OPERATING DATA REPORT

DOCKET NO. 50-346
DATE July 8, 1983
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 196

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: June, 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
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	720.0	4,343.0	43,104.0
12. Number Of Hours Reactor Was Critical	720.0	4,002.8	24,898.3
13. Reactor Reserve Shutdown Hours	0.0	313.9	3,678.0
14. Hours Generator On-Line	720.0	3,950.4	23,710.0
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	1,819,447	10,331,316	55,704,077
17. Gross Electrical Energy Generated (MWH)	598,563	3,446,312	18,551,966
18. Net Electrical Energy Generated (MWH)	566,951	3,264,490	17,379,930
19. Unit Service Factor	100.0	90.9	55.0
20. Unit Availability Factor	100.0	90.9	59.0
21. Unit Capacity Factor (Using MDC Net)	90.1	86.0	46.1
22. Unit Capacity Factor (Using DER Net)	86.9	82.9	44.5
23. Unit Forced Outage Rate	0.0	9.0	18.9

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
July 29, 1983 Refueling Outage Duration: Approximately 8 weeks

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 8, 1983
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000, Ext. 196

REPORT MONTH June, 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
									No unit shutdowns or power reductions this month.

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, 1983

The unit remained at approximately 90 percent reactor power the entire month of June. The power is still limited due to increased condensate flow associated with a realignment of feedwater heater drains following an erosion of moisture separator reheater drain piping to the condenser.

REFUELING INFORMATION DATE: June, 1983

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: July 29, 1983

3. Scheduled date for restart following refueling: September 23, 1983

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, 1983

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) 92 - 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: 77-384

SYSTEM: Miscellaneous

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: The specification for field painting, 7749-A-24 was revised October 20, 1978 to allow the use of Amercoat 90, with the thickness specified in Attachment 1 of this specification for Amercoat systems in all areas where nuclear quality paint is required by Attachment 2 of the specification. This includes repair and touch up work, as well as any new items added to these areas.

REASON FOR CHANGE: Use of one coating system for all areas where nuclear quality painting is required has eliminated the need to stock various types of paint of limited shelf life. Additionally, Amercoat 90 may be applied by brush, roller, or spray as a primer and a top coat over power or hand sanded surfaces.

SAFETY EVALUATION: This change has not adversely affected the safety function of the paint system. A complete engineering evaluation has been made of the flame spread ratings, heat of combustion, and nuclear testing of the Amercoat 90. The results of this evaluation confirm the acceptability of this change in the painting specification. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-389

SYSTEM: Process Monitoring

COMPONENT: Radiation Elements 2024, 2025, 5029, 5030, 5032, 5403, 5405, 5327, 5328, 1003A, and 1003B

CHANGE, TEST OR EXPERIMENT: The enclosure for the above radiation elements and piping was modified so that the bypass and outlet filters for the air pumps are mounted on the outside of the enclosure. A removable cover was fabricated for the filters. Work was completed June 8, 1982.

REASON FOR CHANGE: Prior to this modification, when the pump was shut down for filter cleaning, the unit had to cool for four to six hours as the enclosure becomes very hot. Personnel may now clean the filters in a reasonable amount of time without receiving burns.

SAFETY EVALUATION: These changes are an improvement to the system and will not have an adverse effect on the safety of the plant.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-342

SYSTEM: Pressurizer Spray

COMPONENT: Motor operator for valve RC-HV2

CHANGE, TEST OR EXPERIMENT: Velan drawing R35216-4 and any other affected drawings were revised to reflect that pressurizer spray valve RC-HV2 has a 1.6 HP motor operator rather than a 1.0 HP. The drawing changes were verified May 3, 1983.

REASON FOR CHANGE: Babcock and Wilcox stated in a letter dated June 2, 1978, that records show a 1.0 HP motor should have been supplied. However, it was determined that the use of the 1.6 HP motor may continue with no change as long as no operational problems develop.

SAFETY EVALUATION: This drawing change did not create any new adverse environments by making these as-built revisions and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-271

SYSTEM: 480 Volt and 240 Volt Motor Control Centers

COMPONENT: 480 Volt and 240 Volt Feeders

CHANGE, TEST OR EXPERIMENT: On May 3, 1982, the work implemented by FCR 79-271 was completed. This involved the replacement of all thermal overload heaters with shorting bars for all Class 1E motor operated valves.

REASON FOR CHANGE: This change made the system more reliable and simpler.

SAFETY EVALUATION: Overload heaters do not have any safety function in our system. Therefore, their exclusion made the system simpler and more reliable. This is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-137

SYSTEM: Reactor Coolant System

COMPONENT: Reactor Coolant Pumps and Motors

CHANGE, TEST OR EXPERIMENT: An interlock was added to the reactor coolant pumps motor start circuit. This will ensure that the seal return valves, MU59A through D, are open before the reactor coolant pump motor may be started. This was completed June 1, 1982.

REASON FOR CHANGE: A major contributor to reactor coolant pump seal failures had been operation of the pump with the seal return valve shut. Although the pump pre-start checklist verifies that seal return flow has been established, there was no assurance that the seal return valve was open when the motor was started.

SAFETY EVALUATION: This FCR was designated as nuclear safety related since the FCR package contained a nuclear safety related drawing change notice. The negative pressure boundaries and fire barriers were sealed properly. Hence, this is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-168

SYSTEM: Reactor Coolant Pump

COMPONENT: Reactor Coolant Pump 1-2-2 Elevation 596' Platform

CHANGE, TEST OR EXPERIMENT: Work implemented by FCR 80-168 was completed March 26, 1982. This involved the removal of a 2 inch by 9 inch section of the top flange of a 14WF219 I-beam and a 2 inch by 9 inch section of grating from the Reactor Coolant Pump 1-2-2 596' elevation platform.

REASON FOR CHANGE: This change was necessary because two of the component cooling water return line spool piece flange bolts could not be properly tightened. This was due to the fact that the top flange of the I-beam interferred with the installation of a wrench on these nuts.

SAFETY EVALUATION: This FCR work did not degrade the integrity of the platform and did not affect any safety fuctions. It is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-222

SYSTEM: Emergency Diesel Generators

COMPONENT: Various

CHANGE, TEST OR EXPERIMENT: This FCR was implemented for the installation of the following improved replacement parts for the emergency diesel generators:

New Part Number	Old Part Number	Description
8470154	8442661	Stubshaft assembly
9515338	8419151	Gear-spring loaded drive
181702	181433	Bolt, ½-20 x 1-3/4

This was completed January 4, 1983.

REASON FOR CHANGE: These replacement parts have been recommended by the vendor as a result of the discovery of a 5/8" bolt found to have "backed" out of the spring loaded drive gear.

SAFETY EVALUATION: The changes reflect design product improvements which should preclude such failures from recurring and thus improve component reliability. The vendor has supplied a certificate of conformance to certify that the components meet original specification requirements. No unreviewed safety question was involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-132

SYSTEM: Station Administrative Procedures

COMPONENT: Not Applicable

CHANGE, TEST OR EXPERIMENT: The following station administrtaive procedures, which were listed in Appendix 13 of the Final Safety Analysis Report, have been deleted:

MP 1401.15	AD 1828.19
HP 1601.02	AD 1830.00
HP 1605.03	AD 1831.00
AD 1828.16	AD 1847.06

This change was verified March 21, 1983.

REASON FOR CHANGE: These procedures were deleted for the following reasons:

1) MP 1401.05 - Pressurizer Spray Valve Removal and Replacement

   This procedure was deleted because it was a compilation of sections of other procedures, all of which are related to the removal and replacement of safety related valves.

2) HP 1601.02 - Guides

   This procedure was deleted as its contents were included in HP 1601.04 and HP 1605.02.

3) HP 1605.03 - Working Limits for Contamination

   This procedure was deleted as its contents were included in HP 1601.04.

4) AD 1828.16 - Inspection Engineering Training

5) AD 1828.19 - Designated Inspector Training

6) AD 1830.00 - Inspection Engineering

7) AD 1831.00 - Quality Verification by Station Personnel

8) AD 1847.06 - Materials Inspection Procedure

The last five procedures were deleted as their functions are now met within the Quality Assurance Division.

SAFETY EVALUATION: These changes did not result in the loss of any safety related function information or instructions. Therefore, the deletions did not constitute an unreviewed safety question, as procedures may be combined or separated to conform with the procedure's plan, see R.G. 1.33-1972, paragraph C.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-034

SYSTEM: Safety Features Actuation System (SFAS)

COMPONENT: Radiation Monitors RE2004, RE2005, RE2006, and RE2007

CHANGE, TEST OR EXPERIMENT: This FCR, which was completed June 28, 1982, called for changes for the SFAS containment radiation monitor trip setpoints and alarm setpoint. The new setpoint for Modes 1, 2, 3, and 4 will be 1.8 times background at the rated thermal power ± 10% of the background reading at the rated thermal power. In Mode 6, the new setpoints will be 15 mr/hr, 25 mr/hr, 25 mr/hr, and 15 mr/hr for RE2004, RE2005, RE2006, and RE2007, respectively. Guidelines for determination of background radiation at rated thermal power for Modes 1, 2, 3, and 4 have been incorporated into existing procedure ST 5031.01. Trip setpoints for the monitors in Mode 6 were determined by adding 15 mr/hr to the expected background radiation at their repective locations. The 15 mr/hr would be the contribution from a fuel handling accident.

REASON FOR CHANGE: This change will ensure that Technical Specification requirements are not violated. These requirements are two times background at the rated thermal power for Modes 1, 2, 3, 4, and 6.

SAFETY EVALUATION: The safety function of the SFAS containment radiation monitor is to provide the automatic signal for the initiation of containment isolation in the event of a loss of coolant accident. The monitors are located in the annulus during normal operation and moved into containment to monitor, in case of a fuel handling accident, in Mode 6. It has been concluded that this change provides adequate assurance of compliance with Technical Specification requirements.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-044

SYSTEM: 125 Volt DC Essential Power

COMPONENT: AC Distribution Panel Y4

CHANGE, TEST OR EXPERIMENT: Resistor RY100, located in essential instrument panel Y4, was temporarily replaced with a resistor with the same electrical characteristics but different physical characteristics. This FCR requested an evaluation of the temporary mounting of the resistor which was completed June 8, 1982. A permanent resistor was installed July 30, 1982.

REASON FOR CHANGE: The replacement resistor was supplied by Cyberex, Inc., the manufacturer of panel Y4. However, the original resistor model is now obsolete, making a cutom made permanent replacement necessary to be compatible with the original design.

SAFETY EVALUATION: Since the electrical values of the replacement resistor were identical with those of the original resistor, the voltmeter characteristics were not affected. The replacement resistor was mounted safely to prevent the creation of an adverse environment in panel Y4. The ty-rap mounting and the negligible difference between the weights of the original and temporary resistors did not adversely alter the seismic characteristics of the panel. Therefore, the panel retained the capability to perform its safety function as required.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-085

SYSTEM: Once Through Steam Generator (OTSG)

COMPONENT: Steam Generator Tubes

CHANGE, TEST OR EXPERIMENT: Work implemented by FCR 82-085 was completed August 1, 1982. The purpose was to perform tube plugging and stabilization of no more than ten tubes in each of the OTSGs.

REASON FOR CHANGE: This stabilization helped reassure steam generator tube integrity following the 1982 Refueling Outage.

SAFETY EVALUATION: The safety function of the steam generator tubes are as follows:

(a) Provides a pressure boundary for the reactor coolant system.

(b) Provides a heat transfer surface for the exchange of heat from the reactor coolant system to the steam generator secondary side.

In regard to item (a), the structural adequacy of this change was verified by Babcock and Wilcox on Sheet 13 of Field Change Authorization 04-3831-00. This authorization states that with this change, the steam generators meet the pressure boundary requirements of ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, 1968 Addenda, and the original equipment specification.

Item (b), was addressed on Sheet 2 of Facility Change Authorization 04-3831-00 which states that the removal of 300 tubes per steam generator will have no measurable effect on the performance as long as they are randomly distributed.

Prior to this, a total of 14 tubes had been removed from service in Steam Generator 1-2 and 6 from Steam Generator 1-1. Thus, with the removal of no more than 10 tubes from each steam generator, there is no impact on the performance or reliability of the steam generators.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-103

SYSTEM: Containment Air Sample System

COMPONENT: Valves HV5010E and HV5011E

CHANGE, TEST OR EXPERIMENT: This FCR was implemented to reduce the torque switch settings for Limitorque actuators for valves CV5010E and CV5011E, the Containment Hydrogen Analyzer #2 and #1, respectively, discharge line valves. The new settings for both opening and closing are 1.0. Work was completed August 11, 1982.

REASON FOR CHANGE: The original torque switch settings of these two valves caused the valve stems to be overtorqued and, consequently, to bend. The reduced settings will help prevent this.

SAFETY EVALUATION: The safety function of these valves is to isolate containment on a Safety Features Actuation System (SFAS) Incident Level 1. The new settings of 1.0, both for closing and opening, have enhanced the equipment operation and are still sufficient for the valves to perform their function. Therefore, this change has not affected the safety function of these valves. Hence, no unreviewed safety question is involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-302

SYSTEM: Auxiliary Feedwater

COMPONENT: Auxiliary Feedwater Pump

CHANGE, TEST OR EXPERIMENT: The original speed changer motor addition, by the Terry Turbine Company, consisted of one mounting bracket, a Bodine motor, and a Waldron Type 4 coupling which was 2.25 inches in axial length. A Waldron Type 4 coupling which is 3.5 inches in axial length is now being used. This necessitated the fabrication and installation of a spacer plate between the Woodward Governor Bodine motor baseplate and the motor mounting bracket. Work was completed January 25, 1982.

REASON FOR CHANGE: The Waldron Type 4 coupling, which was 2.25 inches, had cracked longitudinally due to insufficient mass to resist the torque applied by the Bodine motor. The use of the longer Waldron Type 4 coupling has eliminated this problem as it contains sufficient mass to withstand the torque of the Bodine motor.

SAFETY EVALUATION: All components of the Auxiliary Feedwater Pumps are Seismic Class 1. The addition of the longer Waldron coupling and spacing plate has a negligible effect on the seismic analysis. This change has increased the reliability of the Auxiliary Feedwater Pumps to perform their intended function.

July 8, 1983

Log No. K83-992
File: RR 2 (P-6-83-06)

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, 1983
<u>Davis-Besse Nuclear Power Station Unit 1</u>

Enclosed are ten copies of the Monthly Operating Report for Davis-Besse Nuclear Power Station Unit 1 for the month of June, 1983.

Yours truly,

Terry D Murray

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

TDM/BMS/ljk

Enclosure

cc: Mr. James G. Keppler
Regional Administrator, Region III
Encl: 1 copy

Mr. Richard DeYoung, Director
Office of Inspection and Enforcement
Encl: 2 copies

Mr. Tom Peebles
NRC Resident Inspector
Encl: 1 copy

IE24
1/1

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346
UNIT Davis-Besse Unit 1
DATE July 8, 1983
COMPLETED BY Erdal Caba
TELEPHONE 419-259-5000 Ext. 196

MONTH June, 1983

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	793	17	784
2	793	18	787
3	795	19	785
4	797	20	784
5	796	21	788
6	798	22	785
7	795	23	785
8	798	24	784
9	788	25	784
10	783	26	784
11	785	27	784
12	786	28	782
13	787	29	784
14	786	30	782
15	781	31	---
16	783		

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.