliability to respond to various transient events initiated in the feedwater system.

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*Includes 1,199.4 h of reserve shutdown equal to 13.7% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/2	7	F	Loss of "A" inverter	A	3	Electric power (ED)	Generators (Inverter)
2)	1/5	15	F	Inadvertently opened a control rod drive breaker	G	3	Reactor (RB)	Circuit closers/ interrupters
3)	1/20	9	S	Maintenance to torque feedwater nozzle on OTSG	B	1	Steam and power conversion (HH)	Pipes, fittings
4)	2/25	13	S	Shutdown to facilitate adding oil to RCP "C" motor upper bearing	B	1	Reactor coolant (CB)	Motors
5)	2/25	57	F	Repair main generator seal oil system	A	1	Steam and power conversion (HA)	Generators (main generator)
6)	4/22	16	F	Loss of "A" inverter	A	3	Electric power (ED)	Generators (Inverter)

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	4/28	1536	S	Shutdown for reevaluation of safety systems as a result of the TMI-2 accident and to provide turbine trip and loss of feedwater trip	D	1	Steam and power conversion (HH)	Instrumentation and controls
8)	7/1	98	F	Repaired weld on auxiliary feedwater line	A	3	Steam and power conversion (HH)	Pipes, fittings
9)	7/12	5	F	Pressure transmitter malfunction which was equated to turbine overspeed	A	3	Instrumentation and controls (IA)	Instrumentation and controls
10)	7/20	54	F	Inspection and modification of pipe supports	D	1	Engineered safety features (SF)	Shock suppressors
11)	9/12	8	F	Turbine overspeed protection control transmitter failure	A	3	Steam and power conversion (HA)	Instrumentation and controls
12)	9/13	8	F	Power/flow/imbalance on reactor protection system	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	10/10	70	F	Completion of pipe support work on the nuclear service cooling water system	F	1	Auxiliary water (WA)	Shock suppressors
14)	11/17	23	S	Inspect snubbers and initiate under/over voltage protection scheme for vital buses	B	1	Engineered safety features (SF)	Shock suppressors
15)	11/18	2	F	"B" reheater safety valve lifted and would not reseat	A	1	Steam and power conversion (HB)	Valves
16)	11/18	2	F	"B" reheater safety valve lifted and would not reseat	A	1	Steam and power conversion (HB)	Valves
17)	11/18	3	F	"B" reheater safety valve lifted and would not reseat	A	1	Steam and power conversion (HB)	Valves
18)	11/26	56	F	Repair weld leak at intersection of drain valve line and HPI header	A	1	Engineered safety features (SF)	Pipes, fittings

DESIGN ELEC. RATING = 918 MAX. DEPEND. CAP. = 873 (100%) RANCHO SECO 1

ROBINSON 2

I. Summary

Description

Location: Hartsville, S.C.
Docket No: 50-261
Reactor Type: PWR
Capacity (MWe-Net): 665
Commercial Operation: 3/7/71
Plant Age: 9.3 Years

Performance

Net Electrical Energy Generated (MWH):	4,005,007
Unit Availability Factor (%):	70.8*
Unit Capacity Factor (%) (Using MDC):	68.8
Unit Capacity Factor (%) (Using Design MWE):	65.3

Outages

Total No.	16		
Forced	13		
Scheduled	3		
Total:	2,584	Hours,	29.5%*
Forced	272	Hours,	3.1%
Scheduled	2,312	Hours,	26.4%

II. Highlights

Operation was routine throughout the year, with refueling accomplished in May and June. On June 29, the licensed thermal power limit was increased from 2200 to 2300 MW(t).

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*Includes 23.2 h of reserve shutdown equal to 0.3% availability.

DETAILS OF PLANT OUTAGES

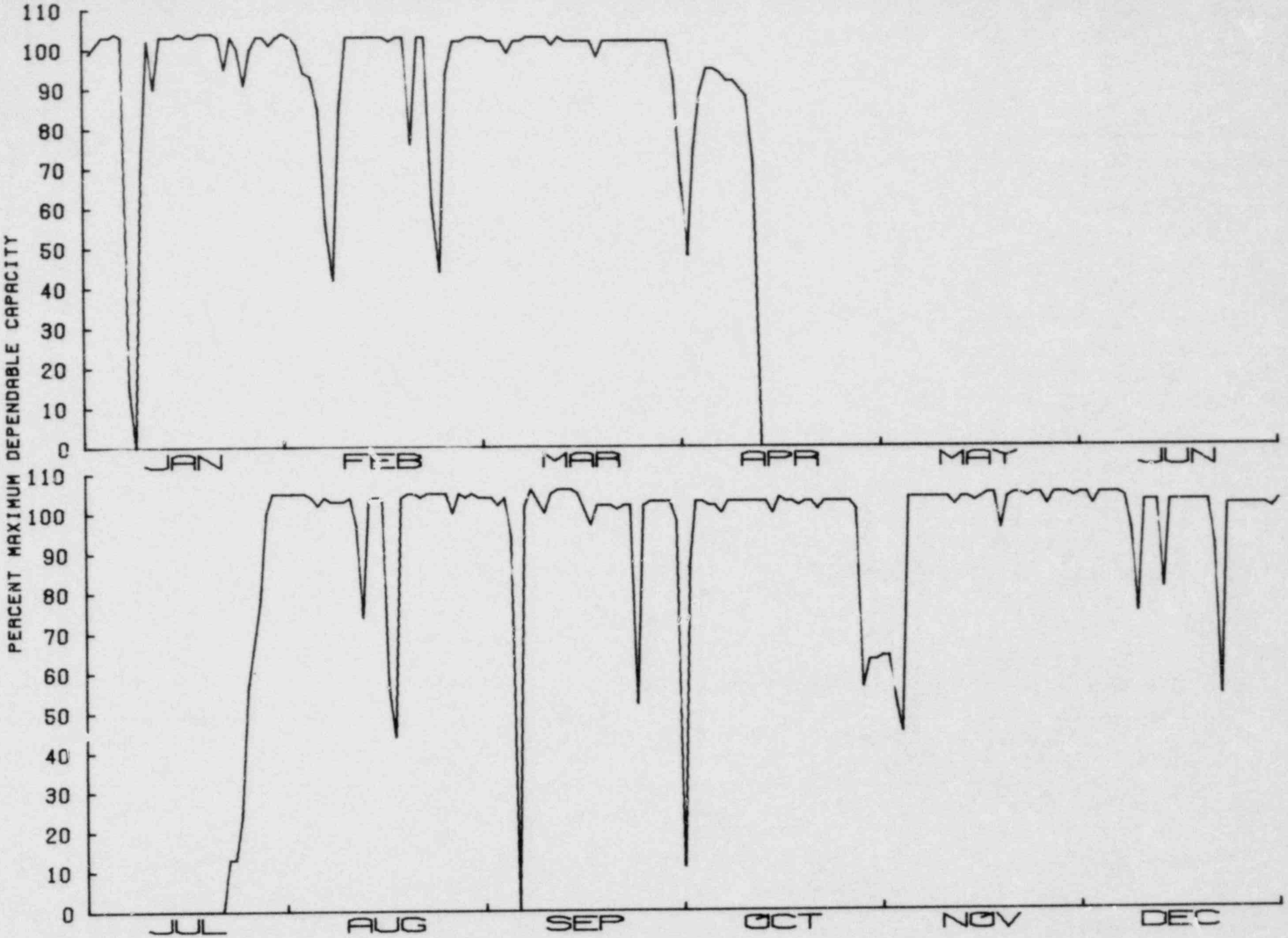
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/6	38	F	"C" steam generator hi level	A	3	Steam and power conversion (HH)	Instrumentation and controls
2)	1/7	5	F	"B" steam generator hi level	A	3	Steam and power conversion (HH)	Instrumentation and controls
3)	2/6	11	F	Safety injection on high containment pressure indication	A	3	Instrumentation and controls (IB)	Instrumentation and controls
4)	2/21	3	F	Reactor tripped while performing surveillance tests on steam generator controls	A	3	Steam and power conversion (HH)	Instrumentation and controls
5)	4/11	147	F	Repair steam generator tube leak	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
6)	4/18	2277	S	Refueling	C	1	Reactor (RC)	Fuel elements

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	7/22	7	F	Turbine balance	A	1	Steam and power conversion (HA)	Turbines
8)	7/23	7	F	Electro-hydraulic oil leak	A	3	Steam and power conversion (HA)	Pipes, fittings
9)	8/16	9	F	Contract personnel inadvertently bumped control valve pressure transmitter	G	3	Instrumentation and controls (IE)	Instrumentation and controls
10)	8/16	10	F	Intermediate range NIS opened due to loose wires	A	3	Instrumentation and controls (IA)	Electrical conductors
11)	9/4	23	S	Precautionary method during Hurricane David	H	1	Electric power (EA)	Electrical conductors
12)	9/23	9	F	Malfunction in governor valve control system due to failed capacitor in EH system	A	3	Steam and power conversion (HA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	9/30	17	F	Fire on cold leg in "C" RCP bay due to thrust bearing oil leak	A	3	Reactor coolant (CB)	Pumps
14)	11/2	12	S	Perform turbine evaluation due to a gradual increase in turbine vibration	B	1	Steam and power conversion (HA)	Turbines
15)	12/13	3	F	"A" steam generator high level	G	3	Steam and power conversion (HH)	Instrumentation and controls
16)	12/22	6	F	Turbine trip during valve testing - lever not in full test position	G	3	Steam and power conversion (HB)	Valves



DESIGN ELEC. RATING = 700 MAX. DEPEND. CAP. = 665 (100%) ROBINSON 2

SALEM 1

I. Summary

Description	Performance		Outages		
Location: Salem, New Jersey	Net Electrical Energy Generated (MWH):	2,042,610	Total No.	7	
Docket No: 50-272			Forced	6	
Reactor Type: PWR	Unit Availability Factor (%):	25.5	Scheduled	1	
Capacity (MWe-Net): 1079			Total:	6,528 Hours,	74.5%
Commercial Operation: 6/30/77	Unit Capacity Factor (%) (Using MDC):	21.6	Forced	4,413 Hours,	50.4%
Plant Age: 3.0 Years			Scheduled	2,115 Hours,	24.1%
	Unit Capacity Factor (%) (Using Design MWE):	21.4			

II. Highlights

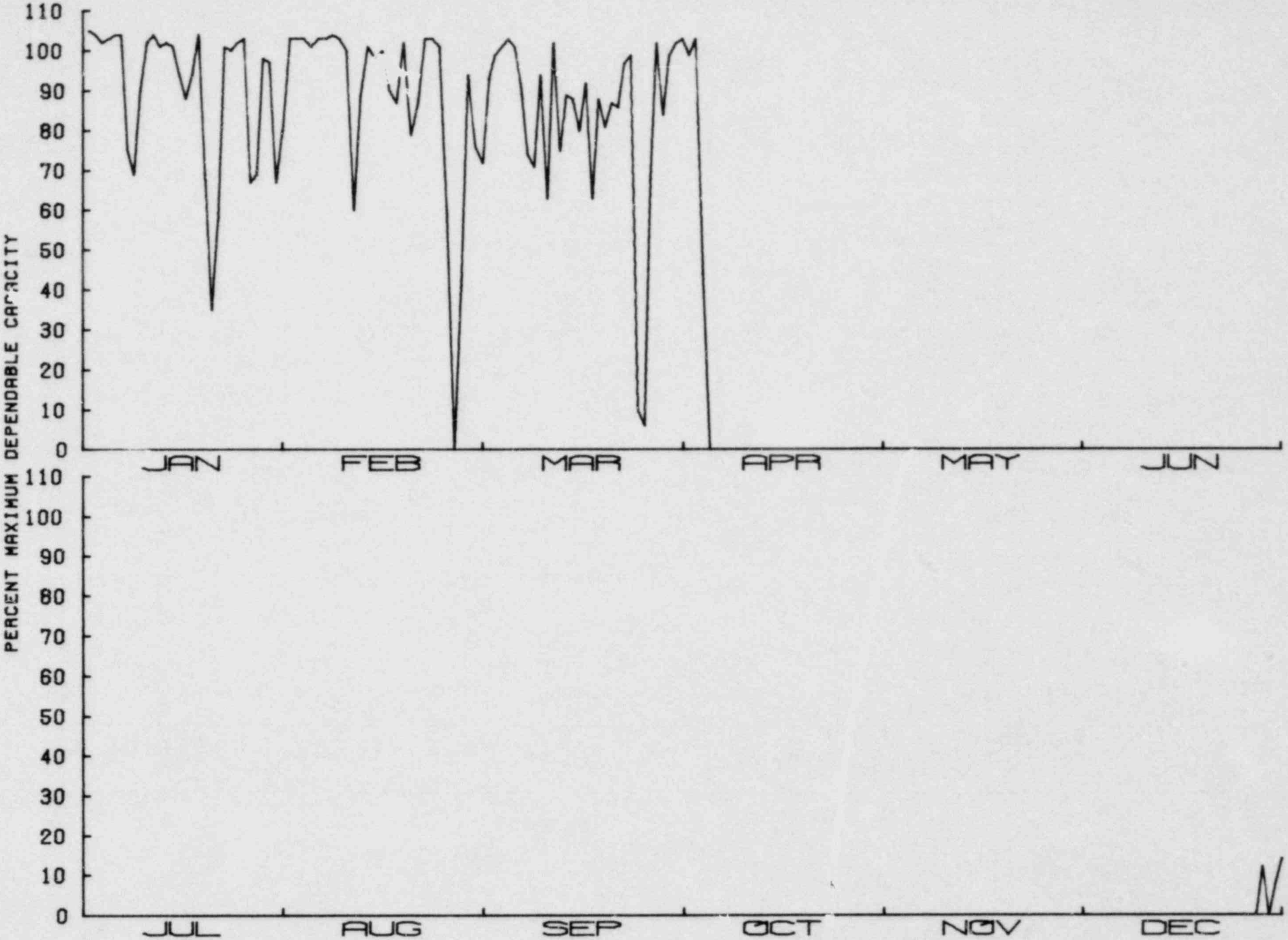
There were numerous power reductions during the first 3 months of the year because of problems with the circulating water system due primarily to fouling by submerged grass. A refueling and maintenance outage which began on April 3 was extended to December 28 for turbine blade maintenance, repair of cracks in steam generator feedwater nozzles, and inspection and repair of seismic hangers and anchor bolts.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/23	31	F	Auto trip on low S/G level caused by technician accidentally shorting test leads	G	3	Instrumentation and controls (IA)	Instrumentation and controls
2)	3/24	36	F	Failure of voltage regulator on main generator	A	3	Steam and power conversion (HA)	Generators (main generator)
3a)	4/3	2115	S	Refueling	C	1	Reactor (RC)	Fuel elements
3b)	4/3 (cont.)	2389	F	Turbine blades replacement	A	4	Steam and power conversion (HA)	Turbines
3c)	4/3 (cont.)	720	F	Modifications to seismic hangers and anchor bolts per I&E bulletin 79-07	D	4	Engineered safety features (SF)	Shock suppressors
3d)	4/3 (cont.)	639	F	NRC requirements	D	4	Engineered safety features (SH)	(not given)

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
3e)	4/7 (cont.)	564	F	Feedwater heater and nozzle block inspection and repair	A	4	Steam and power conversion (HH)	Pipes, fittings
4)	12/27	1	F	Switchgear problems	D	9	Engineered safety features (SH)	Circuit closers/ interrupters
5)	12/29	19	F	Required inspection	D	9	Engineered safety features (SH)	(not given)
6)	12/30	6	F	Required inspection	D	9	Engineered safety features (SH)	(not given)
7)	12/31	8	F	Steam generator high level	A	3	Steam and power conversion (HH)	Pumps



DESIGN ELEC. RATING = 1090 MAX. DEPEND. CAP. = 1079 (100%) SALEM 1

SAN ONOFRE 1

I. Summary

Description

Location: San Clemente, Calif.
Docket No: 50-206
Reactor Type: PWR
Capacity (MWe-Net): 436
Commercial Operation: 1/1/68
Plant Age: 12.5 Years

Performance

Net Electrical Energy Generated (MWH):	3,355,531
Unit Availability Factor (%):	90.2
Unit Capacity Factor (%) (Using MDC):	87.9
Unit Capacity Factor (%) (Using Design MWE):	87.9

Outages

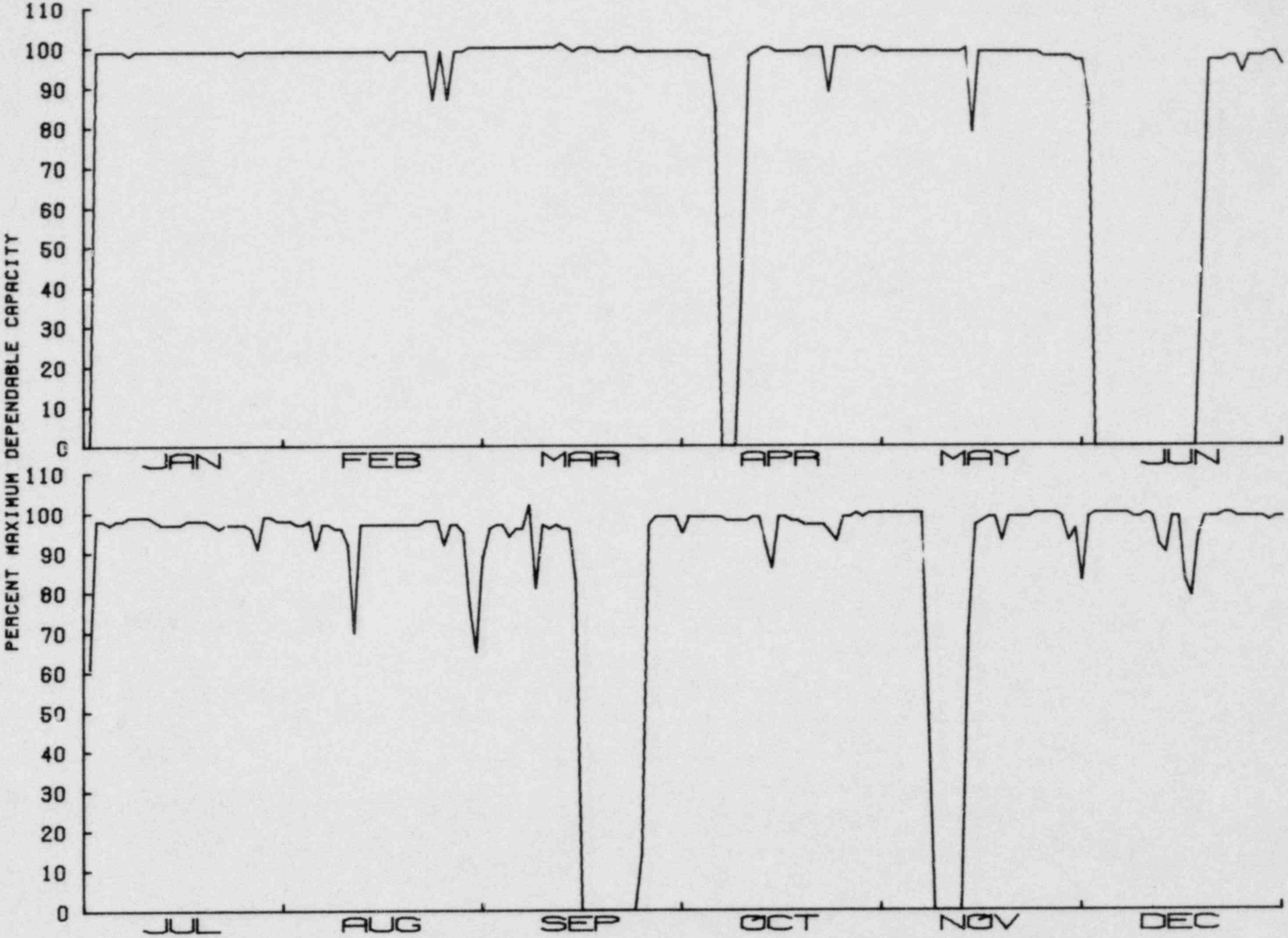
Total No.	6		
Forced	4		
Scheduled	2		
Total:	855	Hours,	9.8%
Forced	453	Hours,	5.2%
Scheduled	402	Hours,	4.6%

II. Highlights

Operation was routine during the year except for an outage in June to replace the steam generator feedwater nozzles. There were 6 months of uninterrupted operation; from January through March operation was continuous.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	4/5	82	F	Repair a major condenser tube leak and the feedwater flow straighteners	A	1	Steam and power conversion (HC)	Heat exchangers (condenser)
2)	5/14	4	F	Unit trip from 2 out of 3 variable low pressure trip channels while performing Delta T and TAVE tests	A	3	Instrumentation and controls (IA)	Instrumentation and controls
3)	6/1	394	S	Steam generator tube leak-tubes plugged	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
4)	8/30	8	S	Condenser tube leak	B	1	Steam and power conversion (HC)	Heat exchanger (condenser)
5)	9/14	234	F	Repair refueling water pump suction piping and replace pipe section on safety injection line	A	1	Engineered safety features (SF)	Pipes, fittings
6)	11/7	133	F	480 V Bus No. 1 failure	A	2	Electric power (EB)	Relays



DESIGN ELEC. RATING = 436 MAX. DEPEND. CAP. = 436 (100%) SAN ONOFRE 1

ST. LUCIE 1

I. Summary

Description

Location: Fort Pierce, Florida
Docket No: 50-335
Reactor Type: PWR
Capacity (MWe-Net): 777
Commercial Operation: 12/21/76
Plant Age: 3.7 Years

Performance

Net Electrical Energy Generated (MWH):	4,885,058
Unit Availability Factor (%):	74.0*
Unit Capacity Factor (%) (Using MDC):	71.8
Unit Capacity Factor (%) (Using Design MWE):	69.5

Outages

Total No.	16		
Forced	12		
Scheduled	4		
Total:	2,290	Hours,	26.1%*
Forced	212	Hours,	2.4%
Scheduled	2,078	Hours,	23.7%

II. Highlights

Operation was routine during the year, with refueling accomplished in April and May. There were 5 months of uninterrupted operation.

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*Includes 11 h of reserve shutdown equal to 0.1% availability.

DETAILS OF PLANT OUTAGES

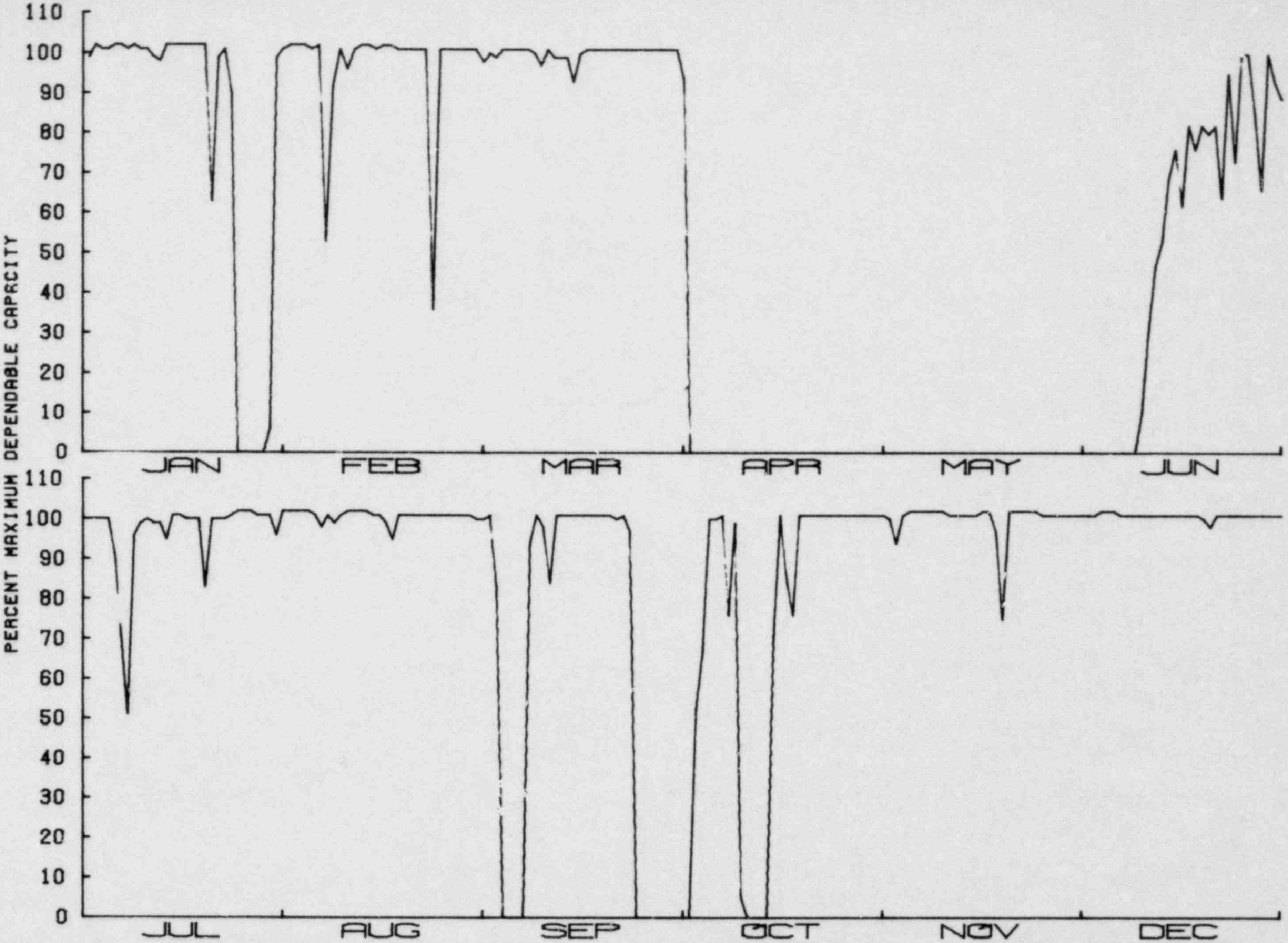
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/19	7	F	A rel y was actuated by severe vibration from construction activities on unit 2	H	3	Electric power (ED)	Relays
2)	1/22	140	F	Replace failed gasket on pressurizer manway closure. Outage was extended to restore CEA No. 43 to operable condition	A	1	Reactor coolant (CB)	Pressure vessels
3)	2/6	10	F	Periodic chemistry tests show low boron concentration in two safety injection tanks	A	2	Engineered safety features (SF)	Accumulators
4)	2/21	12	F	Sudden closure of feedwater isolation valves during a transient condition due to loss of vital instrument power	A	3	Electric power (ED)	Generators (Inverter)
5)	4/1	1644	S	Refueling	C	1	Reactor (RC)	Fuel elements
6)	6/8	4	F	Unit tripped during a transient condition caused by loss of steam generator feedwater pumps	A	3	Steam and power conversion (HH)	Pumps

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	6/9	9	F	Spurious signal from reactor protection system	A	3	Instrumentation and controls (IA)	Instrumentation and controls
8)	6/10	5	F	Unit was tripped during a transient condition by the steam generator level protection system	A	3	Steam and power conversion (HH)	Instrumentation and controls
9)	6/21	6	F	Second set of reactor trip breakers was opened before the first set was properly reset	G	3	Instrumentation and controls (IA)	Circuit closers/ interrupters
10)	6/23	3	F	Repair oil leak in the turbine control system	A	1	Steam and power conversion (HA)	Pipes, fittings
11)	9/2	95	S	Precautionary measure during Hurricane David and to inspect startup transformer No. 1B	H	1	Electric power (EA)	Electrical conductors
12)	9/23	219	S	Perform inspections required by NRC	D	1	Steam and power conversion (HH)	Pipes, fittings

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	10/3	6	F	Loss of condensate pump 1B causes transient	A	3	Steam and power conversion (HH)	Instrumentation and controls
14)	10/7	4	F	Loss of instrument air supply to feedwater control valve	A	3	Auxiliary process (PA)	Valve operators
15)	10/9	120	S	Repair mechanical seal on reactor coolant pump No. 1B2	B	1	Reactor coolant (CB)	Pumps
16)	10/17	6	F	Repair leak in turbine control oil system piping	A	1	Steam and power conversion (HA)	Pipes, fittings



DESIGN ELEC. RATING = 802   MAX. DEPEND. CAP. = 777 (100%)   ST LUCIE 1

SURRY 1

I. Summary

Description	Performance		Outages		
Location: Surry, Virginia	Net Electrical Energy Generated (MWH):	2,255,180	Total No.	4	
Docket No: 50-280			Forced	4	
Reactor Type: PWR	Unit Availability Factor (%):	75.3*	Scheduled	0	
Capacity (MWe-Net): 775			Total:	5,714 Hours,	65.2%*
Commercial Operation: 12/22/72	Unit Capacity Factor (%) (Using MDC):	33.2	Forced	5,714 Hours,	65.2%
Plant Age: 7.5 Years			Scheduled	0 Hours,	0%
	Unit Capacity Factor (%) (Using Design MWE):	31.3			

II. Highlights

The year began with the unit shut down for steam generation tube leak repair. Operation began January 2, but on March 15 a shutdown was ordered due to seismic design deficiencies in safety-related piping (IE Bulletin 79-14). This shutdown lasted until October 24. At the end of the year, the unit was again shut down for replacement and testing of a reactor coolant pump motor.

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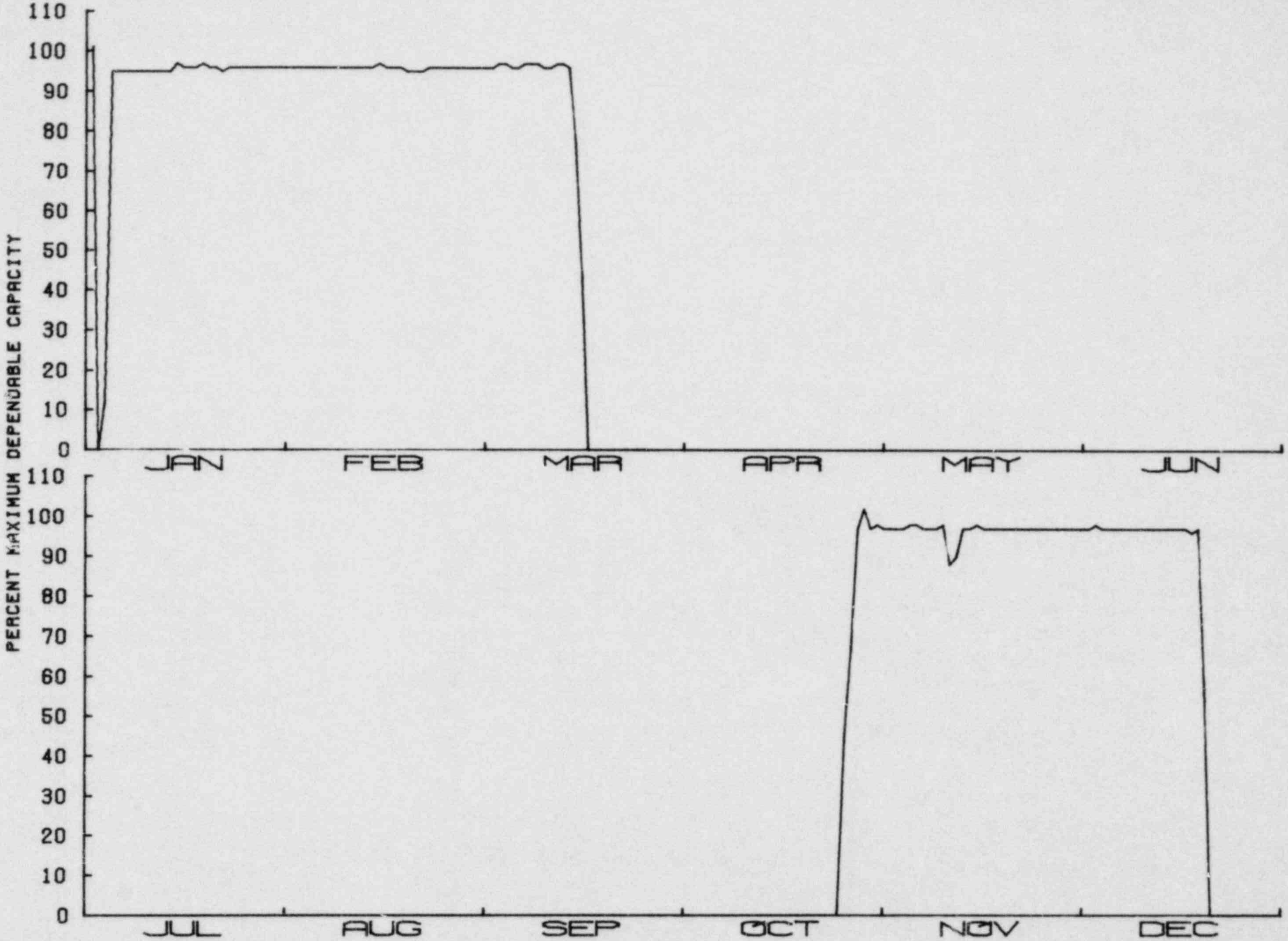
*Includes 3552.6 h of reserve shutdown equal to 40.5% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	12/12/78 (cont.)	41	F	Continuation of previous shutdown for steam generator tube leak repairs	A	4	Steam and power conversion (HB)	Heat exchangers (steam generator)
2)	3/15	5373	F	NRC show-cause order for reevaluation of stress calculations and modifications	D	1	Engineered safety features (SF)	Pipes, fittings
3)	10/24	2	F	Reactor tripped on feed flow/steam flow mismatch coincident with low s/g level signal while feeding s/g's in manual during power increase following startup	G	3	Steam and power conversion (HH)	Instrumentation and controls
4a)	12/19	250	F	Failure of IA reactor coolant pump motor on ground fault	A	3	Reactor coolant (CB)	Motors
4b)	12/19	48	F	NRC requirement to test RCP snubber prior to startup	D	4	Reactor coolant (CB)	Shock suppressors



DESIGN ELEC. RATING = 822 MAX. DEPEND. CAP. = 775 (100%) SURRY 1

SURRY 2

I. Summary

Description		Performance		Outages			
Location: Surry, Virginia		Net Electrical Energy Generated (MWH):	611,521	Total No.	1		
Docket No: 50-281		Unit Availability Factor (%):	9.3	Forced	0		
Reactor Type: PWR		Unit Capacity Factor (%) (Using MDC):	9.0	Scheduled	1		
Capacity (MWe-Net): 775		Unit Capacity Factor (%) (Using Design MWE):	8.5	Total:	7,941	Hours,	90.7%
Commercial Operation: 5/1/73				Forced	0	Hours,	0%
Plant Age: 6.8 Years				Scheduled	7,941	Hours,	90.7%

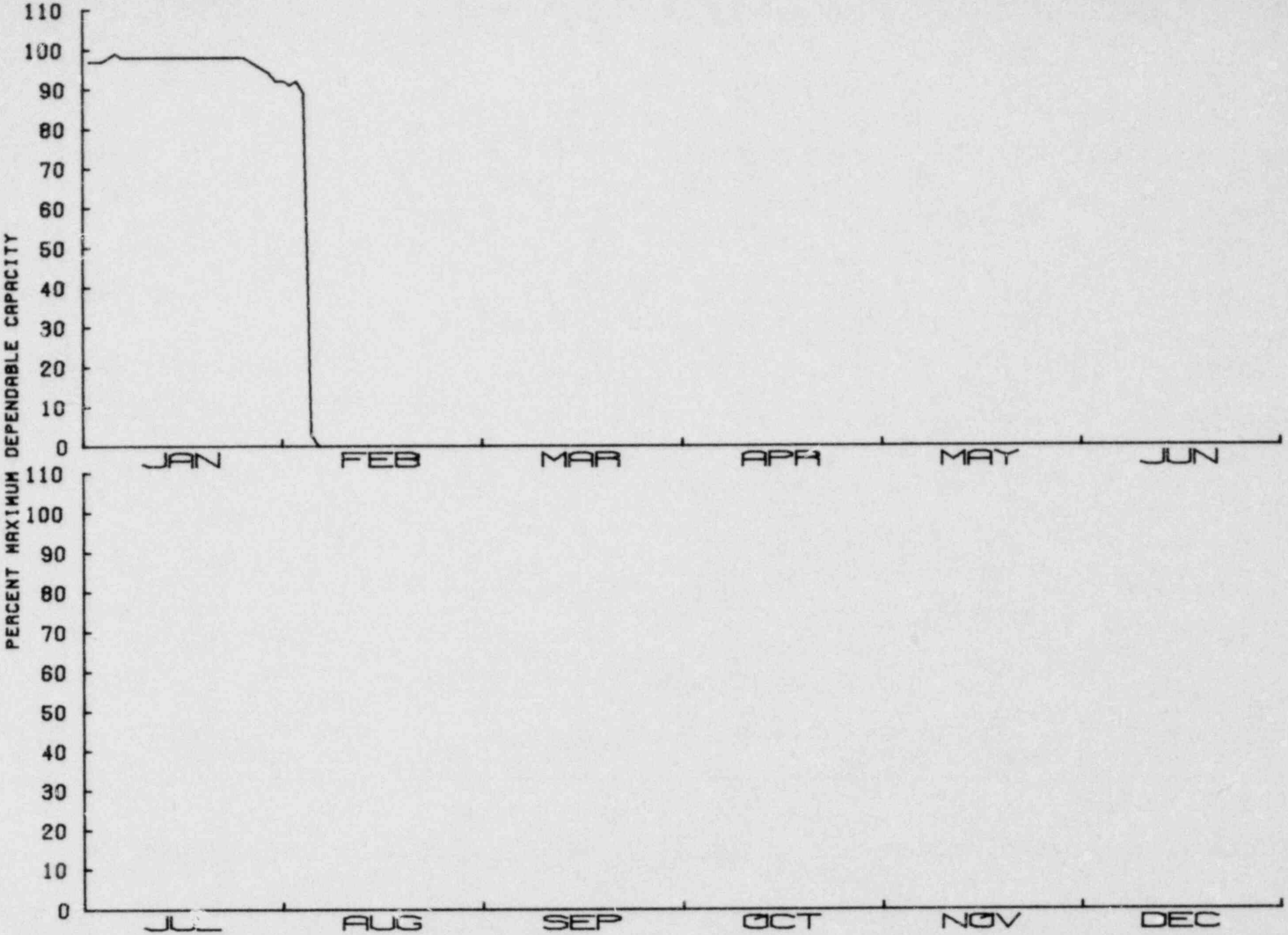
II. Highlights

Operation was uninterrupted until February 4 when the unit was shut down for refueling and replacement of the steam generators. The unit was still shut down at the end of the year.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a)	2/4	1000*	S	Refueling; the unit remained shut down for s/g replacement	C	1	Reactor (RC)	Fuel elements
1b)	2/4 (cont.)	6941*	S	Refueling; the unit remained shut down for s/g replacement	B	4	Steam and power conversion (HB)	Heat exchangers (steam generators)

*Estimated.



DESIGN ELEC. RATING = 822 MAX. DEPEND. CAP. = 775 (100%) SURRY 2

THREE MILE ISLAND 1

I. Summary

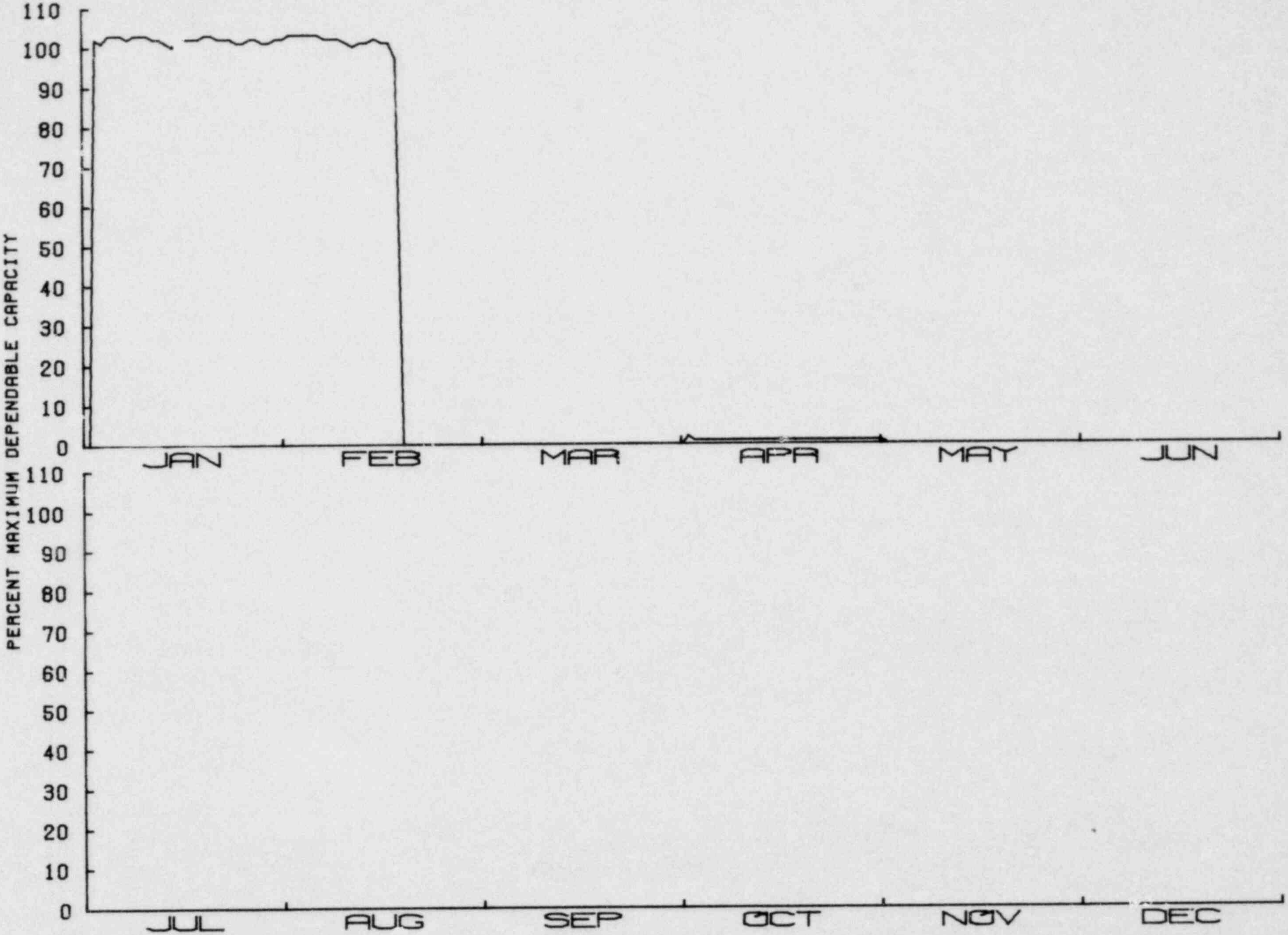
Description	Performance		Outages		
Location: Middletown, Pa.	Net Electrical Energy Generated (MWH):	848,038	Total No.	1	
Docket No: 50-289	Unit Availability Factor (%):	12.9	Forced	0	
Reactor Type: PWR	Unit Capacity Factor (%) (Using MDC):	12.5	Scheduled	1	
Capacity (MWe-Net): 776	Unit Capacity Factor (%) (Using Design MWE):	11.8	Total:	7,632 Hours,	87.1%
Commercial Operation: 9/2/74			Forced	6,692 Hours,	76.4%
Plant Age: 5.5 Years			Scheduled	940 Hours,	10.7%

II. Highlights

Operation was uninterrupted until the refueling outage was started February 17. Plant startup scheduled for March 28 was aborted due to the accident at TMI-2. Resumption of power generation was deferred for an undetermined period pending investigation of, and response to, that accident. The unit remained shut down the remainder of the year.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a)	2/17	940	S	Refueling	C	1	Reactor (RC)	Fuel elements
1b)	2/17 (cont.)	6692	F	The unit remained shut down for investigation of possible safety problems related to the TMI-2 accident	D	4	Steam and power conversion (HH)	Instrumentation and controls



DESIGN ELEC. RATING = 819   MAX. DEPEND. CAP. = 776 (100%)

THREE MILE ISLAND 1

THREE MILE ISLAND 2

I. Summary

Description

Location: Middletown, Pa.
Docket No: 50-320
Reactor Type: PWR
Capacity (MWe-Net): 880
Commercial Operation: 12/30/78
Plant Age: 1.7 Years

Performance*

Net Electrical Energy Generated (MWH):	1,318,113
Unit Availability Factor (%):	33.6
Unit Capacity Factor (%) (Using MDC):	31.2
Unit Capacity Factor (%) (Using Design MWE):	30.3

Outages

Total No.	5	
Forced	4	
Scheduled	1	
Total:	3,185 Hours,	66.4%
Forced	3,172 Hours,	66.1%
Scheduled	13 Hours,	0.3%

II. Highlights

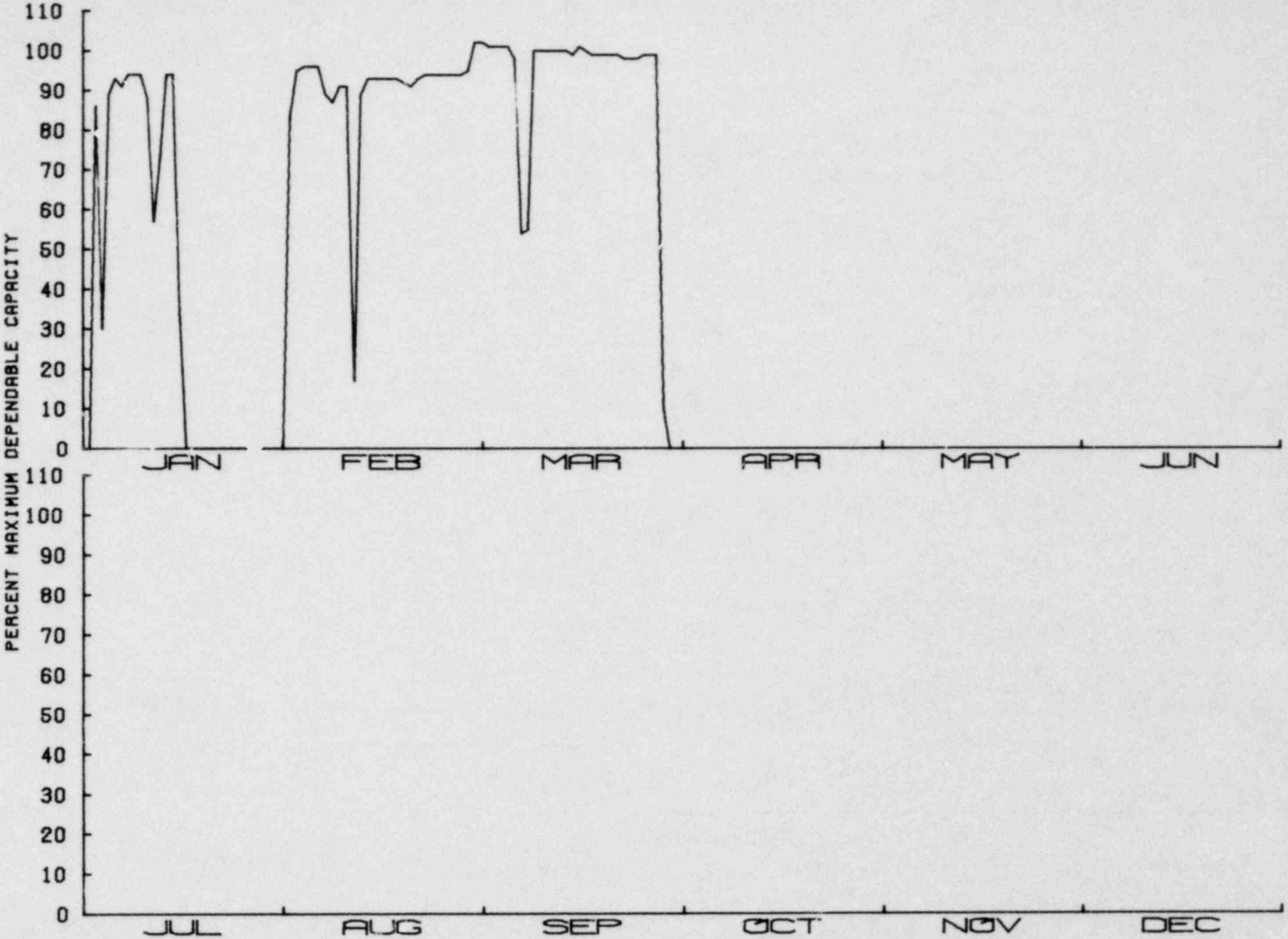
On January 15, a secondary system transient caused the rupture of both discharge piping bellows for an atmospheric relief valve. Operation resumed on January 31 after repairs. Routine operation continued until March 28 when a severe secondary-primary system transient resulted in partial uncovering of the core. The incident was considered a general emergency. The plant was placed in cold shutdown, and analysis to determine long-term corrective action was initiated. On July 20, the NRC issued an order suspending authority to operate the unit. At year-end, the unit remained shut down indefinitely, with no decision yet made on future operation.

*Based on data through July 19. License was suspended effective July 20, 1979. Total hours in the period were 4799 h.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/2	12	F	Repair hydraulic leak on GV-1	A	9	Steam and power conversion (HB)	Valves
2)	1/14	413	F	Repair leaking primary valves. During attempted startup the reactor tripped on low pressure, and outage continued for replacement of the atmospheric dump valve bellows	A	1	Reactor coolant (CB)	Valves
3)	2/10	13	S	Repair turbine EHC leaks	B	1	Steam and power conversion (HA)	Pipes, fittings
4)	3/6	17	F	Turbine generator trip followed by a reactor trip from core power imbalance	A	3	Instrumentation and controls (IA)	Instrumentation and controls
5)	3/28	2730*	F	Feedpump, turbine, and a reactor trip on high pressure resulted in partial uncovering of the core. Unit remains shut down pending investigations and recovery actions	A	3	Steam and power conversion (HH)	Pumps

*Based on data through July 19. License was suspended effective July 20, 1979. Total hours in the period were 4799 h.



DESIGN ELEC. RATING = 906 MAX. DEPEND. CAP. = 880 (100%)

THREE MILE ISLAND 2

TROJAN

I. Summary

Description

Location: Prescott, Oregon
Docket No: 50-344
Reactor Type: PWR
Capacity (MWe-Net): 1,080
Commercial Operation: 5/20/76
Plant Age: 4.0 Years

Performance

Net Electrical Energy Generated (MWH):	5,266,720
Unit Availability Factor (%):	58.1
Unit Capacity Factor (%) (Using MDC):	55.7
Unit Capacity Factor (%) (Using Design MWE):	53.2

Outages

Total No.	10		
Forced	6		
Scheduled	4		
Total:	3,671	Hours,	41.9%
Forced	102	Hours,	1.2%
Scheduled	3,569	Hours,	40.7%

II. Highlights

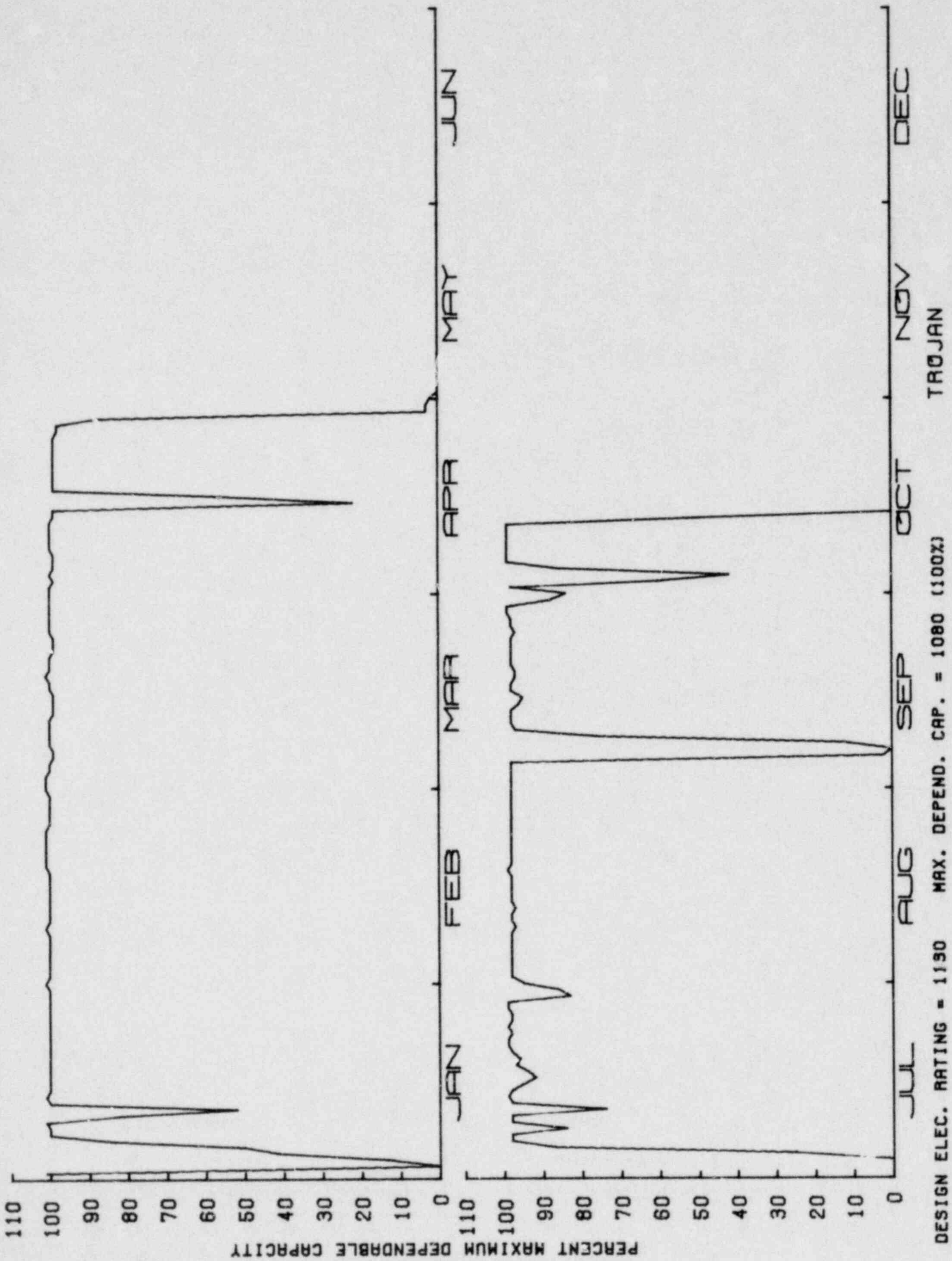
There were two extensive outages during the year. One outage during May and June was for maintenance and surveillance testing. A second outage, beginning October 12, was required by the NRC for inspection of pipe hangers and restraints inside the containment. This outage lasted through the remainder of the year, with reactor startup taking place on December 31 in preparation for power generation.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	5/26/78 (cont.)	32	S	Completion of outage for seismic qualification of control building structure	D	4	Other (control building)	Other (control building)
2)	1/9	13	F	Loss of EHC power while working on the turbine generator thrust bearing wear detector	A	3	Steam and power conversion (HA)	Mechanical function units
3)	4/14	17	F	Steam generator "A" tripped due to capacitor problems	A	3	Steam and power conversion (HB)	Instrumentation and controls
4)	4/14	2	F	Steam generator "A" tripped due to capacitor problems	A	3	Steam and power conversion (HB)	Instrumentation and controls
5a)	4/27	608	S	Maintenance, surveillance, and containment leak rate testing	B	3	Engineered safety features (SA)	Pressure vessels
5b)	4/27 (cont.)	1000	S	(Not needed because of excess of hydro power)	H	4	Steam and power conversion (HA)	N/A

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	7/4	1	S	Turbine-generator control valve testing	B	1	Steam and power conversion (HA)	Valves
7)	7/11	4	F	Turbine-generator underfrequency trip due to failure of underfrequency relay	A	3	Steam and power conversion (HA)	Relays
8)	9/5	63	F	Turbine control valves inadvertently opened during maintenance on the EHC system and caused a safety injection signal	A	3	Steam and power conversion (HA)	Valves
9)	10/2	3	F	Steam line "A" MSIV closed accidentally when workmen disturbed an air line to the control solenoid valve	G	3	Auxiliary process (PA)	Pipes, fittings
10)	10/12	1928	S	NRC required inspection of hangers and restraints within the containment	D	1	Engineered safety features (SF)	Shock suppressors


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DEC
DESIGN ELEC. RATING = 1130 MAX. DEPEND. CAP. = 1080 (100%)
TROJAN

TURKEY POINT 3

I. Summary

Description

Location: Florida City, Florida
Docket No: 50-250
Reactor Type: PWR
Capacity (MWe-Net): 666
Commercial Operation: 12/14/72
Plant Age: 7.2 Years

Performance

Net Electrical Energy Generated (MWH):	2,874,917
Unit Availability Factor (%):	51.8*
Unit Capacity Factor (%) (Using MDC):	49.3
Unit Capacity Factor (%) (Using Design MWE):	47.4

Outages

Total No.	21		
Forced	14		
Scheduled	7		
Total:	4,248	Hours,	48.5%*
Forced	146	Hours,	1.7%
Scheduled	4,102	Hours,	46.8%

II. Highlights

A refueling outage was in effect from January 1 to April 16. At the end of the year, the unit was again in a refueling outage that began December 1. Except for problems with leaking steam generator tubes during the first half of the year, operation was routine.

*Includes 24.5 h of reserve shutdown equal to 0.3% availability.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a)	1/1	2000*	S	Refueling	C	1	Reactor (RC)	Fuel elements
1b)	1/1 (cont.)	528*	S	Outage was continued to repair mechanical seals on coolant pumps	B	4	Reactor coolant (CB)	Pumps
2)	4/16	1	F	Loss of turbine system oil pressure	A	3	Steam and power conversion (HA)	Pumps
3)	4/16	5	F	Balance generator exciter	A	9	Steam and power conversion (HA)	Generators (exciter)
4)	4/16	8	F	Balance generator exciter	A	9	Steam and power conversion (HA)	Generators (exciter)
5)	4/17	14	F	Balance generator exciter	A	9	Steam and power conversion (HA)	Generators (exciter)
6)	4/19	85	F	High temperature in generator stator	A	1	Steam and power conversion (HA)	Generators (main generator)

*Estimated.

DETAILS OF PLANT OUTAGES

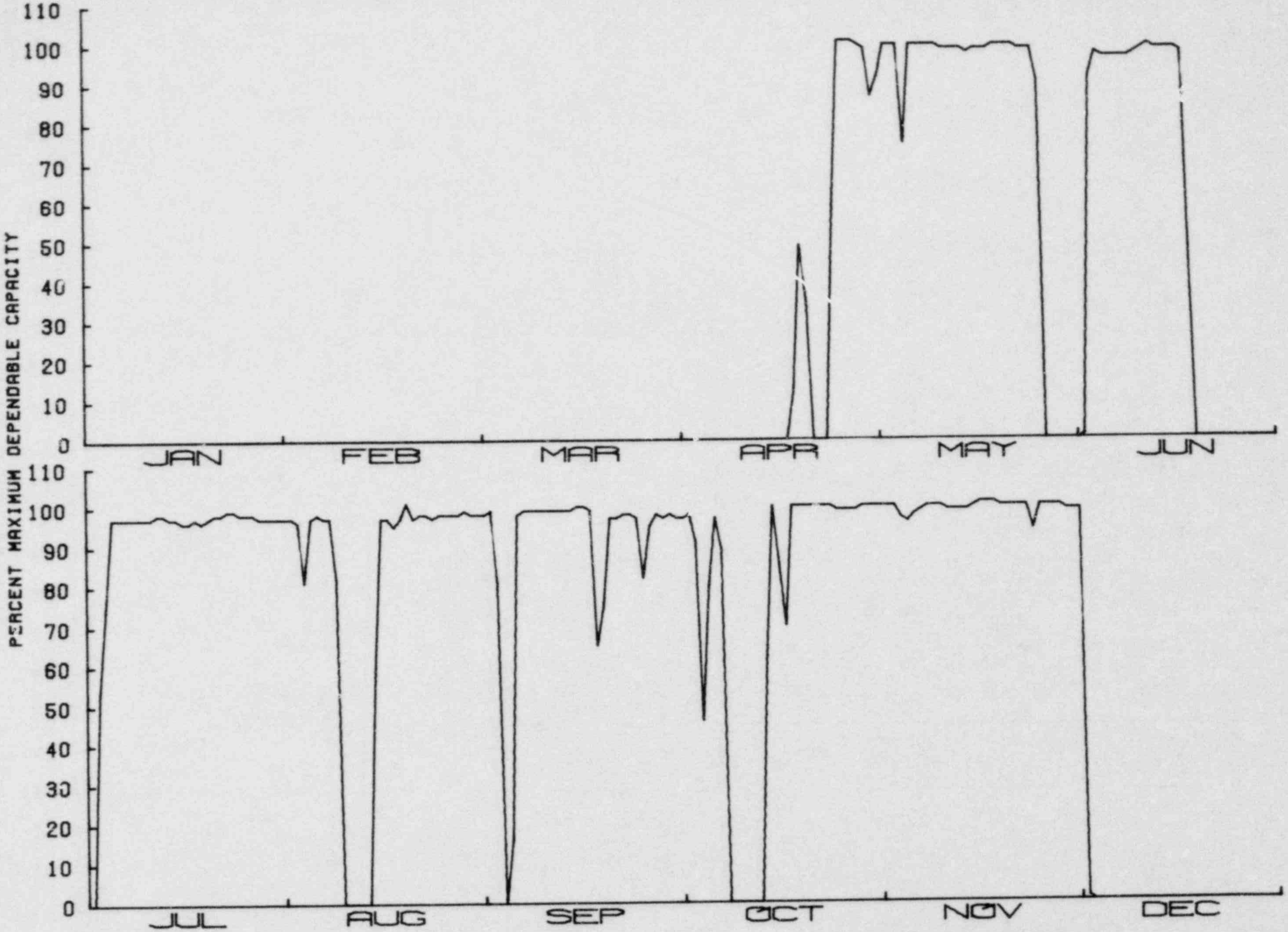
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	4/29	3	F	Unit was tripped by s/g level protection system. Condensate pump discharge check valve was repaired	A	1	Steam and power conversion (HH)	Valves
8)	5/4	3	F	S/G repair	A	3	Steam and power conversion (HB)	Heat exchangers (steam generator)
9)	5/25	167	S	Perform safeguards surveillance tests	B	1	Engineered safety features (SF)	Instrumentation and controls
10)	6/17	344	S	Reactor coolant pump seal repairs	B	1	Reactor coolant (CB)	Pumps
11)	7/1	5	F	Unit was tripped by S/G 3A level protection system during a transient condition	A	3	Steam and power conversion (HH)	Instrumentation and controls
12)	7/2	7	F	Unit was tripped by S/G 3A level protection system during a transient condition	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	8/3	3	F	Temporary loss of power to the rod position indication system	A	3	Instrumentation and controls (ID)	Instrumentation and controls
14)	8/8	128	S	Repair severe packing leaks on valves inside containment	B	1	Reactor coolant (CB)	Valves
15)	9/2	45	S	Precautionary measure against Hurricane David; outage extended to replace fittings	F	1	Electric power (EA)	Electrical conductors
16)	9/17	2	F	Failed diaphragm in the low vacuum trip device	A	3	Steam and power conversion (HC)	Instrumentation and controls
17)	9/17	6	F	Failed diaphragm in the low vacuum trip device	A	1	Steam and power conversion (HC)	Instrumentation and controls
18)	9/24	2	F	Unit tripped by reactor protection system during a periodic surveillance test	G	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
19)	10/6	148	S	Repair tube leaks in moisture separator reheater	B	1	Steam and power conversion (HB)	Heat exchanger (MSR)
20)	10/13	2	F	Unit was tripped by S/G No. 3C level protection system	A	3	Steam and power conversion (HH)	Instrumentation and controls
21)	12/1	742	S	Refueling	C	1	Reactor (RC)	Fuel elements



TURKEY POINT 3

DESIGN ELEC. RATING = 693 MAX. DEPEND. CAP. = 666 (100%)

TURKEY POINT 4

I. Summary

Description	Performance		Outages		
Location: Florida City, Florida	Net Electrical Energy Generated (MWH):	3,845,291	Total No.	16	
Docket No: 50-251			Forced	13	
Reactor Type: PWR	Unit Availability Factor (%):	72.9*	Scheduled	3	
Capacity (MWe-Net): 666			Total:	2,396 Hours,	27.3%*
Commercial Operation: 9/7/73	Unit Capacity Factor (%) (Using MDC):	65.9	Forced	384 Hours,	4.4%
Plant Age: 6.5 Years			Scheduled	2,012 Hours,	22.9%
	Unit Capacity Factor (%) (Using Design MWE):	63.3			

II. Highlights

The year began with the unit at reduced power to extend core life until the refueling outage that began on April 5 and was completed June 21. There were some problems with leaking steam generator tubes during the first half of the year, but otherwise operation during the year was routine.

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*Includes 18.9 h of reserve shutdown equal to 0.2% availability.

DETAILS OF PLANT OUTAGES

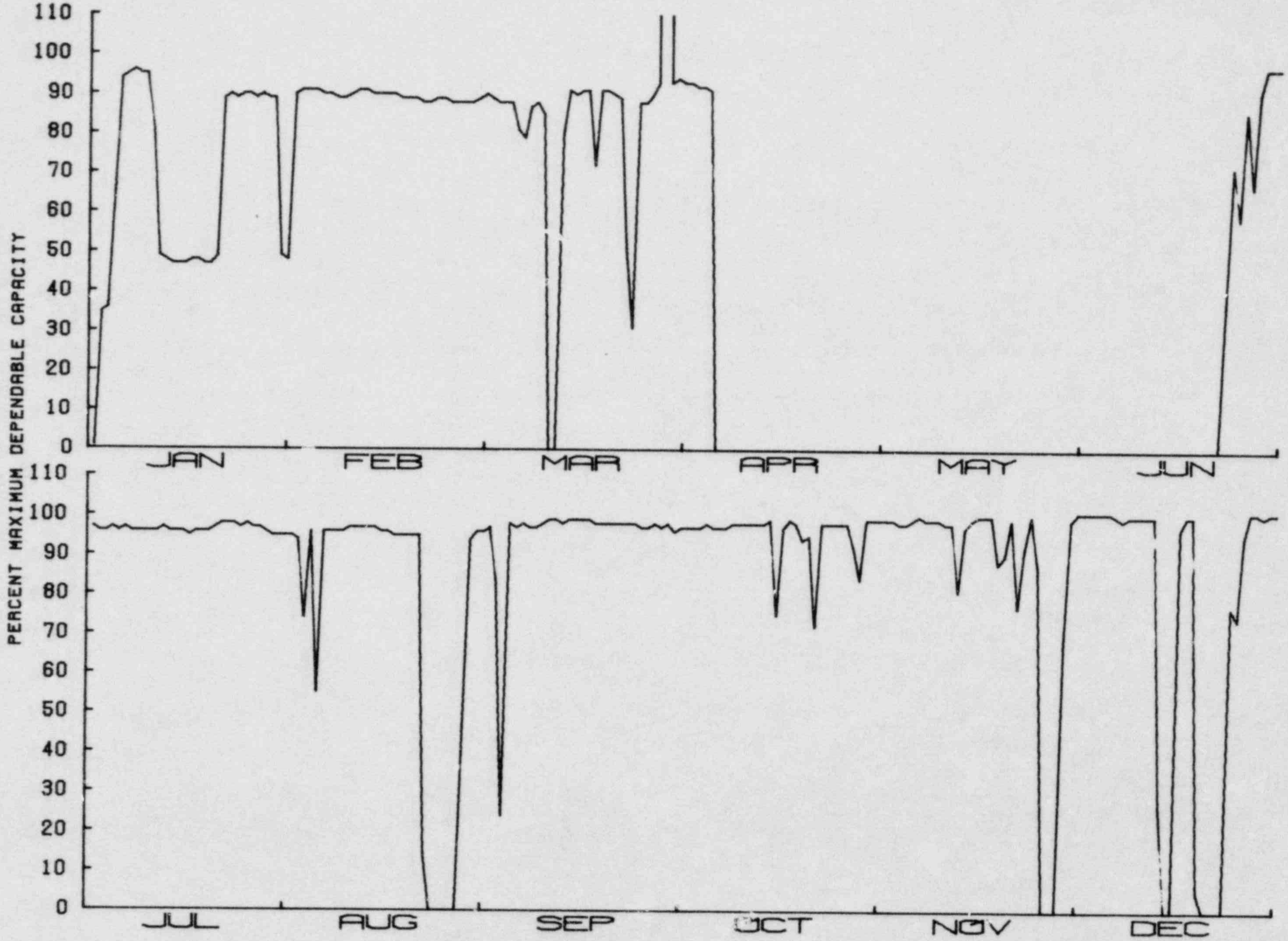
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/9	48	S	Perform safeguards surveillance tests	B	1	Engineered safety features (SF)	Instrumentation and controls
2)	3/17	1	F	Unit was tripped by reactor protection system during tests when the second of two channels was placed in test mode in error	G	3	Instrumentation and controls (IA)	Instrumentation and controls
3)	3/17	1	F	Sudden closure of a turbine stop valve caused by a malfunction of the turbine control system	A	3	Steam and power conversion (HA)	Instrumentation and controls
4)	3/22	24	F	Repair leaks on valves inside containment	A	2	Reactor coolant (CB)	Valves
5a)	4/4	4	F	Unit tripped due to system load conditions	H	3	Electric power (EA)	Electrical conductors
5b)	4/4 (cont.)	1866	S	Refueling	C	4	Reactor (RC)	Fuel elements

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
6)	6/24	4	F	Loss of control signal to feed-water control valve	A	2	Steam and power conversion (HH)	Instrumentation and controls
7)	8/3	4	F	Unit was tripped due to spurious signal from reactor protection system	A	3	Instrumentation and controls (IA)	Instrumentation and controls
8)	8/5	9	F	Turbine trip relatch device failed to reset properly	A	3	Steam and power conversion (HA)	Instrumentation and controls
9)	8/22	148	F	Excessive vibration of reactor coolant pump shaft	A	2	Reactor coolant (CB)	Pumps
10)	8/28	2	F	Unit tripped due to S/G level	A	3	Steam and power conversion (HH)	Instrumentation and controls
11)	9/2	19	S	Precautionary measure due to Hurricane David	H	1	Electric power (EA)	Electrical conductors

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12)	11/12	2	F	Loss of signal to feedwater control system	A	3	Steam and power conversion (HH)	Instrumentation and controls
13)	11/24	79	S	Repair feedwater pump discharge check valve	B	1	Steam and power conversion (HH)	Valves
14)	12/13	62	F	Repair steam leak on S/G 4B steam flow sensing line that could not be isolated	A	1	Steam and power conversion (HB)	Pipes, fittings
15)	12/15	1	F	Repair generator disconnect switches	A	9	Steam and power conversion (HA)	Circuit closers/ interrupters
16)	12/19	122	F	Unit trip by reactor protection system due to RCP motor overcurrent relay trip. Repaired motor leads	A	3	Reactor coolant (CB)	Electrical conductors



DESIGN ELEC. RATING = 693 MAX. DEPEND. CAP. = 666 (100%) TURKEY POINT 4

VERMONT YANKEE

I. Summary

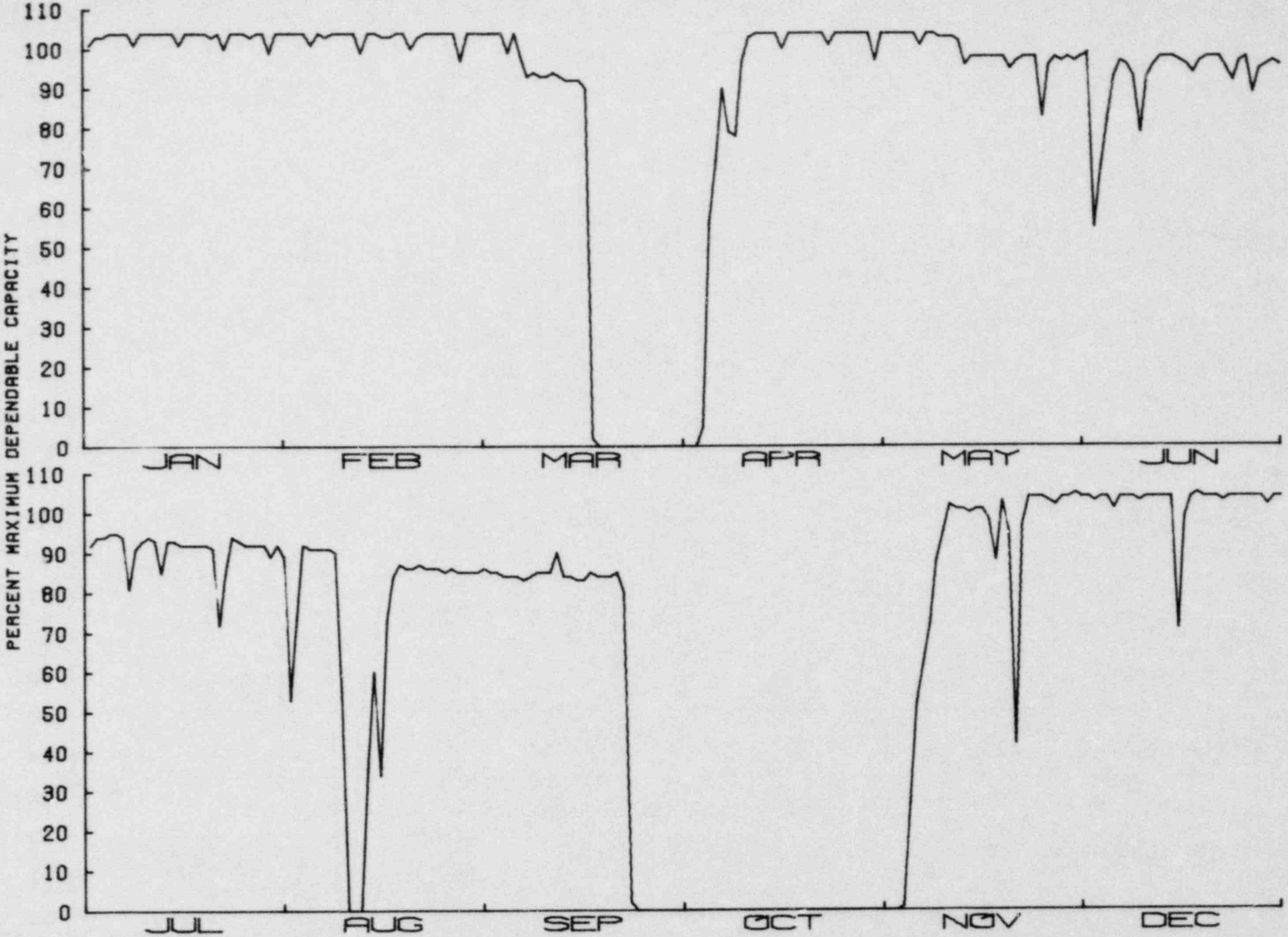
Description	Performance		Outages			
Location: Vernon, Vermont	Net Electrical Energy Generated (MWH):	3,448,842	Total No.	5		
Docket No: 50-271			Forced	3		
Reactor Type: BWR	Unit Availability Factor (%):	82.1	Scheduled	2		
Capacity (MWe-Net): 504			Total:	1,565	Hours,	17.9%
Commercial Operation: 11/30/72	Unit Capacity Factor (%) (Using MDC):	78.1	Forced	115	Hours,	1.3%
Plant Age: 7.3 Years			Scheduled	1,450	Hours,	16.6%
	Unit Capacity Factor (%) (Using Design MWE):	76.6				

II. Highlights

There were two refueling outages during the year — one from March 16 to April 3 and the other from September 22 to November 2. During the latter outage, abnormal wear was observed on 8 × 8R—type fuel water rod end plugs. There were 6 months of uninterrupted operation; from May through July operation was continuous.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	3/16	421	S	Refueling	C	2	Reactor (RC)	Fuel elements
2)	8/9	88	F	Repair bonnet leak in the "A" recirculating pump discharge valve	A	2	Reactor coolant (CB)	Valves
3)	8/14	13	F	Power spikes caused by instabilities in the electronic pressure regulator	A	3	Reactor coolant (CC)	Instrumentation and controls
4)	9/22	1029	S	Refueling	C	1	Reactor (RC)	Fuel elements
5)	11/19	14	F	Inadvertent striking of a protection system instrument panel	G	3	Instrumentation and controls (IA)	Instrumentation and controls



DESIGN ELEC. RATING = 514 MAX. DEPEND. CAP. = 504 (100%) VERMONT YANKEE 1

YANKEE-ROWE

I. Summary

Description	Performance		Outages		
Location: Rowe, Mass.	Net Electrical Energy Generated (MWH):	1,232,264	Total No.	5	
Docket No: 50-29			Forced	3	
Reactor Type: PWR	Unit Availability Factor (%):	81.6	Scheduled	2	
Capacity (MWe-Net): 175			Total:	1,611 Hours,	18.4%
Commercial Operation: 7/61	Unit Capacity Factor (%) (Using MDC):	80.4	Forced	175 Hours,	2.0%
Plant Age: 19.1 Years			Scheduled	1,436 Hours,	16.4%
	Unit Capacity Factor (%) (Using Design MWE):	80.4			

II. Highlights

An outage from September 8 through November 5 was required to perform inspection in accordance with IE Bulletins 79-2, 79-13, and 70-17. During the outage, code defects were found in all four steam generator feedwater nozzle welds. Operation was routine the remainder of the year. The unit generated power without interruption from March 4 until the shutdown on September 8, the equivalent of 6 months. In addition, after resuming operation on November 5, the unit operated continuously the remainder of the year.

DETAILS OF PLANT OUTAGES

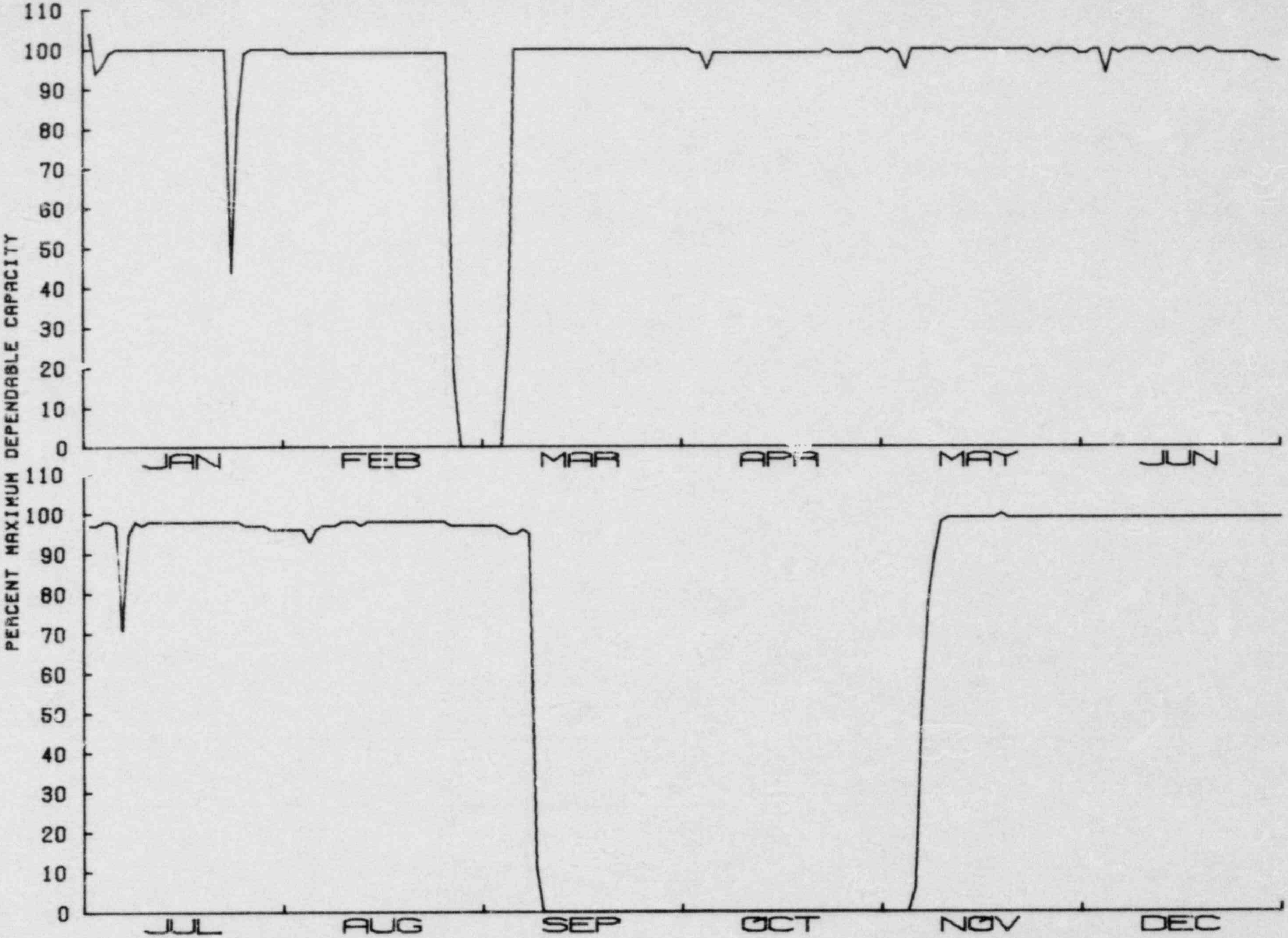
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	1/22	10	F	Turbine control valve oscillation during low power	A	9	Steam and power conversion (HB)	Valves
2)	2/24	33	S	Repair turbine control valve motor	B	3	Steam and power conversion (HA)	Valve operators
3)	2/25	80	F	Control rod No. 2 failed to move beyond 42"	A	1	Reactor (RB)	Control rod drives
4)	3/1	85	F	Control rod No. 2 stopped withdrawal at 42"	A	1	Reactor (RB)	Control rod drives
5a)	9/8	468	S	Perform inspections of pipe and supports per I&E bulletins 79-2 (pipe support base plate anchor bolts), 79-13 (cracking in feedwater piping), and 79-17 (pipe cracks in stagnant borated water systems)	D	1	Engineered safety features (SX)	Shock suppressors and supports
5b)	9/8 (cont.)	468*	S	Perform inspections per I&E bulletins 79-2, 79-13, & 79-17	D	4	Steam and power conversion (HH)	Pipes, fittings

*Estimated

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
5c)	9/8 (cont.)	467*	S	Perform inspections per I&E bulletins 79-2, 79-13, & 79-17	D	4	Engineered safety features (SX)	Pipes, fittings

*Estimated



DESIGN ELEC. RATING = 175   MAX. DEPEND. CAP. = 175 (100%)   YANKEE-ROWE 1

ZION 1

I. Summary

Description

Location:	Zion, Illinois
Docket No:	50-295
Reactor Type:	PWR
Capacity (MWe-Net):	1,040
Commercial Operation:	12/31/73
Plant Age:	6.5 Years

Performance

Net Electrical Energy Generated (MWH):	5,537,168
Unit Availability Factor (%):	68.1
Unit Capacity Factor (%) (Using MDC):	60.8
Unit Capacity Factor (%) (Using Design MWE):	60.8

Outages

Total No.	22		
Forced	21		
Scheduled	1		
Total:	2,790	Hours,	31.9%
Forced	1,689	Hours,	19.3%
Scheduled	1,101	Hours,	12.6%

II. Highlights

Operation during the year was normal. Some problems were experienced with the feedwater system during the first quarter, and on October 6 the unit was shut down for refueling and feedwater nozzle repair in accordance with IE Bulletin 79-13. This outage lasted the remainder of the year, with resumption of operation scheduled for January 1980.

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/1	17	F	1B feedwater pump oscillations	A	3	Steam and power conversion (HH)	Pumps
2)	2/2	11	F	High level in the 1D s/g	A	3	Steam and power conversion (HH)	Instrumentation and controls
3)	3/2	18	F	S/G level control problems	A	3	Steam and power conversion (HH)	Instrumentation and controls
4)	3/5	11	F	Feedwater pump problems	A	3	Steam and power conversion (HH)	Pumps
5)	3/16	18	F	Feedwater pump problems	A	3	Steam and power conversion (HH)	Pumps
6)	3/21	15	F	Spurious power range positive rate trip	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

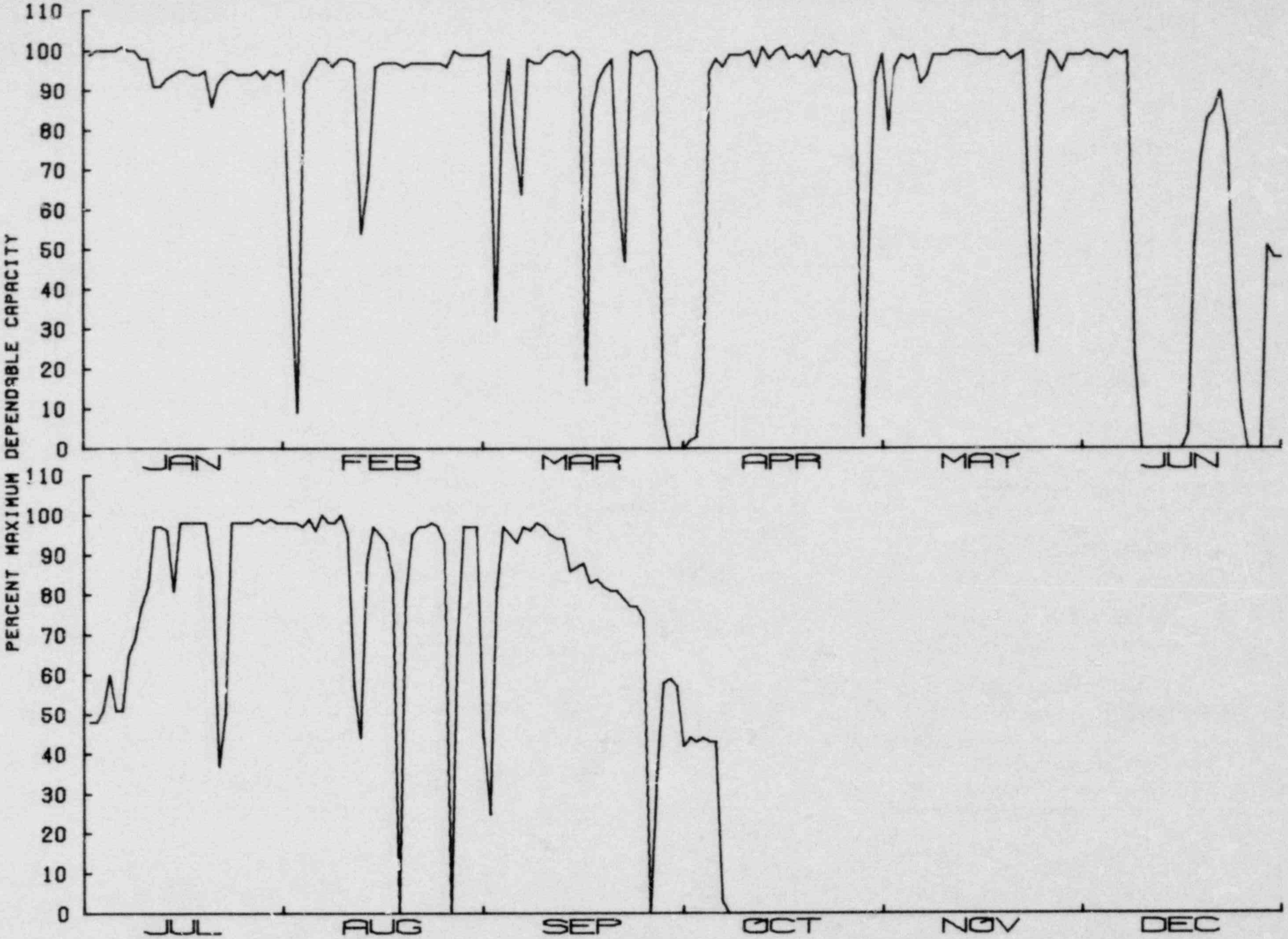
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	3/22	4	F	S/G "C" high level	A	3	Steam and power conversion (HH)	Instrumentation and controls
8)	3/28	156	F	Primary system coolant leaks	A	1	Reactor coolant (CB)	Pipes, fittings
9)	4/26	15	F	Low-low level in S/G due to loss of 1B FW pump	A	3	Steam and power conversion (HH)	Pumps
10)	4/27	7	F	During startup, reactor tripped on low level in coincidence with feedwater flow/steam flow mismatch	G	3	Steam and power conversion (HH)	Instrumentation and controls
11)	4/27	4	F	Turbine trip (EHC problems)	A	3	Steam and power conversion (HA)	Mechanical function units
12)	5/23	26	F	Safety injection and reactor trip occurred during surveillance testing due to spurious signal	G	3	Engineered safety features (SF)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	6/8	206	F	Spurious safety injection signal caused by a water hammer	G	3	Engineered safety features (SF)	Instrumentation and controls
14)	6/23	3	F	Repair exciter bearing	A	1	Steam and power conversion (HA)	Generators (exciter)
15)	6/23	4	F	Repair exciter bearing	A	1	Steam and power conversion (HA)	Generators (exciter)
16)	6/24	90	F	Repair exciter bearing	A	1	Steam and power conversion (HA)	Generators (exciter)
17)	8/17	24	F	Severe lightning	H	3	Electric power (EA)	Electrical conductors
18)	8/26	28	F	Repair minor secondary steam leaks in containment	A	1	Steam and power conversion (HA)	Pipes, fittings

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
19)	8/31	26	F	Power supply failure in rod control system	A	3	Reactor (RB)	Control rod drives
20)	9/25	26	F	Printer circuit card failure for control rods	A	3	Reactor (RB)	Control rod drives
21)	10/6	1101	S	Refueling	C	1	Reactor (RC)	Fuel elements
22)	11/21	980	F	Feedwater nozzle repair per NRC bulletin	D	9	Steam and power conversion (HH)	Pipes, fittings



DESIGN ELEC. RATING = 1040   MAX. DEPEND. CAP. = 1040 (100%)   ZION 1

ZION 2

I. Summary

Description

Location: Zion, Illinois
Docket No: 50-304
Reactor Type: PWR
Capacity (MWe-Net): 1,040
Commercial Operation: 9/17/74
Plant Age: 6.0 Years

Performance

Net Electrical Energy Generated (MWH):	4,759,996
Unit Availability Factor (%):	67.2
Unit Capacity Factor (%) (Using MDC):	52.2
Unit Capacity Factor (%) (Using Design MWE):	52.2

Outages

Total No.	16		
Forced	15		
Scheduled	1		
Total:	2,874	Hours,	32.8%
Forced	1,920	Hours,	21.9%
Scheduled	954	Hours,	10.9%

II. Highlights

The unit experienced a refueling outage from March 9 through March 17. A major outage was initiated on October 27 for inspection and repair of feedwater nozzles in accordance with IE Bulletin 79-13. This outage was in effect for the remainder of the year, with resumption of operation scheduled for January 1980.

DETAILS OF PLANT OUTAGES

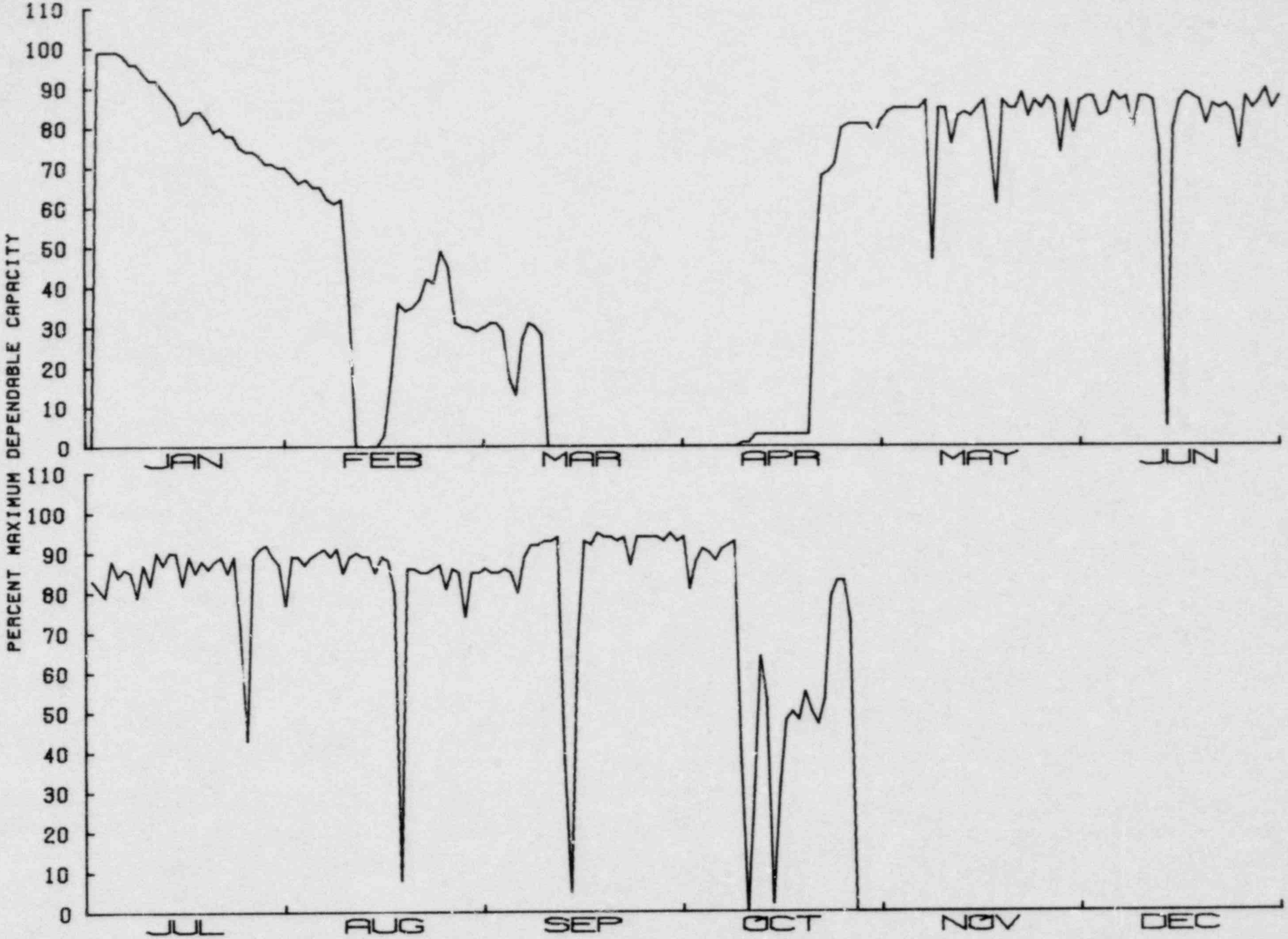
No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1)	2/9	106	F	High s/g conductivity	A	1	Steam and power conversion (HG)	Demineralizers
2)	2/14	20	F	Feedwater pump problems	A	2	Steam and power conversion (HH)	Pumps
3)	2/15	5	F	Generator reverse power trip occurring when the EHC initial valve position did not automatically open the turbine governor valves as the generator was synchronized with the system	A	3	Steam and power conversion (HA)	Mechanical function units
4)	3/4	19	F	Repair pressurizer level channels	A	1	Reactor coolant (CB)	Instrumentation and controls
5)	3/9	954	S	Refueling	C	1	Reactor (RC)	Fuel elements
6)	4/19	17	F	Reverse power due to a pressure sensor mismatch	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7)	5/8	9	F	Low level in 2A s/g in coincidence with steam flow/feed flow mismatch due to a loss at the 2B main feedwater pump	A	3	Steam and power conversion (HH)	Pumps
8)	6/13	20	F	Breaker latching mechanism was bent and holding latch would not clear. While attempting to rack out the breaker a trip occurred	G	3	Electric power (EB)	Circuit closers/ interrupters
9)	7/24	14	F	A reactor/turbine trip occurred while returning the 2C feedwater pump to operation	G	3	Steam and power conversion (HH)	Instrumentation and controls
10)	8/17	17	F	Severe lightning	H	3	Electric power (EA)	Electrical conductors
11)	8/18	3	F	Feedwater flow control and steam generator level problems during startup	A	3	Steam and power conversion (HH)	Instrumentation and controls
12)	9/12	30	F	DC bus interlock key improperly removed	G	3	Electric power (ED)	Circuit closers/ interrupters

DETAILS OF PLANT OUTAGES

No.	Date (1979)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13)	9/13	1	F	Generator trip from reverse power during startup	A	3	Steam and power conversion (HA)	Generator (main generator)
14)	10/9	48	F	Turbine/generator trip — cause unknown	A	3	Steam and power conversion (HA)	Instrumentation and controls
15)	10/14	26	F	Trip during shutdown to repair condenser vacuum leak	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
16)	10/27	1585	F	Inspect feedwater nozzles per NRC bulletin	D	1	Steam and power conversion (HH)	Pipes, fittings



DESIGN ELEC. RATING = 1040 MAX. DEPEND. CAP. = 1040 (100%) ZION 2

<u>Appendix C</u>

ABNORMAL OCCURRENCE CRITERIA

For this report, the following criteria for abnormal occurrence determinations were used. These criteria were promulgated in an NRC policy statement which was published in the *Federal Register*, Vol. 42, pp. 10950—52, February 24, 1977.

Events involving a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to: (1) moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the NRC; (2) major degradation of essential safety-related equipment; or (3) major deficiencies in design, construction, use of, or in management controls for, licensed facilities or material.

Examples of the types of events that are evaluated in detail using these criteria are:

<u>For All Licensees</u>

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual to 150 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms of any individual to 375 rems or more of radiation [10 CFR Part 20.403(a)(1)]; or equivalent exposures from internal sources.
2. An exposure to an individual in an unrestricted area such that the whole-body dose received exceeds 0.5 rem in one calendar year [10 CFR Part 20.105(a)].
3. The release of radioactive material to an unrestricted area in concentrations which, if averaged over a period of 24 hours, exceed 500 times the regulatory limit of Appendix B, Table II, 10 CFR Part 20 [10 CFR Part 20.403(b)].
4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as: (a) a radiation dose rate of 1000 millirems per hour three feet from the surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit [10 CFR Part 71.36(a)].
5. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.
6. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.
7. Any substantiated loss of special nuclear material or any substantiated inventory discrepancy which is judged to be significant relative to normally expected performance and which is judged to be caused by theft or diversion or by substantial breakdown of the accountability system.

8. Any substantiated breakdown of physical security or material control (i.e., access control, containment, or accountability systems) that significantly weakens the protection against theft, diversion, or sabotage.
9. An accidental criticality [10 CFR Part 70.52(a)].
10. A major deficiency in design, construction, or operation having safety implications requiring immediate remedial action.
11. Serious deficiency in management of procedural controls in major areas.
12. Series of events (where individual events are not of major importance), recurring incidents, and incidents with implications for similar facilities (generic incidents) which create major safety concern.

For Commercial Nuclear Power Plants

1. Exceeding a safety limit of license Technical Specifications [10 CFR Part 50.36(c)].
2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.
3. Loss of plant capability to perform essential safety function such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core-cooling system, loss of control rod system).
4. Discovery of a major condition not specifically considered in the Safety Analysis Report or Technical Specification that requires immediate remedial action.
5. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core-cooling system, loss of control rod system).

For Fuel Cycle Licensees

1. A safety limit of license Technical Specifications is exceeded and a plant shutdown is required [10 CFR Part 50.36(c)].
2. A major condition not specifically considered in the Safety Analysis Report or Technical Specifications that requires immediate remedial action.
3. An event which seriously compromises the ability of a confinement system to perform its designated function.

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16. ABSTRACT (200 words or less)

This report is the sixth in a series of reports issued annually that summarizes the operating experience of nuclear power plants in the United States. Power generation statistics, plant outages, reportable occurrences, fuel element performance, and occupational radiation exposure for each plant are presented and discussed and summary highlights are given. The report includes 1979 data from 67 plants - 25 boiling-water reactor plants, 41 pressurized water reactor plants, and 1 high-temperature gas-cooled reactor plant.

17. KEY WORDS AND DOCUMENT ANALYSIS

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