AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse #1

DATE

COMPLETED BY Bilal M. Sarsour

TELEPHONE (419)259-5000 Ext. 384

MONTH November, 1984

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	
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)

IE24
1/1

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE
COMPLETED BY Bilal M. Sarsour
TELEPHONE (419)259-5000 (ext. 384)

OPERATING STATUS

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

Notes

8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

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	8,040.0	55,561.0
12. Number Of Hours Reactor Was Critical	0.0	5,529.0	33,031.5
13. Reactor Reserve Shutdown Hours	0.0	134.8	4,014.1
14. Hours Generator On-Line	0.0	5,489.5	31,641.3
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	0.0	13,941,608	74,985,422
17. Gross Electrical Energy Generated (MWH)	0.0	4,554,151	24,846,344
18. Net Electrical Energy Generated (MWH)	0.0	4,291,557	23,290,256
19. Unit Service Factor	0.0	68.3	56.9
20. Unit Availability Factor	0.0	68.3	60.1
21. Unit Capacity Factor (Using MDC Net)	0.0	61.1	48.0
22. Unit Capacity Factor (Using DER Net)	0.0	58.9	46.3
23. Unit Forced Outage Rate	0.0	11.0	17.3

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		

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse #1
DATE
COMPLETED BY Bilal M. Sarsour
TELEPHONE (419)259-5000 Ext. 384

REPORT MONTH November, 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
5	84-09-14	S	720	C	4	N/A	N/A	N/A	The unit outage which began on September 14, 1984 was still in progress through the end of November, 1984. See Operation 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

NOVEMBER, 1984

The unit outage which began on September 14, 1984, was still in progress through the end of November 1984.

The following are the more significant outage activities performed during November, 1984.

1. Fuel reload was successfully completed.

2. Condenser tube eddy current testing was completed with 12 tubes plugged out of 8,763 tested.

3. Replacement of Reactor coolant pump seal for reactor coolant pumps.

4. Significant cooling tower repairs involving grouting of the basin and support column repairs completed.

5. Functional Snubber testing was completed with no problems noted, however visual inspection of hydraulic snubbers revelaed two failures.

6. Internal turbine work was completed and turbine reassembly is in progress.

7. All four high pressure injection (HPI) swing check valves have been modified.

8. All lower thermal shield bolt and surveillance specimen holder tube bolt work was completed.

REFUELING INFORMATION DATE: November, 1984

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: Spring 1986

3. Scheduled date for restart following refueling: December 28 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: 78-311

SYSTEM: Auxiliary Feedwater

COMPONENT: Auxiliary Feedwater Pump Turbines 1 & 2 Speed Monitors

CHANGE, TEST OR EXPERIMENT: FCR 78-311 was incorporated to update the electrical drawings for SY-815 and SY-816, Auxiliary Feedwater Pump Turbines (AFPT's) 1 and 2 speed monitors, respectively. This will correct the drawing changes made as a result of MWO IC-323-77. This FCR is a drawing change only FCR. Changes were completed July 14, 1983.

REASON FOR CHANGE: This change was made to reflect the correct condition of the plant electrical drawings that are concerned with the AFPT 1 and 2 speed monitors, SI 815A and SI 816A, respectively.

SAFETY EVALUATION: The wiring change is a correction to ensure the correct polarity of the analog signal of the E/I convertor, therfore, reflecting the present condition of the AFPT 1 and 2 speed monitors. Since the safety function of the AFPT 1 and 2 speed monitors are not reduced, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-475

SYSTEM: N/A

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR performed a 10CFR 50.59 review justify running a loss of external load (including loss of off-site power) test. This FCR also revised USAR Section 14.1.8.2 to substitute the unit load transient test with Unit Load Rejection Test, TP 800.13. Work on this FCR was completed June 13, 1984.

REASON FOR CHANGE: This change was incorporated to address NUREG 1.68. A review was performed to justify running loss of external load including the Loss of Offsite Power Test.

SAFETY EVALUATION: Pursuant to the above, it is concluded that there is no unreviewed safety question involved with this FCR.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-223

SYSTEM: 480 VAC Systems

COMPONENT: Essential and Non-essential Breakers

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to provide ground fault protection on essential unit and non-essential unit substation breakers. This was accomplished by incorporating solid state overcurrent/ground fault trip devices. Work involved with this FCR was completed August 31, 1983.

REASON FOR CHANGE: FCR 77-223 was issued on the result of ground faults that occurred previously on the 480 VAC Systems.

SAFETY EVALUATION: The changes involved with this FCR do not create any new adverse environment. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-355

SYSTEM: 480V Power Supply System

COMPONENT: PORV Block Valve (RC 11)

CHANGE, TEST OR EXPERIMENT: FCR 79-355 was implemented to allow the PORV Block Valve (RC 11) to be supplied with control and motive power from the emergency power source when offsite power is not available. This FCR also provided the revision of circuit breakers in MCC F12A during Mode 5 and associated control scheme for the 480V power supply. Work was completed May 23, 1980.

REASON FOR CHANGE: These changes were required by NUREG-0578 and the TMI-2 Lessons Learned Task Force Report.

SAFETY EVALUATION: The function of the PORV block valve, RC 11 is to prevent the de-pressurization of the reactor coolant system by isolating the electromatic relief valve (RC 11) if it has stuck open. By upgrading the power supply to the block valve operator to Class IE, the normal function of the block valve is not affected. This change will facilitate the operation of the block valve during a loss of offsite power since the valve operator will be supplied from the essential bus.

This FCR does not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-408

SYSTEM: 20-10

COMPONENT: Containment Vessel Sump

CHANGE, TEST OR EXPERIMENT: FCR 79-408 incorporated a continuous narrow range indicator/recording of containment water level in the control room. This FCR also allows for the provision of two electrical analog input signals to the plant process computer from source transmitters LY-4617 and LY-4618 with a variable range of 0' to 4' using computer points L317 and L318, respectively. Work was completed August 9, 1982.

REASON FOR CHANGE: This is direct action for the NRC letter received September 13, 1979 on follow up actions resulting from the NRC staff reviews regarding the Three Mile Island - 2 incident.

SAFETY EVALUATION: The work authorized by this FCR does not create any new adverse environmental effect and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-409

SYSTEM: N/A

COMPONENT: Containment Vessel

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to provide a continuous wide range indication/recording of containment water level in the control room. The indicators covered a range from the bottom of containment to the elevation equivalent to a 600,000 gallon capacity. The indication was taken from transmitters LT-5494 and LT-5495. FCR 79-409 also incorporated two electrical analog input signals into the plant computer from LT-5494 and LT-5495 which are representative of computer points L321 and L322, respectively. Work was completed August 6, 1982.

REASON FOR CHANGE: This is direct action for the NRC letter received September 13, 1979 on follow up actions resulting from the NRC staff reviews regarding the Three Mile Island Unit-2 accident.

SAFETY EVALUATION: This change will not degrade any safety-related instrumentation and will enhance information of containment water level. Therefore, this is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-115

SYSTEM: Non-nuclear Instrumentation

COMPONENT: Incore Thermocouples

CHANGE, TEST OR EXPERIMENT: This FCR provided eight incore thermocouple inputs to each Tsat meter. Two thermocouples from each quadrant was input to each Tsat meter. All eight were input to a selector switch. Therefore, the selected input is sent to the Tsat meter. Work on FCR 80-115 was completed October 5, 1982.

REASON FOR CHANGE: The reason for the changes in this FCR is listed in NUREG-0667, Section 2.2, Item 7. This states: "All B&W plants should provide the flexibility to substitute appropriate combinations of incore thermocouples for the loop resistance temperature detectors presently used for primary temperature input to the subcooling meters. All B&W plants should provide the capability of having continuous or trending display of incore thermocouples."

SAFETY EVALUATION: The effect of this change will be to increase the operability of the Tsat instrumentation because

1. More optional inputs are available (8 vs 1).
2. Able to select the worst case temperature (1 out of 8).
3. More indicative of reactor coolant temperature (measured at core exit).
4. All of the components of this system will be class IE (except for the actual temperature elements).

This change will not degrade the Tsat meter function. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-186

SYSTEM: Emergency Diesel Generator System 24-01

COMPONENT: Emergency Diesel Generators Accessory Racks, KS-1 and KS-2

CHANGE, TEST OR EXPERIMENT: This FCR allowed for the modification of the accessory racks for the Emergency Diesel Generators, KS-1 and KS-2. This is done in an effort to increase the natural frequency of the accessory racks. Work involved with this FCR was completed August 19, 1982.

REASON FOR CHANGE: During the Seismic reevaluation for a 0.20g SSE, required by operating license NPF-3, Section 2.C (3)(r) it was determined that a shift in the accessory racks natural frequency had occurred, thus making their factor of safety against failure less than desirable.

SAFETY EVALUATION: These modifications are for enhancing the factor of safety of the Emergency Diesel Generators accessory racks. Because of the increase in the factor of safety, these modifications made, no new adverse environment was created. This will not result in an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-004

SYSTEM: Component Cooling Water (CCW)

COMPONENT: Pipe Anchor A399/A402

CHANGE, TEST OR EXPERIMENT: FCR 84-004 was performed to modify CCW pipe anchor A399/A402. This involved increasing the size of the existing fillet welds and adding stiffener plates to the anchor. Work was completed April 10, 1984.

REASON FOR CHANGE: While conducting the piping support walkdown under NRC IE bulletin A-14, CCW pipe anchor A399/A402 was identified as missing welds in four locations. Because these missing welds were in inaccessible areas, the above modifications were made to allow long term plant operation without overstressing the anchor components.

SAFETY EVALUATION: These modifications will enhance the design of the piping anchor by decreasing the structural stresses, thus ensuring that the piping stress will not exceed the allowable limits for long term plant operation. Therefore, this modification will not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-006

SYSTEM: Heating, Ventilation, and Air Conditioning (HVAC)

COMPONENT: Fire Dampers

CHANGE, TEST OR EXPERIMENT: This FCR allowed for the revision of the Auxiliary and Turbine Building HVAC and P & ID drawings which are associated with fire dampers FD 1001, FD 1003, FD 1010, FD 1026, FD 1029, FD 1063, FD 1064 and FD 1079. This is a drawing change only FCR. The drawing modification was completed July 6, 1984.

REASON FOR CHANGE: These drawings were revised to reflect the "as-built" conditions in the plant.

SAFETY EVALUATION: Since this change does not require revision of the Fire Hazard Analysis Report (FHAR) the change does not adversely affect the analysis set forth in the FHAR.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-050

SYSTEM: Fire Protection

COMPONENT: Drawings A-201F thru A-210F

CHANGE, TEST OR EXPERIMENT: FCR 84-050 allowed for the revision of Fire Delineation drawings A-201F thru A-210F. No physical changes were required in the plant for the implementation of this FCR. Drawing changes were completed by September 10, 1984.

REASON FOR CHANGE: This FCR was generated to reflect the as-built conditions not previously addressed as a result of modifications implemented to upgrade the plant fire protection features.

SAFETY EVALUATION: These changes do not adversely affect the analysis set forth in the Davis-Besse FHAR. Therefore, an unreviewed safety question does not exist.

TOLEDO EDISON

December 7, 1984

Log No. K84-1377
File: RR 2 (P-6-84-11)

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, November 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 November 1984.

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/bec

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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