AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse #1

DATE Nov. 5, 1984

COMPLETED BY Bilal M. Sarsour

TELEPHONE (419)259-5000
Ext. 384

MONTH October, 1984

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)

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1/1

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE Nov. 5, 1984
COMPLETED BY Bilal M. Sarsour
TELEPHONE (419)259-5000
Ext. 384

OPERATING STATUS

1. Unit Name: Davis-Besse #1
2. Reporting Period: October, 1984
3. Licensed Thermal Power (MWt): 2772
4. Nameplate Rating (Gross MWe): 915
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:

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	7,320.0	54,841.0
12. Number Of Hours Reactor Was Critical	0.0	5,529.0	33,031.5
13. Reactor Reserve Shutdown Hours	0.0	134.8	4,014.1
14. Hours Generator On-Line	0.0	5,489.5	31,641.3
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	0.0	13,941,608	74,985,422
17. Gross Electrical Energy Generated (MWH)	0.0	4,5[illegible]4,151	24,846,344
18. Net Electrical Energy Generated (MWH)	0.0	4,291,557	23,290,256
19. Unit Service Factor	0.0	75.0	57.7
20. Unit Availability Factor	0.0	75.0	60.9
21. Unit Capacity Factor (Using MDC Net)	0.0	67.1	48.6
22. Unit Capacity Factor (Using DER Net)	0.0	64.7	46.9
23. Unit Forced Outage Rate	0.0	11.0	17.3

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		

Notes

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-346
UNIT NAME Davis-Besse #1
DATE Nov. 5. 1984
COMPLETED BY Bilal M. Sarsour
TELEPHONE (419)259-5000 Ext.384

REPORT MONTH October, 1984

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
5	84-09-14	S	745	C	4	NA	NA	NA	The unit outage which began on September 14, 1984 was still in progress through the end of October, 1984. 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
OCTOBER 1984

The unit outage which began on September 14, 1984, was still in progress through the end of October, 1984.

The following are the more significant outage activities performed during September, 1984 and October, 1984.

1. The reactor core was successfully defueled. Control components had been shuffled except for one control rod due to a fuel assembly requiring hold down spring repair.

2. Surveillance specimen activity was successfully completed.

3. Significant cooling tower repairs are in progress involving grouting of the basin and support column repairs.

4. The inspection of upper and lower core barrel, flow distributor and upper thermal shield bolts was completed, and no defective bolts were found.

5. The inspection of lower thermal shield bolts and surveillance specimen holder tube bolts was completed. An inspection found a problem with 35 of the 96 lower thermal shield bolts and 18 of the 72 surveillance specimen holder tube bolts. The lower thermal shield bolts and half of the surveillance specimen holder tube bolts will be replaced.

6. Steam generator eddy current testing was completed. No. 1 OTSG had no pluggable indications. No. 2 OTSG had one pluggable indication.

7. The inspection of the turbine indicated some damage. A high pressure turbine first stage diaphragm blade was found missing and two subsequent rows of blades were badly nicked. Additional cracking of the High Pressure Turbine diaphragm were also discovered. The HPT casing had been steam cut at the horizontal joint requiring heat treatment weld repairs. Turbine repair currently is being performed.

8. During an inspection of the high pressure injection (HPI) swing check valves, it was found that the disc could be spun with sufficient force by hand to cause the anit-rotation stop on the hanger arm to ride up on the disc stop. The root cause appears to be due to a defective component as was originally supplied. The HPI swing check valves are being modified to prevent binding.

9. Condenser tube eddy current testing was still in progress through the end of October.

10. Reactor coolant pump seal rebuild work was still in progress through the end of October.

REFUELING INFORMATION DATE: October 1984

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: June, 1986

3. Scheduled date for restart following refueling: December 22 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.

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COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-427

SYSTEM: Reactor Coolant System

COMPONENT: Reactor Coolant Pumps (RCP)

CHANGE, TEST, OR EXPERIMENT: This FCR allowed for Reactor Coolant Pump seal cartridge to be made of ASME A-182 Gr. F316 instead of ASME A-351 Gr. CF8. This FCR was completed October 19, 1984.

REASON FOR CHANGE: This change was made to reduce the long delivery time for spare seal cartridges for the reactor coolant pumps.

SAFETY EVALUATION: This FCR does not affect the safety function of the RCP seals and does not represent an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-341

SYSTEM: Core Flood

COMPONENT: N/A

CHANGE, TEST, OR EXPERIMENT: Work on FCR 78-341 was completed March 29, 1982. This FCR allowed a change in the piping class sheets to permit the use of 1", 600# ANSI, A-182, F304 stainless steel socket weld flanges for the following equipment:

FSK-M	Pieces
FCB-8-1	16 & 17
FCB-9-2	10
FCB-9-5	19
FCB-10-3	21

This FCR also implemented the core flood tank vent flanges on RO 3753 to be changed from 150# flanges to 600#, A-182, F304 Stainless steel.

REASON FOR CHANGE: FCB piping class sheets require 400# ANSI flanges, forged steel.

SAFETY EVALUATION: Since this change does not reduce system integrity for performing its intended function, this is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-475

SYSTEM: N/A

COMPONENT: N/A

CHANGE, TEST, OR EXPERIMENT: This FCR requested a 10 CFR 50.59 review to justify running a loss of external load, including loss of off-site power, test. This FCR also revised USAR Section 14.1.8.2 to substitute the unit load transient test with unit load rejection on test, TP 800.13. Work on this FCR was completed June 13, 1984.

REASON FOR CHANGE: This FCR addressed the NRC Regulatory Guide 1.68 concerning a loss of offsite power during the power ascension test.

SAFETY EVALUATION: This FCR does not involve an unreviewed safety question

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-045

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST, OR EXPERIMENT: Work implemented by FCR 79-045 was completed October 8, 1980. This involved the installation of a sprinkling system in room 227, which is located in the north-south corridor of the auxiliary building and at an elevation of 565'0".

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

SAFETY EVALUATION: This FCR is non-nuclear safety related except for a "Q" drill. Installation was in accordance with PICA and the core drill report which precludes these portions to create any new adverse environments.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-046

SYSTEM: Fire Protection

COMPONENT: Sprinkler System

CHANGE, TEST, OR EXPERIMENT: This FCR requested the installation of a sprinkler system in room 209 which is located in the east-west corridor of the auxiliary building. This room is at an elevation of 565'0". Work was completed January 13, 1981.

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

SAFETY EVALUATION: Installation was in accordance with the "Q" core drill report and PICA, which precluded those portions from creating any new adverse environment. An unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-161

SYSTEM: High Pressure Injection (HPI)

COMPONENT: HP 4957 and HP 4961

CHANGE, TEST, OR EXPERIMENT: This FCR was incorporated to revise P&ID drawing M-033 to reflect the as-built piping of the HPI lube oil system. The change removed from M-033 HP 4957A and HP4961A and changed HP4957B and 4961B respectively to HP4957 and HP4961. Work involved with this FCR was completed May 1, 1984.

REASON FOR CHANGE: Drawing M-033 had always shown the HPI piping to have an isolation valve on the high side of the differential pressure indicator switches (PDIS) 4957 and 4961 when in reality these did not exist in the plant.

SAFETY EVALUATION: Revision of drawing M-033 to show the removal of valves HP4957A and HP4961A will not affect the HPI pumps safety function. Therefore, this is not an unreviewed safety question. piping anchor by decreasing the structural stresses, Thus insuring that the piping stress will not exceed the allowable limits for long term plant operation. Therefore, this modification will not constitute an unreviewed safety question.

<u>COMPLETED FACILITY CHANGE REQUEST</u>

<u>FCR NO</u>: 79-161

<u>SYSTEM</u>: High Pressure Injection (HPI)

<u>COMPONENT:</u> HP 4957 and HP 4961

<u>CHANGE, TEST, OR EXPERIMENT:</u> This FCR was incorporated to revise P&ID drawing M-033 to reflect the as-built piping of the HPI lube oil system. The change removed from M-033 HP 4957A and HP 4961A and changed HP 4957B and 4961B respectively to HP 4957 and HP 4961. Work involved with this FCR was completed May 1, 1984.

<u>REASON FOR CHANGE:</u> Drawing M-033 had always shown the HPI piping to have and isolation valve on the high side of the differential pressure indicator switches (PDIS) 4957 and 4961 when in reality these did not exist in the plant.

<u>SAFETY EVALUATION</u>: Revision of drawing M-033 to show the removal of valves HP 4957A and HP 4961A will not affect the HPI pumps safety function. Therefore, this is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-178

SYSTEM: Reactor Coolant

COMPONENT: Reactor Coolant Pumps (RCP)

CHANGE, TEST, OR EXPERIMENT: This FCR requested a 10 CFR 50.59 review for starting the fourth reactor coolant pump at 50% power so the interlock setting can be raised to 60% full power. Work was completed August 13, 1979.

REASON FOR CHANGE: This FCR was the result of Document BWT-1781.

SAFETY EVALUATION: Since the margin of safety as defined in the basis for any technical specification was not reduced, this change did not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-089

SYSTEM: Containment Spray

COMPONENT: Pipe Supports/Anchors/Restraints

CHANGE, TEST, OR EXPERIMENT: This FCR was implemented to modify fifteen pipe supports, two pipe restraints and six pipe anchors in the containment spray system. Work related to this FCR was completed September 9, 1983.

REASON FOR CHANGE: These modifications were required as a result of reanalysis of the supports and anchors in accordance with IE Bulletins No. 79-02 and/or 79-14.

SAFETY EVALUATION: These modifications reduce the stresses to an acceptable level and increases the factor of safety. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-186

SYSTEM: N/A

COMPONENT: K5-1 and K5-2

CHANGE, TEST, OR EXPERIMENT: This FCR was implemented to modify the accessory rack for the emergency diesel generators in an effort to increase their natural frequency. Work was completed July 15, 1983.

REASON FOR CHANGE: During the Seismic reevaluation for 0.20g SSE, required by operating license NPF-3, Section 2.C (3)(r), it was determined that due to a shift in the accessory rack's natural frequency its factor of safety against failure was less than desired.

SAFETY EVALUATION: These modifications will not result in any unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-196

SYSTEM: N/A

COMPONENT: Drawing 7749-E17B

CHANGE, TEST, OR EXPERIMENT: This FCR was inacted to modify drawing 7749-E17B to represent the "As-built" conditions of the plant. Work was completed October 22, 1982.

REASON FOR CHANGE: During a review of test results of the SFAS sequencer operation test, MC 7500.34, discrepancies were found between Drawing 7749-E17B and the as built conditions in the plant.

SAFETY EVALUATION: Since this FCR involves no physical changes to plant equipment, no adverse safety question is involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-235

SYSTEM: Auxiliary Feedwater

COMPONENT: N/A

CHANGE, TEST, OR EXPERIMENT: FCR 81-235 was implemented to perform a test to determine the pressure at the AFW suction. Work involved with this FCR was completed September 28, 1982. This FCR allowed for the generation of test procedure 520.48, Auxiliary Feedwater Pump Suction Pressure Dip Test.

REASON FOR CHANGE: The above test provided data for a better scheme to transfer AFW pump suction from the condensate storage tank to the service water system on low suction pressure and switching back to the condensate storage tanks when the pressure had recovered to a satisfactory value.

SAFETY EVALUATION: The safety function of the auxiliary feed water system remains unaffected, therefore, it is concluded that an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 81-249

SYSTEM: Control Room Chlorine Detection System

COMPONENT: AE4863A and AE4893B

CHANGE, TEST, OR EXPERIMENT: FCR 81-249 was performed to lower the freeze protection heated enclosure temperature control setpoint for each chlorine detector from 100°F to 70°F and to alarm at 50°F decreasing. Work involved with this FCR was completed November 4, 1981.

REASON FOR CHANGE: The manufacture of the chlorine detectors, AE4863A and AE4863B, preferred the setpoints of these components to be close to ambient temperatures of 70 to 80°F. The former setpoint of 100°F was within the normal operation window. However, it was higher than necessary causing excessive evaporation of electrolyte, reducing detector reliability.

SAFETY EVALUATION: This change will allow the chlorine detectors to function with less maintenance and more reliable operation. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-071

SYSTEM: High Pressure Injection (HPI)

COMPONENT: HP48, HP49, HP56 and HP57

CHANGE, TEST, OR EXPERIMENT: This FCR modified the valve disc seating surfaces on the HPI system stop check valves, HP48, HP49, HP56 and HP57.

REASON FOR CHANGE: The valve discs were sticking in the closed position. The modifications made will reduce this problem.

SAFETY EVALUATION: These changes do not involve any unreviewed safety questions.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-179

SYSTEM: Containment Hydrogen Purge

COMPONENT: CV5037 and CV5038

CHANGE, TEST, OR EXPERIMENT: This FCR requested a 10 CFR 50.59 evaluation which justified the use of the 4 hydrogen purge exhaust path for venting containment during plant startup instead of the 48" containment purge and exhaust system. Work was completed March 22, 1984.

REASON FOR CHANGE: During previous plant startup, the containment exhaust valves were momentarily cracked opened to compensate for the expansion of the containment vessel atmosphere from ambient to operational temperature. However, because of NRC commitments to have the valves remain closed in Modes 1-4 an alternate approach for venting containment is now required.

SAFETY EVALUATION: It has been concluded that the use of the hydrogen purge line for containment venting during plant startup does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-066

SYSTEM: Station AC Electrical Power Distribution

COMPONENT: Miscellaneous

CHANGE, TEST, OR EXPERIMENT: This FCR implemented the testing of the station electrical distribution system via the degraded bus voltage test. Work involved with this FCR was completed July 23, 1983.

REASON FOR CHANGE: Testing was required to verify the assumptions of the analytical study performed to determine the adequacy of the electrical power system from the standpoint of operability of class 1E equipment during a degraded grid voltage condition.

SAFETY EVALUATION: Since this test does not affect the safety function of the class 1E buses, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-068

SYSTEM: Radiation Monitoring

COMPONENT: RE-8442

CHANGE, TEST, OR EXPERIMENT: This FCR allowed a 10 CFR 50.59 evaluation of the existing secondary plant drainage system discharge radiological monitoring configuration. Work on this FCR was completed February 10, 1984.

REASON FOR CHANGE: This FCR was requested for two reasons. First, to justify the continued plant operation until the new storm sewer radiation monitoring equipment was installed; and second, to close out CNRB Action Item #82-06-501.

SAFETY EVALUATION: This FCR does not constitute and unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-069

SYSTEM: Reactor Coolant System

COMPONENT: Core Barrel

CHANGE, TEST, OR EXPERIMENT: This FCR provided for the performance of a 10 CFR 50.59 review for the potential core barrel bolts failure that possibly existed in light of failures discovered in other B&W plants. Work was completed March 12, 1984.

REASON FOR CHANGE: This FCR was initiated to provide justification for the continued safe operation of the plant.

SAFETY EVALUATION: It was concluded from this FCR that: (1) the degraded system could withstand a seismic event plus a design break LOCA should this remote possibility occur; (2) if the core did drop, it would be detected by the neutron noise monitoring; and, if the core did drop, the safe shutdown of the reactor is ensured. Thus, an unreviewed safety question is not involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-373

SYSTEM: Containment Gas Analysis System

COMPONENT: AIT 5027 and AIT 5028

CHANGE, TEST, OR EXPERIMENT: This FCR was implemented to bring about several modifications to the containment Gas analysis system. Work for this FCR was completed November 23, 1983.

First, the containment Hydrogen Analyzers, AIT5027 and AIT5028 were reading values of 0.6 to 0.8% Hydrogen concentration when the actual hydrogen concentration was 0.2%. Because of this, two modifications were performed. First, the desiccant dryer was removed because it was redundant to the sample cooler and heat tracing. Second, an instrument zeroing system was added to allow the analyzer to be zeroed by containment gas.

Second, this change required the continuous indication of containment hydrogen concentration in the control room. This indicator maintained a range of 0 to 10% hydrogen concentration under both a negative and a positive pressure.

Finally, this FCR allowed for the incorporation of two electrical analog input signals to the plant computer from source transmitters AT 5027 and AT 5028 with a variable range of 0 - 10%. The 0 to 10% range was added to a selector switch range position 2 by a logic network change and new meter scales.

REASON FOR CHANGE: This FCR was inacted for three major reasons. First, the desiccant dryer was removed and the instrument zeroing system was added to eliminate the erroneous high indication of hydrogen in containment. Secondly, the control room indicator with a range of 0 to 10% is the direct result of the NRC's review regarding the Three Mile Island-2 incident. Finally, the computer points were added to the plant computer so an indication of hydrogen concentration would be available.

SAFETY EVALUATION: This FCR and all of its supplements have been subjected to a fire hazard analysis review. However, this FCR does not adversly affect the analysis set forth in the Davis-Besse Unit 1 Fire Hazard Analysis Report.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-355

SYSTEM: 48V Power Supply System

COMPONENT: PORV Block Valve RC11

CHANGE, TEST, OR EXPERIMENT: FCR 79-355 was implemented to allow the PORV block valve (RC11) to be supplied the control and motive power from the emergency power source when offsite power is not available. This FCR also provided the revision of circuit breakers in MCC BF12A in Mode 5 and associated control scheme for the 480V power supply. Work was completed May 23, 1980.

REASON FOR CHANGE: These changes were required by NUREG-0578 and the TMI-2 Lessons Learned Task Force Report.

SAFETY EVALUATION: An unreviewed safety question does not exist.

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TOLEDO EDISON

November 9, 1984

Log No. K84-1356
File: RR 2 (P-6-84-10)

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, October 1984
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 October 1984.

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

Yours truly,

Stephen M. Quennoz /wso

Stephen M. Quennoz
Plant Manager
Davis-Besse Nuclear Power Station

SMQ/BMS/bec

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

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