AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO.	50-346
UNIT	Davis-Besse Unit 1
DATE	February 10, 1986
COMPLETED BY	Morteza Khazrai
TELEPHONE	(419) 249-5000 Ext.

MONTH January, 1986

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.

(9/77)

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1/1

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE February 10, 1986
COMPLETED BY Mort[illegible]a Khazrai
TELEPHONE (419) 249-5000, Ext. 7290

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: January 1986
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): 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:

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	744	65,809
12. Number Of Hours Reactor Was Critical	0.0	0.0	35,877.1
13. Reactor Reserve Shutdown Hours	0.0	0.0	4,058.8
14. Hours Generator On-Line	0.0	0.0	34,371.8
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	0.0	0.0	81,297,600
17. Gross Electrical Energy Generated (MWH)	0.0	0.0	26,933,622
18. Net Electrical Energy Generated (MWH)	0.0	0.0	25,233,177
19. Unit Service Factor	0.0	0.0	52.2
20. Unit Availability Factor	0.0	0.0	54.9
21. Unit Capacity Factor (Using MDC Net)	0.0	0.0	44.6
22. Unit Capacity Factor (Using DER Net)	0.0	0.0	42.3
23. Unit Forced Outage Rate	100.0	100.0	26.[illegible]

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 28, 1986
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

DOCKET NO.	50-346
UNIT NAME	Davis-Besse Unit
DATE	February 10, 1986
COMPLETED BY	Morteza Khazrai
TELEPHONE	(419) 249-5000 Ext. 7290

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH January, 1986

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
7 Con't	85 06 09	F	744	A	4	LER 85-013	JK	SC	The unit remained shutdown following the reactor trip on June 9, 1985 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
JANUARY 1986

The unit remained shutdown the entire month of December following the reactor trip on June 9, 1985. Investigation of the causes of the event and corrective actions continue. See NUREG 1154 for further details.

Below are some of the major activities performed during this month:

1) Completed reinstallation of Power Operated Relief Valve (PORV)

2) Completed condenser hydro leak test

3) Repair and replacement of all the hydro motor actuators

4) Completed work on Decay Heat Loop #2

5) Repair condenser internal damage (heater shroud metal)

6) Replacing gasket and studs on steam generator primary side manway

R UELING INFORMATION DATE: J nuary, 1986

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: March 1, 1987

3. Scheduled date for restart following refueling: May 10, 1987

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, 1986

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.

BMS/005

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-221

SYSTEM: Auxiliary Feedwater

COMPONENT: Auxiliary Feed Pump Turbine (AFPT) Speed Controller

CHANGE, TEST OR EXPERIMENT: This FCR replaced the CW/AUX and CCW/AUX relays with relays and contacts that have adequate current ratings.

This FCR also replaced the dynamic braking resistor with a resistor with adequate wattage and voltage rating.

This FCR and work was completed June 1, 1979.

REASON FOR CHANGE: The change was instituted due to the failures of the Couch/Deutsch relays used in the CW/AUX and CCW/AUX application.

Supplement 3 to FCR 77-221 involved the addition of a 50 ohm, 90 watt resistor in the AFPT control circuit. This resistor had been previously removed by the EIT for adverse environment.

SAFETY EVALUATION SUMMARY: The safety evaluation of the Couch relays for CW/AUX and CCW/AUX in disconnect switch cabinets CDE 12A-1 and CDF 12A-1 did not work satisfactorily for the following reasons:

1. DC interrupting rating of contacts were not capable to break the Bodine motor locked rotor current.

   Contacts arced on opening, causing excessive wear.

2. The Agastat relay is a more robust relay with larger contacts and much better DC interrupting rating.

3. The addition of a 50 ohm, 90 watt resistor in the AFPT control circuit. This additional resistor in JT2703 and JT2704 has been previously reviewed by the EIT for adverse environment.

These changes do not change the function of the AFPT speed control circuit. Reliability of the circuit and speed control will be enhanced by this modification. This modification will not adversely affect the safety function of the Auxiliary Feedwater System.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-297

SYSTEM: Reactor Coolant System

COMPONENT: RCP Neutron Power Interlock

CHANGE, TEST OR EXPERIMENT: This FCR is requesting approval of testing to obtain operational data on the restarting of the fourth Reactor Coolant Pump (RCP) above 22% power. This will confirm the simulator results obtained by B&W. The basic test would be to establish reactor power at 50% ± 2% with three RCPs running and then start the fourth RCP.

This FCR requested relief from Section 15.2.6.1 of the FSAR (Page 15-47) which describes the 22% interlock.

This FCR test and request were completed February 28, 1979.

REASON FOR CHANGE: By resetting the setpoint of RCP interlock to a higher setpoint, the fourth RCP could be started at a higher power level then the present power level. B&W recommends this test be performed in SOM #378 dated June 7, 1978.

SAFETY EVALUATION SUMMARY: An analysis of the data provided by the B&W simulator at 50% full power reveals that the highest and lowest pressure after the starting of the fourth RCP are 2192 psig and 2090 psig, respectively, which are well within the pressure trip setpoints. If the trip setpoints are reached, the reactor will trip and any limits of the accident analysis in Chapter 15 of the FSAR will not be exceeded.

The 22% interlock is a non-nuclear safety related interlock and was introduced in the original Integrated Control System (ICS) to avoid the possibility of frequent reactor coolant pressure trips.

Based on the analysis of the data, it is informed that starting a fourth RCP at 50% full power will not compromise nuclear safety and does not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-278

SYSTEM: Safety Features Actuation System (SFAS)

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR changed the automatic transfer of the suctions of the decay heat and containment spray pumps from the BWST to the containment emergency sump to a manual transfer. Work was completed March 18, 1982.

REASON FOR CHANGE: This modification protects the decay heat and containment spray pumps from the possibility of damage due to an inadvertent transfer of pump suction to the dry containment emergency sump.

SAFETY EVALUATION SUMMARY: The safety function of this transfer is to assure proper net positive suction to the decay heat and containment spray pumps during recirculation mode of operation.

Manual transfer from the Control Room will be performed when BWST level indication is between 49.5 inches and 96 inches.

Minimum level requirements for the BWST are met when transfer from BWST is performed within four minutes after BWST level has reached 96 inches. This manual transfer provides the same safety function as the previous automatic transfer; hence, this does not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-046

SYSTEM: N/A

COMPONENT: Borated Water Storage Tank (BWST)

CHANGE, TEST OR EXPERIMENT: FCR 83-046 provided the addition of spray on foam insulation to the exterior sides of the Borated Water Storage Tank (BWST). To protect the BWST from any direct contact with the foam insulation material, the tank had a protective base coating applied to its surface. The BWST was also sealed and top coated. Work was completed October 5, 1985.

REASON FOR CHANGE: This change is necessary due to stratification of temperature levels caused by tank heat losses, the uneven mixing of water, and suspected icing on the inside of the BWST walls. Because of this, inaccurate temperature readings were received during cold weather. Insulation added to the tank's external walls limits the tank heat losses such that the stratification of temperature levels are minimized and icing conditions are eliminated.

SAFETY EVALUATION SUMMARY: The safety function of the BWST is to provide a water source for the Emergency Core Cooling System and the Containment Spray System. The BWST temperature should be maintained between 50°F and 90°F. The low temperature level listed in the Technical Specifications is 35°F. However, a minimum of 50°F should be maintained based on a containment spray initiation. Because this modification will not adversely affect the integrity of the BWST or the operation of the BWST from performing its design function, an unreviewed safety question does not exist.

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TOLEDO EDISON

February 10, 1986

Log No. KB86-0177
File: RR 2 (P-6-86-01)

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, January 1986
Davis-Besse Nuclear Power Station Unit 1

Enclosed are ten copies of the Monthly Operating Report for Davis-Besse Nuclear Power Station Unit 1 for the month of January 1986.

If you have any questions, please feel free to contact Morteza Khazrai at (419) 249-5000, Extension 7290.

Yours truly,

Louis F. Storz

Louis F. Storz
Plant Manager
Davis-Besse Nuclear Power Station

LFS/MK/ljk

Enclosures

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

Mr. James M. Taylor, Director, w/2
Office of Inspection and Enforcement

Mr. Walt Rogers, w/1
NRC Resident Inspector

LJK/002

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