AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE April 9, 1985

COMPLETED BY Bilal M. Sarsour

TELEPHONE (419) 249-5000 Ext. 384

MONTH March, 1985

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	827	17	400
2	826	18	423
3	827	19	416
4	825	20	427
5	824	21	316
6	824	22	0
7	824	23	0
8	822	24	0
9	827	25	0
10	818	26	0
11	827	27	0
12	827	28	0
13	827	29	0
14	825	30	0
15	822	31	0
16	665		

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 April 9, 1985
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 249-5000
Ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: March, 1985
3. Licensed Thermal Power (MWt): 2772
4. Nameplate Rating (Gross MWe): 915
5. Design Electrical Rating (Net MWe): 906
6. Maximum Dependable Capacity (Gross MWe): 904
7. Maximum Dependable Capacity (Net MWe): 860
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons: Items 6 and 7 changed to obtain an acceptable offset from the Reactor Protection System trip setpoints
9. Power Level To Which Restricted, If Any (Net MWe):
10. Reasons For Restrictions, If Any:

Notes

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744	2,160.0	58,465.0
12. Number Of Hours Reactor Was Critical	501.0	1,567.1	34,598.6
13. Reactor Reserve Shutdown Hours	0.0	0.0	4,014.1
14. Hours Generator On-Line	499.9	1,475.9	33,117.2
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	1,153,596	3,350,351	78,335,773
17. Gross Electrical Energy Generated (MWH)	382,649	1,108,804	25,955,148
18. Net Electrical Energy Generated (MWH)	358,144	1,030,413	24,320,669
19. Unit Service Factor	67.2	68.3	56.6
20. Unit Availability Factor	67.2	68.3	59.6
21. Unit Capacity Factor (Using MDC Net)	56.0	55.5	48.4
22. Unit Capacity Factor (Using DER Net)	53.1	52.7	45.9
23. Unit Forced Outage Rate	0.2	.07	16.7

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: April 10, 1985
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH March 1985

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 249-5000, Ext. 384

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
2	85 03 16	F	0	A	5	N/A	N/A	N/A	A manual runback to approximately 46% power was initiated due to dropped Control Rod 5-3.
3	85 03 21	F	1.1	A	3	NP-33-85-06	JA	IMOD	The reactor tripped by the Anticipatory Reactor Trip System (ARTS) due to SFRCS trip on OTSG low level caused by oscillations on the #2 Main Feed Pump.
4	85 03 21	S	243.0	B	1	N/A	N/A	N/A	Scheduled maintenance outage. See Operational Summary 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-Continuation from Previous Month
5-Load Reduction
9-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)

OPERATIONAL SUMMARY
MARCH, 1985

3/1/85 - 3/10/85:

Reactor power was maintained at approximately 94% (reactor power was limited to 94% due to an inoperable main steam safety valve) until 0405 hours on March 10, 1985, when a manual power reduction to approximately 89% was initiated due to low load requirements.

Reactor power was slowly increased to approximately 94% which was reached at 0900 hours on March 10, 1985.

3/11/85 - 3/17/85:

Reactor power was maintained at approximately 94% until 1300 hours on March 16, 1985, when a manual power reduction to approximately 85% was initiated to perform turbine valves testing and control rod drive exercise testing. During the control rod exercise testing, Rod 5-3 dropped and could not be withdrawn. This placed the unit in a Technical Specification action statement requiring power to be reduced to less than 60% full power. At 1630 hours on March 16, 1985, the reactor power was reduced to less than 60%.

Reactor power was slowly increased to approximately 54% which was reached at 2000 hours on March 17, 1985.

3/18/85 - 3/31/85:

Reactor power was maintained at approximately 54% until 1600 hours on March 21, 1985, when a manual power reduction was initiated to begin a scheduled maintenance outage to investigate the cause of dropped Rod 5-3.

During the shutdown and while the unit was at approximately 26% power, a reactor trip occurred. The reactor tripped by the Anticipatory Reactor Trip System (ARTS) due to a Steam and Feedwater Rupture Control System (SFRCS) trip on Steam Generator low level caused by oscillations on the #2 Main Feedwater Pump. A plant cooldown was initiated to perform scheduled maintenance. Details on the work items performed during the outage will be presented in next month's operational summary.

REFUELING INFORMATION DATE: March 1985

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: Spring, 1986

3. Scheduled date for restart following refueling: Summer, 1986

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)?

   Ans: Expect the Reload Report to require standard reload fuel design Technical Specification changes (3/4.1 Reactivity Control Systems and 3/4.2 Power Distribution Limits).

5. Scheduled date(s) for submitting proposed licensing action and supporting information: Winter, 1985

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.

   Ans: None identified to date.

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

   (a) 177 (b) 204 - Spent Fuel Assemblies

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: 735 Increase size by: 0 (zero)

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

   Date: 1992 - assuming ability to unload the entire core into the spent fuel pool is maintained.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-159

SYSTEM: Process and Area Radiation Monitoring

COMPONENT: RE1003A, RE1003B, RE5327, and RE5328

CHANGE, TEST OR EXPERIMENT: FCR 78-159 revised vendor drawing 12501-E-11, Revision B1 to show that radiation monitors RE1003A, RE1003B, RE5327, and RE5328 have the Reliant type motors while the other radiation monitors have Dayton type motors. The updating of drawing 12501-E-11 was completed January 28, 1980.

REASON FOR CHANGE: This change was made so drawing 12501-E-11 would represent correct, actual plant condition.

SAFETY EVALUATION SUMMARY: Since the integrity of the Process and Area Radiation Monitoring System is not reduced, this change did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-372

SYSTEM: Component Cooling Water

COMPONENT: RC3607 and RC3608

CHANGE, TEST OR EXPERIMENT: This FCR corrected the improper labeling of terminal blocks within cabinets RC3607 and RC3608. This was done by changing the nameplates of terminal blocks as specified on connection diagrams E-544B, Sheets 11 and 12. This FCR also updated vendor drawings 7749-E-33-67-5 and 7749-E-33-68-4 to represent the above changes. Work on FCR 78-372 was completed June 22, 1984.

REASON FOR CHANGE: This change was made because the nameplates for the terminal blocks of cabinets RC3607 and RC3608 were not in their correct locations according to connection diagrams E-544B, Sheets 11 and 12. The vendor drawings needed updating to agree with the connection diagrams. The connection diagrams are representative of the correct configuration for the terminal blocks in cabinets RC3607 and RC3608.

SAFETY EVALUATION SUMMARY: Installation of these labels will not affect the safety function of the components within cabinets RC3607 and RC3608. This FCR will assure that the correct relay module is being inspected during troubleshooting and maintenance. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-019

SYSTEM: Fire Detection

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: Work implemented by FCR 79-019 was completed September 28, 1980. This FCR installed smoke detectors and associated components in Rooms 600, 601, 602, and 603. All of these rooms are located at an elevation of 643' in the Auxiliary Building.

REASON FOR CHANGE: This change was completed to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION SUMMARY: Since the margin of safety of the Fire Detection System as outlined by the Technical Specifications is not reduced, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-196

SYSTEM: Safety Features Actuation System (SFAS)

COMPONENT: Drawing 7749-E17B

CHANGE, TEST OR EXPERIMENT: This FCR modified drawing 7749-E17B to represent the correct "as-built" conditions of the plant. Work was completed October 22, 1982.

REASON FOR CHANGE: During performance of MC 7500.34, The SFAS Sequencer Operation Test, when Channel 1 was activated, HV-5090 actuated. According to drawing 7749-E17B, the activation of Channel 2 should have activated HV-5090, and the actuation of Channel 1 should have actuated HV-5065. In reality, HV-5090 actuates on activation of Channel 2.

SAFETY EVALUATION SUMMARY: The safety function of the SFAS is to prevent, limit, or mitigate the release of any radioactive material to the environment and to start the Emergency Core Cooling System.

The "as-built" conditions reflect the correct design of the plant. Therefore, the correction of the drawing will not affect the safety function of SFAS. This FCR does not involve any unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-061

SYSTEM: Emergency Diesel Generators

COMPONENT: K5-1 and K5-2

CHANGE, TEST OR EXPERIMENT: This FCR modifies the Emergency Diesel Generators, K5-1 and K5-2, exhaust manifolds. This modification involved replacing the rear exhaust chamber assembly with a newer Power System's design. This design includes an improved exhaust screen and an inspection port for the screen. Work involved with this change was completed May 21, 1982.

REASON FOR CHANGE: This change was made due to the original exhaust screen failing during light or no load condition. Therefore, Power Systems, the Emergency Diesel Generator vendor, recommended the rear exhaust chamber assembly be replaced with a more updated assembly to increase Emergency Diesel Generator performance.

SAFETY EVALUATION SUMMARY: The safety function of the Emergency Diesel Generators is to provide a power source for plant shutdown in the event of a loss of offsite power. The installation of the exhaust chamber assembly will not degrade the safety function of the Diesel Generators, but instead will increase their performance. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-0[illegible]

SYSTEM: Various

COMPONENT: SW-223, SW-224, AF-7, AF-8, and BW-6

CHANGE, TEST OR EXPERIMENT: This FCR revised P&IDs M-041, M-006B, and M-033 to represent the present correct position of valves SW-223, SW-224, AF-7, AF-8, and BW-6. The following changes were made:

P&ID No.	Valve No.	Valve Position
M-041/G9	SW-223	Locked Close
M-041/G9	SW-224	Locked Close
M-006B/J8	AF-7	Locked Throttled
M-006B/J10	AF-8	Locked Throttled
M-033/B12	BW-6	Remove Locked Closed

This is a drawing change only FCR. Work involved with this FCR was completed December 12, 1983.

REASON FOR CHANGE: This FCR updated the P&IDs to represent changes made by FCR 79-158. FCR 79-158 provided locks for valves in various systems to control their positions per AD 1839.02, Operation and Control of Locked Valves.

SAFETY EVALUATION SUMMARY: This change involves only drawing changes, and the safety function of the affected systems is not degraded. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-128

SYSTEM: Spent Fuel Pool

COMPONENT: Spent Fuel Pool Cask Crane

CHANGE, TEST OR EXPERIMENT: FCR 83-128 installed a permanent jumper wire on the timer relay of the main hoist brake control circuit of the spent fuel cask crane. The jumper was installed between terminals L2 and C1. Work was completed May 15, 1984.

REASON FOR CHANGE: The installation of the jumper was required for the operation of timer relay. Without the jumper installed, the time delay relay coil circuit cannot be completed, and therefore, the operation of the main hoist would be prevented.

SAFETY EVALUATION SUMMARY: The spent fuel cask crane is designed to handle critical loads over the spent fuel pool. The safety function of the spent fuel cask crane main hoist brakes is to maintain the main hook and its load at a desired elevation. Since the installation of the jumper is required for the operation of the main hoist, this change will support the safety function of the spent fuel cask crane main hoist brakes. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-035

SYSTEM: Radwaste Systems

COMPONENT: Tiger Lock Storage Tank, T-136

CHANGE, TEST OR EXPERIMENT: This FCR removed the Tiger Lock Storage Tank, the tank's fill and transfer piping, the tank's platform, and the tank's level instrumentation. Work was completed August 30, 1984.

REASON FOR CHANGE: The Tiger Lock Storage Tank was removed to allow for more space in the Spent Fuel Pool Area. The structural components supporting the tank made the floor space inaccessible. By removing the tank and its supporting components, an additional 300 square feet was created for outage support and critical operational activities.

SAFETY EVALUATION SUMMARY: Since the Tiger Lock Storage Tank was not an active piece of equipment, the safety function of the radioactive waste systems is not decreased. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-098

SYSTEM: Core Flood System

COMPONENT: CF26, CF32, CF33, CF37, CF39, and CF47

CHANGE, TEST OR EXPERIMENT: FCR 84-098 updated drawings M-033 and ISID-033. It changed the notation of Core Flood Valves CF26, CF32, CF33, CF37, CF39A, and CF47 from 1/2 inch gate valves to 1/2 inch globe valves. This is a drawing change only FCR. This FCR was closed out on February 8, 1985.

REASON FOR CHANGE: The drawing changes required were the result of Non-Conformance Report 18-77 which identified the valves as being tagged as 1/2 inch gate valves, but having the identification number of 1/2 inch globe valves. This FCR updated drawings M-033 and ISID-033 to represent the correct, actual condition of the plant.

SAFETY EVALUATION SUMMARY: This FCR allows for correct plant conditions to be represented. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-136

SYSTEM: Service Water

COMPONENT: Hanger FSK-M-HBC-70-3H

CHANGE, TEST OR EXPERIMENT: FCR 84-136 revised drawing 7749-FSK-M-HBC-70-3H, Service Water Discharge From Cooling Coil Condenser, to represent actual plant conditions. These revisions to drawing 7749-FSK-M-HBC-70-3H were completed October 16, 1984.

REASON FOR CHANGE: Originally, this drawing listed the lower two bolts of the lower baseplate of hanger FSK-M-HBC-70-3H as through bolts when actually these are anchor bolts. The drawing change was made to represent the anchor bolts that are presently installed in the plant.

SAFETY EVALUATION SUMMARY: The safety function of the Service Water System is to supply cooling water to the component cooling heat exchangers, the containment air coolers, and the turbine plant cooling water heat exchangers during normal operation. Also, this system provides, through automatic valve sequencing, a redundant supply path to the engineering safety features components during an emergency. The changes made by this FCR do not affect the safety function of the Service Water System. Therefore, a new adverse environment is not created, and an unreviewed safety question does not exist.

TOLEDO EDISON

April 9, 1985

Log No. K85-630
File: RR 2 (P-6-85-03)

Docket No. 50-346
License No. NPF-3

Mr. Norman Haller, Director
Office of Management and Program Analysis
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Haller:

Monthly Operating Report, March 1985
<u>Davis-Besse Nuclear Power Station Unit 1</u>

Enclosed are ten copies of the Monthly Operating Report for Davis-Besse Nuclear Power Station Unit 1 for the month of March, 1985.

If you have any questions, please feel free to contact Bilal Sarsour at (419) 259-5000, Extension 384.

Yours truly,

Stephen M. Quennoz

Stephen M. Quennoz
Plant Manager
Davis-Besse Nuclear Power Station

SMQ/BMS/ljk

Enclosures

cc: Mr. James G. Keppler, w/1
Regional Administrator, Region III

Mr. Richard DeYoung, Director, w/2
Office of Inspection and Enforcement

Mr. Walt Rogers, w/1
NRC Resident Inspector

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