AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE November 7, 1980

COMPLETED BY Bilal Sarsour

TELEPHONE (419) 259-5000, Ext. 251

MONTH October, 1980

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.

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OPERATING DATA REPORT

DOCKET NO. 50-346
DATE November 7, 1980
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000 Ext. 251

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: October, 1980
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): 934
7. Maximum Dependable Capacity (Net MWe): 890

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	745	7,320	27,845
12. Number Of Hours Reactor Was Critical	0	2,078	13,042
13. Reactor Reserve Shutdown Hours	0	0	28,758
14. Hours Generator On-Line	0	2,008.7	11,883
15. Unit Reserve Shutdown Hours	0	0	1,728
16. Gross Thermal Energy Generated (MWH)	0	4,687,305	24,886,812
17. Gross Electrical Energy Generated (MWH)	0	1,593,559	8,307,070
18. Net Electrical Energy Generated (MWH)	0	1,483,787	7,654,365
19. Unit Service Factor	0	27.4	43.2
20. Unit Availability Factor	0	27.4	49.9
21. Unit Capacity Factor (Using MDC Net)	0	22.8	32.9
22. Unit Capacity Factor (Using DER Net)	0	22.4	32.3
23. Unit Forced Outage Rate	0	14.3	25.6

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: November 6, 1980
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH October, 1980

DOCKET NO. 50-346
UNIT NAME Davis-Besse Unit 1
DATE November 7, 1980
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000, Ext. 251

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
4	80 04 07	S	745	C	4	NA	NA	NA	The unit outage which began on April 7, 1980, was still in progress through the end of October, 1980. 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-Other (Explain)~~
4-Continuation
5-Reduction
6-Other

4
Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

5
Exhibit I - Same Source

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OPERATIONAL SUMMARY
October, 1980

The unit outage which began on April 7, 1980 was still in progress through the end of October, 1980. However, heatup was completed at the end of the month with final preparations underway for the Post Refueling Physics Testing.

The outage was extended longer than expected due to problems with a couple of leaking core flood check valves.

Below is a summary of the major items completed during the entire refueling outage:

1) Cycle 2 refueling was completed.

2) Containment and critical pipe hanger modifications for NRC Bulletins 79-02 and 79-14.

3) The high pressure turbine, one low pressure turbine, and the main generator were inspected.

4) Fuel assembly holddown spring work was completed on the fuel assemblies in the core. The spent fuel assemblies in the spent fuel pool will be modified at a later time.

5) Power operated relief valve indication installed.

6) Completion of the control grade T-sat meter installation.

7) Installation of the Reactor Coolant System to Decay Heat Isolation Valve DH-11 and DH-12 interlock with the pressurizer heaters.

8) Installation of the makeup tank level interlock.

9) Containment Integrated Leak Rate Test and Local Leak Rate Test were successfully completed.

10) Steam generator eddy current examination.

11) 13.8 KV bus fast transfer modification

12) Two Reactor Coolant Pump seals were replaced.

13) Fire Protection System modifications required by NRC.

14) All position indicator tubes on the control rod drive mechanism were replaced.

15) A new seal leakage measurement system for the Reactor Coolant Pumps was installed.

16) Refueling and 18 month surveillance tests were completed.

REFUELING INFORMATION

DATE: October, 1980

1. Name of facility: Davis-Besse Nuclear Power Station Unit 1

2. Scheduled date for next refueling shutdown: March, 1982

3. Scheduled date for restart following refueling: May, 1982

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)?

Reload analysis completed, none identified to date.

5. Scheduled date(s) for submitting proposed licensing action and supporting information. January, 1981

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.

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) 44 - Spent Fuel Assemblies
8 - New 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 1988 (assuming ability to unload the entire core into the spent fuel pool is maintained).

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 78-076

SYSTEM: Seismic Monitoring

COMPONENT: Cabinet C5764A

CHANGE, TEST OR EXPERIMENT: The change proposed by FCR 78-076 modified the mounting details of cabinet C5764A, seismic monitoring panel, to provide acceptable mounting. The work was completed on July 2, 1980.

REASON FOR CHANGE: Physical inspection pursuant to Bechtel/NRC 1254 revealed that this panel may not be adequately supported.

SAFETY EVALUATION: This FCR provides for reinforcement of the seismic monitoring cabinet supports this change provides additional assurance of component integrity and does not create a new adverse environment. This does not constitue an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 78-348

SYSTEM: Reactor Coolant System

COMPONENT: Reactor Coolant Pumps

CHANGE, TEST OR EXPERIMENT: On August 29, 1980, the work package for FCR 78-348 was completed. This FCR provided the following changes to the controls which automatically trip the Reactor Coolant Pumps (RCP):

1) The addition of a 90 second time delay to the automatic closure of the seal return isolation valve. The time delay will only actuate if the RCP is not running and there is a loss of seal injection flow.

2) The resetting of the time delay on the tripping of the RCP motor from 120 seconds to 90 seconds.

3) The resetting of the trip setpoints on FIS-4133, FIS-4233, FIS-4333 and FIS-4433 from 45 gpm to 25 gpm.

REASON FOR CHANGE: This FCR has been implemented to improve the reliability of the Reactor Coolant Pump seals. The addition of the 90 second time delay (Item 1) was intended to prevent spurious closures of the seal return isolation valve caused by short time seal injection flow dips.

The sequential time delay of Reactor Coolant Pump trip (Item 2) and seal return valve closure has been implemented to minimize possible damage to the RCP seals. The lowering of the trip setpoint on the above switches (Item 3) will prevent them from being in the trip state during normal operation due to inadequate Component Cooling Water flow.

SAFETY EVALUATION: The work authorized by this FCR does not degrade the containment isolation function of the seal injection and seal return isolation valves. These valves will still close when required by the Safety Features Actuation System(SFAS). This is the only function of these valves, and therefore, and unreviewed safety issue does not exist.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-191

SYSTEM: Component Cooling Water System

COMPONENT: CCW Surge Tank 1-1

CHANGE, TEST, OR EXPERIMENT: On June 7, 1980 the work package of FCR 79-191 was completed. This FCR changed the Component Cooling Water Surge Tank level setpoint to close Makeup Pump Cooling Water Header Isolation Valve, CC 1460, at 35" instead of the original 45" setpoint.

REASON FOR CHANGE: At a decreasing CCW Surge Tank level of 45", LSLL 3758A closes Makeup Pump Cooling Water Header Isolation Valve, CC 1460. As a result, the Component Cooling Water Supply to the M.U. Pump header which provides cooling water to the M.U. Pumps and Emergency air compressor is lost. The M.U. Pumps must be tripped within several minutes after this isolation to prevent overheating the M.U. Pump and motor bearings and gearbox. With the M.U. Pumps inoperable, there is no M.U. to the RCS when the RCS is at its normal operating pressure (2150 psig).

If a "small break" LOCA occurs at this time, no M.U. flow can be initiated until the RCS pressure drops to approximately 1800 psig. At approximately 1800 psig, a HPI-LPI pump "piggyback" combination could be started to stop depressurization and stabilize RCS inventory. As a consequence, there is little control of the RCS inventory or pressure between 2150 and 1800 psig.

Isolating the M.U. Pump Header at a CCW Surge Tank level of 35" decreasing will buy more time for the operators to find and isolate the leak while still maintaining normal makeup capability. It also provides for a more systematic isolation of the leak such that the various Component Cooling Water headers are isolated according to their importance as critical operating equipment; i.e., from a safety standpoint, the Makeup Pump Header is more important than the Auxiliary Building Non-Essential Header, so therefore, the Makeup Pump Header should be isolated after the Auxiliary Building Non-Essential Header. If the leak is in equipment serviced by the Auxiliary Building Non-Essential Header, CCW Surge Tank Level will stabilize with no interruption of Makeup Pump operation.

SAFETY EVALUATION: The change proposed by this FCR lowered the setpoint level of LSLL 3758A from 45" to 35". This will decrease the CCW Surge tank volume by only 500 gallons and leave approximately 1300 gallons. This will allow normal makeup pump operation to continue while providing more time to locate and isolate the leak.

The above change of isolating CC 1460 at 35" does not involve an unreviewed safety question, or Technical Specification change, nor does it alter or change any commitments, designs, descriptions, or failure analysis as described in the Final Safety Analysis Report, section 9.9.2.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 79-318

SYSTEM: Alarm System

COMPONENT: NA

CHANGE, TEST, OR EXPERIMENT: On August 23, 1979 a test to ensure the operability of the "All Plant" page was performed.

REASON FOR CHANGE: This test was performed to satisfy the requirements of IE Bulletin 79-18. Bulletin 79-18 was initiated because of audibility problems encountered on evacuation of personnel from high-noise areas at the Crystal River Unit 3 Nuclear Plant.

SAFETY EVALUATION: This test involves rerunning a portion of nuclear safety related test TP 100.01, which is described in the Final Safety Analysis Report. However, no hardware changes are involved in completing the test. Therefore, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-096

SYSTEM: Nuclear Instrumentation (NNI)

COMPONENT: +24, -24V DC Power Supplies

CHANGE, TEST, OR EXPERIMENT: On July 18, 1980, Items 1 & 2 of FCR 80-096 were completed. This FCR implemented the following changes to the NNI system:

1) The addition of an 118V AC power source from the "YAU" bus to the NNI "X" cabinet.
2) The addition of an 118 V AC power source from the "YBU" bus to the NNI "Y" cabinet. A new +24V, -24V DC power supply was also installed in the "Y" cabinet.

Item 3 of FCR 80-096 has not been implemented but is pending future action.

REASON FOR CHANGE: This FCR reduces the effect of a "YAU" or "YBU" outage on the NNI system. In addition, it also provides fully auctioneered +24V and -24V DC power supplies to both the NNI "X" and "Y" busses.

SAFETY EVALUATION: The change proposed by this FCR deals with running new power cable and conduit for the NNI system power. The work and equipment are non-Nuclear related, however, the required core drills through existing blockouts make this FCR Nuclear Safety Related. Therefore, this change does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-181

SYSTEM: Safety Features Actuation System

COMPONENT: Logic Modules

CHANGE, TEST OR EXPERIMENT: On July 24, 1980 the implementation of FCR 80-181 was completed. This FCR provided a 60 millisecond time delay in the tripping of the following logic modules in the Safety Features Actutation System:

Channel 1: L211, L221, L241, L311, L411, L511
Channel 2: L212, L222, L242, L312, L412, L512
Channel 3: L213, L223, L243, L313, L413, L513
Channel 4: L214, L224, L244, L314, L414, L514

REASON FOR THE CHANGE: During the analysis of the results from ST 5031.07, Integrated Safety Features Actuation System Test, which was conducted in June 1980, it was discovered that several of the above logic modules tripped before their respective sequence step. In the event of a loss of offsite power with Safety Features Actuation System actuation, and under the previously described condition, this could have allowed an undersired instantaneous loading of the Emergency Diesel generator.

SAFETY EVALUATION: The change proposed by this FCR modified the output modules listed above so that they will be timely blocked by the sequencer. The sequencer will then remove the blocks one at a time, allowing the safety loads to start in their turn. This will prevent all of the possible loads from starting at the same time and overloading the Emergency Diesel Generator.

Per the Final Safety Analysis Report, Table 7-7, the Safety Features Actuation System response time (excluding the response time of the actuated equipment) should be 5 seconds for all Safety Features Actuation System measured variables. After adding the required equipment to provide the time delay, the field measurements have indicated that the delay ranges anywhere from 45 to 82 milliseconds. This time delay of 90 milliseconds for the logic modules listed above is negligible in comparison with the response time requirement provided in the Final Safety Analysis Report. Furthermore, completed response time testing on Safety Features Actuation System has indicated a considerable margin between the actual and required times, therefore, the Final Safety Analysis Report and Technical Specification response times will still be met with the added time delay.

Pursuant to the above, the change proposed by this FCR, does not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-206

SYSTEM: Containment Purge and Ventilation System

COMPONENT: Solenoid Valves SV5005, SV5008

CHANGE, TEST, OR EXPERIMENT: On September 11, 1980, the existing solenoid valves SV5005 and SV5008, ASCO # HTX-8210, were replaced with standard Nuclear qualified solenoid valves, ASCO # NP8316A75E. These valves are located on the actuators for the containment purge and vent isolation valves. CV5005 and CV5008.

REASON FOR CHANGE: The original solenoid valves were special fabricated valves for nuclear service. Any spare parts or valves will also require special fabrication, as all parts utilized in the HTX-8210 Unit are non-standard parts. The new valves are standard nuclear qualified valves, which ASCO currently manufactures.

SAFFTY EVALUATION: The actuator manufacturer, G.H. Bettis Corp., has confirmed that a ASCO valve, # NP8316A75E is an acceptable replacement valve for the existing ASCO valve #HTX-8210. Furthermore, they have also stated that the replacement valve qualifications exceed those of the original valve.

This change will have no effect on the safety function of the containment purge and vent isolation valves. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-214

SYSTEM: Main Steam System

COMPONENT: SV 100-1, SV 101-1

CHANGE, TEST, OR EXPERIMENT: On August 30, 1980 the existing solenoid valves, SV 1001- and SV 101-1, were replaced with standard nuclear solenoid valves. These valves are located on the air supply to the Main Steam Isolation valve (MSIV) bypass valve actuators.

REASON FOR THE CHANGE: To provide nuclear qualified solenoid valves of the right class and service.

SAFETY EVALUATION: This FCR provides for the replacement of the existing solenoid valves ASCO #HB8320A90 with ASCO #NP8320A185E solenoid valves. These new valves are standard nuclear solenoid valves and are qualified for nuclear service in accordance with IEEE Standards 323, 382, and 344.
This change will have no impact on the operability of the MSIV bypass valve. An unreviewed safety question does not exist.