AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE June 8, 1979

COMPLETED BY Erdal C. Caba

TELEPHONE 419-259-5000, Ext. 236

MONTH May, 1979

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	0
2	0	18	0
3	0	19	0
4	0	20	0
5	0	21	0
6	0	22	0
7	0	23	0
8	0	24	0
9	0	25	0
10	0	26	0
11	0	27	0
12	0	28	0
13	0	29	0
14	0	30	0
15	0	31	0
16	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.

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7906140408

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE June 8, 1979
COMPLETED BY Erdal C. Caba
TELEPHONE 419-259-5000, Ext. 236

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: May, 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:

Notes

9. Power Level To Which Restricted, If Any (Net MWe): Zero
10. Reasons For Restrictions, If Any: NRC OIE Bulletins and Shutdown Orders

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744	3,623	15,388
12. Number Of Hours Reactor Was Critical	0	1,747.4	8,379.2
13. Reactor Reserve Shutdown Hours	744	892.2	1,682.5
14. Hours Generator On-Line	0	1,675.1	7,408.3
15. Unit Reserve Shutdown Hours	744	744	744
16. Gross Thermal Energy Generated (MWH)	0	3,879,097	14,066,667
17. Gross Electrical Energy Generated (MWH)	0	1,293,268	4,677,023
18. Net Electrical Energy Generated (MWH)	0	1,212,558	4,254,018
19. Unit Service Factor	0	46.2	50.0
20. Unit Availability Factor	100	66.8	55.6
21. Unit Capacity Factor (Using MDC Net)	to be determined		
22. Unit Capacity Factor (Using DER Net)	0	36.9	34.3
23. Unit Forced Outage Rate	0	4.8	22.4

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: June 17, 1979
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

REPORT MONTH ____________

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE ____________
COMPLETED BY Charles N. Alm
TELEPHONE 419-259-5000, Ext. 251

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	744	D	1	N/A	N/A	N/A	The unit remained in an outage the entire month. Refer to the attached Outage Summary of May 1979 for outage activities this month.

POOR ORIGINAL

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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OUTAGE SUMMARY
May, 1979

The unit outage which began at 2142 hours on March 30, 1979 was still in progress through the end of May, 1979. The outage was extended longer than anticipated to respond to NRC IE Bulletins 79-05, 79-05A, and 79-05B. There have also been additional NRC startup restraints which were imposed as a result of an ongoing analysis of the Three Mile Island incident.

The following are the more significant outage activities performed during the month of May:

1. Procedure modification to comply with NRC requests. In addition, modifications of procedures were made due to the re-evaluation of the Babcock and Wilcox small break analysis.

2. Personnel instruction on what the procedure major modifications involved.

3. The installation of an additional anticipatory reactor trip system (Facility Change Request 79-176).

4. The installation of auxiliary feedwater flow indication on the Auxiliary Feedwater System. Also testing was performed on this system to resolve NRC commitments.

5. The performance of eighteen month surveillance tests which were required to be completed prior to the reactor internal vent valve test. This was done to lengthen the available operation time after this outage.

6. The replacement of the Reactor Coolant Pump 1-1-1 seals.

7. The performance of hydraulic snubber surveillance testing which is still in progress.

8. Testing was performed on both Moisture Separator Reheaters. The Moisture Separator Reheater 2 was inspected and several instrument tubes were plugged and one broken instrument tube was repaired.

REFUELING INFORMATION

DATE: May, 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 is awaiting a final NRC Safety Evaluation Report to proceed in time for completion prior to refueling. All licensee submittals are complete including environmental assessment questions.

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 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 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 REQUESTS COMPLETED DURING MAY, 1979

FCR NO. 77-475

SYSTEM: Emergency Diesel Generator 1-2

COMPONENT: Control and Relay Panel C3616

CHANGE, TEST, OR EXPERIMENT: Facility Change Request (FCR) 77-475 was written to revise General Electric Drawing 0132D4596, Sheet 2, to correct an error in the drawing. This drawing is the connection diagram for the Emergency Diesel Generator 1-2 local control panel. The specific change was to revise the external annunciator connections from terminals NN1 and NN2 to NN2 and NN3.

REASON FOR THE FCR: It was found that incorrect wiring terminations had been made on terminal block NN in the diesel engine alarm cabinet because of the error on the General Electric drawing. The terminations were corrected and the local annunciator tested under Maintenance Work Order (MWO) 77-1037 on June 21, 1977. This correction is documented in General Electric Drawing 0132D4596, Sheet 2, Revision B1, dated February 8, 1978.

SAFETY EVALUATION: This work, which corrects the vendor drawing for the external annunciator connections, completes the proper annunciator terminations to terminal block NN. The change will allow the annunciator to function properly. This is not an unreviewed safety question.

FACILITY CHANGE REQUESTS COMPLETED DURING MAY, 1979

FCR NO: 78-001

SYSTEM: Emergency Ventilation System (EVS)

COMPONENT: Valves CV5024 and CV5025 (isolation valves between auxiliary building ventilation systems and EVS)

CHANGE, TEST, OR EXPERIMENT: FCR 78-001 was written to request the revision of Bechtel Piping and Instrument Diagram (P&ID) M-029A, "HV and AC Air Flow Diagram for Containment and Penetration Rooms", to reflect the as built conditions. The change made involved interchanging labeling on the controls and interlocks for valves HV2024 and HV2025 on the P&ID. This change was incorporated in Revision 22 of P&ID M-029A, dated February 16, 1978, by Bechtel, the unit architect/engineer.

REASON FOR THE FCR: It was found that the labeling of the valves was interchanged, and the wiring of the controls and interlocks were interchanged between the valves. The valves are connected in series in the flowpath. Rather than actually rewiring the valves and associated controls to correct this construction error, which would have required extensive plant modifications, a more acceptable solution was to re-label the valves and associated controls on the P&ID.

SAFETY EVALUATION: The only change required on this FCR is to correct P&ID M-029A to reflect as-built locations of valves CV5024 and CV5025 and associated controls and interlocks. Toledo Edison Power Engineering Department has evaluated and confirmed that the function of the EVS will not be affected by this change.

FACILITY CHANGE REQUESTS COMPLETED DURING DECEMBER, 1978

FCR NO: 78-013

SYSTEM: Final Safety Analysis Report (FSAR)

COMPONENT: Chapter 14

CHANGE, TEST, OR EXPERIMENT: FCR 78-013 was written on January 6, 1978, to nullify the erroneous requirement in the FSAR abstract of TP 800.25, "Shutdown From Outside the Control Room", that the reactor be tripped from outside the Control Room as a startup test. However, the NRC requested that the reactor be tripped from outside
1 of the Control room as the FSAR abstract had originally required. The NRC request was in spite of the fact that tripping the reactor from outside the Control Room was inconsistent with the remainder of the FSAR. To comply with the NRC request, Test Procedure TP 800.25 was conducted on January 14, 1979 by tripping the reactor from outside the Control Room using the control rod drive breakers.

REASON FOR THE FCR: All of the accident analyses in the FSAR and the Davis-Besse Unit 1 Fire Hazards Analysis Report assume the reactor to be tripped prior to evacuation of the Control Room.

SAFETY EVALUATION: Administrative Procedures require that the reactor be tripped prior to evacuation of the Control Room. The FSAR abstract of TP 800.25 requires that the reactor be tripped from outside the Control Room as a startup test. This commitment was an error and is not consistent with the remainder of the FSAR; reference the Chapter 7 discussion of the Auxiliary Shutdown Panel.

This change in the Startup Test Program from that described in Chapter 12 of the FSAR does not therefore represent an unresolved or unreviewed safety question.

Conducting the test by tripping the reactor from outside the Control Room was done
1 to satisfy NRC desires and was done with prior NRC knowledge and approval.

FACILITY C'ANGE REQUESTS COMPLETED DURING MAY, 1979

FCR NO: 78-317

SYSTEM: Decay Heat Removal (DHR)

COMPONENT: Motor Operated Valves (MOVs) DH 2733 and DH 2734

CHANGE, TEST, OR EXPERIMENT: Facility Change Request (FCR) 78-317 was written to revise Bechtel Drawing E-15, MOV Data List, torque switch dial settings for DH 2733 and DH 2734 from 2.25 to 3.0. DH 2733 is the suction isolation valve from the Borated Water Storage Tank (BWST) for DHR Pump 1-1 and DH 2734 is the suction isolation valve for DHR Pump 1-2.

This change was incorporated by Bechtel Company, the unit architect/engineer, in revision 5 of Drawing E-15, dated February 2, 1979.

REASON FOR THE FCR: It was found that the previous torque switch dial setting of 2.25 was not adequate to properly seat the valves. Leak thru was occuring due to improper seating of the valves. Thr torque switch dial setting of 3.0 allows proper seating and minimum leakage.

SAFETY EVALUATION: Raising the recommended torque switch setting from 2.25 to 3.0 has been evaluated and was found to be acceptable. This setting provides satisfactory operation and is well below the maximum torque switch setting (set at 3.0 or 450 ft - lb versus maximum of 3.75 or 850 ft - lb). This change will not adversely affect the safety function of the dec·y heat removal system. Valves DH2733 and DH2734 are normally open valves. This is not an unreviewed safety question.

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FACILITY CHANGE REQUEST COMPLETED DURING MAY, 1979

FCR NO: 78-498

SYSTEM: Emergency Core Cooling System (ECCS)

COMPONENT: High Pressure Injection (HPI) Pumps and Low Pressure Injection (LPI) Pumps

CHANGE, TEST, OR EXPERIMENT: FCR 78-498 was written for the purpose of initiating a review to determine the proper interpretation of Subsection 6.3.3.2.3 of the Final Safety Analysis Report (FSAR). This subsection states that, "In the analysis of the loss of coolant accidents, the high pressure and low pressure injection was delayed 30 seconds after receipt of the actuation signal". The analysis was conducted by Bechtel, the unit architect/engineer, and Babcock & Wilcox, the nuclear steam supply system vendor.

REASON FOR THE FCR: The statement in FSAR Subsection 6.3.3.2.3 could be misinterpreted to imply that the LPI pump and HPI pumps would provide full rated flow within 30 seconds of receipt of the actuation signal. This is not a true statement for breaks smaller than the design basis LOCA as amount of flow will be determined by the Reactor Coolant System pressure. For the design basis LOCA, the ECCS will provide the full rated flow within 30 seconds of receipt of the actuation signal.

SAFETY EVALUATION: A review of FSAR Subsection 6.3.3.2.3 and the Babcock & Wilcox safety evaluation as presented in B&W Topical Reports BAW-10105 (ECCS Evaluation of Babcock & Wilcox's 177 FA Raised - Loop NSS) and BAW-10075 (Multinode Analysis of Samll Breaks for Babcock & Wilcox's 177 - Fuel - Assembly Nuclear Plants With Raised Loop Arrangement and Internal Vent Valves) indicates that the requirements for rated HPI and LPI flow within 30 seconds is based on the NSSS design basis LOCA, which is an 8.55 ft.^{2} break in the reactor coolant pump discharge piping. Any break of a smaller size requires less ECCS flow than the design basis event.

The correct interpretation of this FSAR subsection is that the system valves will be in their commanded (open) position in 30 seconds and the pumps (HPI and LPI) will be running at or above the flow corresponding to that assumed in Babcock & Wilcox Topical Reports BAW-10105 and BAW-10075. This interpretation is consistent with the analysis and does not constitute an unreviewed safety question.