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Attachment 1

10 CFR 50.59 Summary Report
of
Facility Changes, Tests, and Experiments
for
Davis-Besse Nuclear Power Station, Unit No. 1

June 2, 1996 - May 23, 1998

(109 pages follow)

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Abbreviations Used:

DCR	=	Design Change Request
FPR	=	Field Problem Report
LM	=	Limited Modification
MOD	=	Modification
SE	=	Safety Evaluation
TM	=	Temporary Modification
UCN	=	USAR Change Notice

10 CFR 50.59 Summary Listing

Initiating Document	Safety Evaluation	Title
DB-OP-00015	98-0030	Safety Tagging of FP 390, Hydrant 18 Isolation Valve
DB-OP-00015	98-0031	Safety Tagging of Turbine Roof Exhaust Fans
DB-OP-00016	97-0067	Temporary Alignment of Valve SC 8, Ammonium Hydroxide Feedpump 1-1 to Low Pressure Feed Water (LPFW) Heaters
DB-OP-00016	97-0068	Temporary Alignment of TPCW Low Level Cooling Water Tank Level (LLCWT) Control Valve Inlet Isolation Valve
DB-OP-02003	98-0001	Disablement of ECCS Alarm Panel 3 Annunciators for Spent Fuel Pool Demineralizer Filter Differential Pressure
DB-PF-04151	98-0009	CCW Augmented Leakage Tests and Check Valve Tests, Isolation of Essential to Non-essential CCW Supply Header
DB-PF-05032	95-0064 R.01	Cable/Hose Routing Alteration to Penetration P59
DB-SP-03135	97-0009	Decay Heat Valve Pit Leakage
DCR 96-0034	96-0046	SFAS Cabinet Power Distribution
DCR 96-0037	97-0015	Fencing Locations
DCR 96-0047	97-0032	Revise the Indicated Capacity for the Reactor Coolant Drain Tank and Pressurizer Quench Tank
DCR 96-0050	97-0037	Revise Tank Capacity for the Miscellaneous Waste Tank and the Miscellaneous Waste Monitor Tank
DCR 96-0052	97-0046	Revise Component Cooling Water (CCW) Valve Positions
DCR 96-0054	96-0090	Change Position of Radiation Element 8432 Sample Line Throttle Valve
DCR 96-0061	97-0011	Resolution of Configuration Concerns for the Station and Instrument Air System
DCR 97-0013	97-0088	Update Circuit Descriptions for L3041
DCR 97-0026	97-0056	Removal of the Turbine Plant Cooling Water Deposit Monitor from Design Documents
DCR 97-0054	97-0045	Revise the Positions of Valves SS-105 and SS-106
DCR 97-0064	98-0010	Boric Acid Evaporator Vent Valve Lineup

Initiating Document	Safety Evaluation	Title
DCR 97-0077	98-0022	Moisture Separator Reheater Second Stage Low Load Valves
FPR 96-0483-901	96-0054	Replacements for the Battery Discharge Alarm Relays
FPR 96-0809-901	98-0002	Sheaves Replacements on EECS Room Cooler Motors
LM 97-0022	97-0023	Service Water Strainer Backwash Valve Logic Change
MOD 93-0056	96-0097	Raising Floor Drains and Leveling Floors in the Auxiliary Building
MOD 95-0056	96-0055 R.08	Resolution to Thermo-Lag Fire Barrier Deficiencies
MOD 95-0057	96-0096	Installation of Condenser Pit Flood Level Switches for Circulation Water Pumps Control
MOD 95-0059	97-0017	Tertiary Demineralizer Vessel Installation
MOD 95-0060	96-0022 R.03	AFPT Main Steam Minimum Flow Lines
MOD 96-0005	97-0025 R.01	Delete CCW Pump Low Flow and High Temperature Trips
MOD 96-0009	96-0041	Add a Reset Button on Door of Breaker BE1401
MOD 96-0034	97-0031	Replacement of Demineralizer Skid Booster Pump
MOD 96-0036	97-0027	Routing the PORV Circuit Through a Different Containment Electrical Penetration
MOD 97-0009	97-0007 R.02	Pressure Relief for Containment Penetration Piping
MOD 97-0015	97-0010	Addition of RCP Motor Lube Oil System Enclosures
MOD 97-0019	98-0003 R.01	OSTG Chemical Cleaning Access
MOD 97-0068	98-0005	Decay Heat Exchanger Supports
MOD 97-0087	98-0025	Protect EDG 1-1 Speed Switch for Appendix R Fire
MOD 98-0013	98-0020	Repair of MS100 and MS101
MOD 98-0018	98-0024	Increase the SFRCS Steam Generator High Level Trip Setpoint and Increase the ICS High Level Limiter Setpoint
SE 96-0040	96-0040 R.01	Cycle 11 Reload Report and Core Operating Limits Report
SE 96-0089	96-0089	Reposition of the Reactor Coolant Drain Tank Nitrogen Regulator Isolation Valves
SE 97-0033	97-0033 R.01	Fuel Assembly Grid Repositioning

Initiating Document	Safety Evaluation	Title
SE 97-0040	97-0040	Potential Radiological Consequences Associated with Floor Drain Extension Removal in the Emergency Diesel Generator Rooms
SE 97-0042	97-0042	Increasing the Utilization of the Low Level Radwaste Storage Facility
SE 98-0017	98-0017	Temporary RCS Letdown Boron Analyzer Indication
SE 98-0021	98-0021R.01	Cycle 12 Reload Report and Core Operating Limits Reports
SE 98-0032	98-0032	Temporary Isolation of MT 19, MT 16A and MT 16B Air Receiver Moisture Traps
TM 96-0024	96-0063	Temporary Modification to Monitor Reactor Coolant Pump (RCP) 2-2 Upper Thrust Bearing
TM 97-0007	97-0060	Install Sight Glass on the Turbine Steam Packing Exhauster and Remove Control Room Annunciator (15-3-A) from Service
TM 98-0001	98-0008	Temporary Demineralizer Connection to the Boric Acid Addition Tanks
TM 98-0016	98-0019	Removal of Polar Crane Pendant and the Bypass of its Weight System Interlocks
TM 98-0024	98-0028	Miscellaneous Waste Monitor Tank Filter Micron Rating
TM 98-0025	98-0029	Gag Main Steam Safety Valve SP17B6
UCN 96-054	96-0080	Grid Description and Stability Analyses
UCN 96-058	98-0006	Control Room Habitability Assessment Posed by Toxic and Explosive Hazards
UCN 96-126	96-0100	Facility Staff Overtime
UCN 96-181	96-0093	Inoperable Fire Barriers or Detection in the Containment Annulus
UCN 96-186	98-0007	Cooldown Following a Seismic Event/Steam Generators Fed with Service Water, Solids Accumulation in the Steam Generators
UCN 96-191	96-0095	Makeup Tank Low Pressure Alarm Setpoint
UCN 96-196	96-0098	USAR and FHAR Changes To Resolve NRC Information Notice 92-018 Hot Short Issues
UCN 96-197	97-0001	Manual Handling of Rod Assemblies
UCN 97-008	97-0005	Implementation of ASME Section XI IWE Requirements

Initiating Document	Safety Evaluation	Title
UCN 97-011	97-0066	Revise Technical Requirements Manual (TRM) Reactor Coolant System Surveillance Frequency
UCN 97-013	97-0013	Changes to the Fire Hazard Analysis Report to Support a 24 Month Fuel Cycle
UCN 97-015	97-0004	Control Room Emergency Ventilation System Operation
UCN 97-023	98-0015	Radiologically Restricted Areas
UCN 97-032	97-0008	Reorganizing Nuclear Operations Responsibilities
UCN 97-035	97-0019	Decay Heat System to AFW Interlock
UCN 97-036	97-0054	Response Time of DH11 and DH12
UCN 97-038	97-0041	Toussaint River Dike Configuration
UCN 97-046	97-0061	Correcting the Room Number Given for Motor Control Center (MCC) E11B
UCN 97-050	97-0016	Revising the USAR for the 24-Month Fuel Cycle
UCN 97-051	97-0034	Primary Filter Media Size Reduction Source Term Reduction Program
UCN 97-055	97-0035	Potential Condition Adverse to Quality Report Initiator's Supervisor Signature
UCN 97-056	97-0021	Fire Hose Houses
UCN 97-057	98-0004	Revise Frequency of Technical Requirements Manual Surveillance Requirement for Channel Calibration of Inside Containment Seismic Sensors to 24 Months
UCN 97-076	97-0026	Changes to USAR Table 5.1-7, Reactor Coolant System Piping Design Data
UCN 97-077	97-0050	Spent Fuel Pool Decay Heat Load
UCN 97-078	97-0028	Physical Separation Criteria for Fluorescent Lighting Cables in Free Air.
UCN 97-087	98-0014	Revision of USAR to Incorporate Additional Wording to Clarify Our Control of Access to Setpoint, Calibration and Test Points
UCN 97-090	97-0030	Spared Circuit BPYBU13A
UCN 97-093	97-0039	Turbine Building Sprinklers
UCN 97-094	97-0055	Fire Hazard Analysis Report (FHAR) Changes Regarding Combustible Loading
UCN 97-095	97-0036 R.01	Filtration Unit for Water Clarity Improvement
UCN 97-102	97-0044	Revision of USAR Figure 9.2-5

Initiating Document	Safety Evaluation	Title
UCN 97-105	97-0065	Revise DC Load List
UCN 97-107	97-0047	Delete Barometric Pressure and Solar Incidence Monitoring
UCN 97-109	97-0053	Procurement Engineering Reorganization
UCN 97-113	97-0057	Staffing Changes in the Operations and Maintenance Organization
UCN 97-118	97-0058 R.01	Main Steam Isolation Valve Bypass Valve Testing
UCN 97-120	97-0059	SFAS Signal Testing
UCN 97-128	98-0011	Eliminate Reference to Specific QA "Hold Tags"
UCN 98-020	98-0023	Reorganization at DBNPS of Manager - Business Service Reporting Responsibility to Vice President Nuclear
UCN 98-021	98-0018	Plenum Removal During Refueling
UCN 98-028	98-0026	Changes to USAR Sections 13.0 and 17.2, Management Responsibility Descriptions
UCN 98-047	98-0033	USAR Changes to Section 11
UCN 98-056	98-0038	Remove Detail on Inoperable CRD Position Indication From USAR Section 4.3.5.3

NOTE

The following are summaries of the Safety Evaluations performed pursuant to 10 CFR 50.59 for the Davis-Besse Power Station, Unit No. 1 from June 2, 1996 through May 23, 1998. For a complete understanding of the safety evaluation of the safety evaluation, the reader must review the actual Safety Evaluation.

SAFETY EVALUATION SUMMARY
FOR
DB-OP-00015 (SE 98-0030)

TITLE:

Safety Tagging of FP 390, Hydrant 18 Isolation Valve

CHANGE:

FP 390, Hydrant 18 Isolation Valve, has been isolated.

REASON FOR CHANGE:

FP 390, Hydrant 18 Isolation Valve, is tagged closed awaiting the delivery of parts to complete repairs to the leaking hydrant. Isolation is necessary for the integrity of the Fire Protection System and personnel safety.

SAFETY EVALUATION SUMMARY:

Operating the Fire Suppression System with valve FP 390 closed, isolating Hydrant 18, will not affect the ability of the Fire Suppression System to ensure that safe shutdown can occur in the event of a fire. This Hydrant is located outside the protected area and is not an Appendix A required hydrant as identified in FHAR Specification 8.2.4.

Operating Fire Suppression System with valve FP 390 closed will not increase any hazards or the release of any radionuclides to unrestricted areas. Closure of the valve does not increase the possibility of fire, flooding, or fire suppression line breaks.

This change in operation will have no adverse effect on safety and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
DB-OP-00015 (SE 98-0031)

TITLE:

Safety Tagging of Turbine Roof Exhaust Fans

CHANGE:

The Turbine Building Roof Exhaust Fans have been removed from service.

REASON FOR CHANGE:

The Turbine Building Roof Exhaust Fans have been removed from service for repair and replacement for an extended period of time.

SAFETY EVALUATION SUMMARY:

The Turbine Building Roof Exhaust Fans remove heat from the Turbine Building by exhausting air to the atmosphere. The Turbine Building Heating and Ventilation System performs no function that is considered important to safe plant operation.

Operating the Turbine Building Heating and Ventilation System with the Turbine Building Roof Exhaust Fans removed from service will not result in temperatures or humidities in the turbine building that would adversely affect Station operation.

The change in system operation of the Turbine Building Heating and Ventilation System with the Turbine Building Roof Exhaust Fans removed from service will not increase any hazards or the release of radionuclides to unrestricted areas.

The operation of the Turbine Building Heating and Ventilation System with the Turbine Building Roof Exhaust Fans removed from service will not result in decreased reliability of the Turbine Building Heating and Ventilation System.

This change does not affect any functions important to safety and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
DB-OP-00016 (SE 97-0067)

TITLE:

Temporary Alignment of Valve SC8, NH₄OH Feedpump 1-1 to Low Pressure Feed Water (LPFW) Heaters

CHANGE:

Valve SC 8 has remained in the open position until the repair can be performed. The pressure boundary will now extend to check valve SC 7.

REASON FOR CHANGE:

During the operation of SC8, NH₄OH Feedpump 1-1 to LPFW Heaters, the valve became damaged and cannot be closed. The valve has remained in the open position until the repair can be performed during an extended shutdown.

SAFETY EVALUATION SUMMARY:

The Condensate System and the Feedwater Chemical Addition System perform no functions important to safety.

Operating the Condensate and Feedwater Chemical Addition Systems with valve SC 8 open will not affect the ability of the condensate system to move the water from the condenser hotwell to the Deaerators. The function to control the chemistry of the feed to the Steam Generators is not affected by having valve SC 8 open, therefore the risk of tube degradation is not increased by this temporary alignment. The change in system operation with valve SC 8 open does not increase the possibility of release of radionuclides to unrestricted areas. Operating with SC 8 open will not increase the degradation of the Steam Generator tubes.

Operating with SC 8 open will not result in an increase in any hazards. The low pressure/temperature and small line size does not increase the possibility of flooding, or high energy line breaks. The piping classifications of the piping to check valve SC 7 are rated at a higher pressure than the condensate piping. Any rupture of this line is bounded by the flood analysis in USAR Section 3.6.2.7.2.

The operation of the Condensate System with valve SC 8 open will not result in a reduction in reliability or performance .

This change in operation will have no adverse affect on the safety functions of the affected SSCs, therefore the proposed temporary alignment change is safe.

SAFETY EVALUATION SUMMARY
FOR
DB-OP-00016 (SE 97-0068)

TITLE:

Temporary Alignment of TPCW Low Level Cooling Water Tank Level (LLCWT)
Control Valve Inlet Isolation Valve (DW 114)

CHANGE:

DW 623, TPCW water fill valve, has been isolated by closing DW 114, LLCWT Level Control Valve Inlet Isolation Valve.

REASON FOR CHANGE:

DW 623, Turbine Plant Cooling Water Fill Valve has been leaking demineralized water past the seat into the Turbine Plant Cooling Water Low Level Tank. The water leaking by the seat causes the level to rise which then results in needing to drain the Low Level Tank. The draining of this tank results in the release of the anticorrosion chemicals to the environment.

SAFETY EVALUATION SUMMARY:

The proposed change does not affect any functions important to safety. Operating the Turbine Plant Cooling Water System with valve DW 114 closed will not affect the ability of the Turbine Plant Cooling Water System to remove heat from the turbine generator accessories and air compressors.

The ability to maintain proper cooling is not affected by the proposed change. The closure of an additional valve does not increase the possibility of flooding or high energy line breaks. This change will not result in a decreased reliability of the Turbine Plant Cooling Water System.

This change in operation will have no adverse affect on the safety functions of the affected SSCs therefore, the proposed change is safe.

SAFETY EVALUATION SUMMARY
FOR
DB-OP-02003 (SE 98-0001)

TITLE:

Disabling of ECCS Alarm Panel 3 Annunciators for Spent Fuel Pool Demineralizer Filter Differential Pressure

CHANGE:

Spent Fuel Pool (SFP) Demineralizer Filter will remain in service until flow decreases to 60 GPM or the monitoring of radiation dose indicates a need for a filter change. The differential pressure instrument is removed from service.

REASON FOR CHANGE:

Changing the filters at 10 PSID results in the generation of unnecessary radioactive waste. The differential pressure instrument is removed from service to prevent over ranging the instrumentation and eliminate the Control Room annunciator that would be sealed into alarm.

SAFETY EVALUATION SUMMARY:

The proposed change does not affect functions important to safety. The ability to maintain proper SPF temperature is not affected by the proposed change. The purification system will still remove fission and corrosion products from the SPF water and the BWST with the filter operating at the higher differential pressure.

The change in system operation does not increase the release of radionuclides to unrestricted areas. This will reduce the number of filters used prior to refueling when the clarity of the water is improved in preparation for fuel handling operations. This should reduce the amount of solid waste generated.

Operating the SFP Filter to a higher differential pressure with the Control Room Alarm disabled will not result in an increase in any hazards. It does not increase the possibility of flooding or high energy line breaks. The annunciator for low or high SFP Purification Flow will alert the operators to the need for filter change or other flow related problems.

The change is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
DB-PF-04151 (SE 98-0009)

TITLE:

CCW Augmented Leakage Tests and Check Valve Tests, Isolation of Essential to Non-essential CCW Supply Header

CHANGE:

Develop and perform integrated CCW leak test to measure leakage between the CCW essential and non-essential headers.

REASON FOR CHANGE:

The purpose of this test is to measure CCW essential leakage by timing a change in CCW Surge Tank level when the CCW non-essential header is depressurized (post seismic event system configuration).

SAFETY EVALUATION SUMMARY:

During this test both essential CCW trains will be in service and the CCW essential loads will be unaffected. One CCW train will be aligned for normal decay heat removal in Mode 5 or 6. The other CCW train will be inservice to provide an essential side pressure source. The CCW non-essential header will be isolated and depressurized. If the CCW non-essential header is required it can be returned to service within a timely manner.

The components listed below were evaluated to determine the effect of securing and depressurizing the CCW non-essential header for approximately four hours in Mode 5 or 6 and were all found to be safe: Component Cooling Water, Decay Heat System, Quench Tank Cooler, RCS Sample Coolers, Boric Acid Evaporators, Waste Gas Compressors, Control Rod Drive Motors and CRD Booster Pumps, RCPS and Letdown Coolers, Makeup Pumps and RCP Seal Injection, Spent Fuel Cooling Heat Exchangers.

Test duration is estimated at four hours. CCW cooling of the spent fuel pool will be isolated during this time. SFP maximum temperature increase should be no more than 2 degrees. It is calculated that the test could last for 20 hours without SFP cooling being required. This test will have no effect on other safety system components.

Water hammer is not a concern because test directs depressurization of the CCW non-essential side with guidance to avoid introducing voids into the CCW system.

SAFETY EVALUATION SUMMARY
FOR
DB-PF-05032 (SE 95-0064 R.01)

TITLE:

Cable/Hose Routing Alteration to Penetration P59

CHANGE:

The proposed activity will temporarily use containment penetration number 59 to route cabling/hoses associated with outage work inside containment.

REASON FOR CHANGE:

The intended cables to be routed through penetration 59 are intended primarily for instruments and communication equipment to be used during the outage. The hoses will be used for various tasks such as steam generator descaling activities.

SAFETY EVALUATION SUMMARY:

The normally installed blind flanges on the exterior and interior of Penetration 59 (P59) will be removed during Modes 5 and 6, including core alterations. A specially fabricated Sealed Penetration Assembly (SPA) will be bolted to the exterior flange of P59 to seal the opening. The SPA will be designed, fabricated, and tested to withstand full containment accident pressure, while allowing for the passage of required services. Hoses will incorporate manual isolation valves on the exterior side of the SPA. Once installed, the only actions necessary to establish full containment closure at P59 will be to close the manual hose isolation valves. The desired cables/hoses will be connected to mating connectors on the interior and exterior faces of the SPA. The safety functions of Containment, the Emergency Ventilation System (EVS), Decay Heat Removal (DHR) and Containment Spray (CS) systems are maintained while DB-PF-05032 is in effect.

The SPA is designed to restrict leakage to less than 10 liters per minute at full containment design accident pressure. The USAP analysis of the fuel handling accident inside containment assumes no restrictions to the release of radionuclides to the environment, whereas installing the SPA conservatively restricts such leakage. Therefore, the USAR analysis of offsite dose consequences is bounding for this situation.

Alteration of Penetration 59 does not pose any undue safety concerns for the plant or personnel. The provisions of DB-PF-05032 preserve the functionality of all the affected systems (i.e., EVS, CTMT, and Hazard Analysis) in the applicable modes.

SAFETY EVALUATION SUMMARY
FOR
DB-SP-03135 and UCN 97-020 (SE 97-0009)

TITLE:

Decay Heat Valve Pit Leakage

CHANGE:

The test acceptance criteria given in surveillance test procedure DB-SP-03135 is being changed based upon the revised calculation of leakage into the Decay Heat Valve Pit following a LOCA.

REASON FOR CHANGE:

Potential Condition Adverse to Quality Report (PCAQR) 97-0071 identified a non conservative assumption in calculation 35.032 rev.0, Decay Heat Pit Vacuum Leakage Test. The assumption was that the potential leak path was limited to the cover at the top of the pit rather than the worse case leak location at the gasketed side wall at the bottom of the pit.

SAFETY EVALUATION SUMMARY:

The Decay Heat Valve Pit Leakage Test is the method used to verify the Decay Heat Valve Pit will provide an adequate water tight enclosure to ensure an acceptable environment for the motor operators of valves DH11 and DH12, the preferred flowpath for post LOCA boron dilution.

The worst case potential leakage path, identified by PCAQR 97-0071, increases the driving head for the leak from 7.2 feet of water to 14.2 feet of water. The revised calculation therefore defines an acceptance criteria that reflects a smaller effective leakage area to assure that water level in the pit does not result in flooding of the valve motor operators.

Several test cases were run to determine the sensitivity of the calculation to both containment temperature and Decay Heat Valve Pit temperature. These test cases demonstrate that a single conservative general acceptance criteria can be generated for all potential conditions for the performance of the test.

It is concluded that revising the USAR description of the surveillance test to reflect the basis of the initial pressure for the Decay Heat Valve Pit Test and the method for determining the final pressure in lieu of defining an explicit acceptance criteria for the surveillance test is safe and will have no adverse affect on plant safety.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0034 (SE 96-0046)

TITLE:

SFAS Cabinet Power Distribution

CHANGE:

This DCR makes changes to the Safety Features Actuation System (SFAS) vendor schematic drawings.

REASON FOR CHANGE:

This DCR will correct discrepancies between the drawings and the field.

SAFETY EVALUATION SUMMARY:

This is a document change only. Past operating experience has shown that the SFAS performs all its safety functions as designed. This change has no effect on any of the functions important to safety of the SFAS. The ability of the SFAS to perform any of its required capabilities will not be impacted in any way.

This change is considered safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0037 (SE 97-0015)

TITLE:

Fencing Locations

CHANGE:

Correct fence locations in Figure 8.2-3 of the USAR.

REASON FOR CHANGE:

As-built fence locations for the south side of the switchyard and Relay House impacts the fencing location as depicted in Figure 8.2-3 of the USAR.

SAFETY EVALUATION SUMMARY:

The 345 kV System does not have any special security requirements since none of the equipment is classified as essential. The fencing only limits access, therefore, there is no effect on plant safety.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0047 and UCN 97-017 (SE 97-0032)

TITLE:

Revise the Indicated Capacity for the Reactor Coolant Drain Tank and Pressurizer Quench Tank

CHANGE:

The DCR revises documents to reflect the actual plant configuration for the reactor coolant system, specifically to modify the indicated capacity for the Quench Tank and the Reactor Coolant Drain Tank. The UCN modifies the tanks' indicated capacity listed in the USAR.

REASON FOR CHANGE:

The UCN and DCR will provide consistency between design drawings and the USAR for both the Quench Tank and the Reactor Coolant Drain Tank.

SAFETY EVALUATION SUMMARY.

Changing the indicated capacity for both the Quench Tank and the Reactor Coolant Drain Tank from a useable volume to a nominal volume has no effect on safety. No physical changes are being made to plant equipment. The equipment will continue to operate as designed.

The change in tank capacity listed in the USAR from 800 ft³ (gross) to reflect a nominal volume of 6700 gallons will have no effect on the ability of the Quench Tank to collect and condense steam. The change in tank capacity listed in the USAR from a useable volume of 655 gallons to reflect a nominal volume of 690 gallons will have no effect on the ability of the Reactor Coolant Drain Tank to collect effluent from plant drains.

Based on the above, changing the indicated capacity in the USAR for both the Quench Tank and the Reactor Coolant Drain Tank to a reflect nominal values under DCR 96-0047 and UCN 97-017 is considered safe.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0050 and UCN 97-018 (SE 97-0037)

TITLE:

Revise Tank Capacity for the Miscellaneous Waste Evaporator Storage Tank and the Miscellaneous Waste Monitor Tank

CHANGE:

The indicated tank capacities of the Miscellaneous Waste Evaporator Storage Tank and the Miscellaneous Waste Monitor Tank will be changed to state the nominal tank capacities.

REASON FOR CHANGE:

A number of different tank capacities have been associated with these tanks in reference to the design document purpose. Design capacity, maximum capacity and usable capacity have all been used with different capacities stated for each on different documents. This change will state the nominal capacities with a note to refer to a controlled procedure for indicated level to volume relationship.

SAFETY EVALUATION SUMMARY:

There are no effects on safety associated with this DCR. The actual tank volumes in the miscellaneous radwaste system have not been changed and no physical change to any SSC is being made by this DCR. No changes have been made in the levels being maintained in the tanks. In addition, the miscellaneous radwaste system is not required to mitigate any accident scenarios. The proposed change will provide for uniformity in the listed information between design documents and the USAR with the ability to determine any tank volume with respect to different tank levels.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0052 and UCN 96-199 (SE 97-0046)

TITLE:

Revise Component Cooling Water (CCW) Valve Positions

CHANGE:

Revise M-036, OS-021 and USAR Figure 9.2-2 to show butterfly valves CC89 and CC90 throttled and to show stop check valves CC147, CC150, CC152 and CC154 normally closed.

REASON FOR CHANGE:

These documents will be revised to ensure that positions for valves in the CCW system are shown consistent in drawings, procedure, and the USAR.

SAFETY EVALUATION SUMMARY:

The above documents currently show CC89 and CC90 as open butterfly valves. These valves are throttled open to ensure that adequate cooling water flow is maintained to the Boric Acid Evaporators. Throttling the valves prevents excessive cooling of the Boric Acid Evaporators, therefore the system function is not affected.

The above mentioned documents currently show CC147, CC150, CC152 and CC154 as stop check valves without a normal position. CC147, CC150, CC152 and CC154 are contained in their respective alternate cooling flowpath for Decay Heat Pump Bearing Housing Cooling. The stop check valves prevent normal cooling flow from back flowing through the alternate flowpath. The alternate flowpaths are isolated, however, by closing the stop check valves, double isolation is provided to prevent cross-connecting the CCW supply loops. The system function is unaffected by closing the stop check valves because all valves in the alternate flowpath require manual operation. Revising the documents to show a stop check valve position convention for CC147, CC150, CC152 and CC154 (normally closed) is an enhancement. Since the normal operation and function of the system are not affected, there is no effect on safety.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0054 and UCN 96-114 (SE 96-0090)

Change:

Change Position of Radiation Element 8432 Sample Line Throttle Valve.

CHANGE:

Radiation Element (RE) 8432 Sample Line Throttle Valve, SW8423A, was changed from an open to a throttled position.

REASON FOR CHANGE:

This change will make procedures and the field configuration consistent with each other.

SAFETY EVALUATION SUMMARY:

The valve, SW8432A, is a 1 1/2" globe valve which can be used for controlling flow to RE8432, Service Water Outlet Header Radiation Monitor. RE8432 is used to monitor/detect radiation in the service water return header.

SW8432A is currently open and is being changed to a throttled position. Throttling this valve will reduce the running time of the sump pumps and possibly extend the life of the pumps.

Throttling SW8432A will have no adverse impact on the function of RE8432. The low flow alarm on RE8432 will continue to sound at a minimum flow requirement of 1.25 gpm.

The proposed valve position change will not adversely impact the operation of the Service Water System. It is concluded that the proposed activity is safe.

SAFETY EVALUATION SUMMARY
FOR
DCR 96-0061 and UCN 97-039 (SE 97-0011)

TITLE:

Resolution of Configuration Concerns for the Station and Instrument Air System

CHANGE:

USAR Figure 9.3-1 is revised to show SA10920, Bypass System Pressure Control Valve, as closed instead of throttled. Also DCR 96-0061 will show the following valves as closed on the Operational Schematics and P & IDs: SA6508, SA6509, SA6510, and SA10920. In addition, a check valve SA6509 will be renumbered as SA6512.

REASON FOR CHANGE:

These changes bring the conventional drawings, Operational Schematics, and P & IDs into agreement with the USAR drawings. Also renumbering the check valve to SA6512 will help to avoid confusing it with solenoid valve SV6509.

SAFETY EVALUATION SUMMARY:

Revising the USAR Figure, the P & ID's and the Operational Schematics to indicate the correct drawing conventions and renumbering the check valve has no effect on the components operation. The operation of the Station and Instrument Air System is not changed. Thus, there is no effect on Plant Safety.

SAFETY EVALUATION SUMMARY
FOR
DCR 97-0013 (SE 97-0088)

TITLE:

Update Circuit Descriptions for L3041

CHANGE:

The number of circuits contained in panel L3041, as indicated on design drawing E-4 Sheet 3, is being deleted.

REASON FOR CHANGE:

PCAQR 97-1602 identified three plant documents that placed the Marsh Transformer Vault Sump Pump on L3041 circuit 5 when it is actually installed on circuit 7.

SAFETY EVALUATION SUMMARY:

The number of circuits on a lighting panel is depicted on E1040A, and it is usually depicted on a lighting schedule. The design and setting of circuit breakers used for protection and coordination of upstream power sources is unaffected by the number of circuits installed on non-Q lighting panels.

The number of circuits on non-safety related lighting panels is not taken credit for in any analysis, is not discussed in the USAR or in the "answers to questions" submitted with the original Safety Analysis Report, or required by Regulatory Guide 1.70. With respect to non-safety related power systems, Regulatory Guide 1.70, Section 8.3 states, "Those portion's that are not related to safety need only be described in sufficient detail to permit an understanding of interactions with the safety related portions." The non-safety related lighting panels are isolated from the safety related portions by protective devices, and their normal loading is so small in comparison to other plant loads that their effect on the safety related load flow conditions is barely detectable. Therefore, the number of circuits on non-safety related panels contributes nothing to understanding any interaction with safety related portions.

These changes are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
DCR 97-0026 and UCN 97-049 (SE 97-0056)

TITLE:

Removal of the Turbine Plant Cooling Water Deposit Monitor from Design Documents

CHANGE:

Removes the Turbine Plant Cooling Water (TPCW) Deposit Monitor from any applicable documents.

REASON FOR CHANGE:

Potential Condition Adverse to Quality Report (PCAQR) 97-0306 identified that the temporary TPCW Deposit Monitor is not installed but is shown on several design documents.

SAFETY EVALUATION SUMMARY:

This portion of the TPCW system affected by this change is not located near any safety related equipment or components and therefore environmental effects such as water spray or pipe whip resulting from a line rupture are not a concern. However, due to the large water volume of the system, flooding resulting from a pipe rupture could be an issue. To address this, an evaluation was performed, which is summarized in USAR Section 3.6.2.7.2.17, that concluded any flooding due to a rupture in the TPCW system will not have any adverse effects on any essential areas in the plant. The USAR also discusses that TPCW piping to the Startup Feedpump in rooms 237 and 238, the AFP rooms, does present a potential flooding hazard.

This change does not affect the evaluations or any accident analysis. The change will not prevent any system from functioning to mitigate the radiological consequences of an accident previously evaluated in the USAR.

The deposit monitor was intended as a temporary installation and is currently not installed. The change will have no adverse effect on any safety related system, component or system operation.

SAFETY EVALUATION SUMMARY
FOR
DCR 97-0054 AND UCN 97-098 (SE 97-0045)

TITLE:

Revise the Positions of Valves SS-105 and SS-106

CHANGE:

Valves SS-105 and SS-106 have been changed from normally open to normally closed.

REASON FOR CHANGE:

Each of these valves is connected to separate sides of the Component Cooling Water Surge Tank baffle and could potentially drain down both sides of this tank in the event of a rupture downstream of these valves.

SAFETY EVALUATION SUMMARY:

The changing of these valves to the closed position will prevent the Component Cooling Water Surge Tank from being drained down through these sample lines in an uncontrolled manner. The position change of these valves will not adversely impact the operation of the Component Cooling Water System. The proposed change to USAR Figure 9.9-2 to reflect valves in the closed position will not adversely impact USAR Chapter 15 - Accident Analysis or other USAR analysis.

The change is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
DCR 97-0064 and UCN 98-0008 (SE 98-0010)

TITLE:

Boric Acid Evaporator Vent Valve Lineup

CHANGE:

Revise design drawings to reflect normal valve line up currently used for the Boric Acid Evaporator vents to reflect that the evaporators are vented to the Station Vent rather than the Gaseous Radwaste System.

REASON FOR CHANGE:

Update the drawings to make them consistent with the operating procedures.

SAFETY EVALUATION SUMMARY:

Valves WC207 and WC386 (vents to Gaseous Radwaste) are changed from open to closed. Valves WC192 and WC208 (vents to Station Vent) are changed from closed to open to reflect that the boric acid evaporators are now normally vented to the station vent.

The proposed activity will have no adverse impact on the ability of the Clean Liquid Radioactive Waste System to perform its function because the Boric Acid evaporators will operate the same regardless of where they are vented. The ability to monitor effluent discharges will not be affected because the station vent is monitored.

This change eliminates the hold up prior to discharge provided by the Waste Gas Decay Tanks. This is not a concern because the requirements of Appendix I to 10CFR50 concerning release of radioactive gases during normal operation will continue to be met. There is a hold up time inherent in the collection and processing of the liquid before it is delivered to the Boric Acid Evaporators which is adequate for significant decay of any fission products that may be in the liquid. Also, the use of a Boric Acid Evaporator is a batch process that typically is performed on a frequency between two weeks and two months. This results in a volume of gases that is small compared to the overall volume of gas discharged from the plant during an operating cycle and adds to the decay time available.

The proposed activity is safe.

SAFETY EVALUATION SUMMARY
FOR
DCR 97-0077 and UCN 98-001 (SE 98-0022)

TITLE:

Moisture Separator Reheater Second Stage Low Load Valves

CHANGE:

Show MS338 and MS353, MSR Second Stage Low Load Valves, as normally closed.

REASON FOR CHANGE:

These valves are operated in manual during startup and shutdown and are closed at full power. Operation in manual has been used due to unsatisfactory system operation in automatic. Manual operation is only necessary for a brief period during turbine startup and shutdown.

SAFETY EVALUATION SUMMARY:

The Moisture Separator Reheater (MSR) Second Stage Low Load Valves, MS338 and MS353 perform no functions important to safe plant operation. They supply Main Steam to the MSR's second stage during startup and shutdown.

The operation of MS338 and MS353 in manual during startup and shutdown is acceptable and has no effect on safe operation of the plant. The normal load change on the turbine is slow enough that the manual control of second stage reheat pressure can be accomplished with out any great difficulty.

With the controller for MS338 and MS353 in manual, the valves will not automatically close following a turbine trip. They are left closed at full power to prevent having a direct path of steam flow to the condenser by way of MS199 or MS314 if one of those two valves should fail to close on a turbine trip. Thus, the automatic function of closing the valves, MS338 and MS353, is now being fulfilled by keeping them closed at full power.

Should they be open and fail to close during a turbine trip, the accident is bounded by the analysis given in USAR Chapter 15.2.11, "Excessive Load Increase", or USAR Chapter 15.4.4.2.3.1, "Steam Line Break Analysis".

Based on the above, this change does not affect safe operation of Davis-Besse.

SAFETY EVALUATION SUMMARY
FOR
REPLACEMENT FPR 96-0483-901 (SE 96-0054)

TITLE:

Replacements for the Battery Discharge Alarm Relays

CHANGE:

Replacement FPR 96-0483-901 selects suitable replacement time delay for the 1P, 1N, 2P and 2N battery discharge alarms.

REASON FOR CHANGE:

The existing battery discharge alarms are implemented by obsolete Agastat 2400 Series relays.

SAFETY EVALUATION SUMMARY:

Class 1E qualified Agastat E7000 Series relays have been determined to be a suitable replacement relay for the obsolete 2400 Series relays. The change will not affect the safety of any structure, system or component. Since the E7000 and 2400 Series relays have comparable weights and mounting, any change in the MCC cubicle's seismic loading is considered negligible.

This change is considered safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
FPR 96-0809-901 AND DCR 98-0006 (SE 98-0002)

TITLE:

Sheaves Replacements on EECS Room Cooler Motors

CHANGE:

Drawings E-1042 and E-1043 contained in the Updated Safety Analysis Report (USAR) as Table 8.3-1 are being revised to incorporate the increase in brake horsepower due to replacing the variable sheaves on Emergency Core Cooling System (ECCS) room cooler motors, MC0314, MC0315 and MC0312, with fixed sheaves.

REASON FOR CHANGE:

The sheaves on all three motors were changed because the variable sheaves were causing vibration. The horsepower change is because the same ratio was not available in the replacement fixed sheaves as the original variable sheaves.

SAFETY EVALUATION SUMMARY:

The changes done in FPR 96-0809-901 have no impact on safety, since the increase of the load on the Emergency Diesel Generator (EDG) is insignificant and the total load is still well below the EDG capacity and the maximum value allowed in Technical Specifications.

The increase in brake horsepower for the EECS room cooler motor from 4.5 horsepower to 5.16 horsepower and corresponding increases in kW and kVA has no effect on safety, since the motor and associated components are rated for 7.5 horsepower.

The proposed actions are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
OF
UCN 98-020 (SE 98-0023)

TITLE:

Reorganization of Manager-Business Service Reporting Responsibility to Vice President
- Nuclear

CHANGE:

Revises the reporting responsibility of the Manager-Business Services from the Director of Engineering and Services to the Vice President - Nuclear.

REASON FOR CHANGE:

Facilitates the organizational realignment at DBNPS.

SAFETY EVALUATION SUMMARY:

The proposed change does not affect the safety function of any SSCs and does not affect the operation of any plant systems. The change is solely administrative in nature as it revises reporting responsibilities.

All functions of the Business Services department remain unchanged and continue to be performed. The technical qualifications and requirements continue to be provided by the Toledo Edison Nuclear Organization.

As required by Technical Specification 6.2.1a, the new organizational structure provides defined lines of authority, responsibility, and communication. The Staff training requirements of Technical Specification 6.4 are not changed by this reorganization.

Commitments to ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel, and ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants, continue to be met as described in USAR Table 17.2-1. The basis for NQAM and Design Control procedures are not altered and continue to identify responsibilities and methods for complying with specific design requirements and activities.

It is therefore determined that the proposed change is safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 93-0056 (SE 96-0097)

TITLE:

Raising Floor Drains and Leveling Floors in the Auxiliary Building

CHANGE:

This modification provides for the raising of four floor drains within the Auxiliary building. The adjacent concrete slabs will be leveled appropriately to match at all locations. The drains will be raised by the addition of a section of pipe to the existing drain pipe. The floor slabs will be leveled to match the elevation of the top of the floor drains by the addition of topping slabs to the existing slabs.

REASON FOR CHANGE:

The proposed change will reduce difficulties encountered while moving equipment on the sloping floors surrounding the drains.

SAFETY EVALUATION SUMMARY:

The slab work associated with this modification is minor in nature. It will take place outside of the slab reinforcing steel, so there will be no diminishing of the strength of the slab. Also, since the original analyses considered the slab weight based on the high point slab thickness and the high point thickness has not changed, the slightly increased slab weight is still within the parameters used for the original slab and supporting structural steel beam analyses.

The floor drains and the surrounding floor will be raised only locally at the drains, so the flood height considered in any flooding analysis will not be affected.

The proposed change is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
OF
MOD 95-0056, UCN 96-021, UCN 96-205, and UCN 97-117
(SE 96-0055 R. 08)

TITLE:

Resolution to Thermo-Lag Fire Barrier Deficiencies

CHANGE:

Fire barriers composed of Thermo-Lag will be replaced with an alternate material. FHAR is being revised to reflect this modification. Other changes include: a requirement to protect Containment Air Cooler Fan 3 and elimination of its fire barriers, the elimination of the need for protecting the structural steel columns in rooms 110, 113A and 114, changes to the fire barriers on the penetrations in the annulus, and the deletion of the fire barrier on the circuit for Service Water Pump (SWP) 3.

REASON FOR CHANGE:

Thermo-Lag was determined to be an "indeterminate" level of protection. DBNPS established an hourly fire watch, however, the NRC stated that this was a temporary measure in response to this issue and is unacceptable as a permanent resolution.

SAFETY EVALUATION SUMMARY:

Material acceptability for the intended environment has been evaluated and found to be acceptable for outage and non-outage and for the effects of radiation and spray inside containment. They are also qualified for a loss of coolant accident environment such that the material will not disintegrate and cause strainers, drains, and sumps to become plugged. It has no adverse chemical reactions to the raceways, cables, pipings, copper wires, supports, penetrations seal materials, existing fire-proofing materials, or any other component with which the material is likely to come into contact. Also the following hazards were evaluated and determined not to be impacted: Pipe breaks, pipe whip, temperature, humidity, radiation, jet effect, pressure, flood, sabotage, heavy loads, toxic gases, hazardous materials, wind, tornado, electrical noise, and missiles.

An engineering evaluation was done to review the impact on plant risk during the time the fire barrier is removed from the protected entity. The conclusion of this evaluation is that the increase in risk to the plant due to the removal of Thermo-Lag per the phased approach is small and acceptable. The continued use of 1-hour roving fire watches is acceptable for these rooms.

The CAC Fan 3 could not be used for shutdown analysis purposes in case of a fire in some areas without additional modification and analysis. It was decided that this flexibility will not be provided for in the design and therefore the CAC-

3 circuit fire barrier requirements in the FHAR are being eliminated. CAC-1 and CAC-2 are the credited trains for shutdown in the analysis.

The elimination of the requirement for a fire barrier on the columns in rooms 110, 113A, and 114 has also been evaluated. The evaluation of the fire hazards in these rooms has determined that the columns in these rooms previously protected with Thermo-Lag do not require protection and the Thermo-Lag material will be removed.

Other changes to the FHAR that were evaluated include; eliminating the need for a fire barrier on penetration P2P5F, adding a 3 hour fire barrier on penetration PBL4E and changing the fire barriers on penetrations P1P3B and P2C5G from Radiant Energy Shields to 3 hour rated barriers and eliminating the need for a fire barrier on conduit 36201D in room 53. The elimination of the barrier on P2P5F is due to a reanalysis of the circuits which go through this penetration and the relocation of one circuit by a previous modification. Eliminating the need for a fire barrier on conduit 36201D in room 53 is due to the decision that Service Water Pump 3 is not required to be protected to meet Appendix R. The other changes are necessary due to the withdrawal of an exemption request.

In summary, the proposed modification and FHAR changes are considered safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 95-0057 (SE 96-0096)

TITLE:

Installation of Condenser Pit Flood Level Switches for Circulation Water Pumps Control

CHANGE:

Sixteen new level switches are being installed in the condenser pit to detect flooding in the pit area. Pressure switches currently installed will be removed.

REASON FOR CHANGE:

Time delay to trip the pumps after detecting a flood is reduced from 60 seconds to 20 to reduce the unnecessary inflow of flood water into the pit. The other switches are being removed due to their redundant function.

SAFETY EVALUATION SUMMARY:

The level switches are considered more reliable because they can detect a variety of flood conditions, because their function is independent of size and source unlike the pressure switches. The proposed modification does not alter the basic function of flood detection or mitigation. The desired function is not only maintained but the proposed modification also increases the safety margin by providing more reliable, and effective flood detection over a wider range of leakages.

The Circulating Water (CW) pump discharge valve closing time is increased to less than 90 seconds. The closing time is not important to safe operation, because even if the valves stay fully open after the pumps are tripped, the flood water flowing through the pumps will establish a level in the pit to balance the water level in the CW canal below the 585 level. USAR analysis assumes the valves do not close during this event.

A new annunciator window is installed in the control room to alarm the flooding condition, in addition to the existing computer alarms from the individual sumps. This improves operator interface and is thus not detrimental to plant safety.

SAFETY EVALUATION SUMMARY
FOR
MOD 95-0059 AND UCN 97-061 (SE 97-0017)

TITLE:

Tertiary Demineralizer Vessel Installation

CHANGE:

USAR Section 9.2.3.2 is being changed to depict the installation of a tertiary demineralizer vessel in the Makeup Water Treatment System.

REASON FOR CHANGE:

The purpose of the tertiary demineralizer vessel is to aid the primary and secondary ion exchangers in the removal of contaminants before entering the Demineralized Water Storage Tank.

SAFETY EVALUATION SUMMARY:

Sodium concentrations shall not exceed 3 parts per billion maximum. The tertiary demineralizer vessel will reduce contamination ingress into secondary system makeup water and primary system makeup water. Reduced contaminants in these systems will increase the life expectancy of components. The tertiary demineralizer vessel will only be used on an as needed basis. The use of the tertiary demineralizer vessel will be controlled by procedure.

The makeup water treatment system is not interconnected with any safety features system and is not essential for safe shutdown of the station.

Therefore, installing the tertiary demineralizer vessel and it's components will not adversely affect any structure, system or component important to safety.

SAFETY EVALUATION SUMMARY
OF
MOD 95-0060 and UCN 97-086 (SE 96-0022 R. 03)

TITLE:

AFPT Main Steam Minimum Flow Lines

CHANGE:

Install two 1-1/2" lines diverting steam flow to feedwater heater E6-2 from the auxiliary feed pump turbine main steam supply lines.

REASON FOR CHANGE:

The AFW pump turbine main steam minimum flow lines provide forward flow through the steam lines in order to reduce the disc impacts on MS734 and MS735. Reducing disc impacts is expected to extend the time between maintenance intervals and ensure that the valves can meet the required reverse flow leakage criteria. MS734 and MS735 are required to close and isolate during certain breaks of the AFW pump turbine main steam supply lines with concurrent failure of the associated steam isolation valves.

SAFETY EVALUATION SUMMARY:

It is concluded that the AFW system is capable of operating from full main steam pressure all the way to the Decay Heat Removal System initiation temperature, during a postulated rupture of the new line in the turbine building. The seismic evaluation concludes that the system design ensures that a crack or break in the turbine building will not preclude access to the AFP rooms by obstructing access with a jet stream. An AOV is installed in the new line to isolate the feedwater heater. Failure to perform its function will have a negligible effect on AFP operation. This isolation will protect the Feedwater Heater from damage.

Check valves are incorporated in the case of a steam line break to isolate the opposite train from a break of the 6" line in the affected AFP room. They merely act as part of the piping pressure boundary and perform no safety function. Relief valves have adequate capacity to protect the heat exchanger from over pressurization. Also the total flow in the new line will not have any adverse effect on the feedwater heater during operation.

A break or crack in the piping routed in the Turbine Building will have an insignificant effect on the AFP room environmental conditions as previously concluded. The piping will be insulated and have an insignificant effect on the heat of the AFP room.

The new piping is designed and constructed to the requirements of ASME/ANSI B31.1 Power Piping. The modification will have no adverse effect on the safety functions of the affected SSCs and will be designed, constructed and installed in accordance with the established standards for these SSCs. It is therefore safe.

SAFETY EVALUATION SUMMARY
OF
MOD 96-0005 (SE 97-0025 R. 01)

TITLE:

Delete CCW Pump Low Flow and High Temperature Trips

CHANGE:

This modification deletes the low flow and high temperature switches from tripping the CCW pump and still maintains the automatic start of the redundant CCW pump and valve transfer logic. This modification also deletes the local control stations that are located in room 328 and could cause a trip because of a fire in room 328.

REASON FOR CHANGE:

The removal of these trip functions will eliminate a potential source for a spurious trip of an operating CCW pump.

SAFETY EVALUATION SUMMARY:

The CCW pump has interlocks to automatically start a pump in the opposite train when flow is low. Automatic start is not affected. Manual start circuits are being affected due to the removal of manual start switch, but there is no effect on safety because of the local control is done at the switchgear rather than the pump room. There are no requirements for the trip features that are being deleted on the CCW pumps. Deletion of these switches from the trip circuit will have no effect on the safety of the CCW pumps or the CCW system. The USAR and Technical Specifications are unaffected by this modification.

SAFETY EVALUATION SUMMARY
FOR
MOD 96-0009 (SE 96-0041)

TITLE:

Add a Reset Button on Door of Breaker BE1401

CHANGE:

Modification 96-0009 will add a reset button on the door of breaker BE1401.

REASON FOR CHANGE:

This modification will allow the closure of the low speed latching starter relay LX contacts without opening the breaker cubicle door.

SAFETY EVALUATION SUMMARY:

The addition of the reset button will not affect the seismic qualification of the Motor Control Center E14. The reset button is also seismically acceptable per calculation C-CSS-060.05-005.

The reset button will only be used in the event of a control room or a cable spreading room fire to close the CAC Fan 1-1 slow speed breaker. There is no change in the operations for any other situations and the CAC fan can be run from the control room.

Previously the FHAR directed the operators, during a control room or cable spreading room fire, to close the CAC Fan 1-1 slow speed breaker (BE1401) at Motor Control Center E14. This was incorrect in that there is no slow speed breaker. The revised operator action is to start the CAC Fan on low speed by pushing the low speed latching starter relay LX reset button on the breaker BE1401 cubicle door.

This change is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
MOD 96-0034 and UCN 97-052 (SE 97-0031)

TITLE:

Replacement of Demineralizer Skid Booster Pump

CHANGE:

This modification removes demineralizer skid booster pump (P261) and replaces it with one cartridge type inlet filter vessel and filter, five stainless steel demineralizer vessels, one cartridge type outlet filter vessel and filter, one positive displacement chemical addition pump, sparger line for the demineralizer vessels, and relief valves for the demineralizer and filter vessels. The USAR is revised to reflect these changes as well as other discrepancies noted in the affected USAR section.

REASON FOR CHANGE:

Inlet filter and demineralizers are equivalent replacements for the existing demineralizers and inlet filters. A post filter is being added to provide a means of capturing any resin fines that may come out of the demineralizers. A sparger line is being added to allow breaking up of the resin to improve the flow before discharging it for disposal. Relief valves provide overpressure protection for each of the vessels as opposed to one valve for all vessels as previously provided.

SAFETY EVALUATION SUMMARY:

This activity has no effect on safety since the affected structures, systems and components have no function important to safety. It is concluded that the proposed change is safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 96-0036 (SE 97-0027)

TITLE:

Routing the PORV Circuit Through a Different Containment Electrical Penetration

CHANGE:

Reroute the circuit which supplies the 125 VDC electrical power for opening the Power Operated Relief Valve (PORV). The new route will be through electrical penetration P2C5G instead of electrical penetration P2P5F.

REASON FOR CHANGE:

Per the FHAR, the PORV (valve RC2A) is required to be operable; therefore, an Appendix R fire barrier was constructed to protect this circuit. Since electrical penetration P2C5G, which is adjacent to P2P5F, is also protected by an Appendix R fire barrier, routing the PORV circuit through P2C5G maintains the Appendix R commitment. Rerouting this circuit will allow a future plant modification to propose eliminating the Appendix R fire barrier currently installed on electrical penetration P2P5F.

SAFETY EVALUATION SUMMARY:

Presently, all circuits routed through P2C5G are essential train 2 circuits. The PORV is an essential train 2 circuit as well, so using P2C5G does not violate electrical separation. The electrical penetration conductor size being used for the PORV circuit is number 2 AWG. The electrical ratings of penetration the will not be exceeded by the PORV circuit which must supply a maximum holding current of 0.35 Amps at 90 to 140 VDC.

The proposed changes to the PORV operation circuit have been evaluated and they still ensure sufficient voltage is available at the PORV solenoid for reliable operation.

The Appendix R fire barriers protecting P2P5F and P2C5G are both designed to provide the same level of fire protection. Therefore, using P2C5G in place of P2P5F will not reduce the degree of fire protection being provided for the PORV operating circuit.

Based on the above discussion, the changes proposed by MOD 96-0036 are considered to be safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 97-0009, Supp. 01 (SE 97-0007 R. 02)

TITLE:

Pressure Relief for Containment Penetration Piping

CHANGE:

Installation of a small bypass check valve around the inside containment isolation valve for penetrations P3 (CCW Containment Header Supply) , P4 (CCW Containment Header Return) , P12 (CCW Control Rod Drive Motors Containment Header Supply) , P47A (Core Flooding Tank Sample) , and P48 (Pressurizer Quench Tank Recirc) , as well as a relief valve for P13 (Containment Normal Sump Discharge).

REASON FOR CHANGE:

Prevent the potential overpressurization (thermally induced overpressurization) of piping associated with the containment isolation valves following a design basis LOCA in response to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

The piping system changes provided by this modification will have no influence on any of the identified safety or important to safety system operations since the addition of the piping and tubing hardware are passive components which are branched off the main process flow line and only become an active part of the system in the event of a LOCA.

The valves being added are part of the containment isolation boundary and satisfy the design requirements of General Design Criteria (GDC) 55 or 56, as applicable. The new check valve materials are suitable for containment accident environment for temperature and radiation.

The new check valves and relief valve were leak tested following installation to satisfy 10CFR50, Appendix J requirements and will continue to be included in the Appendix J leak rate testing program.

When considering the amount of hardware added by this modification, combined with the fact that their installation conforms to existing pressure and seismic design requirements, it can be concluded that the small amount of hardware has a negligible effect on the probability of occurrence of a malfunction of any of the associated piping systems. The overall probability of a malfunction is reduced because this modification will maintain penetration pressures within acceptable pressure limitations following a LOCA and subsequent containment penetration isolation.

SAFETY EVALUATION SUMMARY
FOR
MOD 97-0015 AND FPR 97-0127-701 (SE 97-0010)

TITLE:

Addition of RCP Motor Lube Oil System Enclosures

CHANGE:

Modification 97-0015 will add an enclosure to each Reactor Coolant Pump (RCP) motor to enclose the drain line from the point that it exits the existing enclosure to immediately downstream of the isolation valve.

FPR 97-0127-701 will add enclosures to the pressurized oil lift system manifold located outside the existing oil cooler shield wall.

REASON FOR CHANGE:

Modification 97-0015 and FPR 97-0127-701 were initiated to correct the RCP motor lube oil collection system by the addition of enclosures over high pressure piping and instrumentation and control components located outside the existing lube oil cooler shield and about the lower bearing oil level control drain line and isolation valve for each pump. 10 CFR 50, Appendix R requires that the reactor coolant pump lube oil systems be equipped with an oil collection system capable of the collection of lube oil from potential pressurized and non-pressurized leakage sites.

SAFETY EVALUATION SUMMARY:

The new enclosures are designed to capture any oil leakage from the upper bearing pressurized oil lift system manifold located outside the oil cooler shield wall and the lower bearing level control drain line and isolation valve and to provide a pathway to an existing RCP motor oil drain thereby preventing initiation of a fire due to oil leakage onto hot piping. The enclosures are seismically designed to provide oil collection during and following a seismic event.

The addition of the oil enclosures proposed by Modification 97-0015 and FPR 97-0127-701 bring the oil collection system into compliance with the requirements of 10 CFR 50, Appendix R. These additions are safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
MOD 97-0019 and UCN 97-127 (SE 98-0003 R. 01)

TITLE:

OSTG Chemical Cleaning Access

CHANGE:

The proposed change modifies Steam Generator Blowdown lines, RCS hot leg piping restraints, personnel platforms, handrails and ladders within the Containment. This also affects the pipe whip restraint description in USAR Figure 3.6-30.

REASON FOR CHANGE:

This will provide necessary clearances for insertion of chemical cleaning apparatus into the steam generator hand-holes to facilitate chemical cleaning of the steam generators during Refueling Outage 12.

SAFETY EVALUATION SUMMARY:

The pipe whip restraints on the RCS hot leg piping are no longer required. The Standard Review Plan Section 3.6.3, Leak Before Break, in conjunction with GDC 4 allows the analytical exclusion of the dynamic effects of a postulated pipe rupture. This reanalysis supports the removal of whip restraints on the RCS hot legs.

The Steam Generator Blowdown piping changes will not change the pipe break locations for the blowdown piping as originally determined. The changes meet all the ASME B&PV Section III Code and MEB 3-1 requirements.

The Reactor Coolant Pump platform bolted joints are designed to maintain the seismic II/I design requirements.

The proposed actions are safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 97-0068 and UCN 97-129 (SE 98-0005)

TITLE:

Decay Heat Exchanger Supports

CHANGE:

Provide additional reinforcement for the heat exchangers structural supports and pipe anchors. Additionally, the analytical model of the Decay Heat exchangers, supports and piping during an seismic event is revised.

REASON FOR CHANGE:

The SQUG evaluation of the decay heat exchangers, it was identified that the loads due to the attached piping were not considered in the analysis of the heat exchanger's structural supports nor in their seismic qualification. When all of the appropriate loads are considered, parts of the supports for the heat exchangers and the two welds on adjacent pipe anchors were overstressed and did not meet design basis limits.

SAFETY EVALUATION SUMMARY:

The requalification of the Decay Heat Exchangers required a seismic reanalysis of the Decay Heat Exchangers and the attached piping. The approach used is more rigorous than the initial testing and is consistent with the Davis-Besse Design Criteria Manual and the USAR 3.7.3. This approach is much more exact than the approach currently described in USAR Section 3.9.2.11.3.

The effect of the additional support steel and seismic reanalysis on the Decay Heat Exchangers is addressed in calculation C-CSS-049.02-034. Per this calculation, the heat exchangers are seismically qualified for the revised loading due to the new support configuration and will be within design limits.

Additional supporting members are located such that there will be no interaction with any of the surrounding equipment and will not hinder access to perform operator actions during any postulated accidents.

Based on the above, the safety functions for the systems addressed in this safety evaluation are not adversely affected by this MOD and it is concluded that the modification is safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 97-0087 (SE 98-0025)

TITLE:

Protect EDG 1-1 Speed Switch for Appendix R Fire

CHANGE:

Move EDG 1-1 RPM tachometer SI6222 circuit from speed switch ST6221 to speed switch SS6221A, install isolation fuses in the SS6221A. This will also extend the circuit to control room cabinet C3617 where speed switch SS6221A is located.

REASON FOR CHANGE:

A fire in the cable spreading room or control room could cause a hot short on circuit 1CGD111A which connects EDG 1-1 tachometer SI6222, which could damage speed switch ST6221, making it unable to perform necessary EDG starting functions.

SAFETY EVALUATION SUMMARY:

The safety functions of ST6221 include the following: (1) At 40 rpm- turn on EDG Room Supply fans and energize dampers, control engine restart, permit static exciter functions, open jacket cooling water heat exchanger outlet valve, etc., (2) At 200 rpm- disengages air start motors, permit protective relays to function, shutdown EDG if this switch is not closed within 6 seconds, (3) At 400 rpm- close field flash circuit and inhibit Component Cooling water low flow alarm and (4) At 800 rpm- trips EDG output breaker if speed falls below this point. ST6221 also provides local indication of EDG speed.

After installation of the isolation fuses SS6221A will no longer have any function important to safety. After this modification SI6222 will provide indication of EDG speed, but have no function important to safety.

The control room tachometer provides a ready means to indicate whether the EDG 1-1 is operating properly and some procedures specifically rely upon SI6222 to determine the status of EDG 1-1 but there are other means in the control room to provide the same indication. The alternate means of determining whether the EDG is running include frequency indication, voltage indication, use of the Synchroscope, etc. Fuses are being added to isolate SS6221A to ensure it does not affect the Q power supply.

The other controls of the EDG are unaffected by this modification. Post-installation testing confirmed proper operation of the unaffected EDG circuits.

SAFETY EVALUATION SUMMARY
FOR
MOD 98-0013 and UCN 98-024 (SE 98-0020)

TITLE:

Repair of MS100 and MS101

CHANGE:

Install helical-coil thread inserts into the MSIV stud holes and update the USAR Table 3.9-2 to reflect the addition of the ASME code case reference.

REASON FOR CHANGE:

MOD 98-0013 installs these on an as needed basis.

SAFETY EVALUATION SUMMARY:

The helical-coil threaded inserts become a portion of the valve body pressure boundary.

The installation of helical-coil threaded inserts is being performed in accordance with the requirements defined in ASME Code Case N-496 Helical-Coil Threaded Inserts, Section XI, Division 1, which has been accepted without condition by the Nuclear Regulatory Commission in Regulatory Guide 1.147, Revision 11, October, 1994.

Installing the helical-coil threaded insert does not have a deleterious affect on the MSIV. Analysis demonstrates that the helical-coil thread shear is within stress allowables for the insert material, that there is negligible effect on minimum wall thickness and that the MSIV stud loading analysis is unaffected. Also the valve function is not adversely impacted by this modification.

Based on the above, the installation of helical-coil threaded inserts in the MSIV bonnet to body stud holes is safe.

SAFETY EVALUATION SUMMARY
FOR
MOD 98-0018 (SE 98-0024)

TITLE:

Increase the SFRCS Steam Generator High Level Trip Setpoint and Increase the ICS High Level Limiter Setpoint

CHANGE:

Raise the SFRCS high level trip setpoint for both OTSGs to 280 inches Start Up (SU) level. It also proposes raising the ICS high level limiter for both OTSGs, and associated alarms, to 95.5% Operating Range (OR) level.

REASON FOR CHANGES:

Due to flaking and spalling of contaminants adhering to the sides of the OTSGs, the level increased leaving a 10 inch margin to the high level trip setpoint at 91% power. It is desired to raise the setpoint so that it will be possible to operate the plant at full power.

SAFETY EVALUATION SUMMARY:

Technical Specification 3.4.5 allows operation of the OTSG's with up to 96% OR level, as long as at least 43 °F of superheat is maintained. The amount of superheat will be maintained above 43 °F since full feedwater pre-heating will still be occurring.

By raising the ICS level limiter setpoint, and associated alarms, to 95.5% on the OR, the steady state target of ICS will still be below the Technical Specification limit of 96% OR level. The 96% value was based on ensuring that the aspirator port does not become flooded during normal operation. The 96% value also ensures that the mass of water in the OTSG remains below the mass assumed in the Main Steam Line Break analysis (USAR Section 15.4.4). The ICS will maintain steady state level below the Tech Spec limits. Verification that OTSG level and superheat meets Technical Specification requirements is required on a per shift basis. Having alarms that notify the operators that operation near the Tech Spec limit is occurring ensures that operators are alert to plant conditions that are near the allowed limits. Therefore the plant response to accidents is unchanged.

The selected setpoints provide a separation between the normal steady state level and the SFRCS trip setpoint. This reduces the potential for spurious trips and inadvertent safety system actuations.

Based on the above, it is concluded that raising the SFRCS high level trip to 280 inches SU level and the ICS level limiter, and associated alarms, to 95.5% OR level will have no effect on safety.

SAFETY EVALUATION SUMMARY
FOR
SE 96-0040 R.01

TITLE:

Cycle 11 Reload Report and Core Operating Limits Report

CHANGE:

This revision reflects modifications to the Flux- Δ Flux/Flow trip setpoints in the Reload Report and the Protective Limits figure in the Core Operating Limits Report (COLR). With the implementation of 24-month cycle based Allowable Values Technical Specification (TS) trip setpoints, the COLR Allowable Values Flux- Δ Flux/Flow trip setpoints will exclude drift and calibration error, in accordance with ISA standard RP67.04, Parts I and II. Those efforts will now be included in the Davis-Besse field setpoint calculations.

REASON FOR CHANGE:

NRC approval of the TS Allowable Values for the Reactor Protection System (RPS) trip setpoints for a 24 month operating cycle (Operating License Amendment No. 218).

SAFETY EVALUATION SUMMARY:

The thermal-hydraulics design evaluation in the Reload Report was not affected by the increased flow string errors applicable for a two year cycle's operation. That is, the previous Cycle 11 analyses used flux-flow limits and allowable power levels that bound those resulting from the revised flow string errors analyses.

The Maneuvering Analyses, with Cycle 11's operation beyond the 18 month surveillance interval, also addressed the impact of the difference in the flow string errors on the Limiting Conditions for Operation (LCO) excure axial power imbalance set points. The analyses indicated that the present Cycle 11 excure imbalance set points are still applicable. Similarly, the Flux- Δ Flux/Flow trip setpoints, originally developed for Cycle 9, have been evaluated as conservative and will be maintained in the RPS for the remainder of Cycle 11.

Based on the evaluation of the effects on safety, the proposed action (i.e. implementation of the COLR, by maintaining the previous LCOs and including the revised Protective Limits) and operation of Cycle 11 to 675 EFPDs has been determined safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
SE 96-0089

TITLE:

Reposition of the Reactor Coolant Drain Tank Nitrogen Regulator Isolation Valves

CHANGE:

The Reactor Coolant Drain Tank (RCDT) Nitrogen Regulator (NN 1776) will be placed in continuous operation by repositioning the regulator isolation valves (NN 95 and NN 96) to the open position.

REASON FOR CHANGE:

This change is a result of replacing the nitrogen regulator with a new style capable of supporting the low pressure and low flow conditions inherent to the RCDT nitrogen supply.

SAFETY EVALUATION SUMMARY:

Restoring the RCDT nitrogen regulator to continuous service will ensure a steady supply of nitrogen is available whenever the RCDT pumps down, therefore precluding the need for an operator and possible damage to the RCDT rupture discs. Additionally, due to the replacement of the regulator with a different non-vented design, the possibility of a gaseous release from the RCDT is greatly reduced.

The proposed action has no effect on safety, and will increase the reliability of the Reactor Coolant Drain Tank due to the regulator being continuously in service, thus minimizing the possibility of rupture disc failures and the associated venting of radioactive gases to the Auxiliary building. Returning to continuous operation of the nitrogen regulator is bounded by the original design analysis since it will return the system lineup to that present during the initial plant design. Based on the above, these actions are considered safe.

SAFETY EVALUATION SUMMARY
FOR
SE 97-0033 R. 01

TITLE:

Fuel Assembly Grid Repositioning

CHANGE:

Return spacer grids of fuel assemblies NJ05TM, NJ05V6, NJ05UX, NJ05VN, and NJ05UZ to their design configuration.

REASON FOR CHANGE:

The above mentioned fuel assemblies were observed to have spacer grids which were not at the correct axial position. This inspection was conducted during Refueling Outage 10. The spacer grids were repositioned during Cycle 11.

SAFETY EVALUATION SUMMARY:

The repaired fuel assemblies will not be returned to the core because the assemblies have reached burn-up values which prevent their insertion in the core for operation. Since the fuel assemblies being repaired will not be placed in the core, the fuel assembly functions important for safe plant operation are not affected. However, the repositioning of the spacer grids on an irradiated fuel assembly is a fuel handling process that could potentially result in a fuel handling accident.

The loads applied to the fuel assembly and spacer grid during the repositioning process has been evaluated and does not: 1) affect the spacer grid's integrity ; 2) will not cause fuel rod or spacer grid damage due to grid cocking during the repositioning process; and 3) will not cause fuel rod damage due to lifting a rod against the upper end fitting.

Assuming multiple equipment and administrative failures, the worst case scenario would be the loss of integrity of 28 fuel rods. This worst case scenario is bounded by the fuel handling accident in USAR Section 15.4.7 which assumes the failure of 56 fuel rods having a fission product decay time of 72 hours. The fission products in the fuel assemblies being repaired have been decaying for over a year. This significantly reduces the radiological consequence associated with the loss of fuel rod integrity. With the reduced radiological release and the low number of potentially failed fuel rods, the Emergency Ventilation System will be capable of performing its safety function.

SAFETY EVALUATION SUMMARY
FOR
SE 97-0040

TITLE:

Potential Radiological Consequences Associated with Floor Drain Extension Removal in the Emergency Diesel Generator Rooms

CHANGE:

Removal of floor drain extensions in the Emergency Diesel Generator rooms

REASON FOR CHANGE:

Component Cooling Water (CCW) contaminated with low levels of radioactivity could be released in the Emergency Diesel Generator (EDG) rooms. The water from these drains is routed to the storm sewer system and eventually released from the site to the Toussaint River.

SAFETY EVALUATION SUMMARY:

The only CCW components whose leakage would be collected outside the Radiologically Restricted Area (RRA) are the emergency diesel generator's water cooling jackets. The floor drains in the diesel room drain to the storm sewer. These drains are labeled to inform personnel to contact Radiation Protection prior to draining any components containing potentially radioactive material to the storm sewer system. The low levels of radioactivity present in the CCW do not adversely impact the function of any safety related components.

Any releases to the Toussaint River are administratively controlled. Should an inadvertent release of CCW to the Training Center Pond occur, a monitoring, sampling, and processing program can be established.

The radiological consequences associated with the removal of the floor drain extensions in the EDG rooms are bound by previous analysis. There are no additional safety or radiological control issues associated with the floor drain extension removal. Based on the above analysis, the action of removing the EDG floor drain extensions is deemed safe.

SAFETY EVALUATION SUMMARY
FOR
SE 97-0042

TITLE:

Increasing the Utilization of the Low Level Radwaste Storage Facility

CHANGE:

Permit the following additional operations in the Low Level Radwaste Storage Facility (LLRWSF): Opening, for inventory, sorting and/or repackaging, of Dry Active Waste (DAW); using the truck bay for loading a SeaLand container with DAW and preparing the container for off-site shipment; opening, for retrieval of tools and equipment, of Radioactive Material (RAM) containers stored in the LLRWSF area; refurbishing and/or minor repair of tools and equipment in cell area; and establishing such temporary radiation protection areas as are necessary to accomplish these additional operations.

REASON FOR CHANGE:

Due to space allocation and flow of traffic in the plant, it is desirable to add the above mentioned operations in the LLRWSF.

SAFETY EVALUATION SUMMARY:

The proposed activities do not pose any radiological risks requiring any measures beyond routine RP activities and will have no significant adverse effects on the LLRWSF nor do they create an additional radiation source that could adversely affect outside areas either on or off-site due to the shielding within the LLRWSF.

The proposed activities make no changes to the fire protection, drainage, or shielding systems of the LLRWSF.

The results of the above scenarios on control room habitability result in a maximum activity concentration of $7.6E-8 \mu\text{Ci}/\text{cm}^3$ in the control room during the 2 hour release. This concentration, if breathed for the entire two hour period would result in a dose equivalent of less than one percent of the annual occupational limit.

SAFETY EVALUATION SUMMARY
FOR
SE 98-0017

TITLE:

Temporary RCS Letdown Boron Analyzer Indication

CHANGE:

Provide continuous indication of boron concentration by installing instrumentation to permit monitoring boron concentration up to 2800ppm.

REASON FOR CHANGE:

The anticipated boron concentration during 11 RFO refueling activities, startup and initial Cycle 12 power operation will exceed the existing 0-2050 ppm indication range.

SAFETY EVALUATION SUMMARY:

The RCS Letdown Boron Analyzer AE1999 does not have any important to safety function. The Boron Analyzer's function is to provide continuous indication of reactor coolant boron concentration in the control Room (strip chart), Cabinet Room (analog indicator) and also provides an input to the plant computer. The computer value for boron is utilized in the Reactor Power Distribution Calculations.

The anticipated boron concentration during 11 RFO refueling activities, startup and initial power operation will exceed the existing 0 - 2050 ppm indication range. In order to provide continuous indication of boron concentration during this period, temporary instrumentation will be connected to the existing Boron Analyzer circuitry which will permit monitoring the RCS boron concentration over the required range of boron concentration (approximately 0 - 2800 ppm). The temporary instrumentation is intended to be installed prior to the RCS boron concentration exceeding 2050 ppm during the shutdown for 11 RFO. The temporary instrumentation is intended to be removed after Cycle 12 RCS boron levels have decreased below 2050 ppm, which is expected to be less than one week of power operations.

The proposed temporary instrumentation and specified compensatory sampling will provide an adequate means to monitor RCS boron concentration trends during the time that the RCS concentrations exceed the range of the current Boron Analyzer indicators. Any differences resulting in the accuracy of the temporary instrumentation in comparison to the installed indicator should not prevent the indication from being useful in monitoring boron concentration trends.

Based on the above evaluation, the proposed activity is safe.

SAFETY EVALUATION SUMMARY
FOR
SE 98-0021 R. 01

TITLE:

The Cycle 12 Reload and Core Operating Limits Reports

CHANGE:

The Cycle 12 core loading, as described in the Reload Report, consists of the following:

- 76 FCF Mark-B10M Fuel Assemblies (FAs) with a 4.47 weight percent (w/o) U235 (sub-batches 14A through D). All FAs contain uranium-gadolinia rods, that is depending on the sub-batch: 4, 8, or 12 UO₂ fuel rods with enrichments of 3.80 and 3.13 w/o U235 contain a mixed absorber, Gd₂O₃, with concentrations of 3 and 6 w/o, respectively. FAs also contain axial blankets consisting of 2.50 w/o U235 axially zoned fuel rods. A total of 32 Batch 14 FAs will have Burnable Poison Rod Assemblies (BPRAs) with B₄C concentrations ranging from 1.4 to 2.6 w/o.
- One center assembly, sub-batch 9F re-insert with an initial enrichment of 3.38 w/o U235 and a burnup of 34,540 MWD/MTU.
- The remainder 100 FAs are Mark B10A and B10AZL assemblies (Batches 12A and C, 13A and 13B). Re-inserts are four Batch 12A FAs from Cycle 10, and one Batch 12C FA as part of the core's re-design caused by results of a fuel spacer grid inspection campaign.
- 33 new Extended Life Control Rod Assemblies (ELCRAs) have been installed bringing the total to 53.

REASON FOR CHANGE:

This is to assure that operation of the core with the configuration as described in the Reload Report will not violate operating and safety limits.

SAFETY EVALUATION SUMMARY:

The reference fuel cycle for Cycle 12 is Cycle 11. The nuclear and thermal-hydraulics analyses were based on a duration of Cycle 11 to 637 ± 15 EFPDs, Axial Power Shaping Rods (APSRs) full withdrawal at 620 ± 10 EFPDs, Control Rod Group (CRG) 7 withdrawn to 97% and an End-Of-Cycle (EOC) reactor coolant average temperature reduction of 7°F followed by a planned power coastdown. Cycle 12 was analyzed to 684 EFPDs, and the Cycle 12 Operating limits and Setpoints reflect that licensed design length.

Four Batch 13B FAs had to be substituted with four Batch 12A FAs that were discharged during 10RFO. One is a recaged FA with a stainless steel rod. Also FA NJ08GN had minor damage to its no. 3 grid corner cell but was evaluated acceptable for use. FA NJ07LX, a Batch 12C with 43 GWD/MTU, had been originally a Cycle 12 re-insert, but had two broken corner grid welds and was replaced with another batch 12C discharged FA, NJ07KH, with a slightly higher exposure.

Operation with FA NJO8EX (the most bounding FA with grid damage) and three other FAs with grid damage has been evaluated as having no impact on mechanical safety or result in any operational concern.

The following are analyzed and found to be acceptable for Cycle 12 operation: Core design and loading, reactivity controls, fuel system design, nuclear design/core physics, thermal-hydraulic design, accident and transients analyses, operating limits and allowable values, startup program physics testing, and chemistry issues.

Based on the above evaluations of the effects on safety, the proposed action and operation of Cycle 12 has been determined safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
SE 98-0032

TITLE:

Temporary Isolation of MT 19, MT 16A and MT 16B Air Receiver Moisture Traps

CHANGE:

MT 19, EIAC Receiver Moisture Trap has been tagged out of service. MT 16A and MT 16B, Station Air Moisture Receiver Traps have been removed from service.

REASON FOR CHANGE:

MT 19 is leaking, causing loss of air from the Station and Instrument Air System and has been tagged out of service for repair and replacement. MT 16A and MT 16B have been experiencing air blowing by MT19 and have been isolated to prevent a potential loss of air.

SAFETY EVALUATION SUMMARY:

No adverse effects on the Station and Instrument Air system, or the components served by this system, are created by operating with the moisture traps isolated. In accordance with USAR Section 9.3.1.5 the system is being regularly blown down. The frequency of receiver blow downs has been increased by the operating staff. Operation of the Instrument Air Dryers will prevent any potential buildup of moisture in the air lines.

This change in operation is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
TM 96-0024 (SE 96-0063)

TITLE:

Temporary Modification to Monitor Reactor Coolant Pump (RCP) 2-2 Upper Thrust Bearing

CHANGE:

TM 96-0024 makes wiring changes in Uniform Temperature Reference box in containment.

REASON FOR CHANGE:

TM 96-0024 allows continuous monitoring RCP 2-2 upper thrust bearing which is currently experiencing increasing temperatures.

SAFETY EVALUATION SUMMARY:

RCP 2-2 is currently experiencing an increasing upper thrust bearing temperature. The RCP motors need to be shutdown when bearing temperature reaches 190°F to prevent motor damage. Providing the ability to continuously monitor the spare upper thrust bearing temperature may help determine the cause of the abnormal temperature increase. Additionally, this action will help ensure bearing temperature limits are not exceeded.

RCP 2-2 spare upper thrust bearing temperature will be monitored at the plant computer instead of the motor air inlet temperature. The motor air inlet temperature is used for information purposes only and the stator winding temperature is still available at the plant computer as an indication of motor temperature. The designation for the modified computer point will be changed to reflect the new input.

Disabling the motor air inlet thermocouple and monitoring the spare upper thrust bearing thermocouple under TM 96-0024, together with the required changes to elementary drawings, is considered safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
TM 97-0007 (SE 97-0060)

TITLE:

Install Sight Glass on the Turbine Steam Packing Exhauster and Remove Control Room Annunciator (15-3-A) from Service.

CHANGE:

This Temporary Modification installs a sight glass on the Turbine Steam Packing Exhauster (SPE) and removes Control Room Annunciator (15-3-A), Turbine Steam Packing Exhauster Level High from service.

REASON FOR CHANGE:

This Temporary Modification will allow SPE vacuum adjustments, eliminating excessive gland steam leakage at the Main Feedpump Turbines, while being able to monitor SPE level above the setpoint.

SAFETY EVALUATION SUMMARY:

Functions affected by the proposed changes are removal of SPE high level switch, LSH1987, from service by closing GS1987A and GS1987B. This will prevent Control Room Annunciator (15-3-A) from alarming. Additionally, a sightglass will be installed between GS6 and GS1988 which are normally closed valves.

The purpose of the annunciator is to warn Operations personnel of a SPE tube leak. If a tube leak were to occur, alternate methods of level indication are available to alarm personnel of the SPE drainage problem. The sight glass will allow Operations personnel to continue monitoring for tube leakage.

Overall, there will be an increase in Main Feedpump reliability due to the elimination of gland steam leakage even though the potential for minimal overflow exists. Since the Main Turbine Gland Steam Seals and Drains do not perform a safety function, the normal operation of the system is not affected and the possible SPE overflow conditions do not affect operation, there is no effect on safety.

SAFETY EVALUATION SUMMARY
FOR
TM 98-0001 (SE 98-0008)

TITLE:

Temporary Demineralizer Connection to the Boric Acid Addition Tanks

CHANGE:

Install a demineralizer containing cation resin on one of the boric acid pump lines.

REASON FOR CHANGE:

To remove the higher than desired lithium concentration from the contents of the BAAT during recirculation.

SAFETY EVALUATION SUMMARY:

The Boric Acid Addition System (BAAS) provides no Nuclear Safety Related function. The system is important to safety because it injects boric acid to control reactivity. The BAAS injects boric acid into the reactor coolant system to control reactivity. The BAAS also injects boric acid into the Borated Water Storage Tank System and the Spent Fuel Pool System to control their boron levels and shutdown reactivity margins. The BAAS serves as one source of boric acid for RCS injection, while the BWST provides another.

The proposed USAR change has no adverse effect on safety. The proposed change does not increase the adverse effects from any hazard. The chemical addition system is not credited for mitigation of any USAR Chapter 15 accidents.

The boric acid addition system is credited for providing safe shutdown capability in the event the Borated Water Storage Tank is lost by a tornado missile. Only one BAAT will be recirculated and demineralized at a time, the contents of the other BAAT not being recirculated will still be available. Boric acid addition tank high and low level annunciators and local level indicators are provided in the control room in the event of the loss of BAAT contents. These indications and alarms are unaffected by the use of a temporary demineralizer. Coincidental loss of the contents of a BAAT and occurrence of a tornado which renders the BWST inoperable is an improbable event. The conditional core damage probability for such events is below the applicable screening criteria for the Individual Plant Evaluation of External Events (IPEEE).

The demineralizer and hoses are designed to meet the maximum service conditions of piping class HCC-97 which is the discharge of the boric acid addition pumps.

TM 98-0001 is considered safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
TM 98-0016 (SE 98-0019)

TITLE:

Removal of Polar Crane Pendant and the Bypass of its Weight System Interlocks

CHANGE:

Removal of the pendant containing the digital weight system, lifting of the conductors for the pendant cable, and the use of jumpers to bypass the weight system interlocks.

REASON FOR CHANGE:

The Polar Crane Pendant is unusable, including the digital weight system, and can no longer perform its function.

SAFETY EVALUATION SUMMARY:

This change will not affect in any way the functions important to safety of the Polar Crane or any other system, structure, or component.

Removal of the pendant, the lifting of the conductors for the pendant cable, and the use of jumpers to bypass the weight system interlocks will not affect the structural capability of the Polar Crane or the movement of objects along specified load paths.

The bypass of the weight system interlocks will not negatively impact the polar crane because a "banana scale" indicator, the backup means of load indication, which is calibrated within an accuracy of 1/4 % which is more accurate than the required 1% accuracy of the digital weight system will be used.

The crane operator will use the "banana scale" with established upper and lower limits in the Reactor Vessel Closure Head removal and reinstallation procedure to ensure the maximum analyzed lift (330,000 lbs.) for the polar crane, established by Toledo Edison's response to NUREG 0612 and accepted by the NRC, is not exceeded.

A review of ANSI B30.2 -1983, "Overhead and Gantry Cranes", determined that none of the changes will be in violation of this standard. This standard did not discuss weight systems at all and did not require that a pendant exist.

Based on the above, the proposed change is safe.

SAFETY EVALUATION SUMMARY
FOR
TM 98-0024 and UCN 98-036 (SE 98-0028)

TITLE:

Miscellaneous Waste Monitor Tank Filter Micron Rating

CHANGE:

Revise the filter micron rating in USAR Table 11.2-2 to less than or equal to 10 micron.

REASON FOR CHANGE:

The monitor tank filter is functioning as a post filter for demineralizer skid to remove resin fines that may be present in the flow stream.

SAFETY EVALUATION SUMMARY:

The Miscellaneous Waste Liquid Radwaste System (MWLRS) does not have a safety function and is not required to operate following an accident to mitigate the consequences. The design objective of the system is to reduce the radioactivity concentrations in the effluent at the site boundary to less than the 10 CFR 20 requirements and to keep the releases as low as reasonably achievable to meet the numerical design objectives given in 10 CFR 50, Appendix I.

The miscellaneous waste monitor tank filter function is to remove corrosion products in the processed fluid.

DBNPS liquid radwaste system filters were initially rated at 1 micron because there was not enough data available on the efficiency of filters in removing crud from nuclear power plant waste. Operating experience indicates that increasing filter rating to 10 microns in the other liquid radwaste systems did not significantly impact the liquid radwaste system functions. In the USAR, a decontamination factor of 10 is used for the filters in estimating the releases from the plant. Also no credit is given for higher DF or multiple passes through a filter.

Prior to discharge from the monitor tank, the contents are sampled for radioactivity levels. Further, waste being discharged to the environment is normally monitored by a radiation detector, which will isolate the flow path in case of high radiation levels.

Therefore, increasing the filter size to 10 microns does not impact MWLRS and is safe.

SAFETY EVALUATION SUMMARY
FOR
TM 98-0025 (SE 98-0029)

TITLE:

Gag Main Steam Safety Valve SP17B6

CHANGE:

Temporary Modification (TM) 98-025 installs a gag on Main Steam Safety Valve (MSSV) SP17B6.

REASON FOR CHANGE:

MSSV SP17B6 developed a seat leak. The valve is considered inoperable and may lift at a setpoint below 1050 psig.

SAFETY EVALUATION SUMMARY:

Installation of a gag on a MSSV reduces the total relieving capacity of the MSSV's. Technical Specification 3.7.1.1 permits continued operation with inoperable MSSV's provided that the high flux trip setpoint is reduced. The reduction in high flux trip setpoint, which is a function of the operable relieving capacity, restricts the maximum thermal power at the initiation of a turbine trip. The total design relieving capacity of the MSSV's exceeds the minimum required relieving capacity. Therefore, the TM does not adversely affect safety.

The liquid relief capacity of the MSSV's is credited with preventing over-pressurization of the steam generators in the event of a failed open feedwater control valve. In USAR Section 15.2.10.2.4 it is concluded that four MSSV's per steam generator are sufficient to handle the transient feedwater flow. Gagging of one MSSV will not affect the consequences of this accident because eight MSSV's remain available.

Technical Specification 3.7.1.1 requires that there be at least one operable MSSV per steam generator set at 1050 psig. As stated above, gagging of one MSSV will not affect the consequences of this accident because eight MSSV's remain available.

With any one 1050 psig relief out of service, the estimated initial power level from which Davis-Besse can survive a runback is calculated to be 51.85%. The current ARTS arming setpoint of 45% remains justified with a 1050 psig MSSV gagged. Additional margin actually remains because steam loads (Main Feed Pump Turbines, Gland Steam, Auxiliary Steam) were not considered.

TM 98-0025 is considered safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 96-054 (SE 96-0080)

TITLE:

Grid Description and Stability Analyses

CHANGE:

UCN 96-054 updates USAR Sections 8.1 and 8.2 pertaining to grid descriptions and stability analyses.

REASON FOR CHANGE:

A review of Davis-Besse's USAR indicated that the passage of time has eroded the accuracy of some of the information contained in USAR Sections 8.1 and 8.2.

SAFETY EVALUATION SUMMARY:

Information Notice (IN) 91-29 Supp 3 summarized deficiencies the staff found during Electrical Distribution System Functional Inspections (EDSFIs). One of the findings included the evaluation of the availability and reliability of offsite power sources. UCN 96-054 is related to this issue.

Studies conducted in 1987 evaluated an actual incident that occurred in which two 345 kV circuits near Fostoria substation were faulted. The first study proved that the Fostoria incident caused no stability problems. The second study evaluated the worst case that was expected at Davis-Besse. It was a close-in three phase 345 kV fault in which a breaker at Davis-Besse fails. This second study showed that the unit and area remain stable for this worst-case scenario. One item listed in IN 91-29-03 event is loss of the largest load. This event does not appear to be covered by the studies, so it was questioned as to how the unit would be affected by changes in loads, such as a new 345 kV substation. This new load is considerably smaller than the existing sources which the DBNPS could potentially lose, and therefore does not have as great a potential for disturbance.

On June 23, 1996, the entire Bay Shore 345 kV switchyard shut down, including the line to Davis-Besse. Although no in-depth study is available for this event, the Davis-Besse switchyard responded in a stable manner. Therefore, this event serves to validate previous conclusions of stability for line losses at Davis-Besse.

This change is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 96-058 (SE 98-0006)

TITLE:

Control Room Habitability Assessment Posed by Toxic and Explosive Hazards

CHANGE:

This UCN will revise the appropriate sections of the USAR and add a section titled Habitability Systems, to evaluate and incorporate chemicals known to potentially affect Control Room Habitability.

REASON FOR CHANGE:

This UCN documents the assessment of the current potential hazards posed to Control Room habitability by off site and on site toxic or explosive hazards.

SAFETY EVALUATION SUMMARY:

In response to NUREG-0737, Bechtel Corporation prepared a Control Room Habitability Study for the DBNPS to evaluate explosive and potential toxic material habitability effects to the Control Room from both on site and off site hazards. Analysis of the worst case scenarios of these onsite chemicals and off site chemicals determined respective explosive or toxicity limits would not be exceeded and not pose a hazard for Control Room habitability.

Over the years, other chemicals have been introduced for use at the station. For these chemicals, Engineering performed calculations to evaluate release effects on Control Room personnel. In each case, the chemicals have been shown to have no effect on habitability based on physical properties or when toxic concentrations in the vicinity of the Control Room are calculated for the worst case spill scenario, the concentrations do not exceed the OSHA permissible exposure limit (PEL).

Studies were also conducted to update the hazardous chemicals stored or transported off site. The new calculation assessed 16 types of chemicals that could affect Control Room habitability. In each case, either the chemicals have been shown to have no effect on habitability based on physical properties, the frequency of transportation, or, when toxic concentrations in the vicinity of the control room are calculated, they are below the two minute concentration limit as referenced in Regulatory Guide 1.78.

Explosive or toxic hazardous materials stored on site or off site within the proximity of the Davis-Besse Station, or transported within the vicinity of the Davis-Besse Station, do not pose a hazard to plant safety due to Control Room Habitability concerns.

SAFETY EVALUATION REPORT
FOR
UCN 96-126 (SE 96-0100)

TITLE:

Facility Staff Overtime

CHANGE:

The Technical Requirements Manual is being revised to allow 12 hour work shifts at the Davis-Besse Nuclear Power Station.

REASON FOR CHANGE:

The proposed change clarifies the current work hours of the operations facility staff.

SAFETY EVALUATION SUMMARY:

Facility Staff Overtime and working hours is described in DBNPS USAR Section 13.0, "Conduct of Operations" and controlled by administrative procedures to limit overtime hours for plant staff who perform safety-related functions. These limits are established to assure personnel are not assigned to shift duties while in a fatigued condition that could jeopardize safe plant operation.

Establishing operating personnel work hours at "up to 12-hour shifts under a rotating work week schedule, with a nominal 40-hour work-week," provides increased flexibility in scheduling personnel and does not adversely affect their performance. This change also decreases the risk of miscommunication between shifts by reducing the number of turnovers per day and increases operations and maintenance efficiency by promoting continuity in ongoing plant activities. The probability for operating personnel error due to (1) incomplete or insufficient turnover or (2) interruption of in-plant maintenance and testing is reduced.

The USAR also states that these limits are in accordance with NUREG-0737, "Clarification of TMI Action Plan Requirements," and NRC Generic Letter 82-12, "Nuclear Power Plant Staff Working Hours." The revision to the DBNPS Technical Requirements Manual (TRM) 5.3.2 retains control of overtime by plant facility staff and generally follows the guidance of NUREG-0737 and Generic Letter 82-12 which is consistent with the Improved Standard Technical Specifications.

Revising TRM 5.3.2 to change the normal 8-hour shifts to allow for up to 12-hour work shifts will not jeopardize safe plant operation or impact the ability of plant staff to perform safety-related functions. Changing the normal shift hours remains within TS requirements and is an accepted industry practice.

Based upon the above, the proposed change is considered to be safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 96-181 (SE 96-0093)

TITLE:

Inoperable Fire Barriers or Detection in the Containment Annulus

CHANGE:

The FHAR is being revised to provide additional compensatory measures for inoperable fire barriers and fire detection in the Containment Annulus.

REASON FOR CHANGE:

This change is a result of Potential Condition Adverse to Quality Report 96-0512 regarding the use of Thermo-Lag as a radiant energy shield in containment and the annulus.

SAFETY EVALUATION SUMMARY:

The DBNPS FHAR requires compensatory actions be taken when one or more of the fire barrier/radiant energy shields are inoperable in the Containment annulus or when less than the minimum required fire detection instrumentation is inoperable in the annulus.

This change institutes new compensatory measures such as eight hour watches, video camera inspections, temperature monitoring or other means determined to be adequate by the Fire Protection Engineer.

The proposed change does not affect safety because the additional measures strengthen and improve the compensatory measures for inoperable fire barriers and detectors in the normally inaccessible (at power) containment annulus. The radiation levels, location and nature of the problem are all considered when determining the compensatory measure.

The ability to change the Fire Protection Program is permitted by License Condition 2.C.(4) as long as those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The proposed changes are safe and do not change the ability to safely shutdown in the event of a fire.

SAFETY EVALUATION SUMMARY
OF
UCN 96-186 (SE 98-0007)

TITLE:

Cooldown Following a Seismic Event/Steam Generators Fed with Service Water, Solids Accumulation in the Steam Generators

CHANGE:

Accurately show the effects of plant cooldown following a seismic event using water entrapped in the Seismic Class I area of the forebay (service water via auxiliary feed) and without using the condensate storage tanks or any other Seismic Class II water system.

REASON FOR CHANGE:

Correct a discrepancy in the USAR concerning the quantity of dissolved and suspended solids that would be accumulated in the steam generators (SG) after injecting raw lake water for 24 hours.

SAFETY EVALUATION SUMMARY:

The SSCs affected are the SGs. The quantity of dissolved and suspended solids accumulated in the SGs during cooldown after a seismic event using water entrapped in the Seismic Class I area of the forebay will be greater than is currently listed in the USAR. The service water and auxiliary feedwater systems are designed to allow utilization of raw water to feed the steam generators if necessary.

After an earthquake greater than the Operating Basis Earthquake, Technical Specification 4.4.5.3.c requires that an inservice inspection be performed on the SGs. Degradation of the SG tubing from the initial event or from subsequent off normal operation will be evaluated at this time.

The conditions in the SGs created by raw water injection will not degrade the primary to secondary pressure boundary sufficiently to cause a loss of the boundary. The maximum chemical species loading evaluated, results in approximately 240 pounds of solids accumulated in each SG in the first 24 hours after shutdown. Less than 15 percent (36 pounds) of the solids would be expected to precipitate based on the chemistry of the concentrated solution. The balance of the material would remain in solution. The quantity of the material accumulated in the SGs will not significantly decrease the heat transfer capabilities during an emergency cooldown.

Neither the flow path for auxiliary feedwater nor the flow path for steam flow from the SGs to the auxiliary feedwater pump turbines will be affected. This change is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 96-191 (SE 96-0095)

TITLE:

Makeup Tank Low Pressure Alarm Setpoint

CHANGE:

The Makeup (MU) Tank low pressure alarm setpoint as described in USAR Section 9.3.4.5.1.a is being changed from "approximately 18 psig" to "approximately 17 psig." Correspondingly, the MU Tank low pressure alarm setpoint as referenced on Operational Schematic OS-002 Sheet 3 is being changed from 18 psig to 17 psig.

REASON FOR CHANGE:

This change reflects the actual setpoint, currently 17 ± 1 psig.

SAFETY EVALUATION SUMMARY:

Changing the USAR described value to "approximately 17 psig" has an insignificant impact on the USAR described Makeup and Purification (MU&P) System operation. The actual field setpoint remains unchanged (17 ± 1 psig). Furthermore, the MU tank low pressure alarm will continue to be received prior to reaching the 15 psig low pressure procedural limit which ensures adequate Makeup Pump NPSH during normal plant operations.

The control room operator's response to the low pressure alarm remains unaffected. Also, there are no safety analysis assumptions which specify operator response times related to receiving the low pressure alarm.

The USAR 15.2.4 moderator dilution accident assumptions and analysis are not impacted by these changes. The previously assumed dilution flowrates of up to 500 gpm are unaffected by these changes. USAR 15.3 assumptions are not impacted by these changes, as the MU&P system or HPI may still respond to a small break and match the leakrate, thus ensuring an orderly shutdown. The USAR 15.4.5.2.2 break in instrument lines, methods of analysis, are not impacted by these changes. The MU&P system operational assumptions are still valid and unaffected by these changes.

SAFETY EVALUATION SUMMARY
FOR
UCN 96-196 (SE 96-0098)

TITLE:

USAR and FHAR Changes To Resolve NRC Information Notice 92-018 Hot Short Issues

CHANGE:

This UCN revises the FHAR to show that there could be operator damage to valves DH2733 and DH2734 due to a hot short and the valve may need to be mechanically closed, and to show that there may be damage to valve SW1382 due to a fire and an alternate source of water may be necessary for auxiliary feedwater during a Control Room or Cable Spreading Room fire.

REASON FOR CHANGE:

The changes are due to actions taken to resolve Nuclear Regulatory Commission (NRC) Information Notice (IN) 92-18, Potential for Loss of Remote Shutdown Capability During a Control Room Fire . These issues were reported to the NRC via Licensee Event Report (LER) 96-002, "Potential Loss of Remote Shutdown Capability due to MOV Fire Induced Damage".

SAFETY EVALUATION SUMMARY:

Information Notice 92-018 discussed a potential concern for loss of shutdown capability following an Appendix R fire due to fire induced hot shorts in the control circuitry for motor operated valves (MOV). The shorts could expose the valve to high torque load conditions causing damage to the valve or operator thus resulting the inability to reposition the valve.

Valves DH2733 and DH2734 are potentially affected. These valves need to be closed prior to going onto decay heat removal (DHR) during the latter stages of plant cooldown. Initiation of DHR cooling does not occur until greater than 24 hours after the fire. Appendix R allows for repair activities to be performed on components required to achieve and maintain cold shutdown. An evaluation has concluded that damage to DH2733 and DH2734 is limited to the operator in the event of a hot short condition. The FHAR, Serious Station Fire and Serious Control Room Fire procedure is revised to indicate that the operator on these valves may need to be removed to allow valve closure following a fire in the fire areas of concern prior to the initiation of DHR cooling.

SW1382 is also potentially affected. This valve is opened to provide a backup Seismic Category I source of water (Service Water) to the AFW Pumps in the event that the normal suction source (Condensate Storage Tanks (CSTs)) is depleted during plant cooldown. For fires in the Control Room or in the Cable Spreading Room, an evaluation

has shown that a hot short could cause SW1382 to go closed and not be able to be reopened. In an Appendix R fire scenario, it will be more than a day before the CSTs are depleted and an alternate water source will be needed. The FHAR and the Serious Station Fire procedure have been revised to indicate that if this hot short has occurred, alternate actions will be needed. This is adequate time for plant personnel to carry out these actions.

The hot short conditions described above, is limited to the valve and/or operator internal components and does not affect the structural integrity (i.e., pressure boundary) of the valve.

NRC Generic Letter 86-010, "Implementation of Fire Protection Requirements" allows repairs to be made to achieve cold shutdown. The fire protection system is also available as a backup source of water to the AFW System.

These changes are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 96-197 (SE 97-0001)

TITLE:

Manual Handling of Rod Assemblies

CHANGE:

This change will revise the USAR to describe manual handling of rod assemblies in the auxiliary building.

REASON FOR CHANGE:

The USAR is being revised to add the appropriate level of detail when discussing fuel handling equipment.

SAFETY EVALUATION SUMMARY:

There is no effect on safety with respect to the ability of a rod assembly to perform its functions after manual handling within the auxiliary building fuel handling area. There is no reasonable way for a host fuel assembly to be negatively affected by the withdrawal or insertion of a rod assembly.

There is no danger to the structural integrity of the fuel based on the weight of the tool. The risk of a breach of the ¼ inch thick stainless steel liner, even if these loads were dropped, is minimal. A breach in the liner in these areas would not prevent the liner from performing its safety function since such a breach would be small and well within the capacity of the SFP water makeup system. The area could still be isolated by a gate to stop the loss of water from the fuel storage area.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-008 (SE 97-0005)

TITLE:

Implementation of ASME Section XI IWE Requirements

CHANGE:

The proposed change updates the Inservice Inspection requirements to include the rules of IWE for the inspection of the Containment Vessel and its penetrations.

REASON FOR CHANGE:

This change is a result of new inspection rules for the Containment Vessel which were not previously in effect. These new rules were promulgated through an amendment to 10 CFR 50.55a which was published in the August 8, 1996 Federal Register.

SAFETY EVALUATION SUMMARY:

The proposed changes to USAR Sections 5.2.5, 3.8.2.1.9, and Table 17.2-1 have no adverse affect on safety. The change to each section involves the imposition of ASME Section XI, Subsection IWE, for the inspection of the Containment Vessel and its penetrations. ASME Subsection IWE requires the following examinations: Visual Examination of Containment Surfaces; Visual or Volumetric Examination of Containment Surfaces requiring Augmented Examination; Visual Examination of Seals, Gaskets, and Moisture Barriers; Visual Examination and Torque Testing of Pressure Retaining Bolting; Pressure Testing of Pressure Retaining Components.

Each of these inspections is intended to ensure the continued ability of the Containment System to perform its intended safety function. The proposed change is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-011 (SE 97-0066)

TITLE:

Revise Technical Requirements Manual (TRM) Reactor Coolant System Surveillance Frequency

CHANGE:

The surveillance frequency for TRM Reactor Coolant System (RCS) Surveillance Requirements 4.4.11.1, 4.4.11.2 and 4.4.11.3, is extended from 18 months to "At least once each Refueling Interval." A paragraph is also added to Bases 4.0.2 to note that the allowable tolerance for performing surveillance activities also provides flexibility to accommodate the length of a fuel cycle. The Definition section is also being revised to add a definition for Refueling Interval.

REASON FOR CHANGE:

These changes are made to accommodate a 24 month operating cycle.

SAFETY EVALUATION SUMMARY:

Historical surveillance test data and maintenance records were reviewed to evaluate the effect on safety. In addition, the licensing basis was reviewed for each proposed change to ensure it was not invalidated. Based on the results of these reviews, it is concluded that there is no adverse effect on plant safety due to increasing the surveillance test intervals from 18 to 24 months and the continued application of TRM section 4.0.2. The licensing basis remains valid. This activity is considered safe. All other changes are administrative and have no effect on safety.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-013 (SE 97-0013)

TITLE:

Changes to the Fire Hazard Analysis Report to Support a 24 Month Fuel Cycle

CHANGE:

Revisions to surveillance procedures DB-FP-04021 (18 Month Appendix R Wrap Visual Inspection), DB-FP-04023 (18 Month Fire Rated Barrier Inspection), DB-FP-04038 (18 Month 10% Penetration Seal Visual Inspection), DB-MI-04825 (Functional Test of Inaccessible Detectors for Node 5 C2720), and DB-MI-04827 (Functional Test of Inaccessible Detectors for Node 7 C4720). These surveillance procedures are related to Operating Specification Surveillance Requirements 8.1.1.A, 8.1.4.A, 8.1.4.C and 8.2.1.A.

REASON FOR CHANGE:

Extend the frequency of surveillance inspection and testing activities performed for inaccessible fire protection related equipment and systems in support of the 24 month fuel cycle.

SAFETY EVALUATION SUMMARY:

In order to achieve a 24 month fuel cycle with no planned mid-cycle shutdowns requires that the equipment that is inaccessible during plant operation be evaluated to see if there is a basis to extend the testing frequency for this equipment from 18 to 24 months. The format followed for the evaluation is the same as that followed in a previous Fire Protection Surveillance Engineering Evaluation where a number of surveillance frequencies were reviewed.

There were several factors that went into the decision process on what changes could be made. A selected set of plant procedures was reviewed to determine if there were available surveillance frequency extensions that could be adopted, based primarily on the past performance of the equipment, without affecting the reliability or availability of the equipment. The decision to extend surveillance frequencies and by what amounts was a subjective one based on the preponderance of the past performance data and experienced engineering judgment. For the fire detection and alarm system, the upgrade to a new system done via modification 91-0046, provides self monitoring for several of the functions previously evaluated by the performance of surveillance testing.

The evaluation has concluded that the changes to be implemented by this UCN are technically justified and will not adversely affect the reliability and availability of the fire protection equipment and systems, safety of the plant or the ability to achieve and maintain safe shutdown. The proposed changes are therefore considered safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-015 (SE 97-0004)

TITLE:

Control Room Emergency Ventilation System Operation

CHANGE:

Support operation of the Control Room Emergency Ventilation System (CREVS) in the pressurization mode immediately following a loss of coolant accident. This condition was reported via Licensee Event Report (LER) 96-007, Control Room Emergency Ventilation System Design Bases Calculation Error.

REASON FOR CHANGE:

- 1) Resolves concerns reported in LER 96-007. The CREVS must be operated in the pressurization mode much earlier than the 4 days assumed in the USAR to keep the operator doses below the General Design Criteria (GDC) 19 guidelines.
- 2) Resolves concern related to allowable opening size in the Control Room Pressure Boundary (CRPB) for maintenance purposes by tripping equipment room supply fan, C-44, if maintenance activities are planned which involve openings in the CRPB between the control room and the control room HVAC equipment.

SAFETY EVALUATION SUMMARY:

The safety function of the CREVS is to provide a habitable environment in the control room following an accident. It is designed to keep to radiation doses in the control room following an accident within the guidelines of 10CFR50 Appendix A, General Design Criteria (GDC) 19. The CREVS fans can be started from the control room either in the recirculation mode or the pressurization mode. In the recirculation mode, approximately 3300 cfm of control room air will be recirculated through charcoal filter units. In the pressurization mode, approximately 300 cfm of fresh air is brought into the control room through charcoal filters while 3000 cfm of control room air is recirculated through charcoal filters.

The safety function of the CRPB is to minimize the unfiltered inleakage into the control room following an accident. Automatic isolation of the Control Room Normal Ventilation System (CRNVS) establishes the CRPB. The CRPB includes the control room doors, walls, sealed wall penetrations in the control room boundary, CREVS duct work, and CRNVS duct work between the control room and the isolation dampers HV5301A-H and HV5362A-B.

GDC 19 requires that the control room radiation doses shall not exceed 5 rem to whole body or its equivalent to any part of the body, for the duration of the accident.

New calculations were performed to evaluate the impact of starting the CREVS in the pressurization mode immediately following the accident. Operators get to the procedural step requiring them to start the CREVS in the emergency operating procedures for large break LOCA approximately 5 minutes after the accident. However, credit for manual operator action in the control room within 10 minutes should not be considered. Therefore these new evaluations assumed that the operator will start the CREVS 10 minutes after the accident.

The calculated doses are below the NRC SER acceptance criteria.

Additionally, guidelines have been developed to minimize the potential for unfiltered air inleakage into the control room during the performance of maintenance activities involving control room ventilation, access doors and openings in the CRPB and to reestablish the control room pressure boundary during on going maintenance activities. At least one individual will be assigned to monitor CRPB openings and to reestablish the boundary when necessary.

Generic Letter 91-18 states that it is not appropriate to take credit for manual actions in place of automatic action for protection of safety limits to consider equipment operable. CREVS is a manually initiated system for the protection of Control Room personnel and is not required for the protection of a safety limit.

Consideration was given to the ability of the designated individual performing manual actions to recognize input signals requiring action, the ability and timing to reach the area where manual action is required, and occupational hazards (radiation, temperature, visibility, hearing) to be incurred while in the area of concern. The restoration of CRPB will be initiated if any of the following occurs: 1) the control room normal ventilation system trips; and 2) at the request of the control room.

The control room HVAC equipment room has no occupational hazards that would prevent an assigned individual from closing any maintenance or access ports at the start of any of the accidents described in the USAR requiring control room isolation.

Based on the above, the proposed changes are safe and do not involve an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-023 (SE 98-0015)

TITLE:

Radiologically Restricted Areas

CHANGE:

Updates the method and procedure described in the USAR sections 12.1.2.1.6, 12.1.5.3, 12.3.1, and 12.3.2.1 for designating radiologically restricted areas and the requirements for entry into these areas.

REASON FOR CHANGE:

The changes are necessary to bring the referenced paragraphs into literal compliance with the definitions and wording used in the revised 10 CFR 20.

SAFETY EVALUATION SUMMARY:

This USAR change clarifies the authority to declare radiologically restricted areas (RRAs) and to control the use of these areas by the Radiation Work Permit (RWP) process, no SSCs are affected.

These changes: 1) clarify that RRAs may be declared in the Turbine building on an as needed basis, and that a rope is not necessary to declare these areas; 2) clarify that an RWP is required only in designated RRAs to allow for RRAs with minimal radiation and/or contamination hazard to be declared and utilized without a formal RWP, at the discretion of the Manager, Radiation Protection (RPM); 3) clarify that Radiation Control Training (RCT), comprised of Rad Worker Training and Rad Worker Exercise, need be given to those workers assigned to designated RRAs, and not to all workers assigned to any RPA (i.e., turbine building); 4) designate all the areas formerly described in the SAR as the "Radiological Access Control Area (RACA)" as areas requiring an RWP for access at all times, and clarify the authority of the Manager, Radiation Protection to designate other RRAs; 5) clarify that the RWP is the controlling document for Radiation Protection means and methods in designated RRAs; and 6) clarify that invivo bioassay is performed on an annual basis for those individuals who have entered designated RRAs.

These changes have no effect on safety.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-032 (SE 97-0008)

TITLE:

Reorganizing Nuclear Operations Responsibilities

CHANGE:

This reorganization consists of modifying the existing three director functional arrangement to four directors with the associated realignment of activities to accommodate this new structure. Reporting to the VP Nuclear will be (1) the Plant Manager, responsible for Operations, Maintenance, Radiation Protection; (2) the Director - Engineering and Services, responsible for Design Basis Engineering, Plant Engineering, Regulatory Affairs, D-B Business Services, and D-B Supply; (3) the Director - Nuclear Assurance, responsible for Quality Assessment, and Nuclear Safety and Inspections; and (4) the Director - Nuclear Support Services, responsible for Nuclear Training, Security, Quality Services.

REASON FOR CHANGE:

The organizational change creates a new Director of Nuclear Support Services to assume some non-quality assurance functions from the Director of Nuclear Assurance.

SAFETY EVALUATION SUMMARY:

The proposed change to USAR Section 17.2 has no effect on any structures, systems and components or their associated safety functions, and does not affect the operation of any plant systems. The change is solely administrative as it revises the DBNPS site nuclear operations organization procedural responsibilities. All functions continue to be performed.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-035 (SE 97-0019)

TITLE:

Decay Heat System to AFW Interlock

CHANGE:

USAR Section 7.4.1.3.5 was revised to show the realignment of the normal position of valves MS106A and MS107A from normally closed to normally open. Addition of new Auxiliary Feed Water (AFW) System steam admission valves near the AFW turbines were also added. The Steam and Feed Water Rupture Control System (SFRCS) now initiates the AFWS by opening these new steam admission valves.

REASON FOR CHANGE:

PCAQR 97-0356 identified that USAR Section 7.4.1.3.5 required revision because the Decay Heat Removal System (DHRS) to AFW system interlock no longer exists as originally built.

SAFETY EVALUATION SUMMARY:

The original plant feature of preventing an SFRCS start of the AFWS once the DHRS is in service is essentially defeated by this change. However, no Design Basis requirement for the DHRS to AFWS interlock exists. Furthermore, the DHRS to AFWS interlock feature does not come into effect until the plant is in mode 4. Since SFRCS is not required below mode 3, this feature is outside of the operational requirements of the SFRCS.

Procedural controls require that the AFWS be removed from service once plant cooling is aligned to the DHRS. This action will prevent the AFWS from starting if actuated from the SFRCS.

If the AFWS were to be initiated while the plant was on the DHRS, the resultant cooldown rate would be small since the AFWS was not designed to cool the plant below 280°F. Any potential plant overcooling events associated with this condition are less severe than steam line breaks and other cooldown events analyzed in the USAR.

The change is safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-036 (SE 97-0054)

TITLE:

Response Time of DH11 and DH12

CHANGE:

The statement regarding response time for closure of containment isolation valves DH11 and DH12 will be deleted from USAR Section 7.

REASON FOR CHANGE:

The level of detail regarding valve stroke times is unnecessary since electrical power to these motor operated valves is removed and the valves are closed except when intentionally repositioning them.

SAFETY EVALUATION SUMMARY:

USAR Table 6.2-23, Containment Vessel Isolation Valve Arrangements, does not contain a response time since these valves are normally closed with power removed. DH11 and DH12 are not a part of SFAS, so no SFAS response time is imposed. DH11 and DH12 are only capable of being repositioned during plant cooldown or heatup, otherwise either the motor power or control power is removed from the motor actuators. No response time requirements exist in the design bases of these valves.

Deletion of the response time listed in USAR Section 7.6.1.1.1 will have no effect on any Decay Heat Removal System requirements or operations.

This change is considered safe and does not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-038 (SE 97-0041)

TITLE:

Toussaint River Dike Configuration

CHANGE:

Revise the USAR description to accurately depict the effluent flow path to the Toussaint River.

REASON FOR CHANGE:

In the mid 1980's the outside dike was rebuilt and the flow path that was described in USAR Section 2 was modified.

SAFETY EVALUATION SUMMARY:

Any credible levels of radioactivity in the Component Cooling Water System (CCW) would not adversely impact the function of any safety related components. The effects of the leakage from this system has been evaluated in previous Safety Evaluations which provide the conclusion that the dike provides hold-up for sampling, analysis and possible treatment prior to discharge.

The Borated Water Storage Tank (BWST), the largest inventory of radioactive material in an outdoor tank, has been evaluated for a catastrophic rupture without the credit of dikes and was found to be less than the 10CFR50 Appendix I limits for annual release. Also, the dikes provide a controlled release and holding for processing of this water so the resultant dose to the public would be less than previously evaluated.

Based on the above, the installation/repair of the dikes at the Unrestricted Area Boundary is deemed safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-046 (SE 97-0061)

TITLE:

Correcting the Room Number Given for Motor Control Center (MCC) E11B

CHANGE:

UCN 97-046 changes Room Number 303 to 304 as the location for motor control center (MCC) E11B.

REASON FOR CHANGE:

The correct location for MCC E11B is Room Number 304.

SAFETY EVALUATION SUMMARY:

Changing the USAR to list the correct room number for MCC E11B does not affect the safety function of the essential AC power distribution system. Both are within the Seismic Class I Auxiliary Building. Other documents correctly show MCC E11B in room 304. This change does not impact plant analyses or change the operation of any equipment.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-050 (SE 97-0016)

TITLE:

Revising the USAR for the 24-Month Fuel Cycle

CHANGE:

Revised the USAR to reflect the conversion to a 24 month fuel cycle.

REASON FOR CHANGE:

To analyze the impact of a 24 month fuel cycle on the source terms and the resultant effects on radiation protection structures, equipment qualification, hydrogen production and the accidents analyzed in USAR Chapter 15.

SAFETY EVALUATION SUMMARY:

The source terms given in USAR section 11 were based on an equilibrium cycle of 277 days. The reactor coolant system (RCS) activity used in Chapter 11 for the evaluation of radioactive waste management systems was based on 0.1% failed fuel. The plant Technical Specifications (TS), provide requirements for a radioactive effluents control program, for maintaining the doses to the members of the public from radioactive effluents as low as reasonably achievable. These requirements are not being changed by the extended fuel cycle lengths.

The radiation protection in the USAR Chapter 12 evaluation was based on the 1% failed fuel source term. The TS limit for iodine activity for continuous operation is significantly lower than the activity given in USAR Table 11.1-5 for 1% failed fuel. The difference in the isotopic inventory due to longer cycles is minor and does not impact the previously performed evaluations for post accident access. For these reasons, it is concluded that the radiation protection structures' design is not impacted by the extended fuel cycles.

As a part of the 18 month cycle evaluation, the calculated equipment qualification doses using 277 day equilibrium cycles were increased by 20%. The core activity for a 24 month fuel cycle is not significantly different than the core activities calculated for a 277 day fuel cycle. The 20% increase previously applied bounds the 24 month cycle equipment qualification doses.

Among the accidents analyzed in Chapter 15, only the control rod ejection accident (15.4.3), the Loss of Coolant Accident (15.4.6), and the fuel handling accident (15.4.7) utilize the activity in the fuel and /or in the fuel rod gap in the calculation of off site doses. The fuel handling accident has been evaluated in USAR sections 15.4.7.2.5.1 and 15.4.7.3.4.1 for a maximum burnup of 60,000 megawatt days per metric ton. This burnup

limit is not changed by the 24 month fuel cycle. Therefore no changes are necessary to USAR section 15.4.7.

A comparison between the source terms from the original USAR Table 15A-2, those for the 24 month cycle and TID-14844 (used in the analysis of the MHA) is to be added as Table 15A-6. This table shows that, although there are minor variations in the calculated core inventories, the source term for the 24 month cycle is not significantly different than the one given in the original USAR Table 15A-2. The TID's source term is more conservative than either the original USAR's or the newly calculated 24 month cycle's source term, with the exception of I-131. If the original USAR source term were used in the MHA evaluation, the results in Table 15.4.6-2 would show doses that are slightly higher for the thyroid, but significantly lower for the whole body and skin. The use of the new 24 month cycle source term would yield similar results.

The two year cycle source term does not impact hydrogen production due to radiolysis as discussed in USAR section 6.2.5.3. Use of the TID 14844 source term was conservative for that analysis. Therefore the two year cycle source term does not impact the conclusion of the original USAR analyses of section 6.2.5.3.

The proposed USAR changes will not increase the radiological consequences of the control rod ejection accident or, the Loss of Coolant Accident previously evaluated in the USAR because the conservative evaluations performed show that the calculated off site doses are below the safety evaluation acceptance criteria given in the USAR and the NRC Safety Evaluation Report for DBNPS.

Based upon this review the proposed changes are safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-051 (SE 97-0034)

TITLE:

Primary Filter Media Size Reduction Source Term Reduction Program

CHANGE:

Change the 0.2 micron filters to 0.1 micron filters in the Makeup and Purification, Spent Fuel Pool Cooling, and Clean Waste Systems.

REASON FOR CHANGE:

The finer media filters will remove finer particles of metals and metal oxides from the affected systems. More efficient removal of these materials will lead to a reduction of dose rates and CRUD levels in these systems.

SAFETY EVALUATION SUMMARY:

The 0.1 micron filters will provide additional protection to equipment downstream of the filters, but the new design of the filters will produce less ΔP than the current 0.2 micron filters at the same flow rate.

The 0.1 micron filters will remove more corrosion products from the filtered streams which should decrease the amount of activated corrosion products produced over time. The use of finer filters will provide for an overall exposure reduction by lowering the amount of radioactive particulates.

The proposed change has no adverse effect on safety. No functions important to safety are affected. The proposed change does not increase the adverse effects from any hazard. The systems and components affected have no safety function and therefore the proposed action is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-055 (SE 97-0035)

TITLE:

Potential Condition Adverse to Quality Report Initiator's Supervisor Signature

CHANGE:

Modify USAR Section 17.2.15.2 which requires the Potential Condition Adverse to Quality Report (PCAQR) initiator's supervisor's review.

REASON FOR CHANGE:

It was determined that the requirement for a supervisory review of the PCAQR was an unnecessary when the initiator was a supervisor.

SAFETY EVALUATION SUMMARY:

The proposed change revises the USAR procedural action of requiring the PCAQR initiator's immediate supervisor review the PCAQR during it's initiation. Under the change, any cognizant supervisor is allowed to review the PCAQR for completeness, or if the initiator is a supervisor or above, no further management review is required.

The proposed change does modify procedure requirements as outlined in the USAR but does not affect the facility as described in the USAR.

The 10CFR50.54 (a) review determined that this proposed change does not constitute a reduction in commitment. This proposed change is administrative in nature and has no effect on safety.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-056 (SE 97-0021)

TITLE:

Fire Hose Houses

CHANGE:

This USAR change will remove reference to hose houses HH9, HH10, HH12, HH17, HH18, and HH19 on USAR Figure 9.5-1.

REASON FOR CHANGE:

Fire hoses are no longer stored at the above mentioned hose houses located outside the protected area.

SAFETY EVALUATION SUMMARY:

The specific fire hose houses addressed above are located outside the protected area. These hose houses are not credited in the FHAR with the protection of power block components, therefore, this USAR change is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-057T (SE 98-0004)

TITLE:

Revise Frequency of Technical Requirements Manual Surveillance Requirement for Channel Calibration of Inside Containment Seismic Sensors to 24 Months

CHANGE:

Revise TRM Section 3 / 4.3.3.3, "Monitoring Instrumentation - Seismic Instrumentation," to modify the surveillance frequency for CHANNEL CALIBRATION of inside containment seismic sensors from 18 months to "At least once each REFUELING INTERVAL."

REASON FOR CHANGE:

This change is made using the guidance of GL 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

SAFETY EVALUATION SUMMARY:

The seismic instrumentation does not serve as a protective design feature or part of a primary success path for events which challenge fission product barriers, therefore, the Seismic Monitoring Instrumentation System has no nuclear safety-related function.

The proposed change extends the frequency of the surveillance testing for CHANNEL CALIBRATION of inside containment seismic sensors, TRM Table 3.3-7 Items 1a and 1b, to accommodate a 24 month operating cycle. The proposed change does not affect the safety function of any system, structure or component.

Historical surveillance data and maintenance records were reviewed to evaluate the effects on safety. In addition, the licensing basis was reviewed to ensure it was not invalidated. Based on the results of these reviews, it is concluded that there is no adverse effect on plant safety due to increasing the surveillance test interval for CHANNEL CALIBRATION of TRM Table 3.3-7 Items 1a and 1b from 18 to 24 months and the continued application of TRM 4.0.2 for the seismic instrumentation.

This activity is considered safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-076 (SE 97-0026)

TITLE:

Changes to USAR Table 5.1-7, Reactor Coolant System Piping Design Data

CHANGE:

This USAR change will correct the coolant volume and dry weight shown for the Reactor Inlet Piping and the Reactor Outlet Piping.

REASONS FOR CHANGE:

Coolant volume and dry weight for the Reactor Inlet Piping and the Reactor Outlet Piping was incorrect.

SAFETY EVALUATION SUMMARY:

The initiating documents made no physical changes to the facility. The values installed in Table 5.1-7 have been extracted from analysis of the Davis-Besse facility or other USAR Tables and Figures. The volumes for the hot leg and cold leg piping provided in USAR Figure 5.1-1 are approximately the same volumes that have been used in the RELAP and FLASH P models, validated codes that predict with accepted accuracy the response of the reactor vessel, the reactor internal and reactor core, and the reactor coolant system to various accident scenarios. Volumes from USAR Figure 5.1-1, which are consistent with RCS piping segment volumes, are listed in lieu of the values that are listed in Table 5.1-7. The dry weight values are taken from the Davis-Besse spool drawings. These changes do not have an effect on safety.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-077 (SE 97-0050)

TITLE:

Spent Fuel Pool Decay Heat Load

CHANGE:

Replace text description of the heat loads with the actual analyzed values in USAR sections that discuss the Spent Fuel Pool cooling.

REASON FOR CHANGE:

These changes reflect the responses to a Potential Condition Adverse to Quality issue and review the impact of a 24 month core on the Spent Fuel Pool Decay Heat load.

SAFETY EVALUATION SUMMARY:

The replacement of the descriptions of the heat loads with the actual analyzed values has no effect on safety. The inclusion of the plant conditions for normal and abnormal refueling activities assumed in calculation C-NRE-062.02-090 allows simple verification of compliance prior to refueling discharge. If these conditions are not met, then a specific SFP decay heat load analysis, based on actual fuel assembly operation and decay times, would be performed to verify that the analyzed heat load identified in the USAR is not exceeded. Since the refueling discharge heat load analyzed for a 24 month cycle and the core shutdown time assumed prior to a refueling discharge are not changed from those originally described in the USAR, there is no effect on safety. The identification of the additional discharge of 6 once-burned or 8 twice-burned fuel assemblies is to describe in the USAR a condition that can be implemented without requiring additional engineering evaluation of its effects on the SFP decay heat load.

Since the abnormal heat load analyzed for a 24 month cycle does not exceed the previously analyzed abnormal heat load, there is no effect on safety. The 24 month cycle analyzed abnormal heat load credited a minimum of 42 hours for the core's heat load decay. The inclusion of a minimum time to complete core offload is a change from the technical assumptions previously identified in this USAR text. This change is considered acceptable in that it reflects actual plant conditions. Experience has shown that at least 54 hours is required to complete core offload, which is bounded by the 42 hours assumed in the analysis.

The proposed changes to the USAR do not constitute an unreviewed safety question and are safe USAR changes. Since the analyzed decay heat loads have not been changed no functions of the systems are affected by this UCN text modification.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-078 (SE 97-0028)

TITLE:

Physical Separation Criteria for Fluorescent Lighting Cables in Free Air

CHANGE:

USAR Change Notice (UCN) 97-078 develops cable separation criteria to address cases involving free aired electrical cables supplying fluorescent lighting fixtures.

REASON FOR CHANGE:

To correct an error in Section 9.5.3.1 that states, "The pigtailed of fluorescent light fixtures in Cable Spreading Room are run through watertight flexible metallic conduit, and therefore are in conformance with plant cable separation criteria and also reduces fire hazard of having unprotected cables". A walkdown revealed that the 'pigtailed', which are cables in free air, are not enclosed within flexible metallic conduit.

SAFETY EVALUATION SUMMARY:

The proposed separation criteria for the free aired lighting cables will not adversely affect the safety function of the onsite electrical power system. The proposed criteria ensures that the onsite electric power system has sufficient independence to perform its safety function assuming a single failure.

Test results show that a faulted/overloaded conductor can not develop enough heat to damage an adjacent conductor. Electrical continuity was also tested. Electrical continuity was chosen as the acceptance criteria because, the safety function of these cables is to transmit power (or signals) before, during and after the electrical fault in the adjacent cables. Separation distance is specified in IEEE 384-1992 as applied to the free aired lighting cables. Based on the above, fluorescent lighting cables in free air is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-087 (SE 98-0014)

TITLE:

Revision of USAR to Incorporate Additional Wording to Clarify Our Control of Access to Setpoint, Calibration and Test Points

CHANGE:

Clarify how Davis-Besse controls access to setpoint adjustments, calibration and test points for RPS, SFAS, CRDS, and SFRCS for compliance to IEEE 279-1968 and IEEE 279-1971.

REASON FOR CHANGE:

USAR sections listed above say that these items are controlled through use of locked cabinets which are only accessible when the doors are open. This is not completely correct as field instruments are accessible all the time.

SAFETY EVALUATION SUMMARY:

This UCN will add additional wording to USAR sections 7.2.2.1, 7.3.2.3, 7.4.2.1.1, and 7.4.2.3.1 to clarify item number (4.18) of each section. These particular sections provide information as to compliance with either IEEE 279-1968 or IEEE 279-1971 depending on what system is involved. The following systems compliance wording is affected by this change: Reactor Protection System (RPS), Safety Features Actuation System (SFAS), Control Rod Drive Control System (CRDCS) (Trip Portion), and the Steam Feed and Rupture Control System (SFRCS).

The wording in the USAR is being changed to reflect compliance with the applicable sections of the IEEE standard. It is also worth noting that the new standard (IEEE Std 603-1991), under section 5.9, "Control of Access," says that access to the applicable safety system equipment (i.e. switches, transmitters, etc.) shall be controlled by provisions within the safety system, by provision in the generating station design, or by a combination thereof. This is interpreted to mean by either locked cabinets or controlled access via the general station access control to the protected and vital areas as described in the security plan, or administrative control by compliance with station procedures.

This wording change will have no effect on how these respective safety systems perform their safety functions.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-090 (SE 97-0030)

TITLE:

Spared Circuit BPYBU13A

CHANGE:

Spare circuit BPYBU13A was removed from Appendix 3 of the FHAR, which is the Circuit Coordination Evaluation Summary.

REASON FOR CHANGE:

During evaluation of Potential Condition Adverse to Quality Report (PCAQ) 97-0506, Circuit BPYU13A was discovered spared out sometime in the past.

SAFETY EVALUATION SUMMARY:

Elimination of the circuit from breaker 13 has no effect on panel YBU. Panel YBU supplies only non Class 1E loads. Since circuit BPYBU13A is not connected to any equipment, its designation has been changed to BPSPARE31. The need to verify breaker coordination is not required since the spare cable is not electrically connected, no failures can occur on this circuit which will effect panel YBU. The circuit designation BPYBU13A is no longer a valid number. The change to the FHAR is for documentation purposes. There are no safety concerns with identifying this circuit as spare. Plant drawing changes are incorporated with FPR 96-0714-901 which is associated with changing out the computer monitors in the control room.

The changes identified in UCN 97-090 are safe do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-093 (SE 97-0039)

TITLE:

Turbine Building Sprinklers

CHANGE:

This change to the FHAR will clarify the function and coverage of the various sprinkler systems in the Turbine Building.

REASON FOR CHANGE:

This change is a result of an audit finding which indicated that the existing wording was not clear.

SAFETY EVALUATION SUMMARY:

The changes to the wording clarify the description in the FHAR of the Turbine Building sprinklers. The current discussion states that the structural steel in the Turbine Building and Heater Bay does not have a 3-hour fire rating but because of the automatic sprinkler systems in the area, the steel will not fail. This broad statement implies that every sprinkler in every room in this fire area protects structural steel. This is not the case because there are systems in small rooms where there is no structural steel. There are also systems that are small in comparison to the overall space they are in and thus do not provide protection of the steel.

The revised wording makes the FHAR agree with the sprinkler coverage and function. The proposed changes are considered safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-094 (SE 97-0055)

TITLE:

Fire Hazard Analysis Report (FHAR) Changes Regarding Combustible Loading

CHANGE:

This UCN modifies the FHAR to update the combustible loading values in the front of each fire area's description in Section 4; revise Section 4 text relating to the sprinkler systems in room 209 and 405; and updates the list of Appendix A suppression systems in Section 8 for room 249.

REASON FOR CHANGE:

Audit Finding AR-97-FIREP-01-03 addressed revising the combustible loading in the FHAR and the function of the sprinkler systems in room 209 (Corridor Mechanical Penetration Room 1) and room 405 (Storage Room). During the review it was found that the sprinkler system for room 249 (Lube Oil Storage Tanks Room) was not properly classified.

SAFETY EVALUATION SUMMARY:

The combustible loading changes are to the loading values shown in the beginning of each of the fire area discussions in section 4 of the FHAR. The changes since the last revision are the result of several things: the addition of cables or other components to the plant; changes in the functions of rooms; or the change in room contents such as trash receptacles and supply cabinets

A review of calculations that incorporate combustible loading values has been conducted. Where appropriate, they are being updated to reflect the new values. The conclusions of these individual evaluations have not been affected.

The change to section 8 of the FHAR corrects the category of suppression for room 249. It is not listed as an Appendix A automatic suppression system in Table 8-4 of the FHAR. It should be listed there because it contains a significant quantity of combustible loading within the plant structures.

The wording relating to room 209 and room 405, in sections 4.G and 4.V of the FHAR, respectively, will be changed to delete the wording that implies the sprinklers are necessary for the non-rated aspects of the fire barriers.

The changes are safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-095 (SE 97-0036 R. 01)

TITLE:

Filtration Unit for Water Clarity Improvement

CHANGE:

Revise the USAR to add a description regarding the usage of a temporary, portable, submersible filtration unit for water clarity improvement and removal of suspended particulate matter in the Spent Fuel Pool, Fuel Transfer Canal, Cask Fill Pit and Refueling Canal.

REASON FOR CHANGE:

The filtration unit is being added to provide additional water clarity in the above mentioned areas, when needed.

SAFETY EVALUATION SUMMARY:

The unit is controlled administratively and positioned such that there is no interface with any fuel movement. Floor loading is negligible due to the weight of the unit and the design considerations for the tank floors in question. It is a contained unit with no outside piping and the filtered water is returned to the system at the unit. Failure of the filtration unit would result in no change in cooling capacity for stored fuel. The unit is designed to be submerged with an enclosed motor. Due to underwater environment, sufficient shielding is provided by the environment to ensure that there will be no adverse effect on source terms or radiation levels. Filter changeouts and movements of this unit are controlled by Radiation Work Permit to ensure that personnel dose is minimized and that the potential spread of contamination is controlled. The unit is restrained in case of seismic events and the stainless steel composition is compatible with the fluid environments in which it is used. The proposed action is deemed safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-102 (SE 97-0044)

TITLE:

Revision of USAR Figure 9.2-5

CHANGE:

This UCN adds relief valve (DM 6622) on the Auxiliary Domestic Hot Water Heater to the USAR.

REASON FOR CHANGE:

USAR Figure 9.2-5 did not previously show the Relief Valve.

SAFETY EVALUATION SUMMARY:

The addition of the existing safety relief valve on the USAR Figure will have no effect on how equipment is operated. Assignment of an equipment number and placing the valve on the drawing will make periodic testing easier to accomplish and track. Thus, placement of the valve on the drawing will have no effect on the safe operation of Davis Besse.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-105 (SE 97-0065)

TITLE:

Revise DC Load List

CHANGE:

Revision of the DC Load List contained in USAR Section 8.3.2.

REASON FOR CHANGE:

Replacement of Cyberex inverter/rectifier/DC panel assemblies with SCI equipment for channels 2 and 3, reducing the current demand on batteries 2P and 1N. Design changes were made to the plant computer system which increased the load on distribution panels YAU and YBU resulting in an increase in DC current for inverters YVA and YVB.

SAFETY EVALUATION SUMMARY:

Revision 14 of calculation C-EE-002.01-010 demonstrates the capability of the station batteries to support the new load requirements established by these design changes. In addition, this calculation demonstrates that the station batteries will provide sufficient DC power for the loads required to mitigate the worst case postulated accident scenario.

These changes are either conservative or within the allowable design margins previously assumed for the equipment. There are no new hazards created, no changes to existing hazards, and no effect on Fire Protection, Operations, Security, or the station 125 VDC distribution system failure analysis as a result of these design changes.

These changes are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-107 (SE 97-0047)

TITLE:

Delete Barometric Pressure and Solar Incidence Monitoring

CHANGE:

USAR Table 2.3-8 will be revised to delete the barometric pressure and solar incidence sensors.

REASON FOR CHANGE:

Barometric pressure transmitter PT-1171 has failed and is obsolete. This transmitter will not be replaced or repaired. Solar incidence transmitter RT-119 is functional, but will no longer be described in the USAR.

SAFETY EVALUATION SUMMARY:

Barometric Pressure and Solar Incidence are not required by any document committed to by Davis-Besse, including RG 1.23, and have no role in calculating dispersion of radioactive material during releases. Therefore, this change has no effect on safety. No hazards are created by this change, and the change is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-109 (SE 97-0053)

TITLE:

Procurement Engineering Reorganization

CHANGE:

The changes to USAR Chapter 17.2, Nuclear Quality Assurance Program Description, transfers the procurement engineering function from DB Supply to Design Basis Engineering.

REASON FOR CHANGE:

This USAR change reflects the Procurement Engineering reorganization.

SAFETY EVALUATION SUMMARY:

The proposed changes do not affect the safety function of any SSCs and are considered to be safe. The changes proposed by UCN 97-109U will not reduce the effectiveness of any program, will not reduce any oversight or reviews, will not eliminate any activities, and will not add any non-quality assurance functions to a quality assurance group. The proposed changes only involve non-reductions to existing commitments under 10CFR50.54(a). The proposed actions are therefore considered safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-113 (SE 97-0057)

TITLE:

Staffing Changes in the Operations and Maintenance Organizations

CHANGE:

USAR Section 13 will be changed to describe the responsibilities of the Supervisor - Outage Management position and the Supervisor - Operations Work Control position, to reflect the elimination of the Superintendent - Planning position and to update Figure 13.1-2.

REASON FOR CHANGE:

To restructure the maintenance planning and scheduling activities.

SAFETY EVALUATION SUMMARY:

The proposed change to USAR Section 13 has no effect on any structures, systems, or components or their associated safety functions. The proposed change is administrative in nature, and does not effect the operation of any plant system. The proposed change to USAR Section 13 will allow the maintenance organization to integrate maintenance planning into the maintenance units. Outage planning will be separated to allow outage planning to go on without taking away resources from day to day maintenance planning. Maintenance scheduling will be performed by the Supervisor - Operations Work Control Management to integrate the scheduling and tagging activities under Operations.

The changes are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-118 (SE 97-0058 R. 01)

TITLE:

Main Steam Isolation Valve Bypass Valve Testing

CHANGE:

Revise procedures DB-MI-03211, and DB-MI-03212, to direct testing of the MSIV bypass valve signal during power operation and revise the USAR to remove the exemption from testing.

REASON FOR CHANGE:

GL 96-01 has resulted in re-evaluation of the need to functionally test the SFRCS signal to the MSIV bypass valves on a monthly basis.

SAFETY EVALUATION SUMMARY:

The MSIV bypass valve is designed to handle the temperature, pressure, and flow that will occur during intermittent opening of the bypass valve for testing during power operation. Consequently, there is no increased probability of steam line breaks.

Calculation C-NSA-083.01-005 considered the effects of a Main Steam Line Break during testing of the bypass valves. This was done because the MSIV bypass valves stroke closed much slower than the MSIVs. It was determined that the longer stroke time would have insignificant effect on the plant's response to a Main Steam Line Break.

In the event an actual SFRCS signal occurs during functional testing of the SFRCS circuitry to the valve, the valve will respond to the SFRCS signal. Therefore, functionally testing the SFRCS signals to the MSIV bypass valves with the MSIVs open has no effect on plant safety.

The changes are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-120 (SE 97-0059)

TITLE:

SFAS Signal Testing

CHANGE:

Testing of the SFAS signal to the terminating relays of DH-7A, DH-7B, DH9-A, DH9-B, DH-2733 and DH-2734 during power operation.

REASON FOR CHANGE:

Verify the terminating relays respond to the appropriate SFAS signal.

SAFETY EVALUATION SUMMARY:

It has been evaluated that the tests can be performed with no risk of DH-7A or DH-7B going closed or DH-9A or DH-9B going open. This is because power is removed from the valve operators during testing, so that valve repositioning is not possible.

Testing DH-2733 and DH-2734 requires that the associated LPI train become inoperable. This is permitted for a limited time by Technical Specifications and the alternate LPI train will be at 100% capacity so that it will be capable of carrying out its safety function during the test.

Revising the USAR to reflect the above has no effect on safety because there is no need to describe an exemption from testing because the SFAS signals to the valves will be tested during power operation. Also, revising the table will make it consistent with the description in section 7.3.2.6.

These changes are considered safe and do not constitute an unreviewed safety question.

SAFETY EVALUATION SUMMARY
FOR
UCN 97-128 (SE 98-0011)

TITLE:

Eliminate Reference to specific QA "Hold Tags"

CHANGE:

Alter the process of tagging nonconforming hardware by eliminating the reference to specific "Hold Tags" and permit use of generic equipment status tags.

REASON FOR CHANGE:

Eliminates unnecessary detail in the overly prescriptive method of identification of nonconforming hardware by eliminating the reference to specific QA "Hold Tags" for installed and non-installed hardware.

SAFETY EVALUATION SUMMARY:

The change does not affect the safety function of any SSCs and does not affect the operation of any plant system or reduce any commitments to 10 CFR 50 Appendix B, or the Quality Assurance Program. The function of nonconforming parts, materials, and component identification and segregation will continue to be performed as required.

Under the change identification and segregation of nonconforming items will continue with a generically titled equipment status tag. The change does not alter the nonconforming equipment corrective action dispositions or allow use of nonconforming hardware which could affect USAR described equipment or system operation.

This change does not alter the Potential Condition Adverse to Quality Report (PCAQR)/ Condition Report (CR) dispositions of use-as-is or repair, or allow use of nonconforming hardware which could affect USAR, Fire Hazards Analysis Report (FHAR), or Technical Requirements Manual (TRM) described equipment or system operation. Use and disposition (use-as-is or repair) of nonconforming hardware would require reviews be conducted for impact on the facility as described in the USAR via the appropriate design change documents/ processes.

SAFETY EVALUATION SUMMARY
OF
UCN 98-020 (SE 98-0023)

TITLE:

Reorganization of Manager-Business Service Reporting Responsibility to Vice President
- Nuclear

CHANGE:

Revises the reporting responsibility of the Manager-Business Services from the Director of Engineering and Services to the Vice President - Nuclear.

REASON FOR CHANGE:

Facilitates the organizational realignment at DBNPS.

SAFETY EVALUATION SUMMARY:

The proposed change does not affect the safety function of any SSCs and does not affect the operation of any plant systems. The change is solely administrative in nature as it revises reporting responsibilities.

All functions of the Business Services department remain unchanged and continue to be performed. The technical qualifications and requirements continue to be provided by the Toledo Edison Nuclear Organization.

As required by Technical Specification 6.2.1a, the new organizational structure provides defined lines of authority, responsibility, and communication. The Staff training requirements of Technical Specification 6.4 are not changed by this reorganization.

Commitments to ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel, and ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants, continue to be met as described in USAR Table 17.2-1. The basis for NQAM and Design Control procedures are not altered and continue to identify responsibilities and methods for complying with specific design requirements and activities.

It is therefore determined that the proposed change is safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 98-021 (SE 98-0018)

TITLE:

Plenum Removal During Refueling Activities

CHANGE:

Revise USAR Section 9.1.4.2.3, "Loading and Removing Fuel," description to accurately reflect the processes leading up to refueling the core.

REASON FOR CHANGE:

USAR description of refueling activities is meant to be a broad overview. It is not intended to provide sequential steps of the process. A concern was expressed that the description was meant to be sequential. The changes being made are meant to remove any implications of sequence.

SAFETY EVALUATION SUMMARY:

The sequence of activities leading up to refueling the core is under the control of plant procedures. These changes do not affect commitments for safe operations during shutdown as required by Generic Letter 88-17, "Loss of Decay Heat Removal".

Clarifying the discussion when the plenum is lifted to state that it is acceptable to lift the plenum with the refueling canal dry or flooded is consistent with plant analysis. The two issues involved with this are the heavy load lift over the reactor and the radiation fields associated with the plenum. TE letter to the NRC, Serial 952, analyzed the effects of a dropped plenum with no water damping to cushion the fall. The results were accepted by the NRC. During the lift, radiation levels are monitored to ensure that no over-exposure of personnel occurs.

The proposed changes are considered safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 98-028 (SE 98-0026)

TITLE:

Changes to USAR Sections 13.0 and 17.2 Management Responsibility Descriptions

CHANGE:

The proposed changes involve moving responsibility for the trending and analysis program from the Manager - Quality Assessment to the Director - Nuclear Assurance; moving the industrial health and safety function from the Manager - Regulatory Affairs to the Manager - Nuclear Safety and Inspections; and moving the non-radiological environmental compliance function from the Manager - Regulatory Affairs to the Manager - Radiation Protection. Also included are some changes to descriptions such as the Director - Engineering and Services and the Director Nuclear Assurance regarding the transfer of the reporting relationship of the Manager - DB Supply to the Director - Nuclear Assurance.

REASON FOR CHANGE:

Facilitates organizational realignment at the DBNPS.

SAFETY EVALUATION SUMMARY:

The proposed changes involve administrative descriptions of the "Conduct of Operations" and "Nuclear Quality Assurance Program" sections of the USAR and cannot directly affect structures, systems, or components. The proposed changes do not indirectly affect structures, systems, or components since the descriptions only involve the overall responsibilities for the implementation of activities that are described within the affected USAR sections.

The individual activities are not being altered by these proposed changes. Commitments to 10CFR50 Appendix B, ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," or ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants" continue to be met.

These changes are safe.

SAFETY EVALUATION SUMMARY
FOR
UCN 98-047 (SE 98-0033)

TITLE:

USAR Changes to Section 11

CHANGE:

Corrections to USAR Table 11.4-1, Table 11.6-3 and Section 11.6.5.

REASON FOR CHANGE:

USAR Table 11.4-1 is being changed to correct detector sensitivity values to the sensitivities reported by the vendor, to delete the Background column and to correct the range values for RE-1003A. Table 11.6-3 is being changed to correct the low level dose (LLD) values for Cesium detection for Airborne Particulate. Section 11.6.5 is being corrected to provide consistency with the Radiological Effluent Technical Specifications (RETS) and Offsite Dose Calculation Manual (ODCM).

SAFETY EVALUATION SUMMARY:

The instrument sensitivity updates were inadvertently not applied during the modification to digital process channels in 1993. The range value correction for RE-1003A is based on the latest calibration report by the vendor. These changes have no effect on the instruments' abilities to perform their functions or on the processes used to calibrate the instruments.

The "Background" column is not described in Section 11.4.1 and does not correspond to any programmatic or operability requirement. The removal of this information provides additional clarity to the columns remaining in the Table 11.4-1.

The corrections to Section 11.6.5 and Table 11.6-3 provide consistency with the wording and requirements of RETS and ODCM and are in accordance with the requirements of Regulatory Guide 4.8.

The preceding changes are safe and do not constitute any unreviewed safety questions.

SAFETY EVALUATION SUMMARY
FOR
UCN 98-056 (SE 98-0038)

TITLE:

Remove Detail on Inoperable CRD Position Indication From USAR Section 4.3.5.3

CHANGE:

UCN 98-056 removes a statement regarding actions taken when Control Rod Drive (CRD) Position Indication meters are inoperable.

REASON FOR CHANGE:

The actions listed in section 4.3.5.3 of the USAR are not consistent with Technical Specifications.

SAFETY EVALUATION SUMMARY:

Section 4.3.5.3 of the USAR states "For a position indication less than -10%, or greater than 105%, the signal is to be considered bad and replaced by a group average position obtained by averaging the remaining good rod position signals within the same rod group." This is not consistent with Technical Specification 3.1.3.3 which has specific actions to take when a rod position indicator is inoperable. These actions do not allow the use of an averaged rod position signal in place of the bad signal for determining rod position. When a position indication signal is declared inoperable, Technical Specification 3.1.3.3 requires that thermal power be reduced to <60% of allowable for the reactor coolant pump combination (RCP) and the high flux trip setpoint must be reduced to <70% of allowable for the RCP combination or the position indication of the rod must be verified using the zone reference lights and operation must be within the limits of Technical Specifications 3.1.3.5, 3.1.3.6 and 3.1.3.9 as applicable.

The method of replacing a bad position indication signal with an average position that is described in USAR section 4.3.5.3 does not determine the actual position of the rod nor does it offer any benefits to operating the plant. Using alternate indications (zone reference lights) as required by Technical Specifications will determine the actual position of the rod. Replacing a bad position indication signal with an average signal adds no benefit to system operation because Technical Specification 3.1.3.3 is more restrictive and requires a power reduction or that the position of the rod be verified using other indication. The Control Rod Drive Control System requirements in the USAR and Technical Specifications are still being maintained, so there are no effects on safety.