AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE November 9, 1979

COMPLETED BY Erdal Caba

TELEPHONE (419) 259-5000, Ext. 236

MONTH October, 1979

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	879
2	882
3	883
4	884
5	558
6	0
7	0
8	647
9	877
10	883
11	887
12	885
13	886
14	785
15	244
16	0

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
17	0
18	0
19	0
20	0
21	80
22	736
23	653
24	641
25	316
26	0
27	0
28	0
29	0
30	0
31	0

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)

7911140 332

1321 341

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE November 9, 1979
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 236

OPERATING STATUS

	Notes
1. Unit Name: Davis-Besse Unit 1	
2. Reporting Period: October, 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): to be determined	

8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

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	745	7,296	19,061
12. Number Of Hours Reactor Was Critical	419.1	4,043.5	10,675.3
13. Reactor Reserve Shutdown Hours	142.2	2,085.5	2,875.8
14. Hours Generator On-Line	397.6	3,900.6	9,633.8
15. Unit Reserve Shutdown Hours	0	1,728.2	1,728.2
16. Gross Thermal Energy Generated (MWH)	950,616	9,601,764	19,789,334
17. Gross Electrical Energy Generated (MWH)	320,890	3,202,078	6,585,833
18. Net Electrical Energy Generated (MWH)	294,028	3,008,035	6,049,495
19. Unit Service Factor	53.4	53.5	52.3
20. Unit Availability Factor	53.4	77.1	62.4
21. Unit Capacity Factor (Using MDC Net)	to be determined		
22. Unit Capacity Factor (Using DER Net)	43.6	45.5	38.5
23. Unit Forced Outage Rate	42.8	10.1	20.4

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: November 12, 1979

26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

1321 342

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE November 9, 1979
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 236

REPORT MONTH October, 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
16	79 10 5	S	48.8	A	1	NA	CJ	VALVEX	Shutdown to repair pressurizer spray valve RC 2.
17	79 10 15	F	142.2	A	3	NP-32-79-11	HA	INSTRU	Capacitor failure in Integrated Control System (ICS) pulser circuit to the turbine electro-hydraulic control system. Refer to attached summary for further details.
18	79 10 25	F	156.4	A	3	NP-33-79-121	CB	CKTBRK	Loss of Reactor Coolant Pump 2-2 from blown fuse in the DC power supply starting a pump two minute time delay trip relay with Reactor Coolant Pump 1-1 already shutdown.

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)

1321 343

OPERATIONAL SUMMARY FOR OCTOBER, 1979

10/1/79 - 10/5/79	Reactor power was maintained at 100 percent until 2025 hours on October 5, 1979 when the reactor was manually shutdown to repair the pressurizer spray valve RC2.
10/6/79	The unit remained shutdown to repair the pressurizer spray valve.
10/7/79	The turbine generator was synchronized at 2115 hours and reactor power was increased and maintained between 98-99 percent of full power.
10/11/79 - 10/14/79	The unit operated at approximately 100 percent full power with the turbine-generator gross load at 920 ± 5 MWe. At 1678 hours on October 14, 1979, group 7 rod 7 API was declared inoperable and reactor power was reduced to approximately 59 percent.
10/15/79	At approximately 1227 hours on October 15, 1979, a capacitor failed in the Integrated Control System (ICS) pulser circuit to the turbine electro-hydraulic control system. This capacitor failure caused the turbine control valves to open which lowered the main steam line header pressure. The ICS responded to the low header pressure by increasing both reactor power and feed-water which resulted in a reactor protection system reactor trip at the reduced high flux setpoint of approximately 68.8 percent of full power. While reclosing the generator output breaker 34560 at approximately 1250 hours, "J" bus tripped which resulted in a de-energization of the startup transformer 01 and a station loss of off-site power. Both emergency diesel generators automatically started. The Steam and Feedwater Rupture Control System (SFRCS) actuated from the loss of all four reactor coolant pumps, both auxiliary feed pumps started and natural circulation was established in the Reactor Coolant System (RCS). The cause of the station loss of off-site power was a blowing out of the internals of the muffler on generator output breaker 34560 when it opened for the trip which caused a fault to ground when the breaker was reclosed.
10/16/79 - 10/22/79	The unit remained shutdown until October 31, 1979 when the turbine-generator was synchronized. Reactor power was increased to 92 percent of full power at approximately 1400 hours on October 22, 1979 for the xenon equilibrium hold.
10/23/79	Reactor power was decreased to 70 percent of full power when Reactor Coolant Pump 1-1 seal destaged due to first and third stage seal failure. The reactor coolant pump was tripped at 0430 hours.

10/24/79 - 10/25/79

Reactor power was maintained at approximately 70 percent with three reactor coolant pumps in operation on October 25, 1979, when a reactor trip occurred.

At approximately 1226 hours on October 25, 1979, station personnel de-energized E5 bus to remove station transformer ST 1 from service. At approximately 1238 hours, Reactor Coolant Pump 2-2 tripped from a blown fuse in the DC power supply. The tripping of this pump started a two minute time delay trip relay with Reactor Coolant Pump 1-1 already shutdown because of seal staging difficulties. The Reactor Protection System tripped the reactor on a "flux to number of reactor coolant pumps" actuation. A design deficiency was discovered in the reactor coolant pump interlock circuit which has been corrected with the addition of surge suppressors.

10/26/79 - 10/31/79

The unit remained shutdown to replace the seals of Reactor Coolant Pump 1-1. It was later decided to replace the seals of all four reactor coolant pumps when it was determined that the station loss of off-site power incident on October 15, 1979 had an adverse effect on the reactor coolant pump seal performance.

REFUELING INFORMATION

DATE: October, 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)

1321 346

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)

FACILITY CHANGE REQUEST COMPLETED DURING OCTOBER, 1979

FCR NO: 77-478

SYSTEM: Auxiliary Feedwater (AFW)

COMPONENT: AFW Pump speed control transfer switch position indication

CHANGE, TEST, OR EXPERIMENT: On July 11, 1977 station instrument and control technicians found that the wiring of the position indicating lights for the AFW pump speed control local/remote transfer switch was incorrect as was the applicable "as-built" drawings. The circuitry was rewired to make the indicating lights operate properly. Revisions to Bechtel drawings E-45B Sheet 1, E-289 Sheet 1, and E-298 Sheet 1 were completed by the unit architect/engineer, Bechtel Company, in order to document the corrected "as-built" wiring configuration.

REASON FOR THE FCR: FCR 77-478 was written to document the corrected wiring configuration of this indicating lamp circuit.

SAFETY EVALUATION: This FCR involves the rewiring of contacts in the auxiliary feed pump turbine speed control system in order to have the indicating lights associated with the local/remote transfer switch properly reflect the position of the switches. This change enhances operation of the system and does not result in an unreviewed safety question.

FACILITY CHANGE REQUEST COMPLETED DURING OCTOBER, 1979

FCR NO: 78-521

SYSTEM: Post Accident Radiation Monitors

COMPONENT: Sampling Pumps

CHANGE, TEST, OR EXPERIMENT: On September 5, 1979 work was completed for FCR 78-521. This FCR was written to investigate the possibility of reducing the speed of the air sampling pumps of RE5029 and RE5030, the Containment Post Accident Radiation Monitors. After analysis, the unit architect/engineer, Bechtel Corporation determined that the function of these monitors would not be affected by this change. The pump pulley size and the size of the driving belts were changed to reduce the sampling rate from approximately 8.5 CFM to 4 CFM. Also new flow meters were installed in order to properly monitor the lower flow rate. These changes were made with the concurrence of both Bechtel Corporation and the monitor vendor, Victoreen Incorporated.

REASON FOR THE FCR: This reduction in pump speed was undertaken to decrease pump wear and internal heating as well as to decrease the load on the pump drive motors in an attempt to reduce the frequency of motor and pump bearing failures (see Licensee Event Reports NP-33-79-98, NP-33-79-95, NP-33-79-42, NP-33-79-37, NP-33-78-143, NP-33-78-127, NP-33-78-111, NP-33-78-105, NP-33-78-101, NP-33-78-77, NP-33-78-54, NP-33-78-45, and NP-33-78-30). These two radiation monitors are the only ones which have experienced repetitive bearing failures. This is attributed to the fact that these particular monitors are located in areas where the ambient temperature during plant operation is high (mechanical penetration rooms).

SAFETY EVALUATION: The subject radiation detectors RE5029A, B, C and RE 5030A, B, and C are utilized for post-accident monitoring and monitoring of the containment during normal operation. For the post-accident function, the gaseous monitors are needed. For detection of containment radioactivity resulting from a reactor coolant pressure boundary leak, the monitors of interest are the particulate and gaseous monitors.

The Davis-Besse Unit 1 Technical Specifications address these monitors in Sections 3/4.3.3.1, 3/4.3.3.6, 3/4.4.6.1 and 3/4.4.6.2. The only requirements imposed deal with the particulate and gaseous radioactivity monitors and specify that the measurement range be 10 to 10^6 cpm. There is no limit on sensitivity or response time. The iodine radioactivity monitors are not required.

The effect of the proposed reduction in blower flow rate to approximately four (4) CFM has been evaluated. The expected sensitivities for approximately three (3) CFM as noted below are in the range of the values specified in the FSAR which were based in 8.5 CFM and are well below the maximum permissible concentration (MPC) for a restricted area for activity in air, as specified in 10 CFR 20, Appendix B, Table 1, Column 1. This satisfies the statement in Section 11.4.2.2.5 of the FSAR that requires the ability to measure MPC.

Monitor Sensitivity at Three (3) CFM

Monitor	Isotope of Interest	Sensitivity (μc/cc)	MPC (μc/cc)
Particulate	Cs^{137}	3×10^{-11}	6×10^{-8}
Gaseous	Xe^{133}	3×10^{-7}	1×10^{-5}
Iodine	I^{131}	2×10^{-12}	9×10^{-9}

The NRC Safety Evaluation Report (SER) Supplement 1, Section 5.2.4 stated that the leakage detection systems "are generally in accordance with the recommencations of Regulatory Guide 1.45." This guide states that a one (1) gallon per minute (gpm) leak rate should be detected within one hour. The options for detection include the containment particulate and gaseous radioactivity monitors. Calculations indicate that these monitors will be capable of detecting a leak rate of one (1) gpm within one hour, utilizing a blower flow rate of three (3) cfm. This is true whether the containment is being purged or not.

Based on the above, it is concluded that the proposed reduction in blower flow rate to approximately three (3) to four (4) cfm will not result in a change in the Technical Specifications incorporated in the license or an unreviewed safety question per the definition of 10 CFR 50.59.