AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE April 8, 1981

COMPLETED BY Bilal Sarsour

TELEPHONE (419) 259-5000, Extension 251

MONTH March, 1981

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	883	17	659
2	856	18	667
3	547	19	661
4	870	20	660
5	886	21	692
6	840	22	807
7	877	23	806
8	88	24	810
9	0	25	808
10	0	26	807
11	0	27	809
12	0	28	806
13	73	29	808
14	410	30	805
15	474	31	804
16	608		

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)

8104140 379

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE April 8, 1981
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000, Extension 251

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: March, 1981
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): 934
7. Maximum Dependable Capacity (Net MWe): 890
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):
10. Reasons For Restrictions, If Any:

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	744	2,160	31,469
12. Number Of Hours Reactor Was Critical	709.2	1,462.2	15,846.4
13. Reactor Reserve Shutdown Hours	34.8	34.8	2,916.9
14. Hours Generator On-Line	616.6	1,352.6	14,400.4
15. Unit Reserve Shutdown Hours	0	0	1,731.4
16. Gross Thermal Energy Generated (MWH)	1,437,670	3,234,198	30,139,004
17. Gross Electrical Energy Generated (MWH)	478,048	1,080,766	10,056,100
18. Net Electrical Energy Generated (MWH)	446,468	1,010,370	9,274,871
19. Unit Service Factor	82.9	62.6	46.5
20. Unit Availability Factor	82.9	62.6	52.3
21. Unit Capacity Factor (Using MDC Net)	67.4	52.6	35.0
22. Unit Capacity Factor (Using DER Net)	66.2	51.6	34.4
23. Unit Forced Outage Rate	17.1	37.3	26.0

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		

POOR ORIGINAL

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE April 8, 1981
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000, Ext. 251

REPORT MONTH March, 1981

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	81 03 03	S	0	B	NA	NA	NA	NA	Power was reduced to approximately 55% to replace a single twisted pair cable on Main Feed Pump Turbine #1.
3	81 03 08	S	0	B	NA	NA	NA	NA	Power was reduced to 8% to investigate the high combustible gas alarm received from the main transformer.
4	81 03 11	F	127.4	A	1	NA	NA	NA	The reactor was manually tripped following a full Steam and Feedwater Rupture Control System trip. 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-Other (Explain)~~
4-Continuation
5-Reduction
6-Other

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)

POOR ORIGINAL

OPERATIONAL SUMMARY
MARCH, 1981

3/1/81 - 3/3/81	Reactor power was maintained at between 99% and 100% of full power with the turbine generator gross load at approximately 925 $\pm$ 10 MWe. Power was reduced to 55% at 0212 hours on March 3, 1981 to replace a single twisted pair cable (LIP) on Main Feed Pump Turbine (MFPT) #1. The reactor power level was maintained at 55% with the generator gross load at approximately 500 $\pm$ 10 MWe until 0520 hours on March 3, 1981 when reactor power was increased to 99% full power.
3/4/81 - 3/11/81	The reactor power was maintained at 99% full power with the generator gross load at 921 $\pm$ 10 MWe until 0645 hours on March 8, 1981 when the turbine generator was taken off line to investigate the high combustible gas alarm received from the main transformer, but the reactor stayed critical at approximately 8% power. The reactor power was maintained at 8% full power until 0845 hours on March 11, 1981 when the reactor was manually tripped following a full steam and feedwater rupture control system (SFRCS) actuation on low OTSG level.
3/12/81	The reactor was critical at 1935 hours.
3/13/81 - 3/31/81	The turbine generator was synchronized on line at 1350 hours. The reactor power was slowly increased and attained 90% full power on March 22, 1981 with the generator gross load at 845 $\pm$ 10 MWe. The reactor power was maintained at 90% full power for the rest of the month.

REFUELING INFORMATION

DATE: March, 1981

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

2. Scheduled date for next refueling shutdown: March, 1982

3. Scheduled date for restart following refueling: May, 1982

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)?

   Reload analysis is scheduled for completion as of December, 1981. No technical specification changes or other license amendments identified to date.

5. Scheduled date(s) for submitting proposed licensing action and supporting information. January, 1982

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.

   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) 44 - Spent Fuel Assemblies
   8 - New 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 1988 (assuming ability to unload the entire core into the spent fuel pool is maintained)

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 78-159

SYSTEM: Process and Area Radiation Monitoring

COMPONENT: Various Radiation Monitors (see below)

CHANGE, TEST OR EXPERIMENT: On October 18, 1979, the work as required by FCR 78-159 was completed. This FCR involved the replacement of the inboard and outboard bearings with Fafnir #203PP and #205PP sealed bearings in the pump motor of the following gaseous Radiation Monitors:

RE 1003A	RE 5052
RE 1003B	RE 5327
RE 2024	RE 5328
RE 2025	RE 5403
RE 5029	RE 5405
RE 5030	

REASON FOR THE CHANGE: Due to the high ambient temperatures around the Radiation Monitors, motor bearing seizure has been recurrent.

SAFETY EVALUATION: The replacement of the old bearings with sealed bearings will improve the operation of the radiation monitors. In addition, the new bearings have been packed with a qualified high temp grease.

The changes will enhance the operability of the monitors and will not adversely affect the safety of the plant. An unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-031

SYSTEM: Portable Communications

COMPONENT: N/A

PROPOSED CHANGE, TEST, OR EXPERIMENT: On August 17, 1980 the installation of a portable communications system was completed. This system consists of portable radios and a passive antenna system to enable emergency communications between the auxiliary building, containment, and the control room.

REASON FOR THE CHANGE: This system was installed to comply with Toledo Edison's commitment to provide portable communication equipment in selected locations for emergency communications. This commitment was made in the Fire Hazard Analysis Report, Revision 2 (Table 4-1, Section D5, Sheet 26).

SAFETY EVALUATION: This FCR is non-nuclear safety related except for the core drill/ cutouts. Installation in accordance with the core drill report and PICA will preclude those portions from creating any new adverse environment.

SAFETY EVALUATION: An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-159

SYSTEM: Containment Isolation

COMPONENT: NA

CHANGE, TEST OR EXPERIMENT: FCR 79-159 has been implemented to provide administrative controls and verification on the placement of caps on lines associated with containment isolation. The work portion of this FCR was completed on May 21, 1979.

REASON FOR THE CHANGE: This FCR was written to verify containment isolation on capped lines associated with the containment penetrations and to verify the integrity of systems that could be affected by uncapped lines.

SAFETY EVALUATION: FCR 79-159 provides verification of capped pipes to ensure the establishment of containment isolation of capped lines associated with containment penetrations as well as verifies capped conditions on lines that could affect the proper operation of certain safety related systems.

The associated caps have been reviewed to ensure that they will not adversely affect system operation given precaution to preclude overpressurization of lines between closed valves and valve caps.

This does not affect any event described in the Safety Analysis Report nor does it affect the Station Technical Specifications. An unreviewed safety question is not involved, therefore, no license amendment is required.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-414

SYSTEM: Main Steam

COMPONENT: Conduits 2-57112A and 2-57113A

CHANGE, TEST OR EXPERIMENT: On September 3, 1980, the upgrading of four (4) supports to seismic class I was completed. These supports were installed for the essential conduits 2-57112A and 2-57113A, located in Room 500. Drawing E-302A was revised to show the proper support details required in the upgrading of the above listed conduits.

REASON FOR THE CHANGE: It was determined from Nonconformance Report 14-79 that the supports for conduits 2-57112A and 2-57113A should be seismic Class I.

SAFETY EVALUATION: This FCR provided for the upgrading of four (4) supports to Seismic Class I. Installation in accordance with drawing E-302A will preclude creating any new adverse environments. An unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-117

SYSTEM: Hydraulic Snubbers

COMPONENT: Various

CHANGE, TEST OR EXPERIMENT: On September 23, 1980, the work as required by FCR 80-117 was completed. This FCR involved changing the orientation of the reservoir tubing on various snubbers. The snubbers affected were:

EBB-1-H5	12482
EBB-1-H5	12483
EBB-1-SR9	12474
EBB-1-SR9	12480
EBB-1-SR8	12447
EBB-1-SR8	12449

REASON FOR THE CHANGE: Previously, it was virtually impossible to remove or install the snubbers without getting air trapped in the lines. This modification will allow the proper hookup between the snubbers and reservoirs.

SAFETY EVALUATION: The changes in the reservoir tubing orientation on the above listed snubbers will not change the function of the snubbers. The changes will provide added insurance that air will not be trapped in the tubing and adversely affect the snubber operation. On this basis, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-273

SYSTEM: Process and Area Radiation Monitoring

COMPONENT: RE 5029A and 5030A

CHANGE, TEST OR EXPERIMENT: FCR 80-273 was written to decrease the sensitivity of Radiation Monitors RE5029A and RE5030A by at least a factor of 1000. This was accomplished by replacing the detectors in RE5029A and RE5030A with Victoreen detectors, model number 843-20B, which have a 100:1 sensitivity reduction. The work, as required by this FCR, was completed January 10, 1981.

REASON FOR THE CHANGE: In December of 1980, the containment particulate airborne monitors were found nearly off scale. When the readings are off scale, the instruments will be inoperable, and in accordance with T.S. 3.4.6.1.a, operation of the reactor can only continue for 30 days.

SAFETY EVALUATION: The subject radiation detectors RE5029A and RE5030A are utilized for monitoring of the containment during normal operation and for detection of containment radioactivity resulting from a reactor coolant pressure boundary (RCPB) leak.

The Davis-Besse Unit 1 Technical Specifications address these monitors in Sections 3/4.3.3.1, 3/4.3.3.6, and 3/4.4.4.6.1. The requirements specify that the measurement range be 10 to 10^6 cpm. There is no limit on sensitivity or response time.

The effect of a reduction by a factor of 100 in sensitivity has been evaluated. The sensitivity as noted below is below the maximum permissable 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 alarm at MPC.

Monitor Sensitivity at Three (3) GPM

Monitor	Isotope of Interest	Sensitivity (uc/cc)	MPC (uc/cc)
Particulate	Cs^{137}	3×10^{-9}	6×10^{-8}

The NRC Safety Evaluation Report (SER) Supplement 1, Section 5.2.4, stated that the RCPB leakage detection systems "are generally in accordance with the recommendations of Regulatory Guide 1.45." This guide states that a one (1) gallon per minute (gpm) leak rate should be detected within one hour. Calculations indicate that monitors RE 5029A and RE 5030A will be capable of detecting a RCPB leak rate of one (1) gpm within one hour.

Based on the above, it is concluded that the proposed reduction in sensitivity 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.