AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346
UNIT Davis-Besse
DATE August 7, 1979
COMPLETED BY Erdal Caba
TELEPHONE 259-5000 Ext. 236

MONTH July, 1979

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	882
2	0	18	878
3	0	19	881
4	0	20	879
5	0	21	887
6	0	22	869
7	0	23	873
8	0	24	871
9	0	25	872
10	0	26	866
11	0	27	874
12	259	28	873
13	440	29	877
14	739	30	876
15	798	31	872
16	847		

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.

(9/77)

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OPERATING DATA REPORT

DOCKET NO. 50-346
DATE August 7, 1979
COMPLETED BY E. Caba
TELEPHONE 259-5000 Ext. 236

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: July, 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 det.
7. Maximum Dependable Capacity (Net MWe): To be det.
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): Zero (until July 6, 1979)
10. Reasons For Restrictions. If Any: NRC OIE Bulletins and Shutdown Orders

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744	5,087	16,852
12. Number Of Hours Reactor Was Critical	498	2,245.4	8,877.2
13. Reactor Reserve Shutdown Hours	264.2	1,876.4	2,666.7
14. Hours Generator On-Line	479.8	2,154.9	7,888.1
15. Unit Reserve Shutdown Hours	264.2	1,728.2	1,728.2
16. Gross Thermal Energy Generated (MWH)	1,230,451	5,109,548	15,297,118
17. Gross Electrical Energy Generated (MWH)	409,950	1,703,218	5,086,973
18. Net Electrical Energy Generated (MWH)	381,814	1,594,372	4,635,832
19. Unit Service Factor	64.5	42.4	48.3
20. Unit Availability Factor	100	76.3	59.9
21. Unit Capacity Factor (Using MDC Net)	To be det.		
22. Unit Capacity Factor (Using DER Net)	56.6	34.6	33.7
23. Unit Forced Outage Rate	0	3.8	21.3

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):

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		

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UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse
DATE August 7, 1979
COMPLETED BY Erdal Caba
TELEPHONE 259-5000 Ext. 236

REPORT MONTH July, 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
9	79 03 30	S	264.2	D	1	N/A	N/A	N/A	The unit remained in an outage until July 12, 1979. Refer to the outage summary of July, 1979, for further details.

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

(9/77)

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OPERATIONAL SUMMARY FOR JULY, 1979

The unit outage, which began at 2142 hours on March 30, 1979, was in progress the first nine days of the month. The outage was extended longer than anticipated due to additional NRC startup restraints which were imposed as a result of an ongoing analysis of the Three Mile Island incident. NRC released the unit to go to Mode 3 on July 2, 1979, and the NRC shutdown order was lifted on July 6, 1979.

7/10/79	Reactor criticality was established at 2338 hours. At 2339 hours, the startup range nuclear instrument NI-1 failed and a reactor shutdown was initiated.
7/11/79	Reactor criticality was re-established at 0558 hours.
7/12/79	The turbine-generator was synchronized on line at 0011 hours. Reactor power was increased, but was administratively limited to 60%. This limitation was initiated to investigate an asymmetric rod fault for Rod 2 of Group 2.
7/13/79-7/16/79	Reactor power was increased to 90% on July 14, 1979, and maintained until July 16, 1979. Reactor power was increased to 100% full power at 2223 hours on July 16, 1979.
7/17/79 - 7/31/79	The unit was maintained between 99% and 100% reactor power the remainder of the month.

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REFUELING INFORMATION

DATE: July, 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 f 1 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 (73. total)

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

Date March, 1980 - May, 1980 (assuming ability to unload the entire core into the spent fuel pool is maintained.)

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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 In[illegible]n 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 JULY, 1979

FCR NO: 79-151

SYSTEM: Service Water (SW) and Component Cooling Water (CCW)

COMPONENT Component Cooling Water Heat Exchanger Service Water Outlet Valves (SW 1424, SW 1429, SW 1434)

CHANGE, TEST, OR EXPERIMENT: On July 3, 1979, implementation of Facility Change Request 79-151 was completed. This change modified the actuator linkages on valves SW 1424, SW 1429, and SW 1434 to provide a more secure and positive attachment of the actuator linkage arms to the disk arms of the valves.

REASON FOR THE FCR: The retaining nut for the valve linkage was being loosened by vibration which caused slippage and misalignment of the valve operator linkage. See License Event Reports NP-33-79-74, NP-33-78-147, NP-33-78-120 for further details.

SAFETY EVALUATION: This change will prevent the head capscrew from backing out and the valve linkage from vibrating loose. The safety function of the service water system will not be adversely affected. This change is expected to increase the reliability of these service water valves. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING JULY, 1979

FCR NO: 79-170

SYSTEM: Reactor Protection System (RPS)

COMPONENT: High Pressure Trip Bistable Setpoints

CHANGE, TEST, OR EXPERIMENT: On May 26, 1979, work was completed which readjusted the trip setpoints of the Reactor Protection System (RPS) from 2351.4 PSIG to 2296.4 PSIG. The system operating procedure, as well as the applicable surveillance test procedures, have been revised to reflect this change. A request for an amendment to Table 2.2-1 of the Davis-Besse Unit 1 Technical Specifications has been submitted in order to reflect the above change.

REASON FOR THE FCR: The reduction in the RPS high pressure trip setpoint was made to preclude actuation of the pressurizer Power Operated Relief Valve (PORV). This is in response to Nuclear Regulatory Commission Bulletin 79-05B.

SAFETY EVALUATION: The proposed reduction in the RPS high pressure trip setpoint does not degrade the safety of the plant and does not invalidate any of the safety analyses presented in the Davis-Besse Unit 1 FSAR or in the safety evaluation submitted to the NRC on December 22, 1978 (Serial No. 475). The possibility of an accident or a malfunction of a different type than any evaluated in the FSAR is not created. Also, the margin of safety as defined in the bases for technical specification is not reduced. Pursuant to the above, the proposed change does not involve an unreviewed safety question. For further details, see Toledo Edison response to NRC Bulletin 79-05B.

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FACILITY CHANGE REQUESTS COMPLETED DURING JULY, 1979

FCR NO: 79-244

SYSTEM: Main Steam

COMPONENT: Snubber mounts on both SR-7 seismic restraint locations

CHANGE, TEST, OR EXPERIMENT: On June 20, 1979, work was completed which extended the length of the snubber mountings on both SR-7 seismic restraint locations 1½ inches. All applicable drawings were revised by the unit Architecht-Engineer Bechtel Company.

REASON FOR THE FCR: The former mounting arrangement caused these two snubbers to be completely extended when the piping was cold. This change placed these snubbers in the middle of their stroke when cold, as designed. These snubbers were operable and still are operable when at normal operable temperature.

SAFETY EVALUATION: This change moves the mounting of the snubber such that it will be in the designed position with the piston properly centered. Therefore, operation of the snubber in the unlikely event of a seismic event will be as designed. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING JULY, 1979

FCR NO: 79-257

SYSTEM: Main Steam

COMPONENT: Seismic restraints SR-2, SR-4, SR-6, and SR-7

CHANGE, TEST, OR EXPERIMENT: On June 27, 1979, work was completed which added webb reinforcing plates to seismic restraints SR-2, SR-4, SR-6, and SR-7. Since seismic restraints to SR-2, SR-4, SR-6, and SR-7 are located on both main steam lines, a total of eight seismic restraint locations were affected. The necessity for addition of the web reinforcing plates to these eight seismic restraint locations was determined as a result of an investigation conducted by the unit architect/engineer, Bechtel Company. All affected drawings were revised to reflect the change.

REASON FOR FCR: A review of design calculations by Bechtel discovered a design deficiency in that the impact points of these eight seismic restraint locations could have deformed if a design basis earthquake had occured. The addition of the web reinforcing plates corrects this deficiency. (For further details, see License Event Report NP-32-79-08).

SAFETY EVALUATION: Bechtel Engineering evaluation of as-built seismic restraints on the main steam lines showed that eight seismic restraint locations (four on each main steam line) may be subject to I-beam web deformation. To prevent this web deformation, web stiffeners were welded to both sides of the I-beam web. This will stiffen the web and distribute the load, thus preventing web deformation. This modification will not degrade the safety function of the main steam system. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING JULY, 1979

FCR NO: 79-261

SYSTEM: Auxiliary Feed Water (AFW) System

COMPONENT: Auxiliary Feed Pumps and Turbines 1-1 and 1-2

CHANGE, TEST, OR EXPERIMENT: On June 28, 1979, a 72-hour endurance run of both auxiliary feedwater pumps was successfully completed. After a subsequent cool-down period, both pumps were restarted and run for one hour. This test was conducted with the unit in Mode 5. No water was pumped into the steam generators; pump discharge was to the station drainage system or back to the condensate storage tank.

REASON FOR THE FCR: The above mentioned test runs were made to fulfill an NRC commitment to verify the auxiliary feedwater pumps would operate properly for an extended period of time.

SAFETY EVALUATION: This test did not degrade the safety of the unit since it was conducted in Mode 5, and the AFW system is only required in Modes 1, 2, and 3. Also, a surveillance test has been conducted upon the system before the unit is started up in order to ensure its operability. This is not an unreviewed safety question.