AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE July 9, 1984

COMPLETED BY Bilal Sarsour

TELEPHONE (419) 259-5000, Ext. 384

MONTH June, 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	804	17	799
2	804	18	798
3	798	19	804
4	804	20	807
5	799	21	810
6	803	22	810
7	800	23	804
8	796	24	449
9	788	25	272
10	797	26	738
11	805	27	797
12	800	28	800
13	793	29	802
14	802	30	805
15	810	31	
16	807		

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)

IE24
1/1

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE July 9, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000
Ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: June 1984
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): 918
7. Maximum Dependable Capacity (Net MWe): 874
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	720	4,367	51,888
12. Number Of Hours Reactor Was Critical	705.6	3,788.4	31,290.9
13. Reactor Reserve Shutdown Hours	0.0	134.8	4,014.1
14. Hours Generator On-Line	702.8	3,748.9	29,900.7
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	1,804,239	9,656,539	70,700,353
17. Gross Electrical Energy Generated (MWH)	586,349	3,170,548	23,462,741
18. Net Electrical Energy Generated (MWH)	554,780	2,988,070	21,986,769
19. Unit Service Factor	97.6	85.8	57.6
20. Unit Availability Factor	97.6	85.8	61.0
21. Unit Capacity Factor (Using MDC Net)	88.2	78.3	48.5
22. Unit Capacity Factor (Using DER Net)	85.0	75.5	46.8
23. Unit Forced Outage Rate	2.4	14.2	18.0

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
Refueling Outage: Scheduled start 9/1/84, Scheduled end 11/9/84

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		

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE July 9, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

REPORT MONTH June, 1984

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
4 .	84 06 24	F	17.2	H	1	NP-33-84-10	AA	BKR	The control rod drives were de-energized during the performance of surveillance testing of the Reactor Protection System. See Licensee Event Report NP-33-84-10 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
June, 1984

6/1/84 - 6/9/84

Reactor power was maintained at approximately 94% power with the generator gross load at approximately 845 ± 10 MWe until 0100 hours on June 9, 1984, when a manual power reduction to approximately 85% was initiated to perform turbine valve testing. (Reactor power was limited to 94% due to an inoperable main steam safety valve.)

After the completion of turbine valve testing, reactor power was slowly increased and attained approximately 94% at 0500 hours on June 9, 1984.

6/10/84 - 6/25/84

At 1354 hours on June 24, 1984, while the unit was at approximately 95% power, the control rod drives were inadvertently deenergized during the performance of surveillance testing of the Reactor Protection System. Surveillance testing was required due to a loss of essential inverter YV4 caused by a failure of a zener diode and resistor in the logic power supply circuit board which resulted in a loss of Y4 bus. See Licensee Event Report NP-33-84-10 for further details.

The reactor was critical at 0420 hours on June 25, 1984. The turbine generator was synchronized on line at 0704 hours on June 25, 1984.

6/26/84 - 6/30/84

Reactor power was slowly increased and attained approximately 94% power at 1500 hours on June 26, 1984, (with five hour holds at 75% and 90% due to removal of axial power shaping rods) and maintained at this power level for the rest of the month.

REFUELING INFORMATION DATE: June, 1984

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: September 1, 1984

3. Scheduled date for restart following refueling: November 9, 1984

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: July, 1984

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) 140 - 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: 1993 - assuming ability to unload the entire core into the spent fuel pool is maintained.

BMS/005

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-059

SYSTEM: Diesel Generators

COMPONENT: Fuel Oil Day Tanks

CHANGE, TEST OR EXPERIMENT: The emergency diesel generator fuel oil day tanks saddle supports were enlarged to allow the addition of anchor bolts. Work was completed January 31, 1983.

REASON FOR CHANGE: Seismic calculations performed in accordance with Section 3.C.(3)(r) of Operating License NPF-3, showed the subject anchor bolts to have a factor of safety of 2.2. The enlargement of the saddle supports and addition of anchor bolts increased the factor of safety to 4.0, the factor of safety recommended by the manufacturer.

SAFETY EVALUATION: The increased factor of safety for the expansion anchors was in accordance with manufacturer's recommendations and Nuclear Regulatory Commission IE Bulletin 79-02. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-011

SYSTEM: Reactor Coolant Pump (RCP)

COMPONENT: Seal Return Valves

CHANGE, TEST OR EXPERIMENT: The automatic closure circuitry for the RCP seal return valves was disconnected. Closure of the seal return valves is now done manually by the Control Room operator based on limits for seal return temperature and seal leakage flow except for Safety Features Actuation System (SFAS) closure of the valves for containment isolation. Tripping of the RCPs for protection of the mechanical seals is also a manual operation based on RCP seal limits. Work was completed April 14, 1981.

REASON FOR CHANGE: This change has eliminated a potential source for inadvertent seal return valve closures and RCP trips.

SAFETY EVALUATION: This FCR was classified as nuclear safety related since the seal return valves perform the safety function of containment isolation under the SFAS. This function was not affected by this FCR. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-017

SYSTEM: Concrete Masonry Block Walls

COMPONENT: Wall No. 2337

CHANGE, TEST OR EXPERIMENT: Structurally stiffened angles were added at the base on both sides of wall 2337 in Mechanical Penetration Room #2 at Elevation 585'. The connections to the wall and the floor were made with thru-bolts and expansion anchors. Work was completed October 20, 1983.

REASON FOR CHANGE: Reanalysis of block walls required by Nuclear Regulatory Commission IE Bulletin 80-11 had shown that during a seismic event the connection between the floor and wall 2337 could become overstressed.

SAFETY EVALUATION: Wall 2337, designed to function as a radiation shield, could have failed when subjected to a seismic event impacting safety related conduit attached to this wall. This modification has increased the factor of safety for the wall to floor connection to acceptable limits established in section 3.8 of the Final Safety Analysis Report. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-026

SYSTEM: Concrete Masonry Block Walls

COMPONENT: Wall No. 311D

CHANGE, TEST OR EXPERIMENT: A reinforced concrete column was attached to the east edge of wall 311D from Elevation 595' to 610'. Wall 311D is a shield wall located in the Diesel Generator room adjacent to the containment wall at Elevation 585'. Work was completed November 10, 1983.

REASON FOR CHANGE: Reanalysis of block walls required by Nuclear Regulatory Commission IE Bulletin 80-11 had shown that during a seismic event the connections in a four foot section of the lower east edge of wall 311D could have become overstressed.

SAFETY EVALUATION: This modification has reduced the stresses in the wall connections of wall 311D to the allowable limits of section 3.8 of the Final Safety Analysis Report. This has been reviewed and found not to adversely affect the function of wall 311D to function as a shield wall for the diesel generator and did not result in an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-031

SYSTEM: Concrete Masonry Block Walls

COMPONENT: Wall No. 2297

CHANGE, TEST OR EXPERIMENT: Three steel angle braces were added to the web of the floor beam attached to the top of wall 2297. This wall separates Boric Acid Evaporator Rooms 234 and 235 at Elevation 565'. Work was completed December 8, 1983.

REASON FOR CHANGE: Reanalysis of block walls required by Nuclear Regulatory Commission IE Bulletin 80-11 had shown that during a seismic event the floor beam above wall 2297 could have become overstressed.

SAFETY EVALUATION: This modification has reduced the possible stresses in the floor beam above wall 2297 to within the limits established in section 3.8 of the Final Safety Analysis Report for Class 1 structures. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-266

SYSTEM: Reactor Protection System (RPS)

COMPONENT: Reactor Coolant Variable Low Pressure Trip Setpoints

CHANGE, TEST OR EXPERIMENT: On October 14, 1981, work implemented by this FCR was completed. This involved changing the reactor coolant variable low pressure trip setpoint on all four RPS channels. The new pressure setpoint is 12.60 times the hot leg temperature minus 5651.23 with a tolerance of +10 psi and -0 psi.

REASON FOR CHANGE: Engineering evaluation of data received from the April 1982 test on the RPS revealed a possible maximum string error of 1.63°F. The new setpoint trips the RPS at a hot leg temperature of 606.05°F when the pressure has dropped to 1985 psig. This gives a safety margin of 1.65°F.

SAFETY EVALUATION: The safety function of the variable low pressure trip of the RPS is to ensure that the reactor trips before reaching the safety margin as calculated in the accident analysis. This modification has enhanced this function. Hence, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-305

SYSTEM: Reactor Protection System (RPS)

COMPONENT: Resistance Thermal Device (RTD)

CHANGE, TEST OR EXPERIMENT: This FCR temporarily changed the Reactor Coolant System (RCS) hot leg input for RPS Channel 4 from RTD TE-RC-3A2 to TE-RC-3A1. The system was restored to its original configuration April 12, 1982.

REASON FOR CHANGE: RTD TE-RC-3A2 failed high causing a trip of RPS Channel 4. The use of RTD TE-RC-3A1 enabled RPS Channel 4 to remain operable.

SAFETY EVALUATION: The safety function of RTD TE-RC-3A2 is to provide input to the RPS high temperature and low pressure trips. These units were manufactured to the same specifications and are able to perform the same function. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-032

SYSTEM: Main Generator

COMPONENT: Megawatt Meter

CHANGE, TEST OR EXPERIMENT: The megawatt meter, which is required to display the plant's electrical power in the Control Room, was changed from Electronic Research 4010 to General Electric 800 series. Work was completed March 21, 1983.

REASON FOR CHANGE: The original meter was not functioning properly and replacement parts were no longer available.

SAFETY EVALUATION: This meter is not part of a safety system and does not require Class 1E qualification. The unit is, however, mounted in a seismically qualified control panel. Seismic analysis performed by Bechtel had shown that the original seismic qualification remains valid. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-017

SYSTEM: Hydrogen Recombination System

COMPONENT: Piping

CHANGE, TEST OR EXPERIMENT: Piping and a power supply were added to the fuel handling area to allow movable hydrogen recombiners to be connected to containment. Work was completed September 13, 1983.

REASON FOR CHANGE: Nuclear Regulatory Commission rule 10CFR50.44(c)(3)(ii) requires the capability for external movable hydrogen recombiners to be connected to containment approximately 30 days after the design basis Loss of Coolant Accident (LOCA).

SAFETY EVALUATION: The safety function of this system is to prevent the hydrogen concentration in containment from reaching a flammable level following a LOCA event. The hydrogen dilution/purge system will be the primary means of controlling the post accident containment combustible gases. The hydrogen recombination system is intended to be the backup system.

Since the fuel handling area has several radiation monitors, should there be leakage during hydrogen recombiner operation, there will be automatic transfer to the Emergency Ventilation System.

The system utilizes containment penetrations 51 and 69 which are considered to be far enough apart to prevent short circuit recirculation of containment atmosphere.

The piping modifications were made in a manner so as to not degrade containment isolation capability and were designed such that they will not damage any other safety related equipment upon a seismic event.

Pursuant to the above, no unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-109

SYSTEM: Main Feedwater

COMPONENT: Piping

CHANGE, TEST OR EXPERIMENT: Bechtel drawing 7749-C-180 was revised to reflect the as-built condition of Main Feedwater Restraint No. 5 which utilized steel shim plates rather than grout under one of the anchor base plates. This was completed March 22, 1984.

REASON FOR CHANGE: The installation of the shims resulted from analysis performed by Bechtel after it was discovered that the grout was never installed. This drawing now more accurately reflects existing conditions in the plant.

SAFETY EVALUATION: The safety function of Main Feedwater Restraint No. 5 is to restrain the Main Feedwater Line Train No. 1 from whipping following a postulated line break and to control the direction of jet impingement forces resulting from this line break. This condition did not affect the ability of the restraint to perform its intended safety function. As unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-114

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Torque Switch Settings

CHANGE, TEST OR EXPERIMENT: The torque switch settings for the pressurizer electromagnetic relief isolation valve was changed on electrical drawing E-15 to 2.5 for open and 2.0 for close. This was verified February 29, 1984.

REASON FOR CHANGE: This drawing now more accurately reflects existing conditions in the plant. The new setpoints were based on a recommendation from the Torrey Pines Technology report on Limitorque motor operated valves.

SAFETY EVALUATION: The safety function of the torque switch is to break the circuit during the closing of the valve to prevent overtraveling of the valve stem or to break the circuit in case of high mechanical force to prevent damage to the valve. This FCR has not created any new adverse environments and did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-121

SYSTEM: Drawings

COMPONENT: E-302A

CHANGE, TEST OR EXPERIMENT: Electrical drawing E-302A Sheet 39L was revised to incorporate information received from Raychem on an accepted alternate method for sealing electrical connections. This change was verified December 5, 1983.

REASON FOR CHANGE: In cases where the termination lug is considerably larger than the cable outer diameter, it is necessary to modify the termination sealing process. This is done by using a small piece of Raychem heat shrink tubing to build up the outer diameter of the cable. This splice design is acceptable as long as a minimum overlap of 1" is maintained.

SAFETY EVALUATION: The safety function of Raychem heat shrink tubing is the sealing of electrical terminations with an environmentally qualified protective seal. These terminations are made throughout the station and thus affect various safety functions. This modification has added to the environmental qualification of these terminations. This did not constitute an unreviewed safety question.

TOLEDO EDISON

July 9, 1984

Log No. K84-821
File: RR 2 (P-6-84-06)

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, June, 1984
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 June, 1984.

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

Yours truly,

Terry D Murray

Terry D. Murray
Station Superintendent
Davis-Besse Nuclear Power Station

TDM/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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