AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346
UNIT Davis-Besse Unit 1
DATE January 10, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

MONTH December, 1983

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	882	17	561
2	882	18	0
3	861	19	0
4	880	20	0
5	879	21	0
6	882	22	0
7	878	23	0
8	879	24	157
9	880	25	668
10	882	26	743
11	878	27	802
12	883	28	807
13	877	29	813
14	879	30	826
15	878	31	830
16	883		

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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)

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE January 10, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

OPERATING STATUS

1. Unit Name:	Davis-Besse Unit 1
2. Reporting Period:	December, 1983
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):	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	8,759	47,520
12. Number Of Hours Reactor Was Critical	593.5	6,607.0	27,502.5
13. Reactor Reserve Shutdown Hours	0.0	515.2	3,879.3
14. Hours Generator On-Line	581.9	6,391.2	26,150.8
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	1,546,776	15,671,053	61,043,814
17. Gross Electrical Energy Generated (MWH)	512,418	5,186,175	20,292,193
18. Net Electrical Energy Generated (MWH)	482,921	4,883,259	18,998,699
19. Unit Service Factor	78.2	73.0	55.0
20. Unit Availability Factor	78.2	73.0	58.7
21. Unit Capacity Factor (Using MDC Net)	74.3	63.8	45.7
22. Unit Capacity Factor (Using DER Net)	71.6	61.5	44.1
23. Unit Forced Outage Rate	21.8	11.6	18.6

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 Unit 1
DATE January 10, 1984
COMPLETED BY Bilal Sarsour
TELEPHONE 419-259-5000, Ext. 384

REPORT MONTH December, 1983

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
16	83 12 17	F	162.1	A	3	NP-33-83-101	IA	INSTRU	The Reactor Protection System (RPS) tripped the reactor on high flux due to a loss of power to essential instrumentation panels Y-1 and Y-1A. A unit cooldown was initiated due to excessive leakage. 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, 1983

12/1/83 - 12/23/83

Reactor power was maintained at approximately 100% power until 1512 hours on December 17, 1983, when a reactor trip occurred. The Reactor Protection System (RPS) tripped the reactor on high flux due to a loss of power to essential instrumentation panels Y-1 and Y-1A caused by a combination of the essential bus inverters input fuse blowing from momentary fault and the Integrated Control System (ICS) reactor power auctioneer circuit being powered from Y-1. A unit cooldown was initiated due to an excessive leakage of reactor coolant through the packing of letdown valve MU-2B.

The reactor was critical at 2145 hours on December 23, 1983.

12/24/83 - 12/31/83

The turbine generator was synchronized on line at 0917 hours on December 24, 1983.

Reactor power was slowly increased and attained approximately 95% power on December 30, 1983, and maintained at this power level for the rest of the month.

REFUELING INFORMATION DATE: December, 1983

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: August 3, 1984

3. Scheduled date for restart following refueling: October 14, 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.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-232

SYSTEM: Freeze Protection

COMPONENT: Panel C5735

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to relocate freeze protection alarm panel C5735 in the cabinet room to its new location in cabinet C5759A. The movement of this alarm panel required that the cables from this panel be disconnected, rerouted to the new panel, and reterminated. Work was completed October 5, 1982.

REASON FOR CHANGE: The relocation of this panel was necessary to eliminate interference with the new reactor coolant pump diagnostic panels which were mounted at locations C5758A and B.

SAFETY EVALUATION: The freeze protection alarm system, referred to in this FCR, does not provide a specific safety function. This FCR was designated nuclear safety related due to the inclusion of "Q" DCNs which address the details of the physical relocation of the freeze protection alarm panel. The work authorized by this FCR did not create any new adverse environments because the panel was seismically mounted and all associated cables were disconnected and reterminated internal to this panel. This did not constitute as unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-094

SYSTEM: Reactor Coolant System

COMPONENT: Hot leg thermo wells

CHANGE, TEST OR EXPERIMENT: Hot leg thermo wells TE RC 3A1, 3A2, TE RC 3B3, 3B4, TE RC 3B1, 3B2, and TE RC 3A3, 3A4 were sealed with Furmanite Compound Type 2. Work was completed July 29, 1980.

REASON FOR CHANGE: This modification was performed to enhance the sealing of the above thermo wells to prevent leaks.

SAFETY EVALUATION: Babcock & Wilcox's letter of February 2, 1979 clearly indicates that the installation of the Furmanite should not affect the response time of the RTDs or the integrity of the Reactor Coolant System pressure boundary. The response time of the Reactor Protection System was not changed by the addition of Furmanite as it did not change the response time or accuracy of the RTDs. This did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-019

SYSTEM: Masonry block walls

COMPONENT: Walls 3447 and 3467

CHANGE, TEST OR EXPERIMENT: Structural angles were added to the lintel over Door 309, which is part of block wall 3447, to reinforce the lintel-concrete wall connection. Wall 3467 was reinforced for pressure loadings by the addition of steel bracing. Both walls are located between corridors 304 and 310 on Elevation 585'. Work was completed November 10, 1981.

REASON FOR CHANGE: Reanalysis of block walls required by Nuclear Regulatory Commission IE Bulletin 80-11 had shown that during a seismic event, the lintel over Door 309 in block wall 3447 could become unstable after the lintel to concrete wall connections fail. The additional wall bracing to wall 3467 was required to reduce wall stresses to within allowable limits of compartment pressure loadings following a postulated pipe break.

SAFETY EVALUATION: A large number of safety related conduits penetrate walls 3467 and 3447 may have been affected through loss of support or the impact of falling masonry if these walls were to fail.

These modifications have lowered the possible stresses in masonry and wall connections to allowable limits specified in Section 3.8 of the Final Safety Analysis Report. This has not affected the walls ability to function as a fire wall and as a structural support. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-055

SYSTEM: Fire Protection

COMPONENT: Sprinkler system

CHANGE, TEST OR EXPERIMENT: The sprinkler systems in Emergency Diesel Generator Rooms 318 and 319, Elevation 585'-00", were converted from manual to automatic actuation. Work was completed October 8, 1980.

REASON FOR CHANGE: This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION: Pipe supports were designed to preclude seismic hazards from creating any new adverse environments. Stress analysis was also done to determine if any high stress points exist that could create an adverse water environment. No unreviewed safety question existed.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-229

SYSTEM: Emergency Ventilation System (EVS)

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to revise Pioneer Service and Engineering Company drawing 410-05-C-32D to show that two bolts on the upper wall mounted base plate for EVS ductwork were never installed. This revision was verified July 12, 1983.

REASON FOR CHANGE: The drawing referenced now more accurately reflects the "as-built" conditions in the plant.

SAFETY EVALUATION: The safety function of the subject supporting structure is to support the EVS ductwork during normal operation and during safe shutdown earthquake conditions. The as-built conditions were analyzed by Bechtel, and the results indicate that the as-built support is capable of performing its design function, both for short and long term operation. Based on the above, implementation of this FCR did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-178

SYSTEM: Process and Area Radiation Monitoring

COMPONENT: Radiation Monitors

CHANGE, TEST OR EXPERIMENT: This FCR was written to perform a 10CFR50.59 review on the iodine channel of the post accident containment radiation monitors RE-4597AA and BA. The review was completed December 29, 1982.

REASON FOR CHANGE: The Station has experienced numerous failures of the iodine channel in radiation monitors RE-4597AA and BA. This review justified plant operation without the iodine channels in radiation monitors RE-4597AA and BA in operation.

SAFETY EVALUATION: The safety function of the iodine channel in RE-4597AA and BA is to provide indication and alarms of the iodine concentration in containment. The safety function of the iodine channels is not being impaired because the station has a procedure, ST 5099.06, Effluents, Radiological Monitoring Surveillance Test, to ensure grab samples of the containment atmosphere can be taken at any time. There was no uncontrolled leakage path for unmonitored air to leave containment, and the iodine monitors in the station vent were operable to monitor the actual releases offsite. There is no violation of Technical Specifications involved due to the failure of iodine monitoring channels since the requirements for containment post accident radiation monitoring, of Technical Specification 3.3.3.6, are being fulfilled by the noble gas channels of RE-4597AB and BB, and the containment wide range area monitors RE-4596A and B. An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-043

SYSTEM: Nitrogen Purge and Blanket System

COMPONENT: Electrical Penetration Rooms

CHANGE, TEST OR EXPERIMENT: This FCR was initiated to provide a bracket to secure two nitrogen bottles in each electrical penetration room. Work was completed February 23, 1982.

REASON FOR CHANGE: Water, which was determined to be condensation from the nitrogen supply system, was causing grounds in Electrical Penetrations in Room 427. The nitrogen supply lines to both electrical penetration rooms were isolated until the problem could be resolved, and bottled nitrogen was brought in. These bottles are under pressure and, therefore, had the potential to become missiles.

SAFETY EVALUATION: This modification has eliminated the potential of the nitrogen bottles to become missiles. This change was found to have no adverse effect on the structures to which they are attached. Therefore, this did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-042

SYSTEM: Steam and Feedwater Rupture Control System (SFRCS)

COMPONENT: Annunciator alarm window Q963

CHANGE, TEST OR EXPERIMENT: This FCR called for modifying the alarm logic of SFRCS. This ensures that the full trip alarm annunciation will happen only when any one actuation channel receives a trip signal in both logic channels. Also, the low steam pressure alarm will not come on when these trips are blocked. Work was completed August 2, 1982.

REASON FOR CHANGE: Prior to this modification, the full trip alarm annunciation could occur after a half trip of the SFRCS. These changes provide more accurate and meaningful information to the operator.

SAFETY EVALUATION: The safety function of the SFRCS is to start the Auxiliary Feedwater System which provides the capability to remove the decay heat of the reactor. Since the annunciator does not have a safety function, the safety function of SFRCS was not affected. Pursuant to the above, an unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-075

SYSTEM: Steam and Feedwater Rupture Control System (SFRCS)

COMPONENT:

CHANGE, TEST OR EXPERIMENT: A 0.5 second time delay circuit was added in SFRCS cabinets on the inputs from the main feedwater check valve high reverse pressure switches, PDIS-2685A, B, C, and D and PDIS-2686A, B, C, and D. Supplement 1 to this FCR revised the accuracy of these time delay circuits from ±5% to ±20%. Work was completed August 18, 1980.

REASON FOR CHANGE: Closure of the check valves in the main feedwater lines was causing a differential pressure pulse that these PDIS were detecting. The addition of this 0.5 second time delay has prevented this momentary pulse from tripping the SFRCS.

SAFETY EVALUATION: This time delay has added 0.5 seconds to the response time for the Auxiliary Feedwater Pumps to start on a loss of main feedwater. Tests indicate that after the SFRCS is tripped, in 22 seconds the Auxiliary Feedwater Pumps are at full speed, with 10 more seconds for diesel start time. This gives a total of 32 seconds which is 8 seconds less than the 40 seconds requirement of the accident analysis. Thus, the addition of the 0.5 second time delay has not violated the accident analysis. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-184

SYSTEM: Fire Protection System

COMPONENT: Cable trays BCBQ and BLBP

CHANGE, TEST OR EXPERIMENT: Brand Industrial Services Incorporated Drawing C316-2-4 was updated to reflect as-built conditions of cable trays BCBQ and BLBP. This was completed November 24, 1982.

REASON FOR CHANGE: A site inspection had discovered that there exists a fire stop common to both cable trays BCBQ and BLBP that did not appear on the as-built drawing. The fire stop is required at this location in order to meet the eight foot minimum spacing requirement that was specified for these trays.

SAFETY EVALUATION: The updated drawing certifies installation of a fire barrier in accordance with specifications and provides material traceability. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-424

SYSTEM: Service Water

COMPONENT: Control Room Emergency Condensing Units (S33-1 and S33-2)

CHANGE, TEST OR EXPERIMENT: Piping and Instrument Diagram M-041 was revised to reflect the as-built conditions of the Control Room Emergency Condensing Units. Work was completed March 6, 1981.

REASON FOR CHANGE: The previous revision of this drawing showed Service Water Pump 1-1 (supplied from Channel 1 power) feeding Control Room Emergency Condensing Unit S33-2 (supplied from Channel 2 power) and Pump 1-2 feeding Emergency Condensing Unit S33-1. The as-built configuration has Service Water Pumps 1-1 and 1-2 feeding Emergency Condensing Units S33-1 and S33-2, respectively.

SAFETY EVALUATION: Per the arrangement shown in the previous revision to this drawing, if power was lost in one channel, both Control Room Emergency Condensing Units would be lost. In the as-built configuration, Service Water Pumps 1-1 and 1-2 supply Condensing Units S33-1 and S33-2, respectively, and hence meet the single failure criterion. This drawing change was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-052

SYSTEM: Fire Protection

COMPONENT: Sprinkler system

CHANGE, TEST OR EXPERIMENT: A sprinkler system was added to Service Water Valve Room 316 in the intake structure. Work was completed April 14, 1980.

REASON FOR CHANGE: This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION: Installation in accordance with the PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-242

SYSTEM: Fire Protection

COMPONENT: Sprinkler system

CHANGE, TEST OR EXPERIMENT: A sprinkler system was added in the Number 2 Mechanical Penetration Room 236 on Elevation 565'. Work was completed October 29, 1980.

REASON FOR CHANGE: This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION: This work was non-nuclear safety related except for a core drill/cutout. Installation in accordance with "Q" core drill report and PICA has prevented this from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-056

SYSTEM: Fire Protection

COMPONENT: Blowout panels

CHANGE, TEST OR EXPERIMENT: A water curtain was added for the blowout panel in the east wall of the Number 4 Mechanical Penetration Room 314, Elevation 585'-00". Work was completed October 30, 1980.

REASON FOR CHANGE: This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION: Installation in accordance with the core drill report and PICA has precluded these portions from creating any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-058

SYSTEM: Fire Protection

COMPONENT: Blowout panels

CHANGE, TEST OR EXPERIMENT: A water curtain was added for the blowout panel in the north and east walls of Boric Acid Evaporator Room 235, Elevation 565'-00". Work was completed February 26, 1980.

REASON FOR CHANGE: This change was completed to upgrade the Fire Protection System in order to comply with commitments made in the Fire Hazard Analysis Report.

SAFETY EVALUATION: This work is non-nuclear safety related except for four core drill cutouts. Installation in accordance with the core drill report and PICA has precluded any new adverse environments. An unreviewed safety question was not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-445

SYSTEM: Borated Water Storage Tank (BWST)

COMPONENT: Overflow and drain line piping

CHANGE, TEST OR EXPERIMENT: Permanent heat tracing and associated power supplies were added to the following BWST overflow and drain line piping: 2" - HCB18, 8" - HCC37, 6" - HSC122, 3" - HCC21, and 10" - HCC93. The work for this FCR was completed September 10, 1981.

REASON FOR CHANGE: Initially, only a temporary heat tracing and associated power supply were installed on these lines.

SAFETY EVALUATION: The addition of permanent freeze protection to the BWST overflow and drain line piping has not affected the safety function of the Emergency Core Cooling System. In fact, this change has enhanced their functions. This was not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-104

SYSTEM: Reactor Coolant System (RCS)

COMPONENT: Power Operated Relief Valve (PORV) Setpoint

CHANGE, TEST OR EXPERIMENT: This FCR called for changing the field opening setpoint of the PORV to 2395.8 ± 2.2 psig. The setpoint index was revised to reflect the above change. Work was completed August 6, 1982.

REASON FOR CHANGE: This setpoint complies with the Technical Specification requirements of ≥2390 psig while maintaining an adequate margin for instrument calibration.

SAFETY EVALUATION: The function of this valve is to assist the code safety valves which protect the RCS from overpressurization. This setpoint meets all safety requirements. No unreviewed safety questions were involved in this change.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-212

SYSTEM: Makeup and Purification System

COMPONENT: Valve MU271

CHANGE, TEST OR EXPERIMENT: Seal Return Cooler 1-1 Outlet Isolation Valve, MU271, was changed from locked open status to closed status. Work was completed, and all affected drawings and procedures were updated by May 31, 1983.

REASON FOR CHANGE: It was desired to raise the seal injection temperature to improve Reactor Coolant Pump seal performance. For this reason, only one seal return cooler is normally in service. The second seal return cooler will be used if a makeup pump is running on recirculation flow to maintain makeup tank temperature <140°F.

SAFETY EVALUATION: This valve was initially locked open to ensure a minimum recirculating water flowpath for the makeup pumps. This is still accomplished by having Seal Return Cooler 1-2 in service with its outlet isolation valve, MU270, locked open. Therefore, this change had no impact on system safety function. An unreviewed safety question did not exist.

TOLEDO EDISON

January 10, 1984

Log No. K84-009
File: RR 2 (P-6-83-12)

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, 1983
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 1983.

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

Yours truly,

Terry D. Murray/SMQ

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

IE24
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