AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-346

UNIT Davis-Besse Unit 1

DATE April 9, 1986

COMPLETED BY Morteza Khazrai

TELEPHONE (419) 249-5000 Ext. 7290

MONTH March, 1986

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)

IE24
1/1

OPERATING DATA REPORT

DOCKET NO. 50-346
DATE April 9, 1986
COMPLETED BY Morteza Khazrai
TELEPHONE 419-249-5000
Ext. 7290

OPERATING STATUS

1. Unit Name: Davis-Besse Unit 1
2. Reporting Period: March 1986
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): 904
7. Maximum Dependable Capacity (Net MWe): 860
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	744	2160	67,225
12. Number Of Hours Reactor Was Critical	0.0	0.0	35,877.1
13. Reactor Reserve Shutdown Hours	0.0	0.0	4,058.8
14. Hours Generator On-Line	0.0	0.0	34,371.8
15. Unit Reserve Shutdown Hours	0.0	0.0	1,732.5
16. Gross Thermal Energy Generated (MWH)	0.0	0.0	81,297,600
17. Gross Electrical Energy Generated (MWH)	0.0	0.0	26,933,622
18. Net Electrical Energy Generated (MWH)	0.0	0.0	25,233,177
19. Unit Service Factor	0.0	0.0	51.1
20. Unit Availability Factor	0.0	0.0	53.7
21. Unit Capacity Factor (Using MDC Net)	0.0	0.0	43.6
22. Unit Capacity Factor (Using DER Net)	0.0	0.0	41.4
23. Unit Forced Outage Rate	100.0	100.0	28.7

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: October 15, 1986
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

DOCKET NO.	50-346
UNIT NAME	Davis-Besse Unit 1
DATE	April 9, 1986
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TELEPHONE	(419) 249-5000 Ext 7290

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH March, 1986

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
7 Con't	85 06 09	F	744	A	4	LER 85-013	JK	SC	The unit remained shutdown following the reactor trip on June 9, 1985. 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
MARCH, 1986

The unit remained shutdown the entire month of March following the reactor trip on June 9, 1985. Investigation of the causes of the event and corrective actions continues. See NUREG 1154 for further details.

Below are some of the major activities performed during this month:

1) Continued testing as part of the System Review and Test Program.

2) Continued MOVATS testing.

3) Completed rewiring of SFAS Channel 4.

4) The ultrasonic testing for all four Reactor Coolant Pump (RCP) shafts revealed two cracked RCP shafts. This inspection was conducted as a result of the Crystal River Power Plant inspection of the RCPs which revealed a failed shaft.

   All four RCP seals have been removed, and the RCP 2-1 motor and impeller also have been removed from the pump.

   Plans are to replace RCP 2-1 shaft with a spare rotating assembly and conduct hot functional tests. However, the other pump shafts will be replaced after the hot functional tests.

   The RCP shaft replacements will extend the outage longer than expected.

REFUELING INFORMATION DATE: March, 1986

1. Name of facility: Davis-Besse Unit 1

2. Scheduled date for next refueling shutdown: October, 1987

3. Scheduled date for restart following refueling: December, 1987

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: Summer, 1986

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) 204 - 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: 1992 - assuming ability to unload the entire core into the spent fuel pool is maintained.

BMS/005

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 83-075

SYSTEM: Decay Heat

COMPONENT: DH-12A

CHANGE, TEST OR EXPERIMENT: This FCR repaired and replaced valve DH-13A, to its original condition. (Refer to FCR 82-130 for present pinned condition).

This work was completed February 21, 1986.

REASON FOR CHANGE: This valve (DH-13A) had been placed in a pinned position. This FCR will repair and replace valve DH-13A to its original condition so that it can be operated.

SAFETY EVALUATION SUMMARY: The safety function of valve DH-13A is to maintain the pressure boundry of the decay heat system and to fail close upon SFAS actuation to prevent bypass of the decay heat cooler.

This FCR will restore valve DH-13A to its original design condition, which will enable it to perform its safety function.

This FCR based on the above analysis does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO.: 84-160

SYSTEM: Main Feedwater

COMPONENT: MV-601 and MV-612

CHANGE, TEST OR EXPERIMENT: FCR 84-160 changed the open torque switch settings for the Main Feedwater Isolation Valves, MV-610 and MV-612, from 1.5 to 1.0. Work was completed January 9, 1985.

REASON FOR CHANGES: The new torque switch settings are to improve valve reliability. The basis for the new settings is by recommendation from the Torrey Pines Technology Report on Limitorque motor operated valves.

SAFETY EVALUATION SUMMARY: The safety function of the torque switch settings is to close the valve tight enough to prevent any leakage and to break the circuit in case of high mechanical force to prevent overtraveling of the valve stem. The new torque switch settings do not affect the safety function of the torque switch. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 85-003

SYSTEM: Reactor Protection

COMPONENT: Nuclear Instrumentation

CHANGE, TEST OR EXPERIMENT: This FCR will allow taking noise frequency spectra and response data, during a step change on the detector supply of the intermediate and power range channels, during power physics testing following the 1984 refueling outage.

The step changes will occur at 40% and 100% power levels.

This experiment and data gathering was completed January 16, 1985.

REASON FOR CHANGE: This data gathering was a request from the nuclear engineering department of the Ohio State University. This data gathering is in support of an NRC research project. Although there is no commitment to this test, it is presumed to be in the best interest of the industry with respect to possible future requirements on nuclear instrumentation response time testing.

SAFETY EVALUATION SUMMARY: This FCR will allow data to be gathered on the power range and intermediate range detectors. The response to a step change in detector voltage and a noise frequency spectrum are to be recorded. The detector power supplies will be disconnected to perform these tests. The detector string will be made inoperable for a short period. This will constitute an intentional entry into an action statement for each test. These tests will not affect any of the amplification circuitry. Therefore upon reconnection a channel check will be performed to assure that operability has been restored.

Although these tests require an intentional entry into the action statements of Tech. Spec. 3/4 3-1 they will not violate the limiting condition for operation and will not adversely affect any specified safety function. Therefore no unreviewed safety question exists.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-222

SYSTEM: Containment Vessel

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR updated P&ID M-029B and assigned a valve number. This is an additional isolation valve for SFAS and RPS Channel 3 containment pressure sensing line.

Work was completed February 3, 1986.

REASON FOR CHANGE: This valve exists in the plant and this FCR will update P&ID M-029B to reflect field conditions.

SAFETY EVALUATION SUMMARY: No adverse environment has been created by assigning a valve number and updating P&ID M-029 to reflect the field condition. This change will not increase the probability of occurrence or the consequences of an accident. The possibility for an accident of a different type than evaluated in the USAR is not created. This change does not create an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-246

SYSTEM: Auxiliary Feedwater

COMPONENT: HIS-520B/521B

CHANGE, TEST OR EXPERIMENT: This FCR installed mechanical interlock on HIS-520B/521B to prevent movement of switch to ICS position.

This work was completed March 10, 1986.

REASON FOR CHANGE: To prevent movement of mode selector switch to ICS position. This change was incorporated due to an analysis from E. C. Novak of Engineering.

SAFETY EVALUATION SUMMARY: The addition of this mechanical interlock will only prevent ICS control of the Auxiliary Feedwater System. The manual and auto-essential control functions will not be changed in any way. Therefore, the safety function of the Auxiliary Feedwater System will not be affected.

This is not an unreviewed safety question.

COMPELTED FACILTIY CHANGE REQUESTS

FCR NO: 78-181

SYSTEM: Process Radiation Monitors

COMPONENT: RE5327, RE5328

CHANGE, TEST OR EXPERIMENT: This FCR modified the Control Room Emergency vent radiation monitors RE5327 and RE5328 as follows:

1. Change wiring such that when EVS is initiated pumps on monitor will start automatically.

2. Interlock annunciator windows R346 and R347 that under normal operation alarms are off and when EVS is initiated they will perform their intended function.

This work was completed February 22, 1986.

REASON FOR CHANGE: Presently startup of these sample pumps is by manual operation after EVS has been initiated. By automatically starting Control Room air will be monitored as soon as EVS is initiated.

SAFETY EVALUATION SUMMARY: FCR 78-181 has not changed the safety function of the radiation monitors. This FCR has added the following safety features. The modification now provides automatic startup of the monitors with actuation of the Control Room EVS. This feature will prevent any time lag occurring from human error. An interlock for alarm points R346 and R347 was also installed to activate the alarm points after the radiation monitors are started and drawing a sample at normal flow.

Based on the above analysis, this FCR change request does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 85-117

SYSTEM: Main Steam System

COMPONENT: Anchor A3

CHANGE, TEST OR EXPERIMENT: This FCR installed shims beneath the baseplate for anchor A3 on the main steam inlet line to AFPT-1-2.

This work was completed March 10, 1986.

REASON FOR CHANGE: These shims are being added to satisfy NCR 85-0065 by ensuring that bearing for anchor A3 is in full contact with concrete surface.

SAFETY EVALUATION SUMMARY: Anchor A3 has been analyzed by Bechtel in the as-found conditions. The anchor, anchor bolts and piping system, all meet short and long term design criteria. However, the small concrete areas in contact with the base plate was assumed to crush. It cannot be proven that the concrete will or will not crush, and because of this the long term behavior of the anchor under vibration or shock loads cannot be accurately determined by analysis. Shims will be added beneath the baseplate to assure a direct transfer of loads through the shims to the concrete. Based on the above discussion, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-201

SYSTEM: HPI/MU Systems

COMPONENT: HPI and MU valves (HP-2A, B, C, & D; HP-22 & 23; MU-169, 196 and 197).

CHANGE, TEST OR EXPERIMENT: This FCR leak tested the HPI and MU valves using both liquid measuring devices (LMD) at pipe drains or vents, and acoustic emission (AE) inspection equipment for locating any leaks.

This work was completed February 21, 1986.

REASON FOR CHANGE: This information will be used as I.S.I. documentation for ASME Section XI alternative inspection requirements.

SAFETY EVALUATION SUMMARY: The change to both the HPI and MU systems is a temporary test committed to the NRC that will be conducted by providing a measurable leakage path through drain valves in various locations during Modes 5 and 6. As long as a flowpath from the BWST via a decay heat removal pump to RCS is availalbe, the Makeup System does not need to be operable. The test pressure of 1900 psig will not exceed either the preservice hydro OE 2500 psig nor the required mill test of 2500 psig for each length of pipe per ASTM-1530.

The conduct of this test will not affect the integrity of the system tested, therefore, the testing proposed does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-528

SYSTEM: Radiation Monitoring

COMPONENTS: RE 2387 & RE 2389

CHANGE, TEST OR EXPERIMENT: This FCR extended cooling ductwork from RE 2004 to RE 2357 and extended cooling ductwork from RE 2007 to RE 2389.

This work was done September 8, 1983.

REASON FOR CHANGE: Additional cooling will reduce the failure rate of these monitors Ambient air around these monitors is hotter than design specs. Failure of these monitors has been approximately every six months.

SAFETY EVALUAITON SUMMARY: This FCR provides additional cooling to the housing of "Q" listed radiation monitors (RE 2387 - RE 2389) to improve the reliability of these monitors. The stainless steel duct runs are subject to PICA requirements to assure that they do not create any new adverse environment to nearby safety related equipment.

No unreviewed safety question exists.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-440

SYSTEM: Reactor Coolant System

COMPONENT: LTER C3A3, LTER C3A1 and LTER C3B1 Inconel Mounting Bosses

CHANGE, TEST OR EXPERIMENT: This FCR replaced the resistance temperature detector mounting bosses, for LTER C3A3, LTER C3A1, LTER C3B1 in accordance with B&W field change (FC-138).

This work was completed May 13, 1985.

REASON FOR CHANGE: The change was necessary due to damage on the Inconel Mounting Bosses. Loop #1 had developed galled threads on LTER C3B1. Loop #2 (LTER C3A3 and LTER C3A1) bosses were bent and also had galled threads.

SAFETY EVALUATION SUMMARY: The repairs meet the requirements of the applicable specifications for the reactor coolant piping. Field changes 138 and 139 include a verified statement that an appropriate analysis was made and the repairs would be acceptable. Included in field changes 138 and 139 are the required changes to the equipment specification for reactor coolant piping to allow the repairs, with applicable codes and standards. Original design of this piping was 1968 draft of ANSI B31-7.

The repaired bosses have the same dimensions as the original bosses. There will be no adverse effect of the RTE operation.

This is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-018

SYSTEM: Containment Vessel

COMPONENT: Technical Specificaitons

CHANGE, TEST OR EXPERIMENT: This FCR revised tables 3.3.10 and 4.3.10 of the Technical Specification.

This work was completed March 4, 1986.

REASON FOR CHANGE: The revision to Technical Specification Tables 3.3.10 and 4.3.10 was required per NRC requirement NUREG 0737 Item IIF-1-4 containment wide range pressure (FCR-79-425), IIF-1-5 containment normal sump level (FCR-79-408), and containment wide range level (FCR-79-409).

SAFETY EVALUATION SUMMARY: The safety function of the containment pressure normal sump level and wide range level is to monitor the containment pressure and containment level and to inform the operator of post accident condition.

The attached tables for post accident monitoring instrumentation and the surveillance requirements are adequate to verify that these systems are available during post accident conditions and the operability is maintained in the applicable mode. It is concluded that the proposed Technical Specification changes do not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-094

SYSTEM: Main Steam

COMPONENT: Hanger 3A-EBD-19-H12

CHANGE, TEST OR EXPERIMENT: This FCR modified hanger 3A-EBD-19-H12 on the main steam line to the auxiliary feed pump turbines by replacing the present ½" X 1½" tap bolts and ½" concrete fasteners with 3/4" X7" bolts.

This work was completed September 8, 1983.

REASON FOR CHANGE: This modification is required to correct the non-conformance stated in (NCR) 413-82.

SAFETY EVALUATION SUMMARY: The function of this hanger is to allow thermal expansion of the piping system when operating the AFW pumps yet maintain the supporting capabilities. The primary safety function is to provide restraint to the main steam piping during a dynamic loading event, such as an earthquake. This modification will not adversely affect the function of the hanger/piping system. This conclusion is based on the analysis performed by Bechtel as part of their dispostion to (NCR) 413-82.

Based on the above engineering, it is concluded that no unreviewed safety question exists.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-0203

SYSTEM: CTMT Vessel Gas Monitoring System

COMPONENT: FSK-M-HCB-46-4H

CHANGE, TEST OR EXPERIMENT: This FCR revised drawing No. 7749-FSK-M-HCB-46-4H DET-1 to incorporate the changes as noted on drawing NO. 7749-FSK-M-HCB-46-4H. This FCR is for a drawing change only.

This work was completed April 24, 1985.

REASON FOR CHANGE: This drawing change was required to close out Non-Conformance Report No. 84-0090 and reflect the as-built condition.

SAFETY EVALUATION SUMMARY: This FCR involved only the revision of drawing 7749-FSK-M-HCB-46-4H to as-built conditions. An unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-067

SYSTEM: Various

COMPONENTS: MVO-5990, MVO-06010, MVO-6080, MVO-6120, MV-15170, MV-15180, MVRC-020.

CHANGES, TEST OR EXPERIMENT: This FCR replaced the motor and brake with identical motor and ding brake complete with special radiation resistant coils for the following motor operated valves:

MV-1570	MV-5990
MV-15180	MV-6080
MV-06010	MV-020
MV-0612	

This work was completed February 19, 1986.

REASON FOR CHANGE: NRC IE Bulletin 79-01B required a review of environmental qualification of Class IE Electrical equipment. The above equipment must be replaced to meet the requirements of NRC IE Bulletin 79-01B.

SAFETY EVALUATION SUMMARY: The above equipment are being replaced because they do not meet environmental qualification requirements. Since this FCR calls for replacing the motor with identical motor and brake and coil with qualified brake and coil, the operators on these valves will still perform their intended safety function. This replacement will increase reliability of these actuators during accident conditions. The work authorized by this FCR does not create any new adverse environment and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-101

SYSTEM: RCP Makeup/Chem. Feed

COMPONENT: Limit Switches

CHANGE, TEST OR EXPERIMENT: This FCR replaced the Namco type SAI-33 limit switches with Namco model number EA-740-80100 on the following switc. :s:

ZSMU03	ZSMU66A
ZSMU33	ZSMU66B
ZSMU38	ZSMU66C
	ZSMU66D

This work was completed March 3, 1986.

REASON FOR CHANGE: Namco type SAI-33 limit switches or parts are no longer available. These switches must be replaced to meet environmental qualifications of safety related equipment per I. E. Bulletin 79-01B.

SAFETY EVALUATION SUMMARY: The above limit switches are considered part of the safety system display by providing the operator with valve position indications. Existing limit switches are not qualified for nuclear plant use. The replacement switches will increase reliability during accident conditions and meet IEEE 344-1975, 323-1974 and 382-1972 qualifications.

Based on the above analysis, implementation of this FCR does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 78-467

SYSTEM: Seismic Instrumentation

COMPONENT: C5764A

CHANGE, TEST OR EXPERIMENT: This FCR replaced the seismic recording and playback equipment, and also built a new foundation frame for panel C5764A.

This work was completed March 16, 1986.

REASON FOR CHANGE: This equipment was replaced in order to meet the requirements of R. G. 1-12, Rev. 1. Our present equipment does not meet these requirements. This FCR is also changing the foundation frame for panel C5764A from "Q" to "NON-Q".

SAFETY EVALUATION SUMMARY: The support is used to mount panel C5764A. The function of panel C5764A is to house the seismic recording and playback equipment. There is no safety function of the panel support, the panel C5764A and the seismic monitoring system. Since both the support and panel C5764A have no safety function, there is no reason for these items to be "Q". Therefore, this is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-195

SYSTEM: Technical Support Center

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR involves the installation of new cables to the startup test panel for the Technical Support Center. This provided availability at the Technical Support Center multiplexer to monitor the following parameters:

RC PRZR Quench TK LVL (FT)	LT-225
RC PRZR Quench TK Press.	PT-224
MN STM Line 1 RAD (LCPM)	R-787
MN STM Line 2 RAD (LCPM)	R-788
Collect BX out Rad to Lake (LCPM)	R-779
SW out RAD (LCPM)	R-839
SG 1 out STM Temp	T-885
SG 2 out STM Temp	T-901
Shield Bldg. Press, 5014 (IN H_2O)	P-631
Unit Vent Flow (KSCFM)	F-885

Work was completed February 23, 1986.

REASON FOR CHANGE: These parameter are needed at the Technical Support Center Multiplexer for accident assessment purposes.

SAFETY EVALUATION SUMMARY: The FCR involves the installation of cables to the startup test panel for the Technical Support Center. All new cables are being routed through existing raceways or raceways which are being installed by a different FCR. All of these raceways have been previously reviewed. No new adverse environment will be created and no PICA's will be required. Therefore no unreviewed safety question exists.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 77-433

SYSTEM: RC/SP

COMPONENT: Instrumentation

CHANGED, TEST OR EXPERIMENT: This FCR improved instrument transmitter & RTD's inside the containment vessel to promote hermeticity, by replacement and installation of conax wire leads and seals. This FCR also promoted the hermeticity of wire pigtail terminations inside existing Crouse Hinds outlet boxes.

Work was completed May 13, 1985.

REASON FOR CHANGE: To promote hermeticity of seals and wire terminations and instrument logic to withstand C. V. design environment.

SAFETY EVALUATION SUMMARY: The proposed design change promoted and improved the hermeticity of instrumentation inside the containment vessel to withstand adversely affect the operation of the instrumentation or plant, nor the function of the safety protection system.

The proposed change does not involve a technical specification, nor an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-0207

SYSTEM: Reheat Steam/Moisture Separator System

CHANGE, TEST OR EXPERIMENT: This FCR provided guard rails around instrument numbers LSH 140 and LSH 164 (MSR high water level sensors).

This work was completed January 18, 1985.

REASON FOR CHANGE: To protect MSR high water level switches from being bumped by equipment in the vicinity. Plant has tripped at least twice due to bumping of the MSR level sensors in the past. (see DVR 84-142 and LER 84-013).

SAFETY EVALUATION SUMMARY: The modification to the existing design will increase the margin of safety. By increasing the margin of safety, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-100

SYSTEM: NNI

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR modified the NNI system by the following changes:

1. Move the 118 VAC Power supplies for NI-2 and NI-4 from YBU power supply to the YAU power supply.

2. Reconnect TI-RC4A2 (RC Loop 2 TC) from HS-RC42 to TT-RC4A2. Add a new TI-RC4A4 to TT-RC4A4.

3. Reconnect TI-RC4B2 (RC Loop 1 TC) from HS-RC4B2 to TT-RC4B2. Add a new TE-RC4B4 to TT-RC4B4.

4. Reconnect LR-MR16 (MU TK LVL) from HS-MU16 to LT-MU16-1. Add a new LI-MU16-2 to LT-MU16-2.

5. Add a new PI-SP12A1 (SG 2 Press) to PT-SP12A1.

6. Add a new PI-SP12B2 (SG 1 Press) to PT-SP12B2.

This work was completed January 25, 1986.

REASONS FOR CHANGE:

1. Moving the 118 VAC power supplies for one SU range recorder and one intermediate range recorder from YAU will provide redundant recorder power supplies.

2. Changes 2-3 will provide redundant wide range cold leg temp indicators for each RCS loop.

3. Change 4 will provide redundant MU TK LVL indication in the control room.

4. Changes 5-6 will provide redundant SG pressure indicators for each SG.

The above changes satisfy NRC IE Bulletin 79-27.

SAFETY EVALUATION SUMMARY: The changes covered by this FCR will provide redundancy which will allow the unit to go to cold shutdown with an instrument power supply failure as indicated in NRC IE Bulletin 79-27. With these changes the reliability of the unit will be increased as well as the ability to reach cold shutdown with power supply failures in the instrument power supply system. Therefore, an unreviewed safety concern does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 74-160

SYSTEM: Main Feedwater

COMPONENT: MV-601 and MV-612 (Limitorque motor operated valves)

CHANGE, TEST OR EXPERIMENT: FCR 84-160 was originated to change the torque switch settings for the main feedwater isolation valves, MV-601 and MV-612.

Work was completed January 9, 1985.

REASON FOR CHANGE: The new torque switch settings are for improvement of valve reliability. The basis for the new settings were derived from the Torrey Pines Technology Report on Limitorque motor operated valves.

SAFETY EVALUATION SUMMARY: The safety function of the Limitorque motor operated torque switch setting is to allow the valve to close tight enough to prevent any leakage and to deenergize the circuit in case of excessive mechanical force to prevent any damage to the valve, or to deenergize the circuit in closing the valve to prevent overtravel of the valve stem. The new torque switch settings do not affect the switch function of the torque switch. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 83-123

SYSTEM: Decay Heat Removal System

COMPONENT: DH-11 & DH-12

CHANGE, TEST OR EXPERIMENT: This FCR revised Technical Specification page 3/4 5-4 which relates to the Decay Heat Removal System Isolation Valves DH-11 and DH-12.

This work was completed October 17, 1984.

REASON FOR CHANGE: This revision satisfies NRC requirements in NRC letter of September 23, 1983 and the safety evaluation report. This will involve a change in the Technical Specification.

SAFETY EVALUATION SUMMARY: The safety evaluation report in the NRC letter dated September 23, 1983 required Toledo Edison to remove 480 VAC Power from either DH-11 or DH-12 while in modes 1, 2 and 3. Removal of 480 VAC Power will disable the automatic closure signal, which would place us in an action statement of Tech. Spec. 3-5-2. This FCR will revise this Tech. Spec. requirement with the 480 VAC Power disconnected from these valves. There is no practical way that they can open and therefore the interlocks are not needed. Pursuant to the above there is no unreviewed safety question involved.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-132

SYSTEM: Electrical Penetrations

COMPONENT: PILILX/P4LIGX

CHANGE, TEST OR EXPERIMENT: This FCR replaced the Buchanan 600 volt sectional terminal block assembly type CAT-NO-0721 with a Buchanan 600 volt sectional terminal block strap screw medium duty CAT-NO-511 in PILILX and P4LIGX penetration boxes.

This work was completed March 7, 1986.

REASON FOR CHANGE: Existing installed terminal blocks are not environmentally qualified for class IE instrumentation application. These terminal blocks must be replaced to meet environmental qualification of safety related electrical equipment, per NRC IE Bulletin 79-01B.

SAFETY EVALUATION SUMMARY: The terminal blocks are being replaced because they do not meet environmental qualification requirements. Since model changes are only involved, the terminal blocks will still perform their intended safety function after these changes are made. The replacement terminal blocks will increase the reliability. The work authorized by this FCR does not create any new adverse environment and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUESTS

FCR NO: 80-133

SYSTEM: Demin Water

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: This FCR initiated a 10CFR5059 safety evaluation in accordance with action item #3 of NRC Bulletin #80-10 for the following items:

1. High concentration of tritium in the demin water system.
2. Tritium in the component cooling water at abnormally high levels.

This work was completed October 5, 1983.

REASON FOR CHANGE: In response to NRC IE Bulletin No. 80-10.

SAFETY EVALUATION SUMMARY: This FCR calls for a 10CFR5059 review and a safety evaluation on abnormally high concentration of tritium in the demin water and Component Cooling Water System. This review and evaluation was performed by Station personnel in a memo dated July 9, 1980 by W. D. Mills and transmitted to Nuclear Engineering by memo dated February 24, 1981 from T. D. Murray to C. R. Domeck. Nuclear Engineering concurs with the conclusions identified in the July 9, 1980 memo. Based on the above, it is concluded that no unreviewed safety question exists.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-139

SYSTEM: N/A

COMPONENT: Cable Tray Supports

CHANGE, TEST OR EXPERIMENT: This FCR modified cable tray support CS-452-429-116 and CS-202-429-18.

Work was completed December 19, 1984.

REASON FOR CHANGE: This FCR is the result of non-conformance report (NCR) 84-0082 which addresses the non-conforming conditions of cable tray supports CS-452-429-116 and CS-202-429-18.

SAFETY EVALUATION SUMMARY: These cable tray supports do not support a tray that has safety related cables. However these cable supports are located above a cable tray that does contain safety related cables. For this reason these cable supports are required to be seismically qualified. This modification will ensure that these cable trays meet seismic requirements. Therefore, an unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 85-0019

SYSTEM: Auxiliary Bldg. Control Room Air Damper

COMPONENT: SV5311A

CHANGE, TEST OR EXPERIMENT: This FCR changed SV5311A (auxiliary control room damper) from an ASCO Model HTX8210 to an ASCO Model NP 5316A75E in the Control Room Air Handling Unit 2.

This work was completed January 1, 1985.

REASON FOR CHANGE: ASCO Model HTX8210 is being replaced due to failure. This model is no longer available for purchase. ASCO model NP8316A75E is equal or better to the previous ASCO model, and is therefore replacing the old model.

SAFETY EVALUATION SUMMARY: This FCR provides for the changeout of solenoid valve SV5311A from an ASCO Model HTX 8210 to an ASCO Model NP8316A75E which is equivalent or better replacement. No adverse environment will be created.

An unreviewed safety question does not exist.

COMPLETED FACILITY CHANGE REQUEST

FCR NO.: 80-268

SYSTEM: Reactor Core

COMPONENTS: Fuel Assemblies

CHANGE, TEST OR EXPERIMENT: This FCR will allow the performance of the following tasks:

1. Remove 73 designated assemblies from the core.

2. Insert 48 new fuel assemblies (eight of which are in the spent fuel pool) and 25 spent fuel assemblies.

3. Perform fuel assembly and control rod shuffling as directed by B&W.

4. Prepare a reload report and associated technical specification changes for submittal to the NRC.

This work was completed March 31, 1983.

REASON FOR CHANGE: To facilitate nuclear power generation for Davis-Besse Cycle 3.

SAFETY EVALUATION SUMMARY: This FCR provides a refueling specification for refueling of Davis-Besse core for Cycle 3 operation. The safety function of the refueling specification is to ensure loading of fuel in appropriate locations to facilitate operation within reactor care safety limits. Analysis and examination has determined that this core reload will not adversely affect the ability of Davis-Besse Unit 1 to operate safely and the transient evaluation of Cycle 3, which is also attested to by previously accepted analysis.

Based on the above, it is concluded that reloading of the reactor core in accordance with the refueling specification does not involve an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 80-132

SYSTEM: N/A

COMPONENT: N/A

CHANGE, TEST OR EXPERIMENT: FCR 80-132 provided the addition of support brackets to facilitate the installation of temporary shielding over the containment sump tunnel opening. This installation of support brackets was completed July 30, 1982.

REASON FOR CHANGE: The containment sump at elevation 545' needed to be shielded from radiation emitting from the reactor vessel. This protection will allow personnel to enter the containment sump area to perform surveillance test and maintenance work associated with sump operation.

SAFETY EVALUATION SUMMARY: The installation of the support brackets over the containment sump tunnel opening were installed per applicable standards and specifications. This change has resulted in the reduction of personnel radiation exposure while conducting job tasks in the containment sump area.

This does not involve an unresolved safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-169

SYSTEM: Pressurizer

COMPONENT: Valve RC-2A

CHANGE, TEST OR EXPERIMENT: This FCR will change the following setpoints:

1. Change the relief setpoint for the pressurizer pilot operated relief valve from 2255 psig to 2400 psig.

2. Change the pressurizer pilot operated relief valve to close from 2205 psig to 2350 psig.

This work was completed December 13, 1982.

REASON FOR CHANGE: This change was made per the NRC request in NRC I E Bulletin No. 79-05B.

SAFETY EVALUATION SUMMARY: The above change in the setpoint for the PORV in conjunction with the change in the RPS high pressure trip setpoint will avoid actuation of the PORV during anticipated transients. As stated in the attachment to letter from R. E. Ham at B&W to B&W Owners Group Technical Subcommittee for TMI #2 incident related programs (dated April 20, 1979). All safety analyses for B&W plants assume that the vent capacity of the PORV will not be available; thus these analyses are unchanged by the increase in its setpoint.

The change proposed in FCR 79-169 does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 82-127

SYSTEM: 480V Essential MCC

COMPONENT: Breakers BF 1223 and BF 1217

CHANGE, TEST OR EXPERIMENT: This FCR will replace existing 200 amp trip units on essential heaters BF 1223 and BF 1217 with 250 amp trip units.

This work was completed January 14, 1983.

REASON FOR CHANGE: Breakers were causing spurious losses of pressurizer heaters. Current through these breakers is SS 160 amps which is normal for a full heater bank. FCR 78-430 replaced the trip units on the non-essential breakers with 250 amp units. These breakers supply 126 KV loads also.

SAFETY EVALUATION SUMMARY: This FCR will not compromise the integrity of the existing pressurizer essential heater banks or their power supplies.

All modifications are internal to the breaker units and will not prevent the safe shutdown of the plant. Increasing the current rating of the thermal magnetic trip units from 200 amps to 250 amps will improve the reliability of the power supply to the pressurizer essential heater banks by decreasing the susceptibility of these two breakers to improve operation.

Therefore, the work authorized by this FCR does not create any new adverse environment and does not constitute an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 79-019

SYSTEM: Fire Protection

COMPONENT: Smoke Detectors

CHANGE, TEST OR EXPERIMENT: This FCR was initiated for the modification of the fire protection equipment in rooms 600, 601, 602 and 603 on elevation 643. Equipment involved in this modification included smoke detectors, associated conduit and junction boxes.

This modification was completed September 28, 1980.

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

SAFETY EVALUATION SUMMARY: All modifications were made in accordance with PICA and "Q" core drill reports precluded these portions from creating any new adverse environments.

This is not an unreviewed safety question.

COMPLETED FACILITY CHANGE REQUEST

FCR NO: 84-160

SYSTEM: Main Feedwater

COMPONENT: MV-601 and MV-612 (Limitorque torque valves)

CHANGE, TEST OR EXPERIMENT: FCR 84-160 was originated to change the torque switch settings for the main feedwater isolation valves, MV-601 and MV-612.

Work was completed January 9, 1985.

REASON FOR CHANGE: The new torque switch settings are for improvement of valve reliability. The basis for the new settings were derived from the Torrey Pines Technology Report on Limitorque motor operated valves.

SAFETY EVALUATION SUMMARY: The safety function of the Limitorque motor operator torque switch setting is to allow the valve to close tight enough to prevent any leakage and to deenergize the circuit in case of excessive mechanical force to prevent any damage to the valve, or to deenergize the circuit in closing the valve to prevent overtravel of the valve stem. The new torque switch settings do not affect the switch function of the torque switch. Therefore, an unreviewed safety question does not exist.

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TOLEDO EDISON

April 9, 1986

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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, March 1986
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 March 1986.

If you have any questions, please feel free to contact Morteza Khazrai at (419) 249-5000, Extension 7290.

Yours truly,

Louis F. Storz

Louis F. Storz
Plant Manager
Davis-Besse Nuclear Power Station

LFS/MK/ljk

Enclosures

cc: Mr. James G. Keppler, w/1
Regional Administrator, Region III

Mr. James M. Taylor, Director, w/2
Office of Inspection and Enforcement

Mr. Paul Byron, w/1
NRC Resident Inspector

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