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ComEd

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Enclosed for your information is the Annual Report for LaSalle County Nuclear Power Station for the period January 1995, through December 1995.

D. J. Ray
Station Manager
LaSalle County Station

DJR/mkl

Enclosure

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TABLE OF CONTENTS

I. Introduction

The LaSalle County Nuclear Station is a two-Unit facility owned by Commonwealth Edison Company and located near Marseilles, Illinois. Each unit is a Boiling Water Reactor with a designed net electrical output of 1078 Megawatts. Waste heat is rejected to a man-made cooling pond using the Illinois river for make-up and blow-down. The architect-engineer was Sargent and Lundy and the contractor was Commonwealth Edison Company.

Unit one was issued operating license number NPF-11 on April 17, 1982. Initial criticality was achieved on June 21, 1982 and commercial power operation commenced on January 1, 1984.

Unit two was issued operating license number NPF-18 on December 16, 1983. Initial criticality was achieved on March 10, 1984 and commercial power operation commenced on October 19, 1984.

This report was compiled by Michael J. Cialkowski, telephone number (815) 357-6761, extension 2056.

II. Annual Reportable Documentation for Unit 1 and 2

A. Summary of Operating Experience

The summary of the operating experience has been reported monthly in LaSalle's NRC Monthly Reports (Section II.A) dated January 1995 through December 1995. For safety related maintenance (non-outage related) performed during the period of January 1995 thru December 1995, see Attachment A.

B. Unit Outages and Power Reductions

For unit outages, see Attachment B. For unit power reductions see Attachment C.

C. Radiation Exposure

For the radiation exposure of LaSalle Station personnel for the reporting period of January 1, 1995 to December 31, 1995, see Attachment D.

D. Indications of Failed Fuel Elements

During this reporting period, January 1, 1995 through December 31, 1995, there were no indications of failed fuel elements.

E. Tests and Experiments not covered in the Safety Analysis Report

During this reporting period, January 1, 1995 through December 31, 1995, there were no tests or experiments conducted that are not covered in the Final Safety Analysis Report.

F. Changes to Procedures Covered in the Safety Analysis Report

LAP-200-5 Revision 6, Transfer of Control Room Command Functions Between The Control Room Unit Supervisors, The Shift Engineer, And/Or The Field Supervisor

The procedure was revised to allow transfer of control room command functions between Unit Supervisors, Field Supervisor, and the Shift Engineer. This consisted of title changes to reflect the nomenclature describing the present description of the Operating organization at LaSalle Station. The Field Supervisor maintains the same qualification as the Shift Foreman.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this activity.

F. **Changes to Procedures Covered in the Safety Analysis Report** (continued)

LAP-240-7 Revision 6, Defeating Annunciators

This procedure was revised to update the method of addressing control room nuisance alarms. It also incorporates the method of time delaying annunciators from LaSalle Administrative Procedure (LAP) LAP-240-6, Temporary Alterations, therefore, all annunciator problems are addressed by a single administrative procedure.

The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety evaluation associated with the procedure revision.

LAP-300-29 Revision 2, Rigging and Lifting Program

The procedure was revised to incorporate the contents of LaSalle Maintenance Procedure LMP-GM-09, Safe Rigging Practices. LMP-GM-09 was subsequently deleted.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this activity.

LAP-820-10 Revision 10, "Periodic Procedure Review", and Changes to UFSAR Section 12.5.3 and Appendix B, and Addition of UFSAR Section 13.5.3.

LaSalle County Station previously committed to ANSI 18.7, 1972 through the UFSAR and Regulatory Guide 1.33. The station changed this commitment through the Commonwealth Edison Quality Assurance Manual Topical Report, dated 1993.

Changes to the above UFSAR and LAP-820-10 reflect compliance with the Topical Report and document our approved deviation from Regulatory Guide 1.33 (as approved in the Topical Report of 1993). UFSAR Section 12.5.3 was revised to remove the procedure review requirement with information being added which directs that the review be performed per the controls stated in Section 13.5. This review has now been incorporated into the review requirements in Section 13.5, which governs the review of all station procedures.

Section 13.5.3 was added to control procedure review. This new section provides the foundation for the procedure review system. It clearly states what ANS standard procedure reviews are conducted under, and goes further to explain that this standard does not apply elsehwere within the procedure change process. This will assure that the station will remain within the established program guidelines as required by Regulatory Guide 1.33 and the ANSI standards.

The 10CFR50.59 safety evaluation concluded that there were nounreviewed safety question associated with this activity.

F. Changes to Procedures Covered in the Safety Analysis Report
(continued)

LOA-DC-03 Revision 5, "48/24 VDC System Failure"

The procedure revision identifies specific equipment lost due to a partial or complete loss of 24/48 VDC and other associated alarms with the loss of process radiation monitors. Revisions to references contained in the Technical Specifications to reflect loss of the 24/48 VDC system. which includes reference to the Offsite Dose Calculation Manual (ODCM), and reflects UFSAR sections 7.6. 7.7, 8.3, and 13.5.

A 10CFR50.59 safety evaluation concluded that no unreviewed safety question exists as a result of this modification.

LOP-WF-30, Revision 3, Generic Transfers and Draining/Filling Procedure

This change provided direction for transferring chemical radwaste reprocessing (WZ) Collector Tanks to a vendor container located in the Radwaste Truck Bay. The change involved connection of a hose to a hydrolazing port in the WZ feed line located in the upper radwaste pipe tunnel. The transfer is implemented per procedure LOP-WF-30, and it provides a means to transfer a 1200 gallon sample to a vendor shipping container.

The 10CFR50.59 safety evaluation concluded that there was not an unreviewed safety question.

LOP-WX-06, Revision 9, Establishing a Waste Sludge Tank Transfer Loop

The procedure outlines the steps necessary to establish a transfer loop for the waste sludge tank and the solid radwaste system. The revision incorporates allowance the running of the loop without the recycle pumps, by backflushing to the bottom suction of the waste sludge tank. The backflushing with cycled condensate is aimed at eliminating pipe plugging, and to minimize the potential for elevated dose rates in the transfer pump area.

The safety evaluation concluded that there was no unreviewed safety question.

LTP-100-6, Revision 2, Time Delay Relay Calibration Procedure

This procedure was revised to reflect the addition of Reactor Core Isolation Cooling (RCIC) Low Reactor Water Level (L2) I Initiation Time Delay Relays. The relays 2B21A-K710AX and 2B21A-K710CX were installed per Component Replacement C01-2-94-032 (DCP 9400117). Component Replacement C01-2-94-033 (DCP9400118) replaced relays 2B21A-K710BX and 2B21A-K710DX.

F. Changes to Procedures Covered in the Safety Analysis Report
(continued)

LTP-100-6, Revision 2, Time Delay
Relay Calibration Procedure-(continued)

UFSAR Section 7.4.1.2.3 is being revised to reflect that the time delay is now applicable to both units. The revision of the UFSAR in effect at the time of the procedure revision reflects the time delay as being applicable to Unit 1 only.

The 10CFR 50.59 safety evaluation concluded that no unreviewed safety question exists for thie procedure.

LTP-1600-10, Calculating Core Thermal Power,
and Update to UFSAR Figure 1.2-1

The procedure was revised to account for an additional 3 megawatts thermal (MWth) in the heat balance methodology. 3 MWth is added to the constant term for fixed heat losses from the reactor pressure vessel. This accounts for 8 gpm of seal purge flow to the reactor recirculation pumps that is routed from the control rod drive (CRD) flow. During normal operation, 4 - 6 gpm of s.al purge flow exists, however conditions can exist where this flow increases to 7 gpm. 8 gpm per pump is incorporated as a bounding assumption.

UFSAR Figure 1.2-1 is being updated to reflect the revised heat balance methodology.

The 10CFR 50.59 safety evaluation concluded that no unreviewed safety question exists for thie procedure.

LAP-200-10	Revision 2,	NRC Operator License Active Status Maintenance and Reactivitation.
LAP-220-2	Revision 17,	Unit Operators' Log.
LAP-1600-2	Revision 50,	Conduct of Operations.
LAP-200-1	Revision 30,	Operating Department Organization.
LAP-200-3	Revision 27,	Shift Change.
LOS-AA-S1	Revision 52,	Shiftly Surveillance

The above listed procedures were revised to reflect the enhancements made to the control room organization.

This change was done to modify the organization of control room operators. The Center Desk and fourth Nuclear Station Operator (NSO) positions were replaced with Unit Assist NSOs. Duties previously performed by the Center Desk NSO to support unit operations are performed by the Unti Assist NSOs.

UFSAR Section 7.8.1 was revised to reflect the change of the Center Desk operator.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with the UFSAR and procedure revisions.

F. Changes to Procedures Covered in the Safety Analysis Report
(continued)

LOA Procedure Revisions associated
with the Diesel Generators (DGs)

LOA 0DG03J-3-4 Revision 3 includes a revised trip setpoint for the 0 DG neutral ground trip relay to reflect design change E01-0-94-960-D, DG 0 Neutral Ground Relay Replacement.

LOA 1(2) DG03J 4-1 Revision 2 includes a revised trip setpoint for the 1A DG underfrequency relay to reflect design change C01-1-94-007-B, Replacement of Underfrequency Relay for DG 1A.

LOA 1(2) DG03J-3-4 Revision 2 includes a revised trip setpoint for the 1A DG neutral ground trip relay to reflect design change E01-1-94-960-A, DG 1A Neutral Ground Relay Replacement.

LOA 0DG03J 4-1 Revision 3 includes a revised trip setpoint for the 0 DG underfrequency trip relay to reflect design change C01-1-94-007-A, Replacement of Underfrequency Relay for 0 DG.

LOA 1(2) DG03J-3-5 Revision 2 includes a revised trip setpoint for the 1A DG reverse power trip relay to reflect design change E01-1-94-960-B, DG 1A Reverse Power Relay Replacement.

LOA 0DG03J-3-5 Revision 3 includes a revised trip setpoint for the 0 DG reverse power trip relay to reflect design change E01-1-94-960-C, DG-0 Reverse Power Relay Replacement.

The 10CFR50.59 safety evaluations concluded that there were no unreviewed safety questions associated with the procedure revisions.

LaSalle Special Procedure LLP-95-006 Revision 0,
RHR System A/B Valves 1(2)E-F040A/B and 49 A/B
Motor Operated Valve Dynamic Votes Test

The procedure allowed for differential pressure (DP) VOTES testing as required by NRC Generic Letter 89-10 for RHR System A/B Valves 1(23)E-F040A/B and 49 A/B. Operation of the RHR system was conducted through existing operating procedures. The system was placed in the full flow test mode from suppression pool to suppression pool. Water was not injected in to the reactor vessel as part of this test. This procedure was considered as 'data collection' only. The procedure was performed such that the operation of the subject valves did not impact safe plant operation.

The 10CFR50.59 safety evaluation concluded that there was not an unreviewed safety question associated with this design change.

F. <u>Changes to Procedures Covered in the Safety Analysis Report</u>
(continued)

LaSalle Special Procedure LLP-95-038, Installation and Removal of Temporary Connections for Unit 2 RHR Chemical Decon Temporary Heat Exchanger

The procedure provided the control and approval method for installation of temporary connections for hoses for a temporary heat exchanger to support Unit 2 chemical decon. The small temporary heat exchanger was attached to the service water system using hoses on existing service water connections. The heat exchanger was part of the Vectra skid which was evaluated for seismic considerations prior to use. The system was returned to its original condition following completion of the decon.

The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety evaluation associated with the procedure revision.

LaSalle Special Procedure LLP-95-046 Revision 0, Filling and Draining Reactor Recirculation (RR) Loop 'A' with Jet Pump Plugs Installed.

The procedure provided guidance to drain and fill the Unit 2 A loop with the jet pump plugs installed. The plugs were installed to allow draining the RR system for maintenance activities on the 2B33-F067A valve. The plugs are designed to utilize head pressure with a mechanical clamp preventing leakage of coolant on to the drywell floor. A cover plate was installed whenever the valve bonnet was removed and no one was present.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with the procedure.

LaSalle Special Procedure LLP-95-107, Operability Test of the new Agasat Time Delay Relay (2E31- K11B)

The test was performed to to verify the both the operability and the proper installation of a new Agasate Time Delay Relay (2E31- K11B) in the power sensing circuitry for Leak Detection. The RCIC Relay Race was eliminated due to the replacement of the previous Agastat GPI relay with a Agastat TR Timing Relay for 1(2)E31-K11B.

The test only affected circuits in Division 2, and had no effect on instrumentation. The Division 2 circuits were restored to their original design condition after the testing was completed. Valves 2E51-F063 and 2E51-F076 were stroked closed during the testing, thereby rendering RCIC inoperable for the duration of this test.

The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

F. Changes to Procedures Covered in the Safety Analysis Report (continued)

LaSalle Special Procedure LLP-95-109, Operation with 7 Condensate Polishers On Line During Normal Power Operation

The procedure was written to allow operation of a unit for a trial period with all seven condensate polishers are on line, as opposed to the routine operation consisting of 6 polishers, and 1 spare. During the trial period, determinations were to be made if there were any applicable improvements in reactor chemistry and/or other operation practices.

The 10CFR50.59 safety evaluation concluded that there were no unreviewed safety questions associated with the procedure.

LaSalle Special Procedure LLP-95-116, Operation of the Reactor Building Overhead Crane Within 15 Feet of the East or West Wall

This procedure provides for guidance for operating the Reactor Building Overhead Crane (RBOHC) in inaccessible regions of the Refuel Floor. The procedure allows temporary removal of the trolley east and west limit switch arms to allow the crane access to areas within 15 feet of the east or west walls. The limits are restored during closeout of the procedure.

The 10CFR 50.59 safety evaluation concluded that no unreviewed safety question exists for thie procedure.

LaSalle Special Test Procedure LST-95-007 Revision 0, "Unit 2 Reactor Protection System Alternate Power Source Voltage Regulation Test."

The procedure provided instructions for testing the "Solatron" voltage regulator 2APA9E. This was done to verify that the regulator was capable of regulating its output voltage in accordance with manufacturer's operating characteristics.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

LaSalle Special Test Procedure LST-95-039 Revision 0, "Unit 2 Reactor Protection System Alternate Power Source Voltage Recording During Reactor Recirculation Pump 2B33-C001A/B Start"

The procedure provided instructions for the recording of voltages at the Reactor Protection System (RPS) alternate power source during Reactor Recirculation (RR) pump start. The output of "Solatron" Voltage Regulator 2APA9E (the alternate RPS power supply) was electrically isolated from the RPS bus during this test.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

F. **Changes to Procedures Covered in the Safety Analysis Report**
(continued)

LaSalle Special Test Procedure LST-95-055 Revision 0,
Unit 2 RR FCV Oscillation Data Acquisition and Adjustment

The 2A Reactor Recirculation (RR) Flow Control Valve (FCV) was observed to exhibit excessive oscillations while being moved in the open direction. The special test procedure was written to provide instructions for acquisition of data to determine the cause of the oscillations, as well as to provide guidance for system adjustments to correct or mitigate the problem.

Temporary instruments were connected to the Hydraulic Power Unit (HPU) to determine cause of 2A RR FCV oscillations. Pressure transmitters were connected to high point vent connections in the hydraulic system, and electrical connections were made to the Hydraulic Power Unit (HPU), HPU control drawer, Flow control drawer, and Servo control drawer. Instruments were also connected to the 2B RR loop for comparison purposes. The gain settings on controllers before and after the velocity demand limiter were allowed to be adjusted, as required, based on the data gathered.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with the special test procedure.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report**

Modification M01-0-89-016D

The impressed current studies performed by Commonwealth Edison's Corrosion Engineering Group indicate that the level of cathodic protection afforded by the distributed anode system was unsatisfactory due to system failures. The near-surface distributed anodes and wiring were readily distributed and faults were difficult to locate and repair. To mitigate these problems, the new cathodic protection system utilizes deep-bed anodes for the majority of the protective current.

This modification completes design of deep bed anode cathodic protection in the east area of the plant by installing two new deep anodes and by providing a local supply of D.C. current to distributed bed anode systems No. 1 and No. 2. These two near-surface zones protect underground piping from approximately the Lake Screen House to the CSCS Cooling Pond water inlet chute and are less subject to incidental damage. The remaining near-surface distributed bed anode systems 5 and 6 were disconnected and abandoned.

The safety evaluation concluded that there are no unreviewed safety questions associated with this modification.

Modification M01-1-91-008

The 120 volt AC control circuits of dampers 1VQ037 and 1VQ038 were modified by adding a limit switch intermediate open (IO) contact to the closing circuits. This allows closing the dampers on limit rather than on torque, to prevent unnecessary tripping of the thermal overload relays during the damper closure stroke.

The safety evaluation concluded that there was not an unreviewed safety question.

Modification M01-1-93-008

This modification improved the reliability and operability of motor operated valves 1E12-F042B and 1E12-F042C. Specific requirements relating to operability have been presented in NRC Generic Letter 89-10. This modification was selected as the preferred alternative because it has the lowest cost and provides the greatest direct benefit of improved valve reliability in normal operation.

This modification also installed a new 3/C #6 AWG power cable for each of the subject valve operators. These cables replaced 3/C #14 AWG power cables, which were abandoned in place. The new, larger cables provide increased voltage at the motor terminals under all bus voltage conditions.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-1-93-008
(continued)

This modification also changed the overall actuator gear ratio from 42.50:1 to 88.40:1. The gear change was accomplished by replacing the motor pinion and worm shaft gears. The increased gear reduction increases the available operator torque and thrust.

UFSAR Tables 6.3-2 and 6.3-3 are being revised to reflect the valve stroke and system response time changes.

A 10CFR50.59 Safety Evaluation was performed, and concluded that no unreviewed safety question exists as a result of this modification.

Modification M01-1-94-001

The modification installed a passive zinc injection skid to inject zinc solution in to the Unit 1 Feedwater and Condensate Systems. This will help to mitigate the buildup of Cobalt 60 (Co-60) levels in the recirculation system, thereby significantly reducing personnel dose rates, and maintaining exposure ALARA.

UFSAR Sections 10.4 and 12.2 are being revised to reflect the installation of the modification.

The Safety Evaluation concluded that there was not an unreviewed safety question associated with this design change.

Modification M01-1-94-008

Agastat type 7022AB time delay relays were installed in the low suction pressure trip circuitry for pumps 1FC01PA and 1FC01PB, and functionally tested by a temporary system change. They were added to avoid spurious trips. The time delay relays are set at 2 seconds, which doesn't cause excessive cavitation of the pumps. A low suction pressure trip will still occur at the present setpoint, however, the low suction pressure must be sustained for a 2 seconds. This modification and modification M01-2-94-011 will make the change permanent, thereby clearing Temporary System Change (TSC) No. 2-306-93.

A 10CFR50.59 safety evaluation for this Modification concluded that no unreviewed safety questions exist.

G. **<u>Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)</u>**

M01-2-87-087-3 2E12-F050A
M01-2-87-087-4 2E12-F050B
M01-2-87-087-7 2E22-F005
M01-2-87-087-8 2E21-F006
M01-2-87-087-9 2E12-F041A

These modifications replaced each ECCS Testable Check Valve's packing with fewer rings of packing combined with a carbon spacer, to reduce valve shaft friction; removed of the leakoff line to accomodate the newer packing; and replaced limit switches with smaller and lower torque microswitches. The valves previously encountered rotational resistance during low flow testing conditions, which could prevent the valves from closing during this testing. These changes are intended to minimize the rotational resistance. UFSAR Figure 7.3-7 is being revised to reflect these changes.

The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety question associated with these design changes.

Modification M01-2-89-011-A

This modification replaced, for each Turbine Driven Reactor Feed Pump (TDRFP), the speed control system with a state-of art, highly reliable and fault-tolerant micro-processor control system. The replacement system is manufactured by Lovejoy Control Corporation (LCC). The previous design was obsolete, it couldn't adequately maintain turbine speed and feedwater flow and required extensive maintenance.

The redundancy provided in the new system does not allow any single electronics failure to cause a loss of or increase in the turbine speed. The design also provides stable operation at relatively unloaded or zero speeds to pumping speed, pumping speed range regulation for precise but stable control and prevents most accidental overspeeds. The system requires less overall maintenance as there are fewer moving parts.

UFSAR Sections 7.7.4.2.2 (page 7.7-39) and 7.7.4.3 (page 7.7-40) are being revised to reflect the installation on the design change.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with the design change.

Modification M01-2-91-001

The modification provided overpressure protection by installing a relief valve on chiller units 2VP04AA and 2VP04AB (station heating system), and 2VP16A (service water system). The relief valves are set to lift at a pressure of 150 psig.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this activity.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-2-92-001

This modification converted the Primary Containment Ventilation System (VP) dampers serving the Control Rod Drive (CRD) and Reactor Recirculation (RR) Pump areas from control dampers to balancing dampers. This was accomplished by removing the damper actuators and limit switches from the dampers.

The dampers modified were the 2VP01Y, 02Y, 08Y and 09Y. These dampers control the amount of air supplied to the RR Pump area and the CRD area under the vessel, and operate in pairs with one damper opening to the RR Pump area and the other to the CRD area. The dampers that serve the CRD area are closed during normal operating conditions and open only if the CRD area temperature exceeds 150 degrees F. Maximum temperatures are limited to less than 185 degrees F. This modification was implemented due to repeated failures of both the damper actuators and the limit switches. While trouble with ITT actuators is not uncommon, access to the dampers and limit switches is limited to outages. Repeated maintenance required by the dampers is undesirable due to the high dose rate in the under vessel area.

In 1991 temperature data was collected, per procedure LST-91-013, to determine the actual temperature in the CRD area during normal operating condition. Data was collected prior to and during a full core SCRAM. It was postulated that the maximum heat load would occur during and after a SCRAM. Collected temperature data confirmed that the maximum temperature was well below the allowable temperature of 185 F. Based on this data, it was decided that the damper actuators and associated control and indication devices were not needed to insure that CRD temperatures remained below 185 F.

UFSAR Table 3.11-13 is being be revised to reflect this modification. A 10 CFR 50.59 Safety Evaluation concluded that there was no unreviewed safety questions exist due to the design change.

Modification M01-2-92-006

The modification added a full flow recirculation line for CRD pumps 2C11-C001 A and B, added test taps for each CRD pump discharge line, and replaced the CRD pump discharge stop-check valves 2C11-F393A and B with globe type valves.

The full flow recirculation lines will be used during operating periods of reduced flow requirements, such as a refuel outage. The line provides a means of maintaining the pump recommended minimum flow during reduced CRD system flow conditions. Lack of adequate flow has resulted in a considerable amount of pump wear, damage, and degradation in pump performance. The recirculation line is shut down and isolated during reactor power operation.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-2-92-006
(continued)

UFSAR Section 4.6.1.1.2.4.2.1 is being revised to reflect the Unit 2 modification. The safety evaluation concluded that there was not an unreviewed safety question.

Modifications M01-2-93-001, M01-2-93-003, and M01-2-93-005

These modifications replaced the ECCS and RCIC water leg pump discharge lift check and globe stop check valves with swing check and globe stop valves. Some of the replacement valves are repositioned to minimize piping stress and ensure accessibility.

The valves were replaced to reduce the amount of required maintenance and improve the reliability of the valves. The lift check valve has proven to be susceptible to binding due to the small clearances within the valve. The replacement swing check valve, with its simple design and larger clearances, is considered a more reliable valve.

The required functions of the removed valves will are performed by the replacement valves. The number of check valves in each system is decreased from two to one, since only one check valve is needed to perform the backflow isolation function. The globe stop valve will provide the isolation necessary for maintenance.

UFSAR Section 6.3.2.2.5 is being revised to reflect the modifications.

The safety evaluation concluded that there was not an unreviewed safety question.

Modification M01-2-93-008

The modification improved the reliability and operability of MOVs 2E12-F042B and 2E12-F042C. Specific requirements relating to operability were presented in NRC Generic Letter 89-10. Several alternatives were considered to resolve the operability concerns. This modification was selected as the preferred alternative because it has the lowest cost and provides the greatest direct benefit of improved valve reliability in normal operation.

This modification changed the subject valves' overall actuator gear ratio from 42.50:1 to 88.40:1. This was accomplished by replacing the motor pinion and worm shaft gears with Limitorque parts. The increased gear reduction increases the available operator torque and thrust.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-2-93-008
(continued)

The gear changes nearly double the valves' stroke times. The new calculated stroke times are about 37 seconds in each direction. These stroke times are approximate and may vary due to actual valve stroke and motor speed. These stroke times meet the Technical Specification Table 3.3-3 stroke time requirement of 40 seconds or less.

A 10CFR50.59 safety concluded that no unreviewed safety question exists as a result of this modification.

Modification M01-2-93-009

This modification improves the reliability and operability of motor operated valves 2E12-F042A and 2E21-F005. Specific requirements relating to operability have been presented in NRC Generic Letter 89-10. Several alternatives were considered to resolve the operability concerns. This modification was selected as the preferred alternative because it has the lowest cost and provides the greatest direct benefit of improved valve reliability in normal operation.

The overall actuator gear ratio (OAR) for 2E12-F042A was changed from 42.50:1 to 88.40:1. The gear change was accomplished by replacing the motor pinion and worm shaft gears. The increased gear reduction increases the available operator torque and thrust.

The Limitorque operator for 2E21-F005 was changed from an SMB-1 with a 40 ft-lb motor and OAR of 42.50:1 to an SMB-2 with an 80 ft-lb motor and an OAR of 72.01:1.

The gear changes increase the valves' stroke times significantly. The new calculated stroke times are about 37 seconds in each direction for 2E12-F042A and about 30 seconds in each direction for 2E21-F005. These stroke times are approximate and may vary due to actual valve stroke and motor speed. These stroke times are within the Unit 2 Technical Specification stroke time requirements for these valves.

A 10CFR50.59 safety evaluation concluded that no unreviewed safety question exists as a result of this modification.

Modification M01-2-93-011

This modification improved the reliability and operability of motor operated valve 2E22-F004. Specific requirements relating to operability have been presented in NRC Generic Letter 89-10. This modification was selected as the preferred alternative because it has the lowest cost and provides the greatest direct benefit of improved valve reliability in normal operation.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-2-93-011
(continued)

The previous gearing was not self-locking, and required a motor brake to ensure that the operator and valve remain in the desired position when the operator is de-energized. The new gearing is self-locking and does not require a motor brake.

This modification also changed the overall actuator gear ratio from 48.45:1 to 92.12:1. The gear change was accomplished by replacing the motor pinion and worm shaft gears.

UFSAR Tables 6.3-2 and 6.3-3, and Section 7.3.1.2 are being revised to reflect the modification. A 10CFR50.59 Safety Evaluation was performed, and concluded that no unreviewed safety question exists as a result of this modification.

Modification M01-2-93-012

This modification improves the reliability of and decreases the stroke time of motor operated valve 2E12-F024B. Reliability was improved by changing the closing circuit control logic so that the closing circuit can't be re-energized after the initial torque switch trip.

The control circuit revision eliminates the valve "hammering" that sometimes occurs when this valve is closed in the presence of an isolation signal. This situation occurs when the operator gearing relaxes following valve seating, allowing the torque switch "close" contacts to re-close, and re-energizing the closing circuit. This situation occurs only when an isolation signal is present or if the handswitch is manually held in the "close" position after the valve has seated.

No unreviewed safety question exists as a result of this modification.

Modification M01-2-93-013

This modification improved the reliability and operability of MOV 2E12-F024A. Specific requirements relating to operability have been presented in NRC Generic Letter 89-10. Several alternatives were considered to resolve the operability concerns. This modification was selected as the preferred alternative.

The modification installed of a friction element (high pressure drop) trim in the 2E12-F024A valve. The trim is intended to prevent severe cavitation when the valve is used to throttle flow through the "A" RHR full flow test loop. The cavitation results from the large pressure drop that occurs across the valve seat under such flow conditions.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with the modification.

G. Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)

Modification M01-2-93-014

The purpose of this modification is to improve the reliability and operability of motor operated valve 2E12-F021. Specific requirements relating to operability have been presented in NRC Generic Letter 89-10. Several alternatives were considered to resolve the operability concerns. This modification was selected as the preferred alternative.

The modification installed of a friction element (high pressure drop) trim in the 2E12-F021 valve. This type of anti-cavitation trim is used when normal flow is over the seat of the valve. The trim is intended to prevent severe cavitation when the valve is used to throttle flow through the "C" RHR full flow test loop. The cavitation results from the large pressure drop that occurs across the valve seat under such flow conditions.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with the modification.

Modification M01-2-93-016

The modification brought the Unit 2 System Auxiliary Transformer (SAT) 242 protective relaying package to the level required by the Commonwealth Edison Company standards. Overcurrent relay additions C0-2 and C0-7 were added to accomplish this.

C0-2 relay, ground differential scheme:
The low voltage (LV) windings of the SAT is resistance grounded, which limits the amount of ground current available for phase to ground faults. The differential and sudden pressure relays may be insensitive for phase to ground faults in the transformer LV windings or leads. Therefore, a transformer LV lead differential scheme, using high speed overcurrent relays, were needed as primary protection for these faults.

C0-7 relay, transformer low voltage breaker back-up:
This time overcurrent relay provides back-up protection in the event that the SAT LV breaker fails to interrupt for a LV bus multiphase fault. It also provides back-up protection for transformer internal or LV lead multiphase faults.

The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-2-93-019

The modification brought the Unit 2 Unit Auxiliary Transformer (UAT) 241 protective relaying package to the level required by Commonwealth Edison Company standards. Overcurrent relay additions CO-2, CO-7, and ITH were added to accomplish this.

CO-2 relay, ground differential scheme: The low voltage (LV) windings of the UAT is resistance grounded, which limits the amount of ground current available for phase to ground faults. The differential and sudden pressure relays may be insensitive for phase to ground faults in the transformer LV windings or leads. Therefore, a transformer LV lead differential scheme, using high speed overcurrent relays, were needed as primary protection for these faults.

CO-7 relay, transformer low voltage breaker back-up: This time overcurrent relay provides back-up protection in the event that the UAT LV breaker fails to interrupt for a LV bus multiphase fault. It also provides back-up protection for transformer internal or LV lead multiphase faults.

ITH relay, high speed overcurrent back-up protection: The current transformers used for the differential relays may saturate for heavy multiple internal faults. To ensure high speed operation for severe saturation, solid state instantaneous overcurrent relays are provided as back-up protection. The ITH relay addition provides back-up protection to mitigate transformer damage due to heavy internal faults.

The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

Modification M01-2-93-027

The modification installed a passive zinc injection skid to inject zinc solution in to the Unit 2 feedwater system. This will help to mitigate the buildup of Cobalt 60 (Co-60) levels in the recirculation system, thereby significantly reducing personnel dose rates, and maintaining exposure ALARA.

UFSAR Section 3.11.1.1 is being revised to reflect the installation of the modification.

The Safety Evaluation concluded that there was not an unreviewed safety question associated with this design change.

Modification M01-2-94-002

This modification improves the reliability and operability of motor operated valve 2E12-F053A. Specific requirements relating to operability were presented in NRC Generic Letter 89-10. Several alternatives were considered to resolve the operability concerns. This modification was selected as the preferred alternative because it has the lowest cost and provides the greatest benefit of improved valve reliability.

This valve was equipped with a Limitorque SMB-3 motor operator having a 60 foot-pound, 3600 rpm (nominal) motor and an OAR of 61.5:1. The operator's overall ratio (OAR) was changed to 57.4:1. This was accomplished by replacing the motor pinion and worm shaft pinion gears. The motor was also replaced with a 100 foot-pound, 3600 rpm unit with the required upgrade of the power cable also being performed.

The gear change slightly reduces the valve stroke time. The original stroke time was slightly less than the Technical Specification limit of 29 seconds. The new calculated stroke time is about 27 seconds in each direction. This stroke time is approximate and may vary due to actual valve stroke and motor speed. The intent of the gear change is to provide additional margin between the actual stroke time and the required stroke time.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

Modification M01-2-94-003

This modification improved the reliability and operability of motor operated valve 2E12-F053B. Specific requirements relating to operability have been presented in NRC Generic Letter 89-10. Several alternatives were considered to resolve the operability concerns. This modification was selected as the preferred alternative because it has the lowest cost and provides the greatest benefit of improved valve reliability.

This valve is equipped with a Limitorque SMB-3 motor operator having a 60 foot-pound, 3600 rpm (nominal) motor and an overall ratio (OAR) of 61.5:1. The operator's OAR was changed to 57.4:1, by replacing the motor pinion and worm shaft pinion gears. The motor was also replaced with a 100 foot-pound, 3600 rpm unit. These changes, in addition to other benefits, increase the operator's gearing capacity.

A 10CFR50.59 safety evaluation concluded that no unreviewed safety question exists as a result of this modification.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Modification M01-2-94-007

This modification addresses NRC Generic Letter 89-10 and General Electric Nuclear Services Information Letter (SIL) #377 concerns related to the Reactor Core Isolation Cooling (RCIC) Steam Supply Stop Valve, 2E51-F045.

This modification replaced the Rockwell globe valve with an Anchor/Darling globe valve having a special contour plug design. This was done to improve the startup performance of the RCIC System.

The modification also increased the valve's overall actuator gear ratio (OAR) from 34.96:1 to 51.80:1. The increased gear reduction will increase the available operator torque and thrust, a concern of the Generic Letter.

The new 2E51-F045 valve will reduce turbine speed overshoot on RCIC startup and it will have more capacity to perform its function, thus making the RCIC system more reliable.

The safety evaluation concluded that there was not an unreviewed safety question associated with this design change.

M01-1-94-006 Unit 1
M01-2-94-008 Unit 2

The modifications revised subsystem A of both units Rod Worth Minimizer (RWM) so that new control rod position information is stored in RWM A memory and displayed on the CRT at control room panels 1H13-P603 and 2H13-P603 after each scan of the Rod Position Information System (RPIS) during a scram. Also, unknown position codes generated by rods which have not activated any of the reed switches on the rod position probe, such as those which have inserted past the full-in reed switches, will not erase previously stored valid position data.

These design changes were installed to resolve a problem with rod position indication following a scram. Following a scram, position indication has been lost in several instances for a few control rods. A preliminary investigation determined that this problem was caused by the control rod inserting past the full-in position indication reed switch. Following the installation of these changes, the last position indication reed switch that is picked up by the control rod is displayed by RWM A during a scram.

A 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Exempt Change E01-0-93-955

The design change installed an interlock between the Reactor Building Crane Radiation Monitor and the Reactor Building Crane Auxiliary Hoist raise control circuit such that a high radiation alarm stops upward motion of the Auxiliary Hoist. This interlock already existed in the Main Hoist circuit, the change used spare contacts on the existing relays.

The bases for this interlock is to stop upward motion of the crane to mitigate the inadvertent lifting of radioactive material from shielding casks or water to prevent excessively high radiation fields on the Refuel Floor that could contribute to exceeding 10CFR20 exposure limits and violating Regulatory Guide 8.8 (ALARA) requirements.

UFSAR Section 12.3.4 and the original specification, J-2532, state that this interlock already exists, however, the interlock was inadvertently omitted from wiring and schematic diagram drawings during initial construction, and was not installed. This change brings the Auxiliary Hoist controls into compliance with the UFSAR.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this activity.

Exempt Change E01-0-94-956

LaSalle Station requires the addition of 16 sand filled 55-gallon drums at various locations in the plant, as security barriers. 15 of these drums are polyethylene (plastic) and positioned on Elevations 710' and 731' within the Turbine Building. These drums will be free standing, as they are located within the non-seismic Turbine Building and do not require restraints.

The one remaining drum is made of steel, and is located on Elevation 710' within the Unit 2 Reactor Building. It is constrained by chaining to a structural concrete column to prevent if from falling over, since it is in a seismic area.

The following Fire Zones are affected: Units 1 & 2, Elevation 710 Turbine Building, Fire Zone 5C11; Unit 1, Elevation 731 Turbine Building, Fire Zone 5B5; Unit 2, Elevation 731 Turbine Building, Fire Zone 5B6; and Unit 2, Elevation 710 Reactor Building, Fire Zone 3G. This zone remains unaffected due to the use of a steel drum.

Appendix H, Section H.3 and Table H.3-2 of the UFSAR are revised due to increases in the total combustible loadings as a result of using polyethylene drums.

A 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this DCP.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Exempt Change E01-1-93-916

Temporary System Change (TSC)# 1-141-93 was completed in January 1993. This TSC installed a seal-in contact in the closing circuit for valve 1B21-RSCV1. This TSC was initiated in order to eliminate the "hammering effect" the valve was experiencing in the closing direction. The "hammering effect" was occurring in closing the valve, as a result of the torque switch contact relaxation when the valve logic receives a continuous close signal. The TSC has corrected the problem, this design change makes this solution permanent.

The design change implements the design of TSC # 1-141-93 with a minor revision. Wired in parallel with the seal-in contact is an IO contact from the limit switch of the valve. The IO contact maintains the throttling capability for the valve in the close direction. When the valve is closing from a full open position, the IO contact is closed with the valve withdrawing from the backseat and will remain so until just prior to the valve full closed position. Just prior tc full closed position, the IO contact will open and the torque switch contact will operate to stop the valve. With relaxation of the torque switch contact the valve will not close any further because the seal-in contact is open inhibiting any "hammering effect" to the valve.

A 10CFR50.59 safety evaluation concluded that no unreviewed safety question exists for this design change.

Exempt Change E01-2-93-902B

This design change revised the 4160/480V transformer taps for the ESF Division 1 unit substations, 235X and 235Y, to boost the secondary voltage by 2.5%. The worl scope consisted of changing the high voltage connections to decrease the primary to secondary transformer turns ratio. Revising the transformer taps boosts the voltages on the 480V buses by approximately 2.5% and, thus, will substantially reduce the number of electrical components with insufficient terminal voltages at the degraded voltage setpoint.
This design change does not change the function, operation, or design basis of the auxiliary power system. In addition, the increase in operating voltage will not decrease the reliability of this system.

This design change does not alter the function of the affected system, result in any unreviewed safety questions, or require a change to the Technical Specification or FSAR. The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist for this design change.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Exempt Change E01-2-93-910

LVDTs (Equipment Part Numbers (EPNs) 2B21-N575A through V) function to give position indication in the control room for Safety Relief Valves EPN # 2B21-F013A through V. For the ease of maintenance, electrical quick disconnects were installed at both the junction box and LVDT ends of the cable for each LVDT.

A 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this design change.

Exempt Change E01-2-93-937

Motor Operated Valve (MOV) 2HG009 is a normally closed containment isolation valve for the hydrogen recombiner system. The exempt change replaced the 2 foot-pound motor with a 5 foot-pound motor, operating at the same nominal speed, to improve valve operability. The safety evaluation concluded that there was not an unreviewed safety question.

Exempt Change E01-2-93-957A Exempt Change E01-2-93-957D
Exempt Change E01-2-93-957B Exempt Change E01-2-93-957E

These Exempt Changes drilled 1/4" vent holes in the reactor-side disc of each of the in the 2E12-F042A/B, 2E21-F005, and 2E22-F004 valves. The vent hole eliminates the possibility of the valves becoming pressure bound while in the closed position. This work was proposed in response to various industry and regulatory notices.

The 10CFCR50.59 safety evaluations concluded that there were no unreviewed safety questions associated with these design changes.

Exempt Change E01-2-94-802B

This design change revised the logic for the diesel generator 2B reverse power circuit to interlock a DG-2B output breaker auxiliary contact. This interlock prevents unnecessary tripping of the DG when the generator is not connected to the bus. In this mode reverse power protection is not required. When the breaker is closed, the function of the reverse power protective circuit remains unchanged. This change also replaced the previous reverse power relay with a GE type GGP-53C, model no. 12GGP53C1A, directional power relay. The previous GE type GGP-53B relay is obsolete. The new relay performs the same function, and meets or exceeds the requirements of the relay it replaced.

G. Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)

Exempt Change E01-2-94-802B
(continued)

UFSAR Table 8.3-2 is revised to list the new interlock for the reverse power relay and to change the relay type. However, instead of revising this table to identify the type of the new relay, the proposed change consists of replacing the relay type with a note that refers the reader to the Q-List to obtain the relay manufacturer and model number.

A 10CFR50.59 Safety Evaluation concluded that no unreviewed safety question exist.

Exempt Change E01-2-94-803

This exempt change installed a 1" drain line approximately 3-1/2 feet long with two isolation valves to the Reactor Core Isolation Cooling (RCIC) System Turbine Exhaust Pot 2RI02B. It also installed 3/4-inch pressure class, socketweld, break flanges in the 3/4" drain piping located immediately downstream of the RCIC System Turbine Exhaust Drain Pot 2RI02B. A new support was welded to existing support RI02-2811X.

A 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

Exempt Change E01-2-94-809A Exempt Change E01-2-94-809C
Exempt Change E01-2-94-809B Exempt Change E01-2-94-809D

These modifications replaced the Unit 2 outboard Main Steam Isolation Valve (MSIV) ASCO pilot solenoid valves (ASCO NP8323A20V) with Valcor Model V70900-87 solenoid valves. This replacement was needed because the ASCO pilot solenoid valves were nearing the end of their qualified service lives. They were replaced with Valcor solenoid valves because the ASCO NP8323A20V pilot solenoid valve is obsolete and is no longer being manufactured by ASCO.

The ASCO NP8323A20V pilot solenoid valve has been attributed to be the cause of failures at LaSalle, Grand Gulf, and Perry. The failure at LaSalle occurred on 12-17-87 and resulted in failure of MSIV 1B21-F028C to close during routine testing. The failure occurred because the ASCO pilot solenoid valve failed to change position when de-energized. ASCO has discontinued manufacturing the NP8323A20V solenoid valve. The Valcor Model V70900-87 was specifically designed to specifications from the R. A. Hiller Company for use as an air pilot control valve for MSIV applications in boiling water reactor plants.

The 10CFR50.59 safety evaluation concluded that no unresolved safety questions exist as a result of these design changes.

G. **<u>Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)</u>**

Exempt Change E01-2-94-934A Exempt Change E01-2-94-934C
Exempt Change E01-2-94-934B Exempt Change E01-2-94-934D

These Exempt Changes replaced the Limitorque operators on valves 2E12-F016A/B and 2E12-F017A/B with SMB-1-25 operators. The larger operators are required to increase the available valve operator thrust. This requirement was the result of motor operator design reviews performed in response to NRC Generic Letter 89-10. The yokes on the valves were replaced to support the additional load. The valves' molded case circuit breakers (MCCBs) and thermal overload relays were also accordingly replaced.

The net effect of the operator change was to increase the valves' stroke time from approximately 75 seconds to approximately 95 seconds. These new stroke times are outside of the standard operating time range for a motor operated gate valve.

A note is being added to UFSAR Table 6.2-21 specifying the approximate operating times for the Unit 2 RHR 16A/B and 17A/B valves.

A 10CFR50.59 Safety Evaluation concluded that no unreviewed safety questions exist as a result of these design changes.

Exempt Change E01-2-94-938

The design change removed air flow elements and installed nozzles in the Primary Containment HVAC System (VP) duct supplying the head area. This change also eliminated the need to seal the manway access hatches to the drywell head area. The purpose of these changes is to reduce the temperature in the upper head area. The temperature in this area has resulted in the upset/shutdown reactor water level indication condensing pot and instrument leg boiling off during shutdown when reactor pressure is below about 10 psig. As a result the indicated reactor water level was actually much lower than indicated.

The low velocity of supply air entering the head area causes stratification, as a result of the size change in the discharge duct fitting increasing from 18" to 24", as it penetrates the head. This results in a 50% reduction in the air velocity. The nozzles that were added increase the velocity of the air entering the head by a factor of 4.

The increase in VP system pressure due to the nozzles will be offset by the removal of the air flow elements and the manway access hatch covers. The net effect on the VP system air flow rate will be so small as to be immeasurable. There will be no effect on fan horsepower or performance.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Exempt Change E01-2-94-938
(continued)

UFSAR Section 6.2.1.2 is being revised to reflect this design change. This section addresses the differential pressure across the refueling bulkhead plate following a line break in the head or in the general area of the drywell. The current analysis in the in the UFSAR assumes that the manway access hatch covers in the bulkhead are in place. A new analysis of the addition of the nozzles on the VP supply ductwork to the head area and removal of the manway access hatches (during power operation) will not cause the differential pressure across the refueling bulkhead plate to exceed existing UFSAR limits. A note is being added to the UFSAR to identify the changes made as part of this Exempt Change, and the note will indicate that the original analysis bounds these changes.

A 10 CFR 50.59 Safety Evaluation concluded that no unreviewed safety question exists due the design change.

Exempt Change E01-2-94-939-E

This modification improves the reliability and operability of motor operated reactor water cleanup (RWCU) return isolation valve 2G33-F040.

The overall actuator gear ratio (OAR) for 2G33-F040 was changed rom 46.80:1 to 82.00:1. This change increases the available valve actuator thrust to assure that the valve can be repositioned under all design bases conditions. This change increased the valve's stroke time from approximately 21 seconds to approximately 39 seconds.

UFSAR Table 6.2-21 is revised to reflect the new stroke time of the valve. A 10CFR50.59 safety evaluation concluded that no unreviewed safety question exists as a result of this modification.

Exempt Change E01-2-94-946B

Circuit breaker type Klockner-Moeller N2MH6-160/ZM6-63/800 for RPS MG Set B in MCC 236X-2, compartment B3 was replaced with circuit breaker type Klockner-Moeller N2MH6-160/ZM6-100/1200, which has higher thermal and magnetic trip ranges. The circuit breaker was replaced because it was tripping during the MG set start. The replacement circuit breaker allows its thermal and magnetic trips to be set at a higher value and their new settings will prevent nuisance tripping of the circuit breaker during MG set start.

The safety evaluation concluded that there was no unreviewed safety question.

G. Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)

Exempt Change E01-2-94-968G

This design change increased the thrust that the actuator delivers to the stem of valve 2E12-F049B by replacing the 2ft-lbf motor with a 5 ft-lbf motor. The new motor operates at the same nominal speed as the old motor (1800 RPM).

NRC Generic Letter 89-10 requires that nuclear plant licensees evaluate the ability of certain motor operated valves to be repositioned when subjected to design basis conditions of flow and differential pressure. The results of the evaluation for valve 2E12-F049B indicate that additional actuator thrust is required to ensure the valve's reliability under design basis flow and differential pressure conditions.

The safety evaluation concluded that there was not an unreviewed safety question associated with this design change.

Exempt Change E01-2-94-987

This Exempt Change replaced the flex-wedge gate valve for MOV 2B21-F016 with a double disc parallel slide gate valve manufactured by the Anchor/Darling Valve Company. The valve was replaced because the hardfacing on the valve seats was almost completely removed by extensive machining. The machining was required to refinish the seats so that local leak rate test (LLRT) acceptance criteria could be met. This valve has frequently failed to meet its LLRT acceptance criteria in the past. The new valve is expected to provide better LLRT performance, while requiring less maintenance than the previous valve.

A vent hole was put in the new valve's reactor side disc. The intent of the vent hole is to eliminate the potential for thermal binding or pressure locking of the valve discs. No provisions were made for this in the valve's procurement specification. With the vent hole installed, the valve will still seal effectively because the vented side of the valve is always exposed to a higher pressure. The higher pressure on the back side of the non-vented valve disc will force the disc into it's seat and provide flow shut-off.

The valve's live loaded spring packages were also changed under the scope of this design change, at the request of the Mechanical Maintenance Department. They have more confidence in the leak tightness capability of the Garlock Set over the live loaded spring packs.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Exempt Change E01-2-9500001

In the previous design, the SSPVs supported one end of the scram air header. This design change installed separate supports to secure the scram air header for 14 banks of hydraulic control units (HCUs). Previously installed copper tubing was replaced with flexible tubing. This eliminated the situation where SSPVs supported one end of the scram air header. This was a change to the air lines providing the source of air from to and from the SSPVs, not the source of the air. The function of the SSPVs and the HCUs was not affected by this design change. UFSAR Appendix H, Fire Hazards Analysis, is being revised to reflect the change in combustible loading resulting from the change.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed question.

Exempt Change E01-2-9500120

The minor plant change added a test tap on line 2FC11C-10", located between primary containment isolation valves 2FC086 and 2FC 115. The test tap was added toallow the station to obtain consistent local leakage rate by only including these two valves in the test boundary. The location of the previously used test tap resulted in several valves being included in the test boundary.

UFSAR Volume XI Piping & Instrumentation Diagram (P&ID) drawing M-144, sheet 1, is revised. Table 6.2-21 is revised to reflect the new containment valve arrangement for the Unit 2 FC System Valves 2FC086 and 2FC115, including the test tap. Figure 6.2-31 (sheet 10c) is revised to reflect the new Containment Valve Arrangement "Detail AD". UFSAR section 6.2.4.2.4 is revised to add a brief description of the containment valve arrangement for the Reactor Well Bulkhead drain piping valves.

The 10CFR 50.59 safety evaluation concluded that there is no unreviewed safety question associated with the design change.

Exempt Change E01-2-9500158

The exempt change removed valve 2E51-F091 (RHR Steam Condensing Mode Warming Valve) and associated 1" piping up to a 10" header. The electrical power and controls associated with Valve 2E51-F091 were also removed. The one inch line and the one inch valve (2E51-F091) provided a bypass flow path around 10" valve 2E51-064 to serve as the steam warming line prior to initiation of the RHR steam condensing mode and is not required for any operating mode. The reason of this change was that the valve 2E51-F091 did not reseat during testing. The steam condensing line was capped near its connections to process header lines 2RI01A-10" and 2RI41A-10".

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Exempt Change E01-2-9500158
(continued)

The RHR Steam Condensing has been eliminated as a mode of operation of RHR system, therefore, removal of warming line will not affect plant operation. The 10" header can still be used for alternate decay heat removal operation.

LaSalle Updated Final Safety Analysis Report (UFSAR) Section 1.2.2.3.4, and Tables 6.2-21 (Sheet 1) were revised to reflect the design change, and incorporated in new Table 6.2-28 (Sheet 2).

The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety question associated with this design change.

Minor Plant Change P01-1-92-524

The minor change removed the spring and disk from check valve 1DG036. The valve is downstream of the Unit 1 LPCS pump motor cooler cooling water line. The valve body was left installed in the piping system. The design change was in response to valve corrosion concerns. The disk and spring were replaced during L1R04, due to corrosion discovered during an inspection. The permanent removal of the valve spring and disk removes the funtional capability of 1DG036 to perform as a check valve. The design change eliminates future corrosion concerns associated with the valve. An engineering review determined that the chekc valve is not required in the piping system.

The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

Minor Plant Change P01-1-93-501

The minor change allows the 1B Diesel Generator (DG) to be shutdown using an Energize-to-shutdown solenoid configuration, as opposed to the previous Energize-to-run configuration. The change also revises the shutdown control logic of the 1B DG which previously allowed the DG to be shutdown with either the Manual Control Switch/Pushbutton or Emergency Stop Pushbutton, whether a LOCA signal is present or not. The revised control logic allows the 1B DG to be shutdown with the Manual Control Switch/Pushbutton only when a LOCA signal is not present and would leave the Emergency Stop Pushbutton as currently configured. This change makes the 1B DG consistent with the logic currently configured on DG 0, 1A, and 2A.

The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

Minor Plant Change P01-2-92-506

This design change replaced the motor operator on valve 2E51-F013, in order to increase the thrust available to operate the valve. The minor plant change addresses concerns raised by NRC Generic Letter 89-10 and related documents. The Limitorque SMB-1-60 operator replaces the previous SMB-00-25 operator.

An addendum to this design change consisted of drilling a 1/4" vent hole in the valve's reactor side disc. The vent hole is intended to eliminate the potential for pressure binding. Pressure binding may occur in the event that the valve is closed while the bonnet is full of relatively cool water and is subsequently heated.

The 10CFCR50.59 safety evaluations concluded that there were no unreviewed safety questions associated with this design.

Minor Plant Change P01-2-92-523

The minor change removed the spring and disk from check valve 2DG036. The valve is downstream of the Unit 2 LPCS pump motor cooler cooling water line. The valve body was left installed in the piping system. The design change was in response to valve corrosion concerns. The disk and spring were replaced during L2R04, due to corrosion discovered during an inspection. The permanent removal of the valve spring and disk removes the functional capability of 2DG036 to perform as a check valve. The design change eliminates future corrosion concerns associated with the valve. An engineering review determined that the check valve is not required in the piping system.

The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

Component Replacements C01-2-94-032 and C01-2-94-033

These design changes replaced relays 2B21A-K710AX and 2B21A-K710CX, and 2B21A-K710BX and 2B21A-K710DX, respectively. The Agastat model TR-14B3A replacement time delay relays that replaced the Agastat model EGPB control relays were purchased as commercial grade and upgraded for safety related use.

These design changes do not change the function, operation, or design basis of the Auxiliary Power System, RCIC System, or MSIV Leakage Control System as described in the Technical Specifications or UFSAR.

UFSAR Section 7.4.1.2.3 is being revised to reflect that the time delay is now applicable to both units. The current revision of the UFSAR reflects the time delay as being applicable to Unit 1 only.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Setpoint Changes S01-1-92-024, S01-1-92-025 and S01-2-92-030

The design changes lowered the setpoint for these parameters to more realistic values to incorporate NRR approved Technical Specification amendments 105(Unit 1) and 91(Unit 2). The previous setpoints were found to be higher than needed for verifying that the systems were full of water. For the HPCS and RCIC systems, the setpoints resulted in unnecessary activation of the alarms when associated water leg pumps are aligned to take suction from the suppression pool.

These changes did not affect the UFSAR since neither the functions nor descriptions of the ECCS and RCIC discharge line "keep filled" alarm instrumentation channels as described in that document were changed.

The 10CFR50.59 safety evaluations concluded that there were no unreviewed safety questions associated with these design changes.

Setpoint Change S01-1-94-036

UFSAR Section 11.2.2 was revised to delete reference to the radwaste demineralizer effluent conductivity. This allows future demineralizer effluent conductivity monitors setpoint changes to be made without having to revise the UFSAR. Changes are required to UFSAR pages 11.2-9 and 11.2-10.

Setpoint Changes S01-1-94-038 and S01-2-94-037

These setpoint changes increased the calibration trip setpoint for Reactor Core Isolation Core turbine exhaust pressure switches 1E51-N009A and B and 2E51-N009A and B, respectively, from 25.0 psig to 43.7 psig plus 2.3 psig head correction, increasing.

These setpoint changes did not require the replacement of the pressure switches since the new setpoints are within their adjustable range. These changes also did not involve any hardware, wiring, or cable changes.

The purpose of these setpoint changes is to provide sufficient margin between the calibration setpoint and the postulated RCIC turbine backpressure of 24.5 psig at 4 hours and 15 minutes following a Station Blackout (SBO) to account for all known instrument errors. Another purpose is to increase the availability of RCIC following a LOCA by increasing the setpoint as close as possible to the upper analytical limit which was determined to be 50.0 psig.

UFSAR Table 7.4-1, and Sections 15.9.3 and 15.9.4 are being revised to reflect design changes. The 10CFR50.59 safety evaluation concluded that no unreviewed safety questions exist.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Setpoint Changes S01-1-9400446 and S01-2-9400447

The lowering of the interlock allows the RR pumps to be upshifted or downshifted at a lower power. This may result in a slight increase in time in which the RR pumps are in fast speed and the FCV in a reduced position (below 70 Mlb/hr). While this is not a cavitation concern, this is a known point for increased RR pump vibrations.

By allowing RR pump speed changes at lower feedwater flows, the margin from the region of instability can be increased during normal plant evolutions. Since RR pump speed changes can be performed at a lower Flow Control Line (FCL), the overall power change during the evolution can be reduced. Thus, the transient on the plant, (feedwater control, heater response, etc.) is reduced.

UFSAR Sections G.2.1, G.2.3, G.2.5, G.3.3, G.5.2 and G.7 were revised to reflect the setpoint changes. The safety evaluations concluded that there were no unreviewed safety questions.

Setpoint Change S01-1-92-026 Setpoint Change S01-2-92-032
Setpoint Change S01-1-92-027 Setpoint Change S01-2-92-033
Setpoint Change S01-1-92-028 Setpoint Change S01-2-92-034
Setpoint Change S01-1-92-029 Setpoint Change S01-2-92-035
Setpoint Change S01-2-92-031

These design changes lowered the setpoint for the ECCS and RCIC Discharge Line "Keep Filled" Pressure Alarm parameters to more realistic values to incorporate NRR approved Technical Specification amendments 105 (unit 1) and 91 (Unit 2).

The previous setpoints were found to be higher than needed for verifying that the systems were full of water. For the LPCI RCIC, HPCS, LPCS, systems, the setpoints resulted in unnecessary activation of the alarms when associated water leg pumps are aligned to take suction from the suppression pool.

These changes did not affect the UFSAR since neither the functions nor descriptions of the ECCS and RCIC discharge line "keep filled" alarm instrumentation channels as described in that document were changed.

The 10CFR50.59 safety evaluations concluded that there were no unreviewed safety questions associated with these design changes.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Temporary System Change (TSC) 1-0033-95

The Unit 1 A Reactor Recirculation (RR) Flow Control Valve (FCV) actuator drain alarm was up solid. This situation was a nuisance, and masked other valid alarms. The TSC bypassed the control room alarm allowing other hydraulic power unit alarm signals to annunciate in the control room.

Actuator drainage was verified by system engineers weekly rounds and by periodically performing LOS-RR-SR3, Reactor Recirc FCV Actuator Leakoff Line Flow Rate Test.

The safety evaluation concluded that there was not an unreviewed safety question.

Temporary System Change (TSC) 2-0017-95

The TSC consisted of adding test equipment to monitor Suppression Pool temperature element loop 2TE-CM057A-2. This was done to determine the cause of erratic output from element 2. Non-safety related and non-EQ test equipment was connected to safety related and EQ components. System electrical segregation was not affected because an isolation transformer was used to power the test equipment.

The safety evaluation concluded that there was no unreviewed safety question associated with the TSC.

Temporary System Change (TSC) 2-0062-95

This Temporary System Change allowed fire detection zones 2-16, 2-16P, 2-30, 2-31, 2-32, 2-33 to be taken OOS using switch jumpers which are operated by using keys controlled by the Operating department. Detectors in above areas were taken OOS whenever welding, cutting or grinding was performed in these areas to prevent false alarms at the fire protection panel 2FP04JA. The position of these switches was controlled administratively via the equipment out of service procedure LAP-900-4, and a fire impairment was written in accordance with LAP-900-16 whenever a detection zone was taken OOS. The fire impairment specified and established required fire watches and Out of Services for affected fire zones.

The operating modes of subject detection zones were controlled administratively via procedure LAP-900-4 (Equipment Out Of Service) and LAP-900-16 (Fire Protection Impairment).

The safety evaluation concluded that there was no unreviewed safety question associated with this change.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Temporary Alteration (TALT) 2-186-95

This Temporary Alteration was initiated to bypass the alarms when they were up solid. This was due to the alarms masking other alarms. The TALT was accomplished by installing a jumper across flow switch 2B33-N097 ("A" alarm) and 2B33-N908 ("B" alarm). The jumpers were installed as needed whenever an alarm was up solid. This TALT allows other alarm signals to annunciate in the control room.

Actuator drainage, which occurs slowly over time, was monitored by system engineer and operating personnel. The unit would be shutdown before this degradation affected control of the FCV.

Degradation resulting in unreliable FCV control would require that the unit be shut down, to allow the actuator and switch to be repaired.

The 10CFR50.59 safety evaluation concluded that there were no unreviewed safety questions associated with the TALT.

LaSalle County Station Units 1 & 2 Reclassification of Components Associated With Residual Heat Removal Steam Condensing Mode of Operation

LaSalle County Station has eliminated the Steam Condensing Mode of Operation of the Residual Heat Removal (RHR) system. As a result of this elimination, several components previously operated only for the Steam Condensing Mode have now become passive in their safety function or non-safety related.

Passive valves do not require quarterly stroke time testing in accordance with the American Society of Mechanical Engineers (ASME) Section XI In-Service Testing (IST) Program, and components no longer important to safety can be removed from the 10CFR50.49 Environmental Qualification (EQ) Program, thereby, removing stringent procurement and maintenance requirements.

By formally changing the Safety Classification of these components, the cost of procurement and maintenance for these components can be dramatically reduced, resulting in significant savings to Commonwealth Edison Company. Associated RHR Steam Condensing Mode Motor Operated Valves (MOV) were removed from the Generic Letter (GL) 89-10 MOV Testing Program.

UFSAR Sections 5.2, 5.4, and Appendix L is being revised to reflect the reclassification of components.

The safety evaluation concluded that there was no unreviewed safety question associated with the reclassification.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

UFSAR Revision to Reflect the New Edition of 10CFR20 and the Facility Improvement Project (FIP)

UFSAR Chapter 12, Radiation Protection, and Appendix I of the UFSAR are being revised to incorporate the current revision of 10CFR20, and the Facility Improvement Project. The update of Chapter 12 also reflects the present structure of both the Radiation Protection and Chemistry Departments.

The safety evaluation concluded that there was no unreviewed safety question associated with this change to the UFSAR.

Revision to UFSAR Figure 5.4-3, Reactor Core Isolation Cooling System Process Diagram

UFSAR Figure 5.4-3 Revision 9 - April 1993 improperly shows the E51-F064 valve in series with the reactor core isolation system (RI, RCIC) turbine and fails to show the E51-F008 valve. The change will make this diagram agree with the controlled station drawings and the actual configuration.

This change will correct this UFSAR figure to properly match the as designed, as built configuration of this system. No physical plant changes are occurring. No procedures or analyses are changed, since the existing controlled P & ID's, other UFSAR sections and drawings accurately depict the system configuration.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this activity.

Revision to UFSAR Section 5.4.8.2 and Figure 5.4-7

Delete Table III and V (a list of operating parameters) and the referenced notes from UFSAR Figure 5.4-7. Replace with a note to Figure 5.4-7, "For Reactor Water Cleanup System operating values, see current station procedures.". This revision permits changes to the air and water flow rates during a RWCU filter backwash or precoat, without the administrative burden associated with updating the UFSAR. In section 5.4.8.2 (System Description), the reference to Table 5.4-2 is delete, since there is no Table 5.4-2.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this activity.

G. **Summary of Changes to t[illegible] Facility Which are Described in the Safety Analysis Repor[illegible]inued)**

UFSAR U[illegible]e for fixed air sampling
panels and sampling areas.

UFSAR Section 12.3.4.4 and Table 12.3-16 is revised to reflect actual plant configuration of continuous air monitors (CAMs) sample points and locations.

The safety evaluation concluded that there was no unreviewed safety question.

Replacement of Valve 1E51-F357 in Reactor Core
Isolation Cooling (RCIC) Steam Drain line with
Blind Flanges and Section of Piping.

Drain line 1RI09A-2" from the RCIC Steam Supply line (line no. 1MS06A-10") contained manual valve 1E51-F357. This valve was normally closed, and indicated as locked closed on the piping and instrumentation drawing. This change removed this valve from the drain line, and installed a short section of pipe with blind flanges welded at each end in its place.

UFSAR Table 6.2-21, "Summary of Lines Penetrating the Primary Containment", was revised to reflect this activity.

The safety evaluation concluded that there was no unreviewed safety question associated with the change.

UFSAR update to Section 9.3.5 to
reflect valve setpoint change

The standby liquid control (SBLC) Pump discharge relief valves, 1(2)C41-F029A/B, are tested each refuel outage to meet Technical Specification Surveillance requirements for the SBLC system. The valves were previously required to be set and verified at less than or equal to 1400 psig and to verify that the relief valves do not actuate during recirculation to the test tank.

An investigation determined an inconsistency between the Vendor setpoint tolerance to that which was previously specified in the station procedure for testing and adjusting the relief valve setpoint.

A setpoint which is 75 psi above the pressure that is required for SBLC Pump injection was the required criteria. SBLC is required to supply 41.2 gpm at 1220 psig to meet its functional requirements. General Electric recommended that the setpoint be greater than 1295 psig.

The Vendor (Crosby) stated that it was acceptable to adjust the valve relief setting within the range of 1260 psig to 1499 psig with the current spring in each relief valve.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

UFSAR update to Section 9.3.5 to reflect valve setpoint change (continued)

The relief valve setpoint was changed to 1340 ± 40 psig, which provides margin to the upper and lower limits when testing the relief valves.

UFSAR Section 9.3.5 was revised to reflect the SBLC pump discharge relief valves 1(2)C41-F029A/B setpoints being changed from 1400 psig to 1340 ± 40 psig.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

UFSAR update to reflect the deletion of the Rod Sequence Control System (RSCS)

The RSCS is no longer in Technical Specifications as a result of the Rod Worth Minimizer (RWM) Low Power Setpoint (LPSP) being lowered to 10% rated core thermal power. UFSAR Section 7.2.1 is revised to reflect the deletion of the reference to RSCS.

The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety evaluation associated with the procedure revision.

Non-Fire Rated Penetrations in the Fire Barrier Separating Fire Zones 5B2 and 3K.

The existing fire rated penetrations in the boundary wall between fire zones 5B2 and 3K at elevation 687 feet-0 inch to the floor slab of elevation 710 feet-6 inch, and from column lines 18.7 to 21 are reclassified as non-fire rated penetrations.

The boundary wall which separates fire zones 5B2 and 3K at elevation 687 feet - 0 inch to the floor slab of elevation 710 feet - 6 inch does not separate fire zones containing redundant safe shutdown equipment.

The Safe Shutdown Analysis (UFSAR Appendix H, Section H.4) demonstrates that for a fire in any single plant fire area/zone, at least one method exists to achieve and maintain a safe shutdown condition independent of that fire area/zone.

The safety evaluation concluded that there was not an unreviewed safety question.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Evaluation of Turbine Building Flooding

An Operability Assessment and Safety Evaluation was prepared in response to a flooding concern associated with the potential failure of circulating water and service water piping, which could result in a gravity fed flood from the perched cooling lake, installed outside of the flood protected areas as detailed in the SAR.

Calculations determined that piping with no isolation valves in the flood protected zones had no postulated leakage cracks and based on bounding flood rates, sufficient time existed for manual actions to isolate any of the identified lines using valves installed outside of the flood affected area. Therefore, it was concluded that the identified piping installed outside of the flood protected areas does not result in an operability concern.

The safety evaluation concluded that there are no unreviewed safety questions involving the current piping configuration, based on the analytical qualification of the major piping to show that a moderate energy line break is not a credible event, and, even if a break was postulated, a sufficient time frame exists to take actions to isolate the affected piping from the lake prior to the flood challenging any equipment important to safety.

MCPR/EOOS Bases Change, Revision of the Administrative Technical Requirements to reflect the U2C7 Reload and the Implementation of ARTS and EOOS Revisions to the UFSAR.

Technical Specifications Bases Section 3/4.2.3 (Minimum Critical Power Ratio) was changed to delete specific information regarding LaSalle's analyzed EOOS combinations. Instead, the Core Operating Limits Reports, which contain the needed information is referenced.

The Administrative Technical Requirements were revised to incorporate the power and flow dependent ARTS thermal limits and reference a new addition to LaSalle's EOOS analyses.

Sections 4.4.1.3, 7.3, 7.6, 15.A.5 and 15.A.6 of the UFSAR were revised to describe the new ARTS power and flow dependent thermal limits and the addition to LaSalle's EOOS analyses described above.

TS Bases Section 3/4.2.3 was changed to update the Bases with regards to currently allowed EOOs. The Administrative Technical Requirements (ARTs) were revised due to approval of the ARTS amendment by the USNRC and due to L2C7 startup. The UFSAR was changed to reflect the ARTS amendment and the new EOOS combinations allowed.

The safety evaluation concluded that there was not an unreviewed safety question associated with these activities.

G. Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)

Relocation of Technical Specifications Table 3.6.3-1, "Primary Containment Isolation Valves" to the UFSAR

Technical Specification Amendments 102 and 87 (for Unit 1 and 2, respectively), deleted Table 3.6.3-1, "Primary Containment Isolation Valves". Specification 3/4.6.3 was also changed by the amendments to refer to the required components (valves) by description instead of by use of a list of valves. The Unit 1 and 2 tables were added to Administrative Technical Requirements (ATRs) to provide a ready reference of the components/devices that are covered by Specification 3/4.6.3. ATR Table 3.6.3-1 (Unit 1) and Table 3.6.3-2 (Unit 2) have minor format/editorial changes, which do not affect the content of the tables. Additional valves are added from UFSAR Table 6.2-21 to make the list of components required to be Operable per Technical Specification 3/4.6.3 complete. The new ATR tables are also combined and added to the UFSAR as Table 6.2-28.

The safety evaluation concluded that there was no unreviewed safety question.

Drywell Floor Drain and Equipment Drain Sumps - Operability Assessment

An assessment was performed in response to discovery of the cover plates for the drywell floor drain and equipment drain sump in the pedestal area being unfastened. The design drawings indicate that these plates should be bolted in place and the joints sealed with caulking material.

A calculation performed to evaluate the response of the unfastened plates determined that these plates will not significantly move during the occurrence of any SAR accident scenario. The unbolted condition was found to be acceptable.

A 10CFR50.59 Safety Evaluation concluded that there are no unreviewed safety questions involving the sump cover plate configuration.

Update of the functional requirements of the alternate rod insertion (ARI) system in the UFSAR

The time allowed for the start of the insertion of all control rods after receipt of an initiation signal was increased from 15 seconds to 35 seconds; the maximum time delay between the receipt of an initiation signal and the time when all control rods reach the full-in position was increased from 25 seconds to 45 seconds.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Update of the functional requirements of the alternate rod insertion (ARI) system in the UFSAR (continued)

The time requirements for the ARI system are based upon the time needed to fill the scram discharge volume. The original times proposed for the start of motion (15 seconds) and the time to full insertion (25 seconds) are very conservative. Further analysis has determined that the time for all control rods to start inserting should be 35 seconds, while the time for all control rods to be fully inserted should be 45 seconds.

When ARI was installed, a generic evaluation was performed which assumed the volume of the scram discharge piping would be the same for all BWRs. LaSalle's scram discharge volume is larger than that assumed in the evaluation. Therefore, the maximum allowable time delay between receipt of an initiation signal and when all control rods begin to insert is to be changed. Also, the maximum allowable time between the initiation of ARI and all control rods reaching their full-in position is changed.

LaSalle's UFSAR previously stated that for the ARI function, the control rods must start to insert within 15 seconds from the receipt of the initiation signal. It further stated that all control rods must be fully inserted within 25 seconds from the receipt of the initiation signal. These values are changed to 35 seconds and 45 seconds, respectively. UFSAR Section 7.6.5 is revised to reflect the updated functional requirements of ARI. This change is based upon a realistic analysis of the time required to fill the scram discharge volume. Interaction between the ARI system and the control rod drive system will remain unchanged. The ARI system will continue to provide a means of mitigating an Anticipated Transient Without Scram (ATWS).

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with this change.

Bypassed Reactor Water Cleanup (RWCU) Filter Inlet Temperature Element, 2TE-G33-N007

The RWCU Filter Inlet Temperature Element was removed for repairs. The RWCU Filter Inlet Temperature Hi-Hi function was bypassed because of this. An operability assessment was performed to allow for start up of Unit 2 with this function bypassed.

Bypassed Reactor Water Cleanup (RWCU) Filter
Inlet Temperature Element, 2TE-G33-N007
(continued)

Element 2TE-G33-N007 measures the temperature of the water entering the filter demineralizers. The RWCU filter inlet temperature can be monitored by the main control room operator from temperature indicator 2G33-R607, located on panel 2H13-P602. This instrument provides indication only. As a compensatory action, a caution card was placed on annunciator window 2H13-P602, A303, "RWCU Filter Inlet Temp Hi-Hi" during the bypassing of the element.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

Reactor Vessel Bottom Head Drain Line
Effects on LOCA Analyses

An evaluation was performed in response to discovering that per operating procedures, the reactor vessel bottom head drain valve G33-F101, is normally open, while the UFSAR shows the G33-F101 valve as being closed during routine operations.

This evaluation consisted of determining if this placed the station in an unanalyzed condition. The evaluation concluded that this situation was acceptable, based on the water loss through the bottom head drain line with valves G33-F101 and G33-F103 open has minimal impact, and is bounded by analysis.

UFSAR Figure 5.4-6, sheets 1 and 2 are being revised to document the operating mode for motor operated valve G33-F101 and the fluid flow rates through the associated bottom head drain line.

UFSAR Figure 6.2-1 is being revised to incorporate the reactor bottom head drain line configuration.

A 10CFR50.59 safety evaluation concluded that there is no unreviewed safety question associated with this issue.

UFSAR Revisions per impending Diesel
Generator Design Changes

Component Replacement C01-2-94-006-E will replace the Basler type PRC-130 differential relay with a Westinghouse type SA-1 differential relay in Diesel Generator (DG) 2A.

Component Replacements CR 94-006A (DG-2A), CR 94-007A (DG-0), and CR 94-007B (DG-1A) will replace the underfrequency trip relays 1481-DG007, 1481-DG014, and 2481-DG014 in the DG-0, 1A, and 2A shutdown control circuits.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

UFSAR Revisions per impending Diesel
Generator Design Changes
(continued)

Component Replacement No. C01-2-94-006D will replace the Diesel Generator 2A Overcurrent Basler type PRC-110-108 relays (which are obsolete) withABB/Westinghouse type CO-6, model no. 264C898A05 relays.

Exempt Change E01-2-94-960E will replace the Diesel Generator 2A Neutral Ground Relay, Basler type PRV-120 overvoltage relay (which is obsolete), with GE type IAV, model no. 12IAV51D1A.

Exempt Change E01-2-94-960F will replace the Diesel Generator 2A Basler type PRP-110 Reverse Power Relay (which is obsolete), with a GE type ICW, model no. 12ICW51A2A, directional power relay.

UFSAR Table 8.3-2 specifies the type of differential relay used for DG-2A. However, instead of revising this table to identify the type of replacement relays to be installed, the table will be revised by replacing the relay type with a note that refers the reader to the Q-List to obtain the relay manufacturer and model number.

The 10CFR50.59 safety evaluations concluded that there were no unreviewed safety questions.

Continued operations with the high pressure steam
supply not available to one or both Turbine Driven
Reactor Feed Pumps (TDRFP)

Continued operations with the high pressure steam supply not available to either TDRFP due to the unavailability of the TDRFP high pressure steam supply was evaluated. This was due to operational considerations which may require that this supply be isolated.

While unavailability of the high pressure steam supply limits the maximum flow available of the TDRFP, the TDRFP can meet all design requirements, other than being able to support plant startup.

The unavailability of this source does not affect adjacent equipment. The unavailability of this source doesn't affect the control system. The logic inputs for the steam supplies are parallel such that either must be available and the steam supply motor operated valves (MOVs) do not have logic inputs to feedwater control.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Continued operations with the high pressure steam supply not available to one or both Turbine Driven Reactor Feed Pumps (TDRFP) (continued)

As discussed in UFSAR Section 10.3, this source is not required for the feedwater system to meet its design basis, except for the option of being able to start up on a TDRFP. The system as described in UFSAR sections 10.3 and 10.4 has the high pressure steam supply available for operation of a TDRFP prior to the main turbine being place on line. There are no procedural actions are affected by the unavailability of this steam source, since the use of the TDRFP high pressure steam supply for plant startup is not addressed in station operating procedures.

The safety evaluation concluded that there were no unreviewed safety quesitons associated with the evaluation.

LaSalle Station Dry Active Waste (DAW) Storage Facility for Low Level DAW

The evaluation was performed to evaluate the storage of DAW in an on-site warehouse building. The issues of container strength and durability, pallet strength and durability, direct and skyshine radiation levels at the nearest restricted area boundary and the nearest off-site receptor, fire protection, and accident scenarios were evaluated to determine the acceptability of an on-site DAW storage facility.

The evaluation concluded that an unreviewed safety does not exist.

Special Operating Order 95-29, Provision for Locking the Position of the Reactor Recirculation (RR) Flow Control Valves (FCV)

The operating order provided instructions for dealing with the possibility of the 2A RR FCV oscillating while moving in the open direction, such as during an increase in reactor power. The order allowed for locking the position of the FCV if valve oscillations were to occur.

Locking the FCV position gives operators a means to prevent reactor power spikes which could be caused by FCV position instability. Other instructions in the order are controlled in accordance with station operating procedures.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question.

G. **Summary of Changes to the Facility Which are Described in the Safety Analysis Report-(continued)**

Unit 1 Reactor Recirculation (RR) delta temperature (dT) Resistance Temperature Detector (RTD) failure

On 9/27/95, RTD element #2 of Reactor Recirculation (RR) pump 1B suction temperature instrument 1B33-N028B was found to have failed. The element is part of the RR delta temperature downshift logic, and in its failed condition would prevent the 1B 10.1 dT logic from tripping both RR pumps to slow speed. The circuit is arranged such that both channels of the B RR suction dT logic must sense a 10.1 F dT to cause a downshift. The failed RTD does not feed the dT indicator in the Auxiliary Electric Equipment Room (AEER), therefore the B RR dT indication remains valid. The redundant trip system based on the A RR suction temperature remains available to downshift both RR pumps if the dT reaches 10.1 F. Based on this condition. The following recommendations were made: The unit may be placed in power operation with the pumps in high speed without restrictions; The circuit should be left in the normal, unbypassed condition; and, Once every two weeks the proper function of the 1B33-N028A element #2 RTD which does not feed the dT indication be verified by measuring the resistance of the RTD.

The 10CFR50.59 safety evaluation concluded that there was no unreviewed safety question associated with either the RTD failure or power operation of the unit with the pumps in high speed without restrictions.

Revision to UFSAR Section 7.8

UFSAR Section 7.8 is revised to reflect an update to the Safety Paramater Display System (SPDS) shown in Figure 7.8-1, SPDS Primary Display. This is due the installation of a new process computer graphics system. There is also revision to the text of Section 7.8.2.2 associated with this. The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety question associated with the change.

Changes to UFSAR Sections 12.5.3, 13.5, and Appendix B to reflect Revision 0 of the LaSalle Procedures and Surveillance Writers Guide

UFSAR Sections 12.5, 13.5, Tables 13.5-1, 13.5-2, and 13.5-4, and Appendix B are revised to reflect the publishing of the Writers Guide.

The 10CFR50.59 safety evaluation concluded that there is no unreviewed safety question associated with this activity.

H. **Survey of Evaluation Results of Chlorine Shipments by Barge on the Illinois River**

This survey was not required to be performed for 1995, the survey was last completed and is included with the NRC Annual Report for the year of 1994.

I. **Summary of Events Violating Technical Specification 3.4.5 Primary Coolant Iodine Spiking Exceeding Allowable Limits**

During this reporting period, January 1, 1995 through December 31, 1995, there were no violations of Technical Specfication 3.4.5, Primary Coolant Iodine Spikes Exceeding Allowable Limits.

ATTACHMENT A
SAFETY-RELATED MAINTENANCE COMPLETED
(NON-OUTAGE RELATED)

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950038533-01	AP	1AP000	INSPECT HEA LOCKOUT RELAYS.
950038535-01	AP	1AP000	INSPECT HEA LOCKOUT RELAYS.
950038538-01	AP	1AP000	INSPECT HEA LOCKOUT RELAYS.
950038539-01	AP	1AP000	INSPECT HEA LOCKOUT RELAYS.
950038540-01	AP	1AP000	INSPECT HEA LOCKOUT RELAYS.
930046982-05	AP	1AP04E-3	INSTALL BANANA JACKS.
930047382-01	AP	1AP20E-203C	INSPECT BREAKER.
950112872-01	AP	1AP21E-303B	REPLACE DEFECTIVE RMS-9 UNIT.
950063509-01	AP	1AP21E-303D	REPLACE BREAKER.
950063513-01	AP	1AP22E-404A	REPLACE BREAKER.
940058098-01	AP	1AP75E-C6	REPLACE OVERLOAD RESET PUSH BUTTON.
950117499-01	AP	1AP76E-D5	RCIC WATER LEG PUMP TRIPS BREAKER.
950018769-02	AP	1AP78E-D5	PERFORM MCC CUBICLE INSPECTION.
950096900-01	AP	1AP78E-F1	'A' PHASE FAILED TRIP TEST.
950059602-01	AP	1AP81E-B4	BREAKER FAILED TRIP TEST.
950032658-01	AP	2AP000	REPAIR HOLE IN NON-SEG BUS DUCT.
950009095-01	AP	2AP04E	INSTALL BANANA JACKS.
950026239-01	AP	2AP04E-01	REPAIR CRACKED ARC CHUTE.
940080389-02	AP	2AP04E-2	INSPECT ITE CIRCUIT BREAKERS.
920046985-04	AP	2AP04E-2-C	REPLACE TIMERS AND TEST.
940080411-02	AP	2AP04E-4	INSPECT ITE CIRCUIT BREAKERS.
950099566-01	AP	2AP04E-7	REPLACE PUMP 'TSC' SWITCH.
950026655-01	AP	2AP08E-2	REPLACE DEFECTIVE LUG FOUND.
950032397-02	AP	2AP19E-102A	INSPECT LOW VOLTAGE BREAKERS.
940060582-01	AP	2AP20E	REPLACE BREAKER.
940060583-01	AP	2AP20E	REPLACE BREAKER.
950071362-01	AP	2AP71E-C5	BREAKER TRIPS THERMALS.
940087589-02	AP	2AP72E-AA4	PERFORM MCC CUBICLE INSPECTION.
950019654-01	AP	2AP75E-A1	REPLACE CONTACTOR.
950013821-01	AP	2AP76E-E2	OVERLOAD WOULD NOT RESET.
950032397-04	AP	2AP76E-E3	INSPECT LOW VOLTAGE BREAKERS.
950032397-03	AP	2AP76E-E4	INSPECT LOW VOLTAGE BREAKERS.
950118149-01	AP	2AP78E-A1	TRIP TEST BREAKER.
940087569-02	AP	2AP83E-C4	PERFORM MCC CUBICLE INSPECTION.
950081511-01	B21	1PS-B21N560U	REPLACE PRESSURE SWITCH.
940061618-03	B21	2B21F028A	REPLACE ASCO VALVES WITH VALCOR.
940061619-03	B21	2B21F028B	REPLACE ASCO VALVES WITH VALCOR.
940061621-03	B21	2B21F028D	REPLACE ASCO VALVES WITH VALCOR.
950040842-01	B21	2B21F571	INSPECT CHECK VALVE.
940092472-01	B21	2LIS-B21N702D	POTENTIOMETER IS INTERMITTENT.
950098369-01	B21	2LR-B21R884B	REPLACE DISPLAY, MISSING SEGMENTS.
950055138-01	B21	2LR-B21R884B	RECORDER PEN DOES NOT OPERATE.
950081823-01	B21	2PS-B21N560E	REPLACE PRESSURE SWITCH.
950033931-01	B21	2PS-B21N561D	LOW PRESSURE ALARM IS UP.
950096741-01	B33	1FT-B33N024A	TRANSMITTER SPIKED DOWNSCALE.
950058858-01	B33	2EY-B33K610C	FLOW CONVERTER FAILED LOW.
950011966-01	B33	2FY-B33K606B	NO METER RESPONSE.
940079684-01	CM	1AIR-CM012	REPLACE DISPLAY.
950057432-01	CM	1AIR-CM018	INDICATION IS ERRATIC.
950092407-01	CM	1LY-CM030A	COMPUTER ALARM COMES IN EARLY.
950062869-01	CM	1PY-CM056	CONTROLLER READS HIGHER THAN NORMAL.
950106502-01	CM	1RE-CM017	REPAIR CABLES AND CONNECTORS.
950027653-01	CM	1RIT-CM011	NO INDICATION.
950066796-01	CM	1TR-CM038A	PEN IS INOPERABLE.
950075401-01	CM	1TR-CM038B	PEN IS STICKING.
950100642-01	CM	1TR-CM038B	PEN 2 IS READING HIGH.

ATTACHMENT A

SAFETY-RELATED MAINTENANCE COMPLETED

(NON-OUTAGE RELATED)

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950106216-01	CM	2AIR-CM018	REPLACE DISPLAY.
950106217-01	CM	2AIR-CM047	REPLACE DISPLAY.
950106214-01	CM	2RR-CM017	REPLACE DISPLAY.
950004546-01	CM	2TI-CM037	INDICATION OUT OF TOLERANCE.
950077075-01	CM	2TR-CM037A	RECORDER RED PEN IS STICKING.
950094928-01	CM	2TR-CM038A	RECORDER PEN DOES NOT RESPOND.
950107801-01	C11	1C11D001011-111	REPLACE 111 VALVE ON HCU 10-51.
950009939-01	C11	1C11D001115-SRI-A	REPLACE TEST SWITCH A & B.
950106000-01	C11	1C11D001117	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 34-39,34-55, 22-15 AND 02-39.
940079614-01	C11	1C11000	REPLACE THE DIAPHRAGMS IN THE 117 AND 118 VALVES.
950057779-01	C11	1C11000	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 14-23,46-39,50-15 AND 10-47.
950057778-01	C11	1C11000	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 18-31,42-31,34-15 AND 26-47.
950057782-01	C11	1C11000	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 22-23,38-39,42-15 AND 18-47.
950057780-01	C11	1C11000	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 26-15,34-47,46-23 AND 14-39.
950057783-01	C11	1C11000	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 42-47,18-15,38-23 AND 22-39.
950057743-01	C11	1C11000	REPLACE THE 117 AND 118 SSPV'S ON HCU'S 22-07,30-55,02-31 AND 58-31.
940061246-01	C11	2C11D001002	REPLACE HCU 22-59 WATER ACCUMULATOR.
940061258-01	C11	2C11D001005	REPLACE HCU 22-55 WATER ACCUMULATOR.
940061247-01	C11	2C11D001021	REPLACE HCU 18-43 WATER ACCUMULATOR.
940062012-01	C11	2C11D001044	AIR LEAK AT CHARGING NIPPLE.
940061254-01	C11	2C11D001060	REPLACE HCU 10-23 WATER ACCUMULATOR.
940061253-02	C11	2C11D001079	REPLACE HCU 26-11 WATER ACCUMULATOR.
940061257-01	C11	2C11D001092	REPLACE HCU 18-03 WATER ACCUMULATOR.
940061243-01	C11	2C11D001112	REPLACE HCU 38-47 WATER ACCUMULATOR.
940061242-01	C11	2C11D001127	REPLACE HCU 42-39 WATER ACCUMULATOR.
940061239-01	C11	2C11D001131	REPLACE HCU 58-35 WATER ACCUMULATOR.
940073378-01	C22	2PIS-C22N701B	ABNORMAL READINGS, REPLACE RELAY.
950049582-02	C41	1C41F004A	REPAIR AMPHENOL CONNECTOR.
950036414-02	C41	1C41F029B	RELIEF VALVE LIFTS AT EARLY 1220#.
950067471-01	C41	1C41000	REBUILD RELIEF VALVE.
950063960-01	C41	2C41F004A-AMPH	LOSS OF ALARM.
950022697-01	C51	1C51000	CHECK CONNECTORS FOR LOOSE PINS.
940076713-01	C51	1RR-C51R602	DIGITAL DISPLAY MISSING SEGMENTS.
950068707-01	C51	1RR-C51R603A	DIGITAL DISPLAY MISSING SEGMENTS.
950100645-01	C51	1RR-C51R603C	INACCURATE RECORDER INDICATION.
950070693-01	C51	1RY-C51K002D	EXTREME NOISE LEVEL.
950022332-01	C51	1RY-C51K601A	IRM RESET SWITCH BROKEN.
950097243-05	C51	1RY-C51K601A	REPLACE TRANSISTORS IN PREREGULATORS.
950097243-06	C51	1RY-C51K601B	REPLACE TRANSISTORS IN PREREGULATORS.
950097243-07	C51	1RY-C51K601C	REPLACE TRANSISTORS IN PREREGULATORS.
950097243-08	C51	1RY-C51K601D	REPLACE TRANSISTORS IN PREREGULATORS.
950097243-09	C51	1RY-C51K601E	REPLACE TRANSISTORS IN PREREGULATORS.
940086082-01	C51	1RY-C51K601F	REPLACE IRM F VOLTAGE PREREGULATOR.
950097243-10	C51	1RY-C51K601F	REPLACE TRANSISTORS IN PREREGULATORS.
950097243-11	C51	1RY-C51K601G	REPLACE TRANSISTORS IN PREREGULATORS.
950004226-01	C51	1RY-C51K601H	ERRATIC YIELDING ALARMS.
950097243-12	C51	1RY-C51K601H	REPLACE TRANSISTORS IN PREREGULATORS.

ATTACHMENT A
SAFETY-RELATED MAINTENANCE COMPLETED
(NON-OUTAGE RELATED)

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
940092263-01	C51	1RY-C51K605CS	INACCURATE INDICATION.
910048577-01	C51	1RY-C51K605EM	LPRM FAILED DOWNSCALE.
950009081-01	C51	1RY-C51K605GM	REPLACE K18 RELAYS ON APRM'S.
950015555-01	C51	1RY-C51K605GN	CAUSING ROD BLOCKS TO RMCS.
950009081-02	C51	1RY-C51K605GN	REPLACE K18 RELAYS ON APRM'S.
950009081-03	C51	1RY-C51K605GP	REPLACE K18 RELAYS ON APRM'S.
950114867-01	C51	1RY-C51K605GR	SPURIOUS APRM UPSCALE NEUTRON TRIP.
950009081-04	C51	1RY-C51K605GR	REPLACE K18 RELAYS ON APRM'S.
950058917-01	C51	1RY-C51K605GS	REPLACE K18 RELAYS ON APRM'S.
950009081-06	C51	1RY-C51K605GT	REPLACE K18 RELAYS ON APRM'S.
940078168-01	C51	2C51000	HIGH NOISE LEVELS ON SRMS/IRMS.
940062086-01	C51	2RR-C51R603A	DIGITAL DISPLAY MISSING SEGMENTS.
950031481-01	C51	2RR-C51R603C	DIGITAL DISPLAY MISSING SEGMENTS.
950007552-01	C51	2RR-C51R603D	RECORDER FAILED TO GO FULL SCALE.
950003608-01	C51	2RY-C51K600CA	REPLACE CHASSIS CONNECTOR.
940061879-01	C51	2RY-C51K601A	REPLACE IRM 'A' PREREGULATOR.
950056699-01	C51	2RY-C51K601H	IRM METER, INACCURATE INDICATION.
950054823-01	C51	2RY-C51K605BF	POWER SUPPLY IS READING LOW.
950073660-01	C51	2RY-C51K605BF	SPURIOUS HI ALARM UP.
940089885-01	C51	2RY-C51K605GM	REPLACE K-18 RELAYS.
940089885-02	C51	2RY-C51K605GN	REPLACE K-18 RELAYS.
940089885-03	C51	2RY-C51K605GP	REPLACE K-18 RELAYS.
950075737-01	C51	2RY-C51K605GR	APRM 'D' INDICATOR IS ACTING UP.
950032162-01	C51	2RY-C51K605GR	COMPUTER POINT FAILED TO RESPOND.
940089885-04	C51	2RY-C51K605GR	REPLACE K-18 RELAYS.
950072265-01	C51	2RY-C51K605GS	'E' APRM CAUSED SPURIOUS HALF-SCRAM.
940089885-05	C51	2RY-C51K605GS	REPLACE K-18 RELAYS.
950007295-01	C51	2RY-C51K605GT	REPLACE QUAD TRIP CARD.
940089885-06	C51	2RY-C51K605GT	REPLACE K-18 RELAYS.
940062489-02	C61	1EY-C61K002	CALIBRATION.
950033194-01	C71	1C71AK010D	INSPECT LIMIT FOR CAUSE OF 1/2 SCRAM.
950037621-01	C71	1C71AK010D	RELAY CONTACTS FAILED.
930043774-01	C71	1C71000	INSTALL BANANA JACKS.
940059473-01	C71	2C71000	INSTALL BANANA JACKS.
950012257-01	DC	1DC001E	ADJUST BATTERY ACID CONCENTRATION.
940057641-01	DC	1DC06E	DEVICES FAILED TO TRIP.
950014491-01	DC	1EI-DC055	INDICATION READS LOW.
950018720-01	DC	2DC03E	REPLACE BATTERY CHARGER ALARM BOARD.
940060810-01	DC	2DC16E	REPLACE BATTERY CAPACITOR.
950116676-01	DC	2DC18E	CELL ICV BELOW TECH SPEC LIMITS.
950092691-01	DG	0DG01K	CYLINDER #20 HIGH FIRING PRESSURE.
940062049-01	DG	0DG01K	REPLACE CYLINDER TEST VALVE.
940078453-01	DG	0PS-DG047	REPLACE CRANKCASE PRESSURE SWITCH.
950062401-01	DG	0TS-DG041	TEMPERATURE SWITCH FAILED TO ACTUATE.
950112187-01	DG	1HS-DGS027	REPLACE VOLTAGE ADJUST POT.
940060075-01	DG	1HS-DG015	INSPECT CONTROL SWITCH.
950065758-01	DG	2DG01K	INVESTIGATE LOAD FLUCTUATION.
950033138-01	DG	2DG01P	CLEAN SAND DUST.
950004305-01	DG	2DG01S-B6	INSPECT AND REPAIR CABLE.
950013570-03	DG	2DG02JA	CALIBRATE RELAYS AND METERS.
950055770-01	DG	2PS-DG042B	D826 ALARM COMES IN EARLY.
950104145-01	DR	1DR000	REPLACE NUT AND ROD.
950106202-01	DR	2DR000	REPLACE NUT AND ROD.
950106494-01	DR	2DR000	REPLACE NUTS AND RODS.
950050086-01	D18	0RE-D18N511	SBGT LOW RANGE ERRATIC READINGS.
950048633-01	D18	0RE-D18N513	SBGT WRGM HIGH RANGE ERRATIC READINGS.

ATTACHMENT A

SAFETY-RELATED MAINTENANCE COMPLETED

(NON-OUTAGE RELATED)

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
940088324-01	D18	0RE-D18N537	TEMPERATURE ELEMENT FAILED UP SCALE.
940062079-01	D18	0US-D18R516	DIGITAL DISPLAY FAILS.
950116781-01	D18	1RE-D18N009A	REPAIR LOOP COMPONENTS.
950000295-01	D18	1RIY-D18K751B	READS HIGH.
950034189-01	D18	1RIY-D18K751D	'D' CHANNEL SPIKED HIGH.
950114911-01	D18	2RIY-D18K610A	MONITOR READING HIGH.
950070695-01	D18	2RIY-D18K751B	NO INDICATION LIGHT.
950034200-01	E12	1E12C003	REBUILD WATER LEG PUMP.
940064880-01	E12	1E12C300A	MEGGER/CURRENT READINGS ON MOTORS.
940064885-01	E12	1E12C300B	MEGGER/CURRENT READINGS ON MOTORS.
950061744-01	E12	1PDS-E12N010AA	DIAPHRAGM FAILED.
950040363-01	E12	1PS-E12N413A	INSTRUMENT STOP VALVE LEAKS.
950017820-01	E12	2E12C002A	ADJUST BREAKER ALIGNMENT.
950025994-01	E12	2E12C003	MEGGER/CURRENT READINGS ON MOTORS.
940088684-01	E12	2E12C300A	LOOSE GROUND STRAP AT THE MOTOR BASE.
950068855-01	E12	2E12C300A	MEGGER/CURRENT READINGS ON MOTORS.
950068857-01	E12	2E12C300B	MEGGER/CURRENT READINGS ON MOTORS.
950033163-01	E12	2E12C300C	CLEAN SAND DUST.
950065613-01	E12	2E12C300C	MEGGER/CURRENT READINGS ON MOTORS.
950033169-01	E12	2E12C300D	CLEAN SAND DUST.
950065612-01	E12	2E12C300D	MEGGER/CURRENT READINGS ON MOTORS.
930047047-01	E12	2E12F006B	VALVE HAS PACKING LEAK.
940057847-02	E12	2E12F016A	REPLACE MOTOR.
940057833-02	E12	2E12F016B	REPLACE MOTOR.
940057839-02	E12	2E12F017B	REPLACE MOTOR.
940060100-01	E12	2E12F042A	MEGGER/CURRENT READINGS ON MOTORS.
940075401-01	E21	1E21C002	NOISE COMING FROM PUMP.
950032548-02	E21	1E21F001	PERFORM STEM LUBE.
940085301-01	E22	1E22F362A	INSPECT CHECK VALVE.
940062051-01	E22	1E22S001	REPLACE TEST VALVES.
950111775-01	E22	1HS-E22BS027	REPLACE THE VOLTAGE ADJUST POT.
950061945-01	E22	2E22C003	GEAR BOX HAS OIL LEAK.
950068425-01	E22	2E22C003	GEAR BOX HAS OIL LEAK.
950064065-01	E22	2E22C003	REPLACE PUMP.
950104801-01	E22	2E22C003	CHANGE BREATHER TO ELIMINATE OIL LEAKAGE.
950075886-01	E22	2E22F362A	COMPRESSOR RELIEF VALVE LIFTING.
950044371-01	E22	2E22F362B	CHECK VALVE IS LEAKING BY.
950036734-01	E22	2E22F375A	PACKING LEAK.
950075344-01	E22	2E22F380B	ISOLATION VALVE LEAKS.
930046741-01	E22	2E22P301B	LOW ALARM IS UP WHEN D/G IS RUNNING.
950111361-01	E22	2E22S001	VARS FLUCTUATING DURING RUN.
950057788-01	E22	2E22S001	INSPECT THE 2B DG FOR TRIP.
950058109-01	E22	2E22S001	REPAIR VAR RELATED PROBLEMS.
950111361-02	E22	2HS-E22BS027	VARS FLUCTUATING DURING RUN.
950031791-01	E31	1PDS-E31N013BA	SUPPLY FLOW SWITCH DIAPHRAGM BLOWN.
940084060-01	E31	1TS-E31N601B	RILEY MODULE IS CHATTERING.
950119671-01	E31	1TS-E31N601J	INDICATOR TRIPPED, TEMPERATURE NORMAL.
950111398-01	E31	1TS-E31N601J	RILEY MODULE CHATTERS.
950112849-01	E31	2PDS-E31N013BA	REPLACE SWITCH DIAPHRAM.
950104361-01	E31	2TE-E31N004A	TEMPERATURE ALARMING WITH NORMAL TEMP.
940089867-02	E31	2TE-E31N029B	PERFORM EQ INSPECTIONS.
950097114-01	E31	2TS-E31N601K	RESISTER LEAD BROKEN.
950011942-01	E32	1E32F002J	DUAL INDICATION WHEN CLOSED.
950064432-01	E32	1FI-E32R653E	INDICATION HIGH.
950059342-01	E32	1FT-E32N053A	REPAIR RING LUGS.
950057412-01	E32	1FT-E32N053E	LINE FLOW PEGGED HIGH.

ATTACHMENT A

SAFETY-RELATED MAINTENANCE COMPLETED

(NON-OUTAGE RELATED)

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950092263-01	E32	1PI-E32R661A	REPLACE METER, BROKEN.
950106504-01	E32	2FT-E32N053J	FAILED SURVEILLANCE TEST.
950009233-01	E32	2PS-E32N651A	REPAIR BROKEN LUG.
950044365-01	E32	2PY-E32AK008E	REPAIR WIRING.
950080580-01	E51	1E51C002	REPAIR FLEX CONDUIT.
950025591-01	E51	1E51C003	MEGGER/CURRENT READINGS ON MOTORS.
950118463-01	E51	1E51C003	PUMP TRIPS BREAKER UPON START.
950008699-01	E51	1E51F360	REPLACE SPLIT COUPLING.
940072539-01	E51	1E51000	RESET TO THE CORRECT POSITION.
950046294-01	E51	2E51C001	TURBINE TRIP WOULD NOT RESET.
940079383-01	E51	2E51C003	REPAIR SMALL OIL LEAK.
950065165-01	E51	2E51C003	VIBRATION INCREASING.
950059795-01	E51	2E51F019	REPLACE BREAKER 74 RELAY.
950052189-01	E51	2E51F022	GROUND DETECTOR ALARM WHEN CYCLED.
950013545-03	FC	1FCA000	10 YR PIPING INSPECTION.
950013545-04	FC	1FCB000	10 YR PIPING INSPECTION.
950037023-02	FC	1FC017	VALVE WILL NOT GO FULL CLOSED.
940058871-01	FC	1FC044B	INTERNAL LEAKAGE, REPAIR.
940058870-01	FC	1FC045B	INTERNAL LEAKAGE, REPAIR.
940061986-01	FC	1FE-FC037	INSPECT FLOW ELEMENT.
950041906-01	FC	2FC017	CHECK ANTI-ROTATION CLAMP, COLLAR.
950068859-01	FC	2FC03PA	MEGGER/CURRENT READINGS ON MOTORS.
940074030-01	FC	2FC03PA	HIGH VIBRATION READINGS.
950033154-01	FC	2FC03PB	CLEAN SAND DUST.
930047088-01	FC	2FC03PB	HIGH VIBRATION IN MOTOR.
940061419-01	FC	2FC044A	INSPECT CHECK VALVE.
940061355-02	FP	2FP000	REMOVE THERMO-LAG MATERIAL.
950019655-01	HG	1HG006A	THERMALS TRIPPED WITH OPEN SIGNAL.
950007966-01	HG	1HG009	PERFORM EQ INSPECTION ON ACTUATORS.
950032673-02	HG	1HG009	PERFORM STEM LUBE.
940059980-02	HG	2HG018	PERFORM EQ INSPECTION ON ACTUATOR.
950114627-01	H13	1H13P644	REPLACE BLOWN FUSE.
950044793-01	H13	2H13P601	REPLACE MISSING COVERS.
950059247-01	H13	2H13P609	REPLACE WIRING.
950021719-01	H13	2H13P611	REPLACE TERMINAL BLOCK.
950071146-01	IS	2IS000	DOOR DOES NOT SEAL.
930046195-01	LP	1LP000	REPLACE PUMP AMMETER.
950025820-01	LV	1LV93E	PENETRATION E-26 READS LOW.
940079282-01	NR	2NR000	'A' REM SPIKED UPSCALE.
950037118-01	PC	2PCM111	REPLACE DOOR SEAL.
950001114-01	PL	0PLD2J	PUMP WILL NOT RUN IN AUTO OR MANUAL.
950104030-01	PL	1PL76J	'H2' REAGENT GAS BOTTLE LEAKING.
950105681-01	PL	1PL77J	'H2' SAMPLE READING HIGH.
950068717-01	PL	1PL77J	OSCILLATES FROM 0-5%.
940061207-01	PL	2PL76J	REPLACE MISSING SCREWS/COVER ON PANEL.
940061205-01	PL	2PL77J	INSPECT ASSEMBLY.
950056371-01	RD	1RD000	ROD 30-39 DRIFTED IN DURING TESTING.
950013385-01	RH	1RH000	PERFORM FLOW TEST.
950007075-01	RH	1RH13-1131X	CLAMP BOLT LOOSE.
940086111-01	RH	2RHA000	10 YR HYDRO INSPECTION.
940086111-02	RH	2RHB000	10 YR HYDRO INSPECTION.
950013545-05	SC	1SCA000	10 YR PIPING INSPECTION.
950013545-06	SC	1SCB000	10 YR PIPING INSPECTION.
950075805-01	VC	0FR-VC028	RECORDER NOT FUNCTIONING PROPERLY.
950010245-01	VC	0VC01SA	RELAY CONTACTS DIRTY.

ATTACHMENT A
SAFETY-RELATED MAINTENANCE COMPLETED
(NON-OUTAGE RELATED)

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950027296-01	VC	0VC05CB	CHECK RELAYS AND CONTACTORS.
950076573-01	VC	0VC05YB	DAMPER FAILED CLOSED.
950056970-01	VC	0VC05YB	DAMPER FAILED CLOSED.
950019453-02	VC	0VC24YB	DAMPER FAILED CLOSED.
920049006-01	VC	0VC43Y	INSTALL ACCESS DOOR IN DUCTWORK.
920049007-01	VC	0VC45Y	INSTALL ACCESS DOOR.
940060653-01	VC	0XY-VC125A	ALARMS FAIL TO RESET.
950005646-01	VC	0XY-VC125B	DETECTOR IS INDICATING UPSCALE.
940060653-02	VC	0XY-VC165A	ALARMS FAIL TO RESET.
950056500-01	VE	0FI-VE067	INDICATOR OUT OF CALIBRATION.
950068856-01	VE	0TI-VE044	TEMPERATURE INDICATION READING HIGH.
950069479-01	VE	0TS-VE109	COMPRESSOR TRIPPED ON HI TEMPERATURE.
950100184-01	VE	0VE01AA	REPLACE BOTH DOOR GASKETS.
930047337-01	VE	0VE03CA	EXCESSIVE VIBRATION.
950045529-02	VE	0VE03CA	FAN TRIPPED WHILE STARTING.
950078746-01	VE	0VE04CA	COMPRESSOR TRIPPED ON HI TEMPERATURE.
950027298-01	VE	0VE04CA	CHECK RELAYS AND CONTACTORS.
950107287-01	VE	0VE04CA	CHECK RELAYS.
950060071-01	VE	0VE04CA	TRIPPED MULTIPLE TIMES ON OVERLOAD.
950061484-02	VE	0VE04CA	TRIPPED ON HIGH OIL TEMPERATURE.
950020079-01	VE	0VE04CA	COMPRESSOR NOT OPERATING PROPERLY.
950027297-01	VE	0VE04CB	CHECK RELAYS AND CONTACTORS.
950036286-01	VE	0VE09YB	REPLACE FUSE.
950111839-01	VG	1FC-VG003	REPLACE CAPACITORS IN FLOW CONTROLLER.
950113419-01	VG	1FC-VG003	REPLACE CAPACITORS IN FLOW CONTROLLER.
950025990-01	VG	1FZ-VG003	REPLACE DAMPER ACTUATOR.
950017234-01	VG	1PDS-VG021	ALARM CAME IN EARLY.
940062070-01	VG	1TT-VG012	REPAIR POWER LEAD.
940060241-01	VG	1VG001	PERFORM SPRINGPACK TEST.
940060242-01	VG	1VG003	PERFORM SPRINGPACK TEST.
950038988-01	VG	1VG003	VALVE SHOWS DUAL INDICATION.
930045040-01	VG	1ZS-VG004A	REPLACE CLOSED END LIMIT SWITCH.
930045039-01	VG	1ZS-VG004B	REPLACE OPEN END LIMIT SWITCH.
940090400-01	VG	2FS-VG036	HIGH FLOW ALARM INTERMITENT.
950056028-01	VG	2FZ-VG003	DAMPER FAILED CLOSED.
950016902-01	VG	2VG01C	REPLACE HFA RELAY.
920044926-01	VG	2VG02Y	REPLACE BASE BOLTS.
940060767-01	VG	2ZS-VG004A	REPLACE LIMIT SWITCH.
940060768-01	VG	2ZS-VG004B	REPLACE LIMIT SWITCH.
940062114-01	VQ	1VQ031	REBUILD ACTUATOR.
940059509-02	VR	2VR04YA	REPLACE DAMPER SPRINGS.
940059507-02	VR	2VR04YB	REPLACE DAMPER SPRINGS.
940059508-02	VR	2VR05YA	REPLACE EXISTING SPRINGS.
950022516-01	VX	1VX07Y	DAMPER IS LEAKING OIL.
950058063-01	VY	1TIC-VY024	ALARM UP, TEMPERATURE NORMAL.
950054867-01	VY	2TIC-VY024	PUMP CUBICLE TEMPERATURE HIGH ALARM.
950016388-01	VY	2VY04C	INSPECT WIRING TO PILOT LIGHT.
950104061-01	ZZ	0ZZ000	YOKOGAWA BATTERY REPLACEMENT.
950104057-01	ZZ	0ZZ000	YOKOGAWA RECORDER'S TIME CHANGE.

ATTACHMENT B
II.B UNIT SHUTDOWNS
(UNIT 1)

DATE: 950209 (YYMMDD) GENERATOR OFF-LINE: 140.3 (Hours) OUTAGE TYPE: Forced (L1F29)

REASON: Manual Reactor scram to repair the 1E51F357 valve.

CRITICAL ACTIVITY PATH: Repair of the 1E51F357 valve.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WRNUM	SYS	EPN	Description
940061467-01	C51	1RE-C51N002E	IV PLOTS IRM "E" SHOWS RESISTIVE.
950010998-01	C51	1RY-C51K601A	DOWNSCALE IS ERRATIC.
950011834-01	VG	1FC-VG003	CYCLES EXCESSIVELY WHEN VG IS RUNNING.

DATE: 950611 (YYMMDD) GENERATOR OFF-LINE: 158.2 (Hours) OUTAGE TYPE: Forced (L1F30)

REASON: Manual Reactor scram to replace the Main Steam Isolation Valve solenoid valves.

CRITICAL ACTIVITY PATH: Replacement of solenoid valves.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WRNUM	SYS	EPN	Description
950019766-01	C51	1RE-C51N002A	A IRM OUTPUT METER IS ERRATIC.
950039892-01	C71	1ZS-C71N006D	REPLACE MSV #2 LIMIT SWITCH.
950039893-01	C71	1ZS-C71N006B	REPLACE MSV #1 LIMIT SWITCH.
950039894-01	C71	1ZS-C71N006F	REPLACE MSV #3 LIMIT SWITCH.
950039895-01	C71	1ZS-C71N006H	REPLACE MSV #4 LIMIT SWITCH.
950042448-01	C51	1RY-C51K601H	IRM 'H' FREQUENTLY SPIKES HIGH.
950051999-01	B21	1B21F028A	'A' MSIV REOPENED WITH SWITCH IN THE CLOSED POSITION.
950052000-01	B21	1B21F028B	REPLACE THE ASCO PILOT SOLENOIDS.
950052001-01	B21	1B21F028C	REPLACE THE ASCO PILOT SOLENOIDS.
950052002-01	B21	1B21F028D	REPLACE THE ASCO PILOT SOLENOIDS.
950052089-01	C51	1RE-C51N002E	ERRATIC INDICATION AND 1/2 SCRAM.
950052829-01	PL	1PL77J	O2 MONITOR STUCK AT 17.5%, ACTUAL IS 21%.

ATTACHMENT B
II.B UNIT SHUTDOWNS
(UNIT 1)

DATE: 950816 (YYMMDD) GENERATOR OFF-LINE: 130.4 (Hours) OUTAGE TYPE: Forced (L1F31)

REASON: Automatic Reactor scram from Main Steam Isolation Valve closure due to Main Steam Tunnel high temperature which was caused by the loss of the Reactor Building Ventilation system.

CRITICAL ACTIVITY PATH: Troubleshooting and repair of the EPA logic card.

CORRECTIVE ACTIONS (PIF/LER# if applicable): LER# 95-014.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WRNUM	SYS	EPN	Description
940088674-01	C71	1C71S003A	REPLACE TRIP CARD.
940088675-01	C71	1C71S003B	REPLACE TRIP CARD.
940088675-02	C71	1C71S003B	CALIBRATION INDICATES BAD TRANSFORMER.
940088676-01	C71	1C71S003D	REPLACE TRIP CARD.
940088686-01	C71	1C71S003C	REPLACE TRIP CARD.
950054043-01	C51	1RY-C51K601E	'E' IRM ERRATIC INDICATION.
950054043-02	C51	1RE-C51N002E	REPLACE IRM "E" DETECTOR.
950056549-01	C51	1RY-C51K601A	ERRATIC OUTPUT CAUSING RELJY CHATTER.
950056549-02	C51	1RE-C51N002A	REPLACE IRM "A" DETECTOR.
950071755-01	PL	1PL76J	OXYGEN CONCENTRATION IS UPSCALE HIGH.
950072292-01	E31	1TS-E31N604C	ALARM RELAY CHATTERS.
950072362-01	C71	1C71S003D	RESOLDER CONNECTIONS ON LOGIC BOARD.

DATE: 950924 (YYMMDD) GENERATOR OFF-LINE: 102.1 (Hours) OUTAGE TYPE: Forced (L1F32)

REASON: Manual Reactor scram due to loss of the '1B' Turbine Driven Reactor Feed Pump during surveillance testing.

CRITICAL ACTIVITY PATH: Inspection and troublesooting of the '1B' Turbine Driven Reactor Feed Pump.

CORRECTIVE ACTIONS (PIF/LER# if applicable): LER# 95-016.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WRNUM	SYS	EPN	Description
950092398-01	B33	1B33F060B	INSPECT AND MODIFY THE LVDT LINKAGE AND VALVE INSULATION.
950092399-01	B33	1B33F060A	INSPECT AND MODIFY THE LVDT LINKAGE AND VALVE INSULATION.

ATTACHMENT B
II.B UNIT SHUTDOWNS
(UNIT 2)

DATE: 950218 (YYMMDD) GENERATOR OFF-LINE: 2713.0 (Hours) OUTAGE TYPE: Refuel (L2R06)

REASON: Refuel outage.

CRITICAL ACTIVITY PATH: See Appendix A.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED: See Appendix B.

DATE: 950612 (YYMMDD) GENERATOR OFF-LINE: 1.5 (Hours) OUTAGE TYPE: Scheduled (L2M10)

REASON: Main Turbine overspeed test.

CRITICAL ACTIVITY PATH: Completion of overspeed testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED: None.

DATE: 950916 (YYMMDD) GENERATOR OFF-LINE: 189.5 (Hours) OUTAGE TYPE: Scheduled (L2F23)

REASON: Manual Reactor scram for maintenance of the '2A' Reactor Recirculation Flow Control Valve.

CRITICAL ACTIVITY PATH: Repair of the Flow Control Valve.

CORRECTIVE ACTIONS (PIF/LER# if applicable):

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WRNUM	SYS	EPN	Description
930043461-05	E32	2PT-E32N051J	ISOLATION VALVE LEAKS BY.
930044523-04	E32	2PT-E32N051A	VALVE LEAKING BY.
940058620-03	E32	2E32N051N	VALVE LEAKING BY.
950053381-01	E51	2E51F066	NO INDICATION.
950055503-01	B21	2B21F032A	DUAL VALVE INDICATION AT FULL POWER.
950069089-01	B33	2B33F060A	DISASSEMBLE/REPAIR AS REQUIRED.
950069089-04	B33	2B33F060A	INSTALL LARGER FEED BACK LEVER PINCH BOLT.
950069089-05	B33	2B33F060A	LOOSEN & RETIGHTEN PACKING.
950077175-01	B33	2B33F067A	INSPECT ANTI-ROTATION DEVICES VALVE.
950077175-02	B33	2B33F067B	INSPECT ANTI-ROTATION DEVICES VALVE.
950090461-01	E12	2E12F041A	PACKING LEAK.
950090461-02	E12	2E12F041A	REPLACE ACTUATOR.
950090461-03	E12	2E12F041A	INSPECT ACTUATOR RETURN SPRING.
950092327-01	B33	2B33F060B	INSPECT/MODIFY VALVE INSULATION.

ATTACHMENT C
II.B FORCED REDUCTIONS IN POWER
GREATER THAN 20% IN DESIGN POWER LEVEL
(UNIT 1)

DATE: 950705 (YYMMDD)

OPERATION AT REDUCED POWER: 7.0 (Hours)

REASON: Power reduction for scram time testing of control rods due to replacement of scram solenoid pilot valves.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950057109-01	C11	1C11000	REPLACE SSPV'S ON HCU 14-55.
950057110-01	C11	1C11000	REPLACE SSPV'S ON HCU 46-07.
950057111-01	C11	1C11000	REPLACE SSPV'S ON HCU 10-15.
950057112-01	C11	1C11000	REPLACE SSPV'S ON HCU 50-47.

DATE: 950717 (YYMMDD)

OPERATION AT REDUCED POWER: 7.0 (Hours)

REASON: Power reduction for scram time testing of control rod due to replacement of scram solenoid pilot valves.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950056693-01	C11	1C11000	REPLACE SSPV'S ON HCU 30-39.

DATE: 950718 (YYMMDD)

OPERATION AT REDUCED POWER: 6.0 (Hours)

REASON: Power reduction for scram time testing of control rods due to replacement of scram solenoid pilot valves.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950057784-01	C11	1C11000	REPLACE SSPV'S ON HCU'S 06-23, 54-39, 34-31 AND 26-31.

ATTACHMENT C
II.B FORCED REDUCTIONS IN POWER
GREATER THAN 20% IN DESIGN POWER LEVEL
(UNIT 1)

DATE: 950719 (YYMMDD)   OPERATION AT REDUCED POWER: 6.0 (Hours)

REASON: Power reduction for scram time testing of control rods due to replacement of scram solenoid pilot valves.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950057785-01	C11	1C11000	REPLACE SSPV'S ON HCU'S 30-23, 38-07, 22-55, 10-31 AND 50-31.

DATE: 950722 (YYMMDD)   OPERATION AT REDUCED POWER: 7.0 (Hours)

REASON: Power reduction for scram time testing of control rod.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950062853-01	C11	1C11000	REPLACE PROBE MUX CARD FOR CONTROL ROD 22-55.

DATE: 951027 (YYMMDD)   OPERATION AT REDUCED POWER: 7.5 (Hours)

REASON: Power reduction for scram time testing of control rod due to replacement of scram solenoid pilot valves.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950105037-01	C11	1C11000	REPLACE SSPV'S ON HCU 42-39.

ATTACHMENT C

II.B FORCED REDUCTIONS IN POWER
GREATER THAN 20% IN DESIGN POWER LEVEL
(UNIT 1)

DATE: 951029 (YYMMDD)

OPERATION AT REDUCED POWER: 8.5 (Hours)

REASON: Power reduction for scram time testing of control rods due to scram solenoid pilot valve concerns.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED: None.

DATE: 951108 (YYMMDD)

OPERATION AT REDUCED POWER: 4.5 (Hours)

REASON: Power reduction for scram time testing of control rods due to replacement of scram solenoid pilot valves.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950106007-01	C11	1C11000	REPLACE SSPV'S ON HCU'S 06-47, 18-23, 30-47 AND 46-15.

DATE: 951217 (YYMMDD)

OPERATION AT REDUCED POWER: 30.5 (Hours)

REASON: Scram time testing of control rod 50-23 due to Hydraulic Control Unit water accumulator replacement.

CRITICAL ACTIVITY PATH: Completion of testing.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED:

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
940086644-01	C11	1C11001156	REPLACE THE WATER ACCUMULATOR ON HCU 50-23.

ATTACHMENT C
II.B FORCED REDUCTIONS IN POWER
GREATER THAN 20% IN DESIGN POWER LEVEL
(UNIT 2)

DATE: 951221 (YYMMDD) OPERATION AT REDUCED POWER: 8.0 (Hours)

REASON: Power reduction for maintenance on the '2A' and '2B' Feedwater Heater Drain Pump.

CRITICAL ACTIVITY PATH: Completion of maintenance.

CORRECTIVE ACTIONS (PIF/LER# if applicable): None.

RADIOACTIVITY RELEASE/EXPOSURE OVER 10% ALLOWABLE VALUES: None.

SAFETY RELATED CORRECTIVE MAINTENANCE COMPLETED: None.

ATTACHMENT D
RADIATION EXPOSURE

REGULATORY GUIDE 1.16 REPORT

NUMBER OF PERSONNEL AND PERSON-REM BY WORK AND JOB FUNCTION FOR 1995

WORK AND JOB FUNCTION	NUMBER OF PERSONNEL: STATION EMPLOYEES	NUMBER OF PERSONNEL: CONTRACTORS/ OTHERS	NUMBER OF PERSONNEL: UTILITY EMPLOYEES	TOTAL PERSONS	QUANTITY OF PERSON-REM: STATION EMPLOYEES	QUANTITY OF PERSON-REM: CONTRACTORS/ OTHERS	QUANTITY OF PERSON-REM: UTILITY EMPLOYEES	TOTAL PERSON-REM
REACTOR OPERATIONS AND SURVEILLANCE								
ENGINEERING	76	5	0		8.275	1.438	0.000	
HEALTH PHYSICS	43	7	123		17.090	1.677	1.100	
MAINTENANCE	34	18	1		15.982	3.384	0.057	
OPERATIONS	132	115	0		34.418	5.730	0.000	
SUPERVISORY	100	64	0		6.819	1.648	0.000	
	385	209	124	718	82.584	13.877	1.157	97.618
ROUTINE MAINTENANCE								
ENGINEERING	56	91	0		6.083	25.475	0.000	
HEALTH PHYSICS	36	29	147		14.154	7.041	1.313	
MAINTENANCE	230	647	21		108.542	123.029	1.089	
OPERATIONS	18	0	0		4.718	0.001	0.000	
SUPERVISORY	148	69	0		10.153	1.774	0.000	
	488	835	168	1491	143.650	157.321	2.402	303.373
INSERVICE INSPECTION								
ENGINEERING	8	39	0		0.858	11.022	0.000	
HEALTH PHYSICS	1	21	3		0.238	5.224	0.028	
MAINTENANCE	0	242	0		0.094	46.016	0.000	
OPERATIONS	0	0	0		0.042	0.000	0.000	
SUPERVISORY	6	24	0		0.388	0.615	0.000	
	15	326	3	344	1.620	62.876	0.028	64.524
SPECIAL MAINTENANCE								
ENGINEERING	4	0	0		0.475	0.003	0.000	
HEALTH PHYSICS	1	0	1		0.453	0.017	0.006	
MAINTENANCE	2	38	2		1.023	7.262	0.124	
OPERATIONS	0	0	0		0.121	0.000	0.000	
SUPERVISORY	4	6	0		0.257	0.164	0.000	
	12	45	3	60	2.329	7.445	0.130	9.904
REFUELING								
ENGINEERING	1	57	0		0.110	15.934	0.000	
HEALTH PHYSICS	4	2	13		1.591	0.389	0.114	
MAINTENANCE	3	42	0		1.242	8.079	0.000	
OPERATIONS	7	0	0		1.872	0.000	0.000	
SUPERVISORY	18	3	0		1.214	0.080	0.000	
	33	104	13	149	6.028	24.481	0.114	30.623
WASTE PROCESSING								
ENGINEERING	0	0	0		0.026	0.000	0.000	
HEALTH PHYSICS	1	0	4		0.360	0.000	0.034	
MAINTENANCE	3	10	0		1.267	1.917	0.000	
OPERATIONS	1	31	0		0.380	1.546	0.000	
SUPERVISORY	2	4	0		0.113	0.114	0.000	
	7	45	4	56	2.146	3.577	0.034	5.757

ATTACHMENT D
RADIATION EXPOSURE

REGULATORY GUIDE 1.16 REPORT

NUMBER OF PERSONNEL AND PERSON-REM BY WORK AND JOB FUNCTION FOR 1995

WORK AND JOB FUNCTION	NUMBER OF PERSONNEL				QUANTITY OF PERSON-REM			
	STATION EMPLOYEES	CONTRACTORS/ OTHERS	UTILITY EMPLOYEES	TOTAL PERSONS	STATION EMPLOYEES	CONTRACTORS/ OTHERS	UTILITY EMPLOYEES	TOTAL PERSON-REM
TOTAL BY JOB FUNCTION								
ENGINEERING	146	192	0	338	15.827	53.872	0.000	69.699
HEALTH PHYSICS	86	59	290	435	33.886	14.348	2.595	50.829
MAINTENANCE	271	997	25	1293	128.149	189.686	1.270	319.105
OPERATING	159	146	0	305	41.552	7.277	0.000	48.829
SUPERVISORY	277	170	0	447	18.943	4.394	0.000	23.337
*** GRAND TOTAL ***	939	1564	315	2818	238.357	269.577	3.865	511.799

APPENDIX A
CRITICAL PATH ACTIVITIES
REFUEL OUTAGE (L2R06)

'B' DIESEL GENERATOR RELAY CALIBRATIONS AND TRIP TESTS.
REMOVE REACTOR HEAD.
GROUP 2 AND 4 INBOARD ISOLATION FUNCTIONAL TEST.
PUMP DOWN INNER GRATING.
REMOVE EXCITER HEATER SENSOR.
INSPECTION OF THE INBOARD MSIV ACCUM CHECK VALVES.
FLOOD INNNER GRATING.
A,B AND C MSIV'S ACTUATOR PISTON LEAK CHECK.
INSPECT ALL REACTOR STUDS DURING DETENSIONING.
START 'A' SUPPRESSION POOL CLEANUP PUMP.
MAIN STEAM PRE-MAINTENANCE LEAK RATE TESTS.
REACTOR RECIRCULATION PUMP LOGIC TEST.
REACTOR RECIRCULATION PUMP ATWS TRIP RELAY LOGIC TEST.
LEAKAGE CONTROL SYSTEM MAINTENANCE.
INSTALL TRI-NUCLEAR FILTER EQUIPMENT.
DRAIN HYROGEN SEAL OIL.
CAVITY FLOODUP.
INSTALL CATTLE CHUTE.
HIGH PRESSURE CORE SPRAY LEAK RATE TESTING.
2E22-F004 AND 2E22-F005 LEAK RATE TEST.
FILL REACTOR VESSEL WITH SUPPRESSION POOL CLEANUP.
ESTABLISH WATER CLARITY.
CD/CB SYSTEM BOUNDRY OUT OF SERVICE, SYSTEM DRAINED.
HIGH PRESSURE CORE SPRAY SYSTEM MAINTENANCE.
DISASSEMBLE/REASSEMBLE 2E22-F004 VALVE FOR EQ INSPECTION.
REMOVE FUEL POOL GATE.
FUEL POOL COOLING ESTABLISHED AS ALTERNATE SHUTDOWN COOLING.
UNLOAD RX CORE.
RESIDUAL HEAT REMOVAL 'A' LOOP LEAK RATE TESTING.
REPLACE TIMER RELAYS ON TR232A, TR232B, TR234A AND TR234B.
DIV I RHR RELAY LOGIC FUNCTIONAL TEST.
EQ INSPECTION FOR 2E12-F064A.
SECURE B RESIDUAL HEAT REMOVAL IN SHUTDOWN COOLING.
INSERVICE INSPECTION OF 2E12-F008 AND 2E12-F009.
SHUTDOWN COOLING SYSTEM LEAK RATE TESTING.
2E12-F005, 2E12-F008 AND 2E12-F009 LEAK RATE TEST.
DRAIN COMMON SHUTDOWN COOLING HEADER.
2D18-N009A,B,C AND D REACTOR BUILDING CALIBRATION.
A,B,C AND D REACTOR BUILDING EXHAUST PLENUM CALIBRATION.
VOTES TEST FOR 2E22-F004 AND 2E21-F005.
DYNAMIC VOTES TEST ON 2E12-F064A.
SYSTEM AUXILIARY TRANSFORMER MAINTENANCE.
2AP91E REVISE TRANSFORMER COOLING.
'C' LOOP RESIDUAL HEAT REMOVAL SUCTION DECON.
'A' AND 'B' RHR SYSTEM DECON.
'C' LOOP RESIDUAL HEAT REMOVAL VENTED.
2D18-N015A,B,C AND D FUEL POOL VENT CALIBRATION.
HYDRO TEST CONNECTIONS ON 'B' RHR SYSTEM.
REMOVE DECON FLANGE AT 2RE27M, 2E12-D311A, 2E12-F384A.

APPENDIX A
CRITICAL PATH ACTIVITIES
REFUEL OUTAGE (L2R06)

A,B,C AND D REACTOR BUILDING FUEL POOL EXHAUST CALIBRATION.
RESIDUAL HEAT REMOVAL 'A' LOOP MAINTENANCE.
PERFORM STATION AUX TRANSFORMER CIRCUIT CHECKS.
2DC14E REPLACE POST SEAL ASSEMBLIES.
CHEMICAL DECON FOR REACTOR RECIRCULATION PIPING.
STATION AUX TRANSFORMER RELAY ADDITIONS.
VERIFY TRIPPING FROM NEW STATION AUX TRANSFORMER RELAYS.
MOTOR OPERATED VALVES BREAKERS OUT OF SERVICE.
PERFORM TRANSFORMER TESTING.
2DC14E BATTERY 125 VDC BATTERY INSPECTION.
REACTOR BUILDING HVAC AVAILABLE, CLEAR OOS ON FANS.
SYSTEM AUX TRANSFORMER 243 TESTING.
INSTALL FUEL POOL GATES FOR REACTOR RECIRC DECON DRAIN DOWN.
EQ INSPECTION OF VALVE 2E12-F042A.
OIL CIRCUIT BREAKER 1-6 RELAY CALIBRATION.
RESIDUAL HEAT REMOVAL 'B' LOOP MAINTENANCE.
DIV 1 REACTOR VESSEL LOW LEVEL ROSEMOUNT TRANSMITTERS.
REACTOR LEVEL ECCS TRANSMITTER ZERO SHIFT CALIBRATION.
2E21-C001 RELAY CALIBRATION.
2C11-C001A RELAY CALIBRATION.
2E12-C002A RELAY CALIBRATION.
DIVISION 1 AND 2 DC METER CALIBRATIONS.
BUS 241Y TRIP TESTING.
BUS 235X AND 235Y TRIP TESTING.
CHANGE GEAR SET AND MOTOR FOR 2E12-F042A.
EQ INSPECTION OF VALVE 2E12-F024B.
DIVISION 1 AND 2 125V CHARGER CAPACITY TEST.
DIVISION 1 AND 2 BATTERY PERFORMANCE DISCHARGE TEST.
DIVISION 1 BATTERY LOW/HIGH VOLTAGE CHARGE.
CORE SHROUD INSPECTION AFTER DECON.
POST MAINTENANCE TESTING OF VALVE 2E22-F005.
2E22-F012, F014, F015 AND F023 LEAK RATE TEST.
POST MAINTENANCE LEAK RATE TESTS ON HPCS SYSTEM.
FILL AND VENT THE HIGH PRESSURE CORE SPRAY SYSTEM.
INSERVICE INSPECTION OF HPCS SYSTEM VALVES.
STEM LUBES & GREASE SAMPLES OF HPCS VALVES.
HIGH PRESSURE CORE SPRAY RELAY LOGIC TESTING.
DYNAMIC VOTES TEST ON 2E12-F024B.
CLEAR BOUNDRY OUT OF SERVICE ON B RHR DISCHARGE.
DIV 2 RHR RELAY LOGIC FUNCTIONAL TEST.
LOAD FUEL.
CORE VERIFICATION.
SINGLE ROD SUBCRITICAL CHECK.
FILL MAIN STEAM LINES.
REASSEMBLE REACTOR VESSEL.
INSTALL FUEL POOL GATE, POST RELOAD.
DEWATER INNER CAVITY AND BETWEEN FUEL POOL GATES.
REACTOR LEVEL AT 210 TO 215.
REMOVE CATTLE CHUTE.
DYNAMIC VOTES TEST ON 2E12-F042B.

APPENDIX A
CRITICAL PATH ACTIVITIES
REFUEL OUTAGE (L2R06)

DYNAMIC VOTES TEST ON 2E12-F042C.
DIV 2 ECCS INJECTION VALVES OPERABLE.
DYNAMIC VOTES TEST ON 2E12-F042A.
REINSTALL DRYWELL HEAD.
DRYWELL HEAD LEAK RATE TEST.
REACTOR WATER LEVEL ABOVE MAIN STEAM LINES.
PRESSURIZE TO 40.5 PSIG FOR INTEGRATED LEAK RATE TEST.
DEPRESURIZE PRIMARY CONTAINMENT TO 2.5 PSIG.
VENT SUPPRESSION POOL TO ATMOSPHERIC PRESSURE.
DEPRESSURIZE PRIMARY CONTAINMENT TO 1.5 PSIG.
DRYWELL TO SUPPRESSION POOL BYPASS LEAK TEST.
PRIMARY CONTAINMENT DEPRESSURIZE TO ATMOSPHERIC PRESSURE.
INSPECT PRIMARY CONTAINMENT.
LOWER REACTOR VESSEL LEVEL FOR HYDRO.
REFILL FEEDWATER SYSTEM.
STARTUP INSTRUMENT NITROGEN SYSTEM.
STARTUP REACTOR WATER CLEANUP SYSTEM.
HIGH PRESSURE EXCESS FLOW CHECK VALVE OPERABILITY TEST.
REACTOR PRESSURE VESSEL HYDRO.
SCRAM TIME SELECTED CONTROL ROD DRIVES.
PERFORM WEEKLY NR FUNCTIONAL TEST'S.
MODE SWITCH TO STARTUP.
ROD WORTH MINIMIZER OPERABILITY CHECK.
PULL RODS TO CRITICAL.
SHUTDOWN MARGIN TEST.
REACTOR CRITICAL.
HEATUP FOR RCIC TESTING.
RCIC TURBINE OVERSPEED AT 150 PSI.
ADJUST RCIC OVERSPEED TRIP SETTING.
REPERFORM RCIC TURBINE OVERSPEED TRIP.
RCIC PUMP OPERABILITY TEST.
INSTRUMENT RACK CHECK AT 250 PSI.
START UP MOTOR DRIVEN REACTOR FEEDPUMP (300PSI).
HEAT UP TO 600 PSIG.
500PSI ECCS PERMISSIVES CLEAR.
REACTOR PRESSURE >600 PSIG ON STARTUP.
INSTRUMENT RACK CHECK AT 750 PSI.
PULL RODS UNTIL BYPASSES OPEN.
CHECK IRM TO APRM OVERLAP.
REACTOR PRESSURE AT 920 PSI.
MODE SWITCH TO "RUN".
DYNAMIC VOTES TEST OF 2E51-F013, F019, F022 AND F045.
ROLL MAIN TURBINE.
SYNCH GENERATOR TO THE GRID.

APPENDIX B
REFUELING OUTAGE (L2R06)
SAFETY RELATED CORRECTIVE MAINTENANCE

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950044021-01	B21	2PS-B21N560S	"S" SRV ACCUM LOW PRESS ALARM UP.
950031448-01	E12	2RE40-2921X	ADAPTER ASSEMBLY PSA-3 BENT, REPLACE.
950033931-01	B21	2PS-B21N561D	LOW PRESSURE ALARM UP AFTER CYCLING.
940062012-01	C11	2C11D001044	LEAK AT CHARGING NIPPLE AND INST VALVE.
950040707-01	RP	2C71S003F	APPARENT TRIP DURING AND TRANSFER.
950046606-01	C51	2RY-C51K600BA	B SRM IS SPIKING CAUSING SHORT PERIOD ALARMS.
950020116-01	B21	2PS-B21N413B	REPLACE B/C RHR INJECTION VALVE INTERLOCK.
950027918-01	VP	2VP02CB	BREAKER STAYED CLOSED WHEN C/S TAKEN TO TRIP.
950043052-01	E31	2TDS-E31N615B	DIV 2 GRP 1 ISOLATION DUE TO A BROKEN LEAD.
930048140-01	E22	2E22P301B	REPAIR OR REPLACE METER AS NECESSARY.
950013168-01	E22	2E22F342	CHECK VALVE FAILED TO DRAIN.
950033138-01	DG	2DG01P	CLEAN AND INSPECT.
950033169-01	E12	2E12C300D	CLEAN AND INSPECT.
950033163-01	E12	2E12C300C	CLEAN AND INSPECT.
950033154-01	FC	2FC03PB	CLEAN AND INSPECT.
950032162-01	C51	2RY-C51K605GR	COMPUTER POINT B680 DID NOT RESPOND.
930046458-01	DG	2DG01K	CONNECTION TO ENGINE WAS LEAKING.
950028507-01	H13	2H13P642	CONTACTS ON RELAY NEED TO BE CLEANED.
950016246-01	E31	2PDS-E31N009A	CONTACTS UNSTABLE DURING SURVEILLANCE.
940059105-01	E51	2E51F063	CURRENT TORQUE SWITCH SETTING TOO LOW.
940061655-02	E21	2E21C001	DISASSEMBLE AND CLEAN LPCS MOTOR COOLER.
940061421-01	E51	2E51F082	DISASSEMBLE AND INSPECT CHECK VALVE.
940061422-01	E51	2E51F084	DISASSEMBLE AND INSPECT CHECK VALVE.
950038600-01	D18	2D18000	DISCONNECT CABLE CONNECTOR FOR INSPECTION.
930047650-01	DG	2DG01K	DISCHARGE PIPING LEAKING AT UNION.
950028676-01	B21	2PS-B21N023BA	REPLACE PRESSURE SWITCH.
930048337-01	B21	2B21N402B	PACKING LEAKAGE.
930047328-01	B21	2B21F473	CHECK VALVE HAD TO BE MANUALLY RESET.
930047333-01	B33	2B33F313A	CLEAN AND INSPECT CHECK VALVE.
930047324-01	B21	2B21F328B	CHECK VALVE HAD TO BE MANUALLY RESET.
930047325-01	B33	2B33F311C	CHECK VALVE HAD TO BE MANUALLY RESET.
930047326-01	B21	2B21F328A	CHECK VALVE HAD TO BE MANUALLY RESET.
930047332-01	B33	2B33F307C	CLEAN AND INSPECT CHECK VALVE.
930047329-01	B33	2B33F315A	CHECK VALVE HAD TO BE MANUALLY RESET.
950028675-01	B21	2PS-B21N023AA	REPLACE PRESSURE SWITCH.
950011966-01	B33	2FY-B33K606B	CLEAN E-7 CARD.
950022629-01	AP	2AP75E	SIDE OF MCC COMPT E6 HAS BEEN DAMAGED.
950033875-01	DG	2DG01K	EMERGENCY STOP BUTTON FAILED TO TRIP DIESEL.
950047609-01	HG	2HG19B	PIPING LINE 2HG19B IS BENT.
930047912-01	VP	2VP053A	FAILED LEAK RATE TEST.
950040842-01	B21	2B21F571	VALVE DIRTY.
950029789-01	C11	2C11F423G	FAILED HIGH PRESSURE SEAT LEAKAGE TEST.
950029795-01	E32	2E32F001A	FAILED LEAK RATE TEST.
950029505-01	B21	2B21F022A	FAILED LEAK RATE TEST.
950029506-01	B21	2B21F028A	FAILED LEAK RATE TEST.
950016994-01	VQ	2VQ027	FAILED LEAK RATE TEST.
950033244-01	E12	2E12F041B	FAILED TO CLOSE DURING SURVEILLANCE TESTING.
950042075-01	B21	2B21F032B	FAILED TO CYCLE DURING SURVEILLANCE TESTING.
950028967-01	C11	2C11D001019-107	HCU QUICK DISCONNECT FAILED.
950028968-01	HT	2HT03EE	HEAT TRACE IS NOT ARRANGED CORRECTLY.
950050402-01	E51	2E51F063	HOT TORQUE BONNETT BLOTS.
950028506-01	E22	2E22C003	INCREASED VIBRATION, REPLACE PUMP BEARINGS.
950038601-01	AP	2AP19E-102A	INSPECT AND ADJUST RACKING MECHANISM.
930047327-01	B33	2B33F301B	INSPECT, CLEAN AND REPAIR CHECK VALVE.
930047331-01	B21	2B21F570	INSPECT, CLEAN AND REPAIR CHECK VALVE.
950030399-01	E31	2TDS-E31N614C	INSTRUMENT FAILURE.
940061726-01	H13	2H13P628	REPAIR INSULATED WIRE LUG.

APPENDIX B
REFUELING OUTAGE (L2R06)
SAFETY RELATED CORRECTIVE MAINTENANCE

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950045656-01	C51	2EL-C51AZ008D	IRM D ERRATIC OPERATION AND INDICATION.
950031481-01	C51	2RR-C51R603C	IRM E DIGITAL DISPLAY MISSING SEGMENTS.
950027950-01	B21	2B21F013A	JUNCTION BOX IS DAMAGED.
950004615-01	E12	2E12F065A	GAGGING DEVICE NOT WORKING.
950028966-01	MS	2MS000	LINE HAS ERRODED BEYOND ACCEPTABLE LIMITS.
940078168-01	C51	2C51000	HIGH NOISE LEVELS ON SRMS AND IRMS.
940057292-01	DC	2DC17E	LOST WHEN C/S TAKEN TO EQUALIZE.
950044678-01	NR	2H13P603	LPRM HI ALARM IS UP.
940058672-01	E32	2E32F001N	MAINSTEAM LINES LEAK BY.
950022739-01	G33	2PDIS-G33N025	REPLACE MICRO SWITCH #2.
940058676-01	E32	2E32F002N	MAINSTEAM LINES LEAK BY.
950031471-01	C11	2C11F402A	NO LIGHT INDICATION.
940079588-01	E12	2E12F090B	NO OPEN INDICATION IN CONTROL ROOM.
950045723-01	C51	2RY-C51K601B	NO RESPONSE FROM IRM B ON RANGE 5.
940060399-01	DG	2DG01K	OIL SIGHTGLASS ON LUBE OIL COOLER LEAKING.
950000216-01	B33	2ZSO-B33F020	ONE LIMIT SWITCH FAILING.
950030653-01	B33	2B33F343B	OPEN INDICATION NOT WORKING.
950040481-01	E12	2E12C300C	OUTBOARD BEARING IS HOTTER THAN NORMAL.
950030374-01	AP	2AP71E-C2	REPLACE OVERLOAD AND BREAKER.
950036734-01	E22	2E22F375A	PACKING LEAK.
950041290-01	E12	2E12F008	PACKING LEAK.
950043150-01	IN	2IN114	PACKING LEAK.
940059501-01	E51	2E51F046	PACKING LEAK.
950040080-01	B21	2B21F067D	PACKING LEAK.
950040719-01	B21	2B21F067A	PACKING LEAK.
940060105-01	G33	2G33F040	PACKING LEAK.
940060969-03	DG	2DG060A	PERFORM INSPECTION.
950040854-01	B21	2B21F026A	PIPE CAP AT VALVE IS LEAKING.
950040854-02	B21	2B21F025A	PIPE CAP AT VALVE IS LEAKING.
950022849-01	C71	2PS-C71N002C	PRESSURE TEST SWITCH.
950034809-01	E12	2E12C002B	PUMP SEAL COOLER LINE LEAKING.
950040645-01	IN	2IN038	REGULATOR LEAKING BY SEAT.
950033569-01	DG	2DG03J	RELAY FAILED CALIBRATION.
950014886-01	H13	2H13P632	RELAY IS WORKING INTERMITENTLY.
950025878-01	C71	2C71AK16C	RELAY TIMING IS ERRATIC.
950039188-01	B21	2B21HK1C	RELAY FAILED TO MAKE UP WHEN TRIPPED.
940061317-01	LV	2LV000	REMOVE PROTRUDING CABLES FROM PENETRATION.
940061019-03	E12	2E12D300B	REMOVE END PLATE AND INSPECT.
950032754-01	E12	2E12C300B	INSPECT OUTBOARD PUMP BEARING.
950022348-01	RH	2RH000	REMOVE PIPE CLAMP FOR SUPPORT RH04-2868S.
940061226-01	C11	2C11D001088	REMOVE/ REPLACE CONTROL ROD DRIVE 18-07.
9500[illegible]530-01	AP	2AP02E-8	REPAIR BREAKER GUIDE.
950026239-01	AP	2AP04E-01	REPAIR CRACKED ARC CHUTE.
950032658-01	AP	2AP91E	REPAIR HOLE FOUND IN NON-SEG BUS DUCT.
940081131-02	C51	2C51J003	REPAIR TUBING FROM DRYWELL WALL TO INDEXER.
940061222-01	C11	2C11D001145	REMOVE/REPLACE CONTROL ROD DRIVE 34-31.
940061223-01	C11	2C11D001155	REMOVE/REPLACE CONTROL ROD DRIVE 54-23.
940061228-01	C11	2C11D001181	REMOVE/REPLACE CONTROL ROD DRIVE 38-03.
940061225-01	C11	2C11D001133	REMOVE/REPLACE CONTROL ROD DRIVE 50-35.
950027920-01	AP	2AP75E-D3	REPLACE DEFECTIVE BREAKER.
950026655-01	AP	2AP08E-2	REPLACE DEFECTIVE LUG.
950018720-01	DC	2DC03E	REPLACE ALARM BOARD ON 250V BATTERY CHARGER.
950033547-01	C71	2C71S300D	REPLACE THE LOGIC CARD.
950017190-01	B21	2B21F028A	REPLACE THE 3-WAY NORGREN VALVE.
950020166-01	C71	2C71S003A	REPLACE "A" RPS MG SET EPA LOGIC CARD.
950023756-01	AP	2EI-AP077	REPLACE VOLTAGE RELAY.
940058358-01	E31	2PDS-E31N008A	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
940058358-02	E31	2PDS-E31N009A	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-03	E31	2PDS-E31N010A	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-04	E31	2PDS-E31N011A	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-05	E31	2PDS-E31N008B	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-06	E31	2PDS-E31N009B	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-07	E31	2PDS-E31N010B	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-08	E31	2PDS-E31N011B	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-09	E31	2PDS-E31N008C	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-10	E31	2PDS-E31N009C	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-11	E31	2PDS-E31N010C	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-12	E31	2PDS-E31N011C	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-13	E31	2PDS-E31N008D	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-14	E31	2PDS-E31N009D	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-15	E31	2PDS-E31N010D	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940058358-16	E31	2PDS-E31N011D	REPLACE HI FLOW SWITCH 3 VALVE MANIFOLDS.
940056944-01	H13	2H13P628	RELAY DID NOT ENERGIZE DURING TESTING.
940058078-02	B33	2B33F060A	REPLACE ACTUATOR, INCORRECTLY MACHINED.
950000973-01	B21	2LT-B21N407A	ROSEMOUNT TRANSMITTER ZERO SHIFT OF >.5%.
950028347-01	RE	2RE024	FAILED LOCAL LEAK RATE TEST.
930046719-01	AP	2AP07E	SECONDARY CONTACTS ARE BAD.
950033389-01	B21	2EY-B21K752	TEST POWER SUPPLY.
950033390-01	C22	2JY-C22K601B	TEST POWER SUPPLY.
930047211-01	C11	2C11D045115	DEGRADATION DURING SURVEILLANCE TESTING.
930046621-01	E22	2E22C001	SMALL LEAK ON WEST SIDE OF PUMP.
950039407-01	B21	2B21F032A	SOLENOID BLOWING AIR CONTINOUSLY.
940061960-01	CM	2TE-CM057A	TEMPERATURE INDICATION FAILED.
950027315-01	C11	2C11D001128-112	VALVE FAILED TO GO FULL CLOSED.
950027316-01	C11	2C11D001113-112	VALVE FAILED TO GO FULL CLOSED.
950027317-01	C11	2C11D001142-112	VALVE FAILED TO GO FULL CLOSED.
930042804-01	VQ	2VQ026	ACTUATOR REPLACEMENT.
950019431-01	B21	2PS-B21N015B	SWITCH FAILED RESPONSE TIME TEST.
950017191-01	E31	2PDS-E31N008A	SWITCH WAS ERRATIC DURING SURVEILLANCE.
950017192-01	E31	2PDS-E31N008B	SWITCH WAS ERRATIC DURING SURVEILLANCE.
950043250-01	IN	2IN115	SYSTEM LEAKS.
950024117-01	C22	2JY-C22K601A	TEST DIV I POWER CONVERTERS.
950022091-01	VR	2ZS-VR002AX	RELAY FAILED TO ACTUATE.
950030654-01	E32	2E32F007	TROUBLESHOOT AND REPAIR FOR HIGH AMPS.
950033950-01	B33	2B33F345B	TROUBLESHOOT AND REPAIR.
950046294-01	E51	2E51C001	CONTROL ROOM INITIATED TRIP FAILED TO RESET.
950034208-01	HP	2HP02-2801C	UNPIN PIPE SUPPORT.
950041299-01	E12	2E12F009	VALVE PACKING LEAK.
950040848-01	E32	2E32F001N	VALVE PACKING LEAK.
950043051-01	E51	2E51F091	VALVE FAILS TO CYCLE.
950030184-01	G33	2G33F101	VALVE PACKING LEAK.
930047112-01	E12	2E12F092C	VALVE SEAT LEAKAGE.
950002729-01	B21	2B21N413D-HS	VALVE LEAKAGE.
950050285-01	E51	2LSH-E51N010	VALVE FAILED TO OPEN ON HIGH LEVEL.
930047336-01	B33	2B33F342A	VALVE FAILED TO INDICATE OPEN.
930046196-01	DG	2DG004	VALVE SEAT AND PACKING LEAKAGE.
940061644-01	E12	2E12F332B	VALVE OPERATOR IS NOISY AND HARD TO OPERATE.
950037085-01	E32	2E32F302E	VALVE PACKING LEAKAGE.
950039231-01	C11	2C11D009127	VALVE PACKING LEAKAGE.
940058961-05	E12	2E12F017A	VALVE BODY TO BONNET FLANGE LEAK.
930047223-01	C11	2C11D129101	VALVE LEAKAGE.
950050082-01	C11	2C11D001001-120	DIRECTIONAL CONTROL VALVE LEAKAGE.
940073299-12	B21	2B21F418B	REFURB LIMITORQUE ACTUATOR.
940061172-02	B33	2B33F023A	REPLACE VALVE INTERNALS.

APPENDIX B
REFUELING OUTAGE (L2R06)
SAFETY RELATED CORRECTIVE MAINTENANCE

WORK REQ #	SYS	EQUIPMENT	DESCRIPTION
950016184-01	E12	2E12F041C	VALVE REMAINED OPEN AFTER CYCLING.
950014855-02	E51	2E51F028	FAILED LOCAL LEAK RATE TEST.
950014858-01	E51	2E51F040	FAILED LOCAL LEAK RATE TEST.
950047657-01	E51	2E51F063	VALVE LEAKING.
950004712-02	IN	2IN017	VALVE PACKING LEAK.
950016991-01	VQ	2VQ026	FAILED LOCAL LEAK RATE TEST.
930047564-01	AP	2AP099A	TARGETS FAILED TO DROP ON AN UNDERVOLTAGE.
950025559-01	RH	2RH03-2889X	CLAMP IS NOT MAKING SUFFICIENT CONTACT.