AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE January 10, 1986

COMPLETED BY Morteza Khazrai

TELEPHONE (419) 249-5000 Ext. 7290

MONTH December 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	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)

IE24
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OPERATING DATA REPORT

DOCKET NO. 50-346
DATE January 10, 1986
COMPLETED BY Morteza Khazrai
TELEPHONE (419) 249-5000
Ext. 7290

OPERATING STATUS

Notes

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: December 1985
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:

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	744.0	8,760.0	65,065
12. Number Of Hours Reactor Was Critical	0.0	2,845.6	35,877.1
13. Reactor Reserve Shutdown Hours	0.0	44.7	4,058.8
14. Hours Generator On-Line	0.0	2,730.5	34,371.8
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	0.0	6,312,178	81,297,600
17. Gross Electrical Energy Generated (MWH)	0.0	2,087,278	26,933,622
18. Net Electrical Energy Generated (MWH)	0.0	1,942,921	25,233,177
19. Unit Service Factor	0.0	31.2	52.8
20. Unit Availability Factor	0.0	31.2	55.5
21. Unit Capacity Factor (Using MDC Net)	0.0	25.8	45.1
22. Unit Capacity Factor (Using DER Net)	0.0	24.5	42.8
23. Unit Forced Outage Rate	100.0	64.8	25.3

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

REPORT MONTH December 1985

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE January 10, 1986
COMPLETED BY Morteza Khazrai
TELEPHONE (419) 249-5000, Ext. 7290

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
DECEMBER 1985

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) Continued MOVATS testing.

2) All work completed on Decay Heat Loop #2.

3) Power operated relief valve was removed, modified, and replaced.

4) Steam generator manway studs were replaced with a design which permits tensioning rather than torquing of the bolts.

5) Motor driven feed pump was installed and testing has begun.

REFUELING INFORMATION DATE: December 1985

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: 78-263

SYSTEM: Reactor Coolant System

COMPONENT: FI-4102, FI-4202, FI-4302, FI-4402

CHANGE, TEST OR EXPERIMENT: This FCR installed restraints on the lower support piping for the four Reactor Coolant System (RCS) flow indicators listed above. Work was completed February 21, 1981.

REASON FOR CHANGE: These flow indicators were exposed to stress due to the lack of restraint on the lower support piping. This has caused several of the RCS flow indicators to leak.

SAFETY EVALUATION SUMMARY: The safety function of these flow indicators is to monitor RCS flow which is a parameter that must be monitored closely. The above change will prevent the leakage of the flow indicators and allow for a more accurate RCS flow indication. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-319

SYSTEM: Main Steam

COMPONENT: MS-107, CDF11A-2, and CDF11D

CHANGE, TEST OR EXPERIMENT: FCR 78-319 initiated two changes. The first change was made to update associated drawings involving the circuitry for MS-107 in CDF11A-2 which was necessary to clear jumper and lifted wire tag 1175. The second change made by FCR 78-319 was to verify the as-built conditions for disconnect switches CDF11A-2 and CDF11D. Work was completed March 30, 1985.

REASON FOR CHANGE: The above changes were made to update the drawings associated with the disconnect switches to represent the correct configuration of the plant.

SAFETY EVALUATION SUMMARY: The safety function of the disconnect switch cabinet is to provide local control of equipment in case of losing control capability from the Control Room. The change made by this FCR does not affect this safety function. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-016, Rev. A

SYSTEM: Auxiliary Building Fire Detection System, Elevation 585'

COMPONENT:

CHANGE, TEST OR EXPERIMENT: FCR 79-016 Revision A modifies the Fire Detection System on Elevation 585'.

Fire detectors were installed in Rooms 300, 304, (310-313), 312, 321, 322, 325, 328, 303, 314, the annulus space in the area of the electrical penetrations, Panel C3630 in Room 324, and fire detectors were relocated in Room 323. The modifications to the Fire Detection System were started November 6, 1979, and the job task was completed January 1, 1980.

REASON FOR CHANGE: Detectors were added to Room 301 which adjoins Room 300 to ensure early detection of fires. Detectors were added to Rooms 323 and 324 due to pockets created by structural steel. One detector was added to Room 320 due to a wall being added. Detectors were added to Rooms 318 and 319 to provide fire protection while modifications were being implemented. One detector was added to Room 320A and 321A. Five detectors were added to the Intake Structure. This includes Rooms 51, 52, and 53.

These changes were made to satisfy commitments addressed in the Fire Hazard Analysis Report.

SAFETY EVALUATION SUMMARY: All work was installed with the "Q" core drill reports. Post Inspection Construction Authorizations (PICA) insure those portions from creating any new adverse environments. An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-079

SYSTEM: Main Steam

COMPONENT: Various Hangers

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to modify the hangers listed below. These hangers are support for the Main Steam System. Work was completed February 2, 1984.

3A-EBB-2-H9	3A-EBD-19-H100	3A-EBD-19-H129
3A-EBB-2-H21	3A-EBD-19-H101	3A-EBD-19-H143
3A-EBB-2-H37	3A-EBD-19-H104	3A-EBD-19-H146
3A-EBD-19-H10	3A-EBD-19-H105	3A-EBD-19-H151
3A-EBD-19-H47	3A-EBD-19-H106	3A-EBD-20-H2
3A-EBD-19-H48	3A-EBD-19-H107	SR1 East & West
3A-EBD-19-H58	3A-EBD-19-H108	SR2 East & West
3A-EBD-19-H65	3A-EBD-19-H109	SR4 East & West
3A-EBD-19-H67	3A-EBD-19-H112	SR5 East & West
3A-EBD-19-H85	3A-EBD-19-H113	SR6 East & West
3A-EBD-19-H92	3A-EBD-19-H115	SR7 East & West
SR8 East & West	SR18 East & West	SR19 East & West
SR47 East & West	H3 East & West	

REASON FOR CHANGE: Modifications were made in accordance with IE Bulletins 79-02 and/or 79-14.

SAFETY EVALUATION SUMMARY: The function of the listed hangers is to support the Main Steam System. Modifications were made to the hangers to reduce stress levels to acceptable values, thus increasing the margin of safety. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-058

SYSTEM: Emergency Diesel Generators

COMPONENT: K5-1 and K5-2

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to modify the engine control system of the Emergency Diesel Generators (EDGs) to utilize the engine speed control motor (governor motor) on the mechanical governor to regulate fast start acceleration. This modification changed the 0-900 rpm time from 4-5 seconds to the vendor's recommendation of 8-9 seconds. Technical Specifications require EDGs to start from ambient condition and accelerate to at least 900 rpm in less than or equal to 10 seconds. Work was completed August 30, 1983.

REASON FOR CHANGE: This modification will prolong the life and reliability of the turbocharger and gear train of the EDGs.

SAFETY EVALUATION SUMMARY: The safety function of the EDGs is to provide on-site standby power sources for essential loads required for safe plant shutdown. By increasing the response time of the EDGs, their safety function will be maintained and their reliability will increase. This will not create an adverse environment. Therefore, this does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-050

SYSTEM: Reactor Protection System

COMPONENT: Cable 4LRPSC07

CHANGE, TEST OR EXPERIMENT: This FCR was initiated for the retermination of cable 4LRPSC07 on the leads of Penetration Module C, and to test the cable for resistance of $> 10\mu$. Work was completed June 18, 1982.

REASON FOR CHANGE: During field checks of cable resistance to ground penetration P4L1G, low resistance was encountered which caused FCR 82-050 to be initiated.

SAFETY EVALUATION SUMMARY: The safety function of cable 4LRPSC07 is for reactor coolant loop and hot leg narrow range temperature for reactor protection on reactor coolant high temperature. This change has not degraded the safety function of the cable because it was reterminated to a Class 1E module and was routed to maintain proper safety channel separation and channel designation. There is no unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-102

SYSTEM: Reactor Coolant System

COMPONENT: Reactor Coolant Pump Piping

CHANGE, TEST OR EXPERIMENT. FCR 83-102 was initiated to modify various supports used to maintain the Reactor Coolant Pumps (RCP). Modifications included:

1) deleting support M-1089/H3 on RCP 1-2-1 standpipe flush line from all associated Station drawings

2) modifying the adjacent supports on piping located on RCP 1-2-1

3) removing support on the 3/4" CCB-7 line located on RCP 1-1-1

Work was completed December 19, 1984.

REASON FOR CHANGE: During an inspection, it was discovered that support M-1089/H3 was never installed as designed and, due to obstructions, cannot be installed. For this reason, support M-1089/H3 was deleted from the RCS drawings and adjacent supports on the piping had to be modified. Also, a walkdown required by IEB 79-14 identified interferences (during operation) with an RCP 1-1-1 wire rope whip restraint and the 3/4" CCB-7 line located on RCP 1-1-1. This requires the removal of the support on the 3/4" CCB-7 line.

SAFETY EVALUATION SUMMARY: The safety function of the above listed supports is to maintain the integrity of the RCPs. The above modifications will allow this function to be performed. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-065

SYSTEM: Reactor Coolant System

COMPONENT: RC-49

CHANGE, TEST OR EXPERIMENT: This FCR allowed for the performance of Section 7.4 of Test Procedure . 600.13, the Pressurizer Operational and Spray Flow Test. This was done to reset the flow for RC-49, pressurizer mini-flow valve, as recommended by Babock & Wilcox. The recommended setting range was between 0.75 gpm to 3.0 gpm. Work was completed January 20, 1985.

REASON FOR CHANGE: In the process of connecting and disconnecting the electrical cables to the pressurizer heater bundles, many of the individual heater electrical pins and ceramic insulators were damaged. Damaged heater connectors render the heaters inoperable. Until the connectors are repaired, the required heating capacity during steady state conditions should be reduced, if possible.

The pressurizer's silicon controlled rectifier (SCR) heater bank was sized to provide enough heating capacity to compensate for normal heat loss (ambient losses) and spray valve bypass flow during steady state operations. The original plant design did not require any additional heaters except during load changes and reactor startup.

Presently, the SCR heater bank cannot supply the required heating capacity during steady state operation. Additional heater banks are energized which leaves less capacity available for transient conditions. It is suspected that RC-49 was set above the 0.75 gpm to 3 gpm flow recommended by Babcock & Wilcox, thus, causing an additional heater requirement during normal operating conditions.

SAFETY EVALUATION SUMMARY: The safety function of the mini-flow valve RC-49 is to eliminate the abnormal buildup or dilution of boric acid within the pressurizer and to minimize cooldown of the coolant in the spray and surge lines. The resetting valve RC-49 will not prevent it from performing its intended safety function. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-219

SYSTEM: Reactor Coolant System

COMPONENT: Piping support PSU-R1/PSU-H1

CHANGE, TEST OR EXPERIMENT: FCR 84-219 modified piping support PSU-R1//PSU-H1 located on the 10" pressurizer surge line. Modifications included:

1) the replacement of two 1½" bore hydraulic snubbers with one 2½" bore hydraulic snubber,

2) the installment of a kicker brace under the existing built up beam of PSU-H1,

3) the removal of the upper, west section of the structural tubing from PSU-R1, and

4) the revision of the hot/cold settings of the spring hanger located on PSU-H1.

Work was completed January 3, 1985.

REASON FOR CHANGE: Investigation showed the redesign of the piping support was needed to compensate for the thermal movements of the piping.

SAFETY EVALUATION SUMMARY: The safety function of the piping support on the 10" surge line is to insure the integrity of the surge piping to the pressurizer and, therefore, maintain the RCS pressure boundary under sustained, thermal, and seismic loading. Originally, the thermal movements for this line were not considered in the line's design. The modifications listed above account for the thermal movement of the line, therefore, insuring surge piping integrity. The modifications made do not cause an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: FCR 84-221

SYSTEM: Fuel Handling Area Exhaust

COMPONENT: Support 410-03-7

CHANGE, TEST OR EXPERIMENT: FCR 84-221 modified the Fuel Handling Area Exhaust System support 410-03-7. Modifications included the removal of a tubing section in column HA-7-4 and the addition of stiffener plates to the underside of the 1/8" plate welded to the bottom of the duct. Work was completed December 31, 1984.

REASON FOR CHANGE: The above modifications were made to increase the duct/support flexibility insuring compliance with design requirements for plant operation.

SAFETY EVALUATION SUMMARY: The safety function of support 410-03-7 is to provide a supporting/stabilizing function to the Fuel Handling Area Exhaust System during normal operation and a seismic event. By performing this function, the structural integrity of the ductwork is maintained. The above modifications will enhance this function. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-224

SYSTEM: Emergency Ventilation

COMPONENT: Support 410-07-C-31Z

CHANGE, TEST OR EXPERIMENT: FCR 84-224 was implemented to modify Emergency Ventilation System (EVS) support 410-07-6-31Z. The modification involved cutting and removing a 2" x 2" x 1/4" structural angle from the support. The work was completed December 18, 1984.

REASON FOR CHANGE: After system reanalysis, it was determined that the subject support required modification to comply with short and long term operability requirements.

SAFETY EVALUATION SUMMARY: The purpose of the EVS is to insure that a negative pressure exists in the electrical and mechanical penetration rooms and the containment annulus, and to remove possibly contaminated air from these areas and discharge the air to the atmosphere through the station vent. The safety function of support 410-07-C-31Z is to provide a supporting/stabilizing function to the ductwork during both normal operation and a postulated seismic event. By performing the above modification, the flexibility and structural integrity of the duct and support will be enhanced which will insure compliance with short and long term design criteria for plant operations. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-225

SYSTEM: Containment Purge Exhaust

COMPONENT: 435-04-1B

CHANGE, TEST OR EXPERIMENT: This FCR modified the heating, ventilation, and air conditioning (HVAC) ductwork support 435-04-1B located on the "Q" portion of the Containment Purge Exhaust System. Modifications included adding two extension plates to each upper support plate of support 435-04-1B and installing horizontal bracing from the support frame to the Shield Building. Work was completed December 27, 1984.

REASON FOR CHANGE: The above modifications were made to stabilize the support, thus, insuring compliance with design requirements for plant operation.

SAFETY EVALUATION SUMMARY: The safety function of of the containment purge exhaust support 435-04-1B is to provide restraining/stabilizing action to the ductwork during a postulated seismic event. By performing this function, the structural integrity of the ductwork and the isolation of the negative pressure boundary are maintained. The modifications made by this FCR will enhance the safety function of the support. Therefore, an unreviewed safety question does not exist.

TOLEDO EDISON

January 10, 1986

Log No. KB86-0012
File: RR 2 (P-6-85-12)

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, December 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 December 1985.

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

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