AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE February 8, 1985

COMPLETED BY Bilal M. Sarsour

TELEPHONE (419) 259-5000, Ext. 384

MONTH January 1985

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	30
4	0	20	172
5	0	21	255
6	0	22	280
7	0	23	284
8	0	24	284
9	0	25	287
10	0	26	539
11	0	27	634
12	0	28	667
13	0	29	807
14	0	30	800
15	0	31	778
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 February 8, 1985
COMPLETED BY Bilal M. Sarsour
TELEPHONE (419) 259-5000, Ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: January 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): 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	744	744.0	51,049.0
12. Number Of Hours Reactor Was Critical	394.1	394.1	33,425.6
13. Reactor Reserve Shutdown Hours	0.0	0.0	4,014.1
14. Hours Generator On-Line	304.0	304.0	31,945.3
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	484,324	484,324	75,469,746
17. Gross Electrical Energy Generated (MWH)	152,971	152,971	24,999,315
18. Net Electrical Energy Generated (MWH)	128,895	128,895	23,419,151
19. Unit Service Factor	40.9	40.9	56.0
20. Unit Availability Factor	40.9	40.9	59.0
21. Unit Capacity Factor (Using MDC Net)	19.8	19.8	47.0
22. Unit Capacity Factor (Using DER Net)	19.1	19.1	45.3
23. Unit Forced Outage Rate	0.0	0.0	17.2

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

REPORT MONTH January 1985

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE February 8, 1985
COMPLETED BY Bilal M. Sarsour
TELEPHONE (419) 259-5000, 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
	84 09 14	S	436.7	C	4	NA	NA	NA	Unit outage which began on September 14, 1984 was completed on January 19, 1985 when the unit was placed on line.
1	85 01 19	S	3.3	B	1	NA	NA	NA	Turbine overspeed trip test. 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)

REFUELING INFORMATION DATE: January, 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) 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.

OPERATIONAL SUMMARY
January, 1985

1/1/85 - 1/19/85

Zero power physics testing began on January 14, 1985, and the reactor was deborated to criticality at 0300 hours on January 15, 1985.

At 1230 hours on January 15, 1985, The reactor was tripped by the Anticipatory Reactor Trip System (ARTS) from Steam and Feedwater Rupture Control System (SFRCS) actuation due to a low steam generator level. The initiating event was the decrease in steam generator level with the feedwater valves in manual. The steam generator levels were not manually increased to normal low level limit setpoint since it was not desired to disturb the data being taken for the zero power physics testing. The underlying root cause of this event is the lack of adequate margin between the low level limit setpoint 35 inches and the SFRCS low level trip setpoint 26.5 inches.

The reactor was returned to criticality at 2325 hours on January 15, 1985.

Physics testing was completed on January 18, 1985, and the turbine generator was synchronized on line at 0444 hours on January 19, 1985, marking the completion of the unit outage which began on September 14, 1985.

1/20/85 - 1/22/85

Reactor power was slowly increased until 1927 hours on January 20, 1985, when reactor power was reduced to approximately 24% due to a dropping of Control Rod 7-12. Troubleshooting could not determine any fault, and the rod was repulled at 0128 hours on January 21, 1985.

Reactor power was slowly increased to 40% of full power which was attained on January 21, 1985. Reactor power was maintained at this power level to perform physics testing.

1/23/85 - 1/31/85

On January 23, 1985, Control Rod 7-12 dropped. A detailed investigation determined a pin connector was not properly engaged. Repairs were completed and the rod returned to its normal position at 1758 hours on January 23, 1985. Physics testing at the 40% power level was completed on January 25, 1985.

Power escalation continued until 1300 hours on January 26, 1985, when 75% of full power was achieved.

The reactor power level was maintained at 75% until the completion of physics testing which occurred on January 28, 1985.

At 1600 hours on January 28, 1985, reactor power was slowly increased at a steady rate to approximately 92% which was reached at 2400 hours on January 28, 1985.

Reactor power was maintained at approximately 92% (reactor power was limited at 92% due to an inoperable main steam safety valve) until 2000 hours on January 30, 1985, when reactor power was reduced to approximately 88% to take the #2 Condensate Pump out of service due to high motor bearing temperatures.

Reactor power was maintained at approximately 88% until 1300 hours on January 31, 1985 when reactor power was increased steadily to approximately 90% after Condensate Pump #2 was brought back in service, and remained at that level for the remainder of the month.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-016

SYSTEM: N/A

COMPONENT: Wall No. 2047

CHANGE, TEST OR EXPERIMENT: FCR 81-016 modified pipe hangers 31-HCC-5-H5, 31-HCC-5-H6, and 31-HCC-5-H7 so that they were connected to the ceiling instead of block wall 2047. Work was completed September 13, 1984.

REASON FOR CHANGE: The reanalysis of the block wall required by NRC I.E. Bulletin 80-11 showed that during a seismic event the connection between wall 2047 and the floor would be overstressed. The loads which would cause this overstressed condition were the loads transmitted to the wall from pipe hangers 31-HCC-5-H5, 31-HCC-5-H6, and 31-HCC-5-H7.

SAFETY EVALUATION: These modifications removed the pipe hangers and their resultant loads from wall 2047, thus, reducing the stresses at the connection between the block wall and the floor to within the limits specified in FSAR Section 3.8. This modification will not result in an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-142

SYSTEM: Fuel Oil

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: FCR 83-142 sealed all penetrations passing through the masonry block walls below Elevation 598'-9" in the Emergency Diesel Oil Day Tank Rooms 320A and 321A. Work was completed June 13, 1984.

REASON FOR CHANGE: A review of the Fire Hazard Analysis Report determined these penetrations should be sealed since if the tank had failed, the diesel oil in these rooms could have seeped through the walls to the diesel generator area.

SAFETY EVALUATION: The Emergency Diesel Oil Day Tank Rooms are required to contain the entire contents of the day tanks housed in these rooms per the Fire Hazard Analysis Report. The penetrations in these rooms had to be sealed otherwise the diesel oil in these rooms could have seeped through the walls to the diesel generator area. Since the sealing of these penetrations prevented this from occurring, a new adverse environment was not created. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-029

SYSTEM: High Pressure Injection

COMPONENT: DH64

CHANGE, TEST OR EXPERIMENT: This FCR updated drawing E-15 to show actual, correct plant conditions. This drawing change decreased the open torque switch setting for the low/high pressure injection cross connection valve, DH64, from 4.0 to 2.0 and decreased the close torque switch setting from 2.0 to 1.0. These changes made to drawing E-15 was completed August 2, 1984.

REASON FOR CHANGE: This change was made to update Drawing E-15 to represent the correct condition of the plant. The actual change in valve DH64 came about because the valve experienced problems with its initial settings. Therefore, the settings were changed in accordance with recommendation from the Torrey Pines Technology report on Limitorque motor operated valves.

SAFETY EVALUATION: The safety function of DH64 is to provide a suction path for the high pressure injection system when the Reactor Coolant System pressure is greater than the discharge of the low pressure injection pumps and the Borated Water Storage Tank is depleted. The changes brought about by this FCR does not decrease the safety function of DH64. Therefore, a new adverse environment is not created, and an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-276

SYSTEM: Security

COMPONENT: Security Card Readers

CHANGE, TEST OR EXPERIMENT: FCR 81-276 upgraded the Security System. This requested card user accountability control at the following six assembly areas:

1) Operations Support Center
2) Fifth Floor Office Building
3) Instrumentation and Control Shop
4) Maintenance Shop
5) RACA Entrance
6) Fab Shop (Service Building #6)

This FCR also required software improvements to upgrade two areas. First, a type of system which provided card user accountability control at predefined assembly areas; and second, card controlled keyboards were added to each of the system's keyboards. This feature provides activation of keyboards only for valid card users. Work involved with this FCR was completed July 12, 1984.

REASON FOR CHANGE: These changes will aid in complying with NUREG 0654 and in locating missing or injured persons.

SAFETY EVALUATION: Since these changes upgraded the level of security in the protected area, a new adverse environment is not created. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-086

SYSTEM: Service Water

COMPONENT: Isolation Valve SW-1368

CHANGE, TEST OR EXPERIMENT: FCR 84-086 revised Drawing E-15, Page 7, to delete the torque switch setting for isolation valve SW-1368. Also, E-12B, Page 54, now references E-557, Sheet 69, to show the electrical configuration of SW-1368. This FCR was closed out September 21 1984.

REASON FOR CHANGE: This change corrected the above drawings to reflect the approved design and the "as-built" condition of isolation valve SW-1368.

SAFETY EVALUATION: Valve SW-1368 is an isolation valve for Containment Air Cooler 1-3. This valve is a normally open ball valve. When closed, this valve will isolate Containment Air Cooler 1-3 from the Service Water System. The torque switch for valve SW-1368 is not used in the valve's control circuit. It is not required to control the opening or closing of the valve.

The safety function of the drawings are to provide maintenance personnel the torque switch setting required for a safety related valve. Removal of the torque switch setting for SW-1368 from these documents brings the drawings into agreement with the field, the associated documentation, and the approved design. The removal of the torque switch setting for SW-1368 from the drawings does not create any new adverse environment. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-016

SYSTEM: Process & Area Radiation Monitoring System

COMPONENT: Unit Storm Sewer Radiation Monitor, RE-8442

CHANGE, TEST OR EXPERIMENT: This FCR originally requested RE-8442 be relocated from the old storm sewer to manhole #4. Under revision, this FCR removed RE-8442 and added a new radiation monitor, RE-4686, and flow transmitter, FT-4687, as well as associated conduit, cables, and equipment. Work was completed December 30, 1983.

REASON FOR CHANGE: The original storm sewer location was moved because of the planned construction of Davis-Besse Units 2 and 3. Also, there were many problems experienced with pipes clogging, particularly by the detector. Another problem was the check valves were clogging and holding water, which could freeze. These problems prevented RE-8442 from performing its intended safety function which was as a final check for radioactive releases through the storm sewer system from the plant to the Toussaint River. It was then decided that a new radiation monitor, RE-8486, should replace RE-8442. This decision was made following a 10CFR50.59 evaluation of the secondary plant drainage system discharge radiological monitoring configuration.

SAFETY EVALUATION: The installation of this equipment in manhole #4 required "Q" core drill and PICA reports. It was determined that the change will not create any new adverse environment. Therefore, this FCR does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-396

SYSTEM: Clean Liquid Radwaste

COMPONENT: Spent Resin Transfer Pump

CHANGE, TEST OR EXPERIMENT: This FCR replaced the spent resin transfer pump with two air operated Sandpiper pumps. The addition of these pumps required several modifications. Station air had to be provided for the pumps to run. This air supply required a globe valve with a ball valve downstream to be installed. This globe valve is needed for isolation purposes. Also, ball valves were added at the suction and discharge piping for each Sandpiper pump. A new pressure indicator replaced the original indicator for PI-4950. The new pressure indicator is a compound pressure indicator (rated 30 inches Hg vacuum to 100 psig) used to monitor the pump suction vacuum. Work involved with this FCR was completed January 5, 1984.

REASON FOR CHANGE: This change was needed in that the original spent resin transfer pump was of the Crane-Chempump type and unsuitable for the intended function of the pump, which is to pump resin from the spent resin storage tank to either the drain or the drumming station for offsite shipment. The liquid being pumped is a slurry resin mixture. For this reason, the original Crane-Chempump was not acceptable as the spent resin transfer pump. The Crane-Chempump was qualified for handling clean liquids but not slurry resin mixtures. The problem the slurry resin liquid causes is the resin was wearing the pump's impeller and also plugging the in-line strainer cutting off seal water to the pump.

SAFETY EVALUATION: The purpose of the spent resin transfer pump is to pump resin from the spent resin storage tank to either the drain or the drumming station for offsite shipment. By adding two Sandpiper pumps which are more suited for the intended function of the transfer pump than the Crane-Chempump, the Clean Liquid Radwaste System is upgraded. Because of this increase in quality to the Clean Liquid Radwaste System, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-312

SYSTEM: N/A

COMPONENT: Safety Related Piping

CHANGE, TEST OR EXPERIMENT: This FCR allowed Bechtel Engineering to inspect and document the "as-built" configuration of safety related seismic Category I piping systems per approved "Inspection Procedure for As-Built Configuration of Nuclear Safety Related Piping Components". The final inspection was completed January 14, 1984.

REASON FOR CHANGE: This inspection was to verify the as issued piping and support design drawings agree with the actual plant configuration. Also, this inspection responded to NRC I.E. Bulletin 79-14 which requested the inspection of accessible safety related systems during normal operation and eventually on safety related equipment that is inaccessible during normal plant operation.

SAFETY EVALUATION: This inspection was a passive, visual inspection to obtain measurements and other component verification information. No safety related components or systems were removed from service or altered in their function during the inspection. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-201

SYSTEM: Clean Liquid Radioactive Waste

COMPONENT: Clean Waste Receiver Tank Transfer Pumps, P49-1 & 2

CHANGE, TEST OR EXPERIMENT: FCR 81-201 modified the Clean Liquid Radwaste System and its associated drawings to represent the correct design of the system. Under this FCR, the clean waste receiver tank transfer pumps were altered. The original design of this system had Crane Chempumps of the canned motor design which used Component Cooling Water (CCW) for motor cooling. In 1977, the Crane Chempumps were replaced with Gould pumps which are driven by air cooled motors. Therefore, the CCW supply and return valves were disconnected, and the line's isolation valves were closed. Pipe nipples and pipe caps were installed to each of the pump lines. Drawings were also updated to show as-built conditions. Work involved with this FCR was completed June 12, 1984.

REASON FOR CHANGE: These changes were performed to represent the correct condition of the Clean Liquid Radwaste System.

SAFETY EVALUATION: The purpose of the clean waste receiver tank transfer pumps is to transfer liquid from the clean waste receiver tanks to the boric acid evaporators. Since the above changes enhance the function of the clean waste receiver tanks transfer pumps, no new adverse environment will be created. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-021

SYSTEM: Auxiliary Feedwater

COMPONENT: Supports 3A-EBD-19-H65/H61

CHANGE, TEST OR EXPERIMENT: This FCR modified two supports on steam piping to the Auxiliary Feedwater System. Support 3A-EBD-19-H65 was modified to correct the base of the support pulling away from the wall 7/16 inches. This involved relocating the support's base, increasing the support's bolt size, and relocating the sway strut-to-stanchion connection. These changes reduced the stress levels in the components of support 3A-EBD-19-H65. Support 3A-EBD-19-H61 had its spring hanger readjusted to place the hot/cold settings on the variable spring hanger in more desirable positions for the loads that the piping is subjected to. The modifications made to these supports were completed May 17, 1984.

REASON FOR CHANGE: It was recognized that support 3A-EBD-19-H65 was pulling away from the wall. This was identified as a non-conforming "as-built" condition. Because of this, an analysis was performed on the system with support 3A-EBD-19-H65 inactive and found to be acceptable on a short term basis. Modifications were required to reduce the stress levels in the various support components of 3A-EBD-19-H65. While performing this analysis, support 3A-EBD-19-H61 was found to need its hot/cold settings on its variable spring hanger readjusted. This adjustment was made to compensate for the modifications made to support 3A-EBD-19-H65 and to place the spring hot/cold settings in a more desirable position with respect to safety.

SAFETY EVALUATION: The intended safety function of supports 3A-EBD-19-H65 and 3A-EBD-19-H6 are to restrain/support piping 6"-EBD-19, which is required to deliver steam to drive the auxiliary feed pump turbines powering the auxiliary feed pumps. The changes made under this FCR will not adversely affect the safety function, but will enhance its operation by providing more stability for the steam piping. Therefore, an unreviewed safety question does not exist.

TOLEDO EDISON

February 8, 1985

Log No. K85-345
File: RR 2 (P-6-85-01)

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 1985
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, 1985.

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

Yours truly,

SMQuennoz/jk

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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