AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE April 8, 1982

COMPLETED BY Bilal Sarsour

TELEPHONE (419) 259-5000, Ext. 384

MONTH March, 1982

| DAY | AVERAGE DAILY POWER LEVEL (MWe-Net) | DAY | AVERAGE DAILY POWER LEVEL (MWe-Net) |
|---|---|---|---|
| 1 | 572 | 17 | 0 |
| 2 | 573 | 18 | 0 |
| 3 | 573 | 19 | 0 |
| 4 | 572 | 20 | 0 |
| 5 | 573 | 21 | 0 |
| 6 | 573 | 22 | 0 |
| 7 | 573 | 23 | 0 |
| 8 | 573 | 24 | 0 |
| 9 | 574 | 25 | 0 |
| 10 | 575 | 26 | 0 |
| 11 | 573 | 27 | 0 |
| 12 | 561 | 28 | 0 |
| 13 | 26 | 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)

# OPERATING DATA REPORT

DOCKET NO. 50-346
DATE April 8, 1982
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000, Ext. 384

## OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: March, 1982
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 | 2,160 | 32,161 |
| 12. Number Of Hours Reactor Was Critical | 292 | 1,708 | 17,938 |
| 13. Reactor Reserve Shutdown Hours | 0 | 0 | 3,334.7 |
| 14. Hours Generator On-Line | 291.4 | 1,707.4 | 16,957.6 |
| 15. Unit Reserve Shutdown Hours | 0 | 0 | 1,731.4 |
| 16. Gross Thermal Energy Generated (MWH) | 550,187 | 3,641,078 | 38,762,603 |
| 17. Gross Electrical Energy Generated (MWH) | 178,313 | 1,202,294 | 12,884,545 |
| 18. Net Electrical Energy Generated (MWH) | 162,489 | 1,124,093 | 12,021,378 |
| 19. Unit Service Factor | 39.2 | 79.0 | 52.7 |
| 20. Unit Availability Factor | 39.2 | 79.0 | 58.1 |
| 21. Unit Capacity Factor (Using MDC Net) | 25.0 | 59.5 | 42.8 |
| 22. Unit Capacity Factor (Using DER Net) | 24.1 | 57.4 | 41.3 |
| 23. Unit Forced Outage Rate | 0 | 0 | 23.0 |

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
May 27, 1982

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 April 8, 1982
COMPLETED BY Bilal Sarsour
TELEPHONE (419) 259-5000, Ext. 384

REPORT MONTH March, 1982

| 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 | 82 03 13 | S | 452.6 | C | 6 | NA | NA | NA | A unit outage was initiated to perform scheduled maintenance and refueling work. 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

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OPERATIONAL SUMMARY
March, 1982

The reactor power was maintained at approximately 68% with the generator gross load at approximately 620 ± 10 MWe until 0324 hours on March 13, 1982 when the turbine generator was taken off line to perform a scheduled maintenance outage and refueling work. Full details on the work items performed during the scheduled maintenance outage will be presented in next month's operational summary.

REFUELING INFORMATION

DATE: March, 1982

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: March 12, 1982

3. Scheduled date for restart following refueling: June 4, 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)?

The final reload analysis for Cycle 3 has been completed and submitted to the NRC (See Serial No. 787 dated March 5, 1982). This analysis identifies several technical specification changes relating to core operational limits and reactor protection system setpoints. An option to provide flexibility in the overall cycle length is also provided therein.

5. Scheduled date(s) for submitting proposed licensing action and supporting information. See response to No. 4 above

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
48 - 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 REQUEST

FCR NO: 77-276

SYSTEM: Communi[illegible] n[illegible]

COMPONENT: N/A

CHANGE, TEST, C[illegible]T: FCR 77-276 was implemented by June 9, 1981 to add visual signalin[illegible] [illegible]n the Diesel Generators 1-1 and 1-2 Rooms, their associated Day Tank [illegible], and the Diesel Fire Pump Room. Rotary amber lights were installed to act[illegible] when the visual page pushbutton, located in the Control Room, is held. Rotary [illegible]ed lights were installed with automatic actuation on a fire alarm, evacuation alarm, or a general emergency alarm.

REASON FOR CHANGE: In the previously mentioned rooms, with the Diesel Generator or Diesel Fire Pump in operation, an operator cannot be paged due to high ambient noise. Normally, an operator wears a headset but may occasionally remove it to take readings, etc. During these periods, the operator cannot be contacted.

SAFETY EVALUATION: This change involves installing a visual signal to call an operator in the Diesel Generator Rooms and the Diesel Fire Pump Room. This change will not affect the safety function of the emergency diesel generator or diesel fire pump system since the change was only in the communications system. FCR 77-276 is safety related since PICAs are involved. Installation in accordance with PICAs will preclude creating new adverse environments. Therefore, this is not an unreviewed safety question.

# COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-269

SYSTEM: Miscellaneous

COMPONENT: Cable Trays ICFF10 and ACFG10

CHANGE, TEST, OR EXPERIMENT: Cable tray covers were added to the vertical section of cable trays KFF10 and ACFG10 per FCR 78-269. These trays are located in the corridor to Mechanical Penetration Rooms 3 and 4 (Room 304). The work was completed August 7, 1980.

REASON FOR CHANGE: The Kaowool was becoming dirty and was beginning to show signs of wear and tear from personnel leaning against it when talking on the Cai-Tronics.

SAFETY EVALUATION: This FCR deals with the addition of metallic tray covers on trays ICFF10 and AGFG10. The purpose of these covers is to provide protection for the Kaowool in the trays. The presence of the covers will not affect any equipment required for a safe shutdown.

# COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-012

SYSTEM: Component Cooling Water, Service Water

COMPONENT: Concrete curbs for pumps

CHANGE, TEST, OR EXPERIMENT: The installation of concrete curbs around pumps located in the following rooms was implemented under FCR 79-012:

(1) Component Cooling Water Heat Exchanger and Pump Room (Room 328)

(2) Service Water Pump Room (Intake Structure)

The work was completed December 23, 1980.

REASON FOR CHANGE: To comply with commitments as stated in the Fire Hazard Report.

SAFETY EVALUATION: The work for this FCR is nuclear safety related because the existing Q supports were modified to install curbs under them. The function of these supports will not be affected by these curbs. Therefore, this is not an unreviewed safety question.

# COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-180

SYSTEM: Various

COMPONENT: N/A

CHANGE, TEST, OR EXPERIMENT: Facility Change Request 80-180 was written to revise ITT Grinnell drawings PM-233F and PM-235B.

REASON FOR CHANGE: To reflect as-built conditions

SAFETY EVALUATION: This FCR provides for drawing revisions only to reflect as-built conditions. No adverse environment will be created. An unreviewed safety question is not involved.

# COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-225

SYSTEM: Component Cooling Water (CCW) and Service Water (SW)

COMPONENT: SW 1399 and SW 1395 interlock

CHANGE, TEST, OR EXPERIMENT: The auto closure of SW 1399 and SW 1395 from CCW pump motor breakers was deleted per Facility Change Request 80-225. The safety activation level 2 would still close SW 1399 and SW 1395. Low SW header pressure at PSL 1376 and PSL 1377A would still close SW 1399 and SW 1395. Administrative control must be provided to prevent opening SW 1399 and SW 1395. At the same time, procedure control to prevent SW pump runout can be implemented. The operators can be given the option of using the circulating water for turbine plant cooling water when it is available. The operators can also monitor SW header pressure when SW is being used for both turbine plant cooling water and CCW cooling. The work for this FCR was completed October 8, 1980.

REASON FOR CHANGE: The standby CCW pump most often is placed in service for CCW, decay heat (DH), high pressure injection (HPI), diesel generator testing and makeup pump testing. When this is done during warm weather, or at times in cold weather when the circulating water temperature is high, the circulating water system will not adequately cool the turbine plant cooling water causing upsets in generator hydrogen #2, turbine oil temperature, etc. At times the circulating water loop 2 is not in service and no method of cooling the turbine plant cooling water exists except the SW system. Also during plant startup or shutdown one CCW pump is used to supply the non-essential equipment such as Reactor Coolant Pumps.

SAFETY EVALUATION: This FCR calls for deleting the auto closure of valves SW 1399 and SW 1395 from CCW pump motor breakers. In the present system, if CCW pump 1 (3)* was running, valve SW 1399 will automatically go closed. Similarly if CCW pump 2 (3)* was running, valve SW 1395 will automatically go closed. These valves govern the flow of service water to turbine plant cooling water system. Thus when both trains of CCW system are in operation, SW to turbine plant cooling water is isolated. During normal operation, only one CCW pump is in operation, however, another CCW pump must often be placed in service for CCW, DH, HPI, diesel generator or makeup pump testing. Thus when the second CCW pump is placed in service (especially in warm weather, or at times when circulating water temperature is high), the circulating water system cannot adequately cool the turbine plant cooling water system. This may cause upsets in generator hydrogen, turbine oil temperature, etc. Also, at times, when the loop 2 circulating water system is out of operation, no method of cooling turbine plant cooling water system exists except the SW system. Also, during startup/shutdown of the plant, one CCW pump is supplying the DH cooler and the other CCW pump is used to supply the non-essential equipment such as Reactor Coolant Pumps. This FCR will enable SW supply to turbine plant cooling water system under these conditions.

The subject valves are also automatically closed by Safety Features Actuation System Incident Level 2 and on low pressure on SW pump headers. The PSLs (1376A and 1377A), isolate SW 1399 and SW 1395, respectively, when SW header pressure goes below 50 psig. This provides pump runout protection for the SW pumps. Thus, if the

*depending on CCW pump 3 electrical lineup

CCW pump interlock to these valves is eliminated, the SW pumps will still be adequately protected against runout.

By removal of the CCW pump interlock, valves SW 1395 and SW 1399 can be opened when two CCW pumps are in operation (assuming no Safety Features Actuation System actuation and normal SW header pressure). This FCR also provides for administrative control to prevent opening SW 1399 and SW 1395 to preserve system redundancy and channel separation of the SW system.

The SW system also supplies water to the containment air coolers, CCW heat exchangers and Control Room emergency condensing units. Since these two valves are isolated on Safety Features Actuation System Incident Level 2, adequate capacity will still be available to above units during a Loss of Coolant Accident.

Pursuant to the above, the changes provided by this RCR do not involve an unreviewed safety question.

# COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-244

SYSTEM: Control Room Heating, Ventilating, and Air Conditioning System

COMPONENT: N/A

CHANGE, TEST, OR EXPERIMENT: FCR 80-244 was implemented to install additional solenoid valves on the Control Room normal ventilation system air lines. These are being added to allow for faster bleed-off time which, in turn, decreases the response time of the valves. The work was completed October 10, 1980.

REASON FOR CHANGE: The response times for the Control Room ventilation system isolation valves was not within the time required by Technical Specifications.

SAFETY EVALUATION: This FCR provides for the addition of solenoid valves to the air lines in the Control Room normal ventilation system. The installation was done similar to that of the existing system which includes Class 1E electrical equipment and seismic Class 1 installation.

The testing accomplished under FCR 80-244 has demonstrated that the response time required for these valves will be met.

This change does not constitute an unreviewed safety question.

# COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-108

SYSTEM: Station Administrative Procedures

COMPONENT: N/A

CHANGE, TEST, OR EXPERIMENT: Facility Change Request 81-108 was written to delete procedure AD 1807.01, Action Item Record from the procedures used at Davis-Besse Nuclear Power Station. This procedure is listed in the FSAR Appendix 13c but is not required per Technical Specification 6.8.1.

REASON FOR CHANGE: Deletion of this procedure will, in part, form the corrective action necessary for CAR 81-01.

SAFETY EVALUATION: Action Items Records are intended to assure completion of assigned tasks not normally tracked by other formalized means, such as a MWO, DVR, or FCR.

The Action Item Record System has not been used in the past year but approximately fifty items are still listed as outstanding. Of these, twelve are noted as NRC commitments. A review indicates that the intent of all twelve has been met and should have been procedurally closed. Of the remaining items, a review indicates that all items are either complete or the requested action was not relevant to a Safety Related Item.

Other systems have been developed to supplement the safety related reviews found in the AIR.

The remaining review items under the category of supplier problems, external requests, administrative, or other, would fall outside the definition of safety related as found in Paragraph 2.2.19 of ANSI N18.7-1972 (committed edition). Control of these items can be on a case by case basis by the individual requesting the action. Therefore, it is considered prudent to delete AD 1807.01.

The procedure is found in Appendix 13c of the FSAR, therefore, this safety evaluation is provided to justify the deletion. However, it is not a procedure required by Technical Specification 6.8.1. Deletion of this procedure is deemed not to be an unreviewed safety question.