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Nuclear Power Plant Operating Experience - 1980

Annual Report

Prepared by G. T. Mays, J. A. Haried, C. Kukielka, R. D. Seagren

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory
Commission

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PREVIOUS REPORTS IN THIS SERIES

OOE-ES-004, Nuclear Power Plant Operating Experience During 1973, USAEC, December 1974

NUREG-0227, Nuclear Power Plant Operating Experience, 1974-1975, USNRC, April 1977

NUREG-0366, Nuclear Power Plant Operating Experience - 1976, USNRC, December 1977

NUREG-0483, Nuclear Power Plant Operating Experience - 1977, USNRC, February 1979

NUREG-0618, Nuclear Power Plant Operating Experience - 1978, USNRC, December 1979

NUREG/CR-1496; ORNL/NUREG/NSIC-180, Nuclear Power Plant Operating Experience - 1979, USNRC, May 1981

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1. INTRODUCTION

This report summarizes the operating experience of 67 licensed nuclear power plants during 1980. 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 authority to operate Three Mile Island 2 (TMI-2) was suspended by the Nuclear Regulatory Commission (NRC) on July 20, 1979. However, certain data on TMI-2 are included in this report.

At the end of 1980, there were 70 plants licensed to operate — 68 in commercial operation and 2 (Salem 2 and Sequoyah 1) in power ascension. Three plants were shut down for an indefinite period, with no decision yet made on future operation — Dresden 1, Humboldt Bay, and TMI-2.

The commercial operating experience of 67 plants is reviewed. Included are data for 24 boiling-water-reactor (BWR) plants, 42 pressurized-water-reactor (PWR) plants, and Fort St. Vrain, a plant equipped with a high-temperature gas-cooled reactor (HTGR). In comparison with the 1979 report (NUREG/CR-1496), Arkansas 2 and North Anna 2 have been added to the list of plants reviewed.

2. POWER GENERATION

Electrical Output for 1980

In 1980 the total net electrical output for 67 nuclear power plants in commercial operation was 251.1 billion kilowatt hours, which is 11.0% 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 power in 1980 represents a 0.3% decrease compared with the output for 1979. The TMI-2 accident and the regulatory restrictions resulting therefrom continued to impact operations for plants during the early part of 1980 and were a significant factor in the resulting decrease in the total net electrical energy output generated in 1980 versus 1979 from nuclear power plants. Of the total net electrical energy output of nuclear power plants in 1980, 63.0% was produced by PWRs, 36.7% by BWRs, and 0.3% by the HTGR.

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*See Appendix A for definition.

Plant Availability Factor for 1980

The average plant availability factor for all plants in 1980 was 65.9% for the 67 nuclear power plants in commercial operation. The average BWR and PWR availability factors for this period were 69.4 and 64.2%, respectively. The HTGR had an availability factor of 53.6%.

Plant Capacity Factors for 1980

Individual plant capacity factors were calculated using maximum dependable capacity (MDC)* and design electrical rating (DER),* both in megawatts electrical net (MWe net). The weighted average capacity factors for the 67 commercial nuclear power plants were 58.8% using MDC and 57.2% using DER. These values reflect the lower capacity factors of the HTGR, which were 23.3% using MDC and 23.3% using DER. The combined weighted average values for the BWR and PWR plants were 59.0% using MDC and 57.5% using DER.

3. PLANT OUTAGES

During 1980, the 24 operating BWRs experienced an average of 2677.8 h of outage time compared with an average of 3309.0 h for the 42 operating PWRs plus TMI-2. The percentage of forced outage time at BWRs was 17% compared with 29% at PWRs. The primary cause of forced outages at BWRs and PWRs was equipment failure.

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 PWRs. Regulatory restrictions were a significant cause for a large percentage of scheduled outages for PWRs as a result of continuing action taken with regard to certain aspects of the TMI-2 accident.

Fort St. Vrain, an HTGR, had an availability factor of 53.6%, having experienced 24 forced outages and 2 scheduled outages for a total outage time of 4077.4 h.

4. REPORTABLE OCCURRENCES

Licensee Event Reports

The 67 commercially operating plants covered in this report submitted 3394† Licensee Event Reports (LERs) during 1980, an increase of 520 over the 2874 submitted in 1979. Of these, 1401 were from the 25 BWR plants, 1917 were from the 41 PWR plants, and 76 were from the single HTGR.

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*See Appendix A for definition.

†This total includes LERs from Dresden 1, Humboldt Bay (BWRs), and Three Mile Island 2 (a PWR). See Sect. 1 for more information on these two plants.

Abnormal Occurrences

An abnormal occurrence is an incident or event that the NRC determines is significant from the standpoint of public health or safety. 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 1980, there were six 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 six abnormal occurrences are as follows:

AO 80-1 Occupational Overexposures to Skin and Extremities
AO 80-2 Transient Initiated by Partial Loss of Power
AO 80-5 Loss of Decay Heat Removal Capability
AO 80-6 Failure of Control Rods to Insert Fully During a Scram
AO 80-7 Failure of Saltwater Cooling System
AO 80-9 Significant Flooding of Reactor Containment Building

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) that 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 that 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] excluding TMI-2. Fuel performance has continually improved, yet deviations from the normal occur occasionally.

Specific Fuel-Related Incidents

There were six fuel-related incidents reported to the NRC in the Licensee Event Reports involving leaking fuel elements and cladding degradation; all are briefly described in this 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 69.8% of the total collective dose (man-rems) was incurred by contractor personnel at BWRs compared with 66.6% at PWRs. At PWRs, the largest portion (43.6%) of the collective dose (23,535 man-rems) was incurred by workers involved in special maintenance, while at BWRs the largest portion (42.7%) of the collective dose (27,878 man-rems) was incurred by workers involved in routine maintenance activities.

The average annual dose for individuals who received measurable exposures was 0.67 rems, remaining less than 1 rem as it has every year since 1972.

The total collective dose at light-water reactors for 1980 (53,796 man-rems) increased considerably over last year's value (39,759 man-rems) as it did the year before. Part of the increase could be due to modifications of Mark I toruses and the replacement of certain stainless steel components at BWRs. Also, the activities required by NRC bulletins may have caused an increase in the collective dose received by workers at several plants.

Reference

1. F. Garzarolli, R. von Jan, and H. Steahle, "The Main Causes of Fuel Element Failure in Water-Cooled Power Reactors," *At. Energy Rev.* 17(1), 31 (March 1979).

NUCLEAR POWER PLANT OPERATING EXPERIENCE — 1980

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ABSTRACT

This report is the seventh 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 1980 data from 67 plants — 24 boiling-water-reactor plants, 42 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 67 licensed nuclear power plants during 1980. 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 authority to operate Three Mile Island 2 (TMI-2) was suspended by the Nuclear Regulatory Commission (NRC) on July 20, 1979. However, certain data on TMI-2 are included in this report.

At the end of 1980, there were 69 plants licensed to operate — 67 in commercial operation and 2 (Salem 2 and Sequoyah 1) in power ascension. Three plants were shut down for an indefinite period, with no decision yet made on future operation — Dresden 1, Humboldt Bay, and TMI-2.

The commercial operating experience of 67 plants is reviewed. Included are data for 24 boiling-water-reactor (BWR) plants, 42 pressurized-water-reactor (PWR) plants, and Fort St. Vrain, a plant equipped with a high-temperature gas-cooled reactor (HTGR). In comparison with the 1979 report (NUREG/CR-1496), Arkansas 2 and North Anna 2 have been added to the list of plants reviewed. 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 the 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 ~ 12/31/80[a]

Plant name	Utility	Reactor type	NSSS[b]	Began commercial operation
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
Arkansas 1	Arkansas Power & Light Co.	PWR	BW	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

Table 1.1 (continued)

Plant name	Utility	Reactor type	NSSS[b]	Began commercial operation
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
Arkansas 2	Arkansas Power & Light Co.	PWR	CE	3/80
North Anna 2	Virginia Electric & Power Co.	PWR	W	12/80

[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 1980 is reviewed elsewhere in this report. Dresden 1 (shut down 10/31/78) and Humboldt Bay (shut down 7/2/76) 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 2.1–2.3 summarize the plant availability* and net electrical capacity factors* for the BWRs, PWRs, and HTGR, respectively, for 1980. Table 2.4 is a composite of the BWR and PWR power generation statistics for 1980. Similar information has been reported for the years 1973–1979 for the BWRs and PWRs.[1–6] This report also contains information on Fort St. Vrain, the only commercial HTGR plant in operation in the United States.

2.2 Electrical Output for 1980

In 1980 the total net electrical output for 67 nuclear power plants in commercial operation was 251.1 billion kilowatt hours, which is 11.0% of the total electrical energy generated in the United States for the year from all sources.[7] However, the total net electrical energy output generated by nuclear power in 1980 represents a 0.3% decrease in comparison with the output for 1979. The TMI-2 accident and the regulatory restrictions resulting therefrom continued to impact operations for plants during the early part of 1980 and were significant factors in the resulting decrease in the total net electrical energy output generated in 1980 versus 1979 from nuclear power plants. Of the total net electrical energy output of nuclear power plants in 1980, 63.0% was produced by PWRs, 36.7% by PWRs, and 0.3% by the HTGR.

2.3 Plant Availability Factors for 1980

The average plant availability factor for all plants in 1980 was 65.9% for the 67 nuclear power plants in commercial operation. The average BWR and PWR availability factors for this period were 69.4 and 64.2%, respectively. The HTGR had an availability factor of 53.6%.

The BWR availability factors range from 35.2 for Brunswick 2 to 93.3% for Dresden 2. The BWR reactors had availability factors of less than 50% while 13 reported availability factors of 70% or greater. Brunswick 2 and Oyster Creek had availability factors of 35.2 and 41.7%, respectively, resulting mainly from extended refueling and maintenance outages.

The PWR availability factors ranged from 0 for Three Mile Island 1 (TMI-1) to 96% for Calvert Cliffs 2. Eight PWR units had availability factors of less than 50% while 21 units had availability factors of 70% or greater. Three Mile Island 1 remained shut down by NRC order due to the accident at TMI-2. Beaver Valley 1 had an availability factor of 6.8%, resulting from extensive equipment modifications required by NRC

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*See Appendix A for definition.

Table 2.1. BWR power generation statistics for 1980 (24 plants)

BWR plants	Design electrical capacity (MWe net)	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age[a] (years)
				Using MDC	Using design MWe	
Big Rock Point	72	405,450	78.9	71.5	64.1	18.1
Browns Ferry 1	1,065	6,061,849	72.6	64.8	64.8	7.2
Browns Ferry 2	1,065	5,618,838	69.2	60.1	60.1	6.3
Browns Ferry 3	1,065	6,936,550	79.1	74.1	74.1	4.3
Brunswick 1	821	3,939,624	68.9	56.8	54.6	4.1
Brunswick 2	821	1,864,957	35.2	26.9	25.9	5.7
Cooper Station	778	3,788,053	71.1	56.4	55.4	6.6
Dresden 2	794	4,580,887	93.3	67.6	65.7	10.7
Dresden 3	794	4,329,608	71.8	63.8	62.1	9.4
Duane Arnold	538	2,796,975	73.5	61.8	59.7	6.6
FitzPatrick	821	4,334,505	70.2	60.1	60.1	5.9
Hatch 1	786	4,790,546	81.7	71.4	70.2	6.1
Hatch 2	784	3,644,977	60.0	53.7	52.9	2.3
La Crosse	50	214,545	68.6	50.9	48.8	12.7
Millstone 1	660	3,390,215	69.0	59.0	58.5	10.1
Monticello	545	3,453,799	78.3	73.4	72.1	9.8
Nine Mile Point	620	4,537,788	92.2	84.7	83.3	11.1
Oyster Creek	650	1,957,645	41.7	35.9	34.3	11.3
Peach Bottom 2	1,065	4,343,879	51.6	47.1	46.4	6.9
Peach Bottom 3	1,065	7,233,843	80.7	79.6	77.3	6.3
Pilgrim 1	655	3,044,484	56.4	51.7	52.9	8.5
Quad Cities 1	789	3,441,743	66.5	51.0	49.7	8.7
Quad Cities 2	789	3,614,427	62.5	53.5	52.2	8.6
Vermont Yankee	514	2,979,214	71.4	67.3	66.3	8.3
Total	17,606	91,304,401				
Average	734	3,804,350	69.4	60.1	58.9	8.2
Weighted average[b]				60.1	59.1	

[a]Computed from date of first electrical generation through December 31, 1980.

[b]Averages weighted by the design electrical capacity.

Table 2.2. PWR power generation statistics for 1980 (42 plants)

PWR plants	Design electrical capacity (MWe 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	3,781,602	63.7	51.5	50.6	6.4
Arkansas 2[b]	912	3,647,197	74.0	63.0	59.3	0.8
Beaver Valley 1	852	300,775	6.8	4.2	4.0	4.6
Calvert Cliffs 1	845	4,533,957	72.3	63.7	61.1	6.0
Calvert Cliffs 2	845	6,412,954	96.0	88.5	88.4	4.1
Cook 1	1,054	6,461,827	73.7	70.5	69.8	5.9
Cook 2	1,100	6,691,753	74.4	70.4	69.3	2.8
Crystal River 3	825	3,353,930	53.1	48.8	46.3	3.9
Davis-Besse 1	906	2,093,923	36.2	26.8	26.3	3.3
Farley 1	829	4,603,742	69.6	65.2	63.2	3.4
Fort Calhoun	457	2,010,662	60.4	49.2	49.2	7.4
Ginna	470	3,093,997	76.0	74.9	74.9	11.1
Haddam Neck	575	3,562,845	75.0	73.1	69.9	13.4
Indian Point 2	873	4,264,224	64.8	56.7	55.6	7.5
Indian Point 3	965	3,070,723	53.2	36.2	36.2	4.7
Kewaunee	535	3,631,892	82.1	79.2	77.3	6.7
Maine Yankee	825	4,404,138	72.2	61.9	60.8	8.1
Millstone 2	870	4,881,788	69.2	64.3	63.9	5.1
North Anna 1	907	5,631,557	86.5	75.4	70.7	2.1
North Anna 2[c]	907	349,644	95.5	90.1	89.2	<0.1
Oconee 1	887	5,116,510	75.6	67.7	65.7	7.7
Oconee 2	887	3,878,808	61.5	51.3	49.8	7.1
Oconee 3	887	5,217,839	73.1	69.1	67.0	6.3
Palisades	805	2,379,529	42.9	42.7	33.7	9.0
Point Beach 1	497	2,477,108	78.6	57.0	56.7	10.2
Point Beach 2	497	3,588,294	86.4	82.5	82.5	8.4
Prairie Island 1	530	3,106,335	78.2	70.3	66.7	7.1
Prairie Island 2	530	3,469,271	81.6	79.0	74.5	6.0

Table 2.2 (continued)

PWR plants	Design electrical capacity (MWe net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%) Using MDC	Plant capacity factor (%) Using design MWe	Plant age[a] (years)
Rancho Seco	918	4,415,236	60.4	57.6	54.8	6.2
Robinson 2	700	3,211,350	62.2	55.0	52.2	10.3
Salem 1	1,090	5,684,438	69.2	60.0	59.4	4.0
San Onofre 1	436	816,678	22.3	21.3	21.3	13.5
St. Lucie 1	802	5,199,590	77.5	76.2	73.8	4.7
Surry 1	822	2,473,025	44.9	36.3	34.3	8.5
Surry 2	822	2,241,883	35.8	32.9	31.0	7.8
Three Mile Island 1[d]	819	0	0.0	0.0	0.0	6.5
Trojan	1,130	6,073,440	72.5	64.0	61.2	5.0
Turkey Point 3	693	4,387,391	77.6	77.3	72.1	8.2
Turkey Point 4	693	3,854,024	69.5	67.9	63.3	7.5
Yankee-Rowe	175	291,967	22.0	19.0	19.0	20.1
Zion 1	1,040	6,514,861	81.6	71.3	71.3	7.5
Zion 2	1,040	5,278,833	66.7	57.8	57.8	7.0
Total	33,102	156,459,540				
Average	788	3,725,227	64.2	57.9	56.5	6.8
Weighted average[e]				58.4	56.6	

[a]Computed from date of first electrical generation through December 31, 1980.

[b]Data given are for the period March 26, 1980 (date when commercial operation began), through December 31, 1980.

[c]Data given are for the period December 14, 1980 (date when commercial operation began), through December 31, 1980.

[d]TMI-1 remained shut down during 1980 due to continuation of an NRC regulatory restraint order.

[e]Averages weighted by the design electrical capacity.

Table 2.3. HTGR power generation statistics for 1980 (1 plant)

HTGR plant	Design electrical capacity (MWe net]	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age[a] (years)
				Using MDC	Using design MWe	
Fort St. Vrain[b]	330	675,717	53.6	23.3	23.3	4.1

[a]Computed from date of first electrical generation through December 31, 1980.

[b]Fort St. Vrain is currently restricted to an electrical generating capacity of 231 MWe net pending resolution of in-core temperature fluctuations.

Table 2.4. Composite of BWR and PWR power generation statistics for 1980

Plants	Design electrical capacity (MWe net)	Electrical output [MWh(e) net]	Plant availability factor (%)	Plant capacity factor (%)		Plant age (years)
				Using MDC	Using design MWe	
24 BWRs	17,606	91,304,401	69.4	60.1 (60.1)[a]	58.9 (59.1)[a]	8.2
42 PWRs	33,102	156,459,540	64.2	57.9 (58.4)[a]	56.5 (56.6)[a]	6.8
Total	50,708	247,763,940				
Average	760	3,697,969				
Weighted average by plant			66.1	58.7	57.4	7.3
Weighted average by design electrical capacity				59.0	57.5	

[a]Average weighted by design electrical capacity.

Bulletins 79-02 and 79-14. Yankee-Rowe had an availability of 22%, which resulted mainly from turbine rotor repair and TMI-related modifications. A refueling outage and major repairs of the steam generators were responsible for San Onofre's availability factor of 22.3%. Completion of the steam generator repair on Surry 2, seismic modifications of pipe restraints, refueling, and maintenance outages accounted for Surry 2's availability of 35.8%. An availability factor of 36.2% at Davis-Besse 1 was primarily caused by maintenance, refueling, TMI modifications, and repair of a main coolant pump. Palisades had an availability of 42.9%, which resulted from seismic modifications to pipe hangers and TMI-related modifications. Surry 1 had an availability of 44.9%, resulting from seismic modifications to pipe hangers and both turbine and steam generator repair.

Arkansas Nuclear One Unit 2 and North Anna 2 began commercial operation in 1980. Arkansas Nuclear One Unit 2 began commercial operation on March 20 and had an availability of 74.0%. North Anna 2 began commercial operation on December 14 and had an availability of 95.5%.

2.4 Plant Capacity Factors for 1980

Individual plant capacity factors were calculated using maximum dependable capacity (MDC)* and design electrical rating (DER),* both in megawatts electrical net (MWe net). The weighted† average capacity factors for the 67 commercial nuclear power plants were 58.8% using MDC and 57.2% using DER. These values reflect the lower capacity factors of the HTGR which were 23.3% using MDC and 23.3% using DER. The combined weighted average values for the BWR and PWR plants were 59.0% using MDC and 57.5% using DER.

The weighted average capacity factors for the 24 BWRs were 60.1 and 59.1% using MDC and DER, respectively. The MDC capacity factors varied from 26.9 to 84.7%; the DER capacity factors ranged from 25.9 to 83.3%. Five BWRs had capacity factors below 50% using DER while five were above 70%.

The weighted average capacity factors for the 42 PWRs were 58.4 and 56.6% using MDC and DER, respectively. The MDC capacity factors varied from 0 to 90.1%; the DER capacity factors ranged from 0 to 89.2%. Eleven PWRs and MDC capacity factors were below 50% while 14 were above 70%. Using DER, 12 PWRs had capacity factors below 50% while 14 were above 70%.

Power generation information for 1980 is summarized in Tables 2.1–2.4. More detailed information on individual plants is presented in Appendix B. Tables 2.5–2.8 give the distributions of availability and capacity factors as a function of age. Availability and capacity factor distributions are given in Table 2.9.

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*See Appendix A for definition.

†The weighting of the average capacity factor is based on plant size in terms of design electrical capacity.

Table 2.5. BWR plant availability and capacity factors as a function of plant age for 1980[a]

Plant age group (years)	Number of plants in age group	Average availability factor[b] (%)	Average capacity factor[b] (%)
0-0.9	0		
1-1.9	0		
2-2.9	1	60.0	52.9
3-3.9	0		
4-4.9	2	74.6	65.6
5-5.9	2	52.7	43.0
6-6.9	6	70.5	61.6
7-7.9	1	72.6	64.8
8-8.9	4	63.9	54.3
9-9.9	2	74.4	66.2
10-10.9	2	82.3	62.4
11-11.9	2	66.3	58.2
12-16.9	1	68.6	48.8
17+	1	78.9	64.1

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

[b]Average weighted by design electrical capacity.

Table 2.6. PWR plant availability and capacity factors as a function of plant age for 1980[a]

Plant age group (years)	Number of plants in age group	Average availability factor[b] (%)	Average capacity factor[b] (%)
0-0.9	2[c]	84.7	74.3
1-1.9	0		
2-2.9	2	79.9	69.9
3-3.9	3	52.5	44.7
4-4.9	5	60.6	52.0
5-5.9	3	72.0	64.9
6-6.9	7	59.9	53.0
7-7.9	9	66.6	57.1
8-8.9	4	64.1	59.7
9-9.9	1	42.9	33.7
10-10.9	2	69.0	54.2
11-11.9	1	76.0	74.9
12-16.9	2	52.3	48.9
17-20.0	1	22.0	19.0

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

[b]Average weighted by design electrical capacity.

[c]Includes Arkansas Nuclear One Unit 2, which began commercial operation March 26, 1980, and North Anna 2, which began commercial operation on December 14, 1980.

Table 2.7. HTGR plant availability and capacity factors as a function of plant age for 1980[a]

Plant age group (years)	Number of plants in age group	Average availability factor (%)	Average capacity factor (%)
4.1	1	53.6	23.3

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

Table 2.8. Composite of BWR and PWR plant availability and capacity factors as a function of plant age for 1980[a]

Plant age group (years)	Number of plants in age group	Average availability factor (%)	Average capacity factor (%)
0–0.9	2	87.4	74.3
1–1.9	0		
2–2.9	3	73.2	64.2
3–3.9	3	52.5	44.7
4–4.9	7	64.6	55.9
5–5.9	5	64.3	56.1
6–6.9	13	64.8	57.0
7–7.9	10	67.2	57.9
8–8.9	8	64.0	57.0
9–9.9	3	63.9	45.4
10–10.9	4	76.7	58.3
11–11.9	3	69.5	63.8
12–16.9	3	57.6	48.9
17+	2	50.4	41.5

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

Table 2.9. Distribution of BWR and PWR plant availability and plant capacity factors for 1980[a]

	Number of plants		
	BWRs	PWRs	Total
Plants with availability factors (in percent) of			
90 and over	2	2	4
80–90	2	5	7
70–80	9	14	23
60–70	7	11	18
50–60	2	2	4
Less than 50	2	8	10
Total	24	42	66
Average availability factors, %	69.4	64.2	66.1
Plants with capacity factors (in percent) using MDC of			
90 and over	0	1	1
80–90	1	2	3
70–80	5	11	16
60–70	7	10	17
50–60	8	7	15
Less than 50	3	11	14
Total	24	42	66
Average capacity factors using MDC, % (weighted %[b])	60.1 (60.1)	57.9 (58.4)	58.7 (59.0)
Plants with capacity factors (in percent) using DER of			
90 and over	0	0	0
80–90	1	3	4
70–80	4	7	11
60–70	7	12	19
50–60	7	8	15
Less than 50	5	12	17
Total	24	42	66
Average capacity factors using DER, % (weighted %[b])	58.9 (59.1)	56.5 (56.6)	57.4 (57.5)

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

[b]Averages weighted by the design electrical capacity.

2.5 References

1. U.S. Atomic Energy Commission, Office of Operations Evaluation, *Nuclear Power Plant Operating Experience During 1973*, OOE-ES-004 (December 1974).

2. U.S. Nuclear Regulatory Commission, Office of Management Information and Program Control, *Nuclear Power Plant Operating Experience 1974–1975*, NUREG-0227 (April 1977).*

3. U.S. Nuclear Regulatory Commission, Office of Management Information and Program Control, *Nuclear Power Plant Operating Experience 1976*, NUREG-0366 (December 1977).*

4. M. R. Beebe, *Nuclear Power Plant Operating Experience — 1977: Annual Report*, NUREG-0483 (February 1979).†

5. M. R. Beebe, *Nuclear Power Plant Operating Experience — 1978: Annual Report*, NUREG-0618 (December 1979).†

6. R. L. Scott, D. S. Queener, C. Kukielka, *Nuclear Power Plant Operating Experience — 1979: Annual Report*, NUREG/CR-1496 (ORNL/NUREG/NSIC-180) (May 1981).

7. U.S. Department of Energy, Energy Information Administration, *Annual Report to Congress 1980, Vol. 2: Data*, DOE/EIA-0173(80)/2 (1981).

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*Available for purchase from the National Technical Information Service, Springfield, VA 22161.

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

3. PLANT OUTAGES

3.1 Introduction

A review of the plant outages that occurred during 1980 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 was also classified as a scheduled outage.

The tables in this chapter present plant outage data only for the 66 light-water-reactor (LWR) plants commercially operable in 1980 plus TMI-2. 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. When the outage data are reviewed, note 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 787 outages, requiring 210,633.2 h of shutdown time, reported by the 68 nuclear power plants in commercial operation during 1980. The 67 LWR plants accounted for 761 outages, requiring 206,555.8 h — an average of 35.1% for the year. Forced outage time for the LWRs averaged 9.2%, and scheduled outage time averaged 25.9%. The average total unit availability for the 67 LWRs was 64.9%.

Table 3.1 presents the 1980 performance data for BWRs and lists the systems and components involved in the major outages [i.e., outages lasting 5 d (120 h) or longer]. Table 3.2 presents similar information for PWRs. Seventeen major outages at BWRs involved the reactor coolant systems. Thirty-seven major outages at PWRs involved the steam and power conversion system.

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*See Appendix A for definition.

Table 3.1. Summary of BWR power plant outages during 1980

	Big Rock Point 1	Browns Ferry 1	Browns Ferry 2	Browns Ferry 3	Brunswick 1	Brunswick 2	Cooper	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
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	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	100	100	100	
Scheduled outages during commercial operation																									
Hours	1812.3	1906.3	1997.5	1216.5	1744.7	5103.1	2406.1	216.0	2311.8	1847.8	2361.0	13.5	2975.5	1856.4	2380.5	1339.4	280.6	4819.7	4067.4	816.3	3360.0	2749.3	2801.0	1563.4	51945.9
Percent	20.6	21.7	22.7	13.9	19.9	58.1	27.4	2.5	26.3	21.0	26.9	0.2	33.9	21.1	27.1	15.2	3.2	54.9	46.3	9.3	38.3	31.3	31.8	17.8	24.6
Forced outages during commercial operation																									
Hours	38.8	497.8	710.0	615.4	990.2	591.5	135.7	370.6	152.4	478.3	258.2	1593.0	534.7	901.1	13.2	567.5	405.3	300.7	186.6	874.7	469.6	191.8	496.6	946.8	12320.5
Percent	0.4	5.7	8.1	7.0	11.3	6.7	1.5	4.2	1.7	5.4	2.9	18.1	6.1	10.3	0.2	6.5	4.6	3.4	2.1	10.0	5.3	2.2	5.7	10.8	5.6
Total outage time during commercial operation																									
Hours	1851.1	2404.1	2707.3	1831.9	2734.9	5694.6	2541.8	586.6	2464.2	2326.1	2619.2	1606.5	3510.2	2757.5	2393.7	1906.9	685.9	5120.4	4254.0	1691.0	3829.6	2941.1	3297.6	2510.2	64266.4
Percent	21.0	27.4	30.8	20.9	31.2	64.8	28.9	6.7	28.0	26.4	29.8	18.3	40.0	31.4	27.3	21.7	7.8	58.3	48.4	19.3	43.6	33.5	37.5	28.6	30.5
Unit availability in commercial operation																									
Percent	78.9	72.6	69.2	79.1	68.9	35.2	71.1	93.3	71.8	73.5	70.2	81.7	60.0	68.6	72.7	78.3	92.2	41.7	51.6	60.7	56.4	66.5	62.5	71.4	
Systems and components causing major outages (figures indicate number of outages lasting 5 d or longer)																									
Fuel inspection or replacement	1	1	1	1	1	1	1		1	1	1		1	1	1	1		1	1		1	1	1	1	20
Engineered safety features			1									1	1						1	1					5
Reactor coolant system						1		1		1		2		2		3		1	1	3	1			1	17
Implement requirements of NRC	1									1			1	1											4
Electric power				1	1												1			2					
Control rods				1										1											2
Instrumentation and controls																	1								1
Main generator												1													1
Shock suppressors and supports												1			1										2
Steam and power conversion systems													1												1

Table 3.2. Summary of PWR power plant outages during 1980

	Arkansas 1	Arkansas 2	Beaver Valley 1	Calvert Cliffs 1	Calvert Cliffs 2	Cook 1	Cook 2	Crystal River 3	Davis-Besse 1	Farley 1	Fort Calhoun
Summary of performance data											
Percent of year operational	100	100	100	100	100	100	100	100	100	100	100
Percent of year in commercial operation	100	77	100	100	100	100	100	100	100	100	100
Scheduled outages during commercial operation											
Hours	1324.8	1235.4	7794.7	2075.2	261.3	2021.3	845.8	3779.8	5102.3	2273.3	3458.0
Percent	15.1	14.1	88.7	23.6	3.0	23.0	9.6	43.0	58.1	25.9	39.4
Forced outages during commercial operation											
Hours	1884.7	1902.5	388.2	358.1	94.1	288.8	1400.4	339.0	508.2	397.8	17.1
Percent	21.5	21.7	4.4	4.1	1.1	3.3	16.0	3.9	5.8	4.5	0.2
Total outage time during commercial operation											
Hours	3209.5	3137.9	8182.9	2433.3	355.4	2310.1	2246.2	4118.8	5610.5	2671.1	3475.1
Percent	36.6	35.8	93.1	27.7	4.1	26.3	25.6	46.9	63.9	30.4	39.6
Unit availability in commercial operation											
Percent	63.7	74.0	6.8	72.3	96.0	73.7	74.4	53.1	36.2	69.6	60.4
Systems and components causing major outages (figures indicate number of outages lasting 5 d or longer)											
Fuel inspection or replacement			1	1		1		1	1	1	1
Implement requirements of NRC	1			1	1		1			1	
Steam and power conversion system	1			1		1	1			1	
Instrumentation and controls	1							1			
Reactor coolant	2		1						1		
Electric power		3									
Auxiliary water systems		1									
Auxiliary process systems		1									
Engineered safety features						1					
Control rods											
Other											
Main generator							1				

	Ginna	Haddam Neck	Indian Point 2	Indian Point 3	Kewaunee	Maine Yankee	Millstone 2	North Anna 1	North Anna 2	Oconee 1	Oconee 2
Summary of performance data											
Percent of year operational	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	5	100	100
Scheduled outages during commercial operation											
Hours	2108.8	2124.4	719.7	2629.2	1095.8	1589.2	2651.2	604.8	968.5	1294.6	3272.0
Percent	24.0	24.2	8.2	29.2	12.5	18.1	30.2	6.9	11.1	14.7	37.2
Forced outages during commercial operation											
Hours	0.0	78.9	2373.1	1482.5	475.1	854.5	183.9	584.3	364.0	852.2	112.6
Percent	0.0	0.9	27.0	16.9	5.4	9.7	2.1	6.7	4.1	9.7	1.3
Total outage time during commercial operation											
Hours	2108.8	2203.3	3092.8	4111.7	1570.9	2443.7	2835.1	1189.1	1332.5	2146.8	3384.6
Percent	24.0	25.1	35.2	46.8	17.9	27.8	32.3	13.5	15.2	24.4	38.5
Unit availability in commercial operation											
Percent	76.0	75.0	64.8	53.2	82.1	72.2	69.2	86.5	95.5	75.6	61.5
Systems and components causing major outages (figures indicate number of outages lasting 5 d or longer)											
Fuel inspection or replacement	1	1		1	1	1	1	1		1	1
Implement requirements of NRC										2	2
Steam and power conversion system	1		1	3		1		1		2	
Instrumentation and controls											
Reactor coolant				1	1	1					
Electric power				1	1						
Auxiliary water systems											
Auxiliary process systems				1							
Engineered safety features			1	1			1		1		1
Control rods											
Other									1		
Main generator			1						1		

Table 3.2 (continued)

	Oconee 3	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	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	0	0	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	0	0	100	100	100	100	100	100	
Scheduled outages during commercial operation																						
Hours	1601.1	4549.0	1998.8	498.1	1732.8	1569.2	2873.0	1852.9	2342.1	2612.8	1478.0	4570.9	5542.7	8784.0	8784.0	1740.9	1558.8	2669.5	424.3	480.0	2244.5	100354.3
Percent	18.2	51.8	22.7	5.7	19.7	17.9	32.7	21.1	26.7	29.8	16.8	52.0	63.1	100.0	100.0	19.8	17.7	30.4	4.8	5.5	25.6	17.1
Forced outages during commercial operation																						
Hours	765.4	467.8	42.3	371.9	184.1	45.0	606.7	1463.3	363.8	3509.6	501.5	445.0	100.9			677.3	409.8	17.9	6425.6	1135.6	677.6	41935.1
Percent	8.7	5.3	0.5	4.2	2.1	0.5	6.9	16.7	4.1	40.0	5.7	5.1	1.2			7.7	4.7	0.2	73.2	12.9	7.7	7.1
Total outage time during commercial operation																						
Hours	2366.5	5016.8	2041.1	870.0	1916.9	1614.2	3479.7	3316.2	2705.9	6122.4	1979.5	5015.9	5643.6	8784.0	8784.0	2418.2	1968.6	2687.4	6849.9	1615.6	2922.1	142289.4
Percent	26.9	57.1	23.2	9.9	21.8	18.4	39.6	37.8	30.8	69.8	22.5	57.1	64.3	100.0	100.0	27.5	22.4	30.6	78.0	18.4	33.3	24.2
Unit availability in commercial operation																						
Percent	73.1	42.9	78.6	86.4	78.2	81.6	60.4	62.2	69.2	22.3	77.5	44.9	35.8	0.0	0.0	72.5	77.6	69.5	22.0	81.6	66.7	
Systems and components causing major outages (figures indicate number of outages lasting 5 d or longer)																						
Fuel inspection or replacement	1		1	1	1	1	1	1	1	1	1		1			1	1	1			1	31
Implement requirements of NRC	1	1							1	1		2		1					1			17
Steam and power conversion system	1	2	2	1	2	1	1	2		1		1			1		2	2	1	2	1	37
Instrumentation and controls																						2
Reactor coolant					1			1				1									1	12
Electric power		1																				6
Auxiliary water systems											1											2
Auxiliary process systems								1														3
Engineered safety features													1									7
Control rods																						0
Other																1					1	3
Main generator																						3

3.3 Types of Outages at LWRs

The data on forced and scheduled outages at BWRs and PWRs for plants in commercial operation in 1980 arc summarized in Table 3.3. The average number of forced outages was 8.6 per plant, with each outage averaging 94.2 h. The average number of scheduled outages was 2.7 per plant, with each one averaging 841.9 h (compared with 635 h in 1979 – an increase of 33%). On the average, each plant experienced 11.4 outages, totaling 270.4 h.

3.4 Proximate Causes of Plant Outages at LWRs

Plant outages at LWRs and their proximate 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 that caused a forced outage. Scheduled refuelings required the most outage time of all causes – 92,754.2 h (45%). Equipment failures (forced) accounted for 43,201.6 h (21%) of total outage time. Regulatory restrictions (forced and scheduled) accounted for 32,811.5 h (16%) of total outage time. This is a significant decrease from that accumulated in 1979 when 53,989 h (28%) of total outage time was for regulatory restrictions.

Although the number of LWR plants considered in this review increased by 2 (3%) from 1979 to 1980, the total outage time increased by 24,948.8 h (13%).

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 1980, 24 BWRs were commercially operable 100% of the year (8784 h); therefore, the total number of operating hours considered for BWRs was 210,816 h. For the PWRs, 41 units were commercially operable all year, one unit was commercially operable 77% of the year (6744 h), and one unit was commercially operable 5% of the year (432 h), giving a total of 367,320 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, other, and refueling.

Table 3.3. Summary of BWR and PWR nuclear power plant outages by type for 1980

Plant type (number of plants)	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 (24)	207	12,320.5	83	51,945.9	290	64,266.4
Average per BWR plant	8.6	513.4	3.5	2,164.4	12.1	2,677.8
Average outage duration per BWR plant		59.7		618.4		221.3
PWR plants (43)	372	41,935.1	99	100,354.3	471	142,289.4
Average per PWR plant	8.7	975.2	2.3	2,333.8	11.0	3,309.0
Average outage duration per PWR plant		112.1		1,014.7		300.8
All plants (67)	579	54,255.6	182	152,300.2	761	206,555.8
Average per plant	8.6	809.8	2.7	2,273.1	11.4	3,082.9
Average outage duration per plant		94.2		841.9		270.4

Table 3.4. Proximate causes of outages[a] of light-water-reactor units during 1980

Events	BWRs		PWRs		All plants[b]	
	Number of causes	Outage hours	Number of causes	Outage hours	Number of causes	Total outage hours
Forced outages						
Equipment failure	158	9,635.6	254	33,566.0	412 (49)	43,201.6 (21)
Maintenance or test	25	1,497.1	35	5,762.1	60 (7)	7,259.2 (4)
Regulatory restrictions			2	213.7	2 (<1)	213.7 (<1)
Operational error	23	487.2	56	584.3	79 (9)	1,071.5 (<1)
Administrative			3	153.4	3 (<1)	153.4 (<1)
Other	18	700.6	43	1,655.6	61 (7)	2,356.2 (1)
Scheduled outages						
Maintenance or test	29	2,405.2	60	16,297.3	89 (11)	18,702.5 (9)
Refueling	20	44,930.3	30	47,823.9	50 (6)	92,754.2 (45)
Regulatory restrictions	34	2,515.3	28	30,082.5	62 (7)	32,597.8 (16)
Administrative	1	160.5	3	391.6	4 (<1)	552.1 (<1)
Equipment failure	6	395.7	8	5,277.0	14 (2)	5,672.7 (3)
Other	4	1,538.9	3	482.0	7 (1)	2,020.9 (1)
Total	318	64,266.4	525	142,289.4	843 (100)	206,555.8 (100)

[a] There may be multiple causes for one event.

[b] Numbers in parentheses represent percentages of total.

Table 3.5. BWR and PWR outage ratios (outage hours per 100 h of commercial operation)

	Type of plant	
	BWR	PWR
Refueling	21.3	13.0
Equipment failure	4.8	10.6
Maintenance or test	1.9	6.0
Regulatory restriction	1.2	8.3
Operational error	0.2	0.2
Administrative	0.1	0.2
Other	1.1	0.6
Total	30.6	38.9

3.5 Systems and Components Associated with Plant Outages at LWRs

Graphic representations of plant outages are 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 are listed for major groupings. The system and component classifications used in these tables are listed in Appendix B.

The first four columns in each table are interrelated; for example, Table 3.6 shows that the Vermont Yankee plant accounted for 946.8 h (1%) of the forced outage time associated with valves, pumps, pipes, I&C, heat exchangers, or various other components in the reactor coolant system (RCS). 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 9,635.6 h of forced outage time experienced by all BWRs. This also represents 15% 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 19% of the total outage time at BWRs in 1980. Equipment failures accounted for 15% of the time while maintenance and testing accounted for 2%, other causes accounted for

Table 3.6. Boiling-water-reactor plant outages in 1980[a]

Outage type	Associated system	Associated component	Plants affected	Outage cause
		Valves	946.8 h Vermont Yankee 2%	
			878.9 h Hatch 1 1%	
		1,968.4 h 3%	768.3 h Various 1%	
			663.9 h Peach Bottom 3 1%	
	Reactor coolant	Pumps	600.8 h La Crosse 1%	
		1,568.5 h 2%	548.3 h Monticello 1%	
			417.2 h Duane Arnold 1%	
		945.9 h Pipes 2%	263.4 h Browns Ferry 1 <1%	
		651.1 h I&C 1%	219.1 h Brunswick 2 <1%	
		217.1 h Heat exchangers <1%	195.7 h Pilgrim 1 <1%	
			186.9 h Quad Cities 2 <1%	Equipment failure
	5,874.0 h 9%	523.0 h Various <1%	184.7 h Oyster Creek <1%	
		380.4 h Turbines <1%	496.1 h Various 1%	
Forced outages	Steam and power conversion	324.9 h Heat exchangers <1%	484.7 h Hatch 1 1%	
		230.5 h Valves <1%	392.1 h Hatch 2 1%	
		217.4 h Generators <1%	245.2 h Dresden 2 <1%	
		672.4 h Various 1%	106.3 h Browns Ferry 1 <1%	
	1,825.6 h 3%		101.2 h Pilgrim 1 <1%	
		755.4 h Conductors 1%	417.9 h Browns Ferry 3 1%	
	Electric power	430.0 h Transformers 1%	351.7 h Nine Mile Point <1%	
		215.2 h Circuit closures <1%	236.7 h Brunswick 1 <1%	
		142.6 h Generators <1%	169.6 h Peach Bottom 3 <1%	
		134.4 h Various <1%	143.9 h FitzPatrick <1%	
	1,677.6 h 3%		357.8 h Various <1%	9,635.6 h 15%
	Engineered safety features	346.5 h Pumps <1%	493.3 h Browns Ferry 2 1%	Maintenance and testing
		197.7 h Valves <1%	486.7 h Various 1%	
	980 h 1%+	435.8 h Various 1%		1,497.1 h 2%
	I&C	687.9 h I&C 1%	214.1 h Brunswick 2 <1%	Other
	912.1 h 1%	224.2 h Various <1%	698.0 h Various 1%	700.6 h 1%
	Various			Operator error
12,320.5 h 19%	1,051.2 h 1%+	1,051.2 h Various 1%	1,051.2 h Various 1%	487.2 h 1%
			4,808.5 h Brunswick 2 8%	
			4,641.0 h Oyster Creek 7%	
			3,482.7 h Peach Bottom 2 5%	
			3,256.6 h Pilgrim 5%	
			2,683.1 h Quad Cities 1 4%	
			2,665.8 h Quad Cities 2 4%	
		Fuel elements	2,309.0 h FitzPatrick 4%	
			2,270.9 h Cooper 4%	
			2,158.0 h Dresden 3 3%	
			2,132.7 h Millstone 1 3%	
			1,888.1 h Browns Ferry 1 3%	Refueling
	Reactor		1,882.0 h Browns Ferry 2 3%	
			1,648.0 h Duane Arnold 3%	
Scheduled outages			1,592.7 h Brunswick 1 3%	
			1,464.6 h Big Rock Point 1 2%	
			1,461.2 h Hatch 2 2%	
			1,399.5 h Vermont Yankee 1 2%	
			1,250.6 h La Crosse 2%	
			1,021.0 h Monticello 2%	
		44,930.3 h 70%	914.3 h Browns Ferry 3 1%	44,930.3 h 70%
			618.3 h Various 1%	Regulatory restriction
		Control rods	246.7 h Browns Ferry 3 <1%	
			163.2 h Hatch 2 <1%	2,515.3 h 4%
	46,145.4 h 72%	1,215.1 h 2%	134.9 h Duane Arnold <1%	Maintenance testing
	Engineered safety features	1,401.1 h Not applicable 2%	1,349.2 h Hatch 2 2%	
		472.0 h Valves 1%	767.0 h Various 1%	2,405.2 h 4%
	2,116.2 h 3%	243.1 h Various <1%		Other
	Reactor coolant	416.2 h Pumps 1%	1,357.4 h Various 2%	1,538.9 h 2%
		372.8 h Valves <1%		Equipment failure
	1,357.4 h 2%	568.4 h Various 1%		395.7 h 1%
	Various			Administrative
51,945.9 81%	2,326.9 h 4%	2,326.9 h Various 4%	2,326.9 h Various 4%	160.5 h <1%

[a] BWR plant outages totaled 64,266.4 h (100%).

Table 3.7. Pressurized-water-reactor plant outages in 1980[a]

Outage type	Associated system	Associated component	Plants affected	Outage cause
			8,784.0 h TMI-2 6%	
		Pumps	517.3 h Indian Point 3 <1%	
			162.5 h Palisades <1%	
		9,732.4 h 7%	268.6 h Various <1%	
			3,453.0 h San Onofre 2%	
			628.8 h Oconee 1 <1%	Equipment failure
		Heat exchangers	546.4 h Robinson 2 <1%	
			448.7 h Arkansas 1 <1%	
	Steam and power conversion	7,348.0 h 5%	2,271.7 h Various 2%	
			6,223.8 h Yankee Rowe 1 4%	
			531.7 h Rancho Seco 1 <1%	
		Turbines	167.6 h Calvert Cliffs 1 <1%	
		7,099.4 h 5%	176.3 h Various <1%	
Forced outages		Generators	1,270.3 h Cook 2 1%	
		1,644.4 h 1%	374.1 h Various <1%	
		1,065.3 h Valves 1%	1,065.3 h Various 1%	
		693.5 h I&C <1%	693.5 h Various <1%	
	28,612.8 h 20%	1,029.8 h Various 1%	1,029.8 h Various 1%	33,516.2 h 24%
		2,549.5 h Pumps 2%	2,549.5 h Various 2%	
	Reactor coolant	726.5 h Valves <1%	726.5 h Various <1%	Maintenance and testing
		565.6 h Heat exchangers <1%	565.6 h Various <1%	
	4,929.7 h 3%	1,088.1 h Various 1%	1,088.1 h Various 1%	5,762.1 h 4%
	Electric power 2,066.6 h 1%	2,066.6 h Various 1%	2,066.6 h Various 1%	Other 983.4 h 1%
	Engineered safety features 1,962.9 h 1%	1,962.9 h Various 1%	1,962.9 h Various 1%	N/A 754.9 h <1%
	Auxiliary water 1,139.2 h 1%	1,139.2 h Various 1%	1,139.2 h Various 1%	Operator error 551.4 h <1%
	Auxiliary process 1,099.2 h 1%	1,099.2 h Various 1%	1,099.2 h Various 1%	Regulatory restriction 213.7 h <1%
	I&C 868.2 h 1%	868.2 h Various 1%	868.2 h Various 1%	Administrative 153.4 h <1%
	N/A 754.9 h <1%	754.9 h Various <1%	754.9 h Various <1%	
41,935.1 h 29%	Various 501.6 h <1%	501.6 h Various <1%	501.6 h Various <1%	
			5,098.9 h Davis-Besse 4%	
			3,746.5 h Crystal River 3 3%	
			3,458.0 h Fort Calhoun 2%	
			2,836.0 h Rancho Seco 2%	
			2,240.7 h San Onofre 2%	
			2,040.4 h Haddam Neck 1%	
			1,905.3 h Salem 1 1%	
			1,850.6 h Robinson 2 1%	
			1,788.4 h Zion 2 1%	
			1,733.6 h Trojan 1%	
			1,632.3 h Calvert Cliffs 1 1%	
			1,624.1 h Cook 1 1%	
			1,538.5 h Millstone 2 1%	
	Reactor	Fuel elements	1,480.3 h Maine Yankee 1%	Refueling
			1,368.9 h Oconee 2 1%	
			1,339.7 h Ginna 1%	
			1,335.2 h St. Lucie 1 1%	
			1,296.0 h Farley 1 1%	
			1,288.0 h Prairie Island 1 1%	
			1,274.0 h Turkey Point 4 1%	
			1,148.1 h Prairie Island 2 1%	
			1,087.3 h Kewaunee 1%	
			1,005.7 h Indian Point 3 1%	
			877.4 h Turkey Point 3 1%	
			757.6 h Point Beach 1 1%	
			625.1 h Oconee 3 <1%	
			582.9 h North Anna 1 <1%	
			470.7 h Point Beach 1 <1%	
		47,823.9 h 34%	374.0 h Oconee 1 <1%	47,823.9 h 34%
	47,928.8 h 34%	104.9 h Vessels, pressure <1%	104.9 h Maine Yankee <1%	Other 104.9 h <1%
Scheduled outages			2,605.3 h Surry 1 2%	
		Heat exchangers	1,440.0 h Surry 2 1%	
		8,835.2 h 6%	4,789.9 h Various 3%	
			1,965.6 h Surry 1 1%	
	Steam and power conversion	Pipes, fittings	480.0 h Zion 1 <1%	
		3,303.5 h 2%	857.9 h Various 1%	
		Turbines	719.7 h Indian Point 2 1%	Regulatory restriction
		2,901.6 h 2%	2,181.9 h Various 2%	
		Mechanical function units 269.5 h <1%	269.5 h Various <1%	
		216.1 h Shock suppressors <1%	216.1 h Various <1%	
	15,619.2 h 11%	93.3 h Various <1%	93.3 h Various <1%	
			8,784.0 h Three Mile Island 1 6%	
	N/A	N/A		
		11,607.2 h 9%	976.0 h Oconee 3 1%	
	17,194.0 h 9%	586.8 h Various <1%	2,434.0 Various 2%	
		4,102.7 h Other 3%	4,102.7 h Surry 2 3%	30,063.1 h 21%
	Engineered safety features		3,543.5 h Palisades 3%	
		3,973.7 h Shock suppressors 3%	430.2 h Various <1%	Maintenance and testing
			1,111.4 h Millstone 2 1%	
		1,863.4 h Pipes, fittings 1%	752.0 h Oconee 2 <1%	
	10,841.9 h 8%	902.1 h Various 1%	902.1 h Various 1%	15,961.1 h 11%
	Other	Other	7,794.7 h Beaver Valley 1 5%	Equipment failure
		9,600.5 h 7%	1,805.8 h Various 1%	
	10,069.2 h 7%	468.7 h Pipes, fittings <1%	468.7 h Cook 2 <1%	5,181.2 h 4%
	Electric power 1,630.6 h 1%	1,630.6 h Various 1%	1,630.6 h Various 1%	Other 809.1 h 1%
	Reactor coolant 1,233.1 h 1%	1,233.1 h Various 1%	1,233.1 h Various 1%	Administrative 391.6 h <1%
100,354.3 h 71%	Various 837.5 h 1%	837.5 h Various 1%	837.5 h Various 1%	Operator training 19.4 h <1%

[a]PWR plant outages totaled 142,289.4 h (100%).

1%, and operator error accounted for 1%. The major system involved, accounting for 9% of the time, was the RCS.

Components requiring more significant amounts of time were valves — 1,968.4 h, pumps — 1,568.5 h, and pipes and pipe fittings — 945.9 h.

Scheduled outages. Scheduled outages at BWRs totaled 51,945.9 h (81%) of total BWR outage time. Refuelings accounted for 44,930.3 h (70%). 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.

3.5.2 Pressurized-water reactors

Forced outages. Forced outages accounted for 29% of the total PWR outage time in 1980 (i.e., 41,935.1 of 142,289.4 h). Most of the forced outage time was devoted to the steam and power conversion system (28,612.8 h) and the reactor coolant system (4,929.7 h). The dominant components were pumps, heat exchangers, and turbines.

Equipment failures accounted for 33,516.2 h in 1980, an increase of 5,128.3 h over 1979.

Scheduled outages. Scheduled outages in PWRs totaled 100,354.3 h (71%) of the total PWR outage time. The reactor system accounted for 47,928.8 h, of which 47,823.9 h was for refueling. Regulatory restrictions, accounting for 30,063.1 h, increased considerably from the 1979 total of 8,857 h. Maintenance and testing, accounting for 15,961 h, increased slightly from the 1979 total of 15,090 h.

3.5.3 Comments on BWR and PWR outages

Forced outages. Twenty-four BWR plants experienced 12,320.5 h of forced outage — an overall average of 492.8 h per plant. Forty-three PWR plants experienced 41,935.1 h of forced outage — an overall average of 975.2 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 that contributed 1% or more of the total outage time.

The components that contributed the most to forced outage time at BWRs were valves, accounting for 78.7 h per plant. At PWRs pumps accounted for 226.3 h per plant and heat exchangers accounted for 170.9 h per plant.

Scheduled outages. The 24 BWRs had 51,945.9 h of scheduled outage time for an average of 2,164.4 h per plant. The 43 PWRs accumulated 100,354.3 h for an average of 2,333.8 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 that 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 BWRs was greater than that at

Table 3.8. Components involved in forced outages

System	Components	BWRs (24)		PWRs (43)	
		Percent[a]	Average hours per plant	Percent[a]	Average hours per plant
Reactor coolant	Valves	3	79	1	17
	Pumps	2	63	2	59
	Pipes	1	38		
	I&C	1	26		
Steam and power	Turbines	1	15	5	165
	Pumps			7	226
	Heat exchangers			5	171
	Generators			1	38
	Valves			1	25
Electric power	Electrical conductors	1	30	1	23
	Transformers	1	17		
I&C	I&C	1	28		
Other	Other	1	21		
Engineered safety features	Heat exchangers			1	40

[a]Percent of forced-outage time.

Table 3.9. Components involved in scheduled outages

System	Components	BWRs (24)		PWRs (43)	
		Percent[a]	Average hours per plant	Percent[a]	Average hours per plant
Reactor	Fuel elements	62	1,796	34	1,112
	Control rods	2	49		
Steam and power conversion	Heat exchangers			6	205
	Pipes, fittings			2	77
	Turbines			2	67
System code not applicable	Not applicable	1	18	8	270
Engineered safety features	Other			3	95
	Shock suppressors			3	92
	Pipes			1	43
	Not applicable	2	56		
	Valves	1	19		
Other	Other	1	31	7	223
Electric power	Engines			1	28
Reactor coolant	I&C			1	22
	Pumps	1	17		
	Valves	1	15		

[a]Percent of scheduled-outage time.

PWRs, averaging ~685 h longer. Aside from fuel elements, control rods were the components commanding the most scheduled outage time at BWRs. At PWRs heat exchangers ranked second behind fuel elements, requiring 205.5 h per plant.

3.5.4 HTGR outage experience summary

The Fort St. Vrain unit was in commercial operation throughout 1980. The unit generated 675,717 MWh net. It had an availability factor of 53.6% and a unit capacity factor of 23.3% for both MDC and DER.

The unit experienced 24 forced outages, accounting for 17.9% of the operating period, and 2 scheduled outages, accounting for 28.5% of the 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 1980, the 24 operating BWRs experienced an average of 2677.8 h of outage time compared with an average of 3309.1 h for the 43 operating PWRs. The percentage of forced outage time at BWRs was 17% compared with 29% at PWRs. The primary cause of forced outages at both BWRs and PWRs was equipment failure. 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 TMI-2 accident and with regard to concern for seismic design deficiencies in safety-related piping.

Fort St. Vrain, an HTGR, had an availability factor of 53.6%, having experienced 24 forced outages and 2 scheduled outages for a total outage time of 4077.4 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 file for further analysis and evaluation and for public dissemination. The information reported in the LERs conveys, primarily, negative aspects of plant operations. An 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 in terms of 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, NW, Washington, DC. 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, DC, 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, 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

Licensee Event Reports 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 generally affect safety directly nor do they have an actual impact on or consequence for 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 February 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 follow.

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, (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 1980

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 4.1–4.8 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 LWR plants considered for review in this report with respect to LERs submitted 3318 LERs during 1980, an increase of 490 from the 2828 submitted in 1979. The 25 commercially operating BWRs plus Humboldt Bay submitted 1401, while the 42 commercially operating PWRs plus TMI-2 submitted 1917. Fort St. Vrain, the only HTGR unit, submitted 76 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

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 (1401)
Big Rock Point	1	3	26	0	12	0	3	0	0	0	0	0	0	0	45	3.2
Browns Ferry 1	5	15	14	10	11	0	3	0	7	0	5	3	0	4	83	5.9
Browns Ferry 2	6	18	6	19	1	1	0	0	2	2	1	0	0	0	57	4.1
Browns Ferry 3	5	26	5	6	9	1	0	0	2	0	0	0	0	1	58	4.1
Brunswick 1	9	10	32	25	5	0	2	0	0	0	0	2	0	1	86	6.1
Brunswick 2	11	24	34	25	1	0	7	2	0	1	0	2	1	0	108	7.7
Cooper Station	2	14	16	2	2	0	8	1	0	1	2	1	1	1	49	3.5
Dresden 1	0	1	0	0	0	0	0	0	1	2	0	0	0	0	4	0.3
Dresden 2	7	5	11	8	2	0	0	0	5	1	3	0	0	0	42	3.0
Dresden 3	1	10	18	10	2	0	0	0	0	0	0	1	0	0	42	3.0
Duane Arnold	3	18	22	7	11	0	1	0	0	0	0	0	2	0	64	4.6
FitzPatrick 1	1	8	30	24	5	0	2		8	0	3	2	0	10	93	6.6
Hatch 1	2	29	32	20	7	0	7	2	5	1	5	0	4	7	121	8.6
Hatch 2	7	36	48	24	13	0	6	3	5	0	4	0	1	5	152	10.8
Humboldt Bay	0	0	0	0	3	0	2	0	2	0	0	0	1	0	8	0.6
La Crosse	2	2	7	0	0	0	0	0	2	0	0	0	0	2	15	1.1
Millstone 1	1	4	4	4	0	0	0	0	1	0	1	0	0	1	16	1.1
Monticello	2	7	16	0	0	1	0	0	0	2	1	0	0	0	29	2.1
Nine Mile Point 1	3	3	1	2	2	0	0	0	2	5	2	0	1	4	25	1.8
Oyster Creek	5	4	25	10	2	0	1	0	4	0	0	2	0	2	55	4.0
Peach Bottom 2	1	8	14	5	2	0	0	0	2	2	0	1	0	0	35	2.5
Peach Bottom 3	0	4	14	4	1	0	0	0	0	1	0	1	0	0	25	1.8
Pilgrim	2	15	25	6	4	0	1	0	8	2	3	0	9	4	79	5.6
Quad Cities 1	1	5	10	5	4	0	0	0	0	0	2	0	1	1	29	2.1
Quad Cities 2	1	16	17	2	0	0	1	0	0	0	0	0	0	0	37	2.7
Vermont Yankee	1	9	12	5	3	1	0	1	0	0	2	4	1	4	43	3.1
Total	79	294	437	229	102	4	44	9	46	20	34	19	22	47	1401	100.0
Percent of 1401	5.6	21.0	31.2	16.3	7.3	0.3	3.1	0.6	3.3	1.4	2.4	1.4	1.4	1.6	3.4	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.

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 because these two systems 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 ECCS was involved in a larger number of occurrences, indicating the importance of this system and the attention it consequently receives.

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 (1917)
Arkansas 1	1	8	12	1	3	0	6	1	4	0	0	0	2	3	41	2.1
Arkansas 2	0	11	19	31	4	0	8	5	4	4	3	3	0	1	93	4.9
Beaver Valley 1	7	25	34	12	4	1	10	1	10	1	6	1	0	7	119	6.3
Calvert Cliffs 1	3	10	14	10	8	0	6	1	2	2	1	2	2	8	69	3.6
Calvert Cliffs 2	11	6	5	17	6	0	1	2	0	1	0	7	0	3	59	3.1
Cook 1	0	1	13	6	3	0	1	2	1	0	0	4	0	3	34	1.8
Cook 2	0	4	10	8	2	0	0	0	0	3	0	7	0	3	37	1.9
Crystal River 3	2	8	19	4	4	0	0	0	3	0	0	0	0	22	62	3.2
Davis-Besse 1	5	23	17	22	10	1	2	2	5	2	0	0	3	2	94	4.9
Fort Calhoun 1	0	2	13	7	5	0	0	5	2	0	0	1	0	0	35	1.8
Ginna	0	3	1	1	5	0	0	1	0	0	0	0	0	0	11	0.6
Haddam Neck	1	2	5	0	2	0	1	0	1	2	6	1	0	0	22	1.1
Indian Point 2	0	0	6	2	3	0	0	1	0	3	0	0	2	1	18	0.9
Indian Point 3	0	2	4	1	2	0	0	5	2	1	0	0	0	1	18	0 9
Farly 1	4	6	10	12	20	0	10	0	2	3	0	8	2	2	79	4.0
Kewaunee	0	6	8	6	6	1	7	5	3	1	0	1	0	0	44	2.3
Maine Yankee	2	5	6	4	2	0	0	0	0	1	0	0	0	0	20	1.0
Millstone 2	4	4	5	9	2	0	10	1	1	2	1	0	0	1	40	2.1
North Anna 1	9	12	32	22	7	0	2	3	3	6	2	5	2	3	108	5.6
North Anna 2	10	16	22	36	6	0	0	2	1	6	1	1	0	0	101	5.3
Oconee 1	2	2	4	3	6	0	9	3	5	2	0	0	1	3	40	2.1
Oconee 2	0	4	9	1	0	0	0	7	3	0	0	0	1	0	25	1.3
Oconee 3	0	3	2	1	4	0	0	3	5	1	1	0	0	0	20	1.0
Palisades	4	2	17	4	2	0	0	1	1	3	4	0	1	0	39	2.0
Point Beach 1	0	4	2	1	4	0	1	0	0	1	1	0	0	1	15	0.8
Point Beach 2	0	2	4	3	1	0	1	0	0	0	0	0	0	1	12	0.6
Prairie Island 1	0	5	3	4	7	0	1	0	4	0	0	0	0	1	25	1.3
Prairie Island 2	0	6	3	1	3	0	1	2	0	0	0	0	0	0	16	0.8
Rancho Seco	1	9	11	3	7	0	8	1	0	2	0	2	0	4	48	2.5
Robinson 2	0	4	5	10	2	0	1	3	3	1	0	0	0	0	29	1.5
Salem 1	3	4	16	12	2	2	5	3	11	1	1	4	1	4	69	3.6
San Onofre 1	4	10	5	2	4	0	5	0	0	1	0	0	0	2	33	1.7
St. Lucie 1	23	4	9	11	7	0	4	2	1	2	1	0	0	3	67	3.5
Surry 1	3	10	22	1	3	0	6	3	1	3	20	0	1	1	74	3.8
Surry 2	2	5	17	3	2	0	9	4	5	1	1	0	0	1	50	2.6
Three Mile Island 1	0	3	2	1	3	0	2	0	1	0	0	0	1	6	19	1.0
Three Mile Island 2	1	5	8	3	17	0	0	0	10	0	1	0	4	2	51	2.7
Trojan	2	8	6	3	4	1	1	1	5	0	0	0	1	2	34	1.8
Turkey Point 3	1	2	5	4	3	0	1	1	4	3	1	2	1	1	29	1.5
Turkey Point 4	0	2	1	0	0	0	3	1	3	3	0	1	0	1	15	0.8
Yankee Rowe	1	1	6	3	2	0	0	0	4	1	0	2	0	1	21	1.1
Zion 1	0	2	5	7	4	0	6	1	3	1	14	9	0	1	53	2.8
Zion 2	1	2	6	11	5	0	1	0	1	0	0	2	0	0	29	1.5
Total	107	253	422	303	196	6	129	73	115	64	65	63	25	95	1917	
Percent of 1917	5.6	13.2	22.0	15.8	0.2	0.3	6.8	3.8	6.0	3.3	3.4	3.3	1.3	5.0	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 1980

System	BWRs		PWRs	
	Number of reports	Percent of total reports	Number of reports	Percent of total reports
Reactor	79	5.6	107	5.6
Reactor coolant and connected systems	294	21.0	253	13.2
Engineered safety features	438	31.3	422	22.0
Instrumentation and controls	229	16.3	303	15.8
Electric power systems	102	7.3	196	10.2
Fuel storage and handling	4	0.3	6	0.3
Auxiliary water systems	44	3.1	130	6.8
Auxiliary process systems	9	0.6	73	3.8
Other auxiliary systems	60	4.3	115	6.0
Steam and power conversion systems	19	1.4	64	3.3
Radioactive waste management systems	38	2.7	65	3.4
Radiation protection systems	20	1.4	63	3.3
Other systems	18	1.3	25	1.3
System code not applicable[a]	47	3.4	95	5.0
Total	1401	100.0	1917	100.0

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

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 slightly less than half of the occurrences. Personnel error was the cause of 15.1% of the occurrences in 1980, increasing slightly from 14.9% in 1979.

Table 4.4. Systems and subsystems involved in light-water-reactor LERs for 1980[a]

System and subsystem	BWRs		PWRs		Total	
	Number of reports	Percent of total reports	Number of reports	Percent of total reports	Number of reports	Percent of total reports
Reactor	79	5.6	107	5.6	186	5.6
Reactor vessel internals	2	0.1	3	0.2	5	0.2
Reactivity control systems	61	4.4	78	4.1	139	4.2
Reactor core	16	1.1	26	1.4	42	1.3
Reactor coolant system and connected systems	294	21.0	253	13.2	547	16.5
Reactor vessels and appurtenances	2	0.1	16	0.8	18	0.5
Coolant recirculation systems and controls	31	2.2	45	2.3	76	2.3
Main steam systems and controls	14	1.0	19	1.0	33	1.0
Main steam isolation systems and controls	39	2.8	9	0.5	48	1.4
Reactor core isolation cooling systems and controls	61	4.3	2	0.1	63	1.9
Residual heat removal systems and controls	91	6.5	40	2.1	131	3.9
Reactor coolant cleanup systems and controls	23	1.6	23	1.2	46	1.4
Feedwater systems and controls	6	0.4	63	3.3	69	2.1
Reactor coolant pressure boundary leakage detection systems	15	1.1	12	0.6	27	0.8
Other coolant subsystems and their controls	12	0.9	24	1.3	36	1.1
Engineered safety features	438	31.3	422	22	860	25.9
Reactor containment systems	60	4.3	48	2.5	108	3.3
Containment heat removal systems and controls	27	1.9	55	2.9	82	2.5
Containment air purification and cleanup systems and controls	21	1.5	14	0.7	35	1.1
Containment isolation systems and controls	80	5.7	68	3.5	148	4.5
Containment combustible control systems and controls	61	4.4	15	0.8	76	2.3
Emergency core-cooling systems and controls	147	10.5	147	7.7	294	8.9
Control room habitability systems and controls	6	0.4	31	1.6	37	1.1
Other engineered safety feature systems and their controls	36	2.6	44	2.3	80	2.4
Instrumentation and controls	229	16.3	303	15.8	532	16.0
Reactor trip systems	71	5.1	140	7.3	211	6.4
Engineered safety feature instrument systems	82	5.9	72	3.8	154	4.6
Systems required for safe shutdown	11	0.8	3	0.2	14	0.4
Safety-related display instrumentation	38	2.7	26	1.4	64	1.9
Other instrument systems required for safety	24	1.7	31	1.6	55	1.7
Other instrument systems not required for safety	3	0.2	31	1.6	34	1.0
Electric power systems	102	7.3	196	10.2	298	9.0
Offsite power systems and controls	7	0.5	16	0.8	23	0.7
Onsite power systems and controls (ac)	22	1.6	48	2.5	70	2.1
Onsite power systems and controls (dc)	15	1.1	11	0.6	26	0.8
Onsite power systems and controls (composite ac and dc)	8	0.6	6	0.3	14	0.4
Emergency generator systems and controls	49	3.5	112	5.8	161	4.9
Emergency lighting systems and controls	0	0.0	0	0.0	0	0.0
Other electric power systems and controls	1	0.1	3	0.2	4	0.1
Fuel storage and handling systems	4	0.3	6	0.3	10	0.3
New-fuel storage facilities	0	0.0	0	0.0	0	0.0
Spent-fuel storage facilities	0	0.0	4	0.2	4	0.1
Spent-fuel-pool cooling and cleanup systems and controls	0	0.0	1	0.1	1	<0.1
Fuel handling systems	4	0.3	1	0.1	5	0.2

Table 4.4 (continued)

System and subsystem	BWRs		PWRs		Total	
	Number of reports	Percent of total reports	Number of reports	Percent of total reports	Number of reports	Percent of total reports
Auxiliary water systems	44	3.1	130	6.8	174	5.2
Station service water systems and controls	20	1.4	47	2.5	67	2.0
Cooling systems for reactor auxiliaries and controls	9	0.6	36	1.9	45	1.4
Demineralized water makeup systems and controls	3	0.2	4	0.2	7	0.2
Potable and sanitary water systems and controls	0	0.0	0	0.0	0	0.0
Ultimate heat sink facilities	6	0.4	15	0.8	21	0.6
Condensate storage facilities	3	0.2	11	0.6	14	0.4
Other auxiliary water systems and their controls	3	0.2	17	0.8	20	0.6
Auxiliary process systems	9	0.6	73	3.8	82	2.5
Compressed air systems and controls	3	0.2	1	0.1	4	<0.1
Process sampling systems	3	0.2	4	0.2	7	0.2
Chemical, volume control, and liquid poison systems and controls	2	0.1	67	3.5	69	2.1
Failed-fuel detection systems	0	0.0	0	0.0	0	0.0
Other auxiliary process systems and their controls	1	0.1	1	0.1	2	0.1
Other auxiliary systems	60	4.3	115	6.0	175	5.3
Air conditioning, heating, cooling, and ventilation systems and controls	10	0.7	31	1.6	41	1.2
Fire protection systems and controls	48	3.4	79	4.1	127	3.8
Communication systems	1	0.1	1	0.1	2	0.1
Other auxiliary systems and their controls	1	0.1	4	0.2	5	0.2
Steam and power conversion systems	19	1.4	64	3.3	83	2.5
Turbine generators and controls	2	0.1	8	0.4	10	0.3
Main steam-supply system and controls	6	0.4	19	1.0	25	0.8
Main condenser systems and controls	2	0.1	0	0.0	2	0.1
Turbine-gland-sealing systems and controls	0	0.0	0	0.0	0	0.0
Turbine bypass systems and controls	1	0.1	1	0.1	2	0.1
Circulating water systems and controls	5	0.4	2	0.1	7	0.2
Condensate cleanup systems and controls	3	0.2	0	0.0	3	0.1
Condensate and feedwater systems and controls	4	0.3	30	1.6	34	1.0
Steam generator blowdown systems and controls	0	0.0	4	0.2	4	0.1
Other features of steam and power conversion systems	0	0.0	0	0.0	0	0.0
Radioactive waste management systems	38	2.7	65	3.4	103	3.1
Liquid radioactive waste management systems	9	0.6	19	1.0	28	0.8
Gaseous radioactive waste management systems	7	0.5	12	0.6	19	0.6
Process and effluent radiological monitoring systems	22	1.6	33	1.7	55	1.7
Solid radioactive waste management systems	0	0.0	1	0.1	1	<1.0
Radiation protection systems	20	1.4	63	3.3	83	2.5
Area monitoring systems	4	0.3	16	0.1	20	0.6
Airborne radioactivity monitoring systems	16	1.1	47	2.5	63	1.9
Other systems	18	1.3	25	1.3	43	1.3
System code not applicable[b]	47	3.4	95	5.0	142	4.3
Total	1401	100.0	1917	100	3318	100.1

[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.

Table 4.5. LWR components reported in LERs for 1980[a]

Components	BWRs		PWRs[b]	
	Number of reports	Percent of total reports	Number of reports	Percent of total reports
Accumulators	1	0.1	44	2.3
Air dryers	4	0.3	0	0.0
Annunciator modules	4	0.3	1	0.1
Batteries and chargers	14	1.0	10	0.5
Blowers	6	0.4	12	0.6
Circuit closers/interrupters	56	4.0	87	4.5
Component code not applicable[c]	133	9.5	244	12.7
Control rod drive mechanisms	7	0.5	18	0.9
Control rods	8	0.6	10	0.5
Demineralizers	2	0.1	1	0.1
Electrical conductors	20	1.4	19	1.0
Engines, internal combustion	21	1.5	55	2.9
Filters	8	0.6	13	0.7
Fuel elements	18	1.3	11	0.6
Generators	12	0.9	16	0.8
Hangers, supports, shock suppressors	58	4.1	85	4.4
Heat exchangers	27	1.9	58	3.0
Heaters, electric	1	0.1	21	1.1
Instrumentation and controls	483	34.5	489	25.5
Mechanical function units	13	0.9	15	0.8
Motors	9	0.6	35	1.8
Other components	23	1.6	58	3.0
Penetrations, primary containment	16	1.1	41	2.1
Pipes and/or fittings	49	3.5	72	3.8
Pumps	53	3.8	119	6.2
Recombiners	1	0.1	0	0.0
Relays	38	2.7	58	3.0
Transformers	5	0.4	12	0.6
Turbines	8	0.6	16	0.8
Valve operators	91	6.5	66	3.4
Valves	203	14.5	222	11.6
Vessels, pressure	9	0.6	9	0.5
Total	1401	100	1917	99.8

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

[b]LERs for Arkansas Nuclear One Unit 2 and North Anna 2 include those filed prior to their commercial operation for 1980.

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

Table 4.6. LERs submitted by light-water-reactor plants in 1980 arranged by cause, method of discovery, and reactor status at time of occurrence[a]

	BWRs		PWRs		BWRs and PWRs	
	Number of reports	Percent of BWR reports	Number of reports	Percent of PWR reports	Total reports	Percent of total reports
Approximate cause						
Component failure	749	53.5	859	44.8	1608	48.5
Defective procedures	47	3.4	105	5.5	152	4.6
Design/fabrication error	139	9.9	255	13.3	394	11.9
External cause	16	1.1	27	1.4	43	1.3
Other	265	18.9	354	18.5	619	18.7
Personnel error	184	13.1	317	16.5	501	15.1
Unknown	1	0.1	0	0.0	1	0.0
Total	1401	100.0	1917	100.0	3318	100.0
Method of discovery						
External source	53	3.8	103	5.4	156	4.7
Item not applicable	26	1.9	44	2.3	70	2.1
Observation/evaluation	0	0.0	0	0.0	0	0.0
Operational event	494	35.3	887	46.3	1381	41.6
Routine test or inspection	744	53.1	747	38.9	1491	44.9
Special dosimeter report	0	0.0	0	0	0	0.0
Special test or inspection	84	6.0	136	7.1	220	6.6
Total	1401	100.1[b]	1917	100.0	3318	99.9[b]
Reactor status at time of occurrence						
Construction	1	0.1	5	0.3	6	0.2
Item not applicable	7	0.5	24	1.3	31	0.9
Load change during power operation	64	4.6	42	2.2	106	3.2
Other	19	1.4	121	6.3	140	4.2
Preoperational, startup, power ascension	2	0.1	101	5.3	103	3.1
Refueling	225	16.1	226	11.8	451	13.6
Routine shutdown operations	41	2.9	35	1.8	76	2.3
Routine startup operations	97	6.9	114	5.9	211	6.4
Shut down except for refueling	155	11.1	230	12.0	385	11.6
Steady-state power operation	790	56.4	1019	53.1	1809	54.5
Undetermined	0	0.0	0	0.0	0	0.0
Total	1401	100.1	1917	100.0	3318	100.0

[a]LERs for Arkansas Nuclear One Unit 2 and North Anna 2 include those filed prior to their commercial operation during 1980.

[b]Numerical deviations are due to rounding off of numbers.

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. From 1977 through 1979 a steady decline in personnel errors (as a percentage of the 12-year total) is noted; however, the 501 events reported as personnel errors during 1980 represent an increase (14.8%) as compared to this three-year time period. This is most likely due to an ever-continuing concern and awareness of personnel errors

Table 4.7. Personnel errors vs system for light-water-reactor plants in 1980

System	BWR		PWR		BWRs and PWRs	
	Number of reports	Percent of BWR reports	Number of reports	Percent of PWR reports	Total reports	Percent of total reports
Reactor	19	10.3	19	6.0	38	7.6
Reactor coolant and connected systems	29	15.8	32	10.1	61	12.2
Engineered safety features	40	21.7	74	23.3	114	22.7
Instrumentation and controls	28	15.2	33	10.4	61	12.2
Electric power systems	11	6.0	45	14.2	56	11.2
Fuel storage and handling	0	0.0	0	0.0	0	0.0
Auxiliary water systems	2	1.1	19	6.0	21	4.2
Auxiliary process system	0	0.0	9	2.8	9	1.8
Other auxiliary systems	17	9.2	27	8.5	44	8.8
Steam and power conversion systems	2	1.1	10	3.2	12	2.4
Radioactive waste management systems	7	3.8	15	4.7	22	4.4
Radiation protection systems	2	1.1	7	2.2	9	1.8
Other systems	3	1.6	3	0.9	6	1.2
Not applicable	24	13.0	24	7.6	48	9.6
Total	184	99.9[a]	317	99.9[a]	501	100.1[a]

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

Table 4.8. Personnel errors at light-water-reactor plants for the years 1969 through 1980[a]

System	Number of personnel errors												System totals[a]	Percent of system totals
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980		
Reactor	0	2	2	8	16	27	26	36	31	21	40	38	247	7.3
Reactor coolant and connected systems	2	4	9	16	34	39	73	61	85	56	60	61	500	14.8
Engineered safety features	1	5	11	16	42	80	104	96	115	118	99	114	801	23.7
Instrumentation and controls	0	1	0	6	20	31	28	40	63	60	41	61	351	10.4
Electric power systems	0	2	6	8	13	30	32	42	48	42	42	56	321	9.5
Fuel storage and handling	2	0	0	3	6	6	4	5	4	6	4	0	40	1.2
Auxiliary water systems	0	0	1	3	1	9	15	22	23	11	13	21	119	3.5
Auxiliary process systems	0	1	2	2	12	19	16	19	19	23	13	9	135	4.0
Other auxiliary systems	0	0	0	0	0	3	3	5	8	33	35	44	131	3.9
Steam power and conversion systems	0	0	3	9	13	26	18	11	20	13	4	12	129	3.8
Radioactive waste management systems	0	2	6	7	17	40	46	28	29	11	15	22	223	6.6
Radiation protection system	0	0	0	0	1	2	3	7	8	14	8	9	52	1.5
Other systems	0	0	0	0	3	1	2	6	14	18	5	6	55	1.6
System code not applicable[b]	1	2	2	2	8	3	27	42	53	51	43	48	282	8.3
Total (by year)	6	19	42	80	186	316	397	420	520	477	422	501	3386	100.1[c]
Percent of 12-year total	0.2	0.6	1.2	2.4	5.5	9.3	11.7	12.4	15.4	14.1	12.5	14.8	100.1[b]	

[a]These totals include LERs for Arkansas Nuclear One Unit 2 and North Anna 2 prior to their commercial operation during 1980.

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

[c]Numerical deviations due to rounding off of numbers.

since Three Mile Island. The errors listed for "system code not applicable" (8.3%) 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 76 LERs in 1980. The number of LERs vs the system involved in the reported occurrences were as follows:

System	Number of LERs	Percent of total
Reactor coolant	29	38.2
Electric power	4	5.3
Steam and power conversion	10	13.2
System code not applicable	4	5.3
Other auxiliary systems	8	10.5
Engineered safety features	0	0.0
Fuel storage and handling	1	1.3
Instrumentation and controls	9	11.8
Reactor	2	2.6
Other major systems	9	11.8
Total	76	100.0

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

Components	Number of LERs	Percent of total
Blowers	1	1.3
Component code not applicable	12	15.8
Filters	1	1.3
Generators	1	1.3
Hanger, supports, shock suppressors	9	11.8
Heat exchangers	1	1.3
Instrumentation and controls	24	31.6
Mechanical function units	2	2.6
Other components	4	5.3
Pipes, fittings	4	5.3
Pumps	1	1.3
Relays	1	1.3
Valve operators	1	1.3
Valves	12	15.8
Vessels, pressure	2	2.6
Total	76	99.9*

*Total does not equal 100% because of rounding numbers to the nearest tenth of a percent.

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

Cause	Number of LERs	Percent of total
Component failure	36	47.4
Personnel errors	6	7.9
Design or fabrication error	7	9.2
Defective procedure	1	1.3
Other	26	34.2
Total	46	100.0

4.2.5 Operational events acted on 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). 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 (I&E) 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 were 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 has determined 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 1980, six 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 Occupational overexposure to the skin and extremities (AO 80-1) (Refs. 2 and 3)

On August 28, 1979, six individuals, including a contractor health physics foreman, entered the north makeup valve room in the TMI-2 Fuel Handling Building to inspect and tighten valves leaking highly contaminated reactor coolant. A stay time limit of 4 min in areas not exceeding 15 rem/h of gamma radiation was computed from data gathered using a portable survey instrument. The survey identified gamma radiation dose rates of 10–15 rem/h, generally; however, the beta radiation was grossly underestimated because the survey instrument was designed to operate in beta fields of no more than 2 rem/h. The actual beta exposure rate was 2500 rads/h. The overexposure to beta radiation was determined the following day from thermoluminescent dosimeters worn by the six individuals. Exposure limits per calendar quarter set by the NRC are 7.5 rem to the skin of the whole body and 18.75 rem to the hands and forearms, feet, and ankles. Table 4.9 summarizes the exposures to these workers for the third calendar quarter of 1979.

Table 4.9. Exposure summary

Individual	Skin dose[a] (rems)	Ratio (skin dose/limit)	Hand dose (rems)	Ratio (hand dose/limit)
a	166	22.1	82	4.4
b	161	21.5	38	2.0
c	40	5.3	8	0.4
d	29	3.9	6	0.3
e	26	3.6	16	0.9
f	13	1.7	13	0.7

[a]Since the legs constitute a major portion of the body, leg skin exposures are considered whole body skin exposures.

Metropolitan Edison (Met-Ed) health physics personnel have been retrained in the use and limitation of their radiation survey instruments and in the proper planning and preparation for jobs. Improved survey instruments have been obtained. Personnel dosimetry practices have been upgraded, and appropriate protective clothing requirements have been specified for areas where there are significant beta radiation dose rates.

The NRC has reviewed Met-Ed's health physics program and directed that improvements be made on each of the items identified as a cause of the occurrence. Onsite NRC inspectors are reviewing and observing licensee activities daily on every operating shift to ensure that any necessary corrective action is taken. Enforcement action concerning these overexposures is pending, and the onsite NRC inspectors are reviewing licensee progress in upgrading of the Met-Ed health physics program to prevent recurrence. Generic aspects of the event are under review to assess possible inadequacies of present practices and regulatory requirements for occupational radiation monitoring in postaccident plant environments.

4.3.2 Transient initiated by partial loss of power (AO 80-2) (Ref. 2)

A short to ground in a +24 V nonnuclear instrumentation (NNI power supply) initiated an RCS transient that resulted in the discharge of 43,000 gal of primary coolant into the Crystal River 3 containment building.

The loss of NNI affected automatic plant control systems and about 70% of NNI control board indicators (such as RCS temperature, pressure, and flow; steam generator pressure and level; and pressurizer level). It caused the pressurizer pressure-operated relief valve (PORV) and the pressurizer spray valve to open. The failure also caused false control signals to be sent to the Integrated Control System (ICS), the most significant of which caused a reduction in feedwater flow to the steam generators. Also, the false T_{ave} signal caused the ICS to withdraw the control rods to increase power.

The reduction in feedwater flow reduced the reactor heat removal rate to below the reactor heat generation rate, which caused RCS temperature and pressure to increase in spite of the open PORV and spray valve. As a result, the reactor tripped on high pressure and was subsequently partially depressurized. The operators secured the reactor coolant pumps as required by emergency procedures. High-pressure injection (HPI) was automatically initiated as a result of RCS depressurization due to loss of coolant inventory through the open PORV and the cooling effects associated with the reactor trip. Shortly after receipt of a high reactor coolant drain tank level alarm, the PORV block valve was closed and, with approximately 70% of NNI inoperable or inaccurate, the operator correctly decided that there was insufficient information available to justify terminating HPI. Therefore, the RCS and pressurizer were filled solid, causing RCS pressure to increase to the point where one safety valve lifted, and flow through the safety valve spilled water into the containment through the reactor coolant drain tank rupture disk.

Power was restored to the NNIs about 20 min after the start of the transient. Plant conditions then included the pressurizer filled solid with water, reactor coolant pressure of 2400 psig, a reactor coolant outlet temperature of 556°F, steam generator "A" dry, and the core being cooled by water flow from the high-pressure injection system out the open safety valve and by natural circulation through steam generator "B."

After the restoration of power to the instrumentation, the operators throttled HPI to reduce the flow of water through the open safety valve and into the reactor building. The operators also reestablished the water level in steam generator "A."

About 41 min after the transient began, the licensee declared a Class "B" Emergency based on the fact that coolant was being discharged through the open safety valve and HPI had been automatically initiated. All nonessential site personnel were evacuated and offsite agencies notified.

Actions taken to prevent recurrence included:

1. complete testing and inspection of the NNI system for similar failures,
2. installation of new redundant channels for indication of 23 key plant parameters to provide more reliable information to the operator,
3. comprehensive operator training in response actions for NNI and ICS failures,
4. installation of positive position indication on the PORV and the two-code safety valves,
5. modification of the NNI power supply to provide more reliable power,
6. evaluation of NNI power supply reliability in response to I&E Bulletin 79-27 *(Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation)*, and
7. modification of the control circuitry for the PORV and pressurizer spray valves so that the valves will not open in the event of loss of NNI power.

In addition, the Director, Office of Nuclear Reactor Regulation, established a special task force (i.e., Babcock & Wilcox (B&W) Reactor Transient Response Task Force) on March 12, 1980, to assess the generic aspects of operating experiences of the B&W plants. The task force published their findings in NUREG-0667, *Transient Response of Babcock & Wilcox-Designed Reactors*, dated May 1980. This assessment included consideration of the apparent sensitivity of the B&W plants to transients involving overcooling and undercooling conditions, small-break loss-of-coolant accidents, and the consequences of malfunctions and failures of the ICS and NNI.

4.3.3 Loss of decay heat removal capability (AO 80-5) (Ref. 3)

On April 8, 1980, Davis-Besse 1 was placed in cold shutdown for refueling, maintenance, and modifications. Since the plant was in the refueling mode, many systems or components were out of service for maintenance or testing while others were deactivated to preclude inadvertent actuation. This included draining decay heat loop 1 and energizing channels 1 and 3 of the reactor protection system (RPS) and the safety features actuation system (SFAS) from one electrical source. On April 19, the feeder breaker in the switchgear supplying the source of power to the RPS and SFAS tripped, apparently due to mechanical vibration or from being bumped by construction workers who were working in the area. Since the SFAS logic at Davis-Besse is two out of four taken once, loss of power to input channels 1 and 3 resulted in actuation of all five levels of the SFAS output channels. This precipitated the following sequence of events. Level 1 SFAS actuation closed containment isolation valve DH-12, which caused decay heat pump No. 2 to lose suction. It was then automatically aligned to the borated water storage tank (BWST) in the low-pressure injection mode, SFAS level 3. Actuation of SFAS level 5, however, implies a low level in the BWST; therefore, ECCS operation was automatically transfered from the injection mode to the recirculation mode. This involved closing the supply valve from the BWST and opening the valve to the containment emergency sump, which was dry. The operator secured the No. 2 decay heat pump to prevent its damage from loss of suction. This sequence resulted in a loss of decay heat removal for ~2 1/2 h.

During the time of the event, the reactor coolant temperature increased from 90 to about 170°F (the Technical Specification definition for refueling mode is an average temperature of <140°F); however, the final temperature reached was still considerably below that which could adversely affect the heat transfer characteristics of the fuel such that fuel damage could result. There were no offsite releases of radioactivity, and there were no overexposures or injuries to personnel associated with the event.

To prevent recurrence, the licensee: (1) closed and electrically disabled the isolation valves to the containment emergency sump; (2) kept second decay heat loop in standby until the refueling canal was filled; and (3) reviewed future electrical distribution system maintenance, modification, and testing to provide maximum diversity to the 120-V ac instrument power buses. Appropriate operating procedures were modified.

Long-term corrective actions were taken by the licensee in accordance with the NRC's I&E Bulletin 80-12:

1. Additional revisions were made to EP 1202.32, Loss of DHR Frequency Procedure, to include alternate methods to those previously listed to supply water to the reactor core and reference to appropriate procedures for monitoring core temperatures using the in-core thermocouples.
2. Additional guidance was provided for venting the DHR if air is drawn into the system.
3. Five procedures were revised to ensure that power is removed for emergency sump isolation valves DH-9A and 9B in Modes 5 and 6.
4. SP 1107.09, Instrument AC System Procedure, was revised to allow the 120-V ac instrument power inverter to be supplied from the dc bus when normal ac feed is not available. This will minimize the possible loss of power to two instrument channels at one time.
5. A special procedure was written to require, whenever possible, that redundant decay heat system not be intentionally removed from service in Modes 4, 5, or 6 unless at least one steam generator is available for decay heat removal, the refueling canal is filled, or the decay heat pump can be restored to service or a gravity flow path to the RCS can be established within 4 h. The special order also covers expediting the restoration of redundant or diverse methods if component failure causes loss of alternate decay heat removal methods.

4.3.4 Failure of control rods to insert fully during a scram (AO 80-6) (Ref. 3)

On June 28, 1980, Browns Ferry 3, BWR, reported that 76 out of 185 control rods failed to fully insert during a routine shutdown by a manual scram at about 35% power. The partially inserted rods were all (with one exception) on the east side of the core where reactor power level was indicated to be 2% or less. The west side of the core was subcritical. A second manual scram was initiated 6 min later, and all partially inserted rods were observed to drive inward, but 59 remained partially withdrawn.

A third manual scram was initiated 2 min later, and 47 rods remained partially withdrawn. Six min later, an automatic scram occurred and all the rods inserted fully when the scram discharge level bypass switch was returned from "bypass" to "normal" and there was a high water level in the scram discharge instrument volume. It appears that this was a coincidence in that a manual scram would probably have produced the same result. Core coolant flow, temperature, and pressure remained normal for the existing plant conditions.

The problem has been determined to be hydraulic in nature rather than electrical or mechanical. The control rod drives (CRDs), which insert and withdraw the attached control rods in a General Electric BWR, are essentially water-driven hydraulic pistons. On a scram, a relatively high water pressure is applied to the bottom side of the piston by opening a scram inlet valve; a scram outlet valve opens to relieve water and pressure above the piston, and the rods are rapidly driven up into the reactor core. Water discharged from the 185 individual CRDs during scram insertion is collected in two separate headers consisting of a series of interconnected 6-in.-diam pipes (four on each side of the reactor) called the scram discharge volume (SDV). During normal operation, both SDVs are designed to remain empty by being continuously drained to a separate scram discharge instrument volume (SDIV) tank. The SDVs are therefore normally ready to receive the scram discharge water when a scram occurs. This instrumented tank is monitored for water level and initiates an automatic scram on high level, in anticipation of too much water in the SDV preventing a scram. The CRDs at Browns Ferry 3 are grouped in such a manner that the east and west sides of the reactor core are connected to separate SDVs. Later tests, inspections, and analyses resulted in the conclusion that the east SDV was substantially full of water at the time of the event, leaving insufficient room for the discharge water. Accordingly, upon scram actuation, the CRDs rapidly drove the control rods partially into the core but rod motion prematurely ceased when pressure quickly equalized on each side of the pistons.

There was no danger to the general public or plant employees as a result of this event. No radioactivity was released to the environment. There was no indication of fuel damage.

The unit remained shut down while a series of tests was performed in an attempt to determine the cause of the water accumulation in the SDV. Ultrasonic probes were installed on the SDVs to continuously monitor water level. An evaluation team, consisting of the director and specialists of Region II and NRC headquarters personnel, was assembled at the site to evaluate the significance of this event. On July 3, 1980, I&E Bulletin 80-17 (Ref. 4) was issued to all licensees operating BWRs and required them to (1) conduct prompt and periodic inspections of the SDV, (2) perform two reactor scrams within 20 d while monitoring pertinent parameters to further confirm operability, (3) review emergency procedures to ensure pertinent requirements are included, and (4) conduct additional training to acquaint operating personnel with this type of problem. On July 18, 1980, Supplement 1 to Bulletin 80-17 (Ref. 5) was issued to all licensees operating BWRs. This supplement required an analysis of the "as built" SDV, revised procedures on initiation of the standby liquid control system, specified action to be taken if water is found in the SDV, required daily monitoring of the SDV until a continuous monitor can be installed, and recommended studying of designs to improve the venting of the SDV.

Based on the responses from Supplement 1, Supplement 2 to I&E Bulletin 80-17 (Ref. 6) was issued on July 22, 1980. This required the BWR licensees to provide a vent path from the SDV directly to the building atmosphere without any intervening component except for the vent valve itself. These modifications had to be completed within 48 h for plants operating or prior to startup for plants shut down.

On October 2, 1980, the NRC issued Confirmatory Orders to the licensees of 16 BWR plants requiring the installation of equipment to continuously monitor water levels in all SDVs and provisions for water level indication and alarm for each SDV in the control room. Until the system was installed and operating satisfactorily, the licensees were to increase their surveillance of the SDV water level. The equipment provides information that allows the reactor operator to taken timely action if water accumulates in the SDV. This equipment was to be operable by December 1, 1980, or prior to restart for those reactors in refueling, except for Browns Ferry Units 1 and 2, where installation was required by December 22, 1980 (Browns Ferry already had continuous monitors located outside the control room).

The various aspects of the problem have been and continue to be actively studied by the NRC, the BWR licensees, and the reactor vendor.

4.3.5 Failure of saltwater cooling system (AO 80-7) (Ref. 7)

On March 10, 1980, while San Onofre 1 was operating at 100% power, the south saltwater cooling pump tripped due to a failed shaft. The redundant north pump automatically started but did not supply saltwater cooling since its isolation valve failed due to a deteriorated O-ring in a solenoid valve on the valve operator. An operator then tried to manually start the auxiliary saltwater cooling pump; however, there was insufficient prime water due to an air leak in the primary system and this pump was stopped. Saltwater cooling was finally supplied by manually aligning the screen water pumps to the saltwater cooling system (SWCS), as stated in the emergency procedures. This procedure requires ~15 min to complete. The auxiliary saltwater pump was primed about 20 min later and supplied saltwater cooling.

The SWCS is a safety-related system, and its operation is required for operation of the plant by Technical Specifications. Failure of both north and south saltwater cooling pumps and the auxiliary saltwater cooling pump constituted a loss of the SWCS, and thus an orderly shutdown should have been initiated immediately. A plant shutdown was initiated ~45 min after the loss of the SWCS; however, it was terminated. This constituted a violation of Technical Specifications.

The licensee was to take the following actions to prevent recurrence.

1. The equipment that failed was (or will be) repaired and returned to service. System redesign and changes to the preventive maintenance program will be implemented to improve system reliability.
2. Desiccant was being flushed from the plant service air system during the summer refueling outage. The licensee is preparing a report on the extent of the desiccant contamination and associated problems.

3. Management has taken action to improve the plant staff's knowledge of the Technical Specifications and their basis. Tighter controls were also placed on the administrative process for changing procedures. Clarifications are being made to the Technical Specifications.
4. A review is being made of the capability of the plant to withstand postulated accidents if the SWCS and/or its alternative cooling pathways are unavailable.

The NRC has conducted special inspections of the facility related to this event and through its routine inspection and enforcement process has inspected the adequacy of management and administrative controls, including the preventive maintenance program. Based on the inspection of this event, the licensee has been cited with infractions of NRC regulations for failure to shut the plant down when both salt water cooling pumps and auxiliary salt water cooling pumps were inoperable. As a result of a February 1979 inspection, the licensee was cited in January 1980 for noncompliance with requirements for procedures for pump testing and for in-service testing of pumps and valves and a number of deficiencies related to the preventive maintenance program. The NRC requested the licensee to further assess the implications of a loss of SWCS during postulated accidents. The NRC met with the licensee in October to discuss the evaluations conducted and the planned and completed corrective actions. It is also continuing to review the adequacy of the licensee's corrective actions.

4.3.6 Significant flooding of the reactor containment building (AO 80-9) (Ref. 8)

Upon entry of the Indian Point 2 containment building on October 17, 1980, to repair a malfunctioning power range nuclear detector, ~125,000 gal of water was discovered on the containment floor, in the containment sumps, and in the cavity under the reactor pressure vessel. The source of the water was found to be service water from leaks in the service water piping and from leaks in the containment fan cooling units. Failure of both containment sump pumps to operate, lack of response of the containment sump level indicating light, and miscalibrations of the containment sump level indicating light and the containment moisture indicators allowed the water to accumulate and go undetected.

The flooding directly resulted in the failure of a power range nuclear detector; its repair was the original reason for containment building entry. Because of the flooding, the cavity under the reactor vessel was nearly filled, resulting in the wetting of the lower 9 ft of the reactor vessel and submergence of stainless steel conduits and instrument thimbles located below the reactor vessel.

Evaluations to date indicate that there was no damage to the reactor vessel or other components in the reactor vessel cavity; however, continued operation with abnormal conditions that were not known (the undetected accumulation of water in the containment) did represent some degree of decreased safety.

To prevent recurrence, the licensee has (1) installed alarms in the control room indicating increasing containment sump levels, (2) installed alarms in the control room to indicate when either submersible pump in the reactor cavity operates, (3) repaired the service water leaks, (4) installed guide bushings on the containment sump pump control floats to prevent their binding, and (5) repaired the containment sump water level indicators. Plans were made to replace the containment fan unit cooling coils prior to return to power from the current refueling outage. Further actions are also being evaluated in response to the NRC letters described below.

On October 22, 1980, the NRC Region I office issued an immediate action letter to the licensee confirming the licensee's commitments to specific actions to prevent recurrence prior to restart of the plant. The NRC staff determined that the event demonstrated a serious weakness in the licensee's management control system. As a result, on December 11, 1980, the staff proposed imposition of civil penalties in the amount of $210,000 for violations associated with the event, including failure to promptly report the event. On November 21, 1980, I&E Bulletin 80-24 (Ref. 9) was issued directing all licensees at operating plants to take specific short-term actions and to report information to the NRC. Licensees with plant designs similar to Indian Point 2 were directed to verify or provide specific equipment and procedural controls to preclude events similar to that which occurred at Indian Point Unit 2. NRC will evaluate the reports submitted by all licensees to determine what other generic longer term actions may be required.

4.3.7 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 the 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 piping in BWRs. This occurrence was originally reported in NUREG-75/090, *Report to Congress on Abnormal Occurrences: January-June 1975*, and was updated in NUREG-0090-1, -2, -3, -9, Vol. 1 (No. 3), Vol. 2 (Nos. 2 and 4), and Vol. 3 (No. 2).

The NRC staff published NUREG-0313, Rev. 1, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping*, in October 1979 and requested public comment. Comments from 11 parties were received. All comments were evaluated to determine their significance, and several modifications to the report were made to accommodate those considered significant. A final version of NUREG0313, Rev. 1, detailing the NRC staff's revised guidelines for reducing the susceptibility of intergranular stress-corrosion cracking of BWR piping was issued in July 1980.

76-1 Deficiencies in the Mark 1 containment systems of certain BWRs. This abnormal occurrence was originally reported in NUREG 0090-3, *Report to Congress on Abnormal Occurrences: January-March 1976*, and updated in subsequent reports in this series [i.e., NUREG-0090-4, -6, Vol. 1 (Nos. 1 and 3), Vol. 2 (No. 3), and Vol. 3 (No. 4)]. The NRC staff issued NUREG-0661, *Mark I Containment Long-Term Program Safety Evaluation Report*, in July 1980, thus concluding Task A-7.

This report describes the results of the NRC's review of the proposed generic hydrodynamic load definition and structural assessment techniques and the NRC Acceptance Criteria for the subsequent plant-unique assessments. The plant-unique assessments are currently under way, and most of the affected utilities have performed several of the known plant modifications in order to expedite the resolution of this issue. The Acceptance Criteria, together with schedules for completion of all of the plant modifications needed to conform to these criteria, were formally issued on January 13, 1981, to the Mark I licensees. The completion schedules for modifying the Mark I containment systems in accordance with the Long-Term Program range from October 1981 to January 1983.

76-11 Steam generator tube integrity. Since the last general update of this item [NUREG-0090, Vol. 2 (No. 4)], the following significant developments related to PWR steam generator tube integrity have occurred [Vol. 3 (Nos. 1, 2, and 4)].

Point Beach 1 continued to experience tube degradation due to a phenomenon designated as "deep crevice cracking." The unit has completed the 60-d operating interval allowed under terms of a Confirmatory Order issued on November 30, 1979. The inspection required by the order has indicated that the rate of tube degradation is somewhat retarded. The licensee has ordered some long lead time items such as tubesheets and channel heads for potential replacement of the steam generators at this unit.

Trojan is scheduled to remove some defective tubes for laboratory examinations during the current refueling outage. As reported earlier, Trojan previously experienced a tube leak due to a defect tangent to the inner tube row U-bend.

The Westinghouse topical report on their in situ retubing concept is still under review by the staff.

The staff is continuing their review of a proposed steam generator replacement program for Palisades.

San Onofre 1 shut down on April 8, 1980, because of an increasing primary to secondary coolant leak rate. Subsequent hydrostatic testing of the steam generators revealed confirmed leaking tubes in one steam generator and probable leaking tubes in one of three other steam generators, An exhaustive inspection revealed that caustic intergranular attack was occurring within a 1/4-in. band at the top of the tube sheet for the majority of the tubes with the sludge piles. A hardened sludge pile with a maximum height of ~18 in. exists over approximately two-thirds of the tubesheet on the hot-leg side of each steam generator. Southern California Edison, operator of San Onofre, decided to repair the steam generator tubes by installing leak tight sleeves inside approximately two-thirds of the tubes.

76-16 Feedwater nozzel cracking in BWRs. The following abnormal occurrence was originally reported in NUREG-0090-6, *Report to Congress on Abnormal Occurrences: October—December 1976*, and updated in subsequent reports in this series [i.e., NUREG-0090-7, Vol. 1 (No. 4), Vol. 2 (No. 2), and Vol. 3 (No. 4)]. In April 1980 the NRC staff issued for comment NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*. Public comments were received and incorporated where applicable. A final edition of the report was issued in November 1980. This document provides the NRC staff's approach to the problem's resolution.

77-9 Environmental qualification of safety-related electrical equipment inside the containment. This abnormal occurrence was originally reported in NUREG-0090-10, *Report to Congress on Abnormal Occurrences: October-December 1977*, and updated in subsequent reports in this series [i.e., Vol. 1 (Nos. 1 and 2), Vol. 2 (No. 2), and Vol. 3 (No. 2)]. It is further updated as follows: on May 23, 1980, the NRC issued a memorandum and order that addresses this subject and that directed an accelerated environmental qualification review of safety-related electrical equipment that could be exposed to a harsh environment in the event of a design basis accident at a nuclear facility. The NRC has requested pertinent information for all facilities and has initiated a review of these submittals. The review is scheduled to be completed by February 1, 1981. The order requires that, by no later than June 31, 1982, all safety-related electrical equipment in all operating plants be qualified.

78-2 Fuel assembly control rod guide tube integrity (a generic concern). This abnormal occurrence was originally reported in NUREG-0090, Vol. 1 (No. 2), *Report to Congress on Abnormal Occurrences: April-June 1978*, and updated in subsequent reports in the series [i.e., Vol. 1 (No. 4), Vol. 2 (No. 2), and Vol. 3 (No. 4)].

As previously reported, unexpected wear of rodded guide tubes has been observed in discharged PWR fuel assemblies. Fretting wear of the guide tube wall results when vibrating motion of fully withdrawn control rods in contact with the inner surface of the guide tube is induced by coolant turbulence. The NRC, in conjunction with a review group of vendors and owners, has reviewed the situation and instituted hardware modifications. Surveillance testing has confirmed that the hardware modifications have satisfactorily resolved the problem.

79-2 Deficiencies in piping design. This abnormal occurrence was originally reported in NUREG-0090, Vol. 2 (No. 1), *Report to Congress on Abnormal Occurrences: January-March 1979*, and updated in subsequent reports in this series [i.e., Vol. 2 (Nos. 2 and 4) and Vol. 3 (No. 1)]. It is further updated as follows: as previously reported, 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 shut down were Beaver Valley 1, James A. FitzPatrick, Maine Yankee, and Surry 1 and 2.

The required reanalysis and necessary modifications for the design basis earthquake (DBE) and the operating basis earthquake (OBE) have been or will be completed prior to startup for Maine Yankee and Surry 2. The Show Cause Orders for these plants have been terminated due to satisfactory partial completion of reanalysis and necessary modifications as required by the Show Cause Order.

79-3 Nuclear accident at Three Mile Island. The following abnormal occurrence was originally reported in NUREG-0090 Vol. 2 (No. 1), *Report to Congress on Abnormal Occurrences: January-March 1979*, and updated in subsequent reports in this series [i.e., Vol. 2 (Nos. 2-4) and Vol. 3 (Nos. 1-4)].

Several significant postaccident events took place at TMI-2 Nuclear Power Plant during 1980. The most important ones are listed as follows:

1. Controlled purging of the TMI-2 reactor building began on June 28, 1980, and was completed on July 11, 1980. A total of 43,800 Ci of krypton gas was released.

2. Approximately 55,000 Ci of predominantly cesium and strontium was removed from 500,000 gal of water. This represents the processing of the total basic inventory of accident-generated water that had been stored in the auxiliary and fuel handling building. The decontaminated water is being stored onsite.

3. The *Draft Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from the March 28, 1979, Accident Three Mile Island Nuclear Station, Unit 2* (NUREG-0683) was submitted for public comment. Comments were received through meetings and correspondence. A total of 910 comments were received, 151 in meetings and the remaining through correspondence.

4. Decontamination of the auxiliary and fuel handling buildings is over 90% complete.

5. Five entries were made into the reactor building by several individuals. Equipment was visually inspected, radiation levels were monitored, minor repairs were made, and decontamination techniques were tested.

6. A minidecay heat removal system (MDHRS) was approved for installation by the NRC. The MDHRS provides an appropriately sized forced flow system for removing decay heat from TMI-2 reactor fuel. This will simplify plant operations by eliminating the need for operating various systems required in the current cooling mode.

79-6 Damage to new fuel assemblies. This occurrence was originally reported in NUREG-0090, Vol. 2 (No. 2), *Report to Congress on Abnormal Occurrences: April-June 1979* and updated subsequently in Vol. 3 (No. 3).

The two plant workers who surrendered to Surry County authorities admitted deliberately damaging the new fuel assemblies. They made a number of allegations pertaining to operational and security inadequacies at Surry. A total of 46 allegations were identified by the NRC and were subsequently investigated. Of these, six were found to be wholly or partially substantiated items of noncompliance.

80-1 Occupational overexposure to skin and extremities. This occurrence was originally reported in NUREG-0090, Vol. 3 (No. 1), *Report to Congress on Abnormal Occurrences: January-March 1980* and updated subsequently in Vol. 3 (No. 2).

An independent dose assessment of the six individuals who were overexposed was performed by a consultant. The consultant reported that the calculated beta dose assignment is probably conservative in the limiting organ, and sufficient documentation exists to support this conclusion.

80-2 Transient initiated by partial loss of power. This occurrence was originally reported in NUREG-0090, Vol. 3 (No. 1), *Report to Congress on Abnormal Occurrences: January-March 1980* and updated subsequently in Vol. 3 (No. 3). As previously reported, the Director, Nuclear Reactor Regulation, issued a Confirmatory Order on April 14, 1980, addressing commitments made by the licensee (Florida Power Corporation) to make systems and procedural changes to reduce the probability of recurrence of the event at Crystal River 3. These changes were completed and tested by July 31, 1980. NRC personnel witnessed major portions of the testing of the revised systems as well as reviewed the system and procedural modifications. The plant returned to operation on August 8, 1980, when it was made critical, and began power production on August 10, 1980.

80-6 Failure of control rods to insert fully during a scram. This abnormal occurrence was originally reported in NUREG-0090, Vol. 3 (No. 2), *Report to Congress on Abnormal Occurrences: April–June 1980* and updated subsequently in Vol. 3 (No. 4).

In response to the partial failure to scram at Browns Ferry 3, the NRC issued a Confirmatory Order to 16 BWR licensees, requiring the installation of equipment to monitor the water level in the SDV. Additional surveillance testing was required prior to installation of the equipment. On December 2, 1980, the SDV's continuous monitoring system (CMS) failed to respond satisfactorily at Dresden 2.

The BWR licensees were notified of this event by I&E on December 5, 1980, through Information Notice No. 80-43. Subsequently, on December 18, 1980, the NRC staff proposed a series of actions to improve the reliability of the CMS. The NRC also issued a Generic Safety Evaluation Report discussing the NRC staff's view of the BWR SDV issue. The report specified acceptable bases for continued BWR plant operations and provided recommendations for short- and long-term modifications.

4.3.8 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.

Yankee Rowe turbine failure.[2] On February 15, 1980, the Yankee Rowe turbine experienced multiple disk failures. The reactor was at 2–3% power to provide sufficient steam for turbine startup. Before synchronizing, a thump was heard from an apparent severe jarring of the turbine. The turbine coasted down for about 25 min compared to the normal coastdown time of 45–60 min. The reactor did not experience a transient as a result of the turbine failure. There was negligible decay heat in the core because of the long period of shutdown and no condition to cause a reactor trip. The reactor was left critical for about 0.5–1 h, then was shut down normally, and was placed in a cold shutdown condition. The turbine outer and inner casings were removed and an initial visual inspection was performed. It was found that both first stage disks in the low-pressure rotor were completely failed. They were broken into several major pieces and many smaller fragments. Major damage was also observed at several adjacent rows of blades and stators. Preliminary information indicates extensive cracking in the bore of the first stage disk at the generator end. It appears that one large piece of a first stage disk got wedged against the shaft during turbine coastdown and may have caused significant damage to the shaft.

The disk fragments were shipped to Westinghouse for a detailed investigation.

BWR jet pump assembly failure.[3] On February 2, 1980, Commonwealth Edison reported that a jet pump failed in Dresden 3. The unit was operating at 67% power and preparing for a refueling outage. A remote-camera and visual inspection of the jet pumps and vessel were made after defueling which revealed that the holddown beam assembly had broken. This allowed the jet pump components to disassemble. Subsequent nondestructive examination of the remaining 19 holddown beams identified cracks in 6 additional beams. Investigations by General Electric showed that intergranular stress corrosion cracking under sustained loading was the cause of the beam failure.

Reactor coolant pump seal failure.[3] On May 10, 1980, Arkansas Nuclear One Unit 1 reported the "C" reactor coolant pump seal had failed. The reactor was placed in cold shutdown and the leaking pump was isolated. The maximum leak rate was estimated to be 350 gpm. High-pressure injection was used to maintain pressurizer level and reactor coolant pressure. Damage to the seal was severe. Seals on all four reactor coolant pumps were replaced. The licensee is working with Byron Jackson, the pump manufacturer, and Babcock & Wilcox on the failure analysis investigations.

Development of steam void under vessel head during reactor cooldowns.[3] On June 11, 1980, a loss of component cooling water occurred at St. Lucie 1. The reactor was manually tripped and natural circulation cooldown was initiated to prevent damage to the reactor coolant pumps.

The natural circulation cooldown continued uneventfully until after 6:00 AM. The highest cooldown rate achieved was ~65–70°F/h, which is within operational limits. Between 6:00 and 6:30 AM, RCS pressure was reduced from 1140 to 690 psi by charging water through the pressurizer auxiliary spray line. Around 7:00 AM, while still charging via the auxiliary spray line, pressurizer level increased at rates faster than the rate at which water was being added. Pressurizer level then experienced wide variations, which continued for ~5 h while the cooldown and depressurization continued.

The pressurizer level variations have been shown to be due to the formation of a relatively large steam void in the reactor head area that persisted for a number of hours. The void was due to a temperature lag between the bulk coolant and the vessel head area because of lower cooling flow in the head area during natural circulation cooldown.

Containment sump valve open during reactor operation.[7] On July 27, 1980, a containment sump valve was discovered open at Palisades while the reactor was at 80% power. A reactor operator had apparently opened the valve inadvertently while performing a surveillance test on another system. It was left open for a 36-h period. Analysis of the occurrence showed all safety functions could have been performed in the event of a loss-of-coolant accident.

Concern over licensed operator performance at a power reactor.[7] At 6:00 AM on August 8, 1980, the NRC Senior Resident Inspector at Commonwealth Edison Company's Dresden Nuclear Power Station entered the control room and observed that two of the four licensed reactor operators appeared to be asleep. Units 2 and 3 were in operation. The inspector reported his finding to a manager, who called the control room. When the inspector and the manager arrived at the control room, the operators were awake and attentive. The operators denied that they had been asleep.

An investigation determined that the two operators were not fully attentive to their licensed duties of monitoring reactor conditions, although it could not be determined whether or not the operators were actually asleep. The two operators were issued letters of reprimand by the

NRC Director of the Office of Inspection and Enforcement. Commonwealth Edison is to revise its training program, procedures, and policies to establish operator performance standards and provide for disciplinary action for improper conduct.

Personnel overexposure during steam generator repair.[7] During extensive steam generator repair at San Onofre 1, 66 workers received doses in excess of the regulatory limit of 3 rem per calendar quarter. The highest calculated dose to any single individual for a calendar quarter was 4.9 rem. During the initial periods of work in the steam generators, the licensee failed to make adequate surveys, resulting in overexposures to several personnel.

Inadvertent isolation of auxiliary feedwater system water supply.[8] On May 20, 1980, Calvert Cliffs 1 was manually tripped from full power due to a degradation in the service water system. The reactor was placed in hot standby and auxiliary feedwater (AFW) pumps were used to maintain steam generator water level. About 3 h later the main feedwater pump was started to remove decay heat, and the AFW pump was secured and aligned to take suction from the No. 12 condensate storage tank (CST). The operator inadvertently transposed the nomenclature and valve number which resulted in isolating both condensate storage tanks. The error was detected 15 min later and corrected immediately when the senior control room operator ordered the valve lineup to be verified.

Failure to adequately implement a post-TMI-action item.[8] As a result of the TMI-2 accident, the NRC required all licensees to increase the range of their noble gas monitors. Nine Mile Point responded to the NRC order stating they had satisfied the NRC requirements. However, a health physics inspection performed on October 8, 1980, revealed the licensee had made only a token effort which was technically inadequate. Subsequent investigation revealed that certain key management personnel had been aware that the licensee's actual performance in this area was substantially different from the representation provided by the licensee in a December 31, 1979, letter.

4.4 References

1. U.S. Nuclear Regulatory Commission, *Notification of Significant Events*, I&E Information Notice 80-06 (Feb. 27, 1980).
2. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, January-March 1980*, NUREG-0090, Vol. 3, No. 1 (September 1980).*
3. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, April-June 1980*, NUREG-0090, Vol. 3, No. 2 (November 1980).*
4. U.S. Nuclear Regulatory Commission, *Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR*, I&E Bulletin 80-17 (July 3, 1980).

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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.

5. U.S. Nuclear Regulatory Commission, *Failiure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR*, I&E Bulletin 80-17, Supplement No. 1 (July 18, 1980).
6. U.S. Nuclear Regulatory Commission, Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR, *I&E Bulletin 80-17, Supplement No. 2 (July 22, 1980)*.
7. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, July–September 1980*, NUREG-0090, Vol. 3, No. 3 (February 1981).*
8. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, October–December 1980*, NUREG-0090, Vol. 3, No. 4 (May 1981).*
9. U.S. Nuclear Regulatory Commission, *Prevention of Damage Due to Water Leakage Inside Containment (October 17, 1980 Indian Point 2 Event)*, I&E Bulletin 80-24, Nov. 21, 1980.*

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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.

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) that 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 that 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 1980. None were considered significant enough to be included in NRC's *Report to Congress On Abnormal Occurrences* (NUREG-0090 series). Fuel failure events reported as LERs are discussed in Sects. 5.2.1–5.2.6.

5.2.1 Brunswick 2 (BWR)

A report to NRC dated December 8, 1980, stated that the reactor coolant activity level exceeded the limit of 0.2 μCi/g as result of a reactor scram and leaking fuel elements at Brunswick 2. A report dated December 9, 1980, for Unit 1, where ^{131}I concentrations in samples of milk exceeded limits, stated that the source for this iodine activity was attributed to fuel element leaks in Unit 2. Fuel sipping was planned for the next refueling outage, and defective fuel elements were to be removed from the core and replaced. (LER 80-082 and an environmental report were issued for Unit 1 on December 9, 1980.)

5.2.2 Arkansas Nuclear One Unit 1 (PWR)

The radioactive gas release rate exceeded Technical Specification at Arkansas Nuclear One Unit 1 during the first and third quarters of 1980 as reported to NRC on April 16, 1980, and August 22, 1980, respectively. The average gross gas release rate was 4.3 and 4.47% of the maximum permissible concentration for each quarter, respectively. The excessive release rates were caused by failure of ~0.08% of the fuel accompanied by a purge of the reactor building during the first quarter and by another purge of the reactor building following a plant shutdown to correct a steam generator tube leak during the third quarter (LERs 80-006 and 80-027).

5.2.3 Crystal River 3 (PWR)

Four reports from Florida Power Corporation (January 17, 1980; February 4, 1980; March 10, 1980; and November 5, 1980) described five events in which the dose equivalent of ^{131}I in the reactor coolant exceeded the limit of 1 μCi/g. (Twelve similar events occurred in previous years.) All of these events were caused by an expected iodine spike following either a reactor shutdown or reactor transient with known leaking fuel (LERs 79-109, 80-002, 80-009, and 80-042).

5.2.4 Maine Yankee (PWR)

Nine fuel assemblies were identified during routine fuel sipping as containing a total of ten leaking fuel pins at Maine Yankee and reported to NRC on February 14, 1980. All fuel pins in which through-wall penetrations were observed were removed from the reactor (LER 80-004).

5.2.5 Rancho Seco (PWR)

A report dated March 25, 1980, described a 4-in. section of fuel cladding that was found missing from a fuel pin at Rancho Seco. The discovery was made during a visual examination of the cycle 3 discharged fuel elements. No additional failures were found on examination. A review of the RCS chemistry indicated that the average ^{131}I and ^{133}I levels were not abnormal. No detectable alpha activity in the reactor coolant indicated that insignificant amounts of fuel had been dispersed from the fuel pin (LER 80-015).

5.2.6 Trojan (PWR)

A report dated May 8, 1980, stated that two fuel assemblies were found during a planned fuel inspection program at Trojan to have abnormal clad degradation. A total of two fuel pins had cladding failure. The apparent cause was water-jet impingement on the fuel pins via an enlarged baffle plate point gap (LER 80-006).

5.3 References

1. F. Garzarolli, R. von Jan, and H. Steahle, "The Main Causes of Fuel Element Failure in Water-Cooled Power Reactors," *At. Energy Rev.* 17(1), 31 (March 1979).

6. RADIATION EXPOSURE

6.1 Occupational Radiation Exposure

This section reviews the data on occupational radiation exposure of personnel at BWR and PWR commercial nuclear power plants. Data from 69 plants are considered based on their completion of at least one year of commercial operation as of December 31, 1980. Fort St. Vrain (an HTGR) is included for the first time, and Indian Point 1, although defueled, is still included in the review.

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 collective dose (man-rems) 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, reveal that workers at the 26 BWRs incurred a larger collective dose (27,878 man-rems) than did workers at the 42 PWRs. They also show that 69.8% of the total collective dose was incurred by contractor personnel at BWRs compared to 66.6% at PWRs. Table 6.3 presents a breakdown of these collective doses by work function for the last seven years. One can see that workers performing routine and special maintenance activities continue to receive most (76.1% in 1980) of the total collective dose. Table 6.4 shows the percentage of the collective dose incurred by different types of personnel at BWRs and PWRs by work function. As was the case last year, at PWRs the largest portion (43.6%) of the collective dose was incurred by workers involved in special maintenance activities, whereas at BWRs the largest portion (42.7%) of the collective dose was incurred by workers involved in routine maintenance activities.

Table 6.5 summarizes the exposure information reported pursuant to 10 CFR 20.407 by commercial BWRs and PWRs during the last eight years. The average annual dose for individuals receiving measurable doses is 0.67 rems, remaining less than 1 rem as it has every year since 1972.

The total collective dose at LWRs for 1980 (53,796 man-rems) increased considerably over last year's value (39,759 man-rems). Part of the increase could be due to modifications of Mark I toruses and the replacement of certain stainless steel components at BWRs. Also, the activities required by NRC bulletins may have 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 – 1980* (NUREG-0713, Vol. 2), which will be available from the National Technical Information Service.

Table 6.1. Annual whole body doses at BWRs – 1980[a]

Plant name	Plant and utility personnel		Contractor personnel		Totals	
	Number of workers with doses >0.1 rem	Collective dose (man-rems)	Number of workers with doses >0.1 rem	Collective dose (man-rems)	Number of workers with doses >0.1 rem	Collective dose (man-rems)
Big Rock Point	174[b]	274	101[b]	94	275[b]	368
Browns Ferry 1, 2, 3	2,010	1,231	68	34	2,078	1,265
Brunswick 1, 2	501	731	2,135	2,933	2,636	3,664
Cooper Station	145	205	375	615	520	820
Dresden 1, 2, 3	637[b]	975	1,160[b]	1,053	1,797[b]	2,028
Duane Arnold	97	100	658	564	755	664
FitzPatrick	292[b]	243	1,152[b]	1,892	1,444[b]	2,135
Hatch 1, 2[c]	437[b]	349	454[b]	199	891[b]	548
Humboldt Bay	39	13	4	2	43	15
LaCrosse	71	204	19	11	90	215
Millstone 1	383[b]	283	1,883[b]	1,792	2,266[b]	2,075
Monticello	406[b]	261	334[b]	228	740[b]	489
Nine Mile Point	476[b]	272	177[b]	200	653[b]	472
Oyster Creek	465	572	1,295	1,229	1,760	1,801
Peach Bottom 2, 3	751	651	1,379	1,473	2,130	2,124
Pilgrim	376	481	2,035	2,695	2,411	3,176
Quad Cities 1, 2	441[b]	1,150	2,064[b]	3,560	2,505[b]	4,710
Vermont Yankee	292[b]	402	792[b]	907	1,084[b]	1,309
Totals	7,993	8,397	16,085	19,481	24,078	27,878

[a]Includes only those reactors that had been in commercial operation for at least one year as of December 31, 1980.

[b]Data presented are taken from the annual reports submitted in accordance with Regulatory Guide 1.16 except where the reported number of personnel receiving doses greater than 0.1 rem 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 0.1 rem, as determined from the 10 CFR 20.407 reports.

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

Table 6.2. Annual whole body doses at PWRs — 1980[a]

Plant name	Plant and utility personnel: Number of workers with doses >0.1 rem	Plant and utility personnel: Collective dose (man-rems)	Contractor personnel: Number of workers with doses >0.1 rem	Contractor personnel: Collective dose (man-rems)	Totals: Number of workers with doses >0.1 rem	Totals: Collective dose (man-rems)
Arkansas 1	245	99	398	164	643	263
Beaver Valley	211	68	857	428	1,068	496
Calvert Cliffs 1, 2	423	246	760	359	1,183	605
Cook 1, 2	215	155	573	295	788	450
Crystal River	298[b]	233	434[b]	365	732[b]	598
Davis-Besse	76[b]	60	285[b]	219	361[b]	279
Farley	423[b]	217	272[b]	160	695[b]	377
Fort Calhoun	214	249	340	438	554	687
Ginna	338[b]	409	347[b]	305	685[b]	714
Haddam Neck	296[b]	239	1,095[b]	929	1,391[b]	1,168
Indian Point 1, 2	534	555	534	384	1,068	939
Indian Point 3	178[b]	111	378[b]	236	556[b]	347
Kewaunee	129[b]	55	112[b]	91	241[b]	146
Maine Yankee	179[b]	222	298[b]	233	477[b]	555
Millstone 2	110[b]	117	560[b]	494	670[b]	611
North Anna 1	130[b]	121	174[b]	77	304[b]	198
Oconee 1, 2, 3	1,173	947	333	172	1,506	1,119
Palisades	166	103	360	290	526	393
Point Beach 1, 2	169	175	307	413	476	588
Prairie Island 1, 2	404[b]	197	127[b]	132	531[b]	329
Rancho Seco	167[b]	94	355[b]	199	522[b]	293
Robinson 2	300	450	1,070	1,312	1,370	1,762
Salem 1	264[b]	148	511[b]	249	775[b]	397
San Onofre 1	335	346	1,579	1,895	1,914	2,241
St. Lucie	273	314	445	181	718	495
Surry 1, 2	531	687	2,445	2,978	2,976	3,665
Three Mile Island 1, 2	484[b]	201	488[b]	309	972[b]	510
Trojan	188[b]	123	499[b]	325	687[b]	448
Turkey Point 3, 4	437	477	1,070	1,342	1,507	1,819
Yankee Rowe	150[b]	103	79[b]	76	229[b]	179
Zion 1, 2	321[b]	338	494[b]	526	815[b]	864
Totals	9,361	7,859	17,579	15,676	26,940	23,535

[a]Includes only those reactors that had been in commercial operation for at least one year as of December 31, 1980.

[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 0.1 rem 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 0.1 rem, as reported in their 10 CFR 20.407 annual reports. This total was broken down into the number of personnel types by assuming that the proportion of a type was the same as that shown in the 1.16 reports.

Table 6.3. Percentages of total collective doses incurred by workers at LWRs by work function for 1974–1980

Work function	Percent of total collective dose						
	1974	1975	1976	1977	1978	1979	1980
Reactor operations and surveillance	14.0	10.8	10.4	10.5	13.2	12.2	9.4
Routine maintenance	45.4	52.5	31.7	28.1	31.5	29.2	35.5
In-service inspection	2.7	2.9	5.7	6.4	7.7	9.0	5.5
Special maintenance	20.4	19.0	39.5	42.5	35.9	39.4	40.6
Waste processing	3.5	6.9	4.8	5.8	5.0	3.6	2.9
Refueling	14.0	7.7	7.9	6.7	6.5	6.6	6.0

Table 6.4. Percentages of collective doses incurred by types of workers at BWRs and PWRs by work function in 1980

Work function	BWR personnel type		PWR personnel type	
	Plant and utility	Contractors	Plant and utility	Contractors
Reactor operations and surveillance	6.1	1.5	8.0	3.5
Routine maintenance	14.1	28.6	10.0	17.0
In-service inspection	0.9	2.4	1.2	7.0
Special maintenance	4.9	33.2	8.7	34.9
Waste processing	1.9	1.2	1.5	1.1
Refueling	2.2	2.9	4.0	3.1
Total	30.1	69.8	33.4	66.6

Table 6.5. Summary of annual doses reported by nuclear power facilities, 1973-1980[a]

Year	Reactor type	Number of reactors included	Total collective dose (man-rems)	Number of workers with measurable doses	Total megawatt-years generated	Average annual dose (rems/worker)	Average collective dose per reactor (man-rems)	Average number 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	17,346	0.65	396	614	0.8
	BWR	23	19,042	21,388	9,098	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	38,828	18,249	0.55	510	924	1.2
	BWR	25	18,322	25,245	11,671	0.73	733	1,010	1.6
	Total	67	39,759	64,073	29,920	0.62	593	956	1.3
1980	PWR	42	24,266	46,237	18,287	0.52	578	1,101	1.3
	BWR	26	29,530	34,094	10,868	0.87	1,136	1,311	2.7
	Total	68	53,796	80,331	29,155	0.67	791	1,181	1.8
1980	HTGR	1	3	58	83	0.05	3	58	0.0

[a]The figures in this table are based on the number of nuclear power reactors that had been in commercial operation for at least one year as of December 31 of each of the years indicated. Indian Point 1, although defueled, is counted; Fort St. Vrain is shown for the first time.

Appendix A

GLOSSARY

Abnormal occurrence	See Sect. 4.3 and Appendix C.
Average daily power level, MWe net	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, MWt	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 MWe	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 MWe	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), MWe net	Maximum dependable capacity (gross) less the normal station service loads.
Nameplate rating, gross MWe	The nameplate power designation of the generator, in megavolt-amperes (MV-A), 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 plus hours generator on-line.)
Unit availability factor	$\frac{\text{Unit available hours x 100}}{\text{Period hours}}$
Unit capacity factors	
Using licensed thermal power	$\frac{\text{Gross thermal energy generated x 100}}{\text{Period hours x licensed thermal power}}$
Using nameplate rating	$\frac{\text{Gross electrical energy generated x 100}}{\text{Period hours x nameplate rating}}$
Using DER	$\frac{\text{Net electrical energy generated x 100}}{\text{Period hours x DER}}$
Using MDC gross*	$\frac{\text{Gross electrical energy generated x 100}}{\text{Period hours x MDC gross}}$
Using MDC net*	$\frac{\text{Net electrical energy generated x 100}}{\text{Period hours x MDC net}}$
Unit forced outage rate	$\frac{\text{Forced outage hours}}{\text{Unit service hours + forced outage hours}}$
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	$\frac{\text{Unit service hours x 100}}{\text{Period hours}}$
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 1980

Summaries of the 1980 operating experience for each plant are presented in this appendix. 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 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 outage and S is used for scheduled outage. 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," 1 is manual, 2 is manual scram, 3 is automatic scram, 4 is continuations, and 9 is other.

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, 1980; 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 turbines, 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

Location: Russellville, Arkansas
Docket No.: 50-313
Reactor type: PWR
Maximum dependable capacity (MWe-net): 836
Commercial operation: 12/19/74
Years operating experience: 6.4

Performance

Net electrical energy generated (MWh): 3,781,602
Unit availability factor (%): 63.7
Unit capacity factor (using MDC): 51.5
Unit capacity factor (%) (using design MWe): 50.6

Outages

Total No.: 11
Forced: 9
Scheduled: 2
Total hours: 3,209.5 (36.6%)[a]
Forced: 1,884.7 (21.5%)
Scheduled: 1,324.8 (15.1%)[a]

II. Highlights

The unit was shut down at the beginning of the reporting period for TMI-related modifications. The blocking on the low-pressure turbine "B" was modified during this outage because of cracking in the turbine rotor. Initial 1980 startup was on February 8, and reactor power was limited to 90% because of the turbine modification. Both Arkansas units suffered a loss of offsite power on April 7 due to a tornado. Maintenance during the two-week April outage included Crystal River-related modifications. Five days after the ensuing startup, a reactor coolant pump seal failed, and the unit was down for nearly a month. Steam generator tube leaks resulted in 41 d of downtime in July and September.

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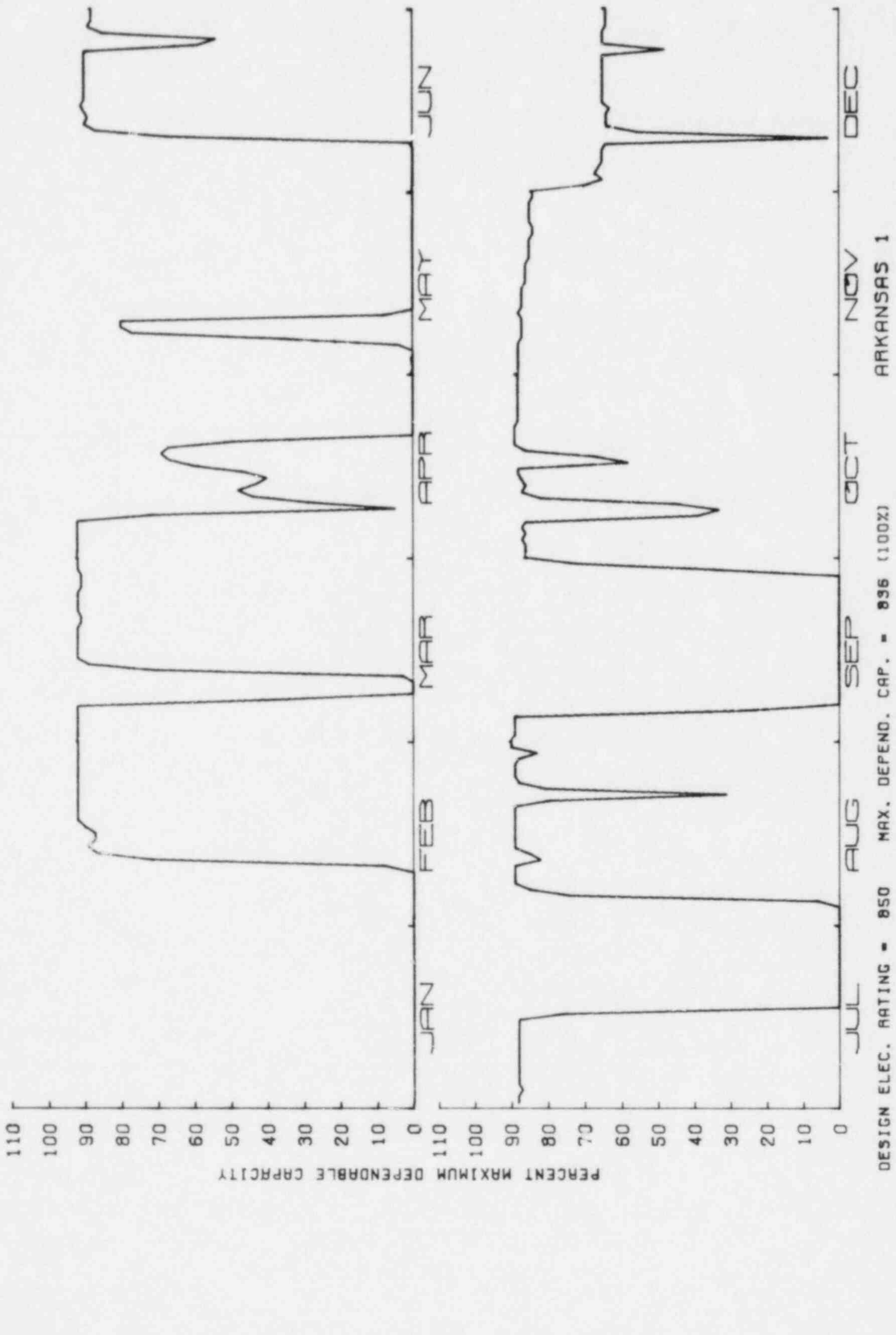
[a]Includes 951.2 h in 1980 from continued shutdown of 12/31/79.

DETAILS OF PLANT OUTAGES FOR ARKANSAS 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/31/79 (cont.)	951.2	S	Commitment to NRC to make TMI-related modifications.	D	4	Reactor coolant (CH)	Instrumentation and controls
1	3/07	104.7	F	Vacuum leak in main condenser due to failure of the rubber expansion.	A	1	Steam and power conversion (HC)	Pipes, fittings
2	4/07	20.7	F	Loss of offsite power due to tornado damage to transmission lines.	H	3	Electric power (EA)	Electrical conductors
3	4/19	369.5	S	Crystal River-related modifications, HP turbine steam seal repair, and LP turbine/condenser expansion joint repairs.	F	1	Instrumentation and controls (IE)	Instrumentation and controls
4	5/05	4.1	S	Turbine overspeed trip test.	B	1	Steam and power conversion (HA)	Turbines
5	5/10	715.3	F	RCP seal failure (LER 80-13).	A	1	Reactor coolant (CB)	Pumps
6	6/24	9.8	F	Partial loss of offsite power (LER 80-22).	H	3	Electric power (EA)	Not applicable
7	7/16	448.7	F	SG A tube leak - two leaking tubes found and plugged, one other defective tube plugged (LER 80-26).	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
8	8/22	9.6	F	A runback was initiated when the B MFWP tripped due to its oil pump tripping because of a motor bearing failure. During the runback the reactor tripped on high RCS pressure when MFWP A control failed. Debris was cleaned from the MFWP governor mechanism.	A	3	Steam and power conversion (HH)	Motors

DETAILS OF PLANT OUTAGES FOR ARKANSAS 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	9/05	549.6	F	One tube found leaking and was plugged in SG A (LER 80-34).	A	1	Reactor coolant (CC)	Heat exchangers (steam generator)
10	10/07	6.1	F	Accidental trip of MFWP A when a mechanic slipped and hit the trip mechanism.	G	3	Steam and power conversion (HH)	Pumps
11	12/08	20.2	F	Secondary plant load oscillations - cause unknown. Thought to be due to a governor valve being close to its break open point.	A	3	Steam and power conversion (HA)	Valves


PERCENT MAXIMUM DEPENDABLE CAPACITY
110
100
90
80
70
60
50
40
30
20
10
0
JAN
FEB
MAR
APR
MAY
JUN
JUL
AUG
SEP
OCT
NOV
DEC
DESIGN ELEC. RATING = 850 MAX. DEPEND. CAP. = 836 (100%)
ARKANSAS 1

ARKANSAS 2

I. Summary

Description

Location: Russellville, Arkansas
Docket No.: 50-368
Reactor type: PWR
Maximum dependable capacity (MWe-net): 858
Commercial operation: 3/26/80
Years operating experience: 2.0

Performance

Net electrical energy generated (MWh): 3,647,197
Unit availability factor (%): 74.0
Unit capacity factor (using MDC): 63.0
Unit capacity factor (%) (using design MWe): 59.3

Outages

Total No.: 29
- Forced: 27
- Scheduled: 2

Total hours: 3,137.9 (35.8%)
- Forced: 1,902.5 (21.5%)
- Scheduled: 1,235.4 (14.1%)

II. Highlights

The unit was in power ascension testing until January 29 when it was shut down for TMI-related modifications (NUREG-0578). A failed diesel generator rotor shaft extended the outage to March 19. Commercial operation was formally declared on March 26. The unit was off-line from September 3–29 for cleaning asian clams, silt, and corrosion products from reactor building cooling coils. The unit had an availability factor of 74.0 in 1980.

DETAILS OF PLANT OUTAGES FOR ARKANSAS 2

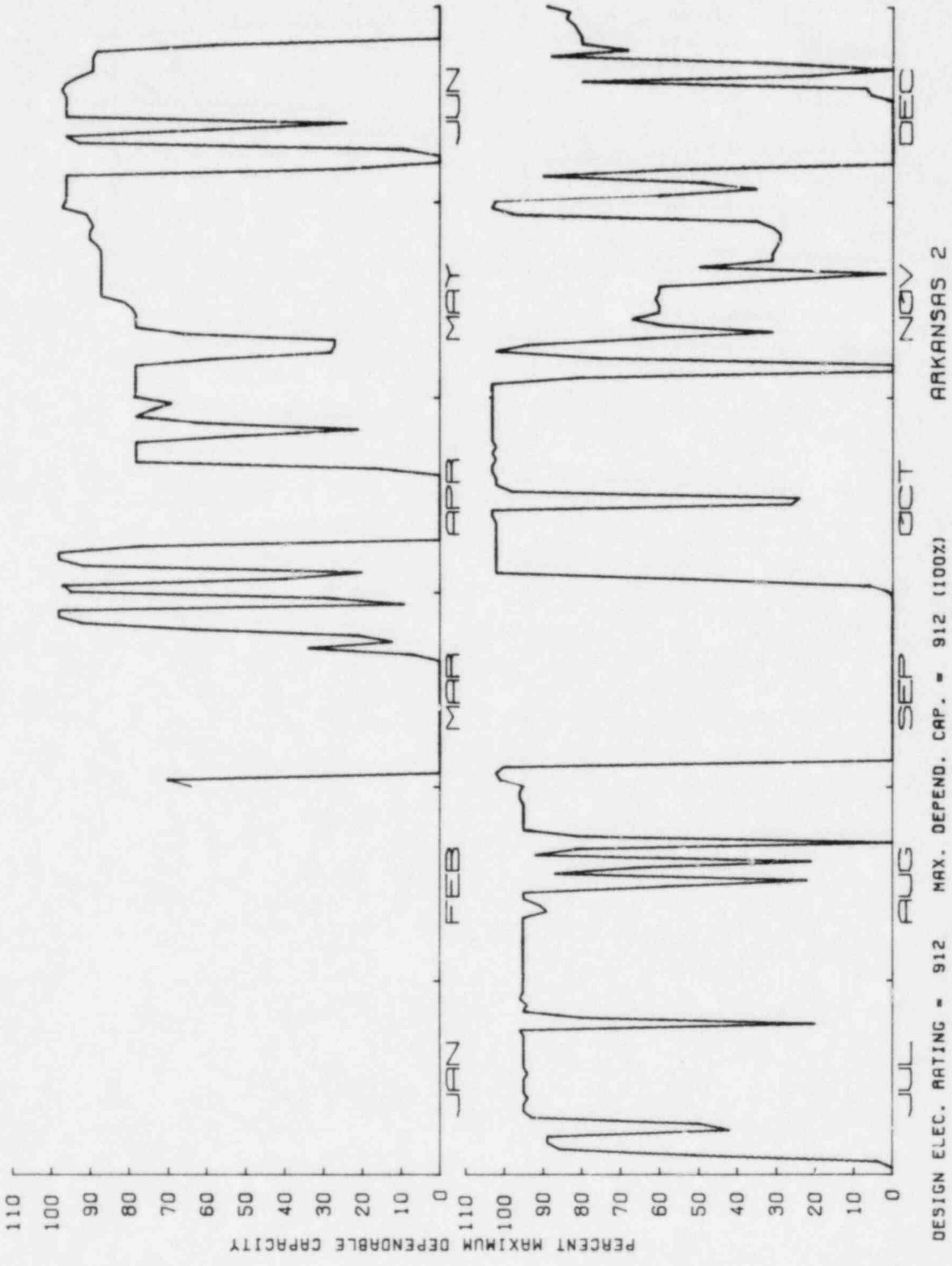
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/17	42.1	F	Feedwater control valve failure due to loss of airline. Airline replaced.	A	3	Steam and power conversion (HH)	Pipes, fittings
2	1/19	14.5	F	Turbine trip due to high vibrations.	A	3	Steam and power conversion (HA)	Turbines
3	1/21	6.3	F	Second trip due to high vibration in turbine.	A	3	Steam and power conversion (HA)	Turbines
4	1/29	1215.6	S	Scheduled reactor/turbine trip per startup testing sequence. During outage, TMI-related modifications were made. Also, a failed diesel generator rotor shaft was discovered during a test, extending the outage.	B	2	Electric power (EE)	Engines, internal combustion
5	3/20	11.6	F	Calculator failure for control element assembly; rod No. 26 dropped and was recovered.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
6	3/22	19.8	S	Power escalation testing.	B	2	System code not applicable (ZZ)	Component code not applicable
	3/26			Declaration of commercial operation.				
7	3/28	15.6	F	MFWP trip due to loss of level indicator in condenser hotwell; loose wire found.	A	3	Steam and power conversion (HC)	Instrumentation and controls
8	3/28	16.0	F	Core protection calculator tripped reactor at axial shape index of 0.6.	H	3	Instrumentation and controls (IA)	Instrumentation and controls
9	4/02	24.7	F	Lost FW control due to inadvertent cycling of control breaker.	A	3	Steam and power conversion (HH)	Relays
10	4/07	278.3	F	Loss of offsite power due to weather. Unit stayed off line due to lack of transmission capabilities.	H	3	Electric power (EA)	Electrical conductors

DETAILS OF PLANT OUTAGES FOR ARKANSAS 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11	4/24	17.7	F	Loss of control rod position indication.	A	3	Reactor (RB)	Instrumentation and controls
12	4/25	4.1	F	Loss of control rod position indication (LER 80-12).	A	3	Reactor (RB)	Instrumentation and controls
13	6/05	75.7	F	Steam leak on main steam line in reactor building.	A	1	Steam and power conversion (HB)	Pipes, fittings
14	6/11	13.5	F	Two spurious DNBR/linear power density (LPD) trips occurred simultaneously on the core protection calculators.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
15	6/24	191.5	F	Partial loss of offsite power due to a ground fault on one of the 500-kV transmission lines (LER 80-42).	A	3	Electric power (EA)	Electrical conductors
16	7/07	13.9	F	CEAC No. 1 in test while CEAC No. 2 inoperable caused trip due to large penalty factor associated with both CEACs inoperable.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
17	7/24	11.8	F	CEA No. 48 dropped due to circuit board failure.	A	3	Reactor (RB)	Instrumentation and controls
18	8/15	13.1	F	Lost stator cooling water while switching control room chillers and main chillers.	G	3	Auxiliary water (WG)	Heat exchangers
19	8/16	5.0	F	Low SG level because of faulty control oil line fitting.	A	3	Steam and power conversion (HH)	Pipes, fittings
20	8/18	8.6	F	Low SG level because of blown fuse in FW control system cabinet.	A	3	Steam and power conversion (HH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR ARKANSAS 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
21	8/21	25.6	F	DNBR/linear power density (LPD) trips on core protection computers while switching inverter power from station battery to AC.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
22	9/03	642.4	F	Cleaning of the reactor building cooling coils of asian clams, silt, and corrosion products (LER 80-72).	B	1	Auxiliary water (WA)	Heat exchangers
23	10/01	16.4	F	Lost RCP B during maintenance in pump breaker cabinet.	A	1	Reactor coolant (CB)	Pumps
24	10/14	28.3	F	Square root extractor in the FW control failed causing on overspeed. High SG level trip followed.	A	3	Reactor coolant (CH)	Instrumentation and controls
25	11/03	49.5	F	Excessive RCS leakage from DP sensing line due to fatigue weld failure (LER 80-86).	A	1	Reactor coolant (CB)	Mechanical function units
26	11/18	29.5	F	MSIV (2CV - 1010) declared inoperable due to excessive stroke time (LER 80-84).	A	1	Engineered safety features (SD)	Valves
27	12/05	283.8	F	Crack on charging pump suction piping (LER 80-90).	A	2	Auxiliary process (PC)	Pipes, fittings
28	12/17	18.1	F	Excess RCS leakage (0.91 gpm) because of packing leak in valve 2CV - 4827.	A	1	Auxiliary process (PC)	Valves
29	12/20	44.9	F	Refueling water tank level sensing lines froze, rendering all level instrumentation inoperable (LER 80-91).	A	2	Instrumentation and controls (IE)	Instrumentation and controls


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 912 MAX. DEPEND. CAP. = 912 (100%) ARKANSAS 2

BEAVER VALLEY 1

I. Summary

Description

Location: Shippingport, Pennsylvania
Docket No.: 50-334
Reactor type: PWR
Maximum dependable capacity (MWe-net): 810
Commercial operation: 10/01/76
Years operating experience: 4.6

Performance

Net electrical energy generated (MWh): 300,775
Unit availability factor (%): 6.8
Unit capacity factor (using MDC): 4.2
Unit capacity factor (%) (using design MWe): 4.0

Outages

Total No.: 6
Forced: 6
Scheduled: 0
Total hours: 8,182.9 (93.1%)[a]
Forced: 388.2 (4.4%)
Scheduled: 7,794.7 (88.7%)[a]

II. Highlights

The unit remained down for major TMI-related modifications and refueling through November 20. After November 26 the unit operated normally until December 18 when leaking pressurizer safety valves forced an outage lasting through the end of the year.

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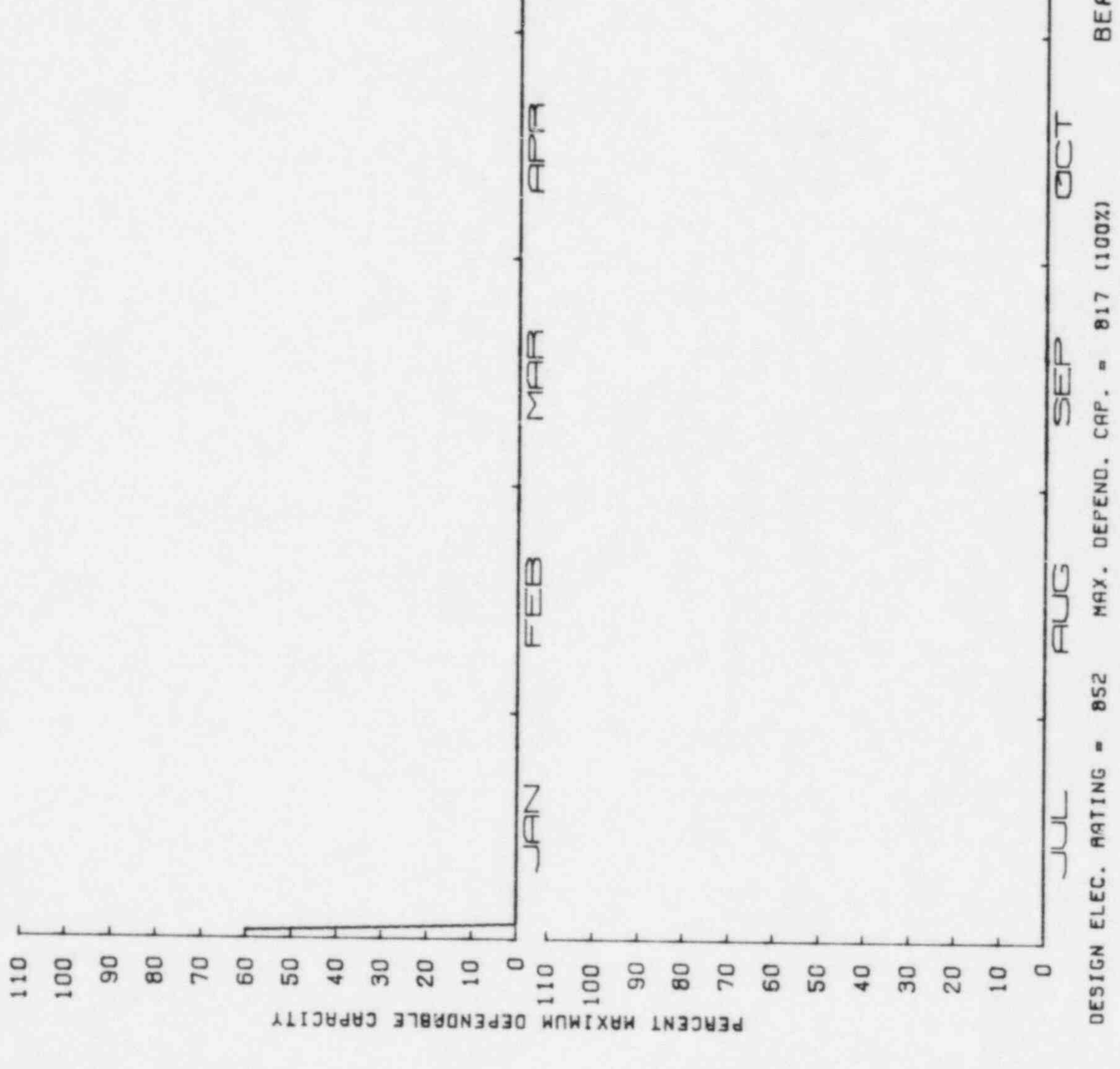
[a]Includes 7,794.7 h in 1980 from continued shutdown of 12/01/79.

DETAILS OF PLANT OUTAGES FOR BEAVER VALLEY 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/01/79 (cont.)	7794.7	S	Refueling and major modifications as required by NRC, including IE Bulletins 79-02 and 79-14.	B	4	Other (XX)	Other
1	11/20	12.0	F	High SG level when 1B and 1C SG bypass feed control valves did not regulate properly.	A	3	Steam and power conversion (HH)	Instrumentation and controls
2	11/21	30.3	F	High SG level when a FW flow signal isolator failed, which caused the 1B main feed regulating valve to go wide open during transfer from bypass flow control (LER 80-96).	A	3	Steam and power conversion (HH)	Instrumentation and controls
3	11/23	17.0	F	Turbine trip while performing turbine thrust bearing oil trip check.	A	3	Steam and power conversion (HA)	Instrumentation and controls
4	11/23	3.1	F	Low SG level when transferring from bypass flow control.	G	3	Steam and power conversion (HH)	Instrumentation and controls
5	11/26	3.7	F	EHC panel power supply inadvertently shorted when trouble shooting an alarm problem.	G	3	Steam and power conversion (HA)	Annunciators
6	12/18	322.1	F	Leaking pressurizer safety valves. The main safety valves had minor corrosive deposits; the pilot valves were not leaking.	A	1	Reactor coolant (CB)	Valves



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DESIGN ELEC. RATING = 852 MAX. DEPEND. CAP. = 817 (100%)
BEAVER VALLEY 1

BIG ROCK POINT

I. Summary

Description	Performance	Outages
Location: Charlevoix, Michigan	Net electrical energy generated (MWh): 405,450	Total No.: 6
Docket No.: 50-155	Unit availability factor (%): 78.9	Forced: 4
Reactor type: BWR	Unit capacity factor (using MDC): 71.5	Scheduled: 2
Maximum dependable capacity (MWe-net): 63	Unit capacity factor (%) (using design MWe): 64.1	Total hours: 1,851.1 (21.0%)[a]
Commercial operation: 3/29/63		Forced: 38.8 (0.4%)
Years operating experience: 18.1		Scheduled: 1,812.3 (20.6%)[a]

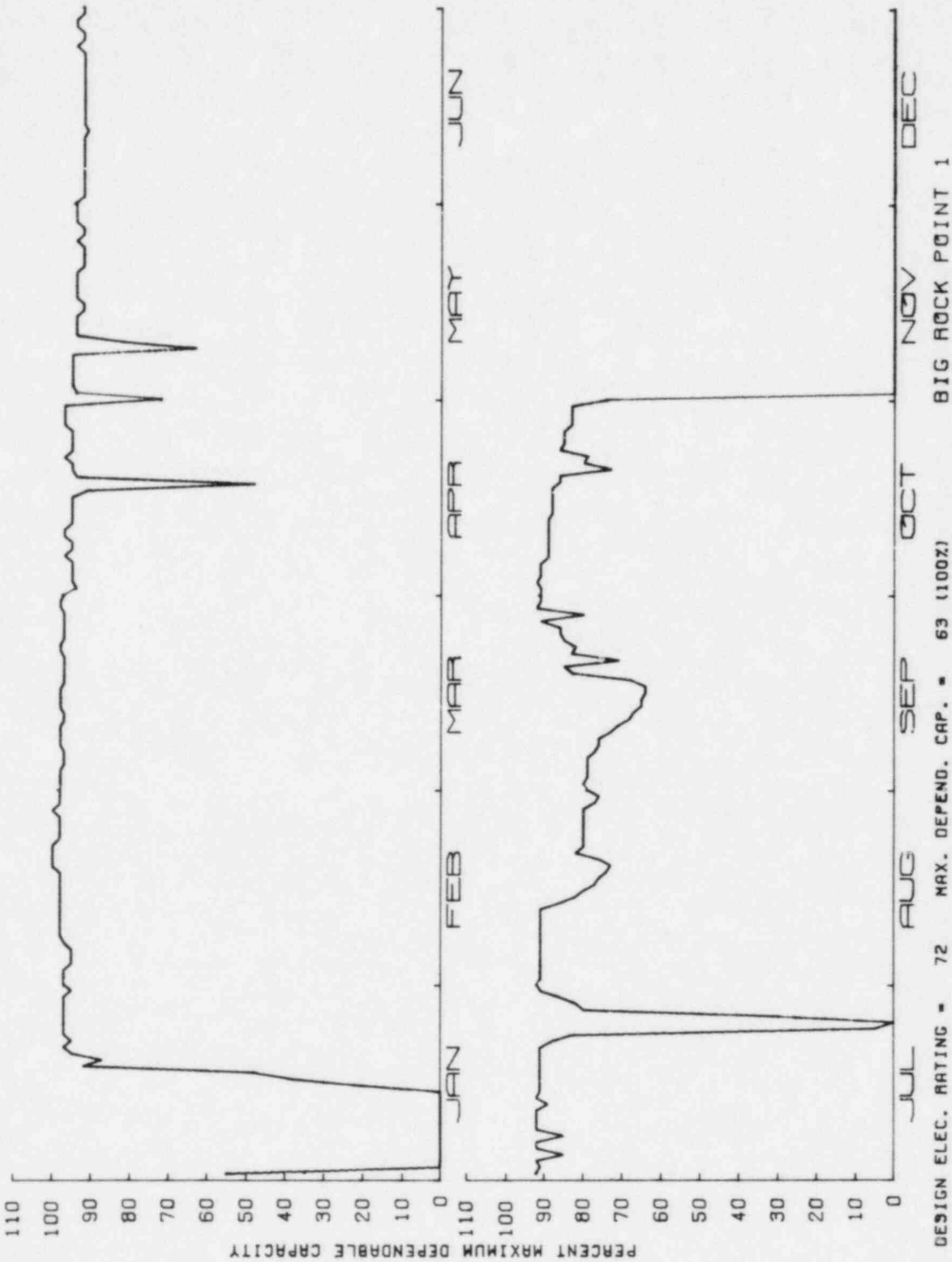
II. Highlights

The unit remained off-line until January 12 per requirements of NUREG-0578. Maximum dependable capacity was decreased from 65 to 63 NWe (net) due to thermal-hydraulic limit margins of the fuel. The plant operated at or near full power from January 15 until October 31 (with the exception of a required scram test on July 24), when refueling commenced. The unit had an availability factor of 78.9 in 1980.

[a]Includes 295.7 h in 1980 from continued shutdown of 12/31/79.

DETAILS OF PLANT OUTAGES FOR BIG ROCK POINT

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/31/79 (cont.)	295.7	S	Implement requirements of NUREG-0578 (TMI related modifications).	D	4	Instrumentation and controls (IB)	Instrumentation and controls
1	1/13	3.7	F	Failure of intermediate power range monitor.	A	3	Instrumentation and controls (IB)	Instrumentation and controls
2	1/13	4.5	F	Failure of intermediate power range monitor.	A	3	Instrumentation and controls (IB)	Instrumentation and controls
3	1/13	15.2	F	Trip of intermediate power range monitor on period due to prompt effect.	H	3	Instrumentation and controls (IB)	Instrumentation and controls
4	1/15	15.4	F	Failure of intermediate pressure regulator.	A	3	Reactor coolant (CC)	Instrumentation and controls
5	7/24	52.0	S	Scram testing per IE Bulletin 80-17.	D	1	Reactor (RB)	Control rods
6	10/31	1464.6	S	Refueling.	C	1	Reactor (RC)	Fuel elements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DESIGN ELEC. RATING = 72
MAX. DEPEND. CAP. = 63 (100%)
BIG ROCK POINT 1

BROWNS FERRY 1

I. Summary

Description

Location: Decatur, Alabama
Docket No.: 50-259
Reactor type: BWR
Maximum dependable capacity (MWe-net): 1,065
Commercial operation: 8/01/74
Years operating experience: 7.2

Performance

Net electrical energy generated (MWh): 6,061,849
Unit availability factor (%): 72.6
Unit capacity factor (using MDC): 64.8
Unit capacity factor (%) (using design MWe): 64.8

Outages

Total No.: 23
Forced: 20
Scheduled: 3
Total hours: 2,404.1 (27.4%)
Forced: 497.8 (5.7%)
Scheduled: 1,906.3 (21.7%)

II. Highlights

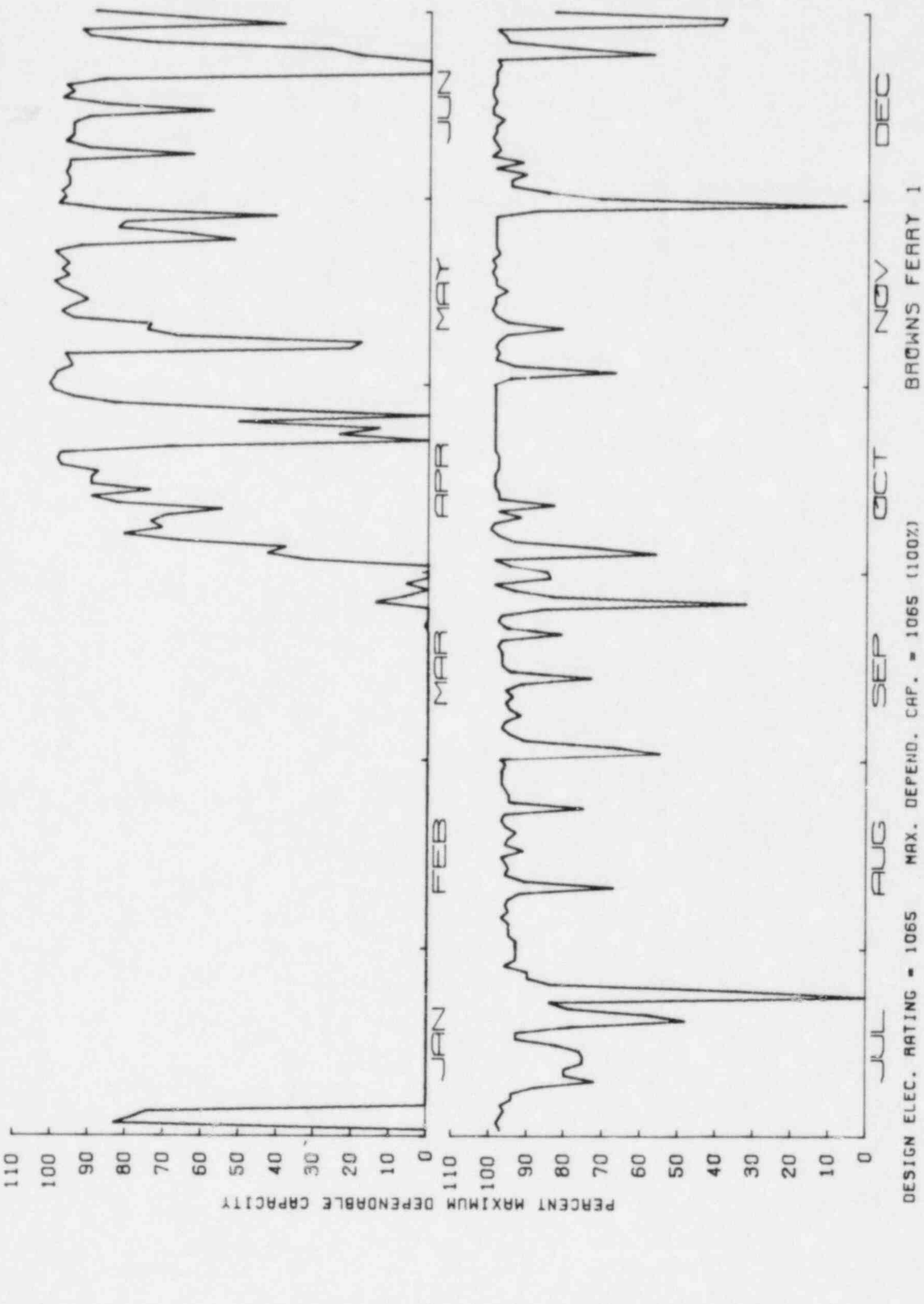
Refueling lasted from January 3 until March 22. On restart, numerous short outages and power reductions, including ones for repair of the main transformer and the "B" recirculation pump thrust bearing, slowed return to full power. The NRC-required manual and automatic scram tests were performed on July 23 and 24. The unit had an availability factor of 72.6 in 1980.

DETAILS OF PLANT OUTAGES FOR BROWNS FERRY 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/03	1888.1	S	Refueling.	C	2	Reactor (RC)	Fuel elements
2	3/22	76.4	F	Blown seal on B recirculation pump.	A	2	Reactor coolant (CB)	Pumps
3	3/27	54.8	F	Generator neutral overvoltage due to problems with main transformers.	A	3	Electric power (EB)	Transformers
4	3/29	74.4	F	High temperature on B recirculation pump thrust bearing.	A	2	Reactor coolant (CB)	Pumps
5	4/20	36.5	F	Maintenance on FW control valve.	A	2	Reactor coolant (CH)	Valves
6	4/23	15.0	F	Balance main turbine.	B	2	Steam and power conversion (HA)	Turbines
7	4/24	30.5	F	Low oil level signal in B recirculation pump due to loose wires (LER 80-35).	A	2	Instrumentation and controls (IF)	Electrical conductors
8	5/06	24.4	F	Generator load rejection due to electrohydraulic control pressure fluctuations during test of intermediate valves.	B	3	Steam and power conversion (HA)	Valves
9	5/07	7.7	F	Blown fuse in A FW inverter and spurious B FW high level signal.	A	3	Reactor coolant (CH)	Generators (inverters)
10	5/27	10.7	F	Load rejection while replacing a PK block in the main transformer.	G	3	Electric power (EB)	Transformers
11	6/17	10.0	F	Moisture separator high level.	A	3	Reactor coolant (CC)	Heat exchangers (MSR)
12	6/17	36.8	F	Recirculation pump low oil level alarm.	A	2	Reactor coolant (CB)	Pumps
13	6/23	8.1	F	Oil leak in EHC.	A	2	Steam and power conversion (HA)	Mechanical function units

DETAILS OF PLANT OUTAGES FOR BROWNS FERRY 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
14	6/24	11.6	F	While performing control valve closure SI, a miswired fast closure pressure switch caused a scram (LER 80-50).	G	3	Reactor coolant (CH)	Relays
15	6/24	10.0	F	While performing control valve closure SI, a miswired fast closure pressure switch caused a scram (LER 80-50). Some control rods failed to insert fully (see IE Bulletin 80-17).	G	2	Reactor coolant (CH)	Relays
16	7/22	17.6	F	EHC system failure.	A	3	Steam and power conversion (HA)	Mechanical function units
17	7/23	11.8	S	Manual scram to test control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
18	7/24	6.4	S	Automatic scram to test control rods per IE Bulletin 80-17.	D	3	Reactor (RB)	Control rods
19	9/01	10.0	F	Turbine trip due to stop valve closure.	A	3	Steam and power conversion (HA)	Valves
20	9/24	12.8	F	Load rejection due to negative ground on main transformer sudden pressure relay.	A	3	Electric power (EB)	Transformers
21	10/03	9.5	F	Turbine trip on low oil level when person inadvertently checked level switch.	G	3	Steam and power conversion (HA)	Turbines
22	11/28	19.3	F	Turbine trip while transferring the shutdown bus from unit 1 station service to unit 2 station service in preparation for steam leak maintenance.	B	3	Electric power (EBO)	Turbines
23	12/29	21.7	F	Replace filters in stator cooling system.	A	2	Steam and power conversion (HA)	Filters



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

BROWNS FERRY 2

I. Summary

Description	Performance	Outages
Location: Decatur, Alabama	Net electrical energy generated (MWh): 5,618,838	Total No.: 19
Docket No.: 50-260	Unit availability factor (%): 69.2	Forced: 15
Reactor type: BWR	Unit capacity factor (using MDC): 60.1	Scheduled: 4
Maximum dependable capacity (MWe-net): 1,065	Unit capacity factor (%) (using design MWe): 60.1	Total hours: 2,707.3 (30.8%)
Commercial operation: 3/01/75		Forced: 710.0 (8.1%)
Years operating experience: 6.3		Scheduled: 1,997.3 (22.7%)

II. Highlights

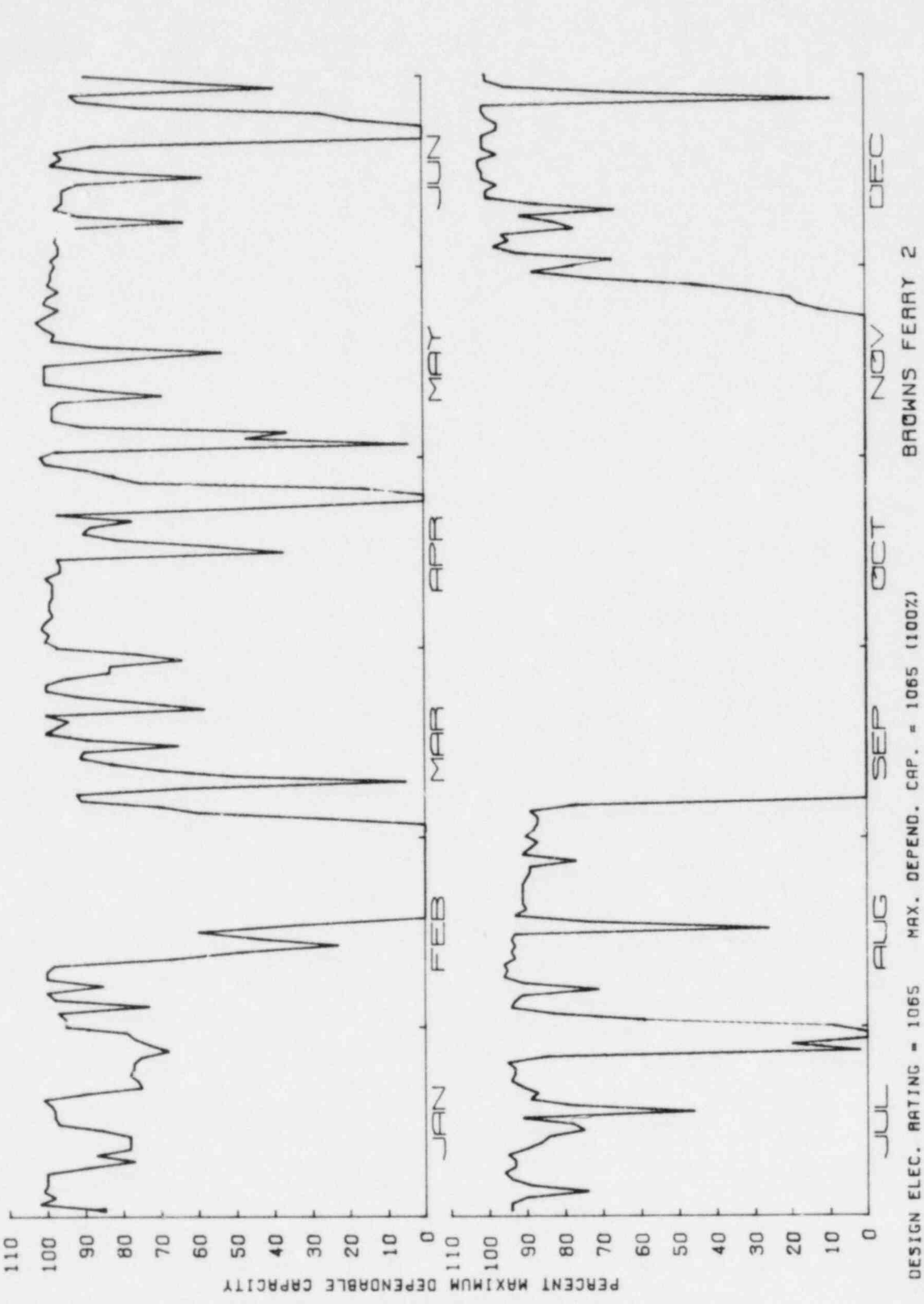
In February, the unit experienced three recirculation pump trips and two reactor protection system channel trips before being shut down on the February 15 for maintenance on the high-pressure coolant system. The shut down lasted two weeks. On April 22, the unit was taken off-line for 3 d when a personnel error caused a high reactor water level during a safety injection. On July 27, the NRC-required manual and automatic scram tests were performed. The unit operated until September 5, when refueling commenced. The unit returned on-line on November 22 and operated at or near full power for most of December.

DETAILS OF PLANT OUTAGES FOR BROWNS FERRY 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/10	10.1	F	RPS channel trip.	H	3	Instrumentation and controls (IA)	Instrumentation and controls
2	2/12	17.9	F	RPS channel trip.	H	3	Instrumentation and controls (IA)	Instrumentation and controls
3a.	2/15	346.5	F	Maintenance of the main cooling system; repaired HPCI pump turbine bearing pedestals (LER 80-10).	H	3	Engineered safety features (SF)	Pumps
3b.	3/01	57.5	F	Primary containment leakage.	A	4	Engineered safety features (SD)	Penetrations
4	3/09	20.5	F	SI channel B solenoids deenergized (cause unknown), causing reactor low water level.	B	3	Instrumentation and controls (IA)	Relays
5	4/15	8.0	F	Low reactor water level due to inadvertent opening of main steam flow control circuit.	G	3	Reactor coolant (CB)	Instrumentation and controls
6	4/22	75.0	F	High reactor water level during SI.	G	3	Engineered safety features (SF)	Instrumentation and controls
7	5/02	20.2	F	Repair steam leak in steam tunnel.	A	2	Reactor coolant (CC)	Pipes, fittings
8	5/03	14.8	F	Repair No. 1 control valve.	A	2	Steam and power conversion (HA)	Valves
9	5/16	8.8	F	Repair control valve servo-motor in electrohydraulic control system.	A	2	Steam and power conversion (HA)	Motors
10	6/19	87.2	S	Scheduled maintenance.	B	2	Other (XX)	Other

DETAILS OF PLANT OUTAGES FOR BROWNS FERRY 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11	6/24	11.0	F	Condenser low vacuum during maintenance on condenser drain valve.	B	3	Steam and power conversion (HC)	Heat exchangers (condenser)
12	6/28	[illegible]4	F	Condenser low vacuum.	B	3	Steam and power conversion (HC)	Heat exchangers (condenser)
13	7/17	[illegible]	F	Low reactor water level due to loss of preferred MG set due to over voltage relay malfunction.	A	3	Electric power (EE)	Relays
14	7/27	12.0	S	Manual scram to test control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
15	7/27	16.1	S	Automatic scram to test control rods per IE Bulletin 80-17.	D	3	Reactor (RB)	Control rods
16	7/28	64.8	F	Failure of condensate short cycle valves.	A	2	Reactor coolant (CH)	Valves
17	8/16	14.3	F	Personnel error while performing SI 4.2.A-8 (reactor building isolation logic test).	G	3	Engineered safety features (SD)	Instrumentation and controls
18	9/05	1882.0	S	Refueling.	C	2	Reactor (RC)	Fuel elements
19	12/27	22.6	F	Main steam line high radiation trip was reset improperly.	A	3	Steam and power conversion (CC)	Instrumentation and controls


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

BROWNS FERRY 3

I. Summary

Description

Location: Decatur, Alabama
Docket No.: 50-296
Reactor type: BWR
Maximum dependable capacity (MWe-net): 1,065
Commercial operation: 3/01/77
Years operating experience: 4.3

Performance

Net electrical energy generated (MWh): 6,936,550
Unit availability factor (%): 79.1
Unit capacity factor (using MDC): 74.1
Unit capacity factor (%) (using design MWe): 74.1

Outages

Total No.: 11
Forced: 8
Scheduled: 3
Total hours: 1,831.9 (20.9%)[a]
Forced: 615.4 (7.0%)
Scheduled: 1,216.5 (13.9%)[a]

II. Highlights

The unit was shut down until January 15 for installation of overhead cables from cooling tower transformers to bus tie boards. Operation was at or near full power with infrequent and short outages until June 28, when during a manual scram to prepare for feedwater pipe maintenance, about one-third of the control rods did not enter the core immediately. It took operators 10 min to get all rods to the bottom of the core. This event precipitated NRC's order in IE Bulletin 80-17 for scram testing at all BWRs. The unit returned to service July 12 and operated until November 23, when it was shut down for refueling. Browns Ferry 3 generated the second highest amount of electricity (net megawatt hours) of any reactor in the United States in 1980.

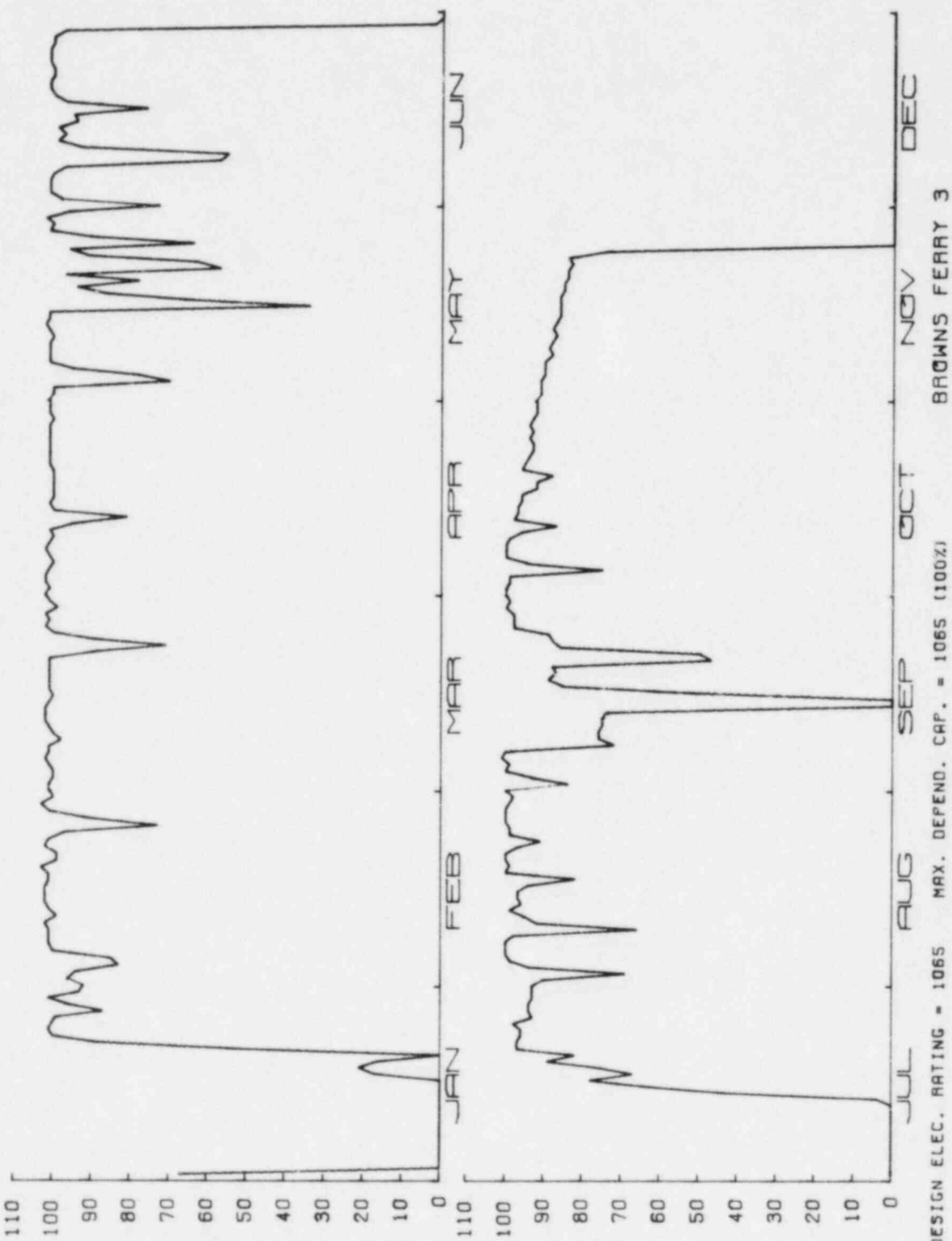
[a]Includes 375.3 h from continued 12/30/79 outage.

DETAILS OF PLANT OUTAGES FOR BROWNS FERRY 3

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/30/79 (cont.)	375.3	F	Installation of overhead cables from cooling tower transformers to bus tie boards.	A	4	Electric power (EA)	Electrical conductors
1	1/17	14.7	F	No. 2 main steam stop valve closes during testing causing a spike in reactor pressure; reactor tripped on high-high flux.	A	3	Reactor (RB)	Circuit closures/ interrupters
2	1/18	45.1	F	No. 2 main steam stop valve closes during testing causing a spike in reactor pressure; reactor tripped on high-high flux. Unit remains down for maintenance on the EHC system.	A	3	Reactor (RB)	Circuit closures/ interrupters
3	5/15	14.5	F	Load rejection due to generator field ground.	G	3	Electric power (EB)	Generators (main generator)
4	5/21	11.0	F	Indication of high flux during reactor low water level SI.	B	3	Reactor (RB)	Instrumentation and controls
5	6/07	13.1	F	Ground protection relay trip.	A	3	Electric power (EA)	Relays
6a.	6/28	76.0	F	Maintenance on FW piping: during this scram about 1/3 of the control rods did not enter the core immediately.	A	2	Reactor coolant (CH)	Pipes, fittings
6b.	7/01	246.7	S	Investigation and testing of control rods and scram discharge volume.	D	4	Reactor (RB)	Control rods
7	7/11	20.8	S	Manual scram to test control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
8	7/12	34.7	S	Automatic scram to test control rods per IE Bulletin 80-17.	D	3	Reactor (RB)	Control rods

DETAILS OF PLANT OUTAGES FOR BROWNS FERRY 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9a.	9/12	41.2	F	Maintenance on MSIV 1-26.	A	2	Reactor coolant (CD)	Valves
9b.	9/14	9.5	F	Maintenance on FW valve 3-219A.	A	4	Reactor coolant (CH)	Valves
10.	9/20	15.0	F	Loss of preferred power when the 480-V shutdown board A failed to transfer back to normal, causing loss of RPS MG Set A (LER 80-39).	A	3	Electric power (EB)	Circuit closers/ interrupters
11.	11/23	914.3	S	Refueling.	C	1	Reactor (RC)	Fuel elements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DESIGN ELEC. RATING = 1065 MAX. DEPEND. CAP. = 1065 (100%) BROWNS FERRY 3

BRUNSWICK 1

I. Summary

Description

Location: Southport, North Carolina
Docket No.: 50-325
Reactor type: BWR
Maximum dependable capacity (MWe-net): 790
Commercial operation: 3/18/77
Years operating experience: 4.1

Performance

Net electrical energy generated (MWh): 3,939,624
Unit availability factor (%): 68.9
Unit capacity factor (using MDC): 56.8
Unit capacity factor (%) (using design MWe): 54.6

Outages

Total No.: 12
Forced: 9
Scheduled: 3
Total hours: 2,734.9 (31.2%)
Forced: 990.2 (11.3%)
Scheduled: 1,744.7 (19.9%)

II. Highlights

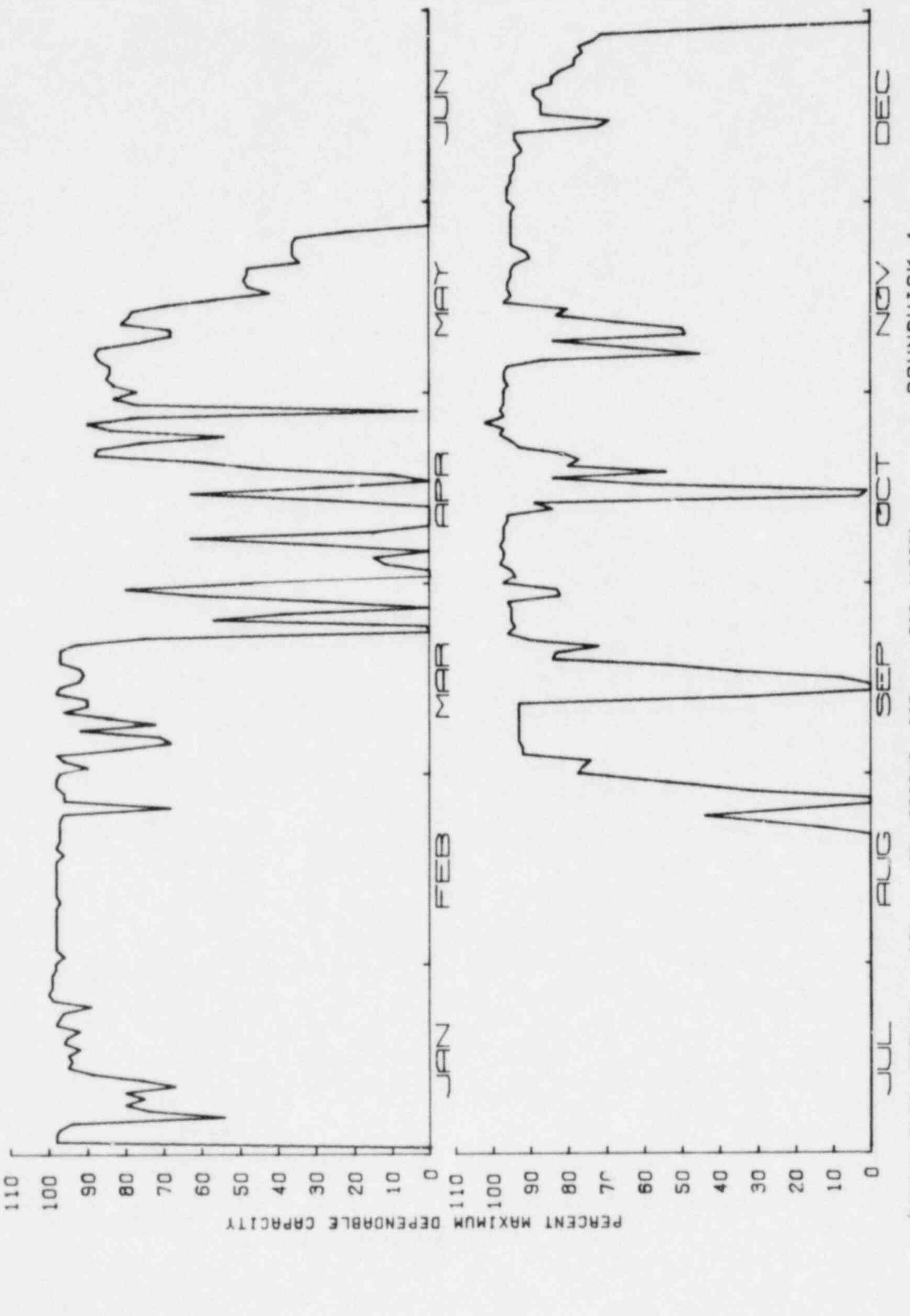
Operation continued until refueling commenced on May 26. The unit returned on-line on August 22, after the refueling outage was extended past its scheduled completion date due to unanticipated maintenance and regulatory problems.

DETAILS OF PLANT OUTAGES FOR BRUNSWICK 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	3/23	43.6	F	Erroneous reactor high water level indications due to stuck pen recorder.	H	3	Instrumentation and controls (IA)	Instrumentation and controls
2	3/26	39.8	F	Excessive leakage on drywell floor; two valves repaired. Nitrogen inert-problem due to inoperable auxiliary boilers extended the outage.	B	1	Reactor coolant (CB)	Valves
3	3/31	61.6	F	Low reactor water level signal due to apparent loss of 1D bus.	A	3	Electric power (EB)	Electrical conductors
4a.	4/05	14.2	F	Turbine trip on high vibration during control valves periodic test.	B	3	Reactor coolant (CC)	Valves
4b.	4/05	16.0	F	Excessive drywell leakage.	A	4	Reactor coolant (CB)	Valves
5	4/08	124.6	F	Turbine trip during DG test due to electrical ground on No. 2 DG and low electrical ground in the distribution system.	B	3	Electric power (EC)	Electrical conductors
6	4/15	50.5	F	Turbine trip during DG test due to electrical ground on No. 2 DG and low electrical ground in the distribution system.	B	3	Electric power (EC)	Electrical conductors
7	4/26	19.7	F	Excessive drywell leakage; packing leak on recirculation line sample valve repaired.	A	1	Engineered safety features (SA)	Valves
8a.	5/26	1592.7	S	Refueling.	C	1	Reactor (RC)	Fuel elements
8b.	8/01	518.0	F	Refueling extended because of unanticipated maintenance and regulatory problems.	B	4	Other (XX)	Other

DETAILS OF PLANT OUTAGES FOR BRUNSWICK 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	8/25	59.1	F	Reactor coolant conductivity exceeded Tech Specs (LER 80-65).	H	2	Radioactive waste management (MA)	Not applicable
10	9/12	70.9	S	Replaced generator bearing and realigned exciter coupling.	B	1	Steam and power conversion (HA)	Generators (main generators, exciter)
11	10/14	43.1	F	High main steam line flow while testing the stop valves because too few main steam lines were available for this total flow test.	G	3	Reactor coolant (CC)	Valves
12	12/28	81.1	S	FW heater maintenance.	B	3	Reactor coolant (CH)	Heaters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DESIGN ELEC. RATING = 821 MAX. DEPEND. CAP. = 790 (100%)
BRUNSWICK 1

BRUNSWICK 2

I. Summary

Description

Location: Southport, North Carolina
Docket No.: 50-324
Reactor type: BWR
Maximum dependable capacity (MWe-net): 790
Commercial operation: 11/03/75
Years operating experience: 5.7

Performance

Net electrical energy generated (MWh): 1,864,957
Unit availability factor (%): 35.2
Unit capacity factor (using MDC): 26.9
Unit capacity factor (%) (using design MWe): 25.9

Outages

Total No.: 14
Forced: 11
Scheduled: 3
Total hours: 5,694.6 (64.8%)[a]
Forced: 591.5 (6.7%)
Scheduled: 5,103.1 (58.1%)[a]

II. Highlights

The unit was shut down until January 4 for modification of safety relief valves. Operation was routine until the unit was shut down for refueling on March 1. Unanticipated maintenance and regulatory problems extended this scheduled outage considerably.

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[a]Includes 106.4 h in 1980 from continued shutdown of 12/25/79.

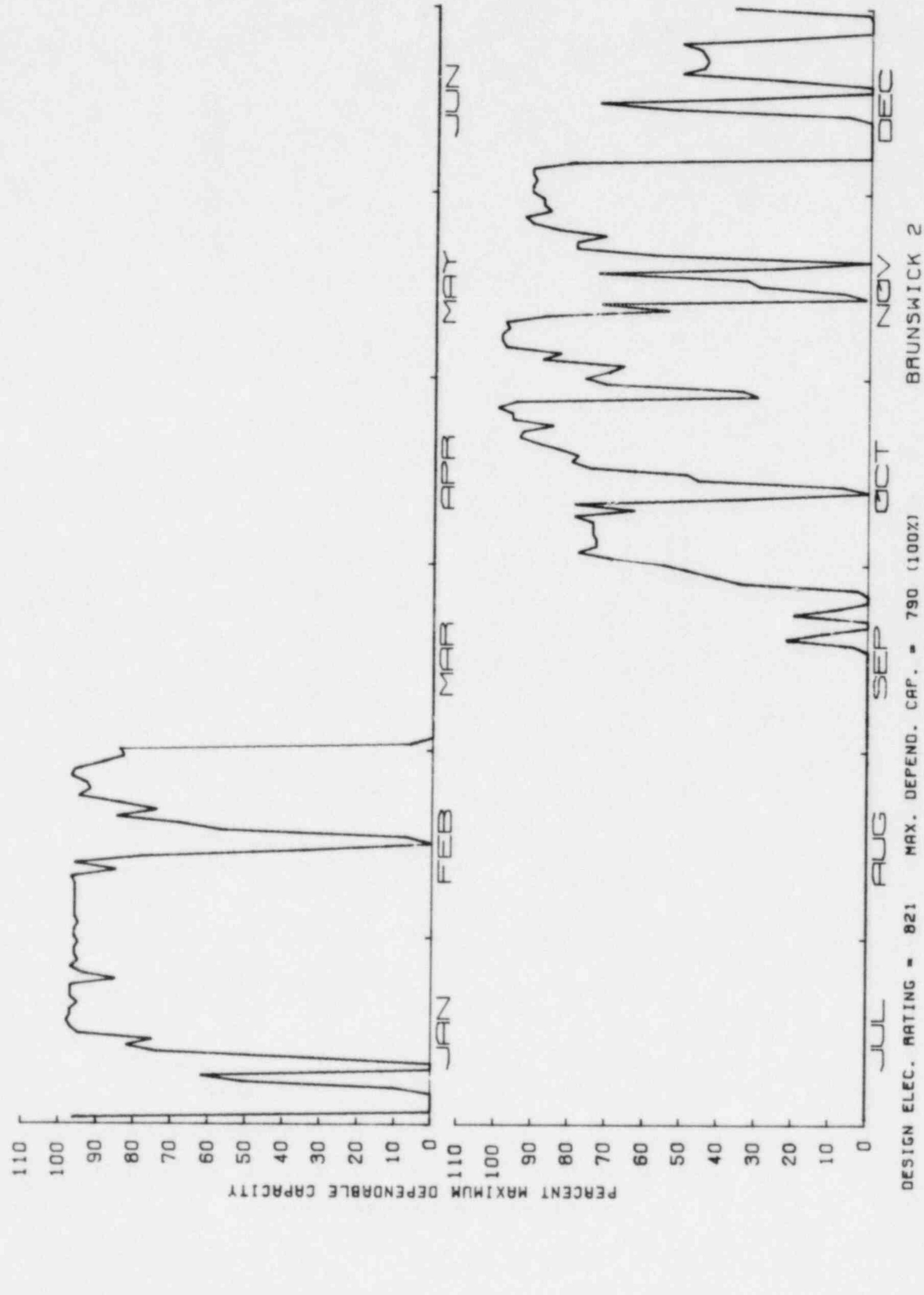
DETAILS OF PLANT OUTAGES FOR BRUNSWICK 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
a.	12/25/79 (cont.)	106.4	S	Safety relief valve position indicating system installed on all primary relief valves. Pipe snubber inspection also completed.	D	4	Engineered safety features (SH)	Valves
b.	12/25/79 (cont.)	*a*	F	The outage was extended due to a leak on a reactor recirculation test connection bushing.	B	4	Reactor coolant (CB)	Valves
c.	12/25/79 (cont.)	*a*	F	Reactor water chemistry went out of specs during attempt to establish a condenser vacuum.	H	4	Steam and power conversion (HG)	Other
1	1/07	51.9	F	High tailpipe temperature: safety relief valve replaced.	A	1	Engineered safety features (SH)	Valves
2	2/19	56.5	F	A slight mechanical jar (bump) of the isolation valve of the instrument being tested was transmitted to another instrument and caused scram.	B	3	Instrumentation and controls (IA)	Instrumentation and controls
3	3/01	4808.5	S	Refueling. Extended due to unanticipated maintenance and regulatory problems.	C	1	Reactor (RC)	Fuel elements
4	9/17	0.2	S	Turbine overspeed trip test.	B	1	Electric power (ED)	Turbines
5	9/19	68.6	F	No reason for the scram could be determined. No tests, maintenance, or other work was in progress. Replaced leaking pilot assemblies on safety relief valves.	A	3	Instrumentation and controls (IA)	Not applicable
6	9/23	76.9	F	High drywell floor drain leakage due to packing leak on valve E51-F007. Upon startup, a drain line from valve FW-FV46 leaked and was repaired.	A	1	Reactor coolant (CE)	Valves

DETAILS OF PLANT OUTAGES FOR BRUNSWICK 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7	10/11	54.2	F	APRM was not reset following I&C test PT-1.3.2P on channel A.	G	3	Instrumentation and controls (ID)	Instrumentation and controls
8	10/28	22.8	F	High reactor water level because of a problem in the FW control system. The inverter in the FW level control circuit tripped on high voltage; the high voltage trip point was reset to 147 V DC.	A	3	Reactor coolant (CH)	Circuit closers/interrupters
9	11/13	34.8	F	RPS power supply group scram due to an insulation breakdown and burn-through of RPS ground cable where it physically rested against a 120-V control power relay contact (LER 80-82).	A	3	Instrumentation and controls (IA)	Electrical conductors
10	11/15	12.2	F	Repair leak in heater drain pumps. Recirculation piping was leaking between heater drain pump 2C and the deaerator tank.	B	1	Reactor coolant (CB)	Pipes, fittings
11	11/18	37.5	F	Power load unbalance sensed by the turbine EHC system. Replaced the control intercept valve amplifier and trigger printed wire board.	A	3	Steam and power conversion (HA)	Instrumentation and controls
12	12/05	188.0	S	FW heater maintenance - plugged identified leaking tubes.	B	1	Reactor coolant (CH)	Heaters
13	12/16	68.9	F	Low vacuum caused by loss of emergency bus E-4.	A	3	Electric power (EB)	Electrical conductors
14	12/26	107.2	F	Reactor feed pump trip. Trigger assembly and dump valve were removed and machined at point of contact to restore dimension to as-built condition.	A	3	Reactor coolant (CH)	Pumps

[a]Total time included in No. a.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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
Maximum dependable capacity (MWe-net): 810
Commercial operation: 5/08/75
Years operating experience: 6.0

Performance

Net electrical energy generated (MWh): 4,533,957
Unit availability factor (%): 72.3
Unit capacity factor (using MDC): 63.7
Unit capacity factor (%) (using design MWe): 61.1

Outages

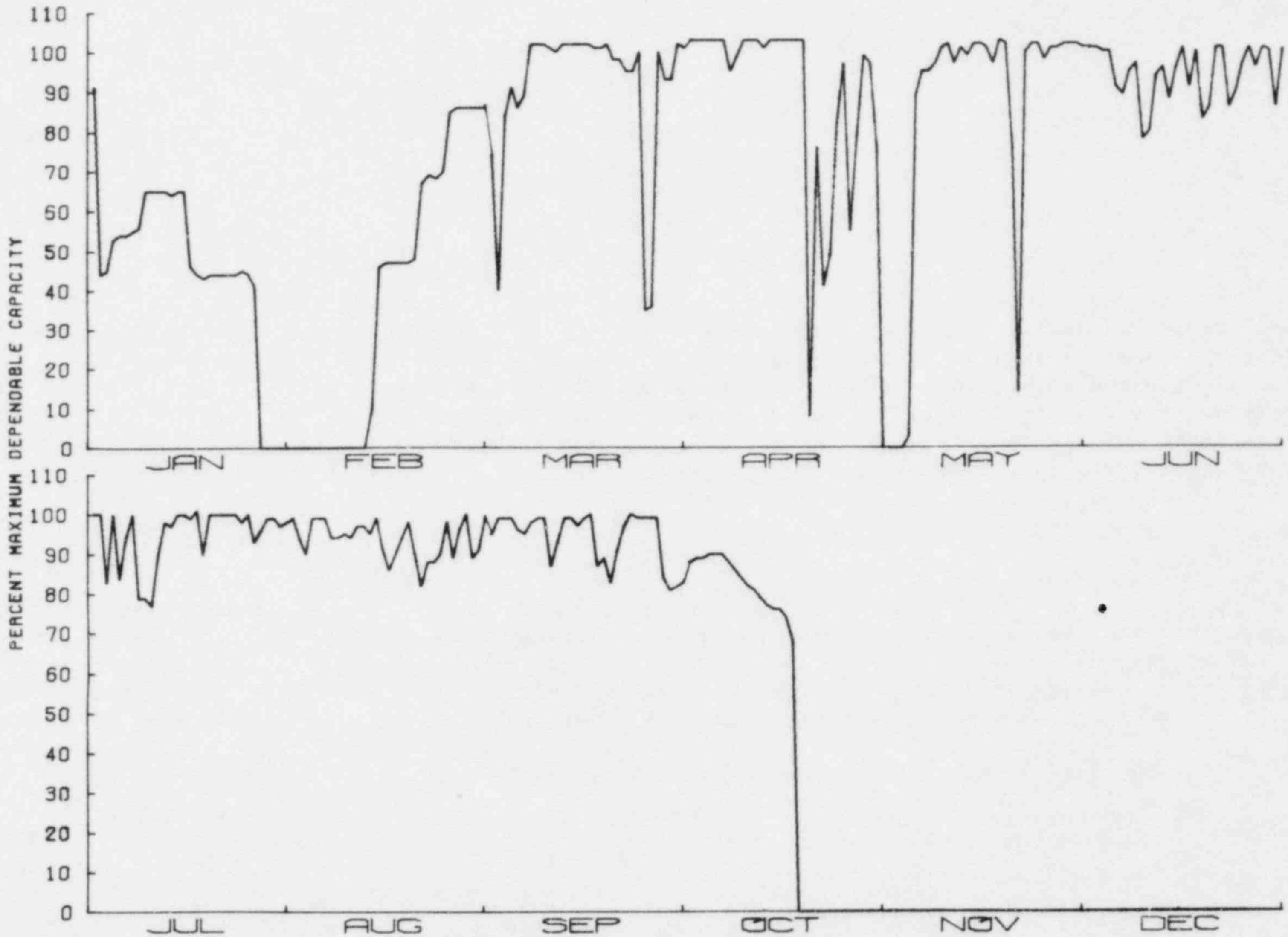
Total No.: 10
Forced: 6
Scheduled: 4
Total hours: 2,433.3 (27.7%)
Forced: 358.1 (4.1%)
Scheduled: 2,075.2 (23.6%)

II. Highlights

January operations were under reduced loads because of unequal power distribution in the core and turbine blade problems. Starting January 25, installation of NRC-required plant modifications took 17 d. The unit was restricted to 97.5% power (790 MWe-net) through February because of turbine blade problems. Power reductions were necessary in March and July to control condenser tube leaks. A refueling outage commenced on October 18 and continued through the end of the year when turbine thrust bearings forced a continuation of the outage on December 25.

DETAILS OF PLANT OUTAGES FOR CALVERT CLIFFS 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/25	412.0	S	TMI-related modification required.	D	1	Other (XX)	Other
2	3/01	9.7	F	Loss of all circulating water pumps due to leak in No. 14 circulating water pump cooler onto the high water level trip circuitry in the intake structure.	A	3	Steam and power conversion (HF)	Pumps
3	3/25	24.1	F	Voltage instability on the reactor trip bus.	A	3	Instrumentation and controls (IA)	Control rods
4	4/19	19.5	S	Hydraulic control valve replaced for No. 1 main turbine intercept valve.	B	1	Steam and power conversion (HA)	Valves
5	4/21	15.5	F	Turbine/reactor trip due to voltage swings on the motor generator sets for the control element drive system.	A	3	Reactor (RB)	Generators (MG set)
6	4/25	13.8	F	Undervoltage to reactor trip breakers while troubleshooting voltage swings on CRDM motor generator sets.	B	3	Instrumentation and controls (IA)	Generators (MG set)
7a.	4/29	11.4	S	Main turbine intercept valves No. 1 and No. 4 repair.	B	1	Steam and power conversion (HA)	Valves
7b.	4/30	104.2	F	Leak on No. 11B RCP control bleed-off line (80-24).	A	4	Reactor coolant (CB)	Pumps
8	5/20	23.2	F	Leak in the aftercooler on No. 12 instrument air compressor (LER 80-27).	A	1	Auxiliary process (PA)	Heat exchangers (coolers)
9	10/18	1632.3	S	Refueling, unit general inspection and TMI modifications.	C	1	Reactor (RC)	Fuel elements
10	12/25	167.6	F	Thrust bearing problems.	A	4	Steam and Power conversion (HA)	Turbines



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

CALVERT CLIFFS 2

I. Summary

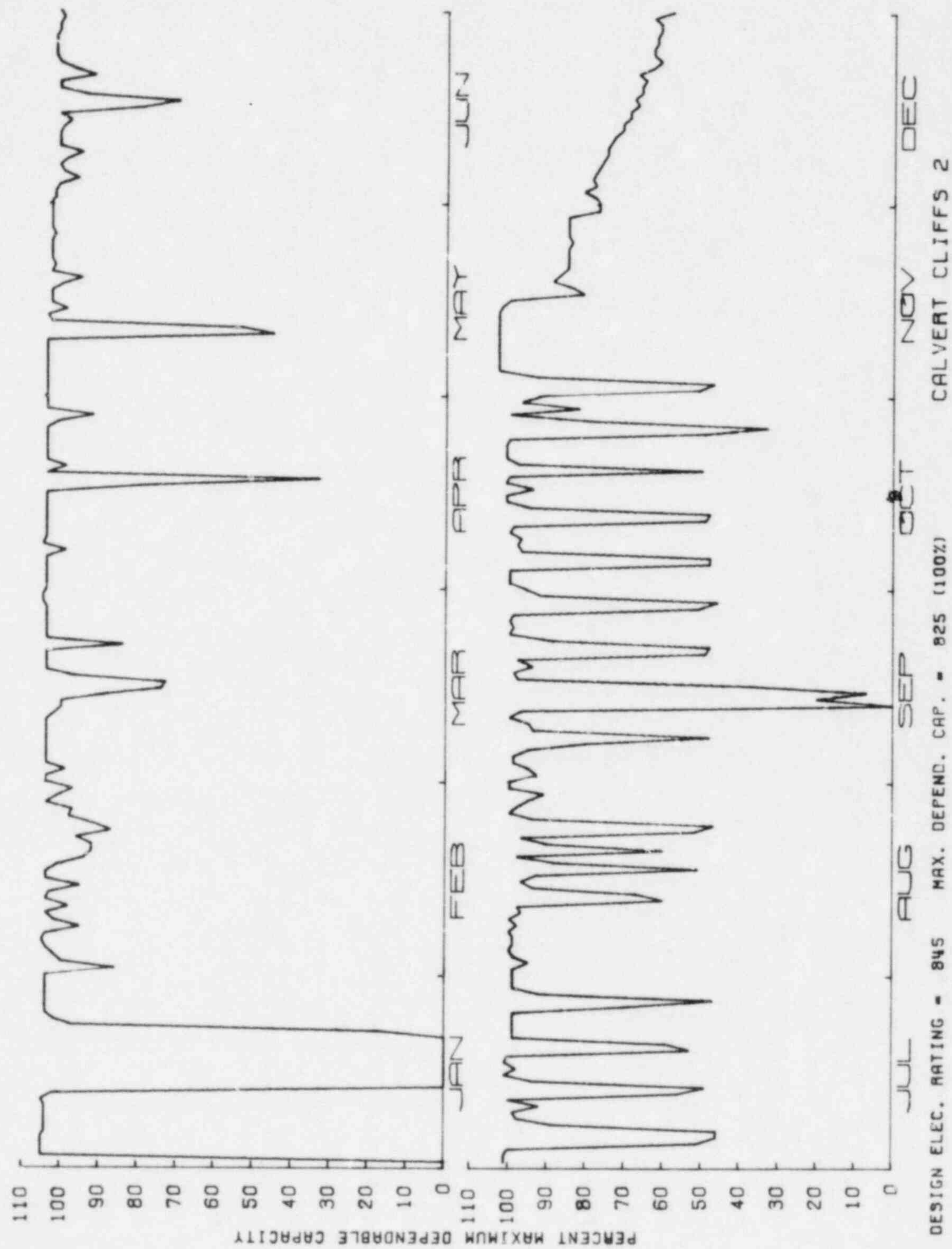
Description	Performance	Outages
Location: Lusby, Maryland	Net electrical energy generated (MWh): 6,412,954	Total No.: 8
Docket No.: 50-318	Unit availability factor (%): 96.0	Forced: 6
Reactor type: PWR	Unit capacity factor (using MDC): 88.5	Scheduled: 2
Maximum dependable capacity (MWe-net): 825	Unit capacity factor (%) (using design MWe): 86.4	Total hours: 355.4 (4.1%)
Commercial operation: 4/01/77		Forced: 94.1 (1.1%)
Years operating experience: 4.1		Scheduled: 261.3 (3.0%)

II. Highlights

The unit had an availability of 96% in 1980. Required TMI-related modifications resulted in 9.5 d of shutdown time in January, and a snubber inspection required 0.5 d in September. The other six shutdowns, all forced, were of shorter duration. Power reductions were necessary in February and July to control condenser tube leaks. Starting in July, power was occasionally reduced on weekends to conserve fuel. In November, the unit operated in a fuel conservation mode during most of the month, and by the end of December the unit had coasted down to 60% power in preparation for a January refueling.

DETAILS OF PLANT OUTAGES FOR CALVERT CLIFFS 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/12	227.3	S	TMI-related modifications required.	D	1	Other (XX)	Other
2	4/16	17.6	F	Letdown control valve repair.	A	1	Reactor coolant (CG)	Valves
3	5/10	20.1	F	Loss of excitation to all main circulating water pumps.	A	1	Steam and power conversion (HF)	Pumps
4	8/12	14.8	F	Loss of No. 22 circulating water pump.	A	1	Steam and power conversion (HF)	Pumps
5	8/20	5.3	F	High pressurizer pressure when a technician inadvertently initiated SG isolation signal.	G	3	Reactor coolant (CB)	Instrumentation and controls
6	9/12	34.0	S	Snubber inspection.	B	1	Engineered safety features (SH)	Shock suppressors and supports
7	9/14	27.5	F	Erratic level transmitter in SI tank (LER 80-43).	H	1	Engineered safety features (SF)	Instrumentation and controls
8	10/26	8.8	F	Loss of condenser vacuum when No. 22.	A	2	Steam and power conversion (HH)	Pumps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DESIGN ELEC. RATING = 845 MAX. DEPEND. CAP. = 825 (100%)
CALVERT CLIFFS 2

COOK 1

I. Summary

Description	Performance	Outages
Location: Bridgeman, Michigan	Net electrical energy generated (MWh): 6,461,827	Total No.: 11
Docket No.: 50-315	Unit availability factor (%): 73.7	Forced: 8
Reactor type: PWR	Unit capacity factor (using MDC): 70.5	Scheduled: 3
Maximum dependable capacity (MWe-net): 1,044	Unit capacity factor (%) (using design MWe): 69.8	Total hours: 2,310.1 (26.3%)[a]
Commercial operation: 8/27/75		Forced: 288.8 (3.3%)
Years operating experience: 5.9		Scheduled: 2,021.3 (23.0%)[a]

II. Highlights

Main feedwater pump problems caused a power reduction on January 24 for pump cleaning, a shutdown on January 27 due to a pump coupling failure, and a power reduction on February 16 for investigation of minor vibrations and oscillation in the pump. Refueling commenced on May 31 and was completed on August 5. On October 11, a safety injection occurred while testing the turbine control valve at 100% power.

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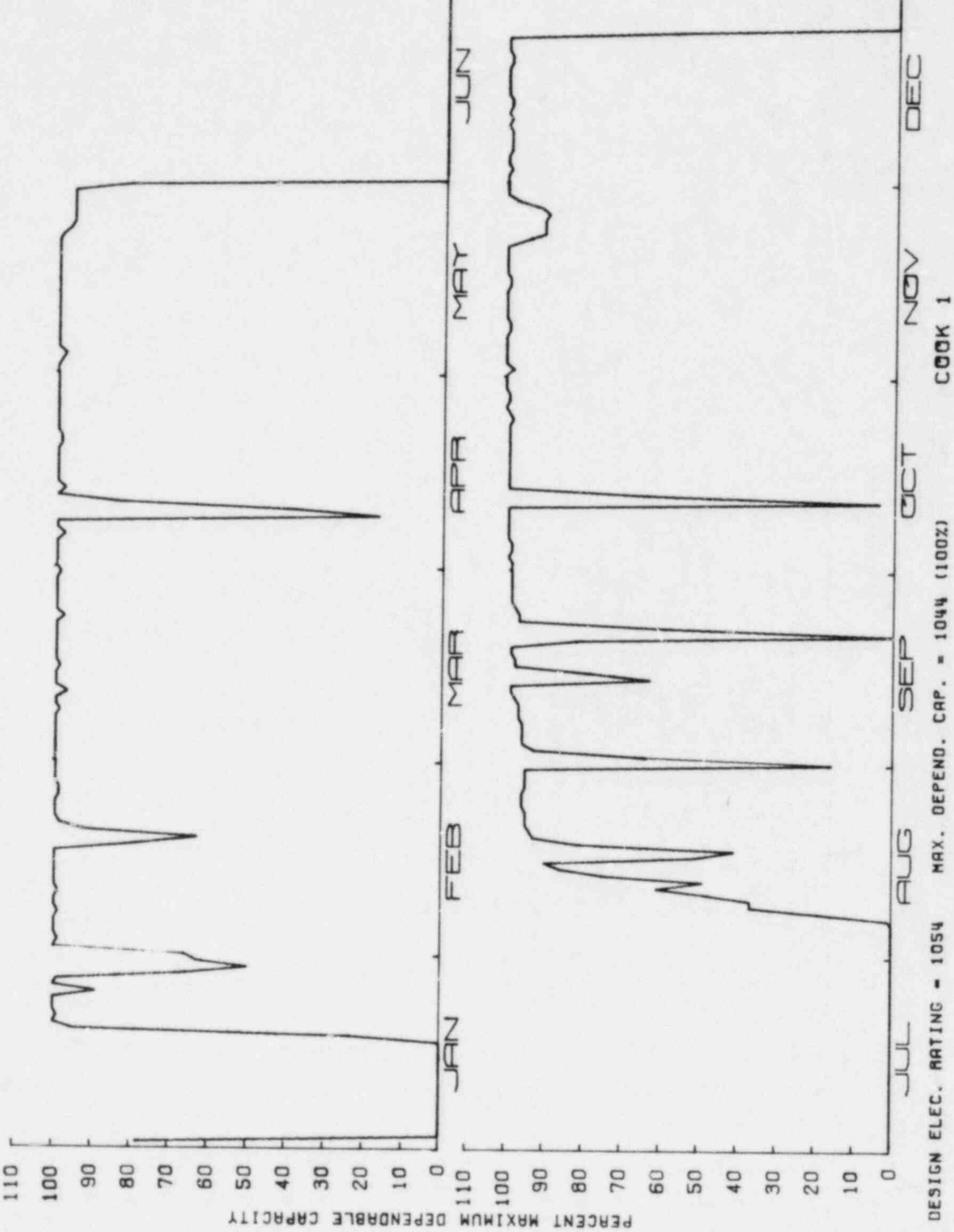
[a]Includes 396.2 h in 1980 from continued shutdown of 12/24/79.

DETAILS OF PLANT OUTAGES FOR COOK 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/24/79 (cont.)	396.2	S	Correct piping support discrepancies in the containment hydrogen skimmer system per IE Bulletin 79-14.	D	4	Engineered safety features (SE)	Shock suppressors and supports
1	1/27	7.2	F	SF/FFMM and low level in SG due to trip of MFWP turbine in turn due to pump coupling failure.	A	3	Reactor coolant (CH)	Mechanical function units
2	4/08	18.4	F	Low-low SG level due to lightning strike in switchyard which led to a load rejection.	H	3	Electric power (EB)	Electrical conductors
3	5/30	1624.1	S	Refueling.	C	1	Reactor coolant (RC)	Fuel elements
4	8/06	7.1	F	Extreme high level in right moisture separator during startup due to an alternate drain valve being closed.	G	3	Steam and power conversion (HB)	Heat exchangers (MSR)
5	8/07	1.0	S	Turbine overspeed trip testing.	B	1	Steam and power conversion (HA)	Turbines
6	8/16	6.2	F	False steam flow/FW flow mismatch and low SG level during instrument surveillance testing.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
7	8/16	2.2	F	High vibration on No. 3 turbine bearing.	B	1	Steam and power conversion (HA)	Turbines
8	8/31	18.3	F	Inverter failure on vital AC instrument bus channel IV (LER 80-20). Safety injection and steam line isolation occurred.	A	3	Electric power (ED)	Generators (inverters)

DETAILS OF PLANT OUTAGES FOR COOK 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	9/19	28.3	F	Repair of sensing line leak in No. 3 SG narrow range level channel BLP-122 (LER 80-23).	B	1	Instrumentation and controls (IB)	Instrumentation and controls
10	10/11	24.0	F	High differential pressure between steam leads caused safety injection actuation. This occurred while testing turbine control valve at 100% power. Control valve oscillations became excessive, creating SG pressure fluctuations. SI was reset and pumps were shutdown after 12 min of operation.	B	3	Steam and power (HA)	Valves
11	12/24	177.1	F	Ice condenser surveillance tests, miscellaneous maintenance, and inspection of SG tube lane blocking devices. Five SG tubes plugged.	B	1	Steam and power conversion (HB)	Heat exchangers (steam generators)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DESIGN ELEC. RATING = 1054 MAX. DEPEND. CAP. = 1044 (100%)
COOK 1

COOK 2

I. Summary

Description	Performance	Outages
Location: Bridgeman, Michigan	Net electrical energy generated (MWh): 6,691,753	Total No.: 12
Docket No.: 50-316	Unit availability factor (%): 74.4	Forced: 11
Reactor type: PWR	Unit capacity factor (using MDC): 70.4	Scheduled: 1
Maximum dependable capacity (MWe-net): 1,082	Unit capacity factor (%) (using design MWe): 69.3	Total hours: 2,246.2 (25.6%)[a]
Commercial operation: 7/01/78		Forced: 1,400.4 (16.0%)
Years operating experience: 2.8		Scheduled: 845.8 (9.6%)[a]

II. Highlights

The unit remained shut down through January 19 for refueling and for correcting discrepancies in safety-related piping. A two-week outage began on June 27 to make the unit 2 auxiliary feedwater system independent of unit 1. Short power reductions were necessary for investigation and repair of tube leaks in the east feed pump turbine condenser on July 25, August 19 and 24, and October 11, 12, and 17. A 7.5-week outage began on October 18 to repair the main electrical generator.

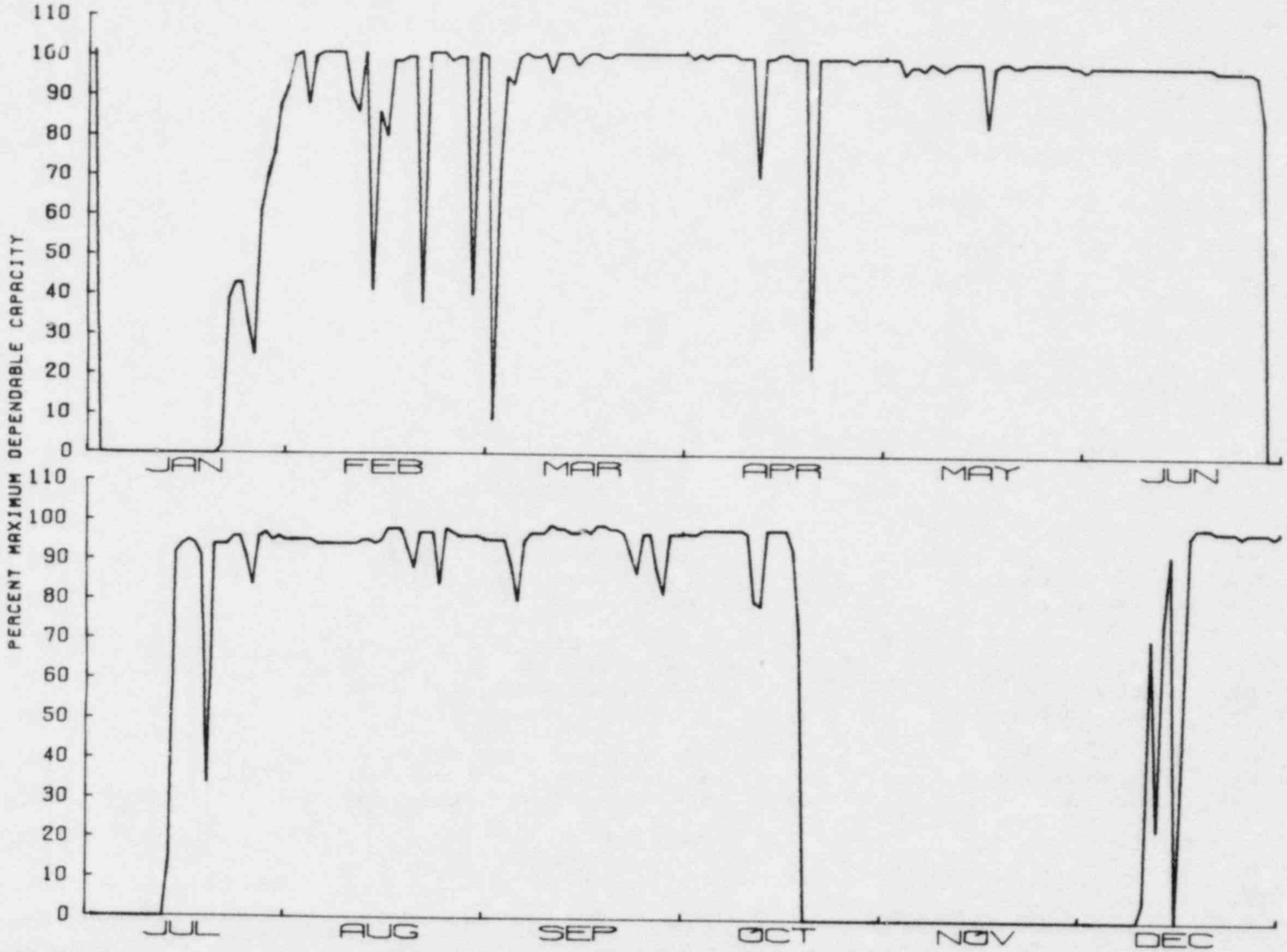
[a]Includes 468.7 h in 1980 from continued shutdown of 12/23/79.

DETAILS OF PLANT OUTAGES FOR COOK 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/23/79 (cont.)	468.7	S	Correct discrepancies in safety related piping per IE Bulletin 79-14.	D	4	Other (XX)	Pipes, fittings
1	1/20	9.6	F	SF/FFMM and low SG level while changing from auxiliary to main FW (LER 80-01).	H	3	Steam and power conversion (HB)	Heat exchangers (steam generators)
2	1/24	15.0	F	Low main condenser vacuum due to operator valving error.	G	3	Steam and power conversion (HC)	Valves
3	2/12	10.6	F	Spurious high level indication in moisture separator reheater.	A	3	Steam and power conversion (HA)	Instrumentation and controls
4	2/19	8.4	F	High vibration on turbine bearing No. 4 due to rapid temperature change while placing moisture separator reheater in service.	H	2	Steam and power conversion (HA)	Turbines
5	2/26	6.7	F	High vibration on turbine generator bearing No. 5 due to rapid temperature change in generator gases.	G	1	Steam and power conversion (HA)	Generators (main generators)
6	3/01	17.9	F	SG level control sluggish after turbine control was left in manual after valve testing.	A	3	Steam and power conversion (HH)	Valve operators
7	4/19	13.7	F	Steam flow/feedwater flow mismatch and low SG level due to failure SG feedwater regulating valve while recovering from an automatic isolation of mositure separator – reheater coil bundles.	A	3	Steam and power conversion (HH)	Valve operators
8	6/27	377.1	S	FW system modification in order to make each unit's AFWS independent.	H	1	Steam and power conversion (HH)	Pipes, fittings
9	7/19	7.9	F	Low-low SG level due to apparent loss of steam supply to east MFWP turbine.	A	3	Steam and power conversion (HH)	Pumps

DETAILS OF PLANT OUTAGES FOR COOK 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10	10/18	1270.3	F	Main generator neutral ground relay actuation; ground fault was in one of the generator stator bars.	A	3	Steam and power conversion (HA)	Generators (main generator)
11	12/12	12.1	F	Low condenser vacuum. During condenser tube leakage repairs, loose tube plug was removed where tube had previously been removed.	H	3	Steam and power conversion (HH)	Heat exchangers (condenser)
12	12/14	28.2	F	Loss of generator excitation due to failure of pilot exciter which was found on fire. Reactor tripped on RCP bus undervoltage and was followed by blackout, startup of emergency diesel generators, and load sequence.	A	3	Steam and power conversion (HA)	Generators (exciter)



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

COOPER

I. Summary

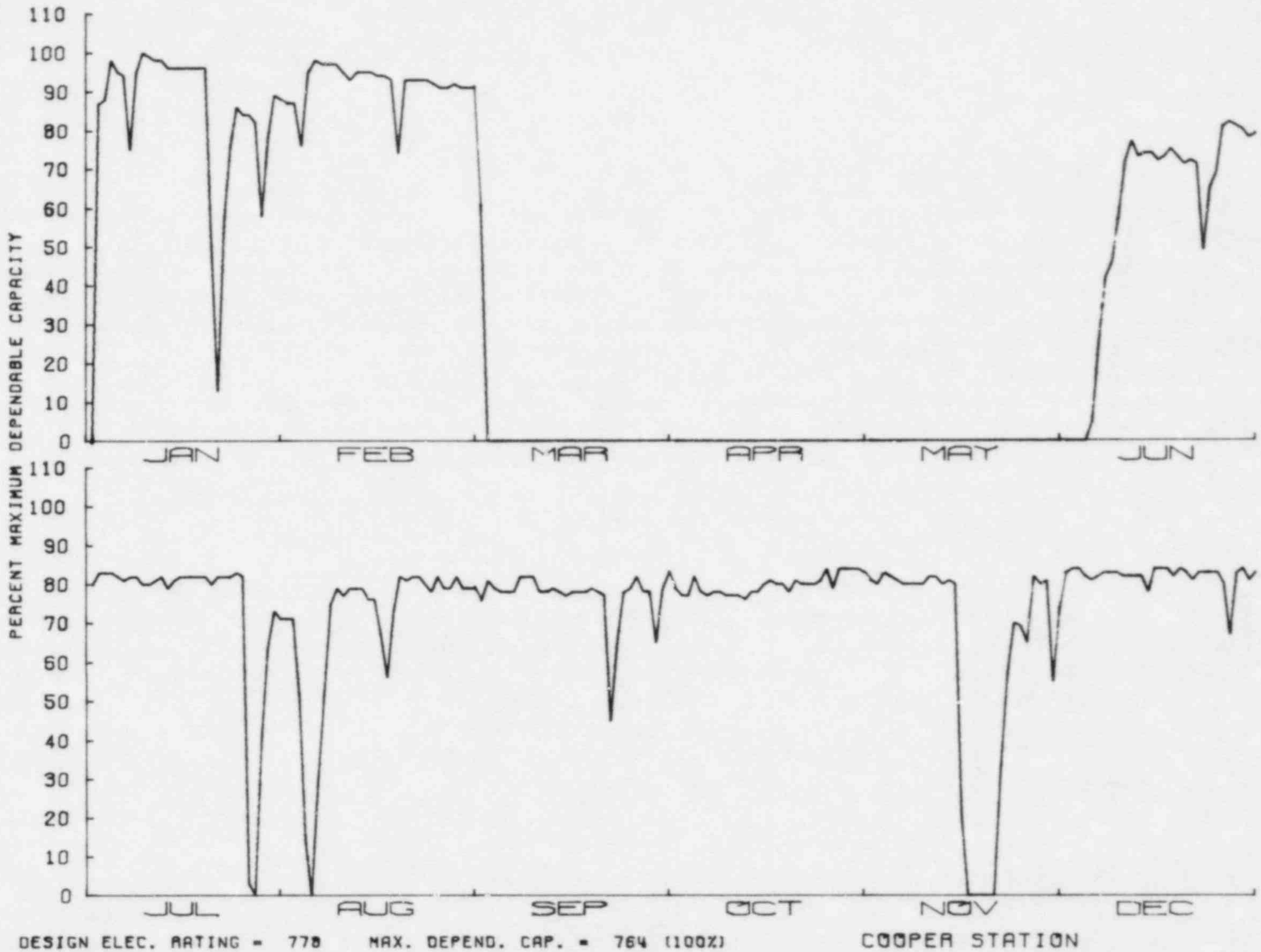
Description	Performance	Outages
Location: Brownville, Nebraska	Net electrical energy generated (MWh): 3,788,053	Total No.: 6
Docket No.: 50-298	Unit availability factor (%): 71.1	Forced: 4
Reactor type: BWR	Unit capacity factor (using MDC): 56.4	Scheduled: 2
Maximum dependable capacity (MWe-net): 764	Unit capacity factor (%) (using design MWe): 55.4	Total hours: 2,541.8 (28.9%)
Commercial operation: 7/01/74		Forced: 135.7 (1.5%)
Years operating experience: 6.6		Scheduled: 2,406.1 (27.4%)

II. Highlights

Power reductions were necessary to adjust control rod patterns on January 6 and 27, February 17, June 22, August 3, September 21, and November 29. A 3.5 month refueling and maintenance outage commenced on March 1. During October, the unit operated at about 80% power because of utility imposed restriction for temporary turbine modifications.

DETAILS OF PLANT OUTAGES FOR COOPER

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/19	24.3	F	Erroneous indication of main generator ground fault led to turbine trip and reactor scram.	A	3	Electric power (EB)	Instrumentation and controls
2	3/01	2270.9	S	Refueling and maintenance.	C	2	Reactor (RC)	Fuel elements
3	6/04	17.4	F	Low RCS level following reactor feed pump overspeed trip due to malfunctioning controller on reactor feed pump.	A	3	Reactor coolant (CB)	Mechanical function units
4	7/26	46.0	S	Special testing of the scram discharge volume and associated system per IE Bulletin 80-17.	D	1	Reactor (RB)	Control rods
5	8/04	46.0	F	MSIV closure because of line fault on 345-kV distribution system due to electrical storm. Subsequent voltage transients affected the turbine control system computer. Main steam pressure control was lost. Group I isolation occurred on low pressure (LER 80-44).	A	3	Reactor coolant (CD)	Electrical conductors
6a.	11/15	48.0	F	Operators tripped both RPS channels while checking APRM trip settings. An unrelated malfunction in the turbine control system caused the main steam bypass valves to fail open allowing a rapid vessel depressurization and cooldown. Replaced a turbine control computer circuit board (LER 80-44).	G	3	Instrumentation and controls (IA)	Instrumentation and controls
6b.	11/17	89.2	S	Scheduled maintenance outage.	B	4	System code not applicable (ZZ)	Not applicable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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): 3,353,930	Total No.: 10
Docket No.: 50-302	Unit availability factor (%): 53.1	Forced: 8
Reactor type: PWR	Unit capacity factor (using MDC): 48.8	Scheduled: 2
Maximum dependable capacity (MWe-net): 782	Unit capacity factor (%) (using design MWe): 46.3	Total hours: 4,118.8 (46.9%)
Commercial operation: 3/13/77		Forced: 339.0 (3.9%)
Years operating experience: 3.9		Scheduled: 3,779.8 (43.0%)

II. Highlights

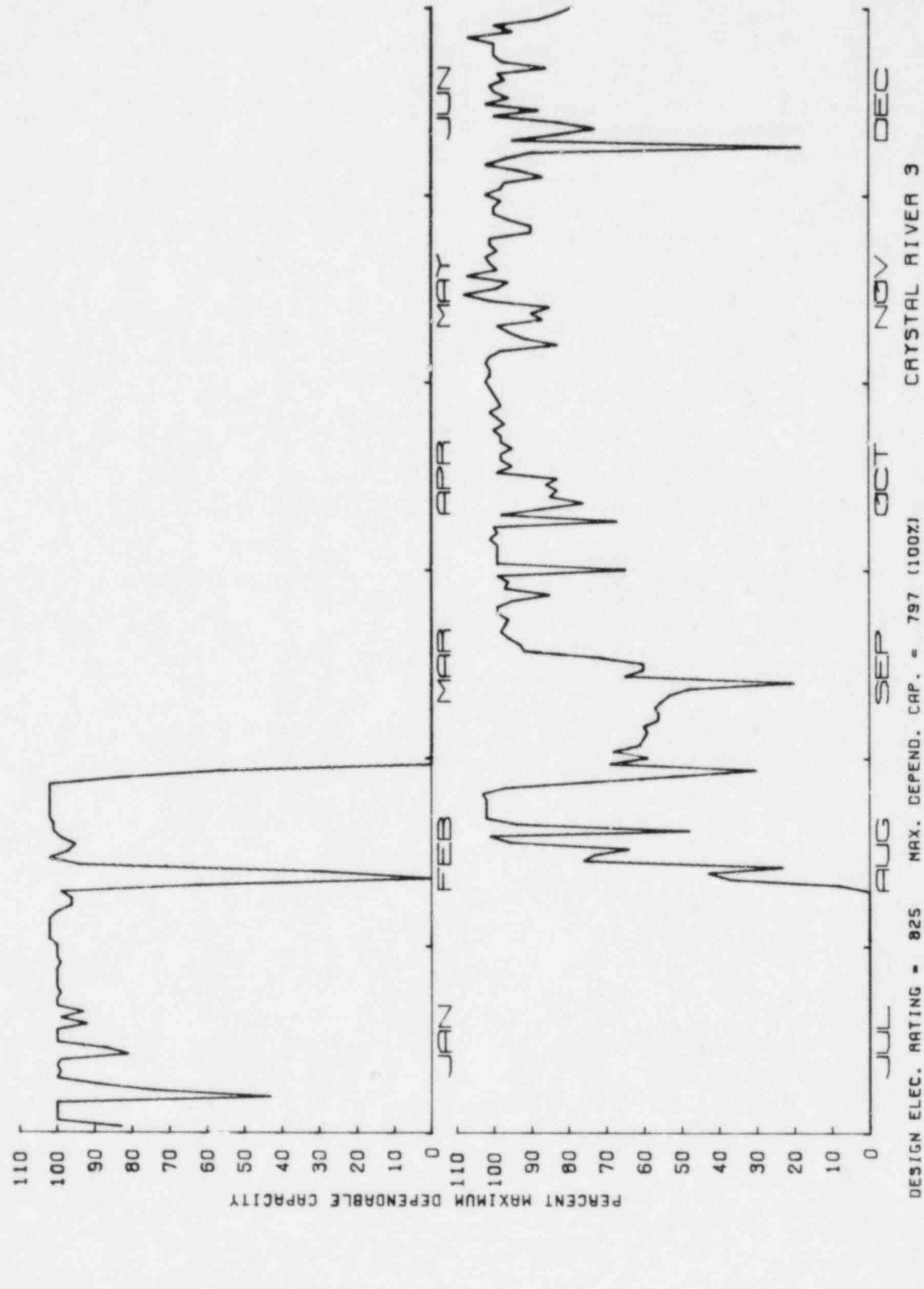
Loss of power to one of two instrumentation buses on February 26 resulted in the loss of several control functions and three-fourths of the instruments. The initiating event probably was either (1) an electrical component failure resulting from an undersized plug-in card which made misalignment of the connector pins possible and likely, (2) inadvertent actions of an instrument technician who was working in the area, or (3) the combined effect of these two circumstances. Some 40,000 gal of reactor coolant water spilled onto the containment building floor mainly through an intermittently open code safety valve. No significant amount of radioactivity was released. A scheduled refueling outage began early because of this event and lasted until August 10. Minor power reductions occurred throughout the year for cleaning condenser waterboxes and inspecting them for saltwater leaks. Several minor power reductions were also necessary because of problems with the main turbine governor valve.

DETAILS OF PLANT OUTAGES FOR CRYSTAL RIVER 3

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/05	14.6	F	To restore RCP oil level. No leaks or problems were discovered.	B	1	Reactor coolant (CB)	Pumps
2	2/09	46.5	F	Dropped rod due to stator failure.	A	1	Reactor (RB)	Control rod drive mechanisms
3a.	2/26	237.5	F	Loss of power to part of the plant instrumentation system (LER 80-10).	A	3	Instrumentation and controls (IF)	Instrumentation and controls
3b.	3/07	3746.5	S	Refueling (begun early).	C	4	Reactor (RC)	Fuel elements
4	8/12	11.0	F	MFWP overspeed trip.	A	3	Steam and power conversion (HH)	Pumps
5	8/19	6.9	F	Turbine trip when new instrument technician mistakenly adjusted the level switch on FW heater 2B instead of on D condenser waterbox.	G	3	Steam and power conversion (HH)	Instrumentation
6	8/29	10.4	F	High RPS pressure trip when turbine governor valve failed open, turbine header pressure decreased, FW increased, then the valve closed, header pressure recovered, and FW dropped.	A	3	Steam and power conversion (HA)	Valves
7	9/11	13.9	S	Cleaned condenser water boxes, put oil in RCP motors, and replaced two PI tubes.	B	1	System code not applicable (ZZ)	Not applicable

DETAILS OF PLANT OUTAGES FOR CRYSTAL RIVER 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8	9/30	6.3	F	High reactor pressure trip during performance of a surveillance procedure on RPS channel A. A clip lead slipped off and shorted the DC power supply. This caused the ICS to see no reactor coolant flow, so it throttled FW.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
9	10/08	5.8	F	Voltage regulator oscillation resulted in an overexcitation trip of the turbine when the regulator was placed in the test position.	A	3	Steam and power conversion (HA)	Instrumentation and controls
10	12/07	19.4	S	Add oil to RCP B motor and replace PI tube.	B	1	Reactor coolant (CB)	Motors


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

DAVIS-BESSE 1

I. Summary

Description

Location: Oak Harbor, Ohio
Docket No.: 50-346
Reactor type: PWR
Maximum dependable capacity (MWe-net): 890
Commercial operation: 11/20/77
Years operating experience: 3.3

Performance

Net electrical energy generated (MWh): 2,093,923
Unit availability factor (%): 36.2
Unit capacity factor (using MDC): 26.8
Unit capacity factor (%) (using design MWe): 26.3

Outages

Total No.: 8
Forced: 8
Scheduled: 0
Total hours: 5,610.5 (63.9%)[a]
Forced: 508.2 (5.8%)
Scheduled: 5,102.3 (58.1%)[b]

II. Highlights

Operation was routine until the refueling outage began on April 7. Refueling, maintenance, and TMI-related modifications were completed on November 6. After the December 3 trip, Davis-Besse 1 operated at a reduced power level of about 50% through the end of the month because of main feedwater pump control problems.

[a] Includes 150.9 h in 1980 from continued 11/30/79 shutdown.

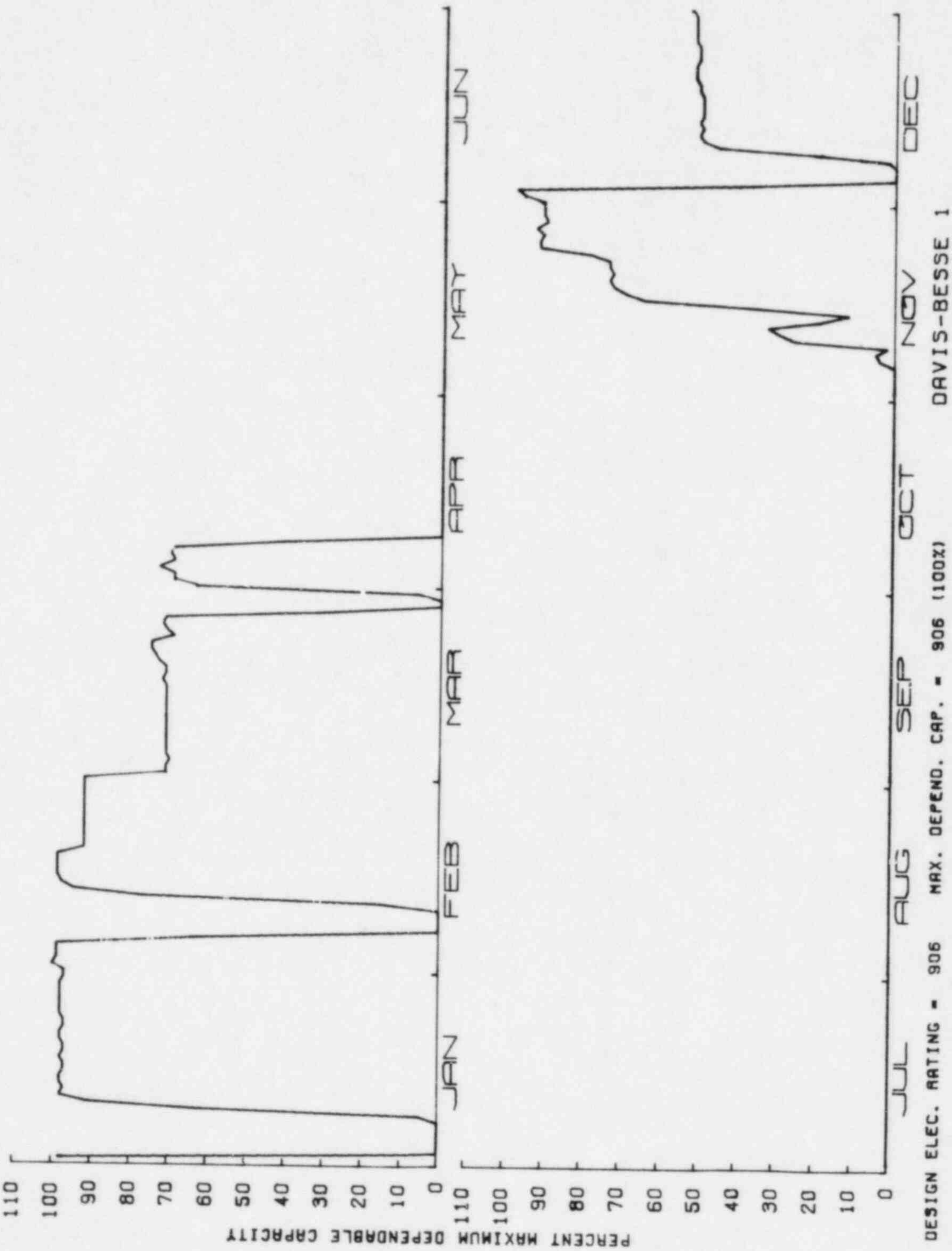
[b] These scheduled hours were continuations from forced shutdowns.

DETAILS OF PLANT OUTAGES FOR DAVIS-BESSE 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	11/30/79 (cont.)	150.9	F	To investigate RCP oil level alarm and to fix control for position indicators.	B	4	Reactor coolant (CB)	Pumps
1	2/05	108.6	F	Trip on high RCS pressure in response to spurious turbine overspeed signal. Replaced all related circuit boards; also repaired condenser tube leaks.	A	3	Steam and power conversion (HA)	Instrumentation and controls
2	3/27	75.1	F	Control rod group 3 dropped and group 4 began to insert and could not be withdrawn by control room operator (LER 80-23).	A	2	Reactor (RB)	Instrumentation and controls
3a.	4/07	0.0	F	Loss of condenser vacuum through high load drain line while testing FW heater.	B	3	Steam and power conversion (HH)	Heat exchangers
3b.	4/07	5098.9	S	Refueling, maintenance, and modifications per NUREG-0578.	C	4	Reactor (RC)	Fuel elements
4	11/06	0.7	F	Generator automatic voltage control problems.	B	1	Steam and power conversion (HA)	Instrumentation and controls
5a.	11/06	22.5	F	Indicated fault on the auxiliary transformer.	A	3	Electric power (EB)	Transformers
5b.	11/07	3.4	S	Turbine overspeed trip test.	B	9	Steam and power conversion (HA)	Turbines
6	11/08	19.4	F	High RCS pressure due to faulty header pressure error signal comparators in the ICS.	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR DAVIS-BESSE 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7	11/12	21.0	F	High RCS pressure when fuse blew in vital 120-V AC bus due to use of a grounded oscilloscope (LERs 80-81, 105).	G	3	Instrumentation and controls (IA)	Instrumentation and controls
8	12/03	110.0	F	RPS trip on flux/delta flux/flow.	A	3	Steam and power conversion (HH)	Instrumentation and controls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DESIGN ELEC. RATING = 906 MAX. DEPEND. CAP. = 906 (100%)
DAVIS-BESSE 1

DRESDEN 2

I. Summary

Description

Location: Morris, Illinois
Docket No.: 50-237
Reactor type: BWR
Maximum dependable capacity (MWe-net): 772
Commercial operation: 6/09/72
Years operating experience: 10.7

Performance

Net electrical energy generated (MWh): 4,580,887
Unit availability factor (%): 93.3
Unit capacity factor (using MDC): 67.6
Unit capacity factor (%) (using design MWe): 65.5

Outages

Total No.: 13
Forced: 10
Scheduled: 3
Total hours: 586.6 (6.7%)
Forced: 370.6 (4.2%)
Scheduled: 216.0 (2.5%)

II. Highlights

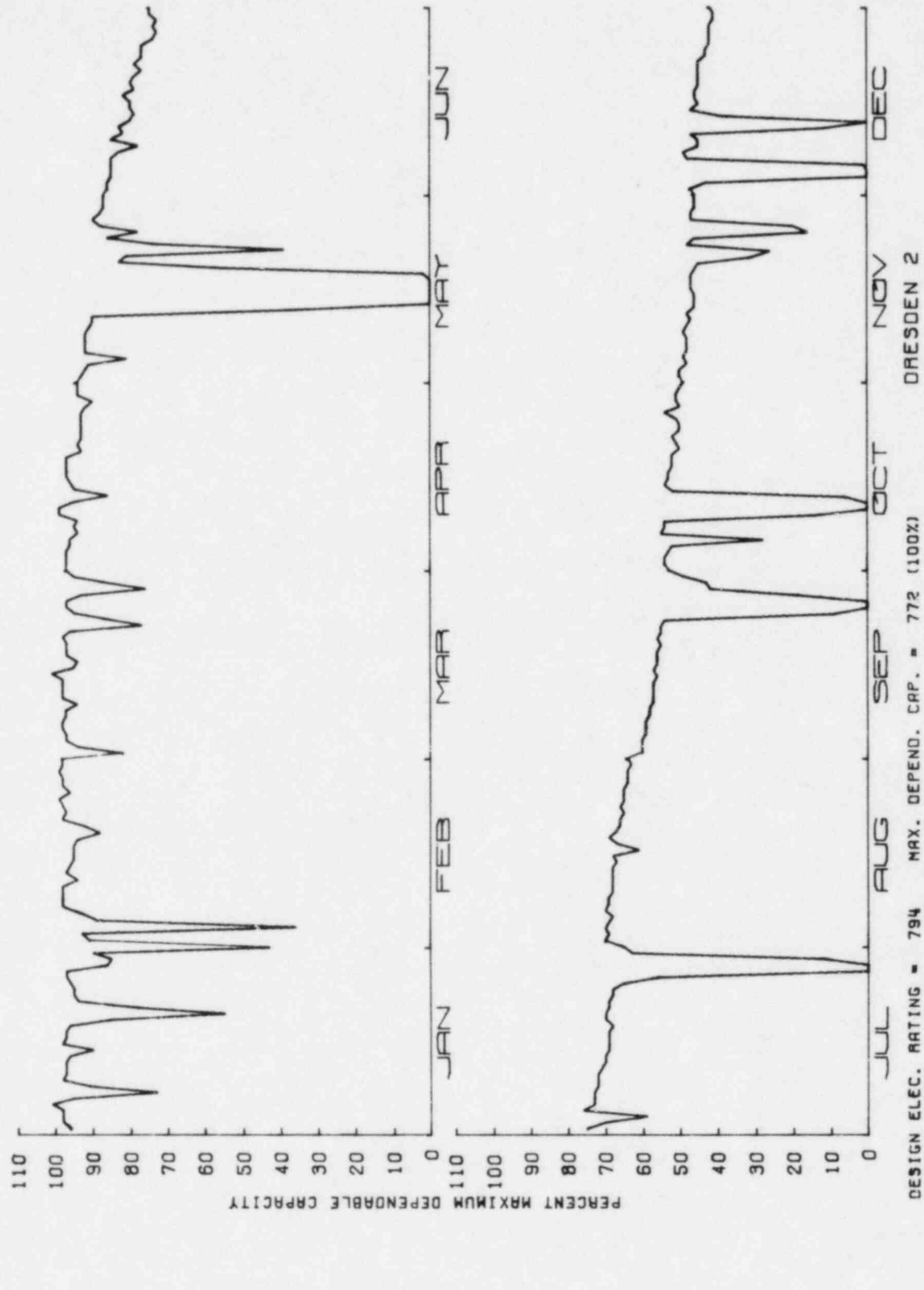
The unit operated with extremely high availability throughout 1980. In April, the unit began a very gradual end-of-cycle coastdown, going from 100% power to about 41% power by the end of December, in preparation for a January refueling. Only four shutdowns lasted longer than 2 d: on May 12 recirculation pump "B" seals were replaced, on July 26 scram testing was performed, on September 23 condenser in-leakage occurred, and on December 2 moisture accumulated in the turbine vibration meter.

DETAILS OF PLANT OUTAGES FOR DRESDEN 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/30	14.3	F	Inadvertent manual scram upon RPS half-scram signal.	G	1	System code not applicable (ZZ)	Not applicable
2	2/03	10.7	F	Low vacuum while transferring relief discharge line from the max-recycle reboiler from the unit 3 condenser to unit 2 condenser.	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
3	5/12	10.0	F	Spurious high radiation signal on main steam line.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
4	5/12	144.6	S	Replace B recirculation pump seals.	A	1	Reactor coolant (CB)	Pumps
5	5/22	10.8	F	Spurious low reactor level signal due to jarring of instrument rack.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
6	7/26	63.1	S	Manual and automatic scram to verify function of control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
7	9/23	78.1	F	Low condenser vacuum because of in-leakage from C turbine hood. Also, B air ejector main steam supply lifted and exhausted to condenser.	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
8	10/05	8.3	S	Tripped turbine to place off gas system back in service; reactor remained critical.	B	9	Radioactive waste management (MB)	Not applicable
9	10/09	83.5	F	Turbine stop valve closure due to EHC pump electrical malfunction.	A	3	Steam and power conversion (HA)	Mechanical function units

DETAILS OF PLANT OUTAGES FOR DRESDEN 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10	11/20	17.5	F	2D CRD pump tripped and failed to restart.	A	2	Reactor (RB)	Pumps
11	11/24	27.8	F	Group I isolation while instrument mechanics were performing main steam line high surveillance DIS 250-1.	G	3	Reactor coolant (CC)	Instrumentation and controls
12	12/02	72.9	F	Moisture in turbine vibration meter.	A	3	Steam and power conversion (HA)	Instrumentation and controls
13	12/11	45.0	F	Scram discharge volume high-high alarm would not reset.	A	2	Instrumentation and controls (ID)	Annunciators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DESIGN ELEC. RATING = 794 MAX. DEPEND. CAP. = 772 (100%)
DRESDEN 2

DRESDEN 3

I. Summary

Description	Performance	Outages
Location: Morris, Illinois	Net electrical energy generated (MWh): 4,329,608	Total No.: 17
Docket No.: 50-249	Unit availability factor (%): 71.8	Forced: 14
Reactor type: BWR	Unit capacity factor (using MDC): 63.8	Scheduled: 3
Maximum dependable capacity (MWe-net): 773	Unit capacity factor (%) (using design MWe): 62.1	Total hours: 2,464.2 (28.0%)
Commercial operation: 11/16/71		Forced: 152.4 (1.7%)
Years operating experience: 9.4		Scheduled: 2,311.8 (26.3%)

II. Highlights

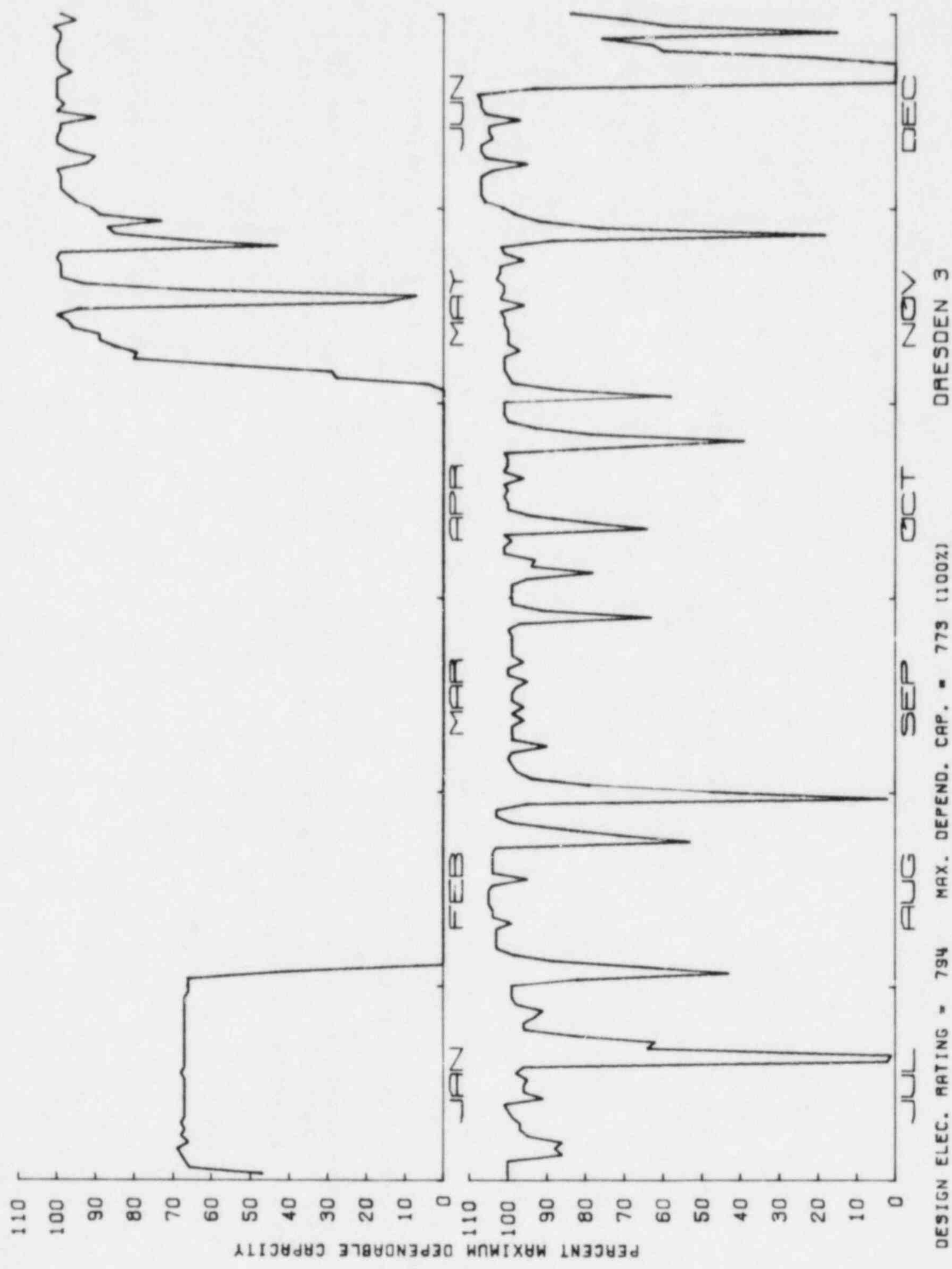
Operation was routine with an administrative-imposed derating to 550 MWe for air ejector evaluations through February 2, when the unit was taken off-line because of an inoperable jet pump. Refueling commenced the following day and was completed May 3.

DETAILS OF PLANT OUTAGES FOR DRESDEN 3

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/02	21.3	F	Jet pump No. 13 found inoperable (LER 80-04).	A	1	Reactor coolant (CB)	Pumps
2	2/03	2158.0	S	Refueling.	C	1	Reactor (RC)	Fuel elements
3	5/03	6.6	F	Turbine trip on high bearing vibration (reactor remained critical).	H	9	Steam and power conversion (HA)	Turbines
4	5/03	1.3	F	Turbine trip on moisture separator high-high signal (reactor remained critical).	H	9	Reactor coolant (CC)	Turbines
5	5/05	9.5	F	Spurious low reactor water level due to jarring of instrument rack.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
6	5/15	9.6	F	Electrohydraulic control oil leak (reactor remained critical).	A	9	Steam and power conversion (HA)	Mechanical function units
7	5/16	5.2	F	Electrohydraulic control oil leak (reactor remained critical).	A	9	Steam and power conversion (HA)	Mechanical function units
8	5/17	5.6	F	Turbine control valve stuck open (reactor remained critical).	A	9	Steam and power conversion (HA)	Valves
9	5/25	9.3	F	Turbine trip on moisture separator high-high level. Reactor scram on turbine stop valve closure.	H	3	Steam and power conversion (HA)	Valves
10	7/19	43.4	S	Manual and automatic scram to verify function of control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
11	8/01	12.8	F	Low reactor level (appears 3B FW regulator valve went closed).	A	3	Reactor coolant (CH)	Valves

DETAILS OF PLANT OUTAGES FOR DRESDEN 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12	8/29	15.8	F	Low reactor water level (appears that 3C feedwater pump minimum flow valve drifted open).	A	3	Reactor coolant (CH)	Valves
13	8/30	11.2	F	Turbine tripped on C moisture separator high-high while resetting heaters. Reactor scrammed on high neutron flux.	H	1	Reactor coolant (CC)	Heat exchangers (MSR)
14	10/24	14.8	F	Essential service M set tripped and APRMs upscale.	A	3	Electric power (EE)	Generators (MG set)
15	11/26	13.8	F	Group I isolation. Contacts on reactor mode switch did not make up properly when switched from run to startup and the unit went below 850 psi.	A	3	Instrumentation and controls (IF)	Relays
16	12/20	110.4	S	Oil leak on No. 2 CIV.	A	1	Steam and power conversion (HA)	Valves
17	12/28	15.6	F	Repair EHC pump.	A	1	Steam and power conversion (HA)	Pumps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DESIGN ELEC. RATING = 794 MAX. DEPEND. CAP. = 773 (100%)
DRESDEN 3

DUANE ARNOLD

I. Summary

Description	Performance	Outages
Location: Palo, Iowa	Net electrical energy generated (MWh): 2,796,975	Total No.: 15
Docket No.: 50-331	Unit availability factor (%): 73.5	Forced: 11
Reactor type: BWR	Unit capacity factor (using MDC): 61.8	Scheduled: 4
Maximum dependable capacity (MWe-net): 515	Unit capacity factor (%) (using design MWe): 59.2	Total hours: 2,326.1 (26.4%)
Commercial operation: 2/01/75		Forced: 478.3 (5.4%)
Years operating experience: 6.6		Scheduled: 1,847.8 (21.0%)

II. Highlights

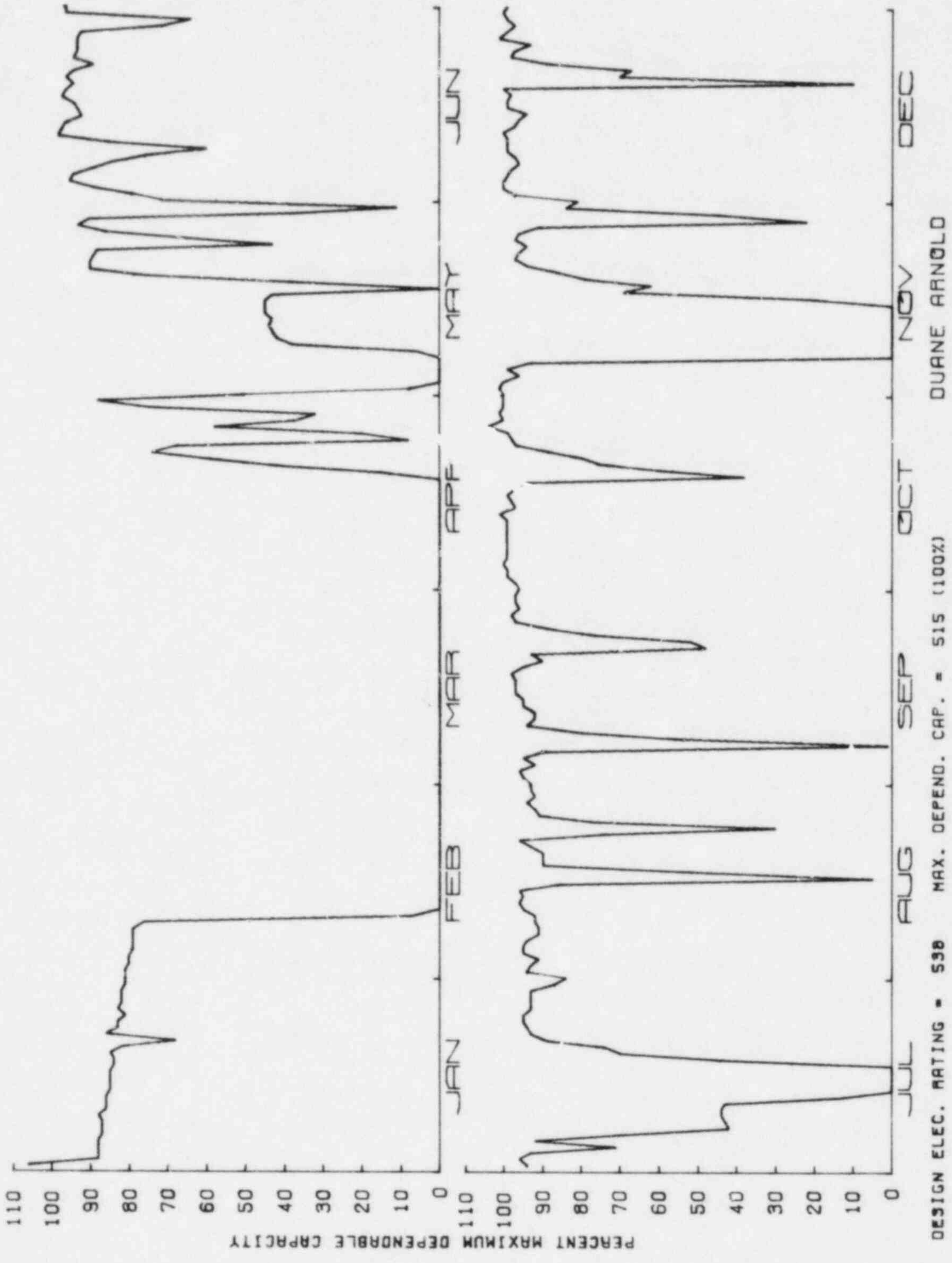
The unit was in an end-of-cycle coast down until refueling began on February 9. The unit was returned to service on April 18. The feedwater and recirculation systems caused problems at Duane Arnold in 1980. The recirculation pumps and motor generator set drive motors of the recirculation system were responsible for the first five shutdowns following the refueling and also for the 50% power reduction between May 7 and May 17. On July 6, power was reduced to about 40% because of the indication of the failure of the inner seal on the B recirculation pump. Then scram testing was performed per IE Bulletin 80-17 on July 12, and the reactor remained down until July 17. In November, the unit was shut down to replace the seal on recirculation pump A.

DETAILS OF PLANT OUTAGES FOR DUANE ARNOLD

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a.	2/09	0.0	F	Scram while testing turbine control valve.	B	3	Steam and power conversion (HA)	Valves
1b.	2/09	1648.0	S	Unit left down for refueling.	C	4	Reactor (RC)	Fuel elements
2	4/23	28.0	S	Test of recirculation pump trip system.	B	2	Reactor coolant (CB)	Pumps
3	4/26	15.6	S	Test of recirculation pump trip system.	B	3	Reactor coolant (CB)	Pumps
4	5/01	151.3	F	Failure of motor generator set drive motor of A recirculation system (LER 80-17).	A	1	Reactor coolant (CB)	Motors
5	5/17	21.3	S	Shutdown to place recirculation system A back in service.	H	1	Reactor coolant (CB)	Pumps
6	5/29	16.4	F	High/low level alarm on B recirculation pump motor; oil level found low (LER 80-22).	A	1	Reactor coolant (CB)	Motors
7	7/12	134.9	S	Scram testing per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
8	8/16	19.9	F	Nitrogen leak on FW check valve.	A	1	Reactor coolant (CH)	Valves
9	8/23	5.1	F	Turbine overspeed trip testing.	B	1	Steam and power conversion (HA)	Turbines
10	9/06	19.0	F	Repair of overspeed trip oil line.	A	1	Steam and power conversion (HA)	Mechanical function units

DETAILS OF PLANT OUTAGES FOR DUANE ARNOLD (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11	9/21	8.8	F	MSIV's less than 90% open due to loss of nitrogen pressure. An earlier group isolation had not been properly reset.	G	3	Reactor coolant (CD)	Valve operators
12a.	11/06	96.0	F	Replace seal on recirculation pump A.	A	1	Reactor coolant (CB)	Pumps
12b.	11/10	124.8	F	Main steam relief valve PSV-4405 was opened during startup and stuck open. Repaired PSV-4405 and wiring error (LER 80-54). Also, containment isolation valves failed to close upon signal (LER 80-55).	A	4	Reactor coolant (CB)	Valves
13	11/27	8.6	F	Power was reduced and the generator taken off line to repair an EHC oil leak.	A	9	Steam and power conversion (HA)	Pipes, fittings
14	11/27	8.2	F	Operator set APRM gain adjustment incorrectly.	G	3	Instrumentation and controls (ID)	Instrumentation and controls
15	12/19	20.2	F	Loss of condenser vacuum due to drain pump problems with air ejector condenser drain tank.	A	3	Steam and power conversion (HC)	Pumps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DESIGN ELEC. RATING = 538 MAX. DEPEND. CAP. = 515 (100%)
DUANE ARNOLD

FARLEY 1

I. Summary

Description	Performance	Outages
Location: Dothan, Alabama	Net electrical energy generated (MWh): 4,603,742	Total No.: 24
Docket No.: 50-348	Unit availability factor (%): 69.6	Forced: 21
Reactor type: PWR	Unit capacity factor (using MDC): 65.2	Scheduled: 3
Maximum dependable capacity (MWe-net): 804	Unit capacity factor (%) (using design MWe): 63.2	Total hours: 2,671.1 (30.4%)
Commercial operation: 12/01/77		Forced: 397.8 (4.5%)
Years operating experience: 3.4		Scheduled: 2,273.3 (25.9%)

II. Highlights

The unit experienced several power reductions for maintenance on condensate pumps, for turbine governor valve surveillance testing, and because of loss of an isophase bus duct cooling fan (1/28). The unit was shut down on January 31 for more than 2 weeks for TMI-related modifications. In February, March, and April, secondary system trouble-shooting, testing, maintenance, and repairs caused 14 power reductions, usually in the range from 5 to 15%. Loss of feedwater pump suction required shutdowns on January 27, February 22 and 23, and March 8 and 10. On June 14 the reactor was shut down for over 2 weeks to repair steam generator tube leaks. The unit was taken off-line on November 7 for the remainder of the year for refueling.

DETAILS OF PLANT OUTAGES FOR FARLEY 1

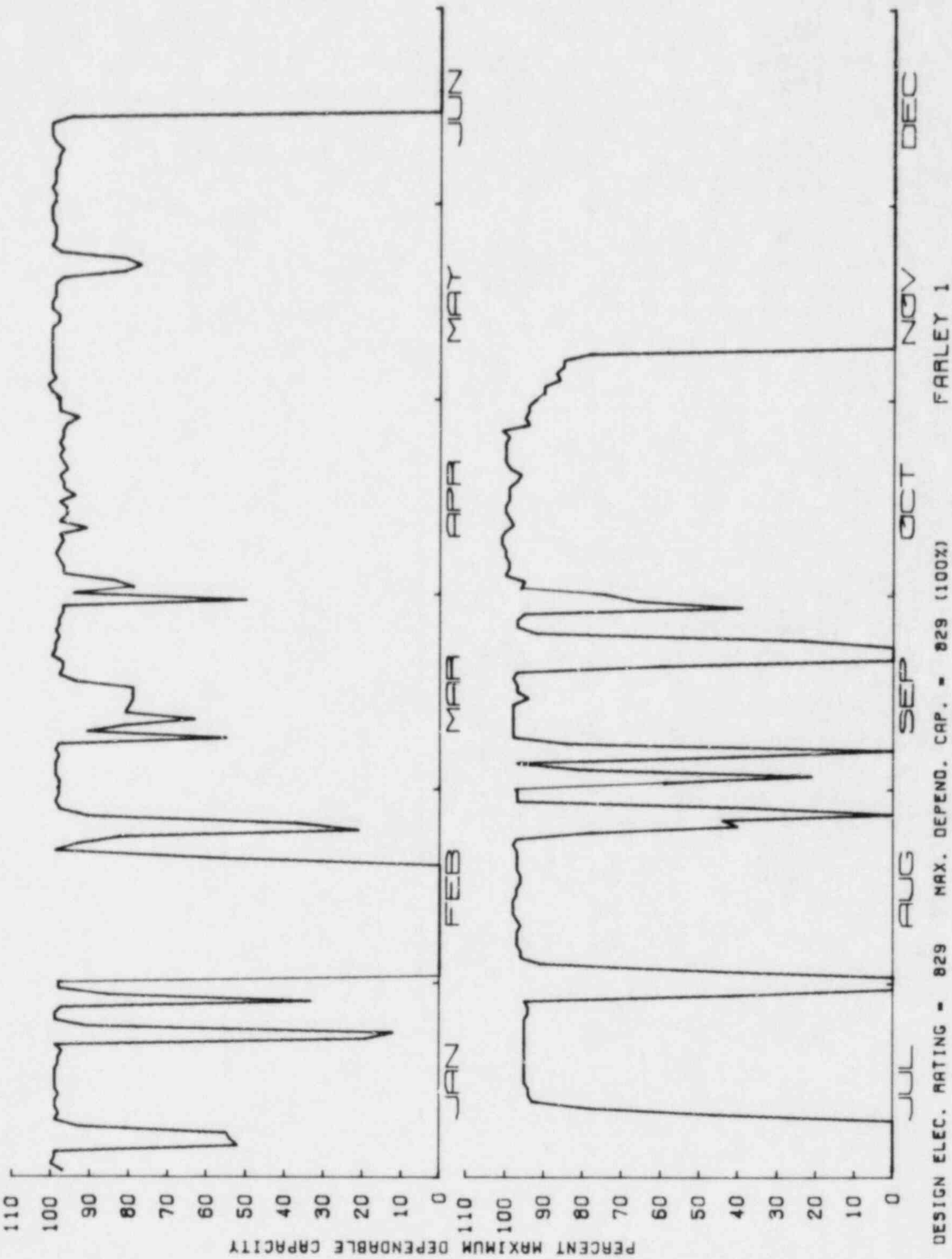
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/21	26.5	F	RCS drain valve leakage.	B	1	Reactor coolant (CI)	Valves
2	1/22	7.4	F	Troubleshooting on breaker trip indicator for MFWP turning gear motor caused unit trip.	G	3	Steam and power conversion (HH)	Circuit closers/interrupters
3	1/27	12.0	F	Loss of both MFWPs on low suction pressure. Instrument air line valves failed open, were replaced.	A	3	Steam and power conversion (HH)	Pipes, fittings
4	1/31	370.1	S	TMI-related modifications.	D	1	Other (XX)	Other
5	2/16	3.8	F	Low-low SG level due to defective control card in B MFWP control system.	A	3	Steam and power conversion (HH)	Instrumentation and controls
6	2/16	17.7	F	Low-low SG level during startup with AFWPs.	G	3	Steam and power conversion (HH)	Instrumentation and controls
7	2/17	16.0	F	Low-low SG level due to unstable MFWP control; revised delta P program in MFWP speed control circuit.	A	3	Steam and power conversion (HH)	Instrumentation and controls
8	2/22	14.6	F	Low suction pressure for both MFWPs.	H	3	Steam and power conversion (HH)	Pumps
9	2/23	8.5	F	Low suction pressure for both MFWPs.	G	1	Steam and power conversion (HH)	Pumps
10	3/08	5.9	F	Low suction pressure for both MFWPs.	H	2	Steam and power conversion (HH)	Pumps
11	3/10	5.9	F	Low suction pressure for both MFWPs.	H	2	Steam and power conversion (HH)	Pumps
12	3/29	11.0	F	No recirculation flow in boron injection tank (LER 80-23).	A	1	Engineered safety features (SF)	Accumulators

DETAILS OF PLANT OUTAGES FOR FARLEY 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13	6/14	607.2	S	"B" SG tube leak repairs.	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
14	7/09	13.3	F	Low-low SG level during startup.	G	3	Steam and power conversion (HH)	Heat exchangers (steam generator)
15	7/29	84.5	F	Break in EH fluid line to governor valve on HP main turbine which caused a MFWP to trip on low EH pressure, resulting in a steam flow feedwater flow mismatch and SG low level.	A	3	Steam and power conversion (HA)	Pipes, fittings
16	8/27	12.1	F	Breaker DH 01 racked out by mistake, causing both MFWPs to trip.	G	3	Electric power (EB)	Circuit closers/interrupters
17	8/27	9.0	F	Low-low level in B SG while putting 1A MFWP on-line and removing 2B MFWP from service to troubleshoot an oscillating governor valve.	A	3	Steam and power conversion (HH)	Valves
18	8/27	23.2	F	Replace cracked disk on No. 2 LP turbine.	A	1	Steam and power conversion (HA)	Turbines
19	9/01	3.8	F	SG 1C high level. While increasing power, MFWP B was left in manual and IC SG level was decreasing. When IC SG FW was increased, high level occurred.	G	3	Steam and power conversion (HH)	Valves

DETAILS OF PLANT OUTAGES FOR FARLEY 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
20	9/02	6.8	F	SG 1B low-low level after transfer to FW regulating valve bypass.	G	3	Steam and power conversion (HH)	Valves
21	9/05	22.3	F	Incoming breaker opened from 4160-V bus 1F to load center 1D, giving the SSPS a RCP breaker open signal. Cause unknown.	H	3	Electric power (EB)	Circuit closers/interrupters
22	9/06	11.5	F	SG 1C low-low level after transfer to FW regulating valve bypass.	G	3	Steam and power conversion (HH)	Valves
23	9/19	82.0	F	Investigation of possible bolt damage due to overtorquing containment equipment hatch bolts. Wrong torque valve in vendor manual (80-54).	F	1	Engineered safety features (SD)	Penetration, primary containment
24	11/07	1296.0	S	Refueling.	C	1	Reactor (RC)	Fuel elements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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
Maximum dependable capacity (MWe-net): 810
Commercial operation: 7/28/75
Years operating experience: 5.9

Performance

Net electrical energy generated (MWh): 4,334,505
Unit availability factor (%): 70.2
Unit capacity factor (using MDC): 60.1
Unit capacity factor (%) (using design MWe): 60.1

Outages

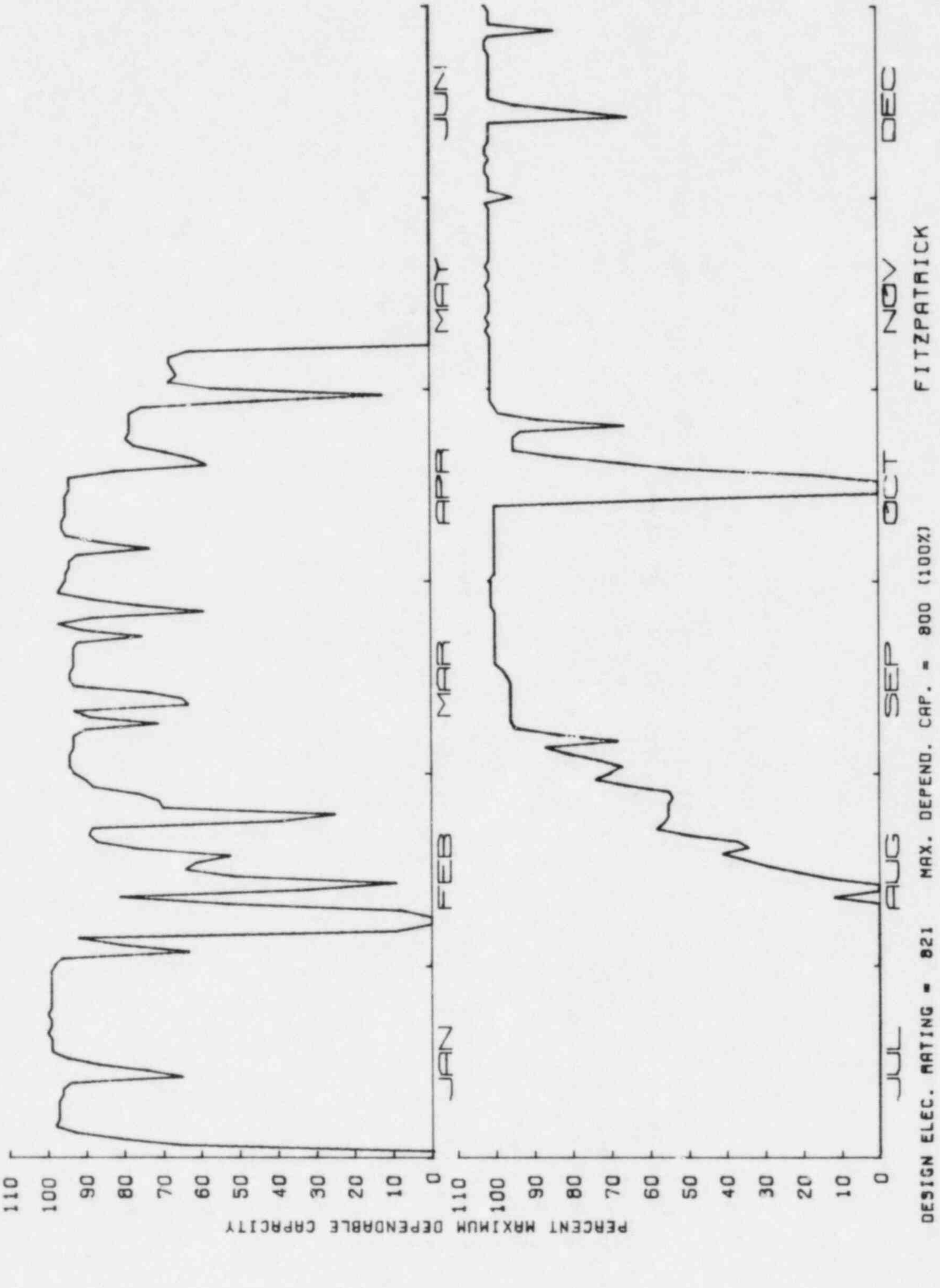
Total No.: 8
Forced: 6
Scheduled: 2
Total hours: 2,619.2 (29.8%)
Forced: 258.2 (2.9%)
Scheduled: 2,361.0 (26.9%)

II. Highlights

Power was reduced nine times in 1980 for control rod pattern changes or sequence exchanges (January 12, February 2, March 8 and 22, April 5, October 24, and December 13). Power was reduced three times because of high reactor vessel conductivity due to condenser tube leaks (March 11 and 22, and April 17). A refueling outage began May 7 and was completed August 11. The NRC scram tests were performed on August 11.

DETAILS OF PLANT OUTAGES FOR FITZPATRICK

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/05	80.8	F	Moisture leakage into valve operator caused motor and control circuit failure. Repaired leak, provided a drainage path, and replaced motor (LER 80-16).	A	1	Engineered safety features (SD)	Valve operators
2	2/11	30.6	F	Operator pulled fuses on undervoltage device for 10300 bus.	G	3	Electric power (EB)	Circuit closers/interrupters
3	2/21	20.1	F	During calibration of main steam hi-radiation monitors a vendor employee accidentally bumped a level switch on CRD scram discharge volume.	G	3	Reactor (RB)	Instrumentation and controls
4	4/28	21.1	F	Generator ground due to condensate forming on generator exciter rectifiers.	H	3	Electric power (EB)	Generators
5	5/06	13.4	F	Malfunction in EHC system while testing CIV/ISV valves.	A	3	Steam and power conversion (HA)	Valves
6	5/07	2309.0	S	Refueling.	C	1	Reactor (RC)	Fuel elements
7	8/11	52.0	S	Scram test of scram discharge volume per IE Bulletin 80-17 and turbine test.	H	2	Reactor (RB)	Control rods
8	10/13	92.2	F	Loss of B RPS MG set.	A	3	Electric power (EE)	Generators


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 821 MAX. DEPEND. CAP. = 800 (100%)
FITZPATRICK

FORT CALHOUN

I. Summary

Description	Performance	Outages
Location: Ft. Calhoun, Nebraska	Net electrical energy generated (MWh): 2,010,662	Total No.: 4
Docket No.: 50-285	Unit availability factor (%): 60.4	Forced: 3
Reactor type: PWR	Unit capacity factor (using MDC): 49.2	Scheduled: 1
Maximum dependable capacity (MWe-net): 457	Unit capacity factor (%) (using design MWe): 49.2	Total hours: 3,475.1 (39.6%)
Commercial operation: 6/20/74		Forced: 17.1 (0.2%)
Years operating experience: 7.4		Scheduled: 3,458.0 (39.4%)

II. Highlights

Refueling commenced on January 18 and was completed on June 11. A power level increase to 1,500 MWt from 1,420 MWt was approved on August 15. The unit was operated at 65% power during October and 75% during November and December to conserve fuel. Only three forced outages occurred all year at Fort Calhoun.

DETAILS OF PLANT OUTAGES FOR FORT CALHOUN

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/18	3458.0	S	Refueling and maintenance.	C	1	Reactor (RC)	Fuel elements
2	7/24	9.6	F	Low SG level due to loss of instrument bus when inverter D output breaker tripped.	A	3	Electric power (ED)	Circuit closers/interrupters
3	9/19	3.6	F	Breaker on control rod clutch power supply tripped while switching power sources to investigate voltage spiking on inverter C.	A	3	Reactor (RB)	Circuit closers/interrupters
4	11/14	3.9	F	Repair packing leak on root valve on reactor coolant sample line.	A	4	Reactor coolant (CB)	Valves

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): 675,717	Total No.: 26
Docket No.: 50-267	Unit availability factor (%): 53.6	Forced: 24
Reactor type: HTGR	Unit capacity factor (using MDC): 23.3	Scheduled: 2
Maximum dependable capacity (MWe-net): 330	Unit capacity factor (%) (using design MWe): 23.3	Total hours: 4,077.4 (46.4%)[a]
Commercial operation: 7/01/79		Forced: 1,575.2 (17.9%)
Years operating experience: 4.1		Scheduled: 2,502.2 (28.5%)[a]

II. Highlights

Installation of region constraint device in the core was completed December 24, 1979. The reactor was critical at less than 2% power for a brief period in January when discovery of a ruptured helium circulator static seal necessitated a shutdown. Replacement of the circulator took until March 5. The unit then operated under an NRC restriction of 231 MWe-net (70% power) pending resolution of temperature fluctuations. A 2-week outage began on July 8 because of a simultaneous trip of all four circulators. A 5.5-week outage for surveillance testing began on August 29. (Although the reactor was critical on September 27, the generator remained off-line.) Fluctuation testing continued in November and December.

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[a] Includes 1,538.8 h in 1980 from continued 10/26/79 shutdown.

DETAILS OF PLANT OUTAGES FOR FORT ST. VRAIN

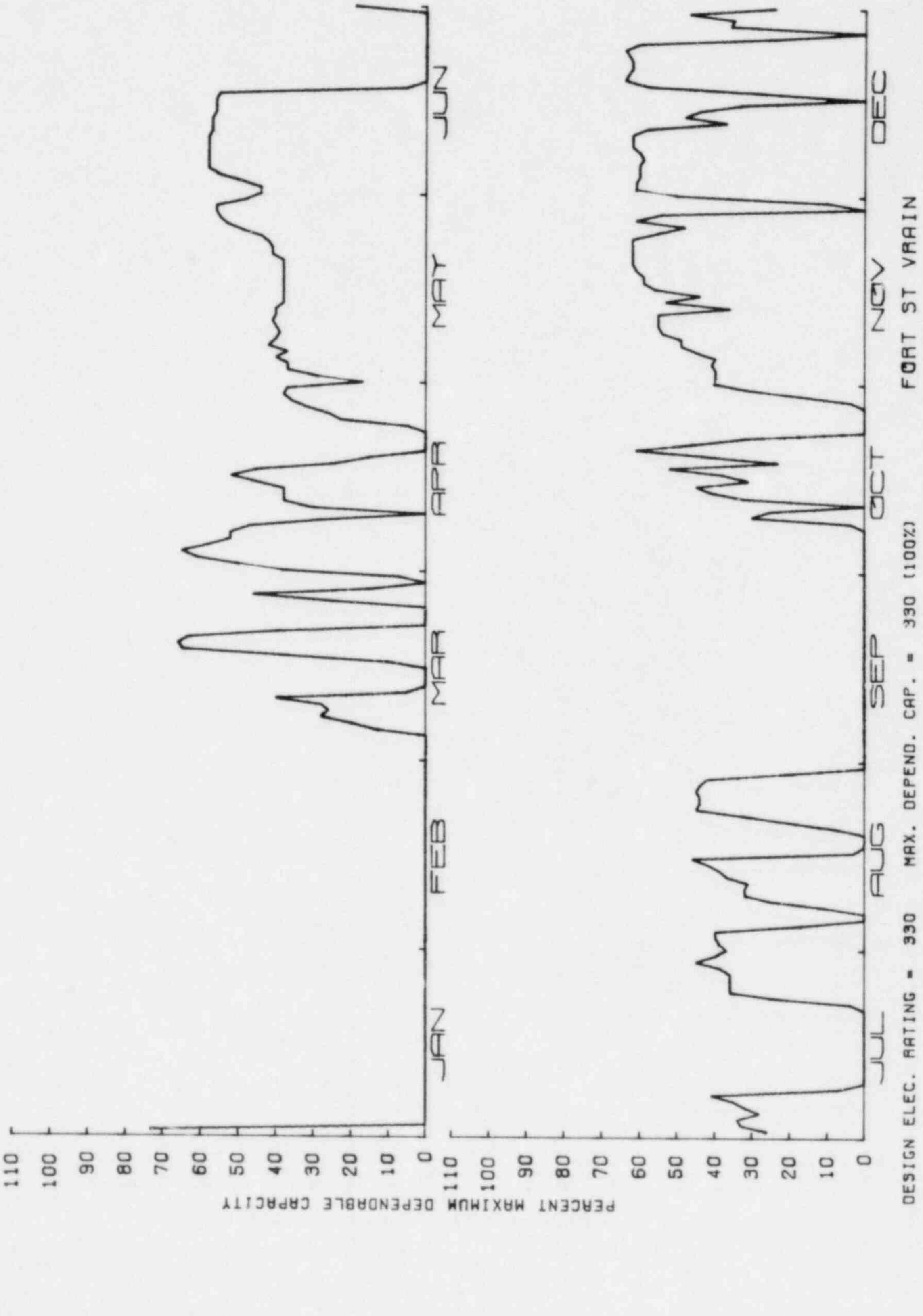
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	10/26/79 (cont.)	1538.8	S	Reactor was critical at ~2% for 36.3 h in January until a static seal in a helium circulator ruptured (LER 80-01). Generator put on line 3/05/80.	B	4	Reactor coolant (CB)	Pumps
1	3/06	4.2	S	Turbine overspeed trip test.	B	2	Steam and power conversion (HA)	Turbines
2	3/11	123.8	F	Circulating water system shutdown due to tower problems.	A	2	Steam and power conversion (HF)	Heat exchangers
3	3/21	56.8	F	Instrument problems.	A	3	Instrumentation and controls (ID)	Instrumentation and controls
4	3/24	49.2	F	Loss of condenser vacuum.	A	2	Steam and power conversion (HC)	Heat exchangers (condensers)
5	3/28	53.2	F	1A circulator trip and system upset.	A	1	Reactor coolant (CB)	Pumps
6	4/08	30.0	F	Turbine generator tripped spuriously during a loop 1 shutdown due to instrument problems.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
7	4/18	0.7	F	Spurious trip while changing instrument modules.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
8	4/18	118.7	F	Turbine trip during loop 1 shutdown due to buffer-mid-buffer problem on RCP.	A	3	Reactor coolant (CB)	Pumps
9	4/25	5.9	F	Turbine taken off-line to recover from a loop shutdown.	A	9	Steam and power conversion (HA)	Turbines
10	4/30	18.8	F	Turbine taken off-line to recover from a loop shutdown.	A	9	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES FOR FORT ST. VRAIN (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11	6/17	313.4	F	Turbine manually shut down after both circulators on one loop tripped. Loop insolated and reactor not shut down.	H	9	Reactor coolant (CB)	Pumps
12	7/08	339.9	F	Simultaneous trip of all four circulators.		3	Reactor coolant (CB)	Pumps
13	7/23	0.4	F	Turbine trip on low main steam pressure. Reactor was not shut down.	H	9	Steam and power conversion (HB)	Not applicable
14	8/04	65.2	F	Loss of all four circulators.	A	3	Reactor coolant (CB)	Pumps
15	8/16	102.7	F	Rupture of hydraulic oil supply line (LER 80-45).	A	2	Other auxiliary (AD)	Pipes, fittings
16	8/29	959.9	S	Following a turbine runback as a result of work on the EHC system, the turbine was manually tripped and the reactor shut down, beginning the scheduled shutdown for surveillance testing. Hydraulic snubbers on main steam supply were repaired or readjusted (LER 80-47).	B	1	System code not applicable (ZZ)	Not applicable
17	10/08	2.2	F	Problems with No. 2 stop valve.	B	1	Steam and power conversion (HA)	Valves
18	10/10	35.0	F	Turbine trip occurred while personnel were investigating problems with bearing vibration.	H	3	Steam and power conversion (HA)	Turbines
19	10/22	1.9	F	Spurious loop shutdown during ultrasonic testing.	H	3	Instrumentation and controls (IA)	Not applicable

DETAILS OF PLANT OUTAGES FOR FORT ST. VRAIN (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
20	10/23	134.9	F	Repair hydraulic oil leaks on system 91 accumulator blind flange, seals of, Loop 1 (System 91 is the hydraulic control system for the secondary coolant system valves).	A	1	Steam and power conversion (HB)	Pipes, fittings
21	11/27	36.7	F	Repair hot reheat drain line.	A	1	Steam and power conversion (HB)	Valves
22	12/15	30.9	F	Test for a possible SG tube leak due to excessive total primary coolant oxidants (80-75).	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
23	12/26	15.0	F	Repair of SG trim valve TV-2228-2.	A	1	Steam and power conversion (HB)	Valves
24	12/27	27.3	F	Hot reheat scram during startup.	A	3	Instrumentation and controls (IB)	Not applicable
25	12/29	5.7	F	Erratic PV-2244 control.	A	1	Steam and power conversion (HB)	Valves
26	12/31	6.9	F	Repair of trim valve V-7202.	A	2	Steam and power conversion (HB)	Valves


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

GINNA

I. Summary

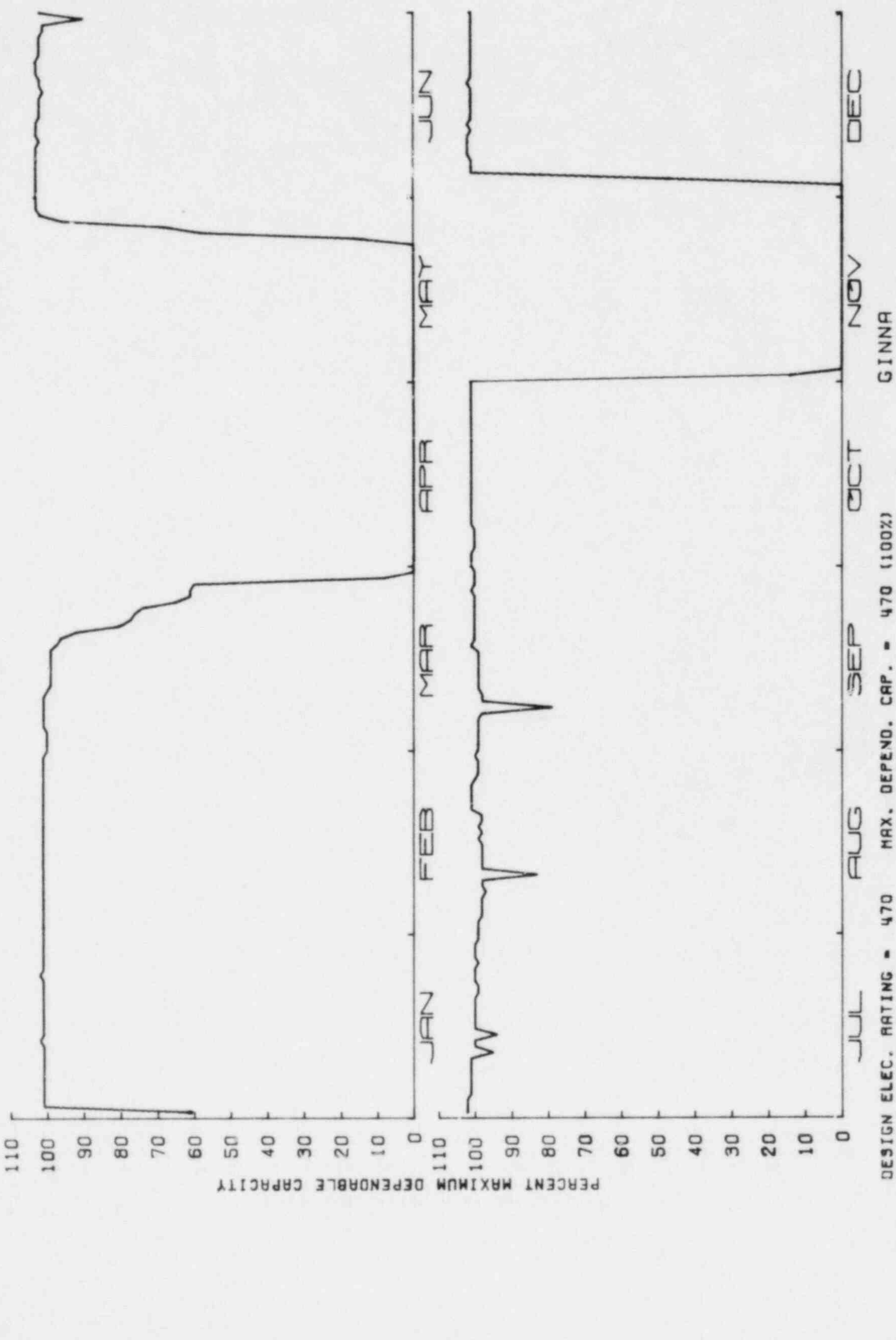
Description	Performance	Outages
Location: Ontario, New York	Net electrical energy generated (MWh): 3,093,997	Total No.: 2
Docket No.: 50-244	Unit availability factor (%): 76.0	Forced: 0
Reactor type: PWR	Unit capacity factor (using MDC): 74.9	Scheduled: 2
Maximum dependable capacity (MWe-net): 470	Unit capacity factor (%) (using design MWe): 74.9	Total hours: 2,108.8 (24.0%)
Commercial operation: 7/70		Forced: 0.0 (0.0%)
Years operating experience: 11.1		Scheduled: 2,108.8 (24.0%)

II. Highlights

Operation was routine at full power until the unit shut down on March 29 for refueling and maintenance. Load reductions occurred on March 10 and 21 because of feedwater heater problems. End of cycle coastdown began on March 17. The unit returned to operation on May 23 and operated until a scheduled steam generator inspection began on November 1. Inspections were finished on December 3 and the unit operated above full power for the remainder of December. Ginna had no forced outages in 1980.

DETAILS OF PLANT OUTAGES FOR GINNA

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	3/29	1339.7	S	Refueling.	C	1	Reactor (RC)	Fuel elements
2	11/01	769.1	S	Steam generator inspection. Previous eddy current testing had indicated intergranular attack in the tube crevice area.	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 470   MAX. DEPEND. CAP. = 470 (100%)
GINNA

HADDAM NECK

I. Summary

Description

Location: Haddam Neck, Connecticut
Docket No.: 50-213
Reactor type: PWR
Maximum dependable capacity (MWe-net): 555
Commercial operation: 1/01/68
Years operating experience: 13.4

Performance

Net electrical energy generated (MWh): 3,562,845
Unit availability factor (%): 75.0
Unit capacity factor (using MDC): 73.1
Unit capacity factor (%) (using design MWe): 69.9

Outages

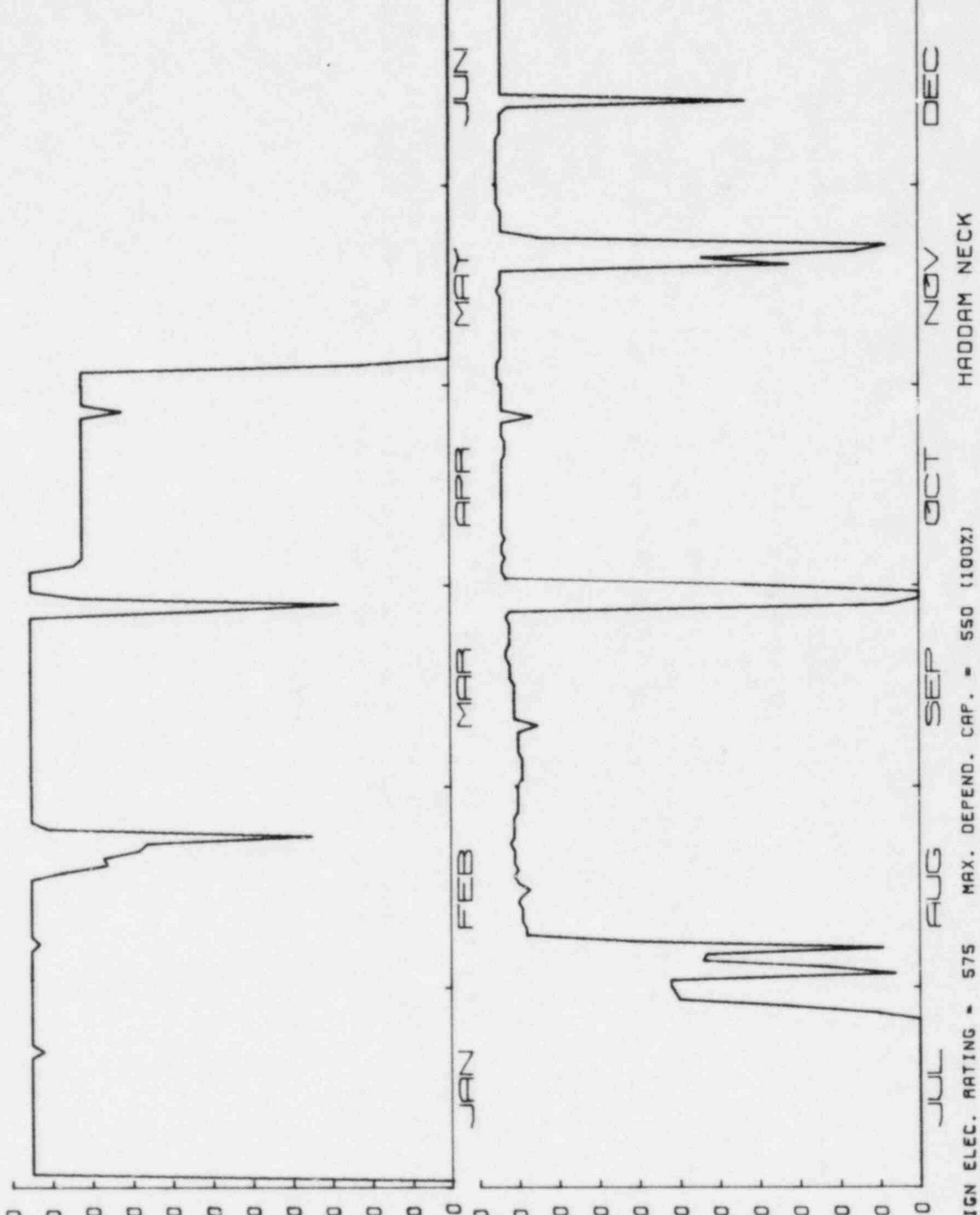
Total No.: 9
  Forced: 4
  Scheduled: 5
Total hours: 2,203.3 (25.1%)
  Forced: 78.9 (0.9%)
  Scheduled: 2,124.4 (24.2%)

II. Highlights

Operation was routine until the refueling outage began May 3, except that power was restricted to 88% from April 3 until the refueling began because of suspected cracks in the low-pressure turbine disk. Power operations resumed on July 27 and were routine for the remainder of the year.

DETAILS OF PLANT OUTAGES FOR HADDAM NECK

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	3/27	13.7	F	Spurious signal for high containment pressure/core cooling actuation during wiring modifications per NUREG-0578.	H	3	Instrumentation and controls (IA)	Instrumentation and controls
2	5/03	2040.4	S	Refueling.	C	1	Reactor (RC)	Fuel elements
3	8/02	18.5	F	Turbine overspeed trip setting improperly adjusted.	G	3	Steam and power conversion (HA)	Mechanical function units
4	8/05	14.8	S	Tie in loop No. 2 after RCP No. 2 repair.	F	1	Reactor coolant (CB)	Pumps
5	9/27	64.7	S	Turbine balancing.	B	1	Steam and power conversion (HA)	Turbines
6	11/18	13.8	F	Failure of movable gripper coils caused two rods to drop (80-16).	A	2	Reactor (RB)	Control rod drives
7	11/20	32.9	F	Mechanical overspeed device on HP turbine out of adjustment.	H	3	Steam and power conversion (HB)	Mechanical function units
8	11/21	0.7	S	Test of turbine overspeed trip setting.	B	1	Steam and power conversion (HB)	Mechanical function units
9	12/13	3.8	S	Test of turbine overspeed trip setting (reactor was not submitted).	B	1	Steam and power conversion (HB)	Mechanical function units


PERCENT MAXIMUM DEPENDABLE CAPACITY
110 100 90 80 70 60 50 40 30 20 10 0
JAN FEB MAR APR MAY JUN
JUL AUG SEP OCT NOV DEC
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): 4,790,546	Total No.: 28
Docket No.: 50-321	Unit availability factor (%): 81.7	Forced: 26
Reactor type: BWR	Unit capacity factor (using MDC): 71.4	Scheduled: 2
Maximum dependable capacity (MWe-net): 764	Unit capacity factor (%) (using design MWe): 70.2	Total hours: 1,606.5 (18.3%)[a]
Commercial operation: 12/31/75		Forced: 1,593.0 (18.1%)
Years operating experience: 6.1		Scheduled: 13.5 (0.2%)[a]

II. Highlights

Numerous shutdowns and power reductions occurred at Hatch 1 in 1980, yet the unit availability was 81.7% and the unit MDC and DER capacity factors were over 70%; Hatch 1 was not shut down for refueling this year. The longest shutdowns were on May 24 for RHR valve repair, June 8 for mechanical snubber repair, on June 26 because the HPCI was out of service, and on July 21 for dry-well fan maintenance. Numerous power reductions were necessary for rod pattern adjustments and weekly turbine tests.

[a]Includes 162.9 h in 1980 from continued 12/13/79 shutdown.

DETAILS OF PLANT OUTAGES FOR HATCH 1

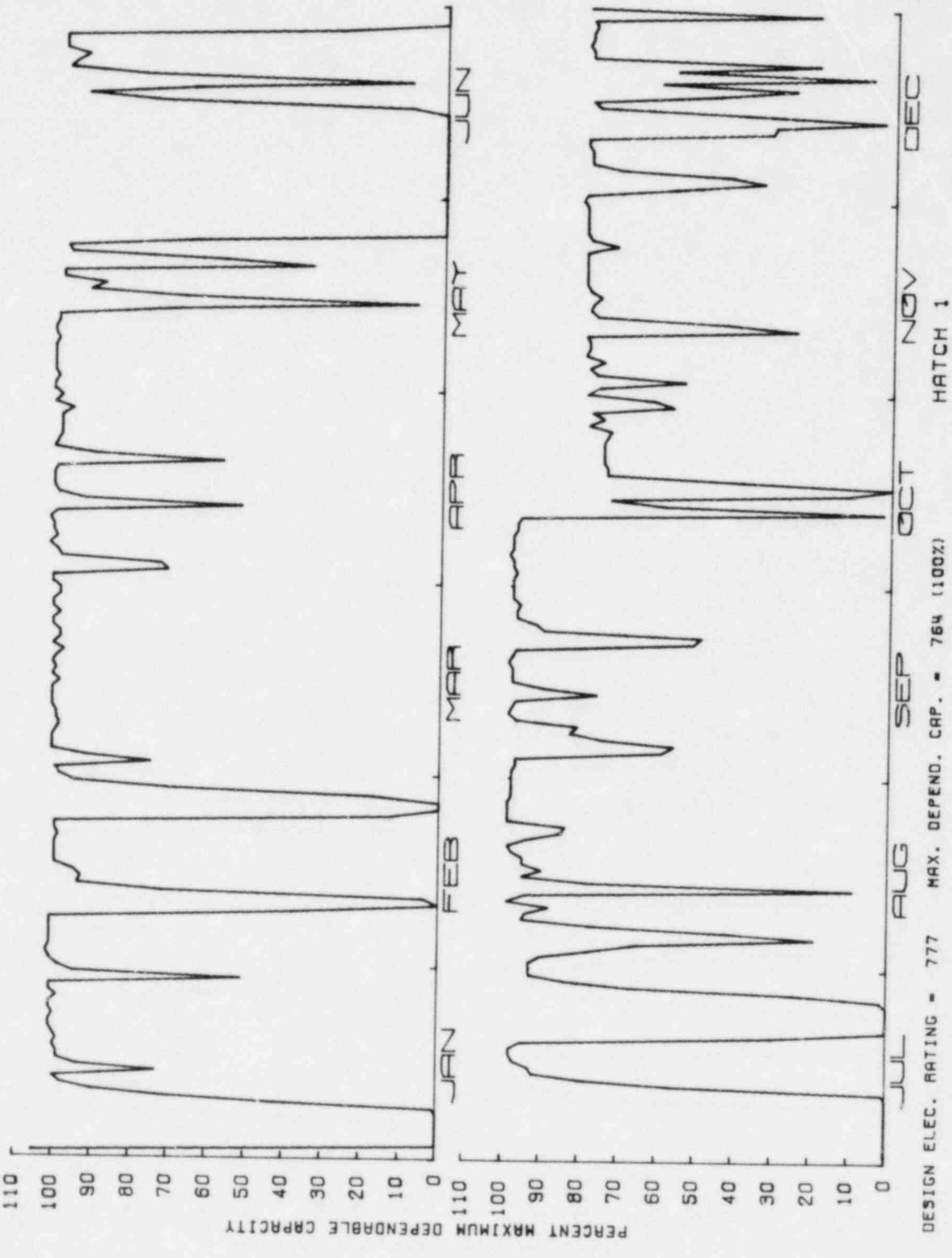
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/13/79 (cont.)	162.9	F	Ground fault in the main generator rotor.	A	4	Steam and power conversion (HA)	Generators
1	2/08	57.7	F	Spurious high RCS pressure signal while RCS pressure switch was being valved in.	H	3	Instrumentation and controls (IB)	Instrumentation and controls
2	2/22	67.3	F	Drywell inspection and steam leak repair.	A	1	Reactor coolant (CH)	Valves
3	2/25	6.3	F	Loss of steam seal pressure on main turbine.	A	1	Steam and power conversion (HD)	Turbines
4	5/13	20.5	F	MSIV not fully open during MSIV test (LER 80-49).	A	3	Reactor coolant (CD)	Valves
5	5/20	8.1	F	Loss of DC power on EHC.	A	3	Electric power (EC)	Mechanical function units
6	5/20	0.5	F	Loss of condenser vacuum.	A	1	Steam and power conversion (HC)	Heat exchangers (condenser)
7	5/24	340.6	F	Inoperable RHR valve (LER 80-53).	A	3	Reactor coolant (CF)	Valves
8	6/08	150.0	F	Unit remained down for mechanical snubber repair (LER 80-33).	A	3	Steam and power conversion (HJ)	Shock suppressors and supports
9	6/14	3.0	F	Loss of condenser vacuum due to steam jet air ejector failure.	A	3	Steam and power conversion (HC)	Not applicable
10	6/15	15.0	F	Condensate demineralizer problems - low DP on suction valves.	A	1	Steam and power conversion (HG)	Demineralizers
11	6/21	1.0	S	Weekly turbine test - extensive check of bypass valves.	B	1	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES FOR HATCH 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12a	6/26	257.0	F	HPCI out of service - problems with electronic and hydraulic systems (LER 80-69).	A	3	Reactor coolant (CJ)	Valves
12b	7/07	115.6	F	Repair to crack in reactor water cleanup return line to feedwater (LER 80-80).	A	4	Reactor coolant (CG)	Pipes, fittings
13a	7/20	12.5	S	Manual scram for control rod testing per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
13b	7/21	130.1	F	Drywell fan maintenance.	B	4	Engineered safety features (SB)	Blowers
14	7/26	10.9	F	Low reactor water level due to loss of feedpump.	A	3	Reactor coolant (CH)	Pumps
15	8/04	18.7	F	Blown fuse in EHC cabinet because operator replaced bulb in control panel with wrong bulb.	G	3	Electric power (EC)	Instrumentation and controls
16	8/12	17.1	F	Flow reactor water level because of FW pump.	A	3	Reactor coolant (CH)	Pumps
17	9/04	11.2	F	False high MSR level caused by LS being grounded.	A	3	Reactor coolant (CH)	Heat exchangers (MSR)
18	9/21	15.8	F	While transferring RPS bus B from alternate to normal, RPS A spuriously tripped.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
19	10/12	20.9	F	TSV closure due to power load in balance on main generator.	A	3	Steam and power conversion (HA)	Valves
20	10/15	45.5	F	Ground fault alarm (LER 80-57).	A	1	Steam and power conversion (HA)	Generators

DETAILS OF PLANT OUTAGES FOR HATCH 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
21	10/29	2.6	F	Ground fault alarm (LER 80-63).	A	1	Steam and power conversion (HA)	Generators
22	11/10	19.2	F	MSIV closure while performing HNP-1-SRV-03005.	B	3	Reactor coolant (CF)	Valves
23	12/02	6.4	F	Investigate ground fault alarm (LER 80-82).	A	1	Steam and power conversion (HA)	Generators
24	12/11	6.5	F	Leaking EHC oil (LER 80-89).	A	2	Steam and power conversion (HB)	Instrumentation and controls
25	12/12	16.2	F	Unit on startup ramp from above shutdown (LERs 80-90,118).	A	9	Steam and power conversion (HA)	Instrumentation and controls
26	12/13	28.9	F	TCV fast closure.	H	3	Steam and power conversion (HA)	Valves
27	12/18	19.5	F	Isolated E11-F060B to restore B loop RHR (LER 80-95).	A	1	Reactor coolant (CF)	Valves
28	12/19	19.0	F	Investigation of EHC low-alarm leak from No. 2 control valve.	A	2	Steam and power conversion (HA)	Valves


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 777
MAX. DEPEND. CAP. = 764 (100%)
HATCH 1

HATCH 2

I. Summary

Description	Performance	Outages
Location: Baxley, Georgia	Net electrical energy generated (MWh): 3,644,977	Total No.: 19
Docket No.: 50-366	Unit availability factor (%): 60.0	Forced: 13
Reactor type: BWR	Unit capacity factor (using MDC): 53.7	Scheduled: 6
Maximum dependable capacity (MWe-net): 773	Unit capacity factor (%) (using design MWe): 52.9	Total hours: 3,510.2 (40.0%)
Commercial operation: 9/05/79		Forced: 534.7 (6.1%)
Years operating experience: 2.3		Scheduled: 2,975.5 (33.9%)

II. Highlights

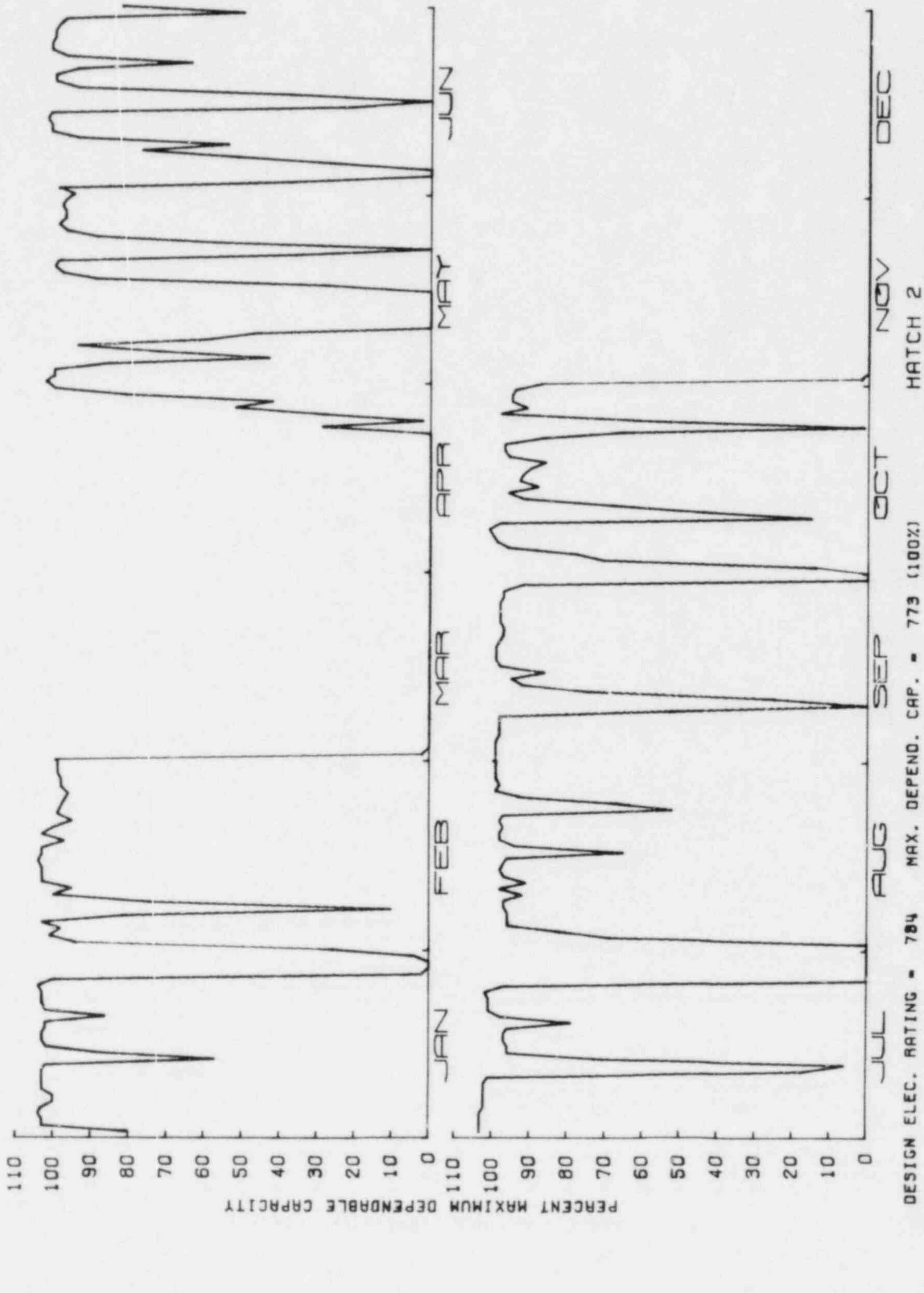
Numerous power reductions were necessary at Hatch 2 in 1980 for rod pattern adjustments and sequence exchanges and weekly turbine tests. On March 1, the unit went down for over seven weeks for an 18-month surveillance outage and maintenance on a vent header deflector. Turbine and turbine control problems caused shutdowns on April 24, May 8, May 22, June 2, June 14, and October 1. Refueling began on November 1 and continued through the end of December.

DETAILS OF PLANT OUTAGES FOR HATCH 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/26	81.4	S	Modifications to primary containment isolation.	D	1	Engineered safety features (SD)	Not applicable
2	1/29	1.4	S	Unit taken off line to perform turbine overspeed test.	B	9	Steam and power conversion (HA)	Turbines
3	2/06	15.7	F	Blown packing on the seal water regulator valve. Also, loss of condenser vacuum due to loss of seal water loop seal.	A	1	Reactor coolant (CH)	Valve
4	3/01	1267.8	S	Vent header deflector maintenance and 18-month surveillance testing.	H	2	Engineered safety features (SH)	Not applicable
5	4/23	0.5	S	Turbine overspeed testing.	B	2	Steam and power conversion (HA)	Turbines
6	4/24	27.4	F	Momentary loss of DC power to electro-hydraulic cooling system.	A	3	Electric power (EC)	Turbines
7	5/08	174.0	F	Ground fault in No. 8 turbine-generator bearing.	A	1	Steam and power conversion (HA)	Turbines
8	5/21	23.7	F	False low level signal due to test shop isolating wrong valve.	H	3	Engineered safety features (SH)	Valves
9	5/22	11.9	F	Generator off line due to 4th stage extraction motor-operated valve not operating.	A	2	Steam and power conversion (HJ)	Valves
10	6/02	68.7	F	Turbine vibrations. Snubber work performed. Shutdown extended for snubber repair in condenser bay.	A	3	Steam and power conversion (HA)	Turbines
11	6/14	46.4	F	Turbine vibration - main bearing No. 9 replaced.	A	3	Steam and power conversion (HA)	Turbines

DETAILS OF PLANT OUTAGES FOR HATCH 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12	7/11	35.5	F	Channel A of RPS deenergized and received a group 1 signal. Channel B of RPS was deenergized before A could be reset. Received group 1 signal through both channels resulting in scram (LER 80-102).	A	3	Instrumentation and controls (IA)	Instrumentation and controls
13	7/26	163.2	S	Manual scram to test scram discharge volume per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
14	9/08	40.3	F	MSIV not fully open.	A	3	Reactor coolant (CD)	Valve operators
15	9/29	43.7	F	Low condenser vacuum.	A	3	Steam and power conversion (HC)	Heat exchangers (condenser)
16	10/01	11.1	F	TSV closure tripped turbine on high water level.	A	1	Steam and power conversion (HA)	Valves
17	10/09	15.7	F	Condenser booster pump trip on low suction which tripped the reactor feed pumps.	A	1	Steam and power conversion (HH)	Pumps
18	10/24	20.6	F	Repair leak in water box.	A	3	Steam and power conversion (HF)	Heat exchanger
19	11/01	1461.2	S	Refueling.	C	1	Reactor (RC)	Fuel elements



PERCENT MAXIMUM DEPENDABLE CAPACITY
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JUL AUG SEP OCT NOV DEC
DESIGN ELEC. RATING = 784 MAX. DEPEND. CAP. = 773 (100%)
HATCH 2

INDIAN POINT 2

I. Summary

Description	Performance	Outages
Location: Indian Point, New York (25 miles north of New York City)	Net electrical energy generated (MWh): 4,264,224	Total No.: 14
Docket No.: 50-247	Unit availability factor (%): 64.8	Forced: 13
Reactor type: PWR	Unit capacity factor (using MDC): 56.7	Scheduled: 1
Maximum dependable capacity (MWe-net): 864	Unit capacity factor (%) (using design MWe): 55.6	Total hours: 3,092.8 (35.2%)
Commercial operation: 8/73		Forced: 2,373.1 (27.0%)
Years operating experience: 7.5		Scheduled: 719.7 (8.2%)

II. Highlights

The unit was taken down for low-pressure turbine inspection on January 11 and remained down for 30 d. Several condenser tube leaks were repaired between February 14 and 20. Several steam generator level trips occurred upon startup. Operation was routine at a power restriction of 834 MWe-net because of removal of a disk on a low-pressure turbine rotor. Lightning strikes caused shutdowns in June and July. The unit tripped on a high pressurizer pressure signal on November 17 and was brought to a cold shutdown to assess equipment damage inside containment on November 22 because of fan cooler heat exchanger leakage. The unit remained down for the remainder of the year.

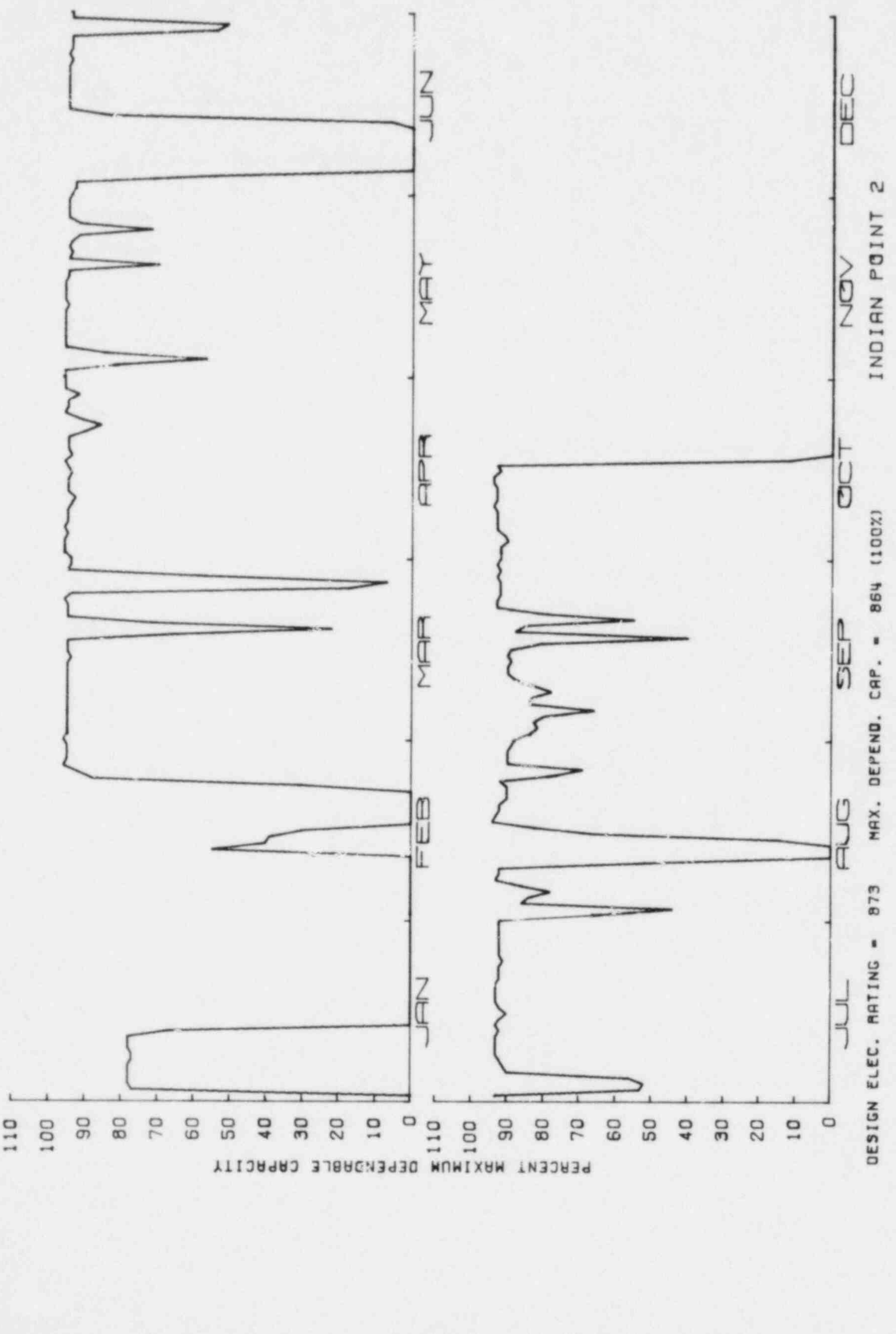
DETAILS OF PLANT OUTAGES FOR INDIAN POINT 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/11	719.7	S	LP turbine inspection.	D	3	Steam and power conversion (HA)	Turbines
2	2/14	134.4	F	Double-ended tube break in the No. 24 condenser.	A	2	Steam and power conversion (HC)	Heat exchangers (condenser)
3	2/20	4.0	F	SG low level during startup.	A	3	Steam and power conversion (HH)	Heat exchangers (steam generators)
4	2/20	7.1	F	SG low level during startup.	A	3	Steam and power conversion (HH)	Heat exchangers (steam generators)
5	2/20	15.5	F	SG high level during startup. Repairs made to No. 24 FW regulator.	A	3	Steam and power conversion (HH)	Valves
6	3/26	36.6	F	SG low level; air line to HDT dump valve to No. 23 condenser parted.	A	3	Steam and power conversion (HH)	Heat exchangers (condenser)
7	5/19	3.6	F	Improper sequencing of breakers while returning No. 22 CRDM motor-generator set to service.	G	3	Reactor (RB)	Control rod drives
8	6/03	217.2	F	Loss of offsite power due to lightning strike (LER 80-06). Also, condenser tube leak repairs.	A	3	Electric power (EA)	Not applicable
9	6/27	10.6	F	Loss of generator exitation due to defective overcurrent relay.	A	3	Steam and power conversion (HA)	Generators
10	7/02	16.6	F	Lightning strike on system.	H	3	Electric power (EG)	Not applicable
11	7/04	5.2	F	Spurious trip signal on main turbine auto stop oil system.	A	3	Steam and power conversion (HA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR INDIAN POINT 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12	8/01	11.7	F	Lo level in No. 23 SG because of No. 22 main boiler feed pump trip.	A	3	Steam and power conversion (HH)	Pumps
13	8/10	89.9	F	Repair main turbine oil cooler.	A	2	Steam and power conversion (HA)	Heat exchangers (cooler)
14a	10/17	120.0	F	High pressurizer pressure signal due to local turbine load limit being moved in decreasing direction.	A	3	Steam and power conversion (HA)	Instrumentation and controls
14b	10/22	1700.7	F	Unit brought to cold shutdown to assess equipment damage from submergence of reactor vessel in raw cooling water resulting from fan cooling unit leakage. Fan cooler heat exchangers to be replaced.	A	4	Engineered safety features (SB)	Heat exchangers (cooler)
14c	10/22	*a*	F	Refueling outage commenced. Also, all three LP turbines have been disassembled for inspection and restacking of the rotor on No. 23 turbine.	A	4	Reactor (RC)	Fuel elements

[a] Total hours included in part 14b.


PERCENT MAXIMUM DEPENDABLE CAPACITY
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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
Maximum dependable capacity (MWe-net): 965
Commercial operation: 8/30/76
Years operating experience: 4.7

Performance

Net electrical energy generated (MWh): 3,070,723
Unit availability factor (%): 53.2
Unit capacity factor (using MDC): 36.2
Unit capacity factor (%) (using design MWe): 36.2

Outages

Total No.: 29
- Forced: 26
- Scheduled: 3

Total hours: 4,111.7 (46.8%)[a]
- Forced: 1,482.5 (16.9%)
- Scheduled: 2,629.2 (29.9%)

II. Highlights

Operation began on February 11 after an extended refueling outage which began September 14, 1979. The unit was shut down for a week on March 6 to flush both main feedwater pump control systems. Nearly two weeks were required to replace bearings in the condensate water pumps beginning April 6. Repairs took two weeks in July because of an electrical fault in feeds associated with the reactor coolant pumps. The unit was off-line from September 20 until December 20 for turbine repairs, fire protection modifications, fan cooler unit inspection and repair, and reactor coolant pump rotor and stator replacement.

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[a]Includes 1,005.7 h in 1980 from continued 9/14/79 shutdown.

DETAILS OF PLANT OUTAGES FOR INDIAN POINT 3

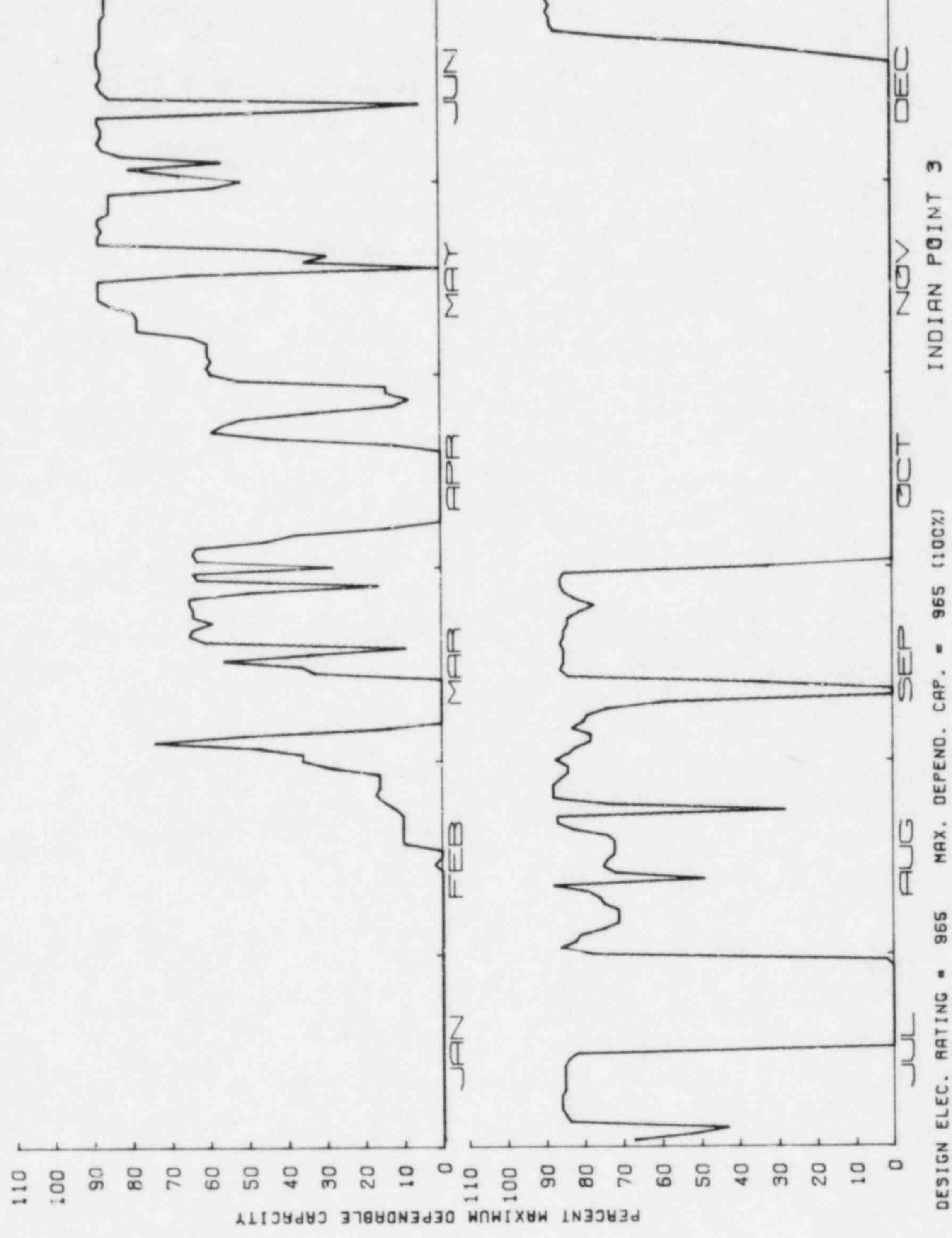
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	9/14/79 (cont.)	1005.7	S	Refueling.	C	4	Reactor (RC)	Fuel elements
1	2/11	32.5	F	Steam leak out of the vents of the moisture separator reloaders.	A	3	Steam and power conversion (HB)	Heat exchanger (MSR)
2	2/13	54.5	S	Surveillance of main turbine.	B	1	Steam and power conversion (HA)	Turbines
3	3/04	7.2	F	Loss of MFWP.	A	3	Steam and power conversion (HH)	Pumps
4	3/05	10.6	F	MSIV inadvertently tripped: installed cages over switches.	G	3	Steam and power conversion (HB)	Valves
5	3/05	4.9	F	Thrust bearing turbine trip: detector cleaned.	A	3	Steam and power conversion (HA)	Turbines
6	3/06	188.6	F	Flushed both MFWP control systems.	B	3	Steam and power conversion (HH)	Pumps
7	3/17	24.7	S	Leaking weld in MFW drain line.	B	1	Steam and power conversion (HH)	Pipes, fittings
8	3/22	0.0	F	Intake screens blocked.	B	9	Steam and power conversion (HF)	Filters
9	3/27	15.7	F	Steam flow/feedwater flow mismatch due to air introduced into system while valving in a condensate pump.	G	3	Steam and power conversion (HH)	Pumps
10	3/28	3.8	F	Thrust bearing turbine trip: detector cleaned.	A	3	Steam and power conversion (HA)	Turbines
11	3/2[illegible]	10.2	F	Static inverter failure.	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR INDIAN POINT 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12	4/06	302.6	F	Circulating water pump repair, replacing bearings.	A	1	Steam and power conversion (HF)	Pumps
13	4/26	4.4	F	345-kV feeder trip due to spurious relay actuation.	H	3	Electric power (EA)	Relays
14	4/26	3.2	F	Turbine overspeed trip due to loose connections.	A	3	Steam and power conversion (HA)	Turbines
15	5/16	15.5	F	Loss of load due to misoperation of transformer relay at substation.	A	3	Electric power (EA)	Relays
16	5/19	22.7	F	Voltage transient on instrument bus No. 33; repaired components within static inverter No. 33.	A	3	Electric power (ED)	Generators
17	5/30	7.7	F	False actuation of independent electric turbine overspeed protection system.	A	3	Steam and power conversion (HA)	Instrumentation and controls
18	6/11	33.4	F	Misoperation of NGA relay at substation caused direct trip of plant.	A	3	Electric power (EA)	Relays
19	6/30	5.0	F	Low SG level due to loss of MFWPs as a result of perturbations in MFW control oil system while shifting oil pumps; pump check valve hung up.	A	3	Steam and power conversion (HH)	Valves
20	7/02	6.7	F	Lightning strike on 345-kV transmission line inducing voltage transient on instrument bus No. 34 coincedent with another protection channel in trip mode for a surveillance test.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
21	7/15	356.2	F	Electrical fault in feeds associated with RCPs.	A	3	Electric power (EB)	Electrical conductors

DETAILS OF PLANT OUTAGES FOR INDIAN POINT 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
22	8/12	3.2	F	Loss of all circulating water to No. 33 condenser bay when No. 36 circulating water pump tripped with No. 35 circulating water pump down for repairs.	A	3	Steam and power conversion (HF)	Pumps
23	8/23	7.2	F	Auto synchronizing device malfunction, while placing repaired motor generator set in service.	A	3	Electric power (EB)	Mechanical function units
24	9/09	61.2	F	SG No. 32 mismatch caused by loss of No. 33 static inverter. Replaced capacitors on static inverter. Outage extended due to fault on unit auxiliary transformer.	A	3	Electric power (ED)	Instrumentation and controls
25a	9/30	913.0	S	Turbine outage.	A	1	Steam and power conversion (HA)	Turbines
26b	11/06	432.0	S	Fire protection modification.	B	4	Auxiliary (AB)	Other
27c	11/25	199.3	S	Fan cooler unit inspection, repair, and testing.	B	4	Engineered safety features (SB)	Heat exchangers (cooler)
28d	12/04	372.9	F	Electrical fault in No. 33 RCP stator. Replaced rotor and stator.	A	4	Reactor coolant (CB)	Motors
29	12/19	7.3	F	Power level drifted above trip setpoint before manual block was applied.	G	3	Instrumentation and controls (IA)	Instrumentation and controls


PERCENT MAXIMUM DEPENDABLE CAPACITY
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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
Maximum dependable capacity (MWe-net): 522
Commercial operation: 6/16/74
Years operating experience: 6.7

Performance

Net electrical energy generated (MWh): 3,631,892
Unit availability factor (%): 82.1
Unit capacity factor (using MDC): 79.2
Unit capacity factor (%) (using design MWe): 77.3

Outages

Total No.: 9
Forced: 7
Scheduled: 2
Total hours: 1,570.9 (17.9%)
Forced: 1,095.8 (12.5%)
Scheduled: 475.1 (5.4%)

II. Highlights

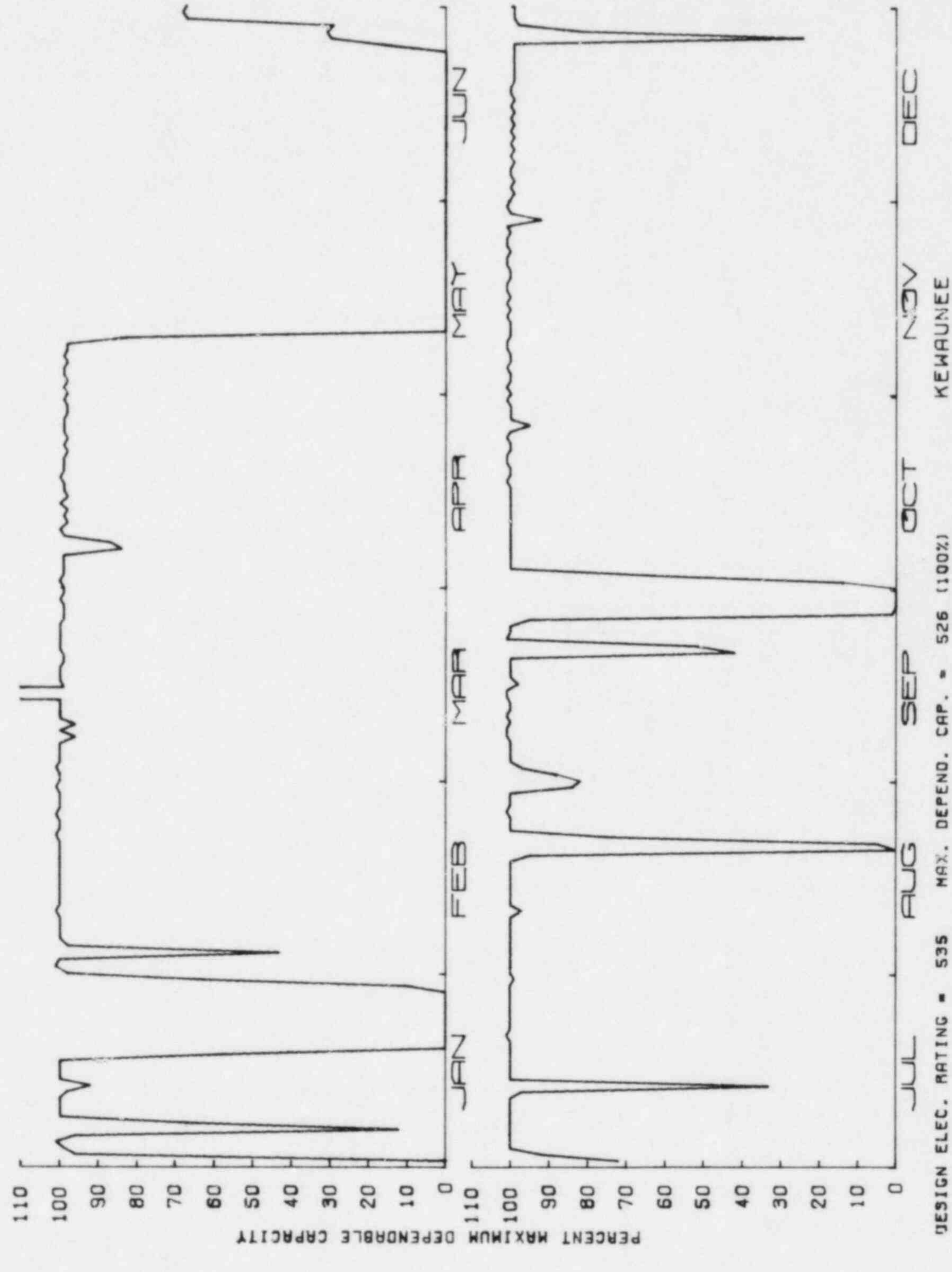
After the main and auxiliary transformers and reactor coolant pump failures in January, operation was at or near full power until the May 9 refueling. The reactor trip on December 26 ended 85 d of continuous power operation. The unit had an 82.1% availability factor and 79.2% (MDC) and 77.3% (DER) capacity factors in 1980.

DETAILS OF PLANT OUTAGES FOR KEWAUNEE

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/04	15.4	F	Bus fault from main auxiliary transformer.	A	3	Electric power (EB)	Electrical conductors
2a	1/17	158.7	F	Bushing failure in reserve auxiliary transformer causes loss of power to all but safeguards buses (LER 80-02).	A	3	Electric power (EB)	Transformers
2b	1/17	106.0	F	RCP seal failure.	A	4	Reactor coolant (CB)	Mechanical function units
3	2/03	4.1	F	Low SG level due to solenoid valve failure on the air system for the 1B MFW control valve.	A	3	Steam and power conversion (HH)	Valve operators
4	5/09	1087.3	S	Refueling.	C	1	Reactor (RC)	Fuel elements
5	6/26	8.5	S	Balance weight adjustment on turbine.	B	1	Steam and power conversion (HA)	Turbines
6	7/13	5.1	F	Low SG level due to a solenoid failure on air system to control valve of 1A MFW.	A	3	Steam and power conversion (HH)	Valve operators
7a	8/19	13.0	F	A lightning strike caused the failure of two instrument bus inverters. Safety injection actuated.	A	3	Instrumentation and controls (IA)	Instrumentation
7b	8/20	29.3	F	A bus fault on the line from the reserve auxiliary transformer to buses 1 and 2 extended the 8/19 outage.	A	4	Electric power (EB)	Electrical conductors

DETAILS OF PLANT OUTAGES FOR KEWAUNEE (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8	9/26	130.7	F	A disk/stem separation in loop isolation valve of loop B RTD by-pass caused loss of flow in the RTD line. Outage was extended when a similar valve failed in the same manner after it was operated to isolate the bypass loop during maintenance (LER 80-32).	A	1	Reactor coolant (CB)	Valves
9	12/26	12.8	F	During monthly stop valve testing, spurious rapid opening of a turbine control valve caused a step increase in steam demand resulting in an SG high level trip.	H	3	Steam and power conversion (HA)	Valves


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 535 MAX. DEPEND. CAP. = 526 (100%)
KEWAUNEE

LA CROSSE

I. Summary

Description

Location: Genoa, Wisconsin
Docket No.: 50-409
Reactor type: BWR
Maximum dependable capacity (MWe-net): 48
Commercial operation: 11/01/69
Years operating experience: 12.7

Performance

Net electrical energy generated (MWh): 214,545
Unit availability factor (%): 68.6
Unit capacity factor (using MDC): 50.9
Unit capacity factor (%) (using design MWe): 48.8

Outages

Total No.: 9
Forced: 6
Scheduled: 3
Total hours: 2,757.5 (31.4%)
Forced: 901.1 (10.3%)
Scheduled: 1,856.4 (21.1%)

II. Highlights

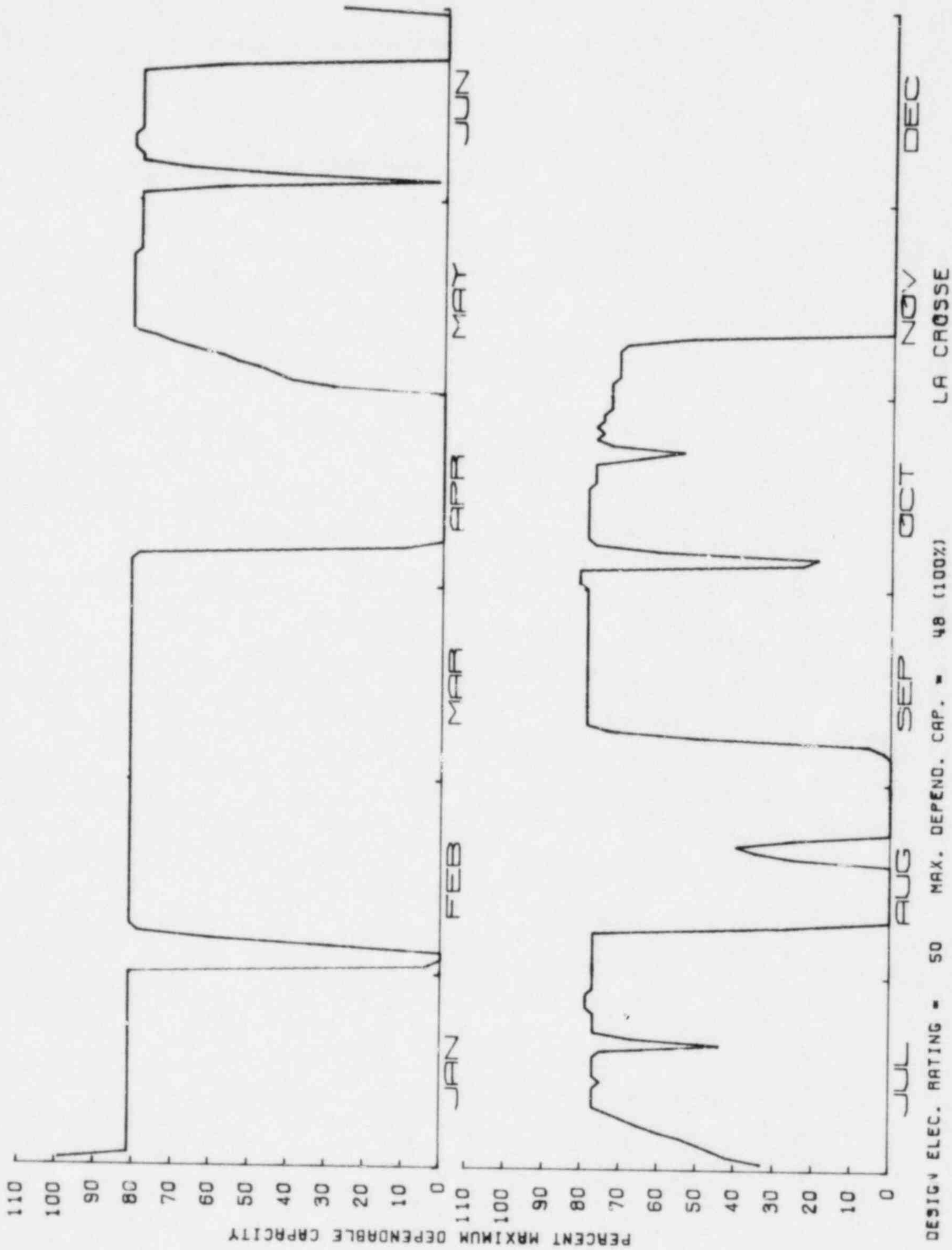
The few outages that occurred at La Crosse this year usually were of substantial duration. The plant was down for 3 d beginning February 1 because of a momentary low voltage signal on a 480-V bus, for over 24 d beginning April 6 for TMI-related modifications, for 8 d beginning June 21 for seal repairs on a control rod drive motor and a circulation pump, for over 13 d starting August 8 for recalibration of level controllers and indicators in the seal injection reservoir, and for the remainder of the year beginning November 9 for a refueling outage.

DETAILS OF PLANT OUTAGES FOR LA CROSSE

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/01	74.5	F	Momentary low voltage signal at turbine building 480-V 1A motor control center; cause undetermined.	A	3	Electric power (EB)	Electrical conductors
2	4/06	581.6	S	NUREG-0578 modification including position indicators on relief valves and manual resets on containment isolations.	D	1	Other (XX)	Other
3	6/02	24.2	S	Operator license examinations.	E	1	System code not applicable (ZZ)	N/A
4	6/21	193.7	F	Seal leak repair in upper CRDM No. 24 and seal repair on forced circulation pump 1A.	A	1	Reactor (RB)	Control rods
5	8/08	253.7	F	Forced circulation pumps tripped due to loss of seal injection flow. Loss of seal injection flow was caused by low level in reservoir due to level controller malfunction. Level controllers and indicators recalibrated.	A	3	Reactor coolant (CJ)	Instrumentation and controls
6	8/22	319.8	F	Reactor feed pump 1A triped due to an electrical short on a printed circuit control card caused by a water leak from an overhead floor. During this shutdown maintenance was performed on the seal injection system and the mechanical seals in CRDMs Nos. 5 and 21.	A	3	Reactor coolant (CH)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR LA CROSSE (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7	9/05	32.1	F	Electrical short of CRDM No. 3 caused by spray from leaking seal injection supply line convection on CRDM No. 1, resulting in a partial scram.	A	3	Reactor (RB)	Control rod drives
8	10/04	27.3	F	High reactor water level due to failure of controller amplifier on reactor feedpump 1B, which caused the pump to fail high. Cause believed to be water-detergent mixture which splashed onto controllers 8/22/80.	A	3	Reactor coolant (CH)	Circuit closers/ interrupters
9	11/09	1250.6	S	Refueling.	C	1	Reactor (RC)	Fuel elements


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

MAINE YANKEE

I. Summary

Description	Performance	Outages
Location: Wincasset, Maine	Net electrical energy generated (MWh): 4,404,138	Total No.: 19
Docket No.: 50-309	Unit availability factor (%): 72.2	Forced: 14
Reactor type: PWR	Unit capacity factor (using MDC): 61.9	Scheduled: 5
Maximum dependable capacity (MWe-net): 810	Unit capacity factor (%) (using design MWe): 60.8	Total hours: 2,443.7 (27.8%)
Commercial operation: 12/28/72		Forced: 854.5 (9.7%)
Years operating experience: 8.1		Scheduled: 1,589.2 (18.1%)

II. Highlights

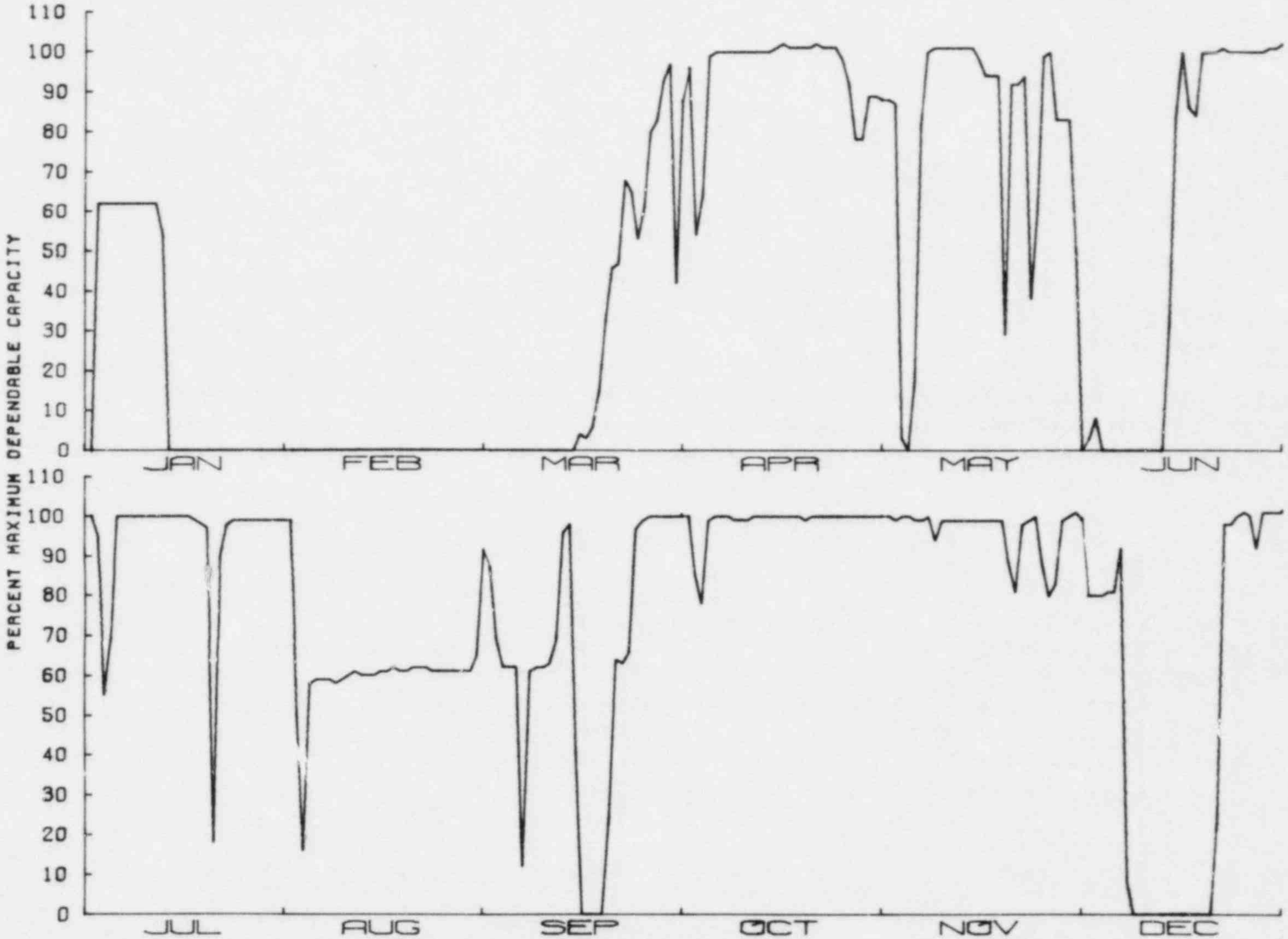
A refueling outage began on January 11, and the generator was put back on-line on March 14. Four brief shutdowns were necessary in the next 3 d to balance the turbine. Core crud was a problem requiring a few shutdowns and power reductions for hydrogen peroxide cleaning. In September and December, reactor coolant pump seal failures caused lengthy outages.

DETAILS OF PLANT OUTAGES FOR MAINE YANKEE

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/11	1480.3	S	Refueling.	C	1	Reactor (RC)	Fuel elements
2	3/10	2.8	S	Low power physics testing. Generator not on line yet.	B	1	System code not applicable (ZZ)	Not applicable
3	3/12	28.6	F	Broken wire in RPS logic ladder during surveillance testing (LER 80-6). Generator not on line yet.	A	3	Instrumentation and controls (IA)	Not applicable
4	3/12	1.2	S	Operator training.	E	1	System code not applicable (ZZ)	Not applicable
5	3/15	8.9	F	Add balance weight to turbine.	B	1	Steam and power conversion (HA)	Turbines
6	3/16	8.0	F	Add balance weight to turbine.	B	1	Steam and power conversion (HA)	Turbines
7	3/16	6.1	F	Add balance weight to turbine.	B	1	Steam and power conversion (HA)	Turbines
8	3/17	5.8	F	Add balance weight to turbine.	B	1	Steam and power conversion (HA)	Turbines
9	4/02	11.3	F	Grounded capacitor in SG level transmitter during RPS surveillance testing.	A	3	Instrumentation and controls (IB)	Instrumentation and controls
10	5/03	57.6	S	Core crud cleanup using hydrogen peroxide.	H	1	Reactor (RC)	Vessels, pressure
11	5/19	9.3	F	False temperature signals cause two RPS channels to trip due to spurious electrical spike.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
12	5/24	14.9	F	False temperature signals cause two RPS channels to trip due to spurious electrical spike.	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR MAINE YANKEE (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13	5/30	47.3	S	Core crud cleanup.	H	1	Reactor (RC)	Vessels, pressure
14	6/02	255.4	F	Failure of No. 1 SG nonreturn valve disk (LER 80-16).	A	1	Steam and power conversion (HB)	Valves
15	7/20	13.1	F	Operator mistakenly opened test valve on turbine thrust bearing system while taking routine readings.	G	3	Steam and power converson (HA)	Instrumentation and controls
16	8/02	25.4	F	Major failure of the P-2A MFWP. Unable to determine exact cause of failure. New pump and rotating assembly installed while plant was at 60-65% power through 8/30.	A	3	Steam and power conversion (HH)	Pumps
17	9/06	14.5	F	Spurious opening of the CRDM MG-set output breakers. Exact cause unknown.	A	3	Instrumentation and controls (IA)	Circuit closers/ interrupters
18	9/14	115.5	F	Failure of two of the four RCP seal states. Seal cartridge unit replaced.	A	1	Reactor coolant (CB)	Pumps
19	12/07	337.7	F	A load reduction to take plant off-line due to an RCP seal failure was in progress when a turbine EHC load limit control stuck in "lower" mode, causing a plant trip. Cause not determined. Numbers 1 and 2 RCP seal cartridges were replaced. During plant heatup, No. 2 RCP seal cartridge indicated two failed stages. No. 2 seal cartridge was again replaced.	A	1	Reactor coolant (CB)	Pumps



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

MILLSTONE 1

I. Summary

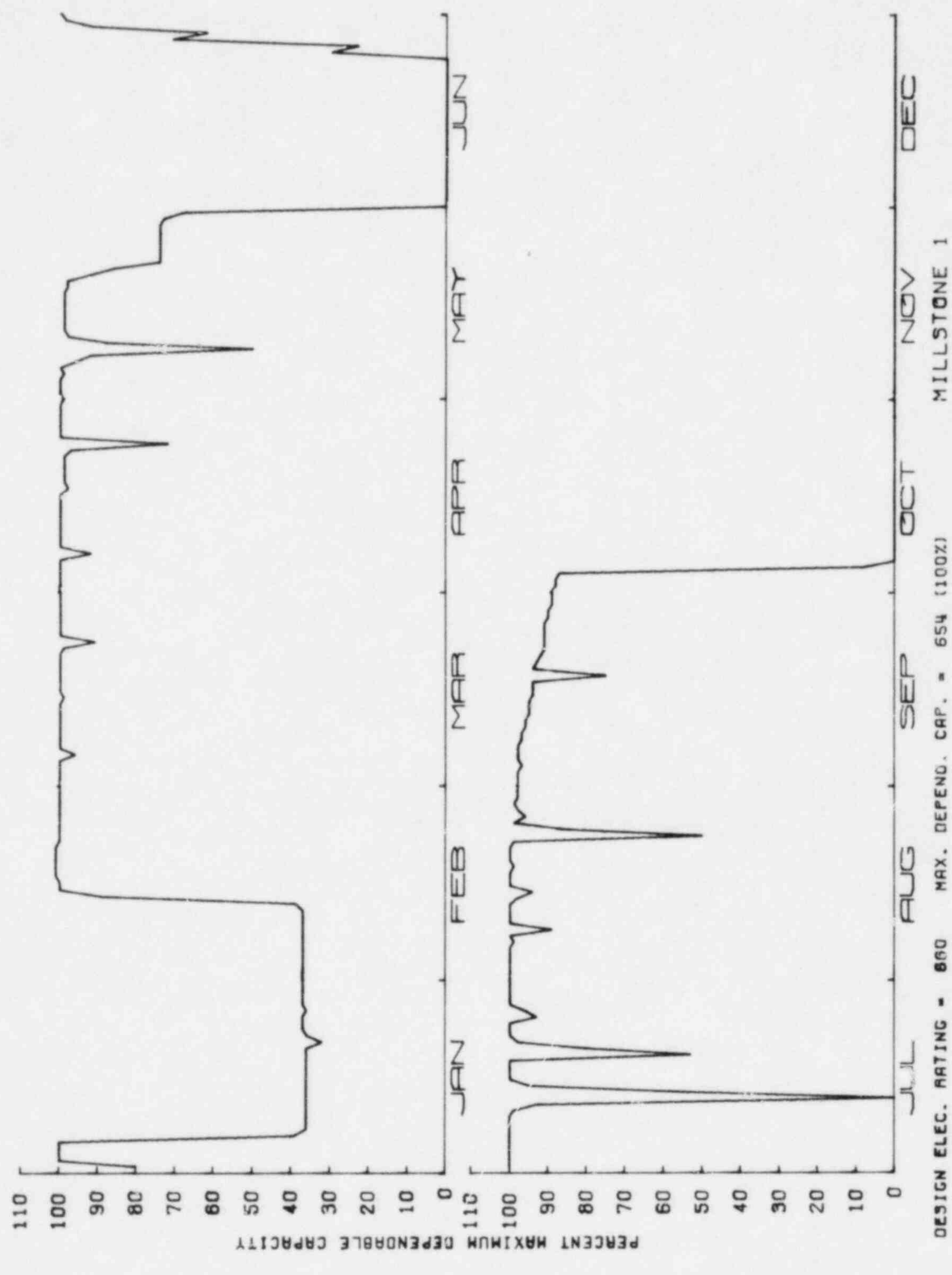
Description	Performance	Outages
Location: Waterford, Connecticut	Net electrical energy generated (MWh): 3,390,215	Total No.: 4
Docket No.: 50-245	Unit availability factor (%): 69.0	Forced: 1
Reactor type: BWR	Unit capacity factor (using MDC): 59.0	Scheduled: 3
Maximum dependable capacity (MWe-net): 654	Unit capacity factor (%) (using design MWe): 58.5	Total hours: 2,393.7 (27.3%)
Commercial operation: 3/71		Forced: 13.2 (0.2%)
Years operating experience: 10.1		Scheduled: 2,380.5 (27.1%)

II. Highlights

Power was restricted to 40% from January 5 to February 11 because of the isolation condenser being out of service. Main condenser tube leaks were repaired during the power reductions of April 23, May 8, July 19, August 23, and September 17. A refueling and maintenance outage lasted from October 4 through the end of the year.

DETAILS OF PLANT OUTAGES FOR MILLSTONE 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a	5/30	23.5	S	Repair steam leak in extraction joint off LP turbine.	A	3	Steam and power conversion (HJ)	Turbines
1b	6/16	197.8	S	Questionable integrity of LPSI subsystem B injection piping at penetration X-45 (LER 80-10).	H	4	Engineered safety features (SF)	Shock suppressors and supports
2	6/25	13.2	F	Electric pressure regulator malfunction induced average power range monitor scram. Pressure control was transferred to mechanical.	A	3	Reactor coolant (CC)	Instrumentation and controls
3	7/12	26.5	S	Manual and then automatic scram testing of control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
4	10/04	2132.7	S	Refueling and maintenance (LERs 80-18, 19).	C	1	Reactor (RC)	Fuel elements


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

MILLSTONE 2

I. Summary

Description	Performance	Outages
Location: Waterford, Connecticut	Net electrical energy generated (MWh): 4,881,788	Total No.: 13
Docket No.: 50-336	Unit availability factor (%): 69.2	Forced: 10
Reactor type: PWR	Unit capacity factor (using MDC): 64.3	Scheduled: 3
Maximum dependable capacity (MWe-net): 864	Unit capacity factor (%) (using design MWe): 63.9	Total hours: 2,835.1 (32.3%)
Commercial operation: 12/26/75		Forced: 183.9 (2.1%)
Years operating experience: 5.1		Scheduled: 2,651.2 (30.2%)

II. Highlights

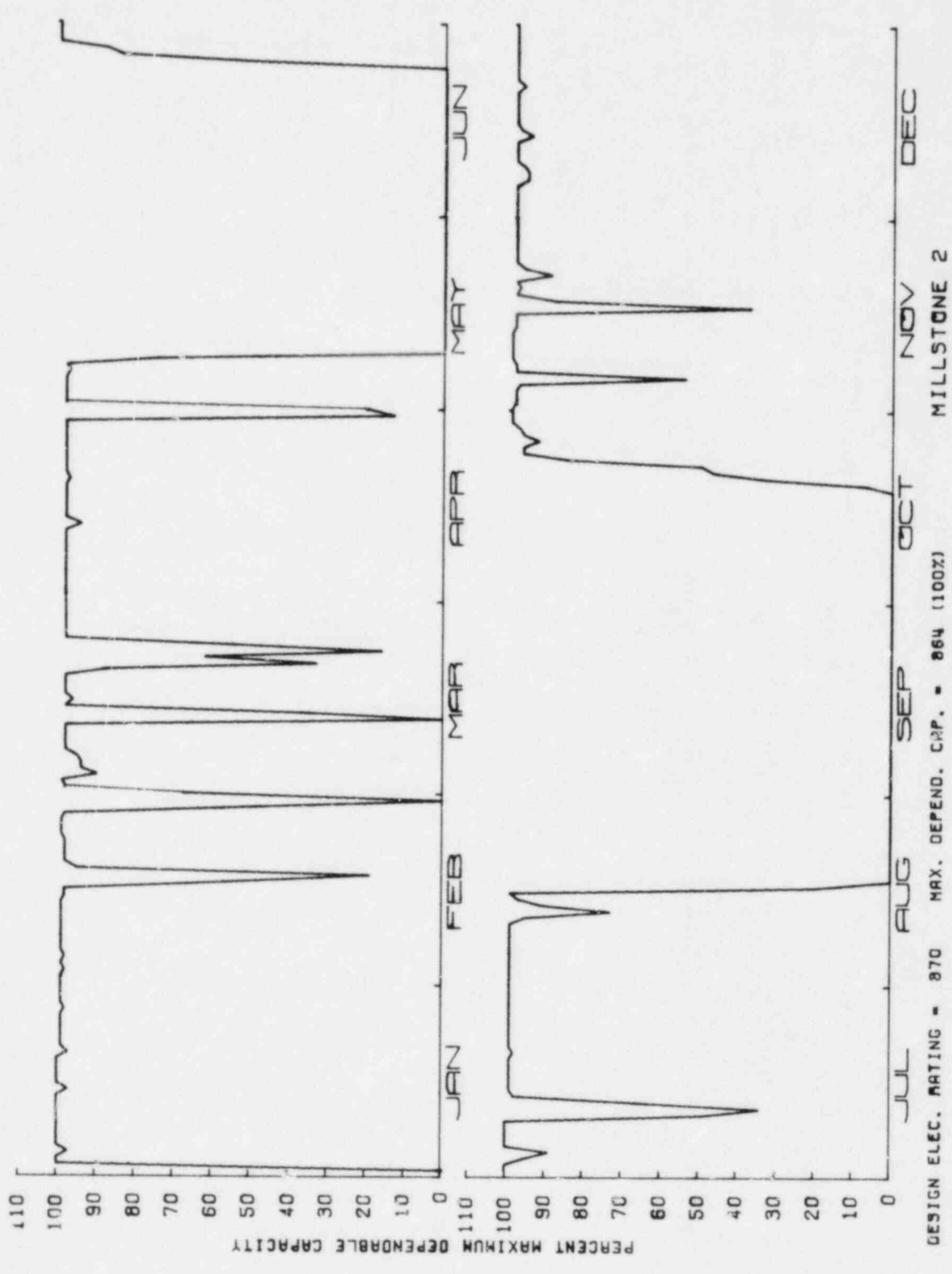
Steam generator regulator valves failed closed and caused five low steam generator level scrams on February 27, March 21 and 23, and April 29 and 30. Control rods dropped into the core and caused power reductions on March 20, August 11, and December 6. Refueling commenced August 16, and the reactor was brought critical October 13. After the turbine overspeed trip test on October 19, the unit operated routinely at or near full power for the remainder of the year.

DETAILS OF PLANT OUTAGES FOR MILLSTONE 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/15	23.5	F	Trip circuit breaker-2 opened during reactor system matrix logic and trip path relay test.	H	3	Instrumentation and controls (IA)	Circuit closers/ interrupters
2	2/26	16.3	F	Main generator lockout due to inadvertent actuation of fault trip circuitry during testing at the site main switchyard.	H	3	Electric power (EE)	Not applicable
3	2/27	19.1	F	Low SG level due to FW regulator valve lockup (closed); valve overhauled.	A	3	Steam and power conversion (HH)	Valves
4	3/12	29.0	F	Low SG level due to FW regulator valve 2-FW-51B lockup (closed).	A	3	Steam and power conversion (HH)	Valves
5	3/21	18.3	F	Low SG level due to instrumentation technician who removed MFWP suction pressure transmitter from service for calibration.	B	3	Instrumentation and controls (IF)	Pumps
6	3/23	18.7	F	Low SG level due to FW regulator valve (2-FW-51A) lockup (closed) during a power reduction for a FW heater problem.	A	3	Steam and power conversion (HH)	Valves
7	4/29	25.5	F	Low SG level due to FW regulator valve (2-FW-51A) locking closed after power reduction. A stuck open pressurizer spray valve had caused the RCS pressure to drop (LER 80-20).	A	3	Steam and power conversion (HH)	Valves
8	4/30	8.4	F	Low SG level due to FW regulator valve (2-FW-51A) lockup (closed).	A	3	Steam and power conversion (HH)	Valves

DETAILS OF PLANT OUTAGES FOR MILLSTONE 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	5/08	1111.4	S	Pipe support reevaluation and modifications per IE Bulletin 79-02 (LER 80-18).	D	1	Engineered safety features (SH)	Pipes, fittings
10	7/10	21.4	F	Low SG level due to instrument air line break causing loss of speed control to MFWP and then loss of MFWP.	A	3	Steam and power conversion (HH)	Pipes, fittings
11	7/12	3.7	F	Low SG level due to problems with heater drains level control causing an MFWP trip on low suction pressure.	A	3	Steam and power conversion (HH)	Instrumentation and controls
12	8/16	1538.5	S	Refueling.	C	1	Reactor (RC)	Fuel elements
13	10/19	1.3	S	Turbine overspeed trip test.	B	1	Steam and power conversion (HA)	Turbines


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 870 MAX. DEPEND. CAP. = 864 (100%)
MILLSTONE 2

MONTICELLO

I. Summary

Description	Performance	Outages
Location: Monticello, Minnesota	Net electrical energy generated (MWh): 3,453,799	Total No.: 9
Docket No.: 50-263	Unit availability factor (%): 78.3	Forced: 4
Reactor type: BWR	Unit capacity factor (using MDC): 73.4	Scheduled: 5
Maximum dependable capacity (MWe-net): 536	Unit capacity factor (%) (using design MWe): 72.1	Total hours: 1,906.9 (21.7%)
Commercial operation: 6/30/71		Forced: 567.5 (6.5%)
Years operating experience: 9.8		Scheduled: 1,339.4 (15.2%)

II. Highlights

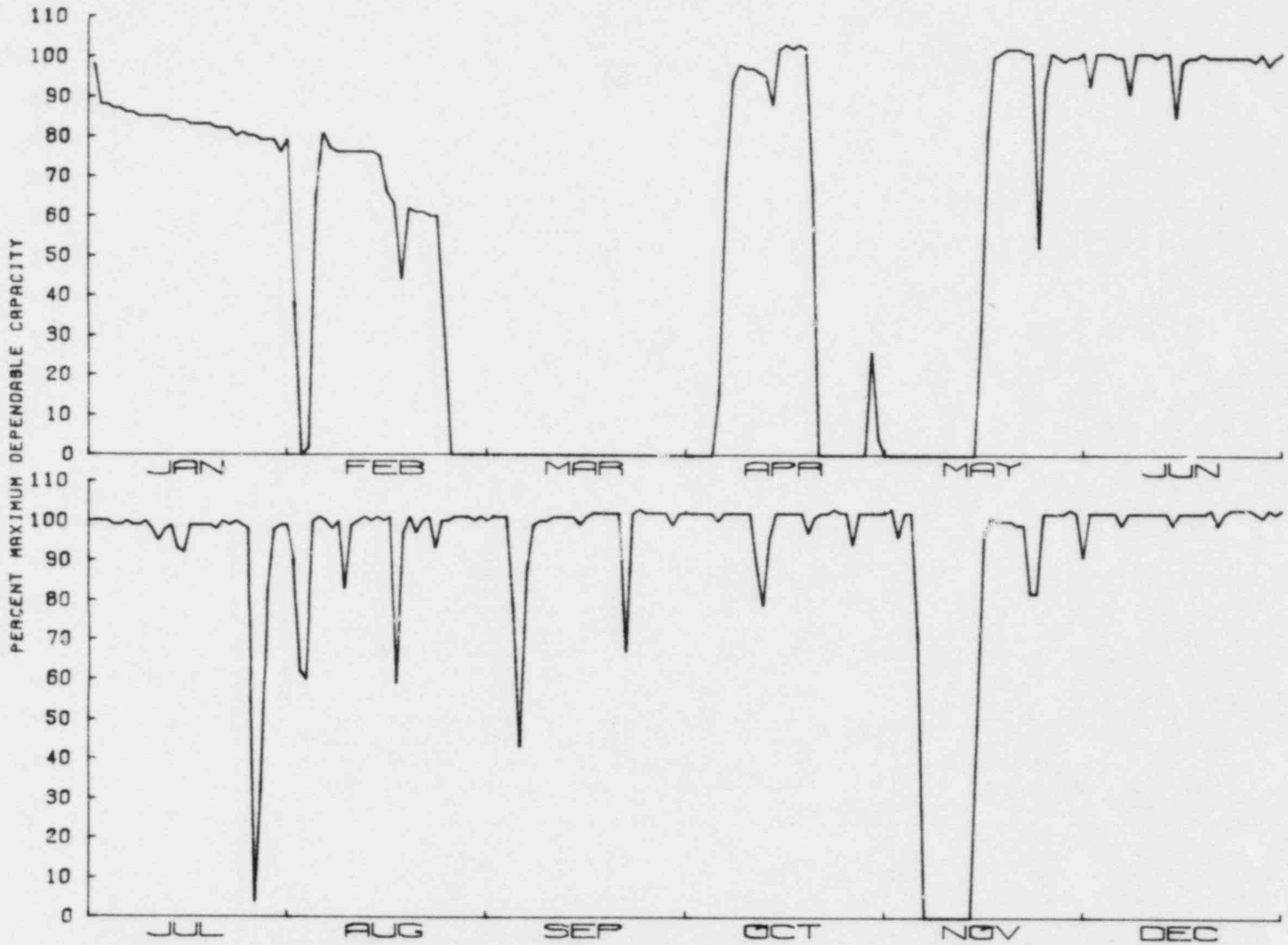
The unit was in an end-of-cycle coastdown until the cycle 7 refueling began on February 22. On April 5, the unit returned to operation and experienced three shutdowns, which accounted for nearly 25 d of downtime for repairs to recirculation pump seals. Availability for the year was 78.3%.

DETAILS OF PLANT OUTAGES FOR MONTICELLO

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/01	45.3	S	Modification on reset logic of primary containment isolation (per NUREG-0578, Item 2.1.4).	D	1	Engineered safety features (SD)	Instrumentation and controls
2	2/03	8.2	F	Scram on spurious upscale spike on intermediate range monitor channel 14; earlier trip on RPS was not fully reset.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
3	2/22	1021.0	S	Refueling.	C	1	Reactor (RC)	Fuel elements
4	4/05	3.2	S	Turbine overspeed test.	B	1	Steam and power conversion (HA)	Turbines
5a	4/19	131.2	F	High RCS level while reducing power after recirculation pump seal failure. Seals on both recirculation pumps replaced.	A	3	Reactor coolant (CB)	Pumps
5b	4/26	57.5	S	Repair recirculation pump seal.	A	4	Reactor coolant (CB)	Pumps
6	4/29	389.6	F	Repair recirculation pump seal.	A	1	Reactor coolant (CB)	Pumps
7	7/26	25.0	S	Manual and automatic scram testing of control rods per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
8	9/04	11.0	F	Failure of backwash operation valve on condensates demineralizer resulted in MFWP trip on low suction. Reactor power was reduced rapidly and one recirculation pump was tripped. During recirculation pump speed increase, a high flux scram was received.	A	3	Steam and power conversion (HG)	Valves

DETAILS OF PLANT OUTAGES FOR MONTICELLO (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9a	11/05	187.4	S	Repair MSIV actuators, replace steam chase cabling, install post-LOCA recombiner penetrations, and repair steam line drains and FW heater leaks.	B	1	System code not applicable (ZZ)	Not applicable
9b	11/13	7.5	F	Intermediate range monitor hi-hi scram received from cold water reactivity insertion after failure of low-flow FW control valve controller.	A	4	Reactor coolant (CH)	Valve operators



DESIGN ELEC. RATING = 545 MAX. DEPEND. CAP. = 536 (100%) MONTICELLO

NINE MILE POINT 1

I. Summary

Description

Location: Scriba, New York
Docket No.: 50-220
Reactor type: BWR
Maximum dependable capacity (MWe-net): 610
Commercial operation: 12/69
Years operating experience: 11.2

Performance

Net electrical energy generated (MWh): 4,537,788
Unit availability factor (%): 92.2
Unit capacity factor (using MDC): 84.7
Unit capacity factor (%) (using design MWe): 83.3

Outages

Total No.: 8
  Forced: 3
  Scheduled: 5
Total hours: 685.9 (7.8%)
  Forced: 405.3 (4.6%)
  Scheduled: 280.6 (3.2%)

II. Highlights

Nine Mile Point 1 achieved 92.2% availability during 1980. Refueling was not undertaken in 1980. The majority of outages were scheduled outages, with the longest outage lasting 14 d because of a high content of explosive gas in the main output transformer.

DETAILS OF PLANT OUTAGES FOR NINE MILE POINT 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/09	17.0	S	Core spray IV quarterly test.	B	1	Engineered safety features (SF)	Not applicable
2	3/01	31.2	S	Relief/safety valve position indicators installed.	D	1	Instrumentation and controls (ID)	Valves
3	3/03	160.5	S	Return to work started in shutdown 3/01/81 which was interrupted by Power Control Center due to grid generation shortage.	F	1	System code not applicable (ZZ)	Not applicable
4	6/28	34.9	S	Quarterly core spray IV test and condenser water box inspection.	B	1	Engineered safety features (SF)	Not applicable
5	7/12	351.7	F	Main output transformer failure (high explosive gas content).	A	1	Electric power (EB)	Transformers
6	7/26	11.0	F	Intermediate range monitor high flux scram due to mechanical pressure regulator failure.	A	3	Instrumentation and controls (ID)	Mechanical function units
7	9/19	42.6	F	Drywell high leakage from shutdown cooling isolation valve packing failure.	A	1	Reactor coolant (CF)	Valves
8	12/19	37.0	S	Installation of ATWS trip hardware and quarterly testing of core spray.	D	1	Instrumentation and controls (IA)	Instrumentation and controls

DESIGN ELEC. RATING - 520 MAX. DEPEND. CAP. - 610 (100%) NINE MILE POINT 1

NORTH ANNA 1

I. Summary

Description	Performance	Outages
Location: Mineral, Virginia	Net electrical energy generated (MWh): 5,631,557	Total No.: 19
Docket No.: 50-338	Unit availability factor (%): 86.5	Forced: 17
Reactor type: PWR	Unit capacity factor (using MDC): 75.4	Scheduled: 2
Maximum dependable capacity (MWe-net): 850	Unit capacity factor (%) (using design MWe): 70.7	Total hours: 1,189.1 (13.5%)[a]
Commercial operation: 6/06/78		Forced: 584.3 (6.7%)
Years operating experience: 2.7		Scheduled: 604.8 (6.8%)[a]

II. Highlights

Refueling outages accounted for the first 21 d and the last 3 d in 1980 at North Anna 1. The unit availability was 86.5%. Five of the seventeen forced shutdowns were attributed to operator error. The only lengthy outage occurred on May 22 when the plant shut down for nearly 13 d to repair a feedwater regulating valve.

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[a]Includes 510.9 h in 1980 from continued 9/25/79 shutdown.

DETAILS OF PLANT OUTAGES FOR NORTH ANNA 1

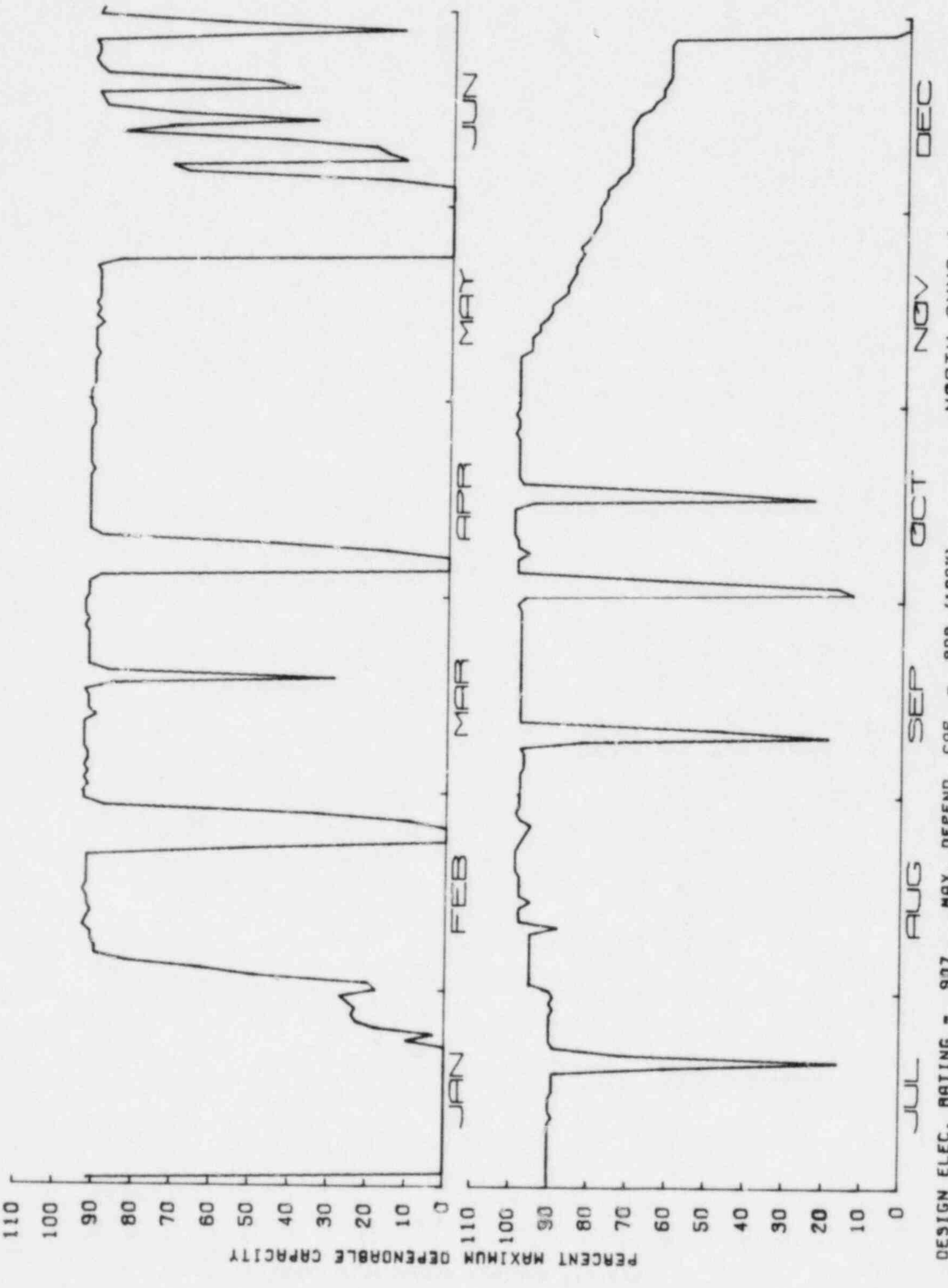
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	9/25/79 (contd.)	510.9	S	Refueling.	C	4	Reactor (RC)	Fuel elements
1	1/22	21.9	S	Turbine overspeed test and maintenance on nuclear instrumentation.	B	1	Steam and power conversion (HA)	Turbines
2	1/29	8.4	F	Loss of feedwater due to clogged strainers in condensate pump suction.	H	3	Steam and power conversion (HH)	Filters
3	2/01	4.0	F	High-high SG level due to flow oscillation while testing SG level control system.	G	3	Steam and power conversion (HH)	Instrumentation and controls
4	2/20	93.4	F	Trip on all four power range channels due to high negative flux rate; a contract laborer's clothing caught on the motor breaker for the CRD motor generator set, causing it to open and rods to drop into core.	H	3	Reactor (RB)	Not applicable
5	2/24	2.7	F	Low-low SG level during startup.	G	3	Steam and power conversion (HH)	Instrumentation and controls
6	4/03	81.0	F	Safety injection on high steam flow and low pressure. While performing a 3° stroke test on the A main steam line trip valve, the valve went fully closed (LER 80-37).	H	3	Steam and power conversion (HB)	Valves
7	4/08	4.3	F	Turbine/reactor trip due to inadvertent operation of the electrohydraulic control system low-low level alarm during investigation of the high level alarm.	G	3	Steam and power conversion (HA)	Instrumentation and controls
8	5/22	305.4	F	High-high SG level due to FW regulating valve FCV-1498 failure in open mode (LER 80-47).	A	3	Steam and power conversion (HH)	Valves

DETAILS OF PLANT OUTAGES FOR NORTH ANNA 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	6/12	8.1	F	Wrong switch operated in protection and control rack while performing periodic test.	G	3	Instrumentation and controls (IB)	Instrumentation and controls
10	6/18	6.4	F	Loose jumper in process rack No. 6.	A	3	Instrumentation and controls (IF)	Electrical conductors
11	6/26	15.9	F	Over-temperature delta T reactor trip while calibrating N43; replaced faulty card in channel II and test recorder installed (LER 80-56).	E	3	Instrumentation and controls (IA)	Instrumentation and controls
12	6/27	2.7	F	Repair broken air line to B main steam trip valve.	A	1	Steam and power conversion (HB)	Pipes, fittings
13	7/19	9.2	F	SG low level and feedwater flow/steam flow mismatch due to water in instrument line which resulted in loss of feed flow control.	A	3	Steam and power conversion (HH)	Instrumentation and controls
14	9/08	6.5	F	SG C low-low level when output breaker from inverter opened causing a loss of power to vital bus IV.	A	3	Electric power (EB)	Generators
15	9/08	0.0	F	SG C high-high level during startup.	G	3	Steam and power conversion (HH)	Instrumentation and controls
16	9/30	6.3	F	High level trip in sixth point heater due to a tube failure.	A	3	Reactor coolant (CH)	Heat exchangers
17	10/02	9.7	F	Unable to operate due to sixth point heater tube failure. Repaired failed tubes. Reactor stayed critical.	A	9	Reactor coolant (CH)	Heat exchangers

DETAILS OF PLANT OUTAGES FOR NORTH ANNA 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
18	12/28	20.3	F	High RCP leakage (LER 80-108).	A	1	Reactor coolant (CB)	Pumps
19	12/29	72.0	S	Refueling and modifications to moisture separator, oil collection, fire protection, generator breakers, and repairs to RCP seals 1A and 1C; in-service inspection of reactor vessel; eddy current testing of all SGs; and sludge lancing.	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

NORTH ANNA 2

I. Summary

Description	Performance	Outages
Location: Mineral, Virginia	Net electrical energy generated	Total No.: 14
Docket No.: 50-339	(MWh): 349,644	Forced: 11
Reactor type: PWR	Unit availability factor (%): 95.5	Scheduled: 3
Maximum dependable capacity	Unit capacity factor (using	Total hours: 1,332.5 (15.2%)
(MWe-net): 898	MDC): 90.1	Forced: 364.0 (4.1%)
Commercial operation: 12/14/80	Unit capacity factor (%) (using	Scheduled: 968.5 (11.1%)
Years operating experience: 0.3	design MWe): 89.2	

II. Highlights

North Anna 2 was granted a full power license on August 21, 1980, and was in power ascension testing until December 14. The unit averaged near full power for the remainder of December.

DETAILS OF PLANT OUTAGES FOR NORTH ANNA 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	8/26	18.1	F	Generator breaker was opened.	H	3	Steam and power conversion (HA)	Circuit closers/interrupters
2	8/27	2.4	F	Overspeed protection controller was activated during test.	H	3	Steam and power conversion (HA)	Mechanical function units
3	8/28	2.0	F	Condensate pump suction strainers clogged up causing loss of FW flow and low SG level.	A	3	Steam and power conversion (HH)	Filters
4	8/28	304.5	S	Compliance with tech spec 3.6.3.1 FW/containment penetration isolation (LER 80-51).	D	1	Engineered safety features (SD)	Valves
5	9/16	3.4	F	Loss of condenser vacuum while cleaning condensate pump suction strainer.	A	3	Steam and power conversion (HH)	Filters
6	9/26	4.0	F	MFWP suction strainers clogged.	A	3	Steam and power conversion (HH)	Filters
7a	9/27	20.6	S	50% test of reactor trip per 2-SU-26.	B	3	Instrumentation and controls (IA)	Not applicable
7b	9/28	7.6	S	No. 2 intercept right valve was indicating open and shut at the same time.	B	4	System code not applicable (ZZ)	Instrumentation and controls
8	10/19	25.2	F	2A station service transformer lock out relay actuated when a low side cable on the 2A transformer blew out.	A	3	Electric power (EB)	Electrical conductors
9	10/20	3.5	F	SG C low-level while restarting the unit.	G	3	Steam and power conversion (HH)	Instrumentation and controls
10a	10/31	2.0	S	100% load reject test.	B	3	Steam and power conversion (HA)	Not applicable

DETAILS OF PLANT OUTAGES FOR NORTH ANNA 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10b	11/01	633.8	S	10% power trip in blackout test. commenced scheduled maintenance outage.	B	4	System code not applicable (ZZ)	Not applicable
11	11/29	286.1	F	Generator leads differential.	A	3	Steam and power conversion (HA)	Generators
12	12/17	5.8	F	Generator overexcitation.	A	3	Steam and power conversion (HA)	Generators
13	12/17	2.7	F	Loose fuses in generator protection relay racks.	G	3	Steam and power conversion (HA)	Circuit closers/interrupters
14	12/31	10.8	F	Loose fuse in generator protection relay racks.	G	3	Steam and power conversion (HA)	Circuit closers/interrupters

DESIGN ELEC. RATING = 907 MAX. DEPEND. CAP. = 898 (100%)

NORTH ANNA 2

OCONEE 1

I. Summary

Description

Location: Seneca, South Carolina
Docket No.: 50-269
Reactor type: PWR
Maximum dependable capacity (MWe-net): 860
Commercial operation: 7/15/73
Years operating experience: 7.7

Performance

Net electrical energy generated (MWh): 5,116,510
Unit availability factor (%): 75.6
Unit capacity factor (using MDC): 67.7
Unit capacity factor (%) (using design MWe): 65.7

Outages

Total No.: 9
Forced: 8
Scheduled: 1
Total hours: 2,146.8 (24.4%)[a]
Forced: 852.2 (9.7%)
Scheduled: 1,294.6 (14.7%)[a]

II. Highlights

Oconee 1 began 1980 in a refueling outage and remained shut down until February 27 for pipe hanger and support inspection and modification, steam generator manway gasket replacement, feedwater chemistry limitations, and operator training. From April 17 until the June 27 shutdown, the reactor was operated at a reduced power level of ~72% because of lower motor bearing problems in a reactor coolant pump. A control rod drop on September 2 forced a reduction in power until the unit was shut down on September 11 for control rod stator replacement.

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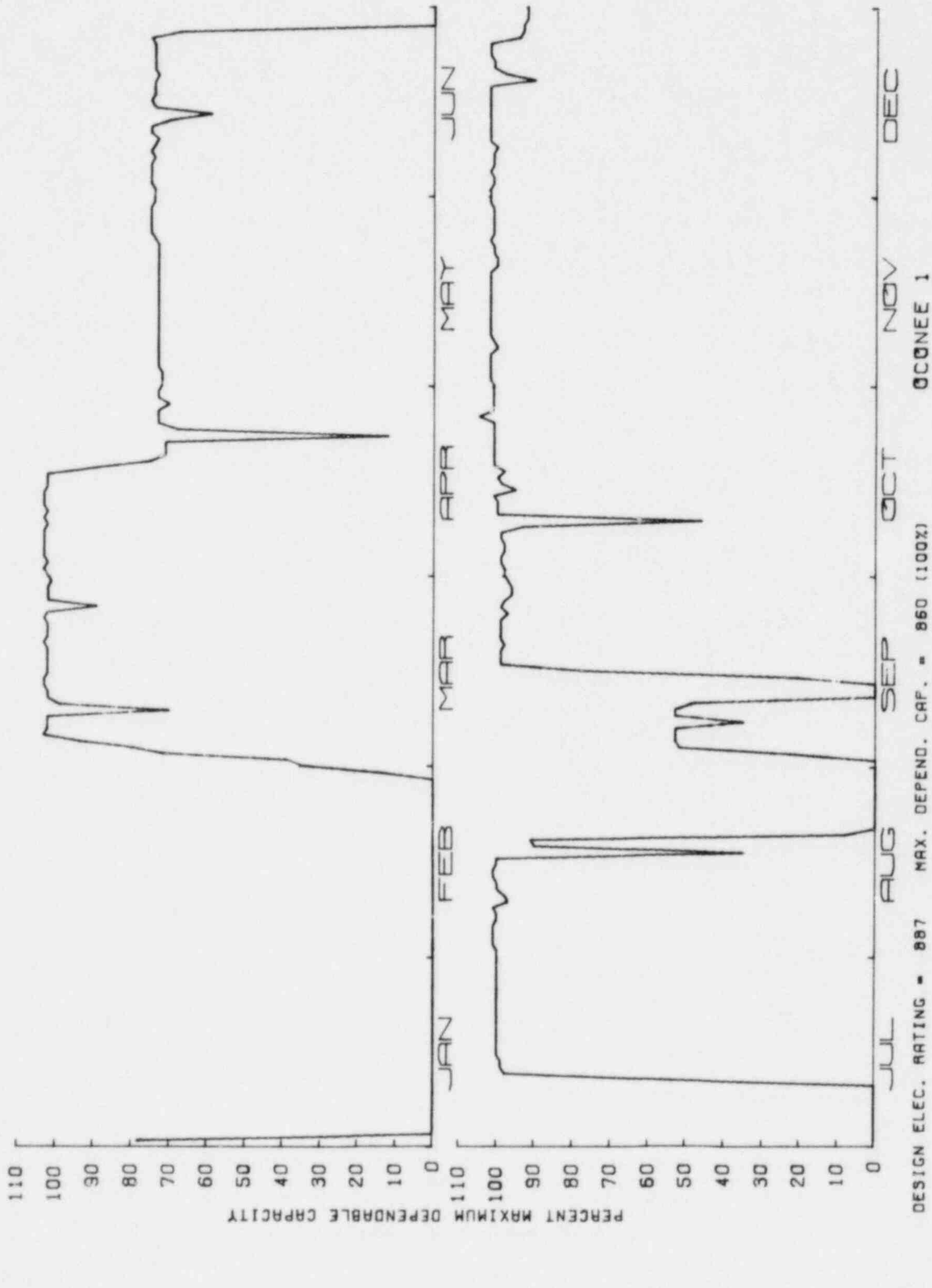
[a]Includes 374.0 h in 1980 from continued 11/21/79 shutdown.

DETAILS OF PLANT OUTAGES FOR OCONEE 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
a	11/21/79 (cont.)	374.0	S	Refueling.	C	4	Reactor (RC)	Fuel elements
b	1/16	579.2	S	Pipe hanger/support inspection and modification per IE Bulletins 79-02 79-14.	D	4	System code not applicable (ZZ)	Shock suppressors and supports
c	2/09	311.3	F	SG A manway gaskets replaced.	A	4	Steam and power conversion (HB)	Heat exchangers
d	2/22	94.5	F	Water chemistry out of limits.	B	4	Auxiliary process (PC)	Not applicable
e	226	18.2	S	Operator training.	E	4	System code not applicable (ZZ)	Not applicable
1	4/22	8.3	F	Low oil level on 1A1 RCP motor bottom bearing.	A	1	Reactor coolant (CB)	Motors
2	4/22	4.7	F	High turbine bearing vibration.	A	1	Steam and power conversion (HA)	Turbines
3	6/27	323.2	S	NRC-required modifications of emergency power supply NSM-1531. Also inspection of 1B1 lower motor bearing.	D	1	Electric power (EE)	Other
4	7/10	11.4	F	Low MFWP discharge pressure.	H	3	Steam and power conversion (HH)	Pumps
5	8/17	11.2	F	Bad cord in EHC control system caused a turbine reactor trip on low EHC oil pressure.	A	1	Steam and power conversion (HA)	Mechanical function units
6	8/20	317.5	F	Tube leaks in the 1B1 FW heater and heater could not be isolated sufficiently.	A	3	Steam and power conversion (HH)	Heat exchangers

DETAILS OF PLANT OUTAGES FOR OCONEE 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
7	9/07	3.8	F	High level in moisture separator drain tank; air line broke allowing valve 1HD-59 to fail closed.	A	3	Steam and power conversion (HC)	Valves
8	9/11	83.3	F	Control rod No. 8 group 7 stator replacement.	A	1	Reactor (RB)	Control rod drive mechanism
9	10/08	6.2	F	Temporary loss of RSVDC power supply to the turbine EHC control cabinet.	A	3	Steam and power conversion (HA)	Turbines


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

OCONEE 2

I. Summary

Description	Performance	Outages
Location: Seneca, South Carolina	Net electrical energy generated (MWh): 3,878,808	Total No.: 4
Docket No.: 50-270	Unit availability factor (%): 61.5	Forced: 2
Reactor type: PWR	Unit capacity factor (using MDC): 51.3	Scheduled: 2
Maximum dependable capacity (MWe-net): 860	Unit capacity factor (%) (using design MWe) 49.8	Total hours: 3,384.6 (38.5%)
Commercial operation: 9/09/74		Forced: 112.6 (1.3%)
Years operating experience: 7.1		Scheduled: 3,272.0 (37.2%)

II. Highlights

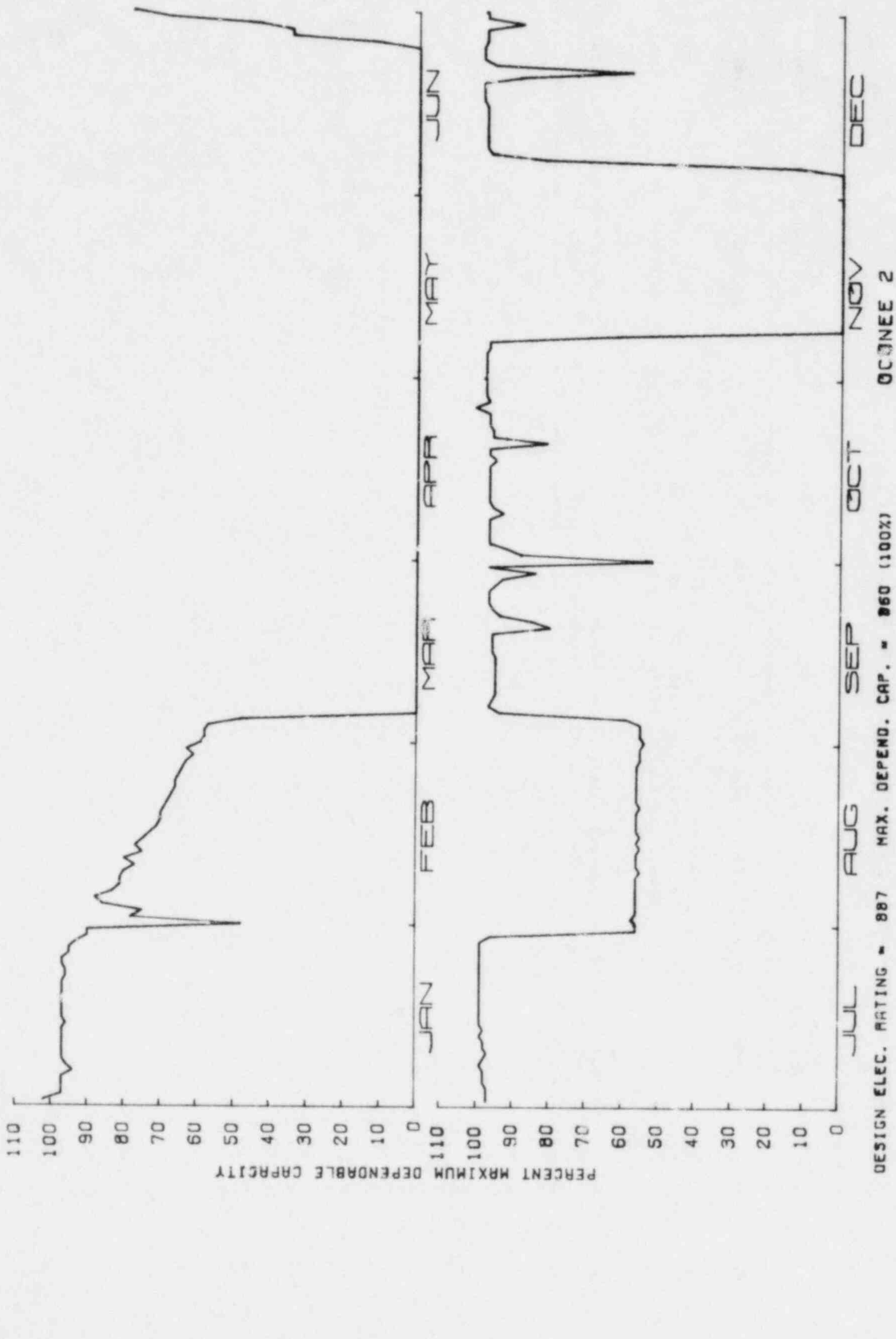
Power reductions because of fuel depletion began February 1 and continued until the refueling began on March 4. A reactor building leak rate test took over 2 weeks to complete in June. The unit returned to service June 25 and operated at or near full power until a power reduction to 59% was required on July 29 because one of the three high-pressure coolant injection pumps was out of service. The same high-pressure pump was out of service for more than 72 h again starting December 20.

DETAILS OF PLANT OUTAGES FOR OCONEE 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/30	12.3	F	Trip due to error in relay testing in the 230-kV substation.	G	3	Electric power (EA)	Relays
2a	3/04	1268.9	S	Refueling and pipe hanger/support inspection and modification.	C	1	Reactor (RC)	Fuel elements
2b	5/01	752.0	S	Pipe hanger/support inspections continue per IE Bulletin 79-02 and 79-14.	D	4	Engineered safety features (SH)	Pipes, fittings
2c	6/01	398.3	S	Reactor building leak rate test.	B	4	Engineered safety features (SD)	Penetrations primary containment
2d	6/17	30.1	F	Low boron concentration in 2A core flood tank.	D	4	Engineered safety features (SF)	Not applicable
2e	6/19	48.4	F	Leaking flange on pressurizer code relief valve ZRC-68.	A	4	Reactor coolant (CB)	Valves
2f	6/21	52.1	S	Zero power physics test.	B	4	System code not applicable (ZZ)	Not applicable
2g	6/23	6.1	S	Power escalation testing.	B	4	System code not applicable (ZZ)	Not applicable
2h	6/23	9.9	S	MSIV closure test.	B	4	Steam and power conversion (HA)	Valves
2i	6/24	5.6	S	Power escalation testing.	B	4	System code not applicable (ZZ)	Not applicable
2j	6/24	9.4	S	SG level control alternative systems testing.	B	4	Steam and power conversion (HB)	Heat exchangers

DETAILS OF PLANT OUTAGES FOR OCONEE 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
2k	6/24	18.0	F	Leaking flange on pressurizer code relief valve ZRC-68.	A	4	Reactor coolant (CB)	Valves
3	9/30	3.8	F	Loss of power to turbine EHC pumps when motor control center 2XA tripped.	F	3	Steam and power conversion (HA)	Turbines
4	11/07	669.7	S	Required modifications per NUREG-0578 and other maintenance.	D	1	System code not applicable (ZZ)	Not applicable


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 887 MAX. DEPEND. CAP. = 860 (100%)
OCONEE 2

OCONEE 3

I. Summary

Description

Location: Seneca, South Carolina
Docket No.: 50-287
Reactor type: PWR
Maximum dependable capacity (MWe-net): 860
Commercial operation: 12/16/74
Years operating experience: 6.3

Performance

Net electrical energy generated (MWh): 5,217,839
Unit availability factor (%): 73.1
Unit capacity factor (using MDC): 69.1
Unit capacity factor (%) (using design MWe): 67.0

Outages

Total No.: 10
Forced: 8
Scheduled: 2
Total hours: 2,366.5 (26.9%)
Forced: 765.4 (8.7%)
Scheduled: 1,601.1 (18.2%)

II. Highlights

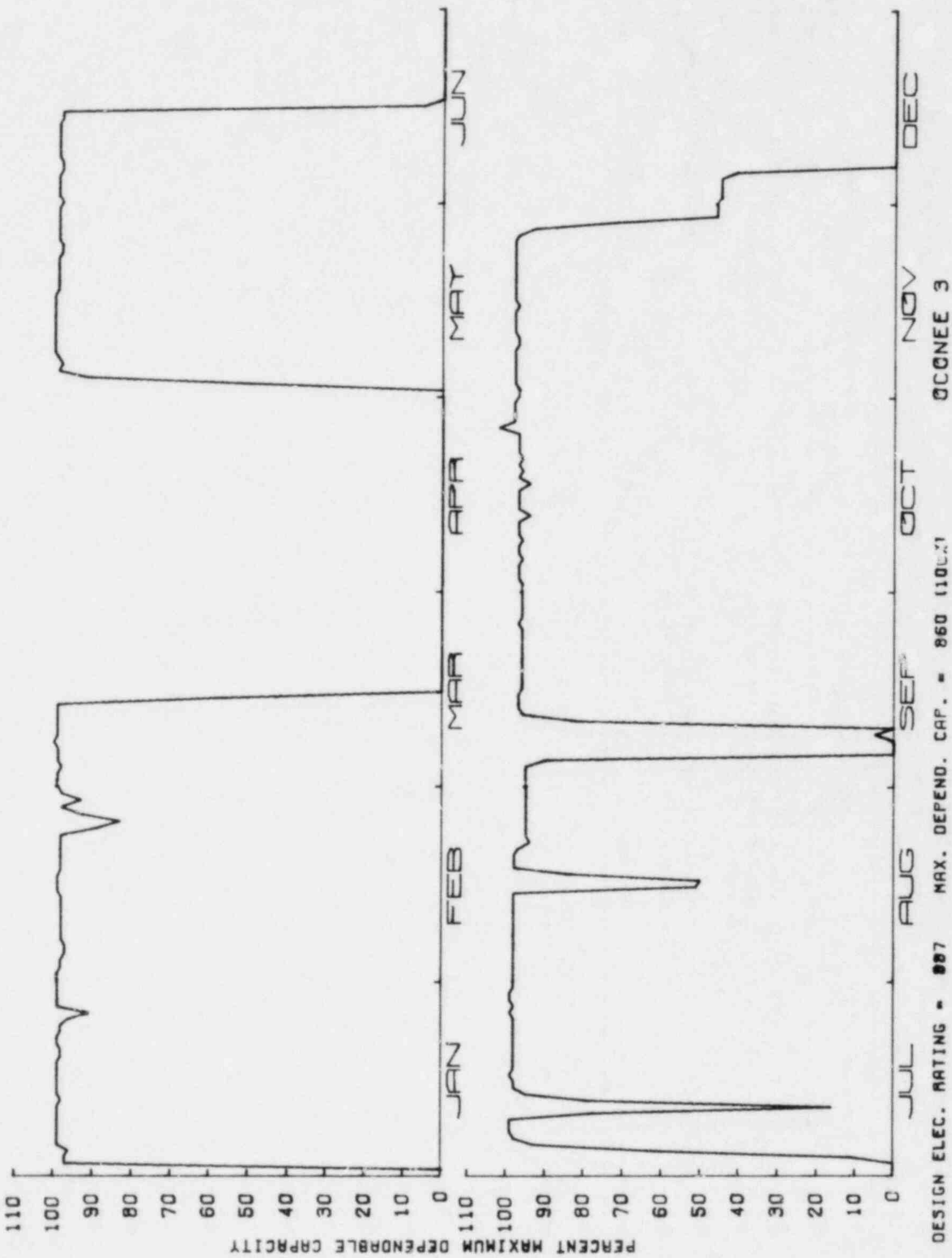
NRC-required modifications took nearly 6 weeks in March and April to complete. A steam generator tube leak required 2.5 weeks to repair in June. Power was at or near full power with the exception of a few short shutdowns when power was reduced on November 27 to extend core life until the refueling outage began December 5.

DETAILS OF PLANT OUTAGES FOR OCONEE 3

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	3/14	9.9	F	Spurious turbine/generator trip.	A	3	Steam and power conversion (HA)	Turbines
2a	3/15	976.0	S	NRC-required modifications.	D	4	System code not applicable (ZZ)	Not applicable
2b	4/24	89.3	F	Leak in pressurizer relief valve flange.	A	4	Reactor coolant (CB)	Valves
2c	4/28	52.4	F	Failed controller on decay heat removal cooler outlet valve.	A	4	Reactor coolant (CF)	Valve operators
2d	4/30	33.2	F	Failure to remove generator bus ground straps before closing the generator field breaker required investigation of possible bus damages.	G	4	Steam and power conversion (HA)	Generators
3	6/15	437.2	F	Tube leak in SG A (LER 80-10).	A	1	Steam and power conversion (HB)	Heat exchangers
4	7/03	3.8	F	Feedwater transient while not on line.	H	3	Steam and power conversion (HH)	Not applicable
5	7/10	5.3	F	High level indication on MSRH's.	A	3	Steam and power conversion (HB)	Instrumentation and controls
6	7/11	11.7	F	Power transient—flux flow imbalance.	H	3	Reactor (RB)	Fuel elements
7a	8/15	6.5	F	Loss of power to turbine EHC pumps when MCC-3XA tripped.	A	3	Electric power (EB)	Circuit closers/ interrupters
7b	8/15	5.6	F	Reactor quadrant power tilt caused a hold in startup.	H	4	Reactor (RC)	Control rods

DETAILS OF PLANT OUTAGES FOR OCONEE 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8	9/05	77.7	F	Repairs to RPS channels A and D precluded necessary test of other channel.	A	1	Instrumentation and controls (IB)	Instrumentation and controls
9	9/08	32.8	F	Change oil in 3A1 and 3A2 RCPs to replace wrong oil recently added.	G	1	Reactor coolant (CB)	Pumps
10	12/05	625.1	S	Refueling.	C	1	Reactor (RC)	Fuel elements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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
Maximum dependable capacity (MWe-net): 620
Commercial operation: 12/23/69
Years operating experience: 11.3

Performance

Net electrical energy generated (MWh): 1,957,645
Unit availability factor (%): 41.7
Unit capacity factor (using MDC): 35.9
Unit capacity factor (%) (using design MWe): 34.3

Outages

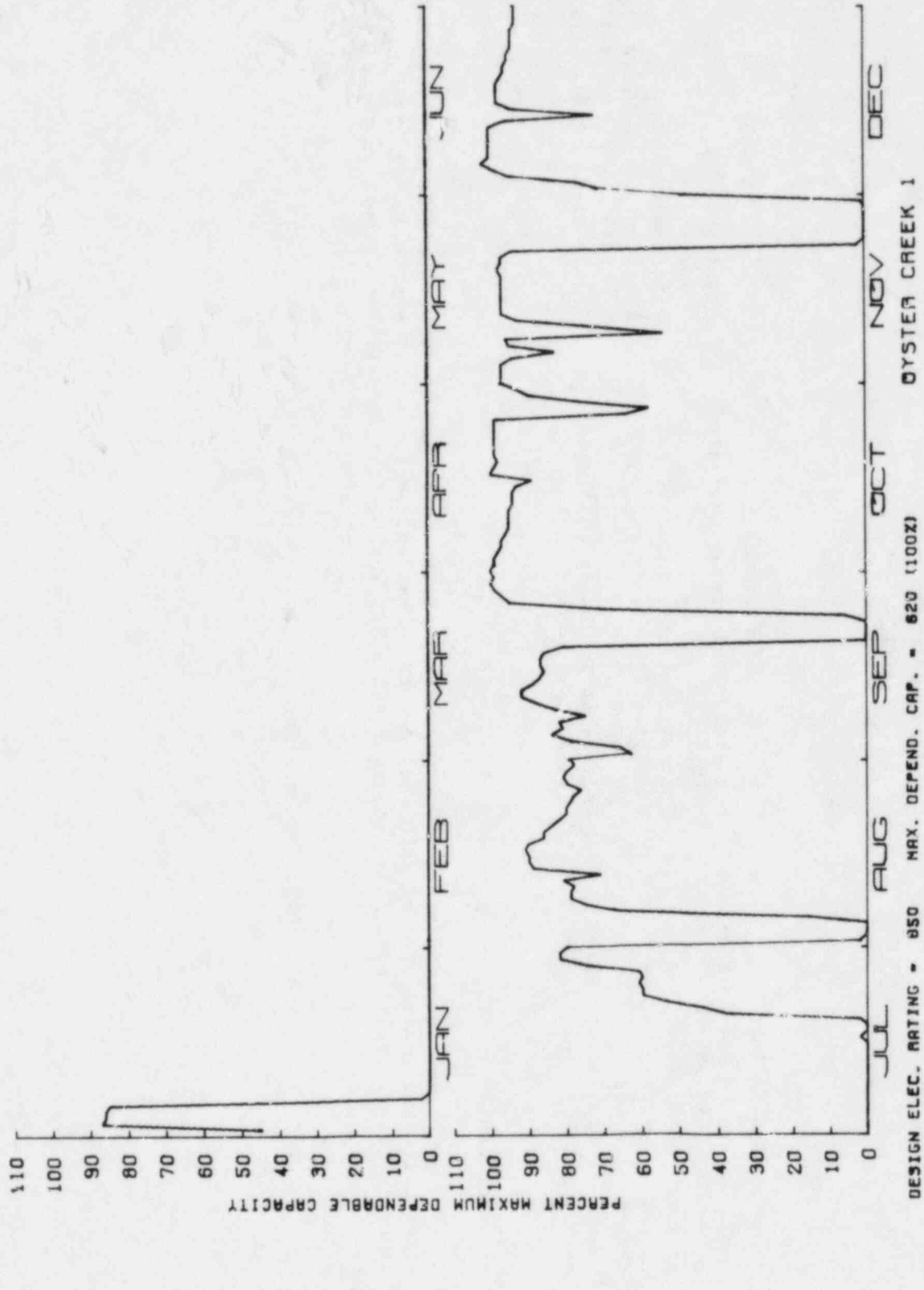
Total No.: 5
 Forced: 2
 Scheduled: 3
Total hours: 5,120.4 (58.3%)
 Forced: 300.7 (3.4%)
 Scheduled: 4,819.7 (54.9%)

II. Highlights

An extended refueling outage began January 5 and lasted until July 16. Only three shutdowns occurred after the outage for scram testing which began on July 16.

DETAILS OF PLANT OUTAGES FOR OYSTER CREEK

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/05	4641.0	S	Refueling and maintenance.	C	1	Reactor (RC)	Fuel elements
2	7/16	73.7	S	Scram testing of scram discharge volume per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
3	8/01	105.0	S	Leak in the nitrogen supply to a MSIV accumulator located and repaired.	B	1	Reactor coolant (CD)	Accumulators
4	9/18	116.0	F	Increasing drywell leak rate.	B	1	System code not applicable (ZZ)	Not applicable
5	11/21	184.7	F	Plugged 27 leaking tubes in 1C3 HP FW heater and repaired seal weld on feed check valve hinge pin.	B	1	Reactor coolant (CH)	Heat exchangers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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
Maximum dependable capacity (MWe-net): 635
Commercial operation: 12/71
Years operating experience: 9.0

Performance

Net electrical energy generated (MWh): 2,379,529
Unit availability factor (%): 42.9
Unit capacity factor (using MDC): 42.7
Unit capacity factor (%) (using design MWe): 33.7

Outages

Total No.: 8
Forced: 6
Scheduled: 2
Total hours: 5,016.8 (57.1%)[a]
Forced: 467.8 (5.3%)
Scheduled: 4,549.0 (51.8%)[a]

II. Highlights

Palisades was in a refueling and modification outage until May 24 for inspection and repair of anchor bolts and performance of TMI-related work. This outage lasted 8.5 months during 1979 and 1980. Operations were routine with four outages until a six-week outage for turbine inspection began October 31.

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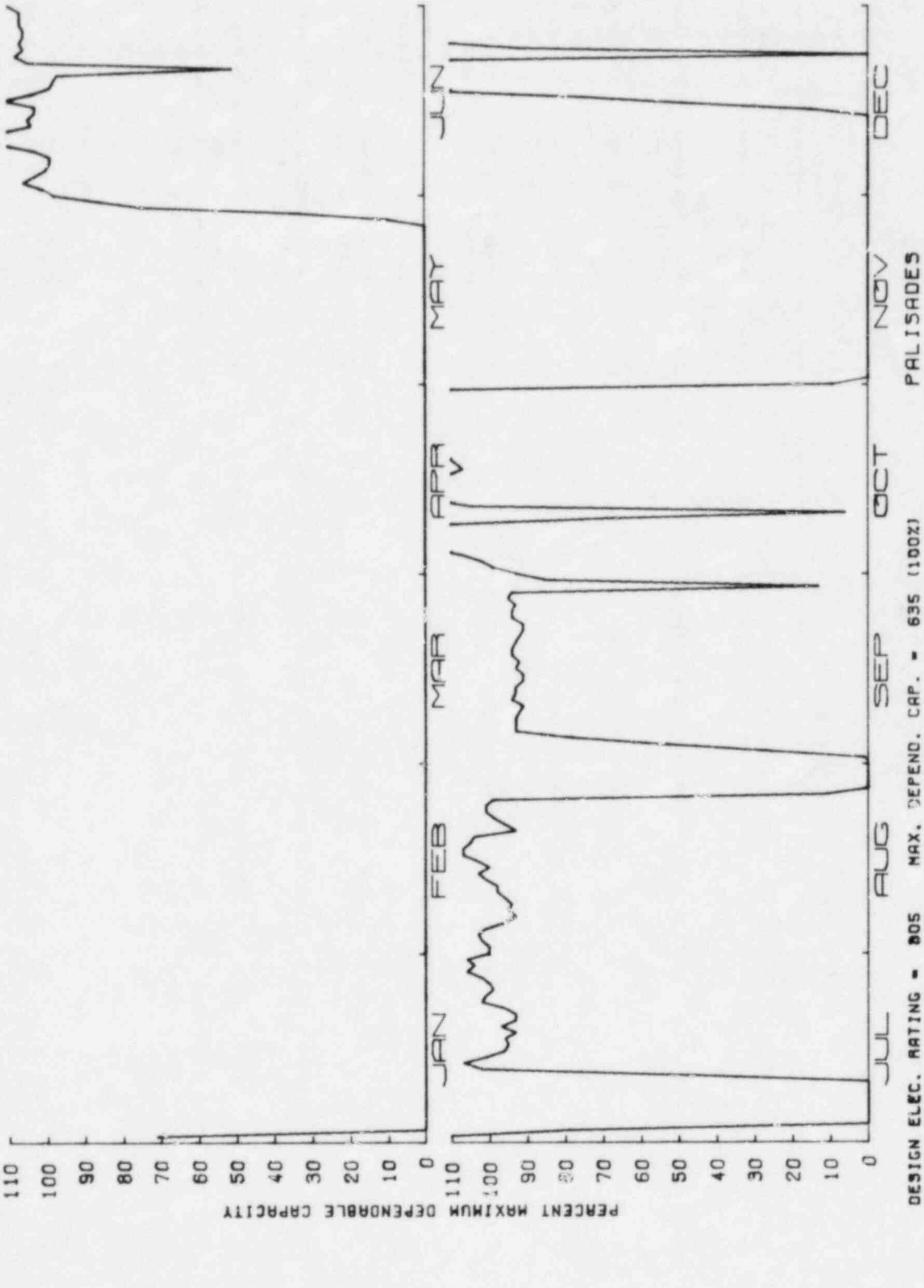
[a]Includes 3,543.5 h in 1980 from continued 9/08/79 shutdown.

DETAILS OF PLANT OUTAGES FOR PALISADES

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	9/08/79 (cont.)	3543.5	S	TMI modifications and hanger/anchor bolt inspections per IE Bulletin 79-02.	D	4	Engineered safety features (SH)	Shock suppressors and supports
1	5/28	0.9	S	Turbine overspeed trip test.	B	1	Steam and power conversion (HA)	Turbines
2	7/02	204.5	F	Oil leak on generator seal oil system.	A	2	Electric power (EB)	Generators
3	8/26	162.5	F	Condensate pump trip.	A	1	Steam and power conversion (HH)	Pumps
4	9/28	16.8	F	Short on power supply to turbine circuitry.	A	1	Steam and power conversion (HA)	Electrical conductors
5	10/09	27.3	F	Severed cables in switchyard.	A	3	Electric power (EB)	Electrical conductors
6	10/31	1004.6	S	Turbine inspection.	B	1	Steam and power conversion (HA)	Turbines
7	12/12	35.0	F	Outage occurred following restart.	H	9	System code not applicable (ZZ)	Not applicable
8	12/23	21.7	F	Reheat intercept valves closed.	A	3	Steam and power conversion (HB)	Valves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DESIGN ELEC. RATING = 805 MAX. DEPEND. CAP. = 635 (100%)
PALISADES

PEACH BOTTOM 2

I. Summary

Description

Location: Peach Bottom, Pennsylvania
Docket No.: 50-277
Reactor type: BWR
Maximum dependable capacity (MWe-net): 1,051
Commercial operation: 7/05/74
Years operating experience: 6.9

Performance

Net electrical energy generated (MWh): 4,343,879
Unit availability factor (%): 51.6
Unit capacity factor (using MDC): 47.1
Unit capacity factor (%) (using design MWe): 46.4

Outages

Total No.: 12
Forced: 6
Scheduled: 6
Total hours: 4,254.0 (48.4%)[a]
Forced: 186.6 (2.1%)
Scheduled: 4,067.4 (46.3%)[a]

II. Highlights

The unit began a power coastdown at the end of January for the March 21 refueling. Refueling and NRC-required scram testing were completed August 17. Operation was routine for the remainder of the year with a few short shutdowns and two power reductions (September 5 and November 22) for control rod pattern adjustments and sequence changes.

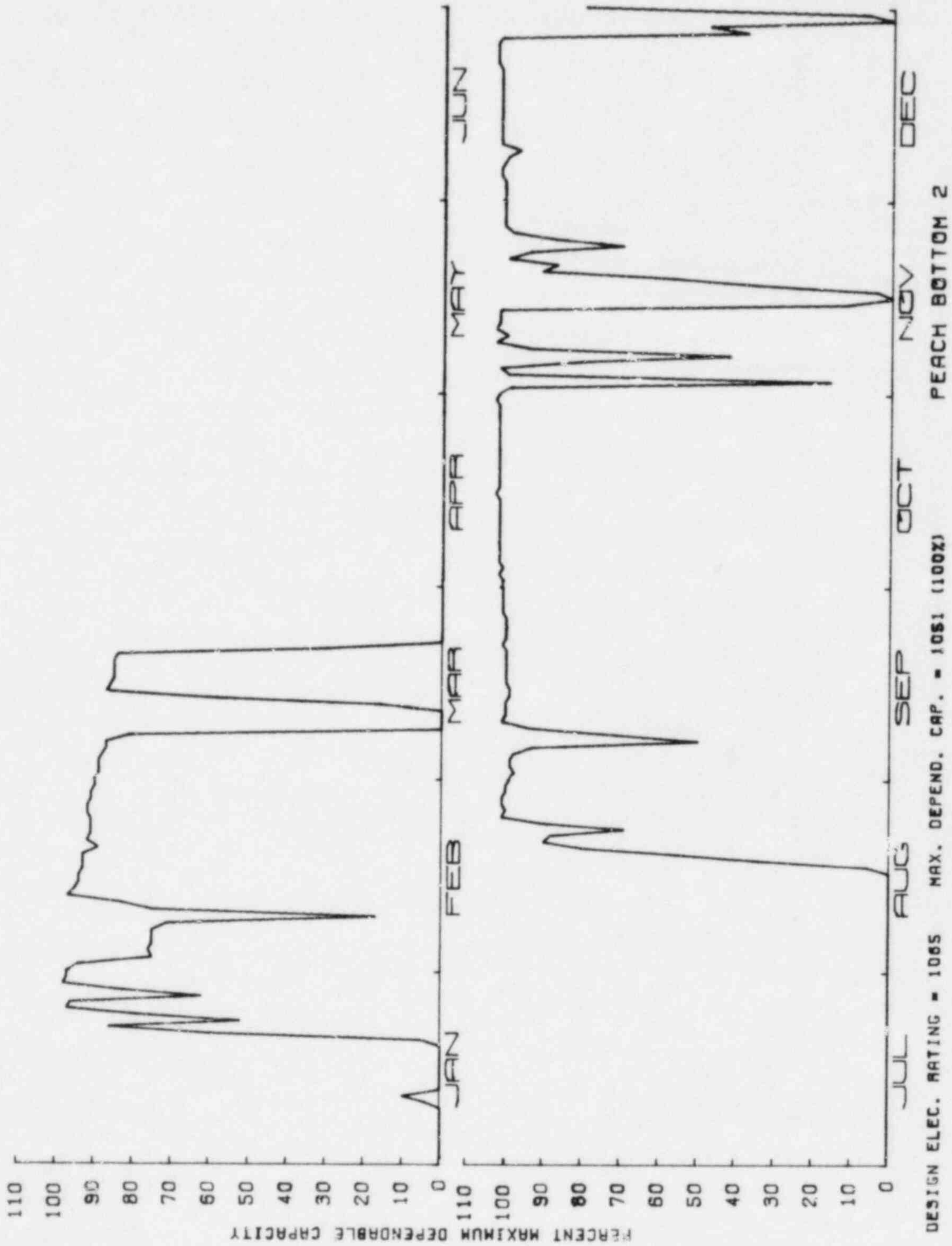
[a]Includes 200.7 h in 1980 from continued 12/31/79 shutdown.

DETAILS OF PLANT OUTAGES FOR PEACH BOTTOM 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/31/79 (cont.)	200.7	S	Core spray valve test at full flow.	B	4	Reactor coolant (CJ)	Valves
1	1/09	12.2	F	Low reactor level scram due to reactor full power trip mechanism malfunction.	A	3	Reactor coolant (CH)	Instrumentation and controls
2	1/10	217.1	S	Check-valve leaks in automatic depressurization system air supply (LER 80-02).	B	1	Engineered safety features (SF)	Valves
3	2/07	5.8	S	Turbine stop valve repair.	B	1	Steam and power conversion (HA)	Valves
4	3/08	104.1	S	Repair of recirculation pump seal and minimum flow valve of RHR.	B	2	Reactor coolant (CB)	Pumps
5	3/21	3482.7	S	Refueling.	C	2	Reactor (RC)	Fuel elements
6a	8/13	25.2	S	Manual scram test of scram discharge volume per IE Bulletin 80-17.	D	2	Instrumentation and controls (IC)	Accumulators
6b	8/14	21.8	S	Automatic scram test of scram discharge volume per IE Bulletin 80-17.	D	3	Instrumentation and controls (IC)	Accumulators
6c	8/15	44.1	F	Recombiner loop seal modification.	A	4	Steam and power conversion (HA)	Turbines
7	11/05	12.6	F	EHC control valve leak.	A	1	Steam and power conversion (HA)	Valves
8	11/14	48.3	F	FW leak at HPCI testable check valve.	A	3	Engineered safety features (SD)	Valves
9	11/17	26.4	F	APRM high-flux scram due to 2A recirculation flow control system failure.	A	3	Instrumentation and controls (IA)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR PEACH BOTTOM 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10	12/27	10.0	S	EHC oil leaks on Nos. 1 and 2 turbine control valves.	A	1	Steam and power conversion (HA)	Valves
11	12/28	40.7	F	High level in D moisture separator drain tank.	A	3	Reactor coolant (CC)	Valves
12	12/30	2.3	F	Insufficient condenser vacuum due to outage of 2B and 2C circulating water pumps for breaker calibration.	A	1	Reactor coolant (CH)	Other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DESIGN ELEC. RATING = 1065   MAX. DEPEND. CAP. = 1051 (100%)
PEACH BOTTOM 2

PEACH BOTTOM 3

I. Summary

Description

Location: Peach Bottom, Pennsylvania
Docket No.: 50-278
Reactor type: BWR
Maximum dependable capacity (MWe-net): 1.035
Commercial operation: 12/23/74
Years operating experience: 6.3

Performance

Net electrical energy generated (MWh): 7,233,843
Unit availability factor (%): 80.7
Unit capacity factor (using MDC): 79.6
Unit capacity factor (%) (using design MWe): 77.3

Outages

Total No.: 14
Forced: 7
Scheduled: 7
Total hours: 1,691.0 (19.3%)
Forced: 874.7 (10.0%)
Scheduled: 816.3 (9.3%)

II. Highlights

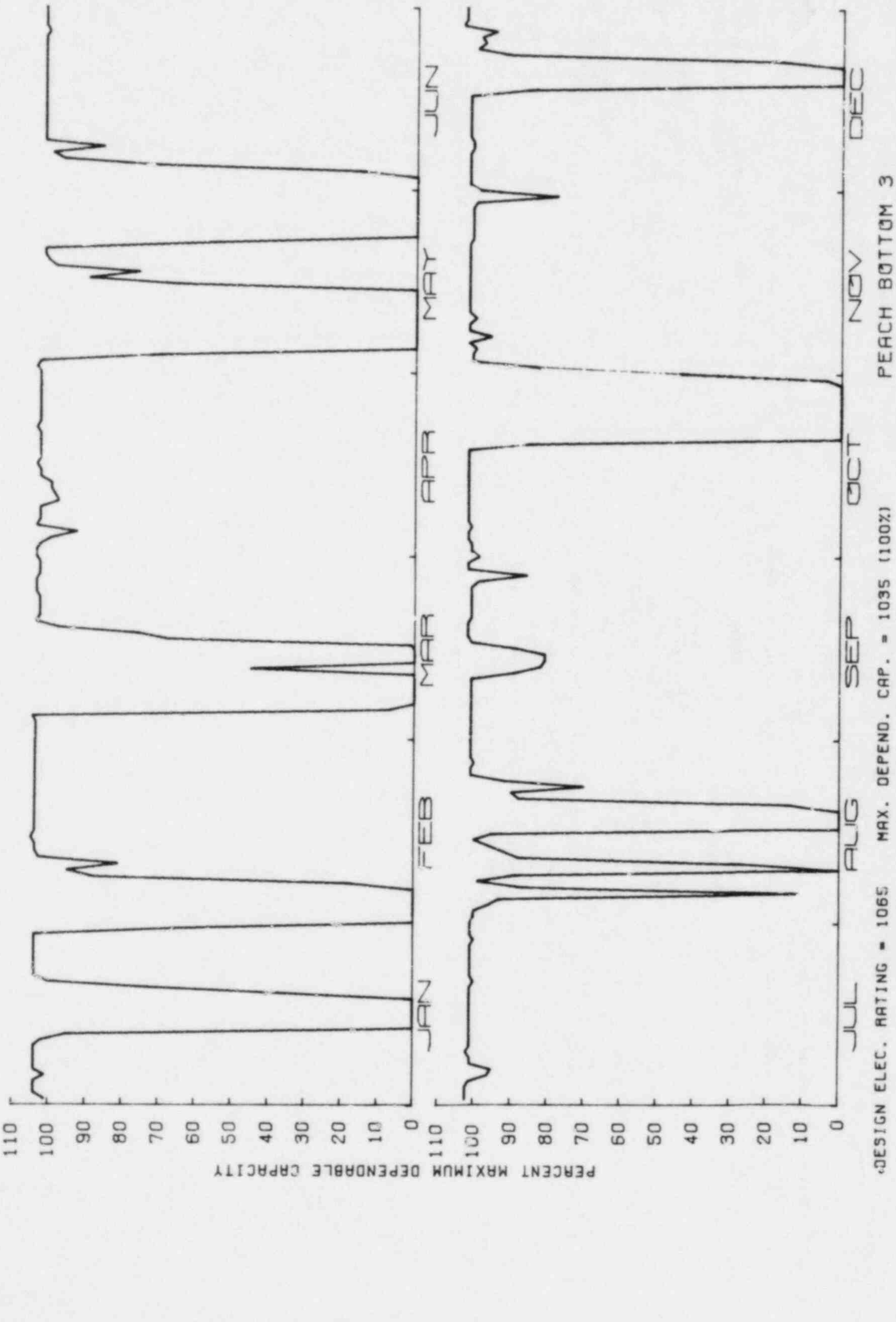
Peach Bottom 3 achieved 80.7% availability and 79.6 and 77.3% MDC and DER capacity factors, respectively. No refueling was performed in 1980. Load reductions for rod pattern adjustments occurred on February 2, March 17, May 17, June 16, and December 27. The 3B reactor coolant pump seal was replaced on January 29, May 3, and again on May 22, accounting for 31 d (44%) of the total 1980 downtime.

DETAILS OF PLANT OUTAGES FOR PEACH BOTTOM 3

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/12	148.5	S	Check valve leaks in automatic depressurization system air supply. Tube repairs to 2B FW heater.	B	1	Engineered safety features (SF)	Valves
2	1/29	189.7	S	RCP 3B seal replacement.	B	3	Reactor coolant (CB)	Pumps
3	3/05	169.6	F	MSIVs drift closed after a loss of offsite power.	A	3	Electric power (EA)	Circuit closure/ interrupters
4	3/13	91.7	S	FW leak from RCIC injection check valve.	B	2	Steam and power conversion (HA)	Valves
5	5/03	265.2	F	RCP 3B seal replacement.	A	1	Reactor coolant (CB)	Pumps
6	5/22	285.9	F	RCP 3B seal replacement.	A	1	Reactor coolant (CB)	Pumps
7	8/04	17.5	F	Low EHC pressure during stop valve testing.	B	3	Steam and power conversion (HA)	Mechanical function merits
8	8/08	15.8	S	Manual scram test of scram discharge volume per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
9	8/09	10.2	S	Automatic scram test of scram discharge volume per IE Bulletin 80-17.	D	3	Reactor (RB)	Control rods
10	8/10	9.6	F	APRM high flux.	H	3	Reactor (RB)	Instrumentation and controls
11	8/15	113.1	S	Repair of valve operator drive sleeve in LPCI injection valve (LER 80-19).	B	3	Reactor coolant (CB)	Valves
12	10/19	247.3	S	Replace transformer 3A.	B	1	[illegible]	[illegible]

DETAILS OF PLANT OUTAGES FOR PEACH BOTTOM 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
13a	10/30	13.6	F	Generator power load unbalance projection.	G	3	Steam and power conversion (HA)	Instrumentation and controls
13b	10/31	0.5	F	Failure to completely remove blocking of electric power instrument associated with outage 10/19/80.	B	9	Steam and power conversion (HA)	Instrumentation and controls
14	12/17	112.8	F	Reactor feed pump trip caused by pressure transient in condensate system. Outage was extended to repair scram discharge volume continuous water level monitoring system.	A	3	Reactor coolant (CH)	Other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•DESIGN ELEC. RATING = 1065 MAX. DEPEND. CAP. = 1035 (100%)
PEACH BOTTOM 3

PILGRIM 1

I. Summary

Description

Location: Plymouth, Massachusetts
Docket No.: 50-293
Reactor type: BWR
Maximum dependable capacity (MWe-net): 670
Commercial operation: 12/72
Years operating experience: 8.5

Performance

Net electrical energy generated (MWh): 3,044,484
Unit availability factor (%): 564
Unit capacity factor (using MDC): 51.7
Unit capacity factor (%) (using design MWe): 52.9

Outages

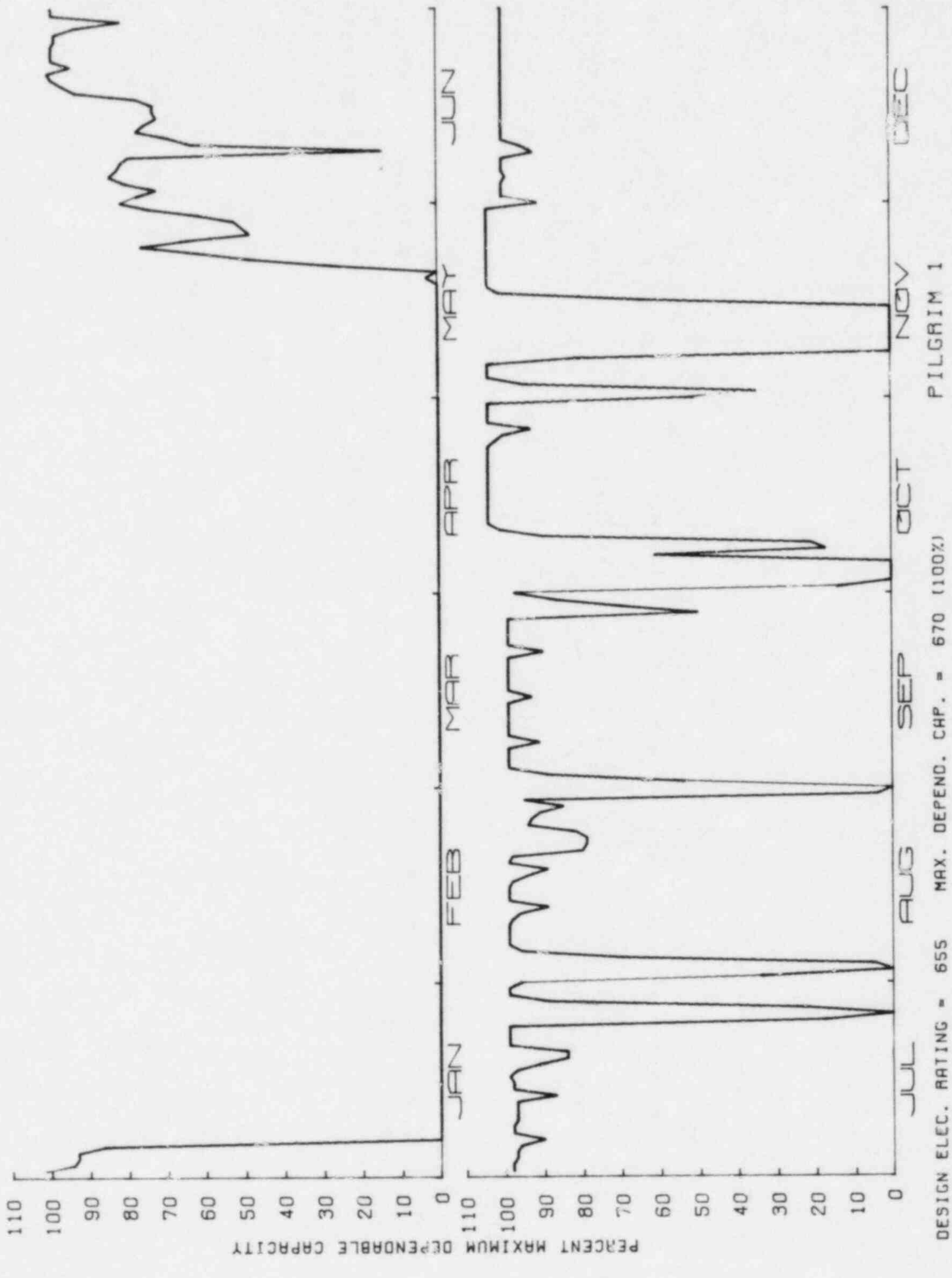
Total No.: 10
Forced: 7
Scheduled: 3
Total hours: 3,829.6 (43.6%)
Forced: 469.6 (5.3%)
Scheduled: 3,360.0 (38.3%)

II. Highlights

Refueling lasted from January 5 until May 19. Otherwise, operation was routine, with nine brief outages for equipment failures and NRC-required scram testing. High nitrogen pressure caused a safety relief valve to close twice in October resulting in shutdowns.

DETAILS OF PLANT OUTAGES FOR PILGRIM 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/05	3256.6	S	Refueling.	C	2	Reactor (RC)	Fuel elements
2	5/19	26.1	F	Reactor scram on low RCS level due to erratic operation of turbine speed control (LER 80-26).	A	3	Steam and power conversion (HA)	Mechanical function units
3	6/08	17.7	F	Low vacuum during condenser backwash.	H	3	Steam and power conversion (HC)	Heat exchangers (condenser)
4	7/25	53.7	S	Scram to test scram discharge volume per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
5	8/01	57.4	F	High conductivity in reactor water due to loose lateral in demineralizer (LER 80-43).	A	2	Steam and power conversion (HG)	Demineralizers
6	8/30	49.7	S	Repair tube leak in 4th point heater.	A	2	Reactor coolant (CH)	Heat exchangers
7	10/01	118.6	F	Air bubble in FW system caused high radiation trip. Monitor trip was set too low.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
8	10/07	33.3	F	High nitrogen pressure to SRV caused SRV to lift. Pressure reduced (LER 80-69).	A	1	Engineered safety features (SF)	Valves
9	10/31	20.8	F	High nitrogen pressure to SRV caused SRV to lift. Pressure reduced (LER 80-80).	A	1	Engineered safety features (SF)	Valves
10	11/06	195.7	F	B recirculation pump discharge MOV-202-5B packing leak.	A	1	Reactor coolant (CB)	Valves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DESIGN ELEC. RATING = 655 MAX. DEPEND. CAP. = 670 (100%)
PILGRIM 1

POINT BEACH 1

I. Summary

Description

Location: Two Creeks, Wisconsin
Docket No.: 50-266
Reactor type: PWR
Maximum dependable capacity (MWe-net): 495
Commercial operation: 12/21/70
Years operating experience: 10.2

Performance

Net electrical energy generated (MWh): 2,477,108
Unit availability factor (%): 78.6
Unit capacity factor (using MDC): 57.0
Unit capacity factor (%) (using design MWe): 56.7

Outages

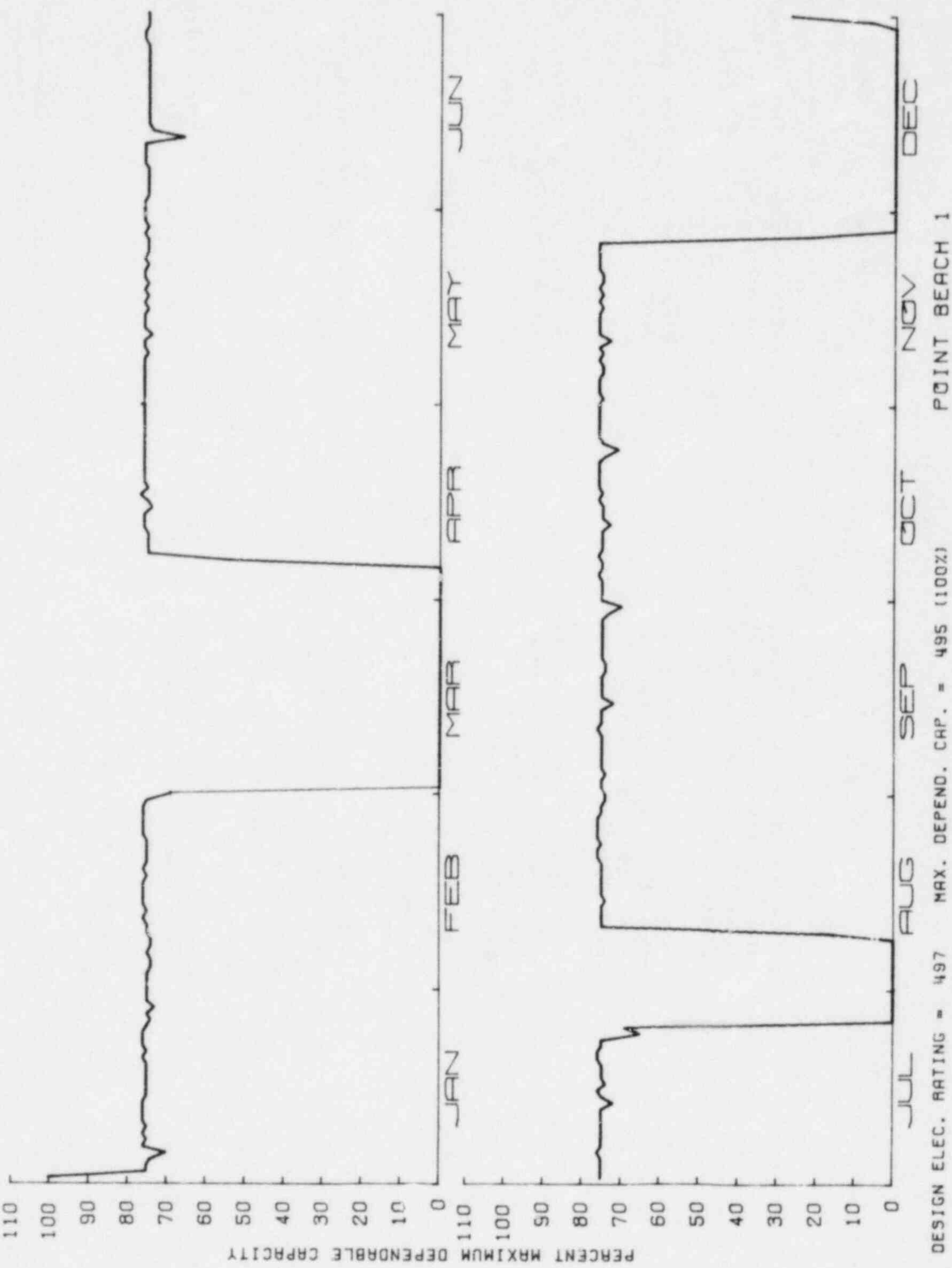
Total No.: 6
Forced: 4
Scheduled: 2
Total hours: 2,041.1 (23.2%)
Forced: 42.3 (0.5%)
Scheduled: 1,998.8 (22.7%)

II. Highlights

Steam generator problems were investigated and repairs made during the outages that began on February 28 and July 26 which accounted for over 7 weeks of downtime, or 97% of the downtime excluding the November 29 refueling outage. Power was restricted to 79% in January and February and 76% after the unit returned on-line on April 6 to reduce steam generator tube corrosion. Refueling was completed December 30.

DETAILS OF PLANT OUTAGES FOR POINT BEACH 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/28	891.5	S	SG tube leak checks, eddy current inspection, explosive tube plugging, tube pulling, weld repair, and hydrostatic leak testing.	D	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
2	6/11	2.2	F	Steam flow/feedwater flow mismatch initiated by removal of vital bus supply inverter from service and a communications mixup (LER 80-07).	G	3	Electric power (EC)	Generators
3	7/24	1.4	F	High differential pressure across circulating water screens caused by buildup of all wires that entered the intake crib.	H	1	Steam and power conversion (HF)	Filters
4	7/26	349.7	S	As requested by NRC confirmatory order of 11/30/79, the unit was taken off line for 90 day steam generator testing (LER 80-09).	D	1	Steam and power conversion (HB)	Heat exchangers
5	11/26	6.6	F	Low SG level and steam flow/feed flow mismatch due to closure of the main feed regulator valve when its air line broke due to excessive vibrations in FW system.	A	3	Steam and power conversion (HH)	Pipes, fittings
6a	11/26	32.1	F	Low SG level and steam flow/feed flow mismatch due to closure of the main feed regulator valve when its air line broke due to excessive vibrations in FW system.	A	3	Steam and power conversion (HH)	Pipes, fittings
6b	11/29	757.6	S	Refueling.	C	4	Reactor (RC)	Fuel elements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DESIGN ELEC. RATING = 497
MAX. DEPEND. CAP. = 495 (100%)
POINT BEACH 1

POINT BEACH 2

I. Summary

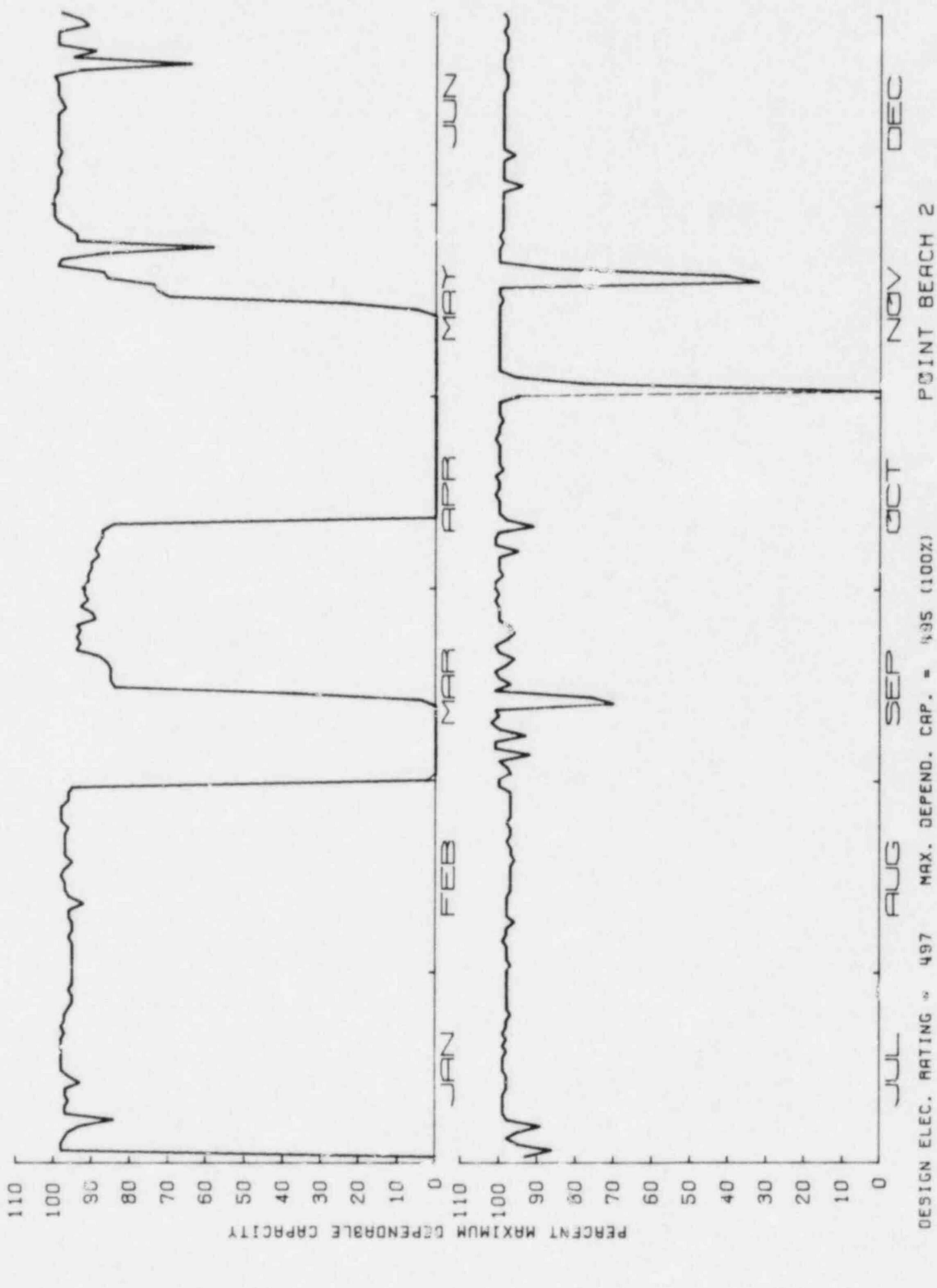
Description	Performance	Outages
Location: Two Creeks, Wisconsin	Net electrical energy generated (MWh): 3,588,294	Total No.: 6
Docket No.: 50-301	Unit availability factor (%): 86.4	Forced: 3
Reactor type: PWR	Unit capacity factor (using MDC): 82.5	Scheduled: 3
Maximum dependable capacity (MWe-net): 495	Unit capacity factor (%) (using design MWe): 82.2	Total hours: 870.8 (9.9%)
Commercial operation: 10/01/72		Forced: 371.9 (4.2%)
Years operating experience: 8.4		Scheduled: 498.9 (5.7%)

II. Highlights

After the outages for steam generator tube testing and repairs starting on February 28 and for refueling starting on April 11, Point Beach 2 operated routinely for the remainder of the year except for a few brief shutdowns. Unit availability was 86.4%.

DETAILS OF PLANT OUTAGES FOR POINT BEACH 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/28	351.6	F	SG tube leakage of 1428 gal/d required shutdown for eddy current testing and plugging of leaking and defective tubes (LER 80-02).	A	1	Steam and power conversion (HB)	Heat exchangers
2	4/11	470.4	S	Refueling and inspection of turbine rotors for crack indications.	C	1	Reactor (RC)	Fuel elements
3	5/23	2.6	S	Turbine overspeed trip testing.	B	1	Steam and power conversion (HA)	Turbines
4	9/12	10.2	F	Failed CRDM power supply caused a group of rods to drop.	A	3	Instrumentation and controls (IF)	Instrumentation and controls
5	11/01	25.9	S	Numerous secondary system repairs.	B	1	Steam and power conversion (HB)	Other
6	11/18	10.1	F	SI pump 2P15A found to have broken keys holding the coupling of both the motor and the pump (LER 80-10).	A	1	Engineered safety features (SF)	Pumps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POINT BEACH 2

PRAIRIE ISLAND 1

I. Summary

Description

Location: Goodhue, Minnesota
Docket No.: 50-282
Reactor type: PWR
Maximum dependable capacity (MWe-net): 503
Commercial operation: 12/16/73
Years operating experience: 7.1

Performance

Net electrical energy generated (MWh): 3,106,355
Unit availability factor (%): 78.2
Unit capacity factor (using MDC): 70.3
Unit capacity factor (%) (using design MWe): 66.7

Outages

Total No.: 5
- Forced: 2
- Scheduled: 3

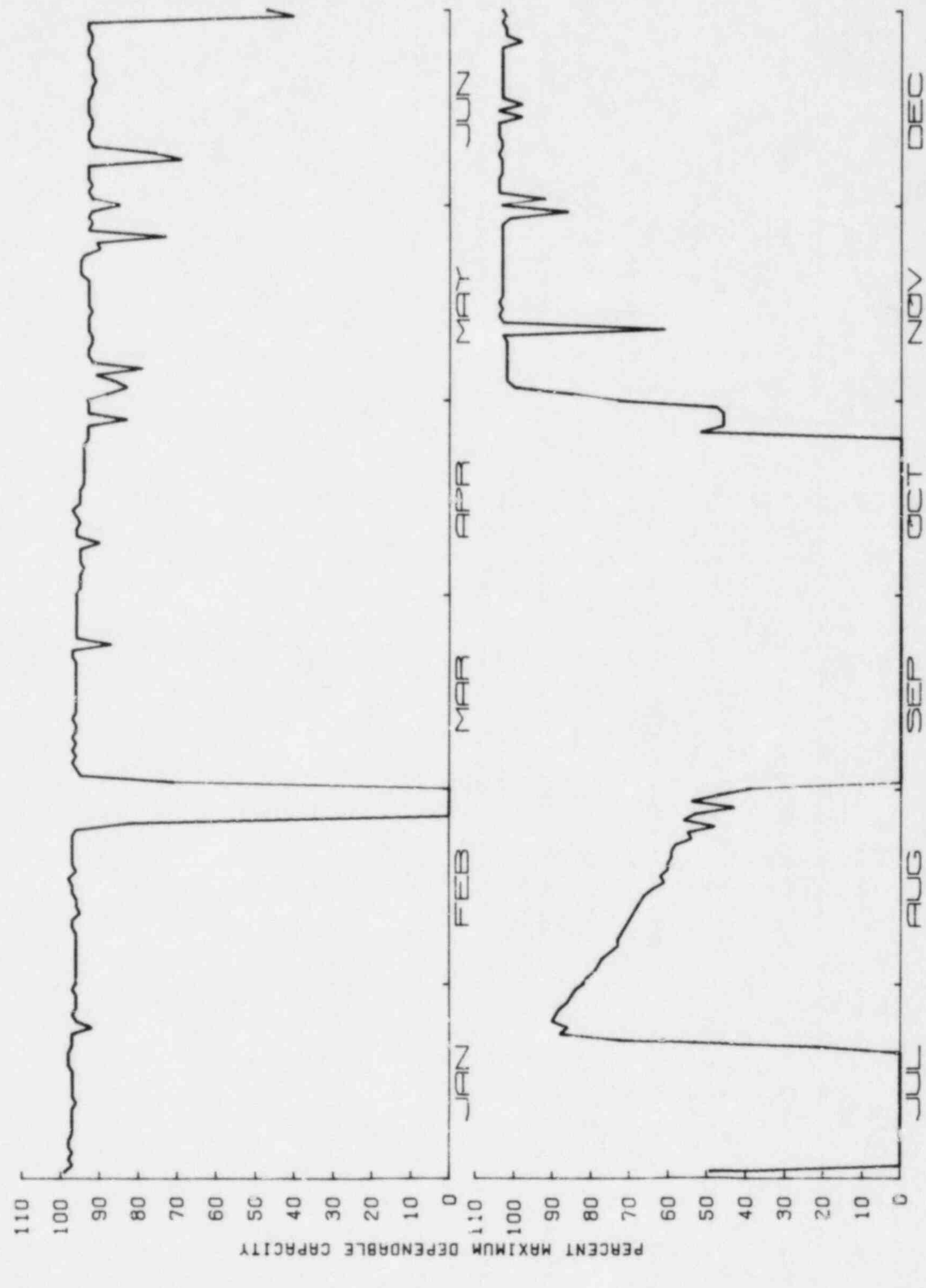
Total hours: 1,916.9 (21.8%)
- Forced: 184.1 (2.1%)
- Scheduled: 1,732.8 (19.7%)

II. Highlights

The unit experienced only five outages in 1980, including a refueling between August 31 and October 26. Steam generator tube inspection and tube plugging required 20 d in July.

DETAILS OF PLANT OUTAGES FOR PRAIRIE ISLAND 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	2/23	136.2	S	No. 11 RCP No. 2 seal repair.	B	1	Reactor coolant (CB)	Pumps
2	7/01	176.0	F	Increasing SG tube leakage, one tube in No. 12 SG plugged (LER 80-18).	A	2	Steam and power conversion (HB)	Heat exchangers
3	7/08	293.1	S	Routine eddy current examination of SG tubes.	B	2	Steam and power conversion (HB)	Heat exchangers
4a	8/31	1288.0	S	Refueling.	C	2	Reactor (RC)	Fuel elements
4b	10/24	15.5	S	Turbine overspeed test and generator short circuit test.	B		Steam and power conversion (HA)	Turbines
5	11/11	8.1	F	Spurious SI while performing safeguards logic test.	G	3	Instrumentation and controls (IA)	Instrumentation and controls



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
Maximum dependable capacity (MWe-net): 500
Commercial operation: 12/21/74
Years operating experience: 6.0

Performance

Net electrical energy generated (MWh): 3,467,271
Unit availability factor (%): 81.6
Unit capacity factor (using MDC): 79.0
Unit capacity factor (%) (using design MWe): 74.5

Outages

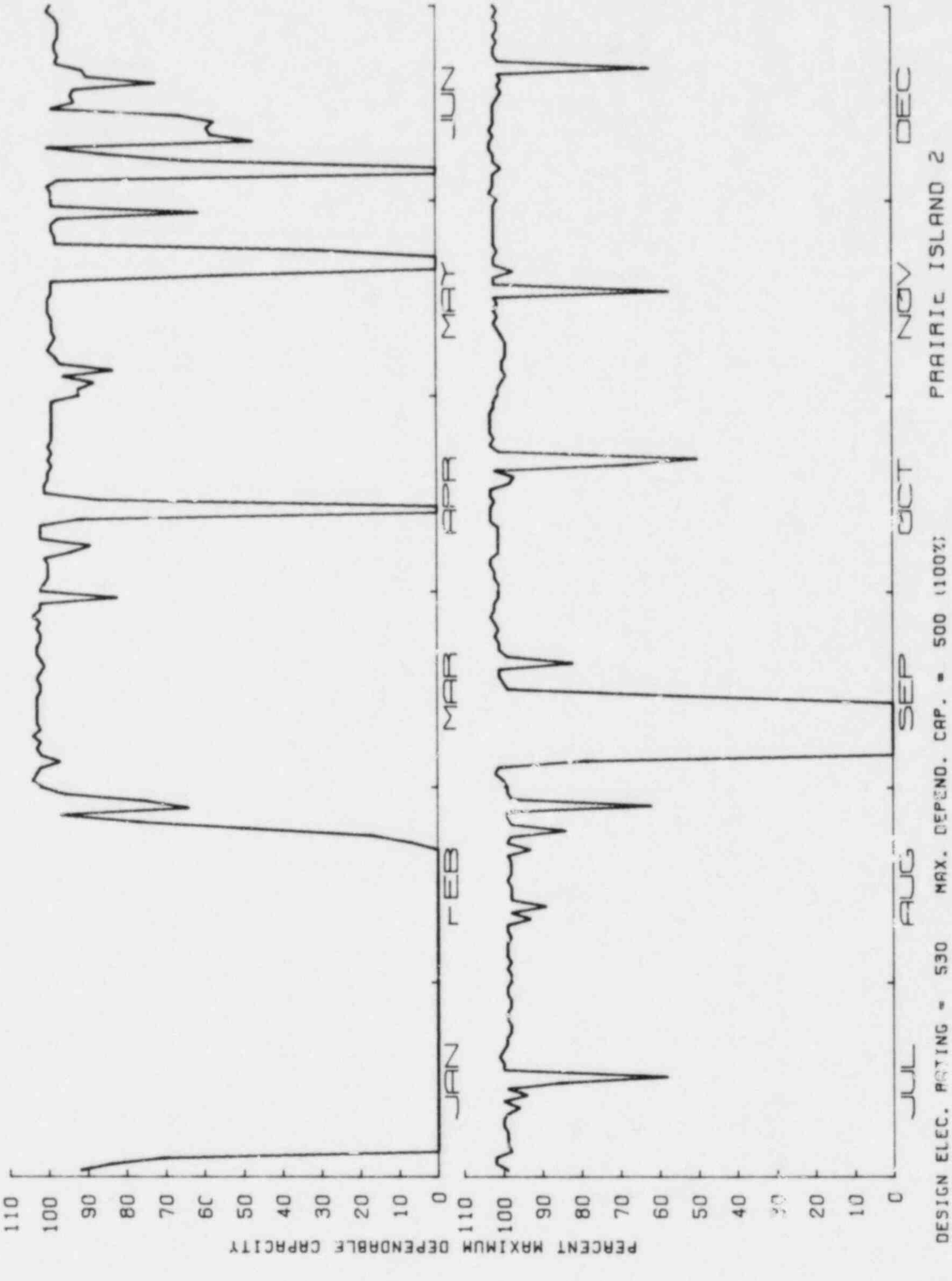
Total No.: 12
Forced: 6
Scheduled: 6
Total hours: 1,614.2 (18.4%)
Forced: 45.0 (0.5%)
Scheduled: 1,569.2 (17.9%)

II. Highlights

As many scheduled outages (six) were reported at Prairie Island 2 as forced outages. Scheduled outages occurred on January 2 for refueling, February 20 at the completion of refueling for a turbine overspeed trip test, April 11 for repair of the turbine oil system, May 19 for reactor coolant pump seal repair, May 29 and June 4 for transformer maintenance, and September 4 for inspection of steam generator support bolts. The unit availability was 81.6%.

DETAILS OF PLANT OUTAGES FOR PRAIRIE ISLAND 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/02	1148.1	S	Refueling.	C	2	Reactor (RC)	Fuel elements
2	2/20	12.0	S	Turbine overspeed test and generator short-circuit test.	B	1	Steam and power conversion (HA)	Not applicable
3	2/21	7.9	F	Turbine steam leak repair.	A	1	Steam and power conversion (HA)	Pipes, fittings
4	2/25	2.6	F	Packing in FW regulating valve replaced.	A	1	Steam and power conversion (HH)	Valves
5	4/11	48.0	S	Repair turbine oil system.	A	1	Steam and power conversion (HA)	Turbines
6	5/19	95.8	S	Repair No. 2 seal on No. 22 RCP.	A	1	Reactor coolant (CB)	Pumps
7	5/29	2.0	F	Remove 2M transformer from service.	B	1	Electric power (EG)	Transformers
8	6/04	49.2	S	Return 2M transformer to service and perform maintenance on turbine oil system.	B	3	Electric power (EG)	Transformers
9	6/06	3.6	F	Manual turbine trip due to high vibration.	H	1	Steam and power conversion (HA)	Turbines
10	7/15	11.4	F	Two sources of offsite power lost during electrical storm (LER 80-20).	H	3	Electric power (EA)	Not applicable
11	9/04	216.1	S	SG support bolt inspection (LER 80-25). Also, unit 1 SG bolts were inspected, and all were replaced.	B	2	Steam and power conversion (HB)	Shock suppressors and support
12	10/20	17.5	F	Construction person accidentally bumped the trip switch for the breaker supplying power to the turbine EHC pumps, and turbine control valves drifted closed.	G	2	Steam and power conversion (HA)	Circuit closure/interrupts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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
Maximum dependable capacity (MWe-net): 769
Commercial operation: 2/18/73
Years operating experience: 8.7

Performance

Net electrical energy generated (MWh): 3,441,743
Unit availability factor (%): 66.5
Unit capacity factor (using MDC): 51.0
Unit capacity factor (%) (using design MWe): 49.7

Outages

Total No.: 7
  Forced: 7
  Scheduled: 0
Total hours: 2,491.1 (33.5%)
  Forced: 191.8 (2.2%)
  Scheduled: 2,749.3 (31.3%)[a]

II. Highlights

Refueling began August 31 following a failure of an electromagnetic relief value to seat. Load reductions occurred on January 26 and March 12 to change the condensate demineralizers, February 2 to check condenser tube leaks, February 24 for control rod sequence exchange, and March 2 and 12 for control rod pattern changes.

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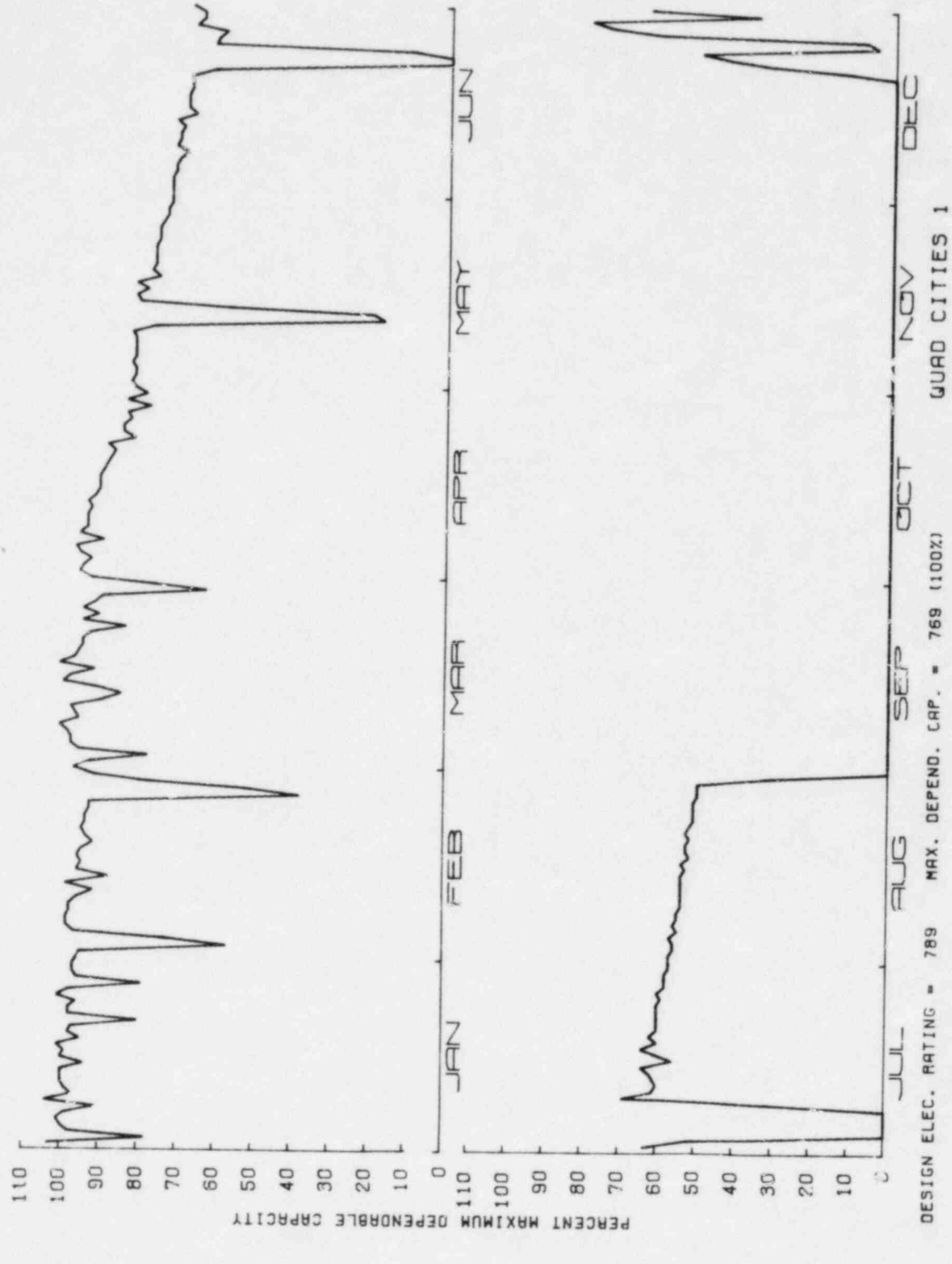
[a]The July 7 scram testing and August 31 refueling hours are included here.

DETAILS OF PLANT OUTAGES FOR QUAD CITIES 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	5/11	12.2	F	1B recirculation motor - generator breaker replacement.	A	1	Reactor coolant (CB)	Circuit closures/interrupters
2	6/20	55.6	F	Low main condenser vacuum.	A	3	Steam and power conversion (HC)	Heat exchangers
3a	7/03	64.0	F	Leak on feedwater check valve.	A	1	Reactor coolant (CH)	Valves
3b	7/03	66.2	S	Scram test of scram discharge volume per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
4a	8/31	0.0	F	Electromagnetic relief valve failed to seat. Generator taken off-line to begin scheduled refueling (LER 80-20).	A	2	Reactor coolant (CB)	Valves
4b	8/31	2683.1	S	Refueling.	C	4	Reactor (RC)	Fuel elements
5	12/20	4.7	F	High vibration turbine trip.	A	3	Steam and power conversion (HA)	Turbines
6	12/25	36.1	F	Average power range monitor hi-hi trip due to recirculation pump transient.	A	3	Reactor coolant (CB)	Not applicable
7	12/30	9.2	F	Spurious reactor vessel low water level signal.	A	3	Instrumentation and controls (IA)	Instrumentation and controls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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
Maximum dependable capacity (MWe-net): 769
Commercial operation: 3/10/73
Years operating experience: 8.6

Performance

Net electrical energy generated (MWh): 3,614,427
Unit availability factor (%): 62.5
Unit capacity factor (using MDC): 53.5
Unit capacity factor (%) (using design MWe): 52.2

Outages

Total No.: 13
Forced: 11
Scheduled: 2
Total hours: 3,297.6 (37.5%)[a]
Forced: 496.6 (5.7%)
Scheduled: 2,801.0 (31.8%)[a]

II. Highlights

Quad Cities 2 resumed power operations April 21 after being down for refueling since November 1979. Operation was routine for the remainder of the year. On November 16 a shutdown was initiated because of simultaneous malfunctions in the RCIC and HPCI systems; load was reduced to 360 MWe only. Two automatic scrams occurred in December because of closure of a main steam isolation valve caused by a spurious main steam line high flow signal.

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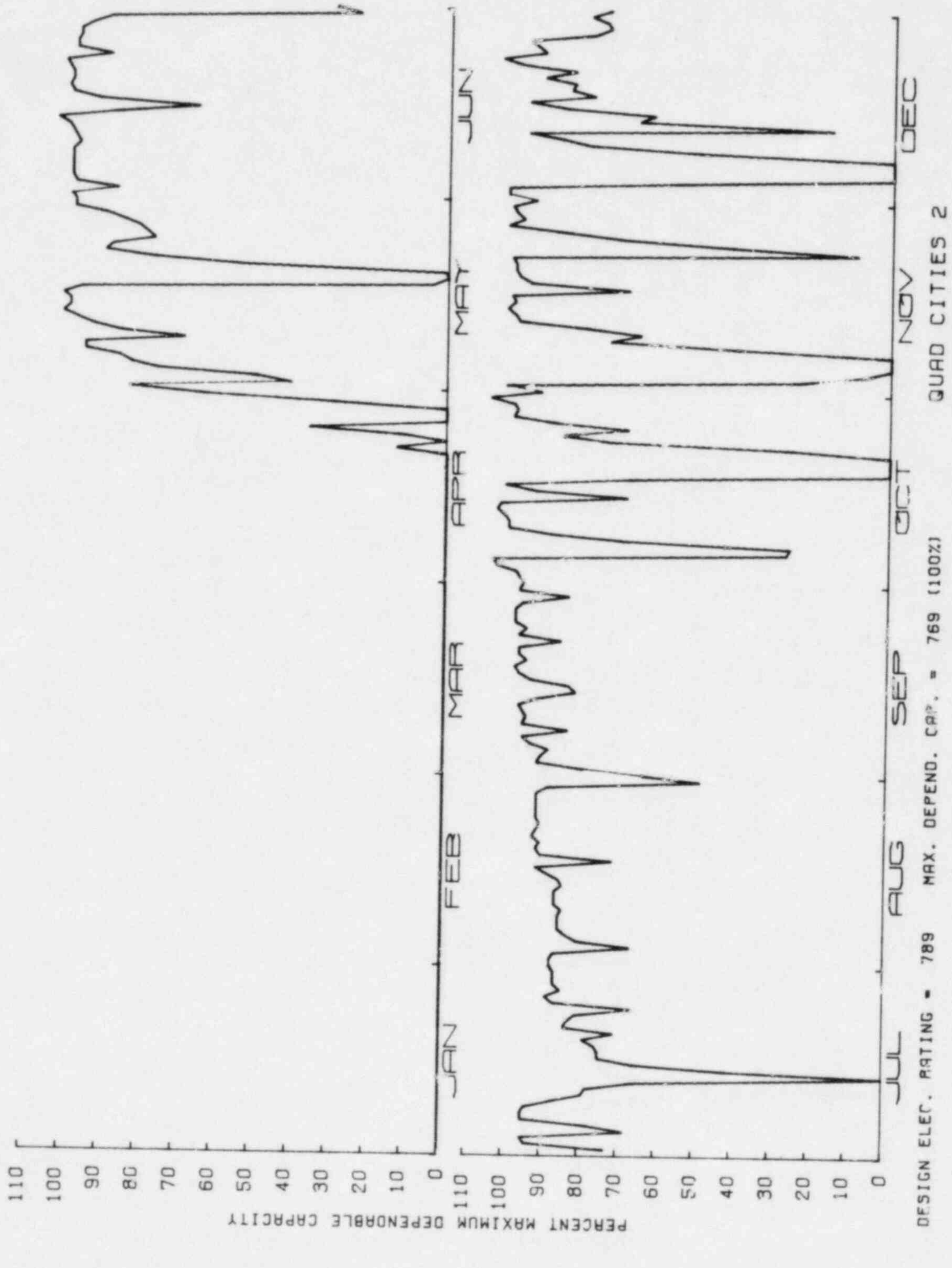
[a] Including 2,665.8 h in 1980 from continued 11/25/79 outage.

DETAILS OF PLANT OUTAGES FOR QUAD CITIES 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	11/25/79 (cont.)	2665.8	S	Refueling.	C	4	Reactor (RC)	Fuel elements
1	4/20	39.4	F	Repair 3C electromatic relief valve (LER 80-11).	A	1	Reactor coolant (CC)	Valves
2	4/22	78.1	F	Repair pressure suppression chamber vacuum breakers.	A	1	Engineered safety features (SE)	Vessels, pressure
3	5/01	12.9	F	Low RCS level due to 2B FW regulation valve failure in closed mode.	A	3	Reactor coolant (CH)	Valves
4	5/17	66.6	F	Main condenser tube leak repair.	B	1	Steam and power conversion (HC)	Heat exchanger
5	6/29	24.8	F	Steam leak on turbine control valve.	A	1	Steam and power conversion (HA)	Valves
6	7/12	28.4	S	Scram testing of scram discharge volume per IE Bulletin 80-17.	D	2	Reactor (RB)	Control rods
7	10/05	21.1	F	Fire caused by oil leaking from speed adjusting valve on MSIV and flashing to fire when contacting hot valve body.	H	2	Reactor coolant (CD)	Valves
8	10/17	106.8	S	Battery tests and miscellaneous maintenance items.	B	1	Electric power (EC)	Batteries and chargers
9	11/02	15.9	F	Average power range monitor high flux due to recirculation pump motor generator set speed transient.	G	3	Electric power (EB)	Motors
10	11/03	91.1	F	Leaking recirculation suction valve.	A	1	Reactor coolant (CB)	Valves

DETAILS OF PLANT OUTAGES FOR QUAD CITIES 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11	11/22	22.4	F	Leaking recirculation suction valve.	A	1	Reactor coolant (CB)	Valves
12	12/03	108.9	F	Spurious main steam line high flow signal caused MSIV closure.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
13	12/11	15.4	F	Spurious main steam line low pressure signal caused MSIV closure.	A	3	Instrumentation and controls (IA)	Instrumentation and controls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DESIGN ELEC. RATING = 789   MAX. DEPEND. CAP. = 769 (100%)
QUAD CITIES 2

RANCHO SECO

I. Summary

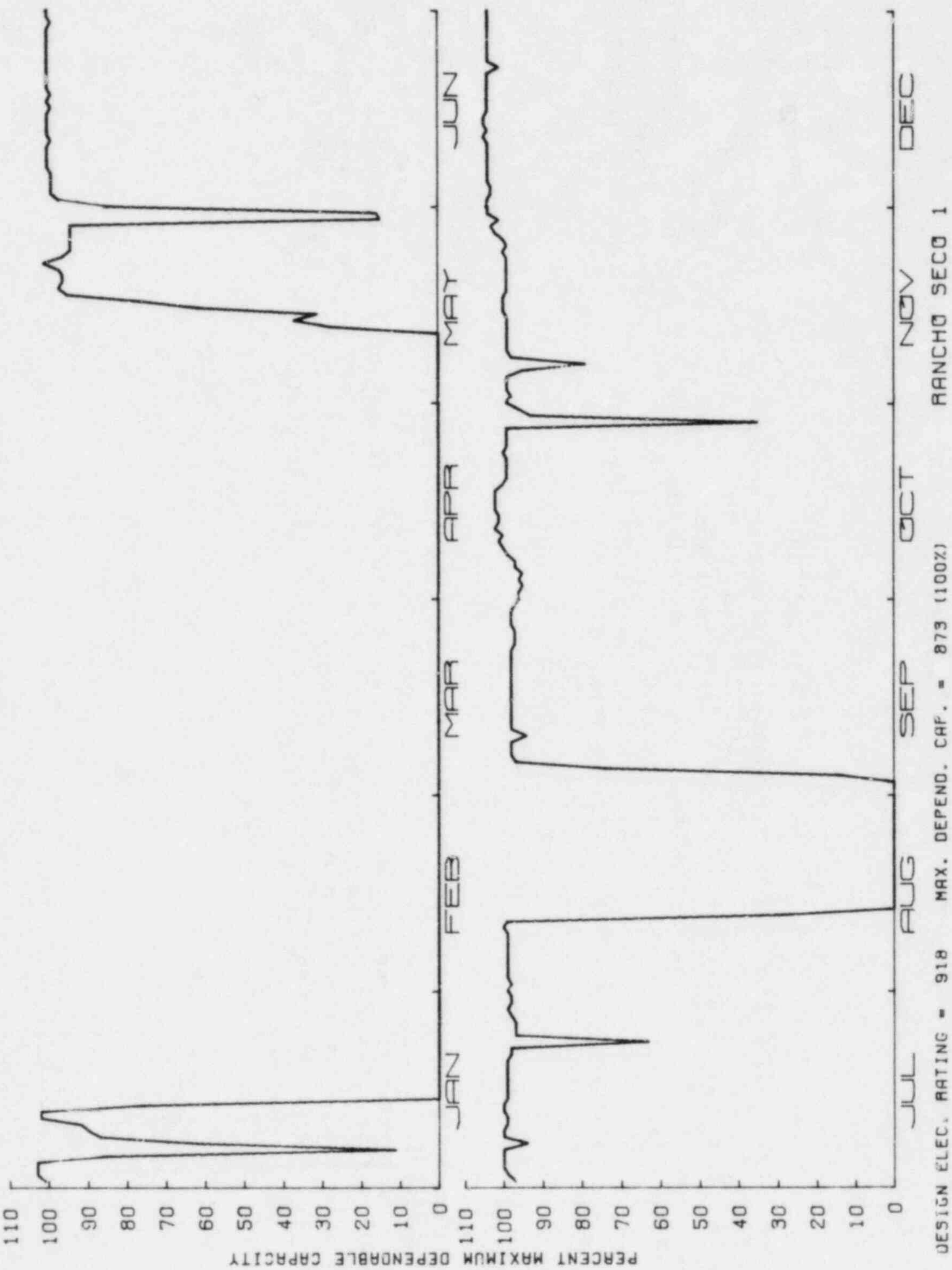
Description	Performance	Outages
Location: Sacramento, California	Net electrical energy generated (MWh): 4,415,236	Total No.: 7
Docket No.: 50-312	Unit availability factor (%): 60.4	Forced: 5
Reactor type: PWR	Unit capacity factor (using MDC): 57.6	Scheduled: 2
Maximum dependable capacity (MWe-net): 873	Unit capacity factor (%) (using design MWe): 54.8	Total hours: 3,479.7 (39.6%)
Commercial operation: 4/17/75		Forced: 606.7 (6.9%)
Years operating experience: 6.2		Scheduled: 2,873.0 (32.7%)

II. Highlights

Refueling commenced after the unit was shut down on January 12 due to a pressurizer spray valve leak and was completed on May 12. Turbine thrust bearing repair required over 3 weeks in August and September to complete. Otherwise, operations were routine.

DETAILS OF PLANT OUTAGES FOR RANCHO SECO

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/05	15.7	S	Analyses by A/E identified support HPCI and RHR common header with safety factor <2 (LER 80-02).	D	1	Reactor coolant (CF)	Shock suppressors and support
2a	1/12	29.4	F	Pressurizer spray valve leaks (LER 80-04).	A	2	Reactor coolant (CB)	Valves
2b	1/14	2856.0	S	Refueling.	C	4	Reactor (RC)	Fuel elements
3	5/14	1.3	S	Turbine trip testing.	B	1	Steam and power conversion (HA)	Turbines
4	5/29	27.4	F	Low EHC oil pressure.	A	1	Steam and power conversion (HA)	Motors
5	5/30	7.6	F	Loss of condenser vacuum due to electrical problem with main circulating water pumps.	A	3	Electric power (EB)	Pumps
6	8/12	531.7	F	Damaged turbine thrust bearing.	A	3	Steam and power conversion (HA)	Turbines
7	10/28	10.6	F	FW flow imbalance due to loss of instrument air pressure.	A	3	Auxiliary process (PA)	Air dryers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DESIGN ELEC. RATING = 918 MAX. DEPEND. CAP. = 873 (100%) RANCHO SECO 1

ROBINSON 2

I. Summary

Description

Location: Hartsville, South Carolina
Docket No.: 50-261
Reactor type: PWR
Maximum dependable capacity (MWe-net): 665
Commercial operation: 3/07/71
Years operating experience: 10.3

Performance

Net electrical energy generated (MWh): 3,211,350
Unit availability factor (%): 62.2
Unit capacity factor (using MDC): 55.0
Unit capacity factor (%) (using design MWe): 52.2

Outages

Total No.: 17
Forced: 15
Scheduled: 2
Total hours: 3,316.2 (37.8%)
Forced: 1,463.3 (16.7%)
Scheduled: 1,852.9 (21.1%)

II. Highlights

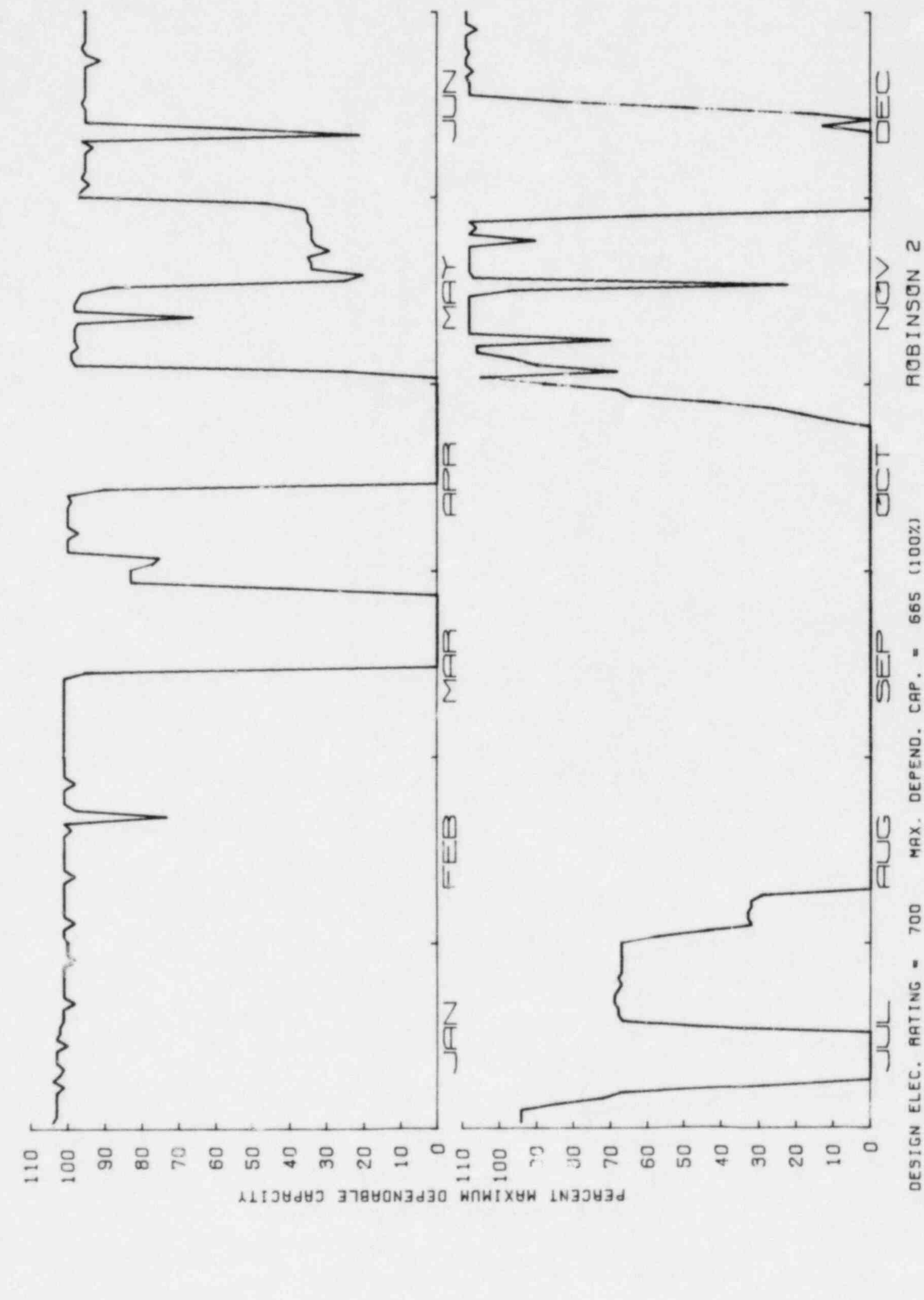
Robinson 2 experienced a short power reduction on February 19 for condenser tube leak repairs and three shutdowns on March 14, April 13, and July 7 for steam generator tube leak repairs. The outages for steam generator tube repair and plugging accounted for 68% of the total forced downtime at Robinson in 1980. Refueling took 11 weeks beginning August 8 and concluding October 25. A 2-week outage beginning at the end of November was necessary because of problems with an auxiliary pressurizer spray valve, the nuclear instrumentation system, a control rod, and the charging system.

DETAILS OF PLANT OUTAGES FOR ROBINSON 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	3/14	321.5	F	SG tube leak test and repair.	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
2	4/13	448.2	F	Seal failure in RCP. Also, 100% eddy current testing in all 3 SGs led to 149 tubes being plugged in B SG.	A	1	Reactor coolant (CB)	Pumps
3	5/17	14.0	F	Radial tilt limit of 1.02 exceeded while reducing power. With N-42 inoperable due to a failed detector, trip setpoints could not be reset with the reactor critical. Detector replaced (LER 80-13).	A	1	Instrumentation and controls (IA)	Instrumentation and controls
4	5/22	2.7	F	I and C technicians shorted-out RCP bearing temperature recorder.	H	3	Reactor coolant (CB)	Instrumentation and controls
5	6/10	2.4	F	High pressurizer pressure due to a defective contact (relay 152X) which caused the turbine governor and intercept valves to close.	A	3	Steam and power conversion (HA)	Circuit closures/ interrupters
6	6/10	4.9	F	Unit was separated from the system as condenser vacuum continued to drop; a condenser vacuum pump was inoperable due to a motor ground and B pump lost seal water flow.	A	1	Steam and power conversion (HC)	Pumps
7	7/07	224.9	F	SG a tube leaking at 0.32 gpm; tube plugged.	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
8	8/08	1850.6	S	Refueling and maintenance.	C	1	Reactor (RC)	Fuel elements
9	10/25	2.3	S	Turbine overspeed trip test.	B	1	Steam and power conversion (HA)	Turbine

DETAILS OF PLANT OUTAGES FOR ROBINSON 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10	10/25	3.5	F	Generator grounding strap reconnected.	B	1	Steam and power conversion (HA)	Generators
11	10/25	7.2	F	Steam flow/FW flow mismatch and 30% SG level.	A	3	Steam and power conversion (HH)	Valves
12	10/26	3.4	F	SG B high-high level due to B FW regulator valve swinging open. Valve was stroked and lubricated.	A	3	Steam and power conversion (HH)	Valves
13	11/02	8.0	F	Feed flow/steam flow mismatch due to FW regulator valve.	A	3	Steam and power conversion (HH)	Valves
14	11/07	6.7	F	Loss of condensate and FW pumps when hotwell level switch was bumped.	G	3	Steam and power conversion (HC)	Instrumentation and controls
15	11/15	18.6	F	Balance turbine and B RCP.	B	1	Steam and power conversions (HA)	Turbines
16	11/27	356.7	F	Repair CVC-311 packing leak in auxiliary pressurizer spray valve. Problems with the nuclear instrumentation system, a control rod, and the charging system extended this outage (LER 80-28).	A	1	Auxiliary process (PC)	Valves
17	12/12	40.6	F	Repair RHR-750 packing leak (LER 80-29).	A	1	Reactor coolant (CF)	Valves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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): 5,684,483	Total No.: 14
Docket No.: 50-272	Unit availability factor (%): 69.2	Forced: 11
Reactor type: PWR	Unit capacity factor (using MDC): 60.0	Scheduled: 3
Maximum dependable capacity (MWe-net): 1,079	Unit capacity factor (%) (using design MWe): 59.4	Total hours: 2,705.9 (30.8%)
Commercial operation: 6/30/77		Forced: 363.8 (4.1%)
Years operating experience: 4.0		Scheduled: 2,342.1 (26.7%)

II. Highlights

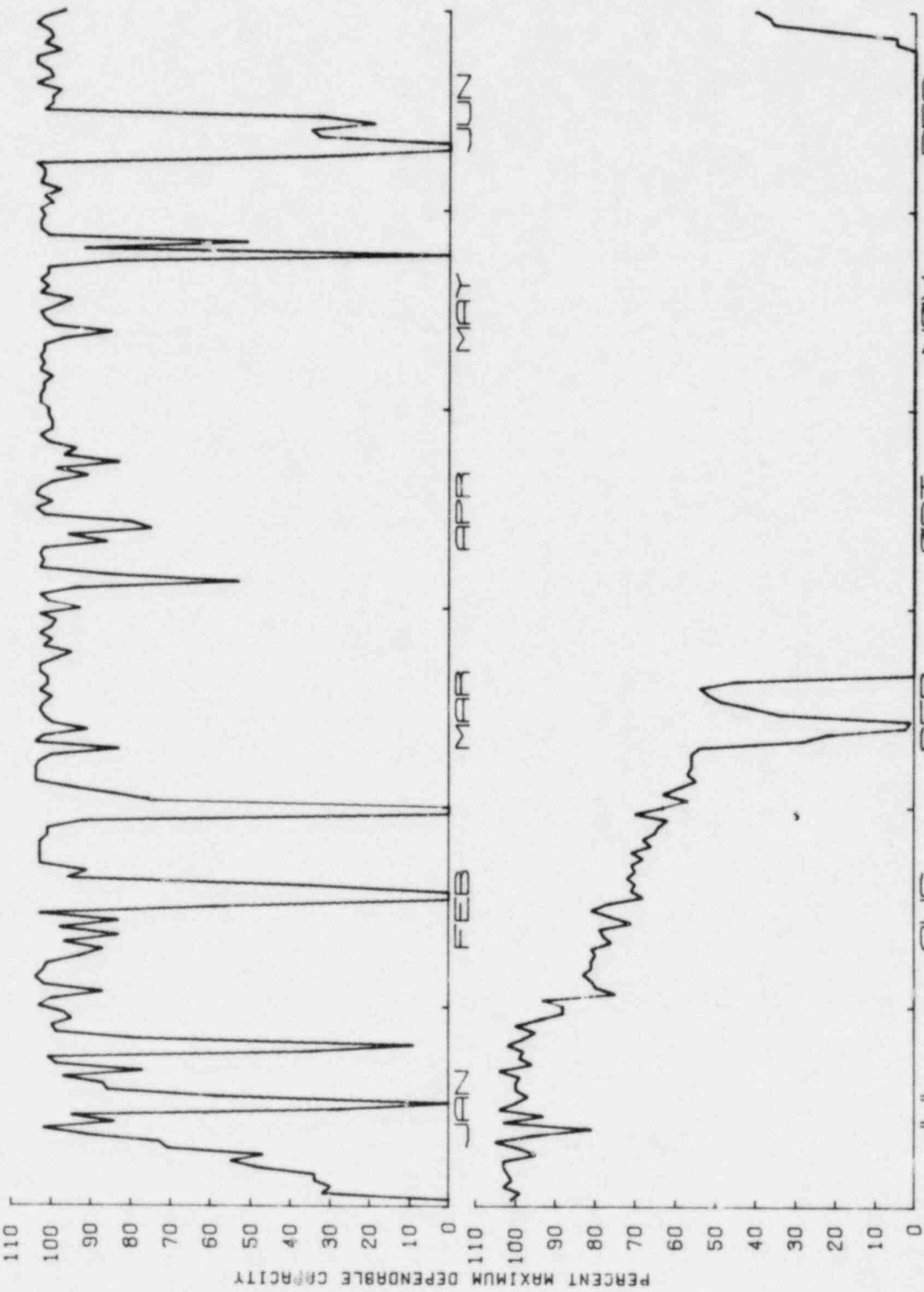
Power escalation testing continued from 1979 maintenance outage until January 26. Power reductions were necessary to clean suction strainers of condensate pumps from January through May, to clean condenser water boxes in February and March, to repair traveling screens in July, and to accommodate fuel depletion in August and September. Refueling commenced September 19 and the unit returned on-line December 26.

DETAILS OF PLANT OUTAGES FOR SALEM 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/14	37.1	F	Spurious noise spike on power range channel N-43 while N-44 was in test.	A	3	Reactor (RB)	Not applicable
2	1/23	25.2	F	Loss of auxiliary transformer.	A	3	Electric power (EG)	Transformers
3	2/14	60.0	F	SG low-level and low-flow due to failure of valve positioner.	A	3	Steam and power conversion (HH)	Mechanical function units
4	2/26	67.4	F	Loss of stator water cooling to the generator due to switch failure.	A	3	Steam and power conversion (HA)	Instrumentation and controls
5	5/23	31.6	F	SG FW control malfunction.	A	3	Steam and power conversion (HH)	Valve operators
6a	6/08	29.2	F	SG channel pressure control failure due to lightning (LER 80-31).	A	3	Steam and power conversion (HH)	Instrumentation and controls
6b	6/09	39.4	F	Repair No. 11 fan coil unit.	A	4	Other auxiliary (AA)	Blowers
7	6/12	4.8	F	Technician error during functional test of N-43 – instrumentation channel.	H	3	Instrumentation and controls (IA)	Instrumentation and controls
8	6/13	8.9	F	1A vital 4160 volt bus trip.	H	3	Electric power (EB)	Circuit closures/interrupters
9	6/14	5.4	F	Low SG level due to MFWP trip.	A	3	Steam and power conversion (HH)	Pumps
10	9/10	18.0	F	SG No. 12 low level due to loss of control air to valve 12BF19.	A	3	Steam and power conversion (HH)	Valve operators
11	9/12	36.8	F	Water in turbine lube oil.	H	2	Auxiliary process (PA)	Turbines
12a	9/19	1905.3	S	Refueling.	C	1	Reactor (RC)	Fuel elements

DETAILS OF PLANT OUTAGES FOR SALEM 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12b	12/08	416.7	S	NRC requirement for modifications to the plants, building reinforcement, and SG testing.	D	4	System code not applicable (ZZ)	Not applicable
13	12/26	10.5	S	SG testing.	B	3	Steam and power conversion (HB)	Heat exchangers (steam generator)
14	12/27	9.6	S	Turbine overspeed testing.	B	3	Steam and power conversion (HA)	Turbines



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

SAN ONOFRE 1

I. Summary

Description

Location: San Clemente, California
Docket No.: 50-206
Reactor type: PWR
Maximum dependable capacity (MWe-net): 436
Commercial operation: 1/01/68
Years operating experience: 13.5

Performance

Net electrical energy generated (MWh): 816,676
Unit availability factor (%): 22.3
Unit capacity factor (using MDC): 21.3
Unit capacity factor (%) (using design MWe): 21.3

Outages

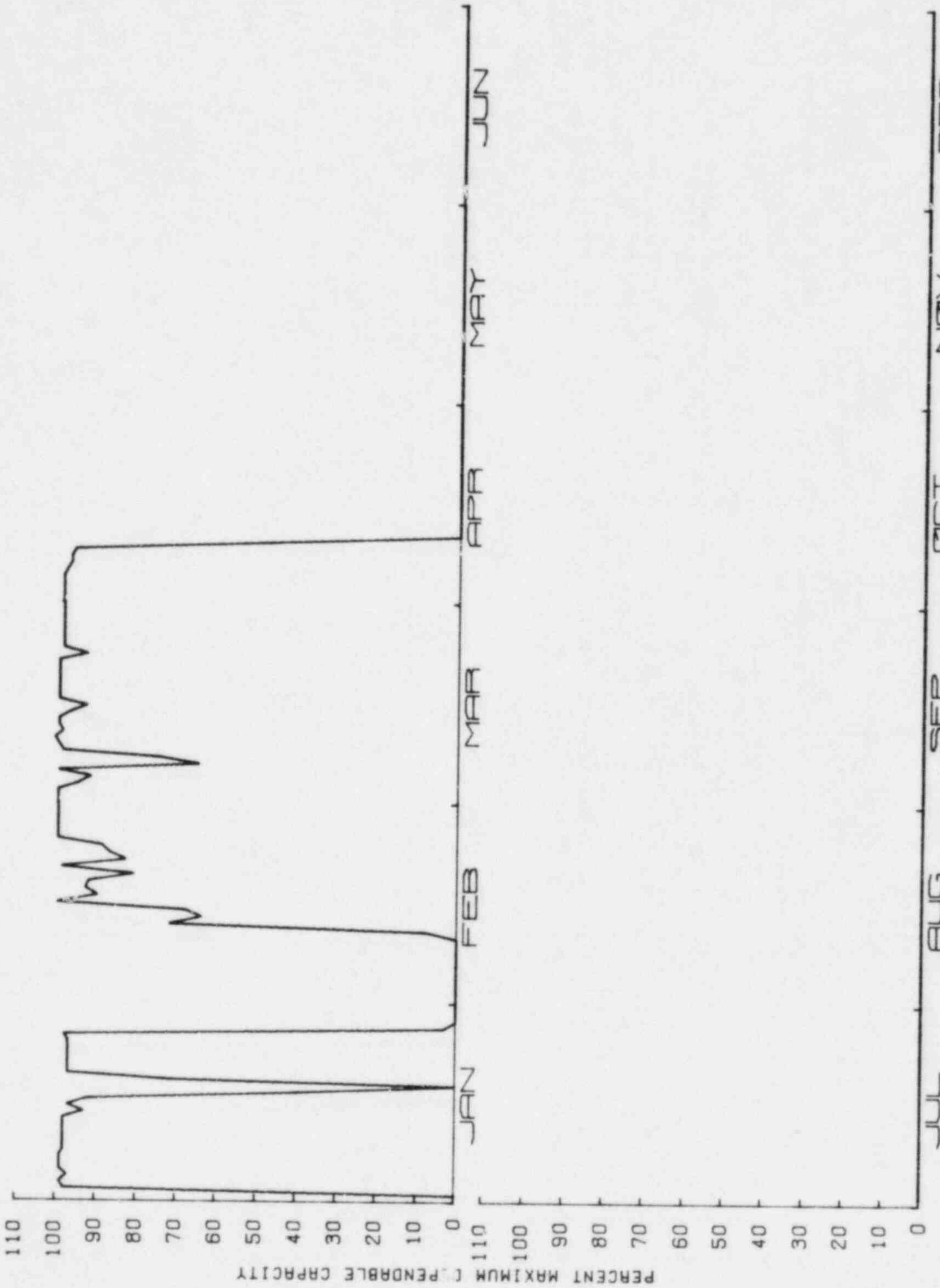
Total No.: 5
Forced: 3
Scheduled: 2
Total hours: 6,122.4 (69.7%)
Forced: 3,509.6 (40.0%)
Scheduled: 2,612.8 (29.7%)

II. Highlights

TMI-related modifications required being shut down 2 weeks starting January 26. Several power reductions were necessary in February and March to clean condenser water boxes. Refueling occurred between April 9 and July 11 at which time the outage was continued through the remainder of the year for steam generator tube repair.

DETAILS OF PLANT OUTAGES FOR SAN ONOFRE 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/16	37.8	F	SF/FFMM due to construction worker who accidentally struck the closing circuit control relay to MFWP normal discharge valve (LER 80-02).	G	3	Steam and power conversion (HH)	Relays
2	1/26	372.1	S	TMI-related modifications.	D	1	Other (XX)	Other
3	2/12	8.0	F	Governor control oil system repair.	A	1	Steam and power conversion (HA)	Turbines
4	3/06	10.8	F	Ruptured pressure relief tank diaphragm due to overfilling (80-11).	A	1	Reactor coolant (CJ)	Vessels pressure
5a	4/09	2240.7	S	Refueling and maintenance.	C	1	Reactor (RC)	Fuel elements
6	7/12	3453.0	F	SG tube repair (LER 80-14).	B	4	Steam and power conversion (HB)	Heat exchangers (steam generator)



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

ST. LUCIE 1

I. Summary

Description

Location: Ft. Pierce, Florida
Docket No.: 50-335
Reactor type: PWR
Maximum dependable capacity (MWe-net): 777
Commercial operation: 12/21/76
Years operating experience: 4.7

Performance

Net electrical energy generated (MWh): 5,199,590
Unit availability factor (%): 77.5
Unit capacity factor (using MDC): 76.2
Unit capacity factor (%) (using design MWe): 73.8

Outages

Total No.: 12
- Forced: 6
- Scheduled: 6

Total hours: 1,979.5 (22.5%)
- Forced: 501.5 (5.7%)
- Scheduled: 1,478.0 (16.8%)

II. Highlights

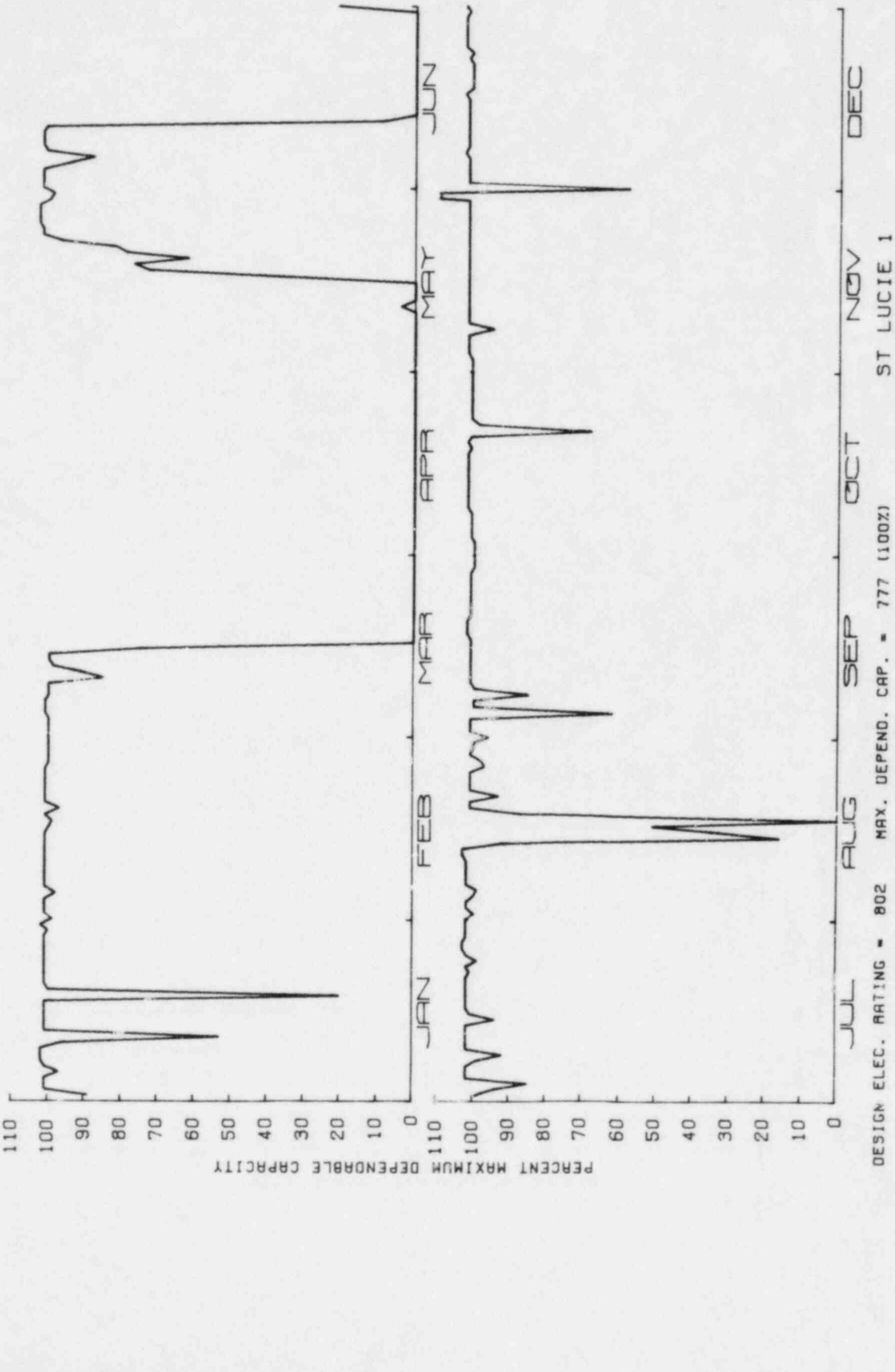
Refueling was performed between March 15 and May 10. A main power transformer oil leak and bushing problems caused three shutdowns and a power reduction in August. Operation was routine for the remainder of the year.

DETAILS OF PLANT OUTAGES FOR ST. LUCIE 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/17	16.1	F	Trip signaled by RPS during periodic surveillance test wher a second breaker failed to remain closed after a test.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
2	3/15	1335.2	S	Refueling, maintenance, and inspections.	C	1	Reactor (RC)	Fuel elements
3	5/11	3.7	S	Turbine overspeed trip test.	B	1	Steam and power conversion (HA)	Mechanical function units
4	5/11	96.5	S	Turbine overspeed trip test and turbine shaft seal No. 3 repair. Outage was extended to replace valve stems on the bypass valves around MSIVs 1A and 1B and to repair valve stem packing leaks on valves inside containment that could not be isolated.	B	1	Steam and power conversion (HA)	Mechanical function units
5	6/11	467.8	F	Loss of component cooling water to RCP mechanical seals. Corrective actions included providing a backup nitrogen supply to the diaphragm operated CCW valves. Outage was extended to inspect mechanical seals on RCPs (LER 80-29).	A	2	Auxiliary water (WB)	Valve operators
6	8/14	10.7	S	Main power transformers No. 1 isolated to repair oil leak. Upon startup, load limited to capacity of main power transformer No. 1A.	A	1	Electric power (EG)	Transformers
7	8/14	7.3	S	Unit was removed from service to protect personnel using a crane to remove faulty bushing from main power transformer No. 1B.	F	1	Electric power (EG)	Transformers
8	8/17	24.6	S	New bushing installed and main power transformer 1B returned to service.	B	1	Electric power (EG)	Transformers

DETAILS OF PLANT OUTAGES FOR ST. LUCIE 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	9/04	3.6	F	Two sequential CEA drops. CEA's 15 V power supply modified (LER 80-50).	A	2	Reactor (RB)	Instrumentation and controls
10	10/21	5.8	F	SG level protection system trip due to spurious control signal to FW control valve 1A.	A	3	Steam and power conversion (HH)	Instrumentation and controls
11	11/30	6.1	F	Loss of power supply to the control system when the output breaker of the second of two MG sets tripped open MG 1A was removed from service while investigating DC ground isolation.	A	3	Instrumentation and controls (IA)	Instrumentation and controls
12	11/30	2.1	F	SG level protection system trip during load increase.	A	3	Steam and power conversion (HH)	Not applicable


PERCENT MAXIMUM DEPENDABLE CAPACITY
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110
DESIGN ELEC. RATING = 802
MAX. DEPEND. CAP. = 777 (100%)
ST LUCIE 1

I. Summary

Description	Performance	Outages
Location: Surry, Virginia	Net electrical energy generated (MWh): 2,473,025	Total No.: 5
Docket No.: 50-280	Unit availability factor (%): 44.9	Forced: 3
Reactor type: BWR	Unit capacity factor (using MDC): 36.3	Scheduled: 2
Maximum dependable capacity (MWe-net): 775	Unit capacity factor (%) (using design MWe): 34.3	Total hours: 5,015.9 (57.1%)[a]
Commercial operation: 12/22/72		Forced: 445.0 (5.1%)
Years operating experience: 8.5		Scheduled: 4,570.9 (52.0%)[a]

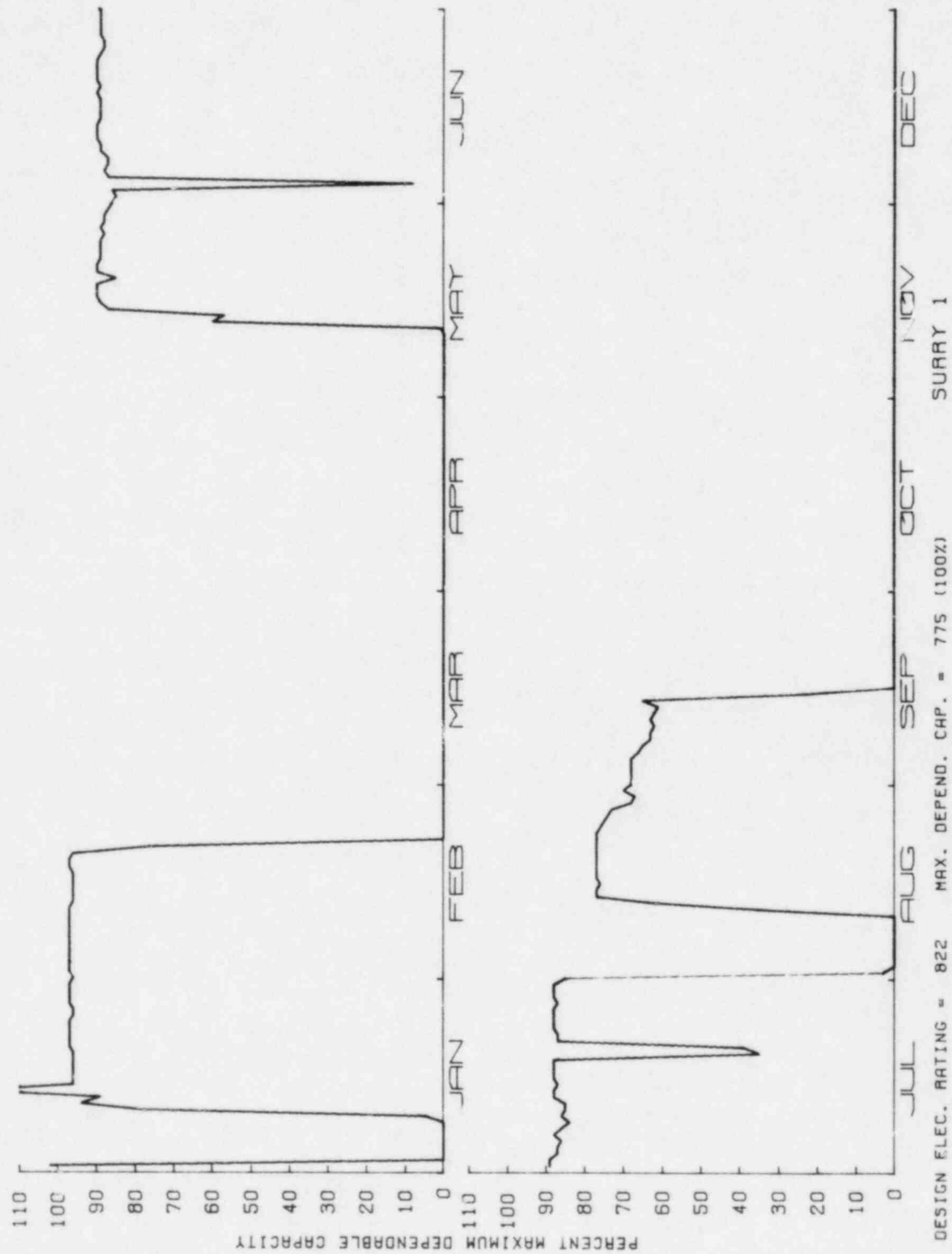
II. Highlights

The unit began 1980 down for replacement of a reactor coolant pump motor and testing of a snubber. Pipe stress reanalysis and turbine inspection per IE Bulletin 79-14 required nearly 12 weeks ending May 11. One steam generator tube was plugged on August 1, but the unit was shut down for the remainder of the year on September 14 to replace the lower shells and tube bundles in all three steam generators.

[a]Includes 183.6 h in 1980 from continued 12/30/79 outage.

DETAILS OF PLANT OUTAGES FOR SURRY 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/30/79 (cont.)	183.6	F	Replacement of RCP motor. Also, NRC requirement to successfully test RCP snubber.	D	4	Reactor coolant (CB)	Shock suppressors and supports
1	2/19	1965.6	S	Pipe stress reanalysis and turbine inspection per IE Bulletin 79-14 (LER 80-18).	D	1	Steam and power conversion (HD)	Pipes, fittings
2	5/11	1.9	F	Feed flow/steam flow mismatch and low SG level during startup.	G	3	Steam and power conversion (HC)	Heat exchangers (steam generator)
3	6/03	18.5	F	Loss of power to I-IV vital bus due to fire in its transformer, safety injection initiated (LER 80-29).	A	3	Electric power (EB)	Transformers
4	8/01	241.0	F	Tube leak in 1C SG greater than 0.3 gpm. The tube plugged (LER 80-40).	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
5	9/14	2505.3	S	SG repair outage leakage in SG B was 0.283 gpm. All 3 SG lower shells and tube bundles will be replaced.	A	1	Steam and power conversion (HB)	Heat exchangers (steam generator)


PERCENT MAXIMUM DEPENDABLE CAPACITY
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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): 2,241,883	Total No.: 11
Docket No.: 50-281	Unit availability factor (%): 35.8	Forced: 10
Reactor type: PWR	Unit capacity factor (using MDC): 32.9	Scheduled: 1
Maximum dependable capacity (MWe-net): 775	Unit capacity factor (%) (using design MWe): 31.0	Total hours: 5,643.6 (64.3%)[a]
Commercial operation: 5/01/73		Forced: 100.9 (1.2%)
Years operating experience: 7.8		Scheduled: 5,542.7 (63.1%)[a]

II. Highlights

The unit began the year in a shutdown mode which continued until August 19 for seismic reanalysis and pipe restraint modifications. Operation was routine for the remainder of the year. Five of the ten forced shutdowns were attributed to operator error. Four of those five involved steam generator level trips.

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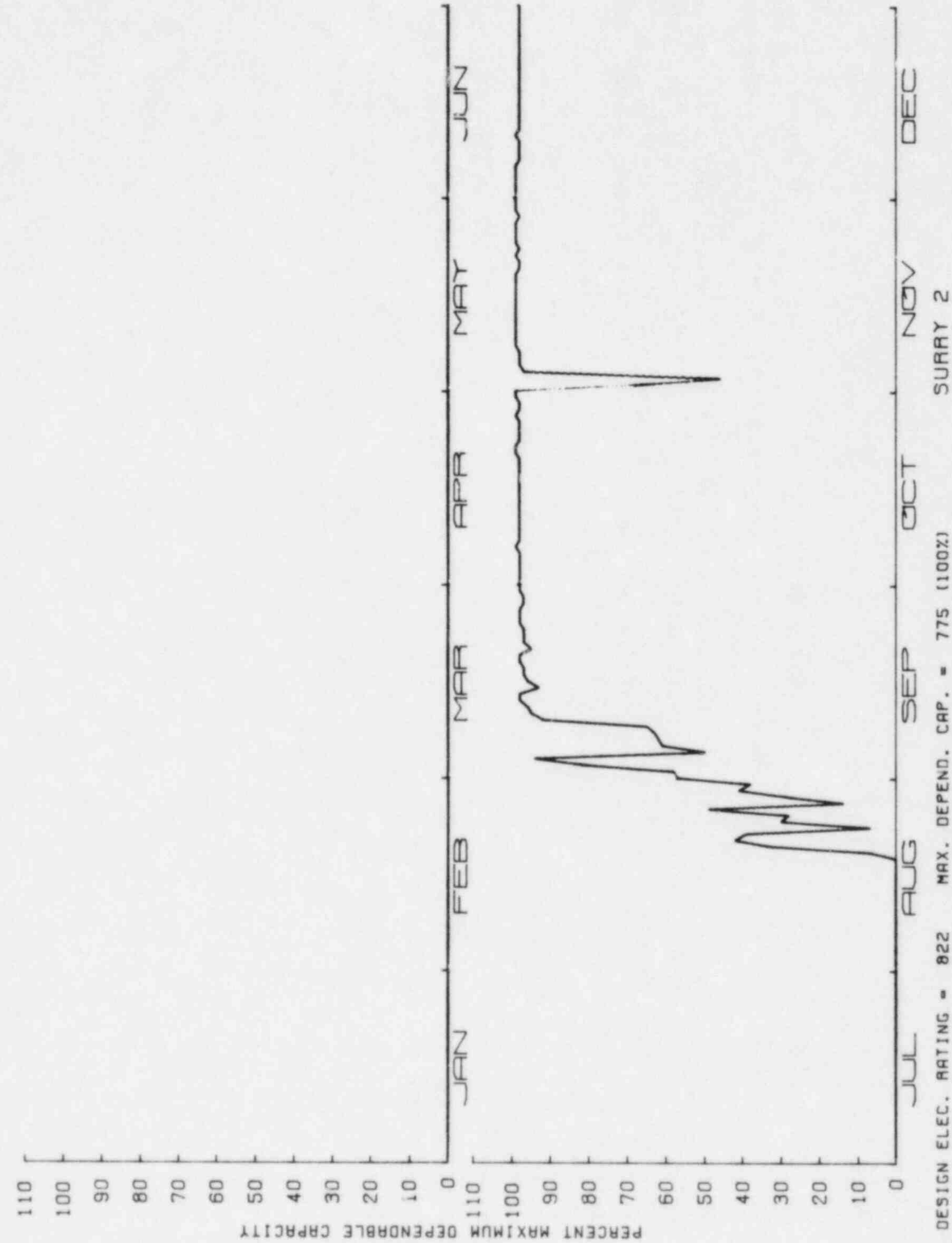
[a]Includes 1,440 h in 1980 from continued 2/04/79 shutdown.

DETAILS OF PLANT OUTAGES FOR SURRY 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	2/04/79 (cont.)	1440.0	S	Refueling and SG replacement.	A	4	Steam and power conversion (HC)	Heat exchangers (steam generator)
1	3/01/80	4102.7	S	Modifications as result of show cause order for seismic analysis and piping restraint modifications per IE Bulletin 79-14.	D	4	Engineered safety features (SA)	Other
2	8/19	15.9	F	SG B high level due to regulating valve failing open. The controller was replaced.	A	3	Steam and power conversion (HH)	Valve operators
3	8/19	1.9	F	SG A low-low level while FW in manual control. Another more experienced operator was placed on the FW control station.	G	3	Steam and power conversion (HH)	Instrumentation and controls
4	8/22	23.6	F	Reactor trip and safety injection on spurious steam header to steam line delta P signal caused by steam from lifting relief valve impinging on steam header pressure transmitter. Stuck NRV's in steam drain line were repaired (LER 80-20).	A	3	Steam and power conversion (HH)	Valves
5	8/24	13.7	F	Loss of auto stop oil pressure through faulty relief valve.	A	3	Steam and power conversion (HA)	Valves
6	8/26	12.1	F	Steam header steam line delta P safety injection signal given due to air trapped in condensate polishing system causing a flow surge.	H	3	Steam and power conversion (HG)	Valves
7	8/27	9.3	F	Power interrupted to train A reactor trip relays by electricians working in instrument racks.	G	3	Instrumentation and controls (IA)	Relays

DETAILS OF PLANT OUTAGES FOR SURRY 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
8	8/30	3.6	F	SG C high level when main FW regulating valve failed open due to instrument technicians working in instrument racks.	G	3	Steam and power conversion (HH)	Circuit closures/interruptions
9	9/01	2.9	F	SG A low-low level as a result of MFWP B trip caused by technicians working in the recirculation flow air circuitry.	G	3	Steam and power conversion (HH)	Instrumentation and controls
10	9/04	3.6	F	SG A low-low level when main FW regulating valve failed shut due to a break in its instrument air line.	A	3	Steam and power conversion (HH)	Instrumentation and controls
11	11/01	14.3	F	SG C high level trip when construction worker grounded one phase of reserve station service transformer C while clipping cement in turbine building basement. Replaced damaged cable (LER 80-35).	G	3	Electric power (EB)	Electrical conductors


PERCENT MAXIMUM DEPENDABLE CAPACITY
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0 10 20 30 40 50 60 70 80 90 100 110
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JUL AUG SEP OCT NOV DEC
DESIGN ELEC. RATING = 822 MAX. DEPEND. CAP. = 775 (100%) SURRY 2

THREE MILE ISLAND 1

I. Summary

Description

Location: Middletown, Pennsylvania
Docket No.: 50-289
Reactor type: PWR
Maximum dependable capacity (MWe-net): 776
Commercial operation: 9/02/74
Years operating experience: 6.5

Performance

Net electrical energy generated (MWh): 0
Unit availability factor (%): 0
Unit capacity factor (using MDC): 0
Unit capacity factor (%) (using design MWe): 0

Outages

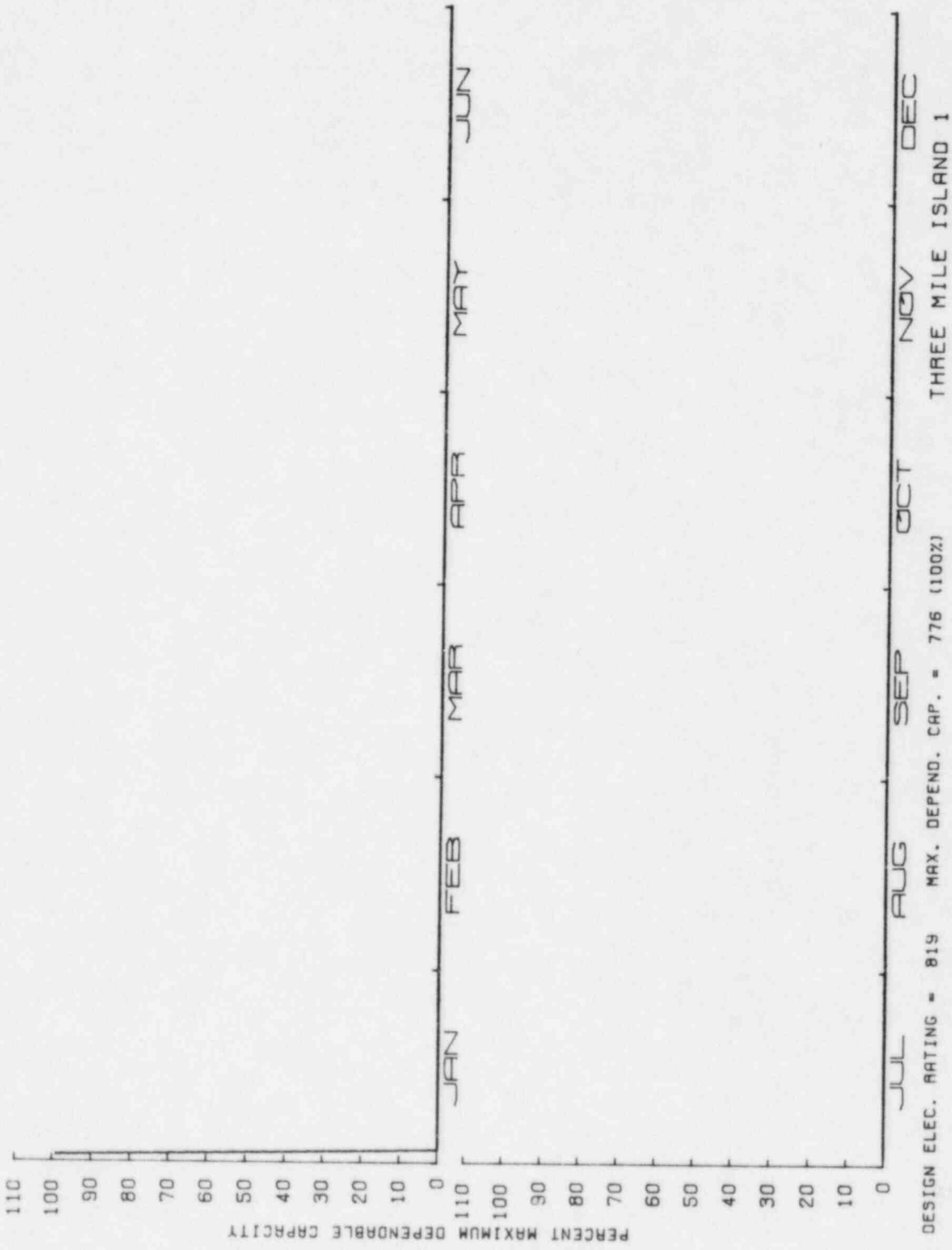
Total No.: 0 (1 continued)
Forced: 0
Scheduled: 0 (1 continued)
Total hours: 8,784 (100%)
Forced: 0 (0%)
Scheduled: 8,784 (100%)

II. Highlights

The plant remains shut down by NRC order pending completion of modifications and other actions related to the TMI-2 accident.

DETAILS OF PLANT OUTAGES FOR THREE MILE ISLAND 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	3/1/79 (cont.)	8784.0	S	Regulatory restraint order.	D	4	System code not applicable	Not applicable


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 819 MAX. DEPEND. CAP. = 776 (100%)
THREE MILE ISLAND 1

THREE MILE ISLAND 2

I. Summary

Description	Performance	Outages
Location: Middletown, Pennsylvania	Net electrical energy generated (MWh): 0	Total No.: 0 (1 continued)
Docket No.: 50-320	Unit availability factor (%): 0	Forced: 0 (1 continued)
Reactor type: PWR	Unit capacity factor (using MDC): 0	Scheduled: 0
Maximum dependable capacity (MWe-net): 0	Unit capacity factor (%) (using design MWe): 0	Total hours: 8,784 (100%)
Commercial operation: 12/30/78		Forced: 8,784 (100%)
Years operating experience: 2.0		Scheduled: 0 (0%)

II. Highlights

On July 20, 1979, the licensee's authority to operate the facility was suspended, and the licensee was required to maintain the facility in the present shutdown cooling mode. Decay heat is being removed through reactor coolant system boundary to the reactor building ambient.

DETAILS OF PLANT OUTAGES FOR THREE MILE ISLAND 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	3/28/79	8784.0	F	MFWP, turbine, and reactor trip on high pressure resulted in a partial uncovering of the core.	A	3	Steam and power conversion (HH)	Pumps

TROJAN

I. Summary

Description

Location: Prescott, Oregon
Docket No.: 50-344
Reactor type: PWR
Maximum dependable capacity (MWe-net): 1,080
Commercial operation: 5/20/76
Years operating experience: 5.0

Performance

Net electrical energy generated (MWh): 6,073,440
Unit availability factor (%): 72.5
Unit capacity factor (using MDC): 64.0
Unit capacity factor (%) (using design MWe): 61.2

Outages

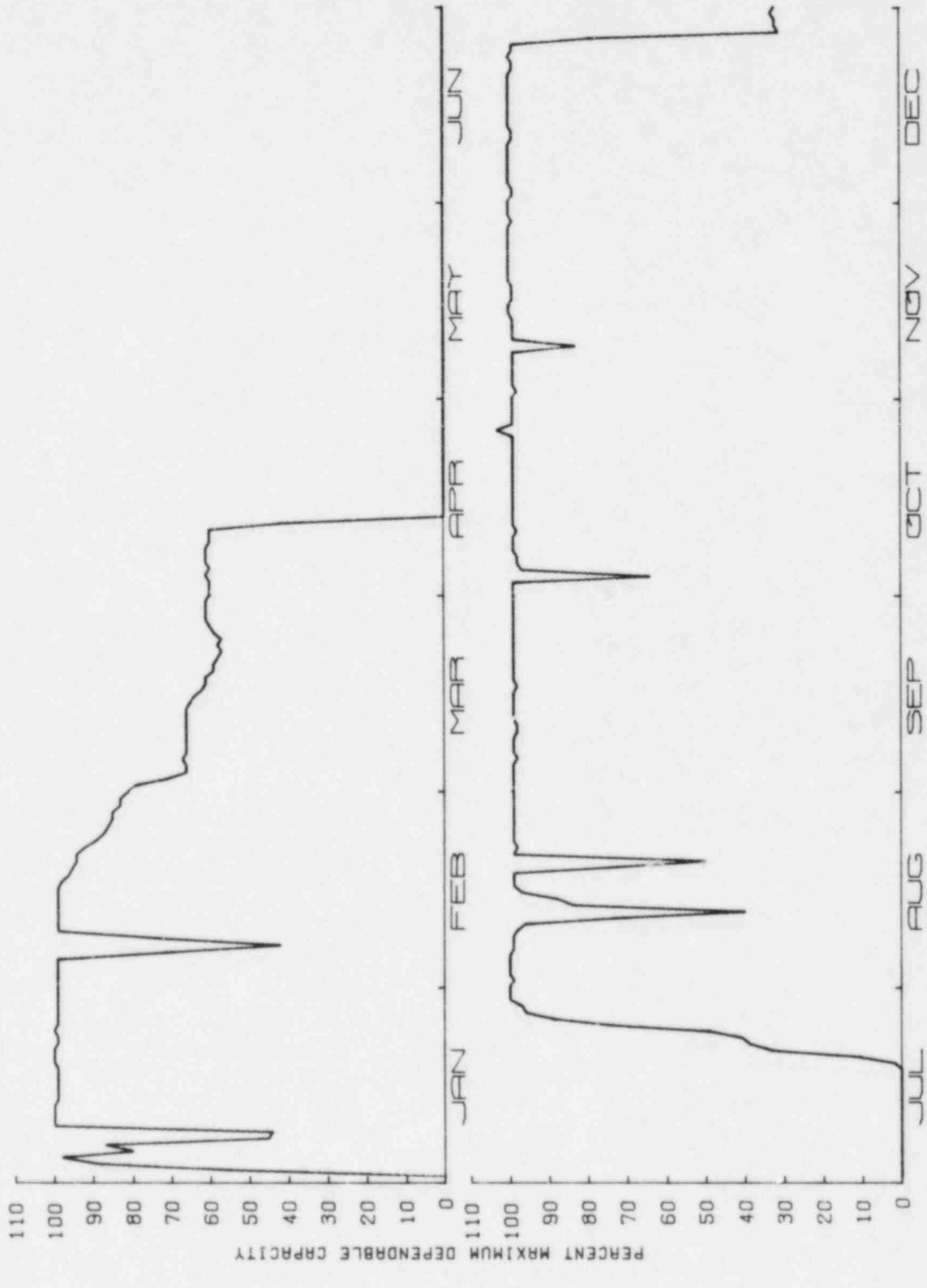
Total No.: 6
Forced: 4
Scheduled: 2
Total hours: 2,418.2 (27.5%)
Forced: 677.3 (7.7%)
Scheduled: 1,740.9 (19.8%)

II. Highlights

Trojan began its end-of-cycle coastdown in February and subsequently shut down for refueling on April 11. Refueling was completed June 18, but the unit remained off-line for modifications to auxiliary building walls. Power operation resumed July 19. Two power reductions were necessary in August to plug leaking condenser tubes. Power was reduced to 40% on December 26 for the remainder of the year because hydroelectric power was available. Trojan's availability was 72.5% for the year.

DETAILS OF PLANT OUTAGES FOR TROJAN

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/06	22.2	F	Broken line in the oil drain from the main bearing pedestal for the turbine generator caused a condenser leak.	A	1	Steam and power conversion (HA)	Pipes, fittings
2	2/05	2.4	F	Inadvertent grounding of a preferred AC bus resulted in a false open indication in a RCP breaker.	G	3	Reactor coolant (CB)	Instrumentation and controls
3a	4/11	1733.6	S	Refueling.	C	1	Reactor (RC)	Fuel elements
3b	6/22	637.2	F	Auxiliary building south-wall discovered not to be connected at top wall to interfacing structures (LER 80-07).	H	4	System code not applicable (ZZ)	Not applicable
4	7/19	7.3	S	Low power physics testing.	B	4	System code not applicable (ZZ)	Not applicable
5	7/20	9.0	F	SG C low-low level during turbine loading at low power while on manual SG level control.	G	3	Steam and power conversion (HH)	Instrumentation and controls
6	10/03	6.5	F	Loss of main generator field occurred when personnel inadvertently removed input lead while setting up plant for power systems stabilizer test. MFW regulating valve B failed to close automatically following plant trip because a manual vent throttle valve did not allow enough air venting. Valve reset and studied for foreign material blockage or malpositioning (LER 80-23). The AFWPs failed to start automatically from low-low SG level following plant trip. Investigation showed that leads to a slave relay had been connected to the wrong terminals (LER 80-20).	G	3	Steam and power conversion (HA)	Electrical conductors



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
Maximum dependable capacity (MWe-net): 646
Commercial operation: 12/14/72
Years operating experience: 8.2

Performance

Net electrical energy generated (MWh): 4,387,391
Unit availability factor (%): 77.6
Unit capacity factor (using MDC): 77.3
Unit capacity factor (%) (using design MWe): 72.1

Outages

Total No.: 21
Forced: 16
Scheduled: 5
Total hours: 1,968.6 (22.4%)[a]
Forced: 409.8 (4.7%)
Scheduled: 1,558.8 (17.7%)[a]

II. Highlights

Even with 21 shutdowns in 1980, Turkey Point 3 experienced a 77.6% availability and 77.3% MDC capacity. Two outages for steam generator inspection and maintenance accounted for nearly 45% of all the outage time. Turbine rotor balancing required four brief outages in February. Loss of power supply to vital instrument buses was responsible for shutdowns on May 6 and 21 and June 10.

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[a]Includes 877.4 h in 1980 from continued 12/01/79 shutdown.

DETAILS OF PLANT OUTAGES FOR TURKEY POINT 3

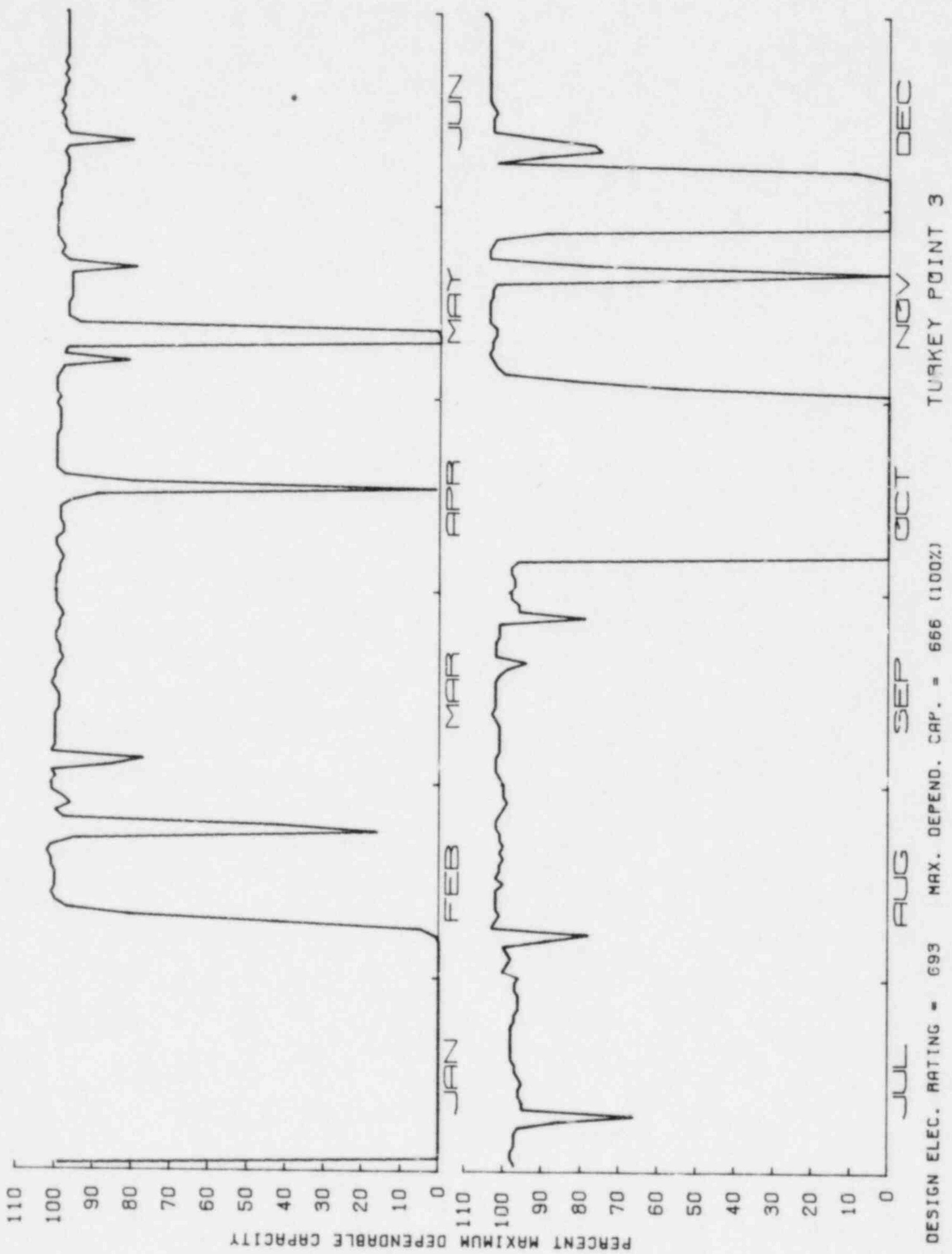
No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	12/01/79	877.4	S	Refueling, maintenance, and inspections. Outage was extended to repair failed mechanical seal on RCP; corrective actions included replacing a sheared lock pin and shaft sleeve on shaft seal assembly No. 1.	C	4	Reactor (RC)	Fuel elements
1	2/16	5.5	F	Turbine rotor balancing.	B	1	Steam and power conversions (HA)	Turbines
2	2/07	17.9	F	Turbine rotor balancing.	B	1	Steam and power conversion (HA)	Turbines
3	2/08	0.8	S	Periodic test on turbine overspeed protection system.	B	1	Steam and power conversion (HA)	Turbines
4	2/08	2.3	F	Loss-of-excitation relay actuated on false signal from voltage regulator.	A	3	Steam and power conversion (HA)	Instrumentation and controls
5	2/20	12.7	S	Turbine rotor balancing.	B	1	Steam and power conversion (HA)	Turbines
6	2/21	11.2	S	Turbine rotor balancing.	B	1	Steam and power conversion (HA)	Turbines
7	3/03	5.8	F	Low oil level in RCD motor.	B	1	Reactor coolant (CB)	Motors
8	4/15	10.8	S	Repair servomotor test valve assembly on turbine control valve.	B	3	Steam and power conversion (HC)	Valves
9	4/16	11.9	F	Repair condenser tube leaks.	A	9	Steam and power conversion (HC)	Heat exchangers (condenser)
10	5/06	2.9	F	Loss of power supply to vital instrument buses 3B and 4A led to SG level protection trip.	A	3	Electric power (ED)	Circuit closures/interrupters

DETAILS OF PLANT OUTAGES FOR TURKEY POINT 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
11	5/09	67.6	F	Modifications to piping supports and restraints inside containment (LER 80-08).	F	1	Reactor coolant (CB)	Shock suppressors and supports
12	5/21	2.5	F	Loss of power supply to vital instrument bus 3A due to malfunction of associated inverter led to SG level protection trip.	A	3	Electric power (ED)	Batteries and chargers
13	6/10	3.0	F	Loss of power supply to vital instrument bus No. 3A caused SG No. 3B level protection trips; replaced SCRs in inverter No. A.	A	3	Electric power (ED)	Generators
14	7/07	7.0	F	Low oil level in RCP motor No. 3B.	B	1	Reactor coolant (CB)	Motors
15	8/06	6.1	F	Low oil level in RCP motor No. 3C.	B	1	Reactor coolant (CB)	Motors
16	9/26	2.2	F	RPS trip due to spurious signal from nuclear instrumentation system channel N-41 while channel N-43 was in trip mode.	A	3	Instrumentation and controls (IA)	Electrical conductors
17a	10/06	645.3	S	SG tube inspection and maintenance.	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)
17b	11/02	0.6	S	Turbine overspeed trip test.	B	4	Steam and power conversion (HA)	Mechanical function units
18	11/19	2.4	F	SG 3A level protection system trip caused by sudden closure at FW control valve 3A. Repaired loose electrical connection.	A	3	Steam and power conversion (HH)	Valves

DETAILS OF PLANT OUTAGES FOR TURKEY POINT 3 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
19	11/19	19.7	F	SG 3C level protection system trip while reducing load. Outage was extended to repair failed weld on line attached to bypass FW line to SG 3B (LER 80-24).	A	3	Steam and power conversion (HH)	Valves
20	11/20	16.9	F	FW flow to SG 3A could not be transferred from the bypass FW control valve to the main FW control valve. Inadequate flow through FW control valve 3A was due to broken valve stem.	A	9	Steam and power conversion (HH)	Valves
21	11/26	236.1	F	Locate and repair (by welding) leaking tube plug in SG 3B.	B	1	Steam and power conversion (HB)	Heat exchangers (steam generator)


PERCENT MAXIMUM DEPENDABLE CAPACITY
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DESIGN ELEC. RATING = 693 MAX. DEPEND. CAP. = 666 (100%)
TURKEY POINT 3

TURKEY POINT 4

I. Summary

Description	Performance	Outages
Location: Florida City, Florida	Net electrical energy generated (MWh): 3,854,024	Total No.: 12
Docket No.: 50-251	Unit availability factor (%): 69.5	Forced: 6
Reactor type: PWR	Unit capacity factor (using MDC): 67.9	Scheduled: 6
Maximum dependable capacity (MWe-net): 646	Unit capacity factor (%) (using design MWe): 63.3	Total hours: 2,687.4 (30.6%)
Commercial operation: 9/07/73		Forced: 17.9 (0.2%)
Years operating experience: 7.5		Scheduled: 2,669.5 (30.4%)

II. Highlights

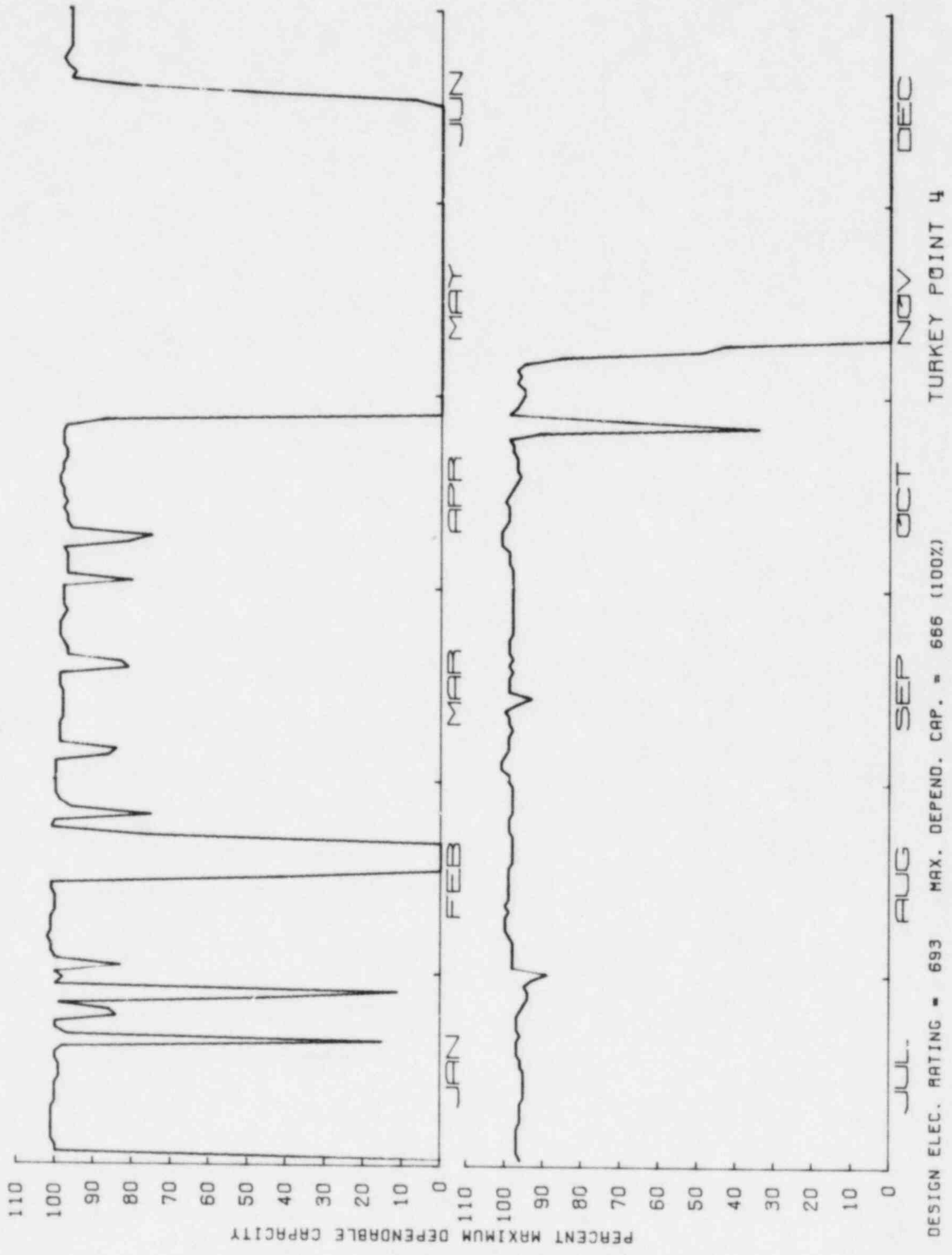
A single outage beginning April 26 for condenser tube leak repair accounted for 86% of the downtime in 1980 at Turkey Point 4, excluding the refueling outage which lasted from November 8 through December. Power reductions because of condenser tube leaks occurred on January 23, February 1 and 23, March 18, and April 1.

DETAILS OF PLANT OUTAGES FOR TURKEY POINT 4

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/19	15.4	S	Integrated safeguards surveillance test.	B	1	Instrumentation and controls (IB)	Instrumentation and controls
2	1/19	2.8	F	Trip by the SG level protection system due to a transient condition during startup.	A	3	Steam and power conversion (HB)	Heat exchangers (steam generator)
3	1/26	29.4	S	Repair turbine control oil system, replaced turbine governor impeller seal sleeve.	A	1	Steam and power conversion (HA)	Mechanical function units
4	2/14	134.8	S	Repair turbine control oil system; replaced turbine governor impeller shaft sleeve.	A	3	Steam and power conversion (HA)	Mechanical function units
5	2/23	1.4	F	Leak at a fitting on a condenser vacuum sensing line caused a false signal in the low vacuum trip device.	A	3	Steam and power conversion (HA)	Mechanical function units
6	3/04	4.2	F	High oil level in RCP motor.	B	1	Reactor coolant (CB)	Motors
7	4/07	3.4	F	SG level protection trip due to failure of condensate pump motor; repaired failed electrical insulation.	A	3	Steam and power conversion (HH)	Motors
8	4/26	1214.1	S	Repair condenser tube leak.	B	1	Steam and power conversion (HC)	Heat exchangers (condenser)
9	6/17	1.8	S	Turbine overspeed trip test. Reactor tripped by SG 4C level protection system.	B	3	Steam and power conversion (HB)	Instrumentation and controls

DETAILS OF PLANT OUTAGES FOR TURKEY POINT 4 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
10	6/17	2.8	F	Unit tripped by SG 4C level protection system.	A	3	Steam and power conversion (HA)	Turbines
11	10/25	3.3	F	RPS trip due to loss of power supply to vital instrument AC bus No. 4B.	A	3	Electric power (ED)	Generators
12	11/08	1274.0	S	Refueling, maintenance, and inspections.	C	1	Reactor (RC)	Fuel elements


PERCENT MAXIMUM DEPENDABLE CAPACITY
110 100 90 80 70 60 50 40 30 20 10 0
JAN FEB MAR APR MAY JUN
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DESIGN ELEC. RATING = 693 MAX. DEPEND. CAP. = 666 (100%)
TURKEY POINT 4

VERMONT YANKEE 1

I. Summary

Description

Location: Vernon, Vermont
Docket No.: 50-271
Reactor type: BWR
Maximum dependable capacity (MWe-net): 504
Commercial operation: 11/30/72
Years operating experience: 8.2

Performance

Net electrical energy generated (MWh): 2,979,214
Unit availability factor (%): 71.4
Unit capacity factor (using MDC): 67.3
Unit capacity factor (%) (using design MWe): 66.0

Outages

Total No.: 8
Forced: 4
Scheduled: 4
Total hours: 2,510.2 (28.6%)
Forced: 946.8 (10.8%)
Scheduled: 1,563.4 (17.8%)

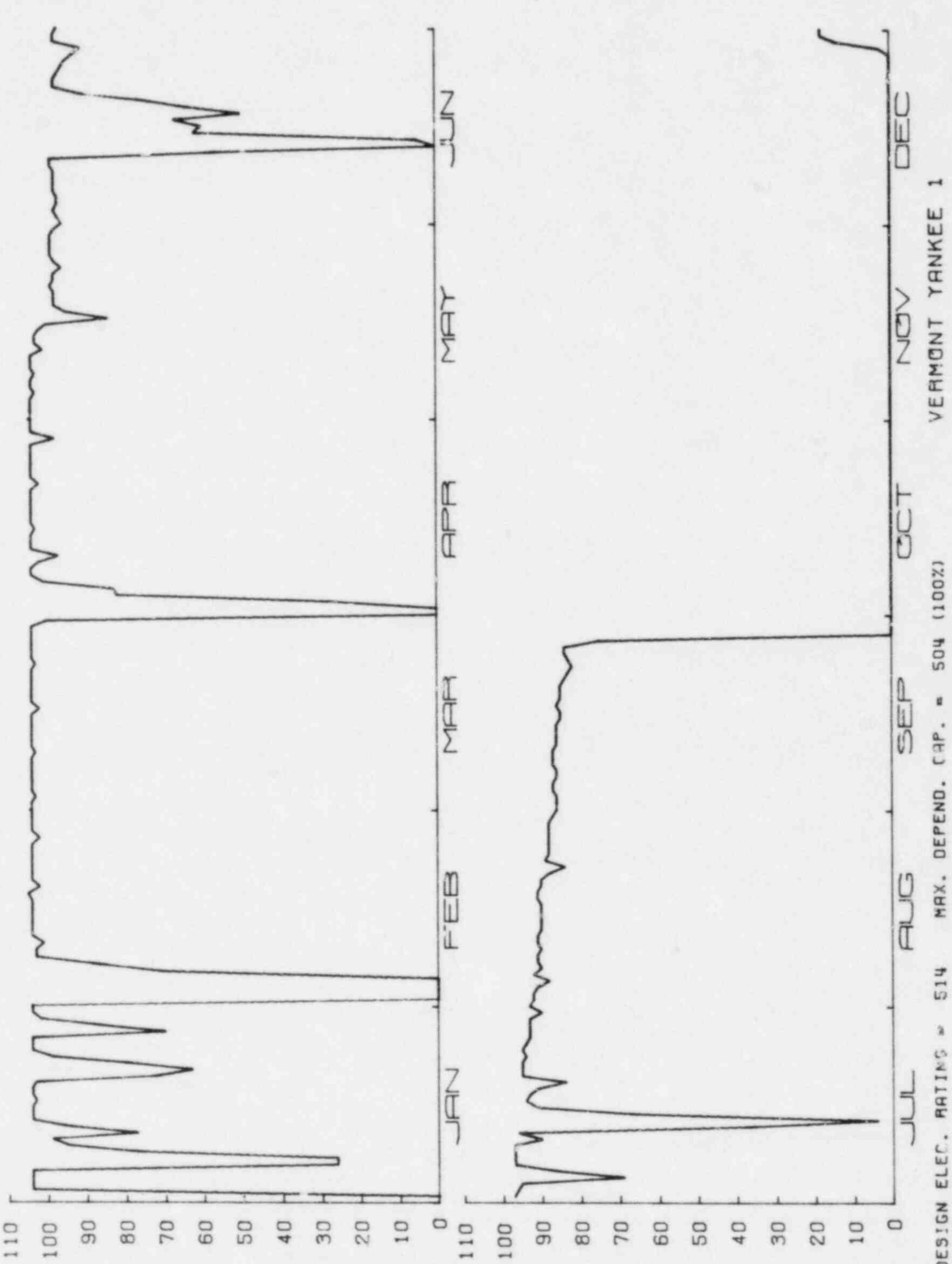
II. Highlights

Vermont Yankee attained 71.4% availability in 1980. TMI-related modifications took 4 d beginning January 31. Refueling began September 26, and the outage was continued to replace cracked pipes and a leaking RHR value.

DETAILS OF PLANT OUTAGES FOR VERMONT YANKEE 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1	1/05	27.0	F	Excessive drywell leakage rate; RHR8 valve 1B was repacked.	A	1	Reactor coolant (CF)	Valves
2	1/13	102.4	S	Modifications per NUREG-0578.	D	1	Other (XX)	Other
3	3/30	59.0	S	Leak in main steam valve which isolates the steam supply to the main turbine steam seal regulator.	B	1	Reactor coolant (CD)	Valves
4	6/11	51.5	F	FW check valve seal failure (LER 80-18).	A	1	Reactor coolant (CH)	Valves
5	6/17	8.5	F	High level in main turbine moisture separator drain tank due to faulty drain valve.				
6a	7/12	30.0	F	Repair bypass valve on B recirculation pump discharge.	B	2	Reactor coolant (CB)	Valves
6b	7/12	*a*	S	Scram testing of scram discharge volume per IE Bulletin No. 80-17.	D	4	Reactor (RB)	Control rods
7a	9/26	1399.5	S	Refueling.	C	1	Reactor (RC)	Fuel elements
7b	11/24	721.9	F	Replacement of cracked pipe and sweep-o-let (LER 80-37).	A	4	Reactor coolant (CG)	Pipes, fittings
7c	12/24	1.0	F	FW instrument failure and turbine-generator mechanical pressure regulator malfunction.	A	9	Reactor coolant (CH)	Instrumentation and controls
7d	12/25	106.9	F	Repair RHR valve leakage.	A	4	Reactor coolant (CF)	Valves
8	12/29	2.5	S	Turbine overspeed testing.	B	1	Steam and power conversion (HA)	Turbines

[a]Total hours for 6b are included in 6a.


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

YANKEE-ROWE

I. Summary

Description

Location: Rowe, Massachusetts
Docket No.: 50-029
Reactor type: PWR
Maximum dependable capacity (MWe-net): 175
Commercial operation: 7/61
Years operating experience: 20.1

Performance

Net electrical energy generated (MWh): 291,967
Unit availability factor (%): 22.0
Unit capacity factor (using MDC): 19.0
Unit capacity factor (%) (using design MWe): 19.0

Outages

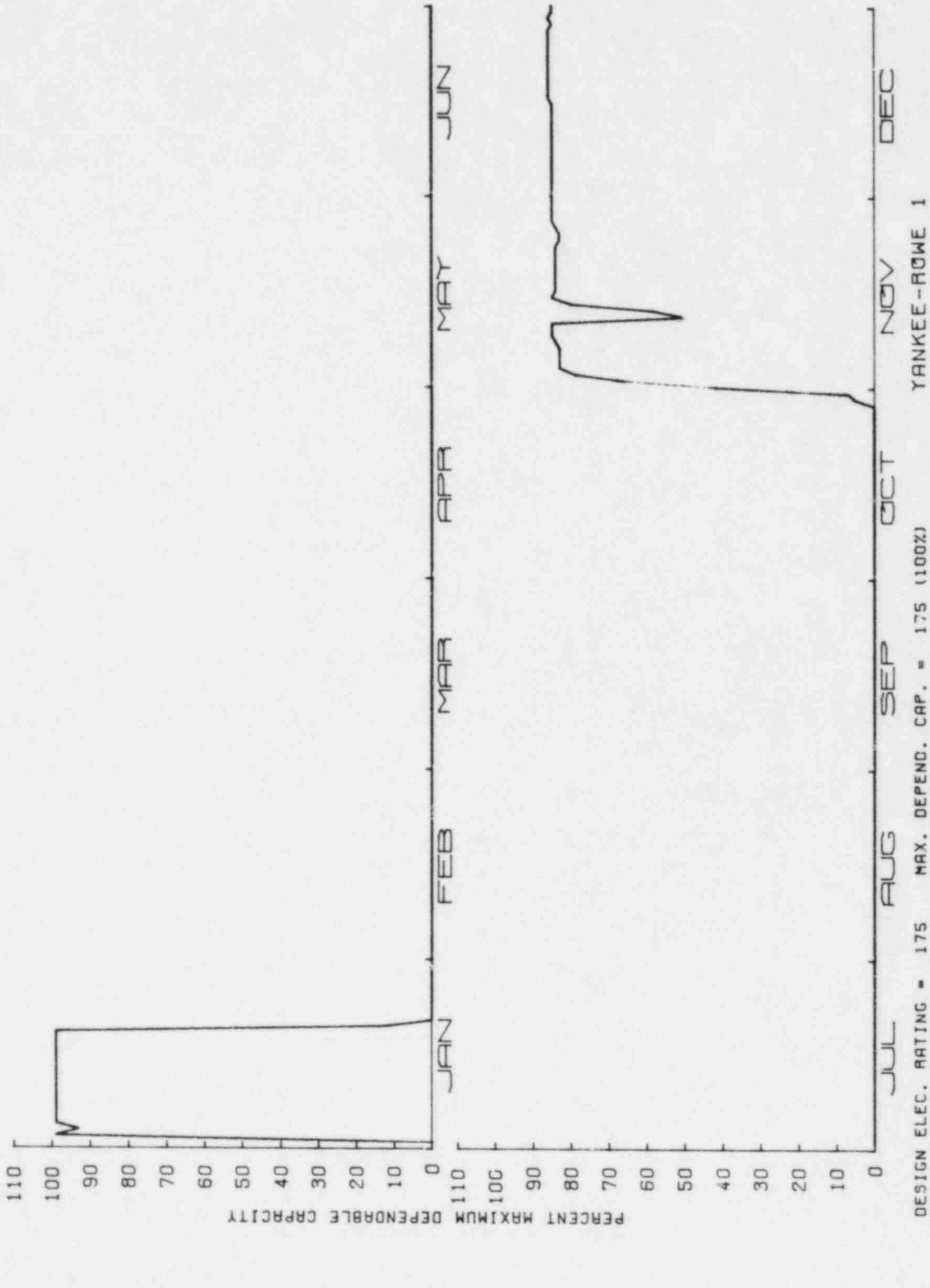
Total No.: 3
Forced: 2
Scheduled: 1
Total hours: 6,849.9 (77.0%)
Forced: 424.3 (4.8%)
Scheduled: 6,425.6 (73.2%)

II. Highlights

Yankee-Rowe was down for TMI-related modifications when a turbine rotor failure required a 37-week shutdown beginning February 12.

DETAILS OF PLANT OUTAGES FOR YANKEE ROWE

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a	1/19	424.3	S	Install voltage regulators and make TMI-related changes.	D	1	Other (XX)	Other
1b	2/05	1680	F	Leak in RCP flange.	A	4	Reactor coolant (CB)	Pipes, fittings
1c	2/12	6223.8	F	Turbine rotor failure.	A	4	Steam and power conversion (HA)	Turbines
2	10/29	24.0	F	Turbine overspeed test. Broken throttle poppet valve was found and repaired (LERs 80-19.20).	B	3	Steam and power conversion (HB)	Valves
3	11/11	9.8	F	Ground to offsite 115-kV line resulting in loss of Z-126 line (LER 80-21).	H	3	Electric power (EA)	Circuit closures/ interrupters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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
Maximum dependable capacity (MWe-net): 1,040
Commercial operation: 12/31/73
Years operating experience: 7.5

Performance

Net electrical energy generated (MWh): 6,514,861
Unit availability factor (%): 81.6
Unit capacity factor (using MDC): 71.3
Unit capacity factor (%) (using design MWe): 71.3

Outages

Total No.: 17
Forced: 16
Scheduled: 1
Total hours: 1,615.6 (18.4%)
Forced: 1,135.6 (12.9%)
Scheduled: 480.0 (5.5%)

II. Highlights

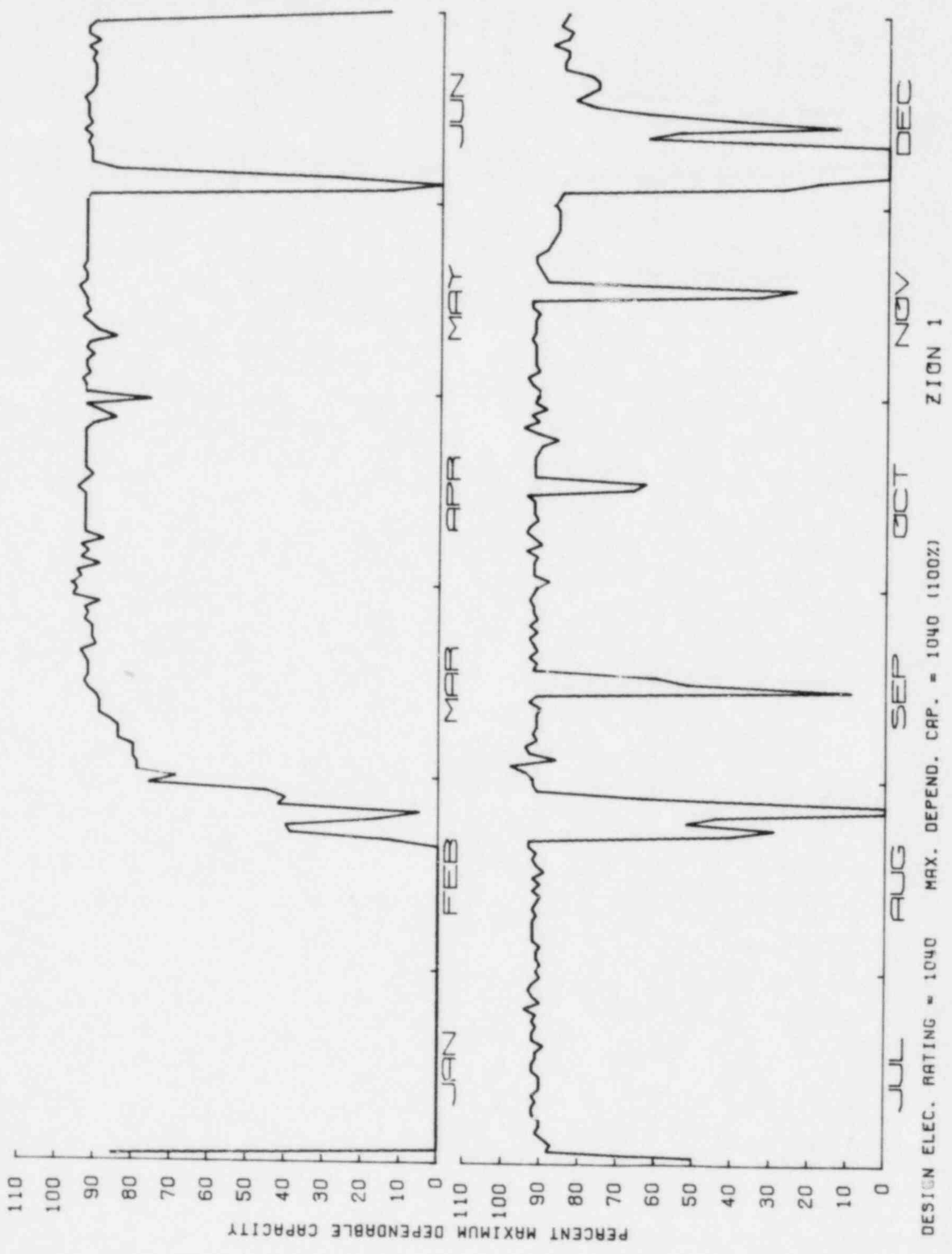
Zion 1 began the year down for feedwater nozzle repairs following refueling. The unit remained down until February 18 for charging pump and reactor coolant pump seal replacement and for charging pump isolation valve repairs. A generator hydrogen cooler leak was the only other lengthy outage, accounting for 5.5 d beginning December 4. The unit availability was 81.6%.

DETAILS OF PLANT OUTAGES FOR ZION 1

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
1a	1/01	480.0	S	FW nozzle repair per IE Bulletin 79-13.	D	4	Steam and power conversion (HH)	Pipes
1b	1/21	72.0	F	Replace 1A charging pump.	A	4	Engineered safety features (SF)	Pumps
1c	1/24	3360.0	F	Replace 1B RCP seals.	A	4	Reactor coolant (CB)	Pumps
1d	2/8	275.5	F	Repair of charging pump isolation valves.	A	4	Auxiliary process (PC)	Valves
2	2/22	8.5	F	Loss of 1B MFWP.	A	3	Steam and power conversion (HH)	Pumps
3	2/22	15.2	F	Governor valve opened.	A	3	Steam and power conversion (HA)	Valves
4	2/23	3.9	F	Repair of 1A MSIV DC solenoid.	A	1	Steam and power conversion (HB)	Relays
5	6/02	0.1	F	SG snubber inoperable.	A	1	Steam and power conversion (HB)	Shock suppressors and supports
6a	6/02	3.0	F	Instrument malfunction.	A	3	Instrumentation and controls (ID)	Instrumentation and controls
6b	6/02	49.8	F	SG snubber inoperable.	A	4	Steam and power conversion (HB)	Shock suppressors and supports
7	6/29	22.4	F	Repair component cooling meter leak on RCP 1B.	A	1	Reactor coolant (CB)	Pumps
8	8/22	19.1	F	Low-low level in SG 1B.	B	3	Steam and power conversion (HH)	Heat exchangers (steam generator)

DETAILS OF PLANT OUTAGES FOR ZION 1 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
9	8/26	44.3	F	Generator off-line due to voltage regulation problem.	A	1	Electric power (EB)	Generators
10	8/27	11.7	F	1B MFWP problem.	A	3	Steam and power conversion (HH)	Pumps
11	9/14	15.6	F	Electrical ground repair LCV-459.	A	2	Electric power (ED)	Electrical conductors
12	9/15	12.6	F	Accidental turbine trip caused by contractor jarring turbine auto-stop trip relay housing.	F	3	Steam and power conversion (HA)	Relays
13	11/16	32.9	F	Surveillance testing.	G	3	Instrumentation and controls (IA)	Instrumentation and controls
14	12/03	19.3	F	SG 1A low-low level.	G	3	Steam and power conversion (HH)	Instrumentation and controls
15	12/04	132.3	F	Generator hydrogen cooler leaks.	A	3	Steam and power conversion (HA)	Heat exchangers
16	12/10	15.9	F	SG low-low level on loop D.	A	3	Steam and power conversion (HH)	Not applicable
17	12/12	21.5	F	Loss of oil pump caused FW pump to trip and SG low-low level.	A	3	Steam and power conversion (HH)	Pumps


PERCENT MAXIMUM DEPENDABLE CAPACITY
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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
Maximum dependable capacity (MWe-net): 1,040
Commercial operation: 9/17/74
Years operating experience: 7.0

Performance

Net electrical energy generated (MWh): 5,278,833
Unit availability factor (%): 66.7
Unit capacity factor (using MDC): 57.8
Unit capacity factor (%) (using design MWe): 57.8

Outages

Total No.: 20
Forced: 19
Scheduled: 1
Total hours: 2,922.1 (33.3%)[a]
Forced: 677.6 (7.7%)
Scheduled: 2,244.5 (25.6%)[a]

II. Highlights

Zion 2 operated with high availability after it came back on-line January 20 from a feedwater nozzle repair outage until May 2 when the unit shut down for refueling.

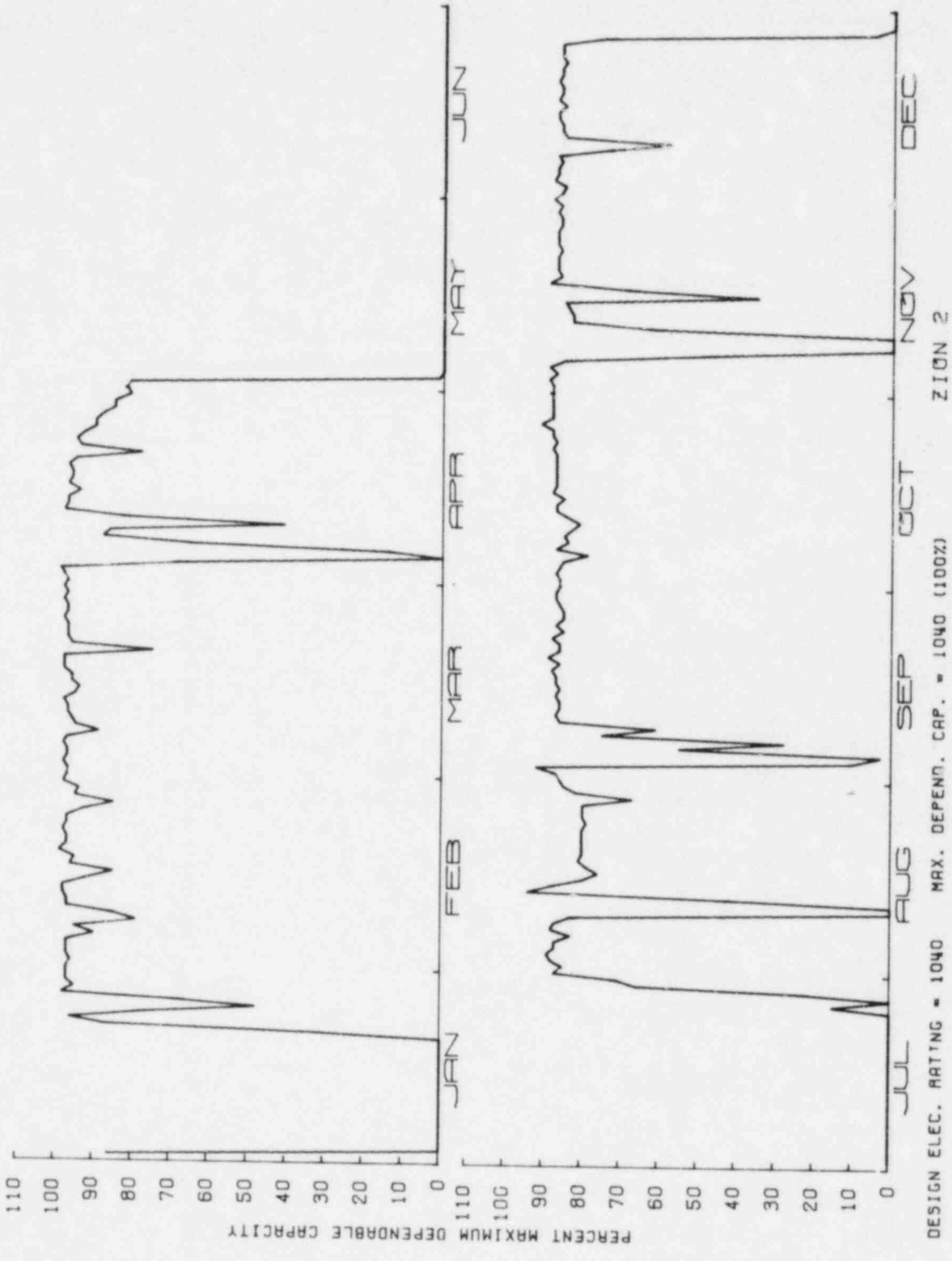
[a]Includes 456.1 h in 1980 from continued 10/27/79 shutdown.

DETAILS OF PLANT OUTAGES FOR ZION 2

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
	10/27/79 (cont.)	456.1	S	FW nozzle repair per IE Bulletin 73-13.	D	4	Steam and power conversion (HH)	Pipes, fittings
1	1/24	15.9	F	Reactor trip on failure of both rod drive MG sets.	A	3	Reactor (RB)	Generators
2	1/24	3.9	F	Steam flow/feed flow mismatch and low-low SG level.	G	3	Steam and power conversion (HC)	Heat exchangers (steam generator)
3	4/03	33.0	F	Reactor trip due to lightning.	H	3	Electric power (EA)	Not applicable
4	4/05	3.7	F	SG low-low level.	A	3	Steam and power conversion (HH)	Not applicable
5	4/05	9.1	F	SG low-low level.	A	3	Steam and power conversion (HH)	Not applicable
6	5/02	1788.4	S	Refueling.	C	1	Reactor (RC)	Fuel elements
7	7/15	11.8	F	SG B low-low level.	A	3	Steam and power conversion (HH)	Heat exchangers
8a	7/16	22.2	F	Manual scram after partial scram from lightning strike.	H	2	Electric power (EA)	Not applicable
8b	7/17	211.2	F	RCP seal repairs.	B	4	Reactor coolant (CB)	Pumps
9	7/25	10.5	F	SG 2B steam flow/feed flow mismatch.	A	3	Steam and power conversion (HH)	Heat exchangers (steam generator)
10	7/26	32.6	F	Steam flow/feed flow mismatch due to EHC problems.	A	3	Steam and power conversion (HA)	Mechanical function units
11	8/09	15.9	F	Repairs on stator water cooling pumps.	B	1	Auxiliary water (WA)	Generators

DETAILS OF PLANT OUTAGES FOR ZION 2 (continued)

No.	Date (1980)	Duration (h)	Type	Description	Cause	Shutdown method	System involved	Component involved
12	8/10	5.1	F	Low level in SG 2D and steam flow/feed flow mismatch due to steam spike while starting the B FW pump.	A	3	Steam and power conversion (HH)	Pumps
13	8/10	36.4	F	Problem with the EHC system caused a generator reverse power trip and led to a low SG level and steam flow/feed flow mismatch trip.	A	3	Steam and power conversion (HA)	Mechanical function units
14	9/03	38.8	F	Low-low SG level due to loss of 2B feedwater pump.	A	3	Steam and power conversion (HH)	Pumps
15	9/05	10.9	F	High SG level due to 2C FW pump flow oscillation.	A	3	Steam and power conversion (HH)	Pumps
16	11/06	82.7	F	Reactor trip and generator trip. Cause unknown.	A	3	System code not applicable (ZZ)	Not applicable
17	11/15	12.4	F	Nuclear rate trip on N43.	A	3	Reactor (RB)	Instrumentation and controls
18	11/15	2.7	F	SG low level during startup.	H	3	Steam and power conversion (HH)	Not applicable
19	12/08	7.5	F	Rod control malfunction (80-32).	A	2	Reactor (RB)	Instrumentation and controls
20	12/27	111.3	F	Improper chemistry caused by condenser tube leak.	A	2	Steam and power conversion (HH)	Heat exchangers (condenser)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DESIGN ELEC. RATING = 1040   MAX. DEPEND. CAP. = 1040 (100%)
ZION 2

Appendix C

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:

For All Licensees

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.

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION

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G. T. Mays, J. A. Haried, C. Kukielaka, R. D. Seagren

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15. SUPPLEMENTARY NOTES

16. ABSTRACT (200 words or less)

This report is the seventh 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 1980 data from 67 plants - 24 boiling-water-reactor plants, 42 pressurized-water-reactor plants, and 1 high-temperature gas-cooled reactor plant.

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