AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE October 8, 1979

COMPLETED BY Jan Stotz/Carl Berge

TELEPHONE 419-259-5000, Ext. 243

MONTH September, 1979

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	877	17	880
2	869	18	470
3	860	19	472
4	877	20	862
5	871	21	885
6	876	22	883
7	785	23	848
8	0	24	881
9	0	25	879
10	174	26	767
11	441	27	0
12	423	28	131
13	655	29	702
14	826	30	876
15	882	31	---
16	882		

INSTRUCTIONS

On this format, list the average daily unit power level in MWe-Net for each day in the reporting month. Compute to the nearest whole megawatt.

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OPERATING DATA REPORT

DOCKET NO. 50-346
DATE October 8, 1979
COMPLETED BY J. Stotz/C. Berger
TELEPHONE 419-259-5000, Ext. 243

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: September, 1979
3. Licensed Thermal Power (MWt): 2772
4. Nameplate Rating (Gross MWe): 925
5. Design Electrical Rating (Net MWe): 906
6. Maximum Dependable Capacity (Gross MWe): to be determined
7. Maximum Dependable Capacity (Net MWe):
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report. Give Reasons:

Notes

9. Power Level To Which Restricted. If Any (Net MWe): None
10. Reasons For Restrictions. If Any:

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	720	6,551	18,316
12. Number Of Hours Reactor Was Critical	634.95	3,624.4	10,256.2
13. Reactor Reserve Shutdown Hours	85.05	1,943.3	2,733.6
14. Hours Generator On-Line	604.1	3,503	9,236.2
15. Unit Reserve Shutdown Hours	0	1,728.2	1,728.2
16. Gross Thermal Energy Generated (MWH)	1,549,866	8,681,528	18,869,098
17. Gross Electrical Energy Generated (MWH)	503,791	2,881,188	6,264,943
18. Net Electrical Energy Generated (MWH)	477,780	2,714,007	5,755,467
19. Unit Service Factor	83.9	53.5	50.4
20. Unit Availability Factor	83.9	79.9	59.9
21. Unit Capacity Factor (Using MDC Net)	to be determined		
22. Unit Capacity Factor (Using DER Net)	73.2	45.7	34.7
23. Unit Forced Outage Rate	8.8	[illegible].9	19.0

24. Shutdowns Scheduled Over Next 6 Months (Type. Date. and Duration of Each):
Refueling outage to start March 15, 1980

25. If Shut Down At End Of Report Period. Estimated Date of Startup:
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE October 8, 1979
COMPLETED BY Jan Stotz/Carl Berger
TELEPHONE 419-259-5000, Ext. 243

REPORT MONTH September, 1979

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
11	79 09 07	S	58.6	B	1	NA	NA	NA	Maintenance outage to find steam leak in containment.
12	79 09 18	F	17.2	A	3	NA	NA	NA	Sticking pump pressure controller on No. 2 electro-hydraulic control pump. Controller to be disassembled and cleaned at earliest convenience.
13	79 09 23	S	0.0	B	4	NA	NA	NA	Reactor power reduced to 75% to allow maintenance to adjust blowdown on SP17A2.
14	79 09 26	F	41.1	A	3	NA	NA	NA	Faulty capacitor on turbine throttle pressure transmitter power supply.

1
F: Forced
S: Scheduled

2
Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

3
Method:
1-Manual
2-Manual Scram.
3-Automatic Scram.
4-Other (Explain)

4
Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

5
Exhibit I - Same Source

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OPERATIONAL SUMMARY FOR SEPTEMBER, 1979

9/1/79 - 9/2/79	The reactor power level was maintained between 99 and 100 percent with the generator gross load at 915 ± 10 MWe.
9/3/79	(Labor Day) Per Load Dispatcher, began reducing power at 0530 hours to 70 percent. Began increasing power at 0850 hours and reached 99 percent at 1230 hours.
9/4/79 - 9/6/79	The reactor power was maintained between 99 and 100 percent with generator gross load at 915 ± 10 MWe.
9/7/79 - 9/14/79	Maintenance outage to find source of steam leak in containment. Reactor subcritical by 0220 hours on September 8, 1979. Returned to critical at 2105 hours on September 9, 1979 and in Mode 1 at 2317 hours. Power was increased to 50% on September 10, 1979 and held until September 13, 1979. Power reached 99 percent on September 14, 1979.
9/15/79 - 9/17/79	Reactor power level maintained between 99 and 100 percent with generation gross load at 915 ± 10 MWe.
9/18/79 - 9/19/79	Turbine trip followed by reactor trip at 1243 hours on September 18, 1979 during electro-hydraulic control pump test. A drop in electro-hydraulic control hydraulic pressure occurred when the pumps were switched over. The Anticipatory Reactor Trip System tripped the reactor. Began recovery and increased power to 99 percent by 2300 hours on September 19, 1979.
9/20/79 - 9/22/79	The reactor power level was maintained between 99 and 100 percent with generation gross load at 915 ± 10 MWe.
9/23/79	Reactor power reduced to 75 percent at 1400 hours to allow maintenance to reset the blowdown on SP17A2. Returned to 99 percent power by 2000 hours.
9/24/79 - 9/25/79	The reactor power level was maintained between 99 and 100 percent with generator gross load at 915 ± 10 MWe.
9/26/79 - 9/29/79	Turbine/reactor trip at 2056 hours on September 26, 1979 caused by a turbine throttle pressure transmitter power supply causing the turbine valves to partially close. Began increasing power on September 28, 1979 and returned to 99 percent power on September 29, 1979.

9/30/79 The reactor power level was maintained between 99 and 100 percent with generator gross load at 915 ± 10 MWe.

REFUELING INFORMATION

DATE: September, 1979

1. Name of facility: Davis-Besse Nuclear Power Station Unit 1

2. Scheduled date for next refueling shutdown: March, 1980

3. Scheduled date for restart following refueling: May, 1980

4. Will refueling or resumption of operation thereafter require a technical specification change or other license amendment? If answer is yes, what, in general, will these be? If answer is no, has the reload fuel design and core configuration been reviewed by your Plant Safety Review Committee to determine whether any unreviewed safety questions are associated with the core reload (Ref. 10 CFR Section 50.59)?

   Yes, see attached.

5. Scheduled date(s) for submitting proposed licensing action and supporting information. December, 1979

6. Important licensing considerations associated with refueling, e.g., new or different fuel design or supplier, unreviewed design or performance analysis methods, significant changes in fuel design, new operating procedures.

   The spent fuel pool capacity expansion program was approved by the NRC in Amendment 19 to the operating license received August 1, 1979.

7. The number of fuel assemblies (a) in the core and (b) in the spent fuel storage pool.

   (a) 177 (b) 0 (zero)

8. The present licensed spent fuel pool storage capacity and the size of any increase in licensed storage capacity that has been requested or is planned, in number of fuel assemblies.

   Present 260 Increase size by 475 (735 total)

9. The projected date of the last refueling that can be discharged to the spent fuel pool assuming the present licensed capacity.

   Date 1989 (assuming ability to unload the entire core into the spent fuel pool is maintained and the unit goes to an 18 month refueling cycle)

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4. The following Technical Specifications (Part A) will require revision:

2.1.1 & 2.1.2 - Reactor Core Safety Limits (and Bases)
2.2.1 - Reactor Protection System Instrumentation Setpoints (and Bases)
3.1.3.6 - Regulating Rod Insertion Limits
3.1.3.7 - Rod Program
3.2.1 - Axial Power Imbalance (and Bases)

The following Technical Specifications (Part A) may also require revision:

3.1.2.8 & 3.1.2.9 - Borated Water Sources (and Bases)
3.2.4 - Quadrant Power Tilt (and Bases)
3.2.5 - DNB Parameters (and Bases)

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FACILITY CHANGE REQUESTS COMPLETED DURING SEPTEMBER, 1979

FCR NO: 79-188

SYSTEM: Auxiliary Feedwater System

COMPONENT: Auxiliary feedwater flow instrumentation

CHANGE, TEST, OR EXPERIMENT: On June 28, 1979, the physical work for FCR 79-188 was completed. This FCR installed instrumentation to provide indication in the control room of auxiliary feedwater flow to each steam generator. This addition was made under the guidance of the unit architect/engineer, Bechtel Company. All affected drawings were revised by Bechtel to reflect the addition of this instrumentation.

REASON FOR THE FCR: This instrumentation was installed in order to comply with a commitment documented in the May 16, 1979 Shutdown Order.

SAFETY EVALUATION: The addition of the auxiliary feedwater flow indication does not affect the function of the auxiliary feedwater system in any way. Ultrasonic flow transmitters were utilized to sense the flowrate through the auxiliary feedwater piping to each steam generator. The flow transmitters are clamped to the outside of the piping and do not penetrate the pressure boundary. It has been verified by the architect/engineer that the small additional weight of the transmitters does not affect the seismic qualification of the piping.

FACILITY CHANGE REQUESTS COMPLETED DURING SEPTEMBER, 1979

FCR NO: 79-251

SYSTEM: Decay Heat Removal

COMPONENT: Hydraulic snubber CCA-4-H3

CHANGE, TEST, OR EXPERIMENT: On June 27, 1979, under Maintenance Work Order 79-2282, hydraulic snubber CCA-4-H3, which was a remote reservoir snubber, was replaced with a local reservoir snubber.

REASON FOR THE FCR: The change to a local reservoir snubber allows for easier removal and replacement of the snubber. It was also found that the original snubber had non-ethylene propylene seal materials (see Licensee Event Report NP-32-79-07).

SAFETY EVALUATION: The hydraulic shock and sway suppressor being installed is the type called for on existing project drawings. The new snubber has ethylene propylene seals and is in accordance with technical specifications. The local reservoir location and new snubber will not adversely affect the safety function of this support. This is not an unreviewed safety question.

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