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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
10CFR50.59 EVALUATION SUMMARY REPORT 0009
COMMITMENT MATERIAL CHANGE EVALUATION
REPORT 0004

Gentlemen:

Please find attached the following periodic reports pertaining to CPSES Unit 1 and Unit 2.

(A) Attachment 1 is the report required by 10CFR50.59(b)(2) for activities since February 1, 1999, at CPSES Units 1 and 2. This report contains a brief description of the changes, tests and experiments implemented or performed pursuant to 10CFR50.59(a), including a summary of the safety evaluations for each. Items in this report are referenced by their 10CFR50.59 Evaluation Numbers. This report includes those activities which were completed or partially completed between February 2, 1999, and August 1, 2000, which were not reported to the NRC in a previous submittal. This report also includes certain activities completed or partially completed after August 1, 2000.

(B) Attachment 2 is the CPSES Units 1 and 2 report (Commitment Material Change Evaluation Report 0004) per the recommendations of NRC document SECY-95-300, "Guidelines for Managing NRC Commitments." The tracking document for this process at CPSES is the "Commitment Material Change Evaluation (CMCE)" which identifies the affected commitments and origin, original criteria, proposed changes and the justifications for the changes. This

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report pertains to commitment material changes (in docketed correspondences) which require reporting between February 2, 1999 and August 1, 2000, which were not addressed in the 10CFR50.59 evaluations.

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

Sincerely,

C. L. Terry

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

JDS/js

Attachments

c - E. W. Merschoff, Region IV
J. I. Tapia, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Attachment 1 to TXX-01013

SE-94-079	**Rev. 2**	**SE-99-040**	**Rev. 0**
SE-97-092	**Rev. 0**	**SE-99-041**	**Rev. 0**
SE-98-034	**Rev. 1**	**SE-99-042**	**Rev. 0**
SE-98-043	**Rev. 0**	**SE-99-043**	**Rev. 0**
SE-98-044	**Rev. 0**	**SE-99-045**	**Rev. 0**
SE-98-045	**Rev. 0**	**SE-99-046**	**Rev. 0**
SE-98-048	**Rev. 0**	**SE-99-047**	**Rev. 1**
SE-99-004	**Rev. 0**	**SE-00-001**	**Rev. 0**
SE-99-008	**Rev. 0**	**SE-00-002**	**Rev. 0**
SE-99-011	**Rev. 0**	**SE-00-005**	**Rev. 0**
SE-99-020	**Rev. 0**	**SE-00-007**	**Rev. 0**
SE-99-021	**Rev. 0**	**SE-00-008**	**Rev. 0**
SE-99-025	**Rev. 0**	**SE-00-009**	**Rev. 0**
SE-99-027	**Rev. 0**	**SE-00-010**	**Rev. 0**
SE-99-028	**Rev. 0**	**SE-00-011**	**Rev. 0**
SE-99-029	**Rev. 0**	**SE-00-012**	**Rev. 0**
SE-99-030	**Rev. 1**	**SE-00-013**	**Rev. 0**
SE-99-031	**Rev. 1**	**SE-00-014**	**Rev. 0**
SE-99-032	**Rev. 0**	**SE-00-015**	**Rev. 0**
SE-99-033	**Rev. 0**	**SE-00-020**	**Rev. 0**
SE-99-034	**Rev. 0**	**SE-00-021**	**Rev. 0**
SE-99-035	**Rev. 0**	**SE-00-022**	**Rev. 0**
SE-99-037	**Rev. 0**	**SE-00-024**	**Rev. 0**
SE-99-038	**Rev. 0**		

Evaluation Number SE-94-079 Unit: 1NN
Revision 2

Activity Title: UNIT 1 STEAM GENERATOR ATMOSPHERIC RELIEF VALVE RETROFIT; DM-91-177, REV. 1

Description of Change(s):

This activity will retrofit the existing Unit 1 Steam Generator Atmospheric Relief Valves (ARV's), utilizing CCI DRAG velocity control trim and new actuators. A fire protection line is being rerouted to avoid an interference. Conduits to the actuator limit switches are being upsized and rerouted as required and connecting ECSA's are being changed. Instrument air tubing to the new actuators is also being rerouted as required.

Summary of Evaluation:

The ARV retrofit will provide tighter valve shutoff, which will reduce steam loss and seat damage to the valves. The retrofit will improve controllability of the valves and will also preclude the need for routine stroke length checks. The fire protection line being rerouted will have no adverse effect on the fire suppression coverage in the affected room. Since the ARV retrofit is being designed and installed to the existing design requirements as delineated in the LBD's, this activity will have no adverse impact on any existing structures, systems or components. This activity will not have any affect on accidents, malfunctions or the margin of safety as described in the LBDs.

Evaluation Number SE-97-092 Unit: NN2
Revision 0

Activity Title: DISSOLVED OXYGEN CONTROL IN THE UNIT 2 REACTOR MAKEUP WATER STORAGE TANK

Description of Change(s):

The Demineralized Water System by design is de-aerated in a vacuum de-aerator to remove dissolved oxygen to 0.1mg/L before it is transferred to the Reactor Makeup Water Storage Tank (RMWST). Periodic nitrogen sparging of the Unit 2 (RMWST) has been required to remove dissolved oxygen from the stored water. The current nitrogen supply and injection path for sparging is installed under temporary modification TM 93-2-17. A temporary pressure indication instrument was installed under TM 93-2-18 to measure gas pressure under the tank diaphragm. DM 97-002 will provide permanent nitrogen sparging injection into the Unit 2 RMWST via the reactor makeup pumps mini-flow recirculation line and a permanent local pressure indication instrument for the RMWST diaphragm. Overpressure protection of the Safety Related Class 3 (DD) piping system is provided to protect the Class 3 piping from postulated failure of the nitrogen regulating valve.

Summary of Evaluation:

The sparging operation currently performed under TM 93-2-17 has proven to be effective in reducing the dissolved oxygen content of water stored in the Unit 2 RMWST. High-pressure cylinders continue to be used as the nitrogen supply for this sparging operation. These cylinders require frequent replacement, they are a drain on plant manpower and are a potential source for personnel injury due to handling accidents. In addition, there is some inherent risk to plant equipment and personnel should a pressurized gas cylinder rupture, causing it to become a projectile. Installation of DM 97-002 will utilize the existing low pressure (100 psi) plant nitrogen system for the source of sparging allowing the permanent removal of the gas cylinders and the restoration of Temp Mods 93-2-017 and 93-2-018. Installation of the permanent nitrogen supply connection does not provide any increase in the probability of an accident or malfunction of equipment important to safety. It does not increase the consequences of an accident or the failure of equipment important to safety. The change does not create the potential for a new accident or failure.

Evaluation Number SE-98-034 Unit: 1N2
Revision 1

Activity Title: REROUTE CPX-SAAPDP-01 & CPX-SAAPDP-02, AUXILIARY STEAM DRAIN PUMPS, DISCHARGE TO TURBINE BUILDING SUMP #2

Description of Change(s):

Correct FSAR to state Contaminated Condensate can be removed from the Auxiliary Drain Tank by draining condensate to the Liquid Waste Management System.

Summary of Evaluation:

There is no impact on plant safety to reroute the Auxiliary Steam Drain Pump (CPX-SAAPDP-01 & CPX-SAAPDP-02) discharge to Turbine building Sump #2. This function to preserve condensate is non-safety related function. The lost condensate is not credited in accident analysis. Removing this volume of condensate from the Auxiliary Drain Tank by transferring to the Waste Management System via the Turbine Building Sump #2 is within the current design basis and licensing basis. Removing this volume of condensate from the Auxiliary Drain Tank by transferring to Liquid Waste Management System is within the current design basis and licensing basis.

Evaluation Number SE-98-043 Unit: NN2
Revision 0

Activity Title: CHANGE SET PRESSURE OF RELIEF VALVE 2-8857 AND INSTALL A DIFFERENT SEAL IN VALVE 2-HCV-0943 IN THE SI ACCUMULATOR N2 SUPPLY SYSTEM

Description of Change(s):

The proposed activity involves changing SI Accumulator Nitrogen Supply Header Relief Valve 2-8857 set pressure from 700 psig to 750 psig, installing a different spring and washer set needed for the revised set pressure, and replacing the existing seal with a different seal in SI Accumulator Nitrogen Vent Control Valve 2-HCV-0943.

Implementation of this modification will restore leak tightness of the above valves to eliminate the continuous pressurizing and relieving of the SI Accumulator nitrogen supply and vent process currently being experienced in Unit 2 containment.

Summary of Evaluation:

Consideration has been given for all potential failure modes for this activity, and it has been determined that there are no credible failure modes that could adversely affect system, structures, or components resulting in an increase in the probability, severity, or consequences of any accident analyzed in LBDs. This activity does not adversely affect any system used for accident mitigation and will not impact plant response to an upset, emergency, faulted condition.

Evaluation Number SE-98-044 Unit: 1NN
Revision 0

Activity Title: RE-ROUTE HEATER DRAIN RECIRCULATION LINES FOR UNIT 1 (DM98-031/1RF07)

Description of Change(s):

The Unit 1 Heater Drains System is being modified to improve system reliability and to reduce the possibility of water hammer due to the starting of the heater drain pumps CP1-HDAPDP-01 and CP1-HDAPDP-02. There is a possibility of water hammer in the Heater Drain system during transients, when the Heater Drain Pumps (HDP) are started, and when the recirculation valves open. This modification is similar to the reroute being done to Unit 2 in DM 97-044.

This modification will ensure the recirculation piping is water solid at all times by moving the piping from the top of Heater Drain Tanks (HDT) to the new 30" equalizing piping installed during 1RF06. This modification will also relocate the Heater Drain Pump recirculation valves, 1-FV-2589A and 1-FV-2589B and their associated isolation valves. The two recirculation lines downstream of the control valve station isolation valves are combined into one 16" header which flows to the 30" header. The re-designed pump recirculation control stations and piping will be entirely below the water level of the HD tanks.

Summary of Evaluation:

The modifications will improve system reliability and reduce water hammer associated with the starting of the HDPs, or the opening of the recirculation valves. This modification ensures the recirculation lines are water solid all the time after the initial fill. Relocating HD pump recirculating control valves, and the entire recirculating lines to evaluation below HD tank water level, will eliminate any void to form after venting the lines. This water solid recirculation subsystem will reduce water hammer concerns during pump start-up, and/or system pump restart. The addition of drain valves, between pump recirculating control valve and inlet isolation valve, will facilitate any maintenance/repair activity to be performed on the recirculating control valves during power operation.

All the systems involved in the modification are non-safety, have no protective functions and cannot impair the ability of protection systems to function. The only accidents in Chapter 15 that could be impacted by the SSCs associated with this proposed activity are those described in Section 15.1, "Increase in Heat Removal by the Secondary System" and Section 15.2, "Decrease in Heat Removal by the Secondary Systems". The probability of occurrence of these accidents is expected to decrease due to this activity. There is no unreviewed safety question.

Evaluation Number SE-98-045 Unit: 1NN
Revision 0

Activity Title: DM 96-113; INSTALLATION OF TELESCOPIC JIB CRANE IN UNIT 1 CONTAINMENT TO PERFORM MISC. LOAD HANDLING ACTIVITIES DURING PLANT OUTAGES

Description of Change(s):

This activity installs a Telescopic Jib Crane in Unit 1 Containment. It is supported by structural steel support, mounted on east-west divider wall between Steam Generator compartments at El. 905'-9". Two such supports are provided one each between Steam Generator compartments 1 & 4 and 2 & 3. Crane can be located, as required, at either of these two supports. This crane will only be used during Plant outages modes 5, 6 and defueled. Jib crane and it's supports are non-nuclear safety related, Seismic Category II. Non-Safety Plant Support Power (plant power from 25kV loop) will be used to provide electrical power to the crane. Plant support power panel Tag No. CP1-EPDPNB-24 will be the power source.

Summary of Evaluation:

The modification installs new load handling equipment in Unit 1 Containment in the form of Telescopic Jib Crane. This will help to reduce demand on Polar Crane during outages and this crane will be available as a back-up to Polar Crane in the event of temporary failure of the Polar Crane. The jib crane, it's supports and platform around the crane at EL 905'-9" are designed Seismic Category II and therefore meet the requirements of current design and licensing bases of CPSES. Also the east-west divider wall of Steam Generator compartments has been evaluated for additional loads due to Jib crane and the wall is determined to be structurally adequate to support the additional loads. Hence this activity does not introduce any new credible potential failure modes.

The Electrical Power addition is designed such that all requirements pertaining to cable sizing, and protection are met in accordance with existing design bases and no new credible potential failure modes are introduced. Heavy Loads movements performed by this crane will be in accordance with Heavy Load Program requirements.

Evaluation Number SE-98-048 Unit: 1X2
Revision 0

Activity Title: SETPOINTS FOR TORNADO DAMPERS IN UPS HVAC CHASE

Description of Change(s):

Section 3.3 of the FSAR currently states that the interior tornado dampers at CPSES are normally closed and are set to open when a 3" w.g. differential pressure across the dampers is reached.

There are four tornado dampers (CPX-TVABTD-13, 14, 15 and 16) located in the UPS HVAC chase which would not remain closed during normal operations at this low pressure differential setpoint. The opening of these dampers results in an undesired HVAC flow path. In order to prevent this condition, the setpoint of these four tornado dampers was increased so that they would open at 7.5" w.g. instead of the 3" w.g.

This activity revises section 3.3 of the FSAR and DBD-ME-009 to include the increased tornado damper setpoints.

Summary of Evaluation:

The Electrical and Control building (E&C) walls and floor slab pressure differentials resulting from a Design Basis Tornado (DBT) with the increased damper setpoints are acceptable. The loads imposed on the UPS HVAC system by the increase in the four damper setpoints does not result in the failure of any system, structure or components. There are no credible failure modes that could be introduced by implementation of this activity. Based on the results of the evaluation, implementation of the proposed activity is acceptable.

Evaluation Number SE-99-004 Unit: 1N2
Revision 0

Activity Title: Change MDAFWP miniflow orifices to ones that flow nominal 200 gpm. (Unit 2 installed during 2RF06. Unit 1 to be installed during 1RF08.)

Description of Change(s):

This mod replaces the nominal 100 gpm Recirc Breakdown Orifices with ones rated at 200 gpm nominal (225 gpm max). Increasing the MDAF Pumps' minimum recirculation flow rate is expected to minimize flow-induced noise and vibration experienced in the suction and discharge piping of the pumps.

Summary of Evaluation:

This safety evaluation shows that the MDAFPs will continue to perform their normal and accident functions with the higher capacity Recirc Breakdown Orifices, while the new orifices are expected to minimize the flow-induced noise and movement in the pump suction and discharge piping. This mode does not introduce any new failure modes or unbounded events. This mode does not impact accidents or malfunctions previously evaluated, nor are events of a different type created. Tech Spec. acceptance limits and basis are not affected. All applicable limits of the AFW System as described in the Tech Specs, licensing basis and DBD remain valid. Safety analysis are unaffected and therefore there are no effects on margins of safety.

Evaluation Number SE-99-008 Unit: 1N2
Revision 0

Activity Title: REVISE TURBINE STOP VALVE SURVEILLANCE TESTING INTERVALS

Description of Change(s):

The current CPSES surveillance requirement 4.3.4.2a for TS 3/4.3.4, "Turbine Overspeed Protection", requires cycling each high and low pressure stop and control valve once per 6 weeks using the manual test or the Automatic Turbine Tester. Surveillance 4.3.4.2c requires the direct observation of the movement of the above turbine valves through one complete cycle once per 6 weeks. The surveillance testing requires moving each turbine valve through one complete cycle and is typically performed by a control room operator with an observer at the valve. The test verifies freedom of movement of the valve components and is beneficial in detecting problems with valve operation and identification of gross outward appearance of valve condition. The surveillance requirement ensures that all turbine steam inlet valves are capable of closing to protect the turbine from excessive over speed which could generate potentially damaging missiles. To minimize the effects while assuring proper protection against overspeed, TXU Electric has received from Siemens a recommendation to increase the valve testing frequency to 12 weeks. This increase in testing frequency contains provisions for additional monitoring of the stop valves and that no degradation is observed. This recommendation was based on a quantitative evaluation performed by Seimens on the probability of failure of the overspeed trip and protection system as a function of the turbine stop and control valve test interval. CPSES will install the additional monitoring sensors before each test.

Summary of Evaluation:

This request proposes to revise the surveillance requirements TRS 13.3.33 entitled "Turbine Overspeed Protection" in the Comanche Peak Steam Electric Station Unit 1 and Unit 2. TRS 13.3.33.2 is revised to require cycling of each high and low pressure turbine stop and control valve once every 12 weeks using the manual test or the automatic turbine tester. Also, Surveillance TRS 13.3.33.3 is revised to require direct observation of the movement of the above turbine valves through one complete cycle once every 12 weeks. FSAR Sections 3.5 and 10.2 require revision also to consistently reflect the testing intervals. Based upon the ACPSI Report ER-504, and the updated stop and control valve failure probability, it is concluded that the implementation of this TRM surveillance revision will not increase the probability or consequences of an accident previously analyzed. The revision of the surveillance results in a net improvement in plant safety by reducing the likelihood of plant trips and stress and wear on plant components.

Evaluation Number SE-99-011 Unit: 1N2
Revision 0

Activity Title: TEST THE GAS COMPRESSORS IN THE GASEOUS WASTE HOLDUP SYSTEM TO SIMULATE REMOVAL OF GAS FROM THE PRESSURIZER RELIEF TANK TO 2 PSIG

Description of Change(s):

This activity is a proposed test of the Gaseous Waste Holdup System (GH) in order to simulate a desired operating practice that may be used for venting the Pressurizer Relief Tank (PRT) during upcoming refueling outages. The waste gas compressors in the GH are to be tested to determine if and how to best operate the GH when it is desired to remove gases from the PRT at a lower than normal system pressure. The GH design is limited to typically remove radioactive gases in the PRT down to a pressure of approximately 7 psig. However, by conducting this test which will involve a specific set of initial GH conditions and an alignment between two different GH Gas Decay Tanks, it is expected to be able to simulate the desired PRT gas venting conditions, observe GH operation, and make adjustments to allow optimized removal of gas from the PRT down to approximately 2 psig. This test will verify that the GH can maintain stable operability at these conditions.

Summary of Evaluation:

The sole safety function provided by the GH is the retention of radioactive gases for decay. The GH is designed so that a single failure in the system will not result in an offsite dose which exceeds the 0.5 REM criterion that is described in the NRC Branch Technical Position ETSB 11-5 and the site boundary requirements of 10CFR100. To meet these requirements, the total radioactivity in a single GH Gas Decay Tank (GDT) is limited to 200,000 Ci noble gas as specified in the CPSES Licencing Basis Documents. Additionally, a limit of 3% oxygen is imposed whenever hydrogen concentration exceeds 4%. This limitation precludes an explosive/flammable gas mixture in the Gaseous Waste Processing System (GWPS) and provides assurance that releases of radioactive materials will be controlled in conformance with the requirements of GDC 60 of 10CFR50 Appendix A.

This test is designed to determine if the GH will remain stable enough to remove more gas from the PRT during refueling outage operations, i.e., allow a pump down of the PRT to approximately 2 psig. vis 7 psig. Presently the minimum pressure that the waste gas compressors can pump the system down to is approximately 7 psig.

This test does have a minor risk associated with it in that a rupture disk between the phase separator and GDT-10 could be degraded or rupture as a result of a compressor trip. (If the rupture disk fails the radioactive gas goes to GDT-10.) There is also a minor chance that surging could occur in the compressor, but not to the point that seals could be failed or the compressor or motor could be damaged. There is no possibility that the Technical Specifications would be violated or the Safety Function of the GH system would be lost or challenged. There is a possibility that a false high product hydrogen alarm could be generated and the compressor could trip due to lost flow inertia.

Expected results (and test termination criteria) are that the normal compressor suction pressure control valve will go full open and/or a low flow alarm for the hydrogen recombiner on the GH control panel could lock in.

However, there are no new unreviewed safety questions involved, nor are there any systems, structures or components important to safety that are degraded or affected by this test.

Evaluation Number SE-99-020 Unit: 1NN
Revision 0

Activity Title: CHANGES IN THE UNIT 1 EMERGENCY DIESEL GENERATORS (EDGS) INSTALLATION

Description of Change(s):

This DM makes the starting of Unit 1 EDGs as an "emergency start" due to the associated bus under voltage, automatically repositions the machine frequency to 60 HZ on receipt of emergency start signal. The DM also replaces a control switch on CB11 with a pull to lock type control switch to enable the operator to manually stop the engine from control room even if it was started in an emergency mode. The DM also makes modifications to the EDGs starting circuitry (150 psi interlock) so that the EDGs trips (with the exception of engine overspeed and generator differential protection) are not reinstated when the starting air pressure falls below 150 psi and the EDG starts in an emergency mode.

Summary of Evaluation:

The impact of this DM on the performance of EDGs, 125 DC system, 118 V AC system and seismic qualification of EDGs control panels were reviewed. It has been determined that there are no credible failure modes associated with the implementation of DM 98-054 activities. The modifications make the EDG installation compliant with IEEE 387-1977, Reg. Guide 1.9, FSAR and CPSES Design Basis documents. Engine emergency start/stop switch use is not restricted under plant normal operating conditions. However, its use to stop the engine under off normal conditions will be administratively controlled. Unwanted stopping of the engine is precluded by changing the existing "Emergency start/stop switch" to "Pull to lock" stop type. This switch returns to the normal position if it is not in the pulled to lock position.

DM 98-054 modification addresses the concerns identified in ONE FORMS 98-1220, 98-1221, and 98-1305.

To resolve the concern on ONE FORM 98-1221, the EDG start due to LOOP has been made as an emergency start. The emergency start by passes delay circuitry in place, following an engine shutdown, and start the engine immediately. In emergency Start of the EDG all trips with the exception of the engine overspeed and generator differentials are also automatically bypassed. The bypassing of these trips on LOOP start resolve part of the problem identified in Feedwater Line Break ONE FORM 98-1217 (SMF 98-448) and DM 98-054 modifications preclude the possibility of loss of a motor driven AFWP due to non critical trips. Since, DM 98-054 modifications take credit to resolve part of the problem identified in SM 98-0448 a Tech Spec Change TS-99-004 has been initiated. NRC's approval of this TS change is required for DM 98-054. This TS changes requires that every 18-month verification be made that DGs automatic trips except the engine overspeed and generator differential are bypassed on actual or simulated; i) loss of voltage on the emergency bus and ii) SI signal.

Based on the above evaluation, it is concluded that the implementation of DM 98-054 does not involve in an unreviewed safety question.

Evaluation Number SE-99-021 Unit: 1NN
Revision 0

Activity Title: CHANGE SET PRESSURE OF RELIEF VALVE 1-8857 AND INSTALL A DIFFERENT SEAL IN VALVE 1-HCV-943 IN THE SI ACCUMULATOR N2 SUPPLY SYSTEM

Description of Change(s):

DM 97-64, DCN 13016 for Relief Valve 1-8857, and DCN 13017 for vent Control Valve 1-HCV-943. The proposed activity involves (i) changing SI Accumulator Nitrogen Supply Header Relief Valve 1-8857 set pressure from 700 psig to 750 psig and installing a different spring and washer set needed for the revised set pressure, and (ii) replacing the existing seal with a different seal in SI Accumulator Nitrogen Vent Control Valve 1-HCV-0943.

Implementation of this modification, DM 97-64 will restore leak tightness of the above valves to eliminate continuous pressurizing and relieving of the SI Accumulator nitrogen supply and vent process.

Summary of Evaluation:

Consideration has been given for all potential failure modes for this activity, and it has been determined that there are no credible failure modes that could adversely affect system, structures, or components resulting in an increase in the probability, severity, or consequences of any accident analyzed in LBDs. This activity does not adversely affect any system used for accident mitigation and will not impact plant response to an upset, emergency, or faulted condition.

Evaluation Number SE-99-025 Unit: 1NN
Revision 0

Activity Title: DM 98-056, MIDLOOP LEVEL INSTRUMENTATION AND REACTOR COOLANT SYSTEM VENT TUBING

Description of Change(s):

This design modification provides a more reliable midloop level monitoring system. The system uses a digital quartz pressure transducer to measure reference pressure at the pressurizer vent and/or reactor head vent, and a liquid head pressure at the number 4 reactor coolant loop. The pressure signals are transmitted to a remote computer system which determines the Reactor Vessel water level. Indication and display are provided remotely in the control room. The following is also be included as part of this design modification: installation of vent tubing from the reactor head to the containment ventilation system with associated isolation valves; new flange connections with a threaded tail pipe on the Pressurizer Relief tank; modification of pressurizer level tree test connections to support the installation of the vent tubing; and the addition of vent valves on 1-LT-3615A, B and C to prevent spillage during instrument venting. This permanent installation of vent tubing and vent valves will reduce the manpower required to install tygon tubing during outages, thus reducing personnel exposure and spread of contamination.

Also, there is an isolation valve (1RC-8154) installed downstream of steam traps CP1-RCSTDR-01, -02 & -03. This isolation valve is used during the filling and venting activities performed during the outage by Operations personnel. Once the activities are completed, the valve will be "locked open."

Summary of Evaluation:

The Midloop Level Monitoring System (MLMS) uses digital quarts pressure transducers to measure reference pressure at the pressurizer vent and/or reactor head vent, and a liquid head pressure at the number 4 RCS loop. All conduit supports and instruments were installed per design specifications and have no adverse affect on any of the existing SSCs. The system only provides indication for monitoring the RCS level in shutdown conditions. Loss of the system does not induce any failure modes. The MLMS installation is isolated in MODES 1-4 and cannot initiate any licensing basis accidents, nor is the probability of a license basis accident affected. Because all accident analyses remain valid and are unaffected by installation of the MLMS, the bases for Technical Specifications remain valid. In addition, because the accident analysis is unaffected, the margin of safety is unaffected by the installation of the MLMS.

Evaluation Number SE-99-027 Unit: NN2
Revision 0

Activity Title: UPDATE OF FSAR TABLE 6.2.2-1, TABLE 6.5-5, SECTION 6.5.2.2.3 AND FSAR FIGURE 6.5-4

Description of Change(s):

Update the number of nozzles on Unit 2, Train A, Region B (on Line # 6-CT-2-124-301R-2) in the FSAR to reflect the actual number of nozzles in the field.

Summary of Evaluation:

The update to the FSAR to reflect the actual Unit 2 plant configuration remains bounded by the current analysis. There are no credible potential failures that have not already been evaluated by bounding calculations. The Unit 2 containment spray systems analysis for accident mitigation remain bounding . (This includes the iodine removal rate and the pressure/temperature analysis). The radiological consequences are unaffected since the number of nozzles in Unit 2 was not a controlling factor in the analysis. The containment spray system function is to mitigate the effects of a LOCA or MSLB. One less nozzle in the Containment Spray System does not significantly affect the capability of the system nor does it affect the margin of safety analysis.

Evaluation Number SE-99-028 Unit: 1N2
Revision 0

Activity Title: REMOVE INEQUALITIES FROM NOMINAL TRIP SETPOINTS FOR REACTOR PROTECTION SYSTEM AND ESFAS INSTRUMENTATION

Description of Change(s):

The proposed activity is the revision of ITS (TS Amendment 64) Bases Tables B3.3.1-1 and B 3.3.2-1 and associated text to reflect that the nominal trip setpoints for the Reactor Protection System and ESFAS instrument are nominal values and, as such, should not have inequalities associated with the specification of the setpoint. The relevant ITS Bases discussions are also reviewed to more accurately discuss the relationship between the nominal trip setpoint, the Allowable Value and the setpoint methodology.

Summary of Evaluation:

If the Improved Tech Specs (Amendment 64), the Limiting Safety System Setting (LSSS) for the Reactor Trip System and the operability limits for the ESFAS instrumentation is explicitly defined to be the Allowable Value. During the development of the Improved Tech Specs (Amendment 64), only the Allowable Value was retained in Tech Specs, and the trip setpoints for these instruments were relocated to the Tech Spec Bases. In the current TS, these trip setpoints are provided with inequalities; however, the use of inequalities in both the current and improved TS format is inconsistent with the approved setpoint methodology, in which two-sided calibration tolerances about the nominal trip setpoint are allowed. Therefore, the inequalities are removed from the ITS nominal trip setpoints. The relevant ITS Bases discussions are also revised to more accurately discuss the relationship between the nominal trip setpoint, the Allowable Value and the setpoint methodology.

Because this change is consistent with both Amendment 64 to the CPSES Tech Specs and the approved methodology for calculating Reactor Trip System and ESFAS instrument setpoints and Allowable Values, it is concluded that no Unreviewed Safety Question exists.

Evaluation Number SE-99-029 Unit: 1N2
Revision 0

Activity Title: INSTALLATION OF SCREENS ON THE AIR INTAKE OPENINGS OF THE SERVICE WATER INTAKE STRUCTURES (SWIS) TO PREVENT INGRESS OF INSECTS

Description of Change(s):

Design Modification 99-023 will install mesh cages to the inlet ventilation floor openings in front of each Service Water Pump and south of the Fire Pumps. Remaining pipe penetrations will be covered with rubber boots, mesh screens or Bisco seals. Existing chemical injection lines will be rerouted for ease of grating removal and piping supports will be modified due to the piping reroute. Pipe unions will be installed in each drain line from the electric fire pump and jockey fire pump north of the eastern 2' x 4' ventilation floor opening. The abandoned chemical injection lines at the 36" sparger penetration below floor elevation 796' will be cut and capped. The existing grating will be scrapped and replaced with aluminum grating.

Summary of Evaluation:

The SSCs that could be affected by design modification implementation are the Service Water Intake Structure (SWIS), the SWIS Ventilation System intake openings, the Station Service Water Pumps, and the Fire Pump in the SWIS. The system parameter that could be affected is the SWIS temperature which is limited by Technical Specification 3/4.7.10 to 127 degrees F during normal conditions and 131 degrees F during abnormal conditions.

The proposed activity will maintain existing air flow rates (even under 50% plugging) and will not resist the SWIS Ventilation System intake openings such that the SWIS temperature limits are exceeded. Operating procedures will ensure plugging does not prevent adequate air flow. The screens are classified NNS, Seismic Category II. The screens also will be designed for the differential pressures associated with MELB flooding and tornado wind pressure. The screens will not increase the probability of failure or consequences of malfunctions of safety related equipment described in the Licensing Basis Documents. The proposed change will have no impact on the Technical Specification acceptance limit, and therefore will not decrease the margin of safety.

Note: TS 3/4.7.10 of NUREG-1468 is transferred to TRM table 13.7.36-1 after implementation of ITS based on NUREG-1431.

Evaluation Number SE-99-030 Unit: 1NN
Revision 1

Activity Title: INSTALLATION OF THROTTLING VALVES IN THE REACTOR COOLANT PUMP SEAL #3 STANDPIPE FILL LINES

Description of Change(s):

Design Modification 99-003, Revision 0, will install two 3/4" socket welded throttling valves in the Reactor Coolant System (1RC-8155 and 1RC-8156). One valve will be installed in each of the Reactor Coolant Pump Seal #3 standpipe fill lines (RC-1-125 and RC-1-126 respectively). During normal operation, level control valves 1-LCV-178, 1-LCV-179, 1-LCV-180 and 1-LCV-181 cycle to allow the Reactor Coolant Pump Seal #3 standpipes to fill. Extensive testing has determined that the cycling of these LCVs causes a pressure transient to occur which lifts relief valves 1RC-0036 and 1DD-0600. Installing the new valves in the LCV supply headers and throttling the valves to the flow required to maintain the RCP#3 seal should minimize any pressure transients, helping to eliminate the spurious lifting problems currently experienced by the relief valves.

Summary of Evaluation:

Installation of the throttling valves in the supply headers to the RCP Seal #3 Standpipes does not impact the ability to makeup sufficient seal water inventory to the seal standpipes in a timely manner. Evaluation indicates that installation of the throttling valves will not adversely affect the operation of the safety related Reactor Coolant System (nor the CVCS or DD systems). As a result of this activity, there is no increase in the probability of a Licensing Basis Accident or safety related equipment malfunction, nor is there a potential for creating a previously unanalyzed event. Also there is no increase in the consequences of a previously analyzed event. As a result of this activity, there is no change to the Technical Specifications.

Evaluation Number SE-99-031 Unit: 1N2
Revision 1

Activity Title: STA-758 & FSAR change - Incorporate changes recommended by the NRC GL 99-002," Lab. Testing of Nuclear Grade Activated Charcoal."

Description of Change(s):

The activity involves implementing changes to the FSAR and procedures for Comanche Peak Steam Electric Station (CPSES) based on the Generic Letter (GL99-02) written by the NRC to Licensees of PWRs & BWRs utilizing activated charcoal in their filtration systems to maintain dose to the operator and public within guidelines of GDC 19 and 10CFR100.

No Technical Specification change is required to implement the testing requirements of ASTM D3803-1989. However, a Technical Specification change request (LAR 99-007, TXX-99231) referencing ASTM D3803-1989 was submitted to the NRC within 180 days of the issuance of GL 99-002.

The NRC intends to exercise the enforcement discretion, consistent with section VII.B.6 of the Enforcement Policy until a TS change is approved.

Summary of Evaluation:

There will be no system or accident analysis parameters affected by this activity. This activity only involves the testing of activated charcoal. The systems will not experience any changes whatsoever in the manner they are operated and what is expected from their intended design function. The systems will continue to operate at their design operating parameters and generate their design flow rates through the filtration units.

There are no failure modes associated with this activity. The activity only involves the testing performed on activated charcoal to determine its ability to adsorb and either to remain in place or be replaced.

Evaluation Number SE-99-032 Unit: 1N2
Revision 0

Activity Title: Update FSAR, TS Bases, DBD & device setting drawings to elaborate on functions and settings of relays used in LOP DG start instrumentation.

Description of Change(s):

LOP DG start instrumentation relay descriptions are revised to elaborate the function of the relays, and their settings are updated per RG 1.105 Rev. 2 to assure that the tech spec allowable and safety limit values will not be exceeded.

Summary of Evaluation:

The relay settings will assure that Class 1E equipment will not be exposed to damaging voltages as a result of degraded grid, the DG will start if the alternate source does not restore adequate voltage, and on bus reenergization the Class 1E motors will not experience damaging voltages. Also the normal relay drifts will not cause Tech Spec allowables to be exceeded and the settings will assure that the equipment voltage experienced will remain within the safe voltage limits.

Evaluation Number SE-99-033 Unit: 1N2
Revision 0

Activity Title: SMF 1999000118 "NON CONSERVATIVE TECHNICAL SPECIFICATION SURVEILLANCE CRITERIA"

Description of Change(s):

This activity involves a revision to the Bases section of Technical Specifications regarding the auxiliary feedwater pumps TDH acceptance criteria. In addition, since the surveillance criteria is also depicted in design basis document DBD-ME-206 Rev. 9, this activity also includes the revision of this DBD.

The acceptance criteria for the AFW Motor driven pumps is being changed from 430 gpm at 1372 psid to 430 gpm at 1380 psid. The before and after values for the Motor Driven pumps shown above does not include instrument uncertainties. The uncertainties are accounted during the testing surveillance.

The acceptance criteria for the AFW Turbine driven pumps is being changed from 860 gpm at 1450 psid to 860 gpm at 1438 psid. The TDH of 1450 psid included the instrument uncertainties. The new value of 1438 does not include instrument uncertainties, therefore, it will be necessary to account for these uncertainties during the surveillance testing.

These changes were necessary because the computer model which was utilized to obtain the original acceptance criteria did not contain sufficient resistance to account for the installed steam generator flow control valves.

Summary of Evaluation:

For all the pumps, the acceptance criteria was increased (made more stringent) this however, is not obvious in the Turbine driven case because the new acceptance criteria does not include the instrument uncertainties.

Although the changes to the acceptance criteria results in a change to the procedures as described in the Technical Specification Bases, there is no effect on the probability of malfunctions, events, or accidents as described in the licensing basis. The consequences of accidents as described in the LBDs are not affected by this change because the more stringent test criteria ensures that the flow/head requirements from the auxiliary feedwater systems are met for the various accident analyses.

Evaluation Number SE-99-034 Unit: 1NN
Revision 0

Activity Title: REVISE UNIT 1 LP TURBINE INSPECTION INTERVAL FROM 50,000 HOURS

Description of Change(s):

The LP Turbine disk inspection as prescribed by the vendor is normally at 50,000 operating hours. This activity prescribes a one time deviation to extend LP turbine 1-02 inspection interval to 35,000 hours. The need to extend the inspection interval is to avoid performing a disk inspection on both LP turbines during 1R07.

Summary of Evaluation:

The LP turbine disk inspection as prescribed by the vendor is normally at 50,000 operating hours. This activity prescribes a one time deviation to extend LP turbine 1-02 inspection interval to 65,000 hours. The interval of inspection is captured in FSAR 10.2.3.6 and 3.5.1.3 and is part of the basis of the turbine missile analysis for CPSES. The missile analysis was performed to ensure the integrity of systems, structures and components. The original CPSES analysis used the vendor engineering reports to determine the probability of missile generation from an LP turbine disk burst. This activity probes for the extension of the inspection interval by the use of a new vendor engineering report which provides a probability of missile generation from the Unit 1 LP turbines. The new report uses improved calculation methods and actual inspection data from previous CPSES LP turbine inspections.

The conclusion of this evaluation is that LP turbine 1-02 can have a one time extension to its inspection interval to increase to 65,000 hours without increasing the probability of a turbine missile and does not introduce an unreviewed safety question.

Evaluation Number SE-99-035 Unit: 1X2
Revision 0

Activity Title: INSTALLATION OF TM TO SUPPLY NITROGEN GAS TO THE COMMON RMUWP (REACTOR MAKEUP WATER PUMP)

Description of Change(s):

The temporary modification that is currently on the Unit 2 Reactor Makeup Water (RMUW) pump will be transferred to the Common RMUW pump. The purpose of the temporary modification will be to inject nitrogen into the recirculation line returning to the RMUW storage tank to scrub excess oxygen from the system.

Summary of Evaluation:

The addition of nitrogen gas will be injected into the common RMUWP mini-flow header downstream of the flow orifice. This allows the Dissolved Oxygen content to be reduced below 100 ppb without having to replace the water in the Reactor Makeup Water Storage Tank. The result will be maintaining water chemistry without increasing any significant to the plant or challenging the Safety Systems. There are no new unreviewed safety questions, nor are there any systems, structures, or components important to safety that are degraded or affected by this modification.

Evaluation Number SE-99-037 Unit: 1NN
Revision 0

Activity Title: PERFORM TWO TRAIN CCW SYSTEM OUTAGE DURING 1RF07 & SUPPLY COOLING SF HEAT EXCH. FROM U2 CCW. ALSO EVALUATE EFFECTS OF UNIT 1 DUAL TRAIN

Description of Change(s):

Spent Fuel Pool cooling is normally supported by the CCW from both Units 1 and 2. A two train CCW system outage in Unit 1 during 1RF07 is required to perform maintenance work activities on the Unit 1 CCW system. During the time that Unit 1 CCW is not available, CCW cooling to the Spent Fuel Pool Cooling and Clean-up System (SF) heat exchangers will be provided from Unit 2 CCW only. This Safety Evaluation addresses the Unit 2 plant conditions during the Unit 1 two train CCW outage evolution, what responses were required for various possible accident or failure scenarios, and the Defense-In-Depth provisions that was in place to perform this activity. This Safety Evaluation was required for the potential deviations from plant processes and configurations as described in the FSAR for Unit 2 and SF.

During 1RF07, a dual train EDG outage was taken and the effects of this on Unit 2 is also being evaluated.

Summary of Evaluation:

This Safety Evaluation evaluates the plant condition required to perform 1RF07 maintenance which requires that both Unit 1 CCW trains be shutdown. CCW cooling to the Spent Fuel Pool Cooling and Clean-up heat exchangers was provided by the operating Units (Unit 2) CCW. This evaluation also evaluates the effects of a dual train Unit 1 EDG outage on Unit 2. This evaluation considers the following potential plant accidents, events, or equipment failures: 1) Loss of SF cooling due to SF equipment failure; 2) Loss of SF cooling due to CCW equipment failure; 3) LOCA in Unit 2 resulting in loss of SF; 4) Fire causing loss of SF cooling; 5) LOCA in Unit 2 coincident with a Single Active Failure; 6) Unit 2 Fire Safe Shutdown; 7) Unit 2 RSB 5-1 cooldown; 8) Unit 2 forced plant cooldown; 9) Unit 2 Station Blackout; 10) CCW pump run out under various plant configurations. This evaluation shows that during the time period when both trains of Unit 1 CCW is not available, and when both Unit 1 trains of EDG are not available, Unit 2 and Spent Fuel Pool cooling continued to meet Licensing Basis requirements. This was accomplished with the Defense-In-Depth provisions that was implemented as part of the 1RF07 outage plan. This activity does not involve an Unreviewed Safety Question.

Evaluation Number SE-99-038 Unit: 1N2
Revision 0

Activity Title: UPDATING THE APPLICATION OF IEEE 450 FROM VERSION 1980 TO VERSION 1995 FOR CLASS 1E BARRIERS

Description of Change(s):

This change to the FSAR, Technical Specification Bases and DBD-EE-044, Revision 9, implements the more recent version of Standard IEEE 450 "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications" on the procedures for the Class 1E batteries. The earlier version (1980) was referenced in the Basis for the Improved Technical Specification; the Bases will now reference the 1995 version as being applied to the Class 1E Batteries.

Summary of Evaluation:

There are no expected effects on the performance of the Class 1E batteries from this change. Five items associated with this change meet the definition of a trivial change because they make the FSAR consistent with the Improved Technical Specifications (ITS). The effects of changing reference in BASES for the CPSES Technical Specification from IEEE-450 (1980) to IEEE -450 (1995) on the Technical Specification Bases are nonexistent or, in two cases, the more conservative approach is retained. The remaining items from the proposed activity (additional inspections of the Class 1E batteries and deletion of periodic equalizing charges) will not degrade or prevent the same level of performance by the Class 1E batteries. No new failure mode will be introduced by the proposed activity. Better or equivalent performance of the Class 1E batteries will be maintained.

Evaluation Number SE-99-040 Unit: 1NN
Revision 0

Activity Title: ALLOW TEMPORARY USE OF A NITROGEN GAS BLANKET IN THE REACTOR COOLANT DRAIN TANK 24 HRS PRIOR TO A UNIT SHUTDOWN FOR REFUELING

Description of Change(s):

Change station procedures and the FSAR to allow a temporary introduction of nitrogen to the Reactor Coolant Drain Tank (RCDT) approximately 24 hours prior to a scheduled Unit shutdown for refueling outage.

During normal plant operations the RCDT is maintained with a hydrogen gas blanket; however, during refueling outages (just prior to the Reactor Coolant System (RCS) chemical cleanup/crudburst activities) all hydrogen blankets must be removed before peroxide is added to the RCS. Each tank that must be purged of hydrogen is currently on the outage critical path because the purging must be performed one tank at-a-time due to lineup limitations. This change is beneficial and desired because the RCDT can then be purged off of the outage critical path resulting in a time savings of approximately 6-8 hours.

Summary of Evaluation:

The requirement to keep a hydrogen blanket on the RCDT is of benefit when the reactor is critical for long extended periods of time. The basis for the requirement of a hydrogen blanket on the RCDT is: to prevent the introduction of air or nitrogen and thus prevent the generation of ammonia that could affect the RCS chemistry; to prevent excessive buildup of nitrogen contaminated with radioactive gases that would have to be stored in the gas decay tanks; and to prevent generation of a volatile mixture of hydrogen and air. Adding nitrogen to the RCDT could potentially create these conditions; however, approximately 24 hours prior to a unit shutdown for refueling, the addition of nitrogen would not generate a significant amount of ammonia in the RCS or result in producing excessive amounts of nitrogen contaminated with radioactive gases. The introduction of air continues to be controlled. The RCS chemistry within 24 hours of a unit shutdown for refueling is characterized by rapidly changing conditions, thus effects from purging the hydrogen blanket with nitrogen would be negligible. The requirements for dissolved hydrogen are reduced 24 hours before shutdown; therefore, the intent of the discussion in the FSAR is unchanged. A review of the EPRI "Primary Water Guidelines" (Rev 4, Volume 1 page 3-16 & Volume 2 page 3-24), shows that the required amount of hydrogen in the RCS should be diminished in preparation for a unit shutdown and can be done so without negative effects on the system. Therefore, the original intent of the hydrogen blanket on the RCDT is not affected.

There is no increased risk to mechanical equipment because one type of gas is simply being replaced with another while the system is in operation. The risk of introducing more dissolved oxygen to the system is also not increased because the only difference in the old method vs the new method is that the RCDT pumps are running in the new method. RCS water chemistry requirements are such that a minimum amount of hydrogen must be present while in operation. EPRI PWR Water Chemistry Guidelines allow the RCS minimum hydrogen to be further reduced approximately 24 hours prior to

shutdown. The amount of RCS hydrogen reduction introduced by this activity is minimal and EPRI guideline information indicated that there is no reason for concern. Introduction of nitrogen to the RCDT while operating will result in some ammonia production as well as the generation of nitrogen contaminated with radioactive gases; however, the amounts will be negligible given the short amount of time operation will be allowed with a nitrogen blanket.

Evaluation Number SE-99-041 Unit: 1NN
Revision 0

Activity Title: Unit 1, Cycle 8 CORE CONFIGURATION

Description of Change(s):

During the next refueling outage for Unit 1 (1RF07), prior to operation of Cycle 8, 60 fresh Region 10A and 24 fresh Region 10B fuel assemblies, in addition to 4 Region 7 partially burned assemblies, all manufactured by Siemens Power Corporation (SPC), along with 1 Unit 2 Region 2 (called Region 2-2) partially burned Westinghouse optimized fuel assembly (OFA), will replace 72 Region 8, 16 Region 7 SPC assemblies and 1 Region 3 Westinghouse Standard fuel assembly. The partially burned assemblies are from the spent fuel pool and were discharged from Unit 1 and the end of cycle 6, except for Region 2-2 which was discharged from Unit 2 at the end of its Cycle 1. For the Unit 1 Cycle 8 core configuration, 84 fresh SPC fuel assemblies will be co-resident with 108 partially burned SPC fuel assemblies and 1 partially burned OFA manufactured by Westinghouse. In addition, provisions for accommodating up to 10% steam generator tube plugging have been made in the analyses.

Summary of Evaluation:

The CPSES U1C8 mixed core configuration has been evaluated for mechanical and thermal-hydraulic compatibility between the different SPC and W fuel assemblies. All applicable design criteria were determined to be satisfied at the current power levels. The neutronic characteristics of the Cycle 8 core configuration have been evaluated for their effect on the accident analyses. In all cases, it was determined that the applicable event acceptance criteria are satisfied. Because all mechanical design criteria continue to be satisfied, there is no reduction in any failure point introduced by the Cycle 8 core configuration. All acceptance criteria of the accident analyses continue to be satisfied; therefore, there is no increase in the consequences of any accident previously analyzed. Based on the foregoing, it is concluded that the Unit 1 Cycle 8 core configuration does not reduce any margin of safety as defined by the plant Technical Specifications.

Evaluation Number SE-99-042 Unit: NN2
Revision 0

Activity Title: INSTALL TEMPORARY HYDROGEN SUPPLY FOR THE UNIT 2 VOLUME CONTROL AND POSITIVE DISPLACEMENT PUMP SUCTION STABILIZER DURING 1RF07

Description of Change(s):

TM to provide hydrogen to the Unit 2 Volume Control Tank (VCT) and Positive Displacement Pump (PDP) suction stabilizer while the hydrogen supply main header is isolated for Unit 1 work. The TM 2-99-000006-00 will safely supply an alternate source of hydrogen to Unit 2 while work is performed on the supply header during the Unit 1 outage 1RF07. The system was isolated at both the supply bottle and at the temporary injection point when injection was not occurring. When hydrogen is being added to the system an operator would be present at all times and would isolate the hydrogen supply bottle if there are any malfunctions, failures or leakages are detected. A fan supplying air would be in operation at all times when the temporary hydrogen supply bottle is open. The hydrogen supply bottle will have 2 discharge regulators, in series, to protect the hydrogen supply header from over pressurization. The tubing was installed to a minimum of ASME B31.3 and no more than 400 cubic feet of hydrogen was in room 2-090, at elevation 832, in the Safeguards building at any time. No more than one hydrogen bottle was connected to the hydrogen header at anytime. Safety services will check the hydrogen content in the room periodically and as needed.

Summary of Evaluation:

There are no conditions that exist where any single or multiple credible failures would cause the Safety Related System to fail or operate in a degraded state. No additional safety systems would have to function or be actuated if the hydrogen system malfunctioned. The TM purposely limits the amount to hydrogen that would be available to inject in the VCT to eliminate the dangers associated with the hydrogen. Double pressure regulators would be installed to make the system single failure proof from possible over pressurization of the supply header. A fan was added to the room to supply sufficient air to maintain hydrogen concentrations low, in case there are leaks that occur after the system is installed. An operator is required to mitigate the consequences of a catastrophic failure of the fitting between the hydrogen bottle and the first regulator.

Safety related failures have already been analyzed and sufficient safety precautions are being taken to eliminate significant fire or detonation hazards. Therefore, the final result is that there is insufficient hydrogen connected to the system to cause significant failures in the safety related equipment.

A review was performed that verified the components, identified in the TM, meet all requirements. The piping between the hydrogen bottle and valve 2HG-0013 are supported in accordance with CPES-I-1018 and tested to 1.5 times the operating pressure down stream of the second regulator, which is in accordance with ASME B31.3.

Evaluation Number SE-99-043 Unit: 1N2
Revision 0

Activity Title: Revision of operability test and inspection and preventative maintenance of circuit breakers, and update for Containment Penetration Conductor overcurrent protection.

Description of Change(s):

The surveillance frequencies for operability tests of the containment penetration conductor protective circuit breakers are changed from 10% of each type of breaker every 18 months to 72 months on a staggered test basis.

Molded Case Circuit Breakers (MCCBs) testing requirements are changed from requirements provided in NEMA AB 2 to NEMA AB 4.

The breaker surveillance for inspection and preventive maintenance of containment penetration conductor protective circuit breakers is split into three surveillances for medium voltage 6.9 kV switchgear circuit breakers, low voltage 480 V switchgear circuit breakers, and 480 V and lower voltages molded case circuit breakers to align with vendor recommendations and industry experience. The preventative maintenance and inspection period for molded case circuit breakers is changed to 72 months, consistent with EPRI recommendations.

The Bases for the Technical Requirements Manual is revised to address the changes to surveillance requirements of containment penetration conductor overcurrent protection circuit breakers.

Summary of Evaluation:

The surveillance frequencies, for operability tests for containment penetration conductor protective circuit breakers are changed to more conservative periods of 72 months with staggered testing, which is same as present testing or adequate for the breaker group.

EPRI document NP-7410-V3, Revision 1, "Circuit Breaker Maintenance, Volume 3: Molded Case Circuit Breaker Application and Maintenance Guide, Revision 1," recommends a period of 4 to 6 years, for all breakers to have maintenance within four refueling outages, where the Class 1E breakers are located in abnormal environment. At CPSES the MCCBs utilized for penetration conductor protection are located in mild environment. Therefore, a time period of 72 months for the maintenance of these breakers will assure the adequacy of the breakers to perform their safety function when required.

The 72 month staggered testing of breaker groups will not affect the assessment for determination of breaker performance, and the breaker tested under this program will adequately perform its intended function when required. NEMA AB 2 standard redirects testing requirements back to the NEMA AB 1 standard required for factory tolerances which is intended for laboratory conditions and not field conditions. NEMA AB 2 guidance has been rescinded and replaced with NEMA AB 4 guidance, which is intended for field conditions. NEMA AB 4 testing will assure the operability of the molded case circuit breakers. However, although manufacturers documents provide guidance for periodic testing and maintenance of molded case circuit breakers, they do not specify periodicity for

preventative maintenance and inspection. The EPRI NP-7410 recommendation of 72 months has been adopted. Therefore, all breakers (including 6.9Kv and 480V switchgear and molded case circuit breakers) will be tested in a 72 month period in lieu of the previous 180 month period. This activity does not affect any accident or equipment important to safety and therefore, will not affect the probability of failure of SSCs to perform their functions. Furthermore, there are no new potential failure modes created and consequently, no possibility for an accident of a different type than previously evaluated. Because the accidents are unaffected by the activity, the bases for all Technical Specifications remain valid and furthermore, there is no change to the margin of safety.

Evaluation Number SE-99-045 Unit: 1N2
Revision 0

Activity Title: REVISE CALIBRATION FREQUENCY OF WIND SPEED AND WIND DIRECTION SENSORS FROM 6 TO 12 MONTHS

Description of Change(s):

ANSI/ANS-2.5 (endorsed by RG 1.23) recommends that the calibration frequency of the meteorological (Met) instrument channels be performed at least semiannually. The purpose is to assure instrument data quality and accuracy; therefore, a comprehensive calibration of the CPSES meteorological system components is performed at a 6 month interval.

However, the manufacturer of the CPSES meteorological wind speed and wind direction sensors has recently certified that their factory calibration of these sensors is good for one year after a sensor is placed in service. Re-calibration per the manufacturer is due 12 months after a sensor is placed in service or three years after the date of the manufacturer's calibration. The manufacturer's calibration remains valid provided a sensor is properly stored before use; for sensors in storage awaiting installation, the time period between a sensor's factory calibration date and installation at CPSES (including storage time in the CPSES warehouse) should not exceed two years in order to support the 12 month installed service interval. Based on the manufacturer's certification, TXU Electric has elected to extend the replacement frequency of CPSES in-service wind speed and wind direction sensors to 12 months.

This change in frequency of replacement of the CPSES wind speed and wind direction sensors requires an exception to Regulatory Guide 1.23, Second Proposed Revision 1 (April, 1986), endorsing ANSI/ANS 2.5 regarding the required calibration frequency for these particular measured parameters. An exception to the ANSI/ANS 2.5 standard guidance calibration frequency of 6 months has been noted in FSAR Section 1A(B) and ODCM Part I, Meteorological Monitoring Instrumentation Controls & Bases.

Summary of Evaluation:

The practice of replacing wind speed and wind direction sensors every 6 months, with "just-in-time" calibrated sensors procured from the manufacturer seemed very conservative for these non-safety-related equipment items. Based on the manufacturer's design, calibration certification, and TXU Electric's review of these sensor's past reliability and performance history at CPSES, it was reasonably concluded that the meteorological system wind speed and wind direction sensors may be assigned a 12 month installed service interval. This assignment is based on the manufacturer's certification that the sensors retain their calibration for at least one year after installation. Administrative controls on calibrated sensors stored in the CPSES warehouse are in place to ensure that the time period from the last calibration to installation does not exceed two years.

This activity maintains the meteorological system data quality standards intended by Regulatory Guide 1.23 and ANSI/ANS 2.5 guidance.

Wind speed and wind direction sensors are replaced after 12 months of service in accordance with the manufacturer's calibration certification. In addition, the sensor that is being replaced is returned to the manufacturer for "as found" data and re-calibration. Changing the replacement frequency of the wind speed and wind direction sensors to every 12 months without requiring "just-in-time" procurement provides a maintenance cost saving and a more flexible meteorological monitoring program.

Evaluation Number SE-99-046 Unit: 1N2
Revision 0

Activity Title: ALLOW BYPASSING OF A HIGH RADIATION SIGNAL THAT AUTOMATICALLY ISOLATES STEAM GENERATOR BLOWDOWN

Description of Change(s):

NOTE: This activity was implemented via a temporary procedure change as part of the Y2K contingency plan.

A high radiation alarm at the Steam Generator Blowdown Radiation monitor (1, 2-RE-5179) or the Steam Generator Bowdown Sample Radiation Monitor (1, 2-RE-4200) automatically isolates the SGBD, the SGBD sampling and the SGBD effluent. The proposed activity would allow the operator to defeat the automatic isolation of the SGBD system and associated sample lines on a high radiation signal. Manual isolation of these systems upon receipt of a high radiation alarm will be achieved in accordance with administrative controls. Defeating the automatic isolation of these system on high radiation indications from the above radiation monitors minimizes the potential for inadvertent isolation of the SGBD & SGBD sample systems which, in turn, minimizes the potential for adverse impacts on the Steam Generator secondary side water chemistry.

Summary of Evaluation:

The proposed activity does not affect the operability or functionality of the radiation monitors or the SGBD System. The proposed activity does not impair the ability for manual isolation of the SGBD nor does it modify any of the other signals that initiate automatic SGBD isolation (e.g, Phase A containment isolation). Existing CPSES procedures proactively respond to the changes in radiation levels corresponding to probable or impending Steam Generator tube leakage. Actions based on RCS-to-SG leak rates are taken before such leakage would be expected to trigger the high radiation monitor alarms. Bypass of the SGBD System isolation on a high radiation monitor signal does not affect any accident sequence and thus, does not result in an increase in any radiological consequence.

Evaluation Number SE-99-047 Unit: NN2
Revision 1

Activity Title: DM 98-061, Revision 0, Convert TM 2-96-008 TO A PERMANENT INSTALLATION

Description of Change(s):

Design Modification, DM 98-061 Revision 0, converts TM 2-96-008 to a permanent mod. The temporary modification installed an alternate relief path for relieving SI header pressure due to back leakage through RCS pressure isolation check valves. The alternate relief path is required to meet RG 1.29, ANSI N18.2 and 18.2a. This modification documents that the process piping and tubing has been installed to meet the above requirements. As part of the modification, the NNS piping and tubing installed has been categorized as a Special Case Seismic Category II. FSAR Sections 6.3, and 17A are being revised to reflect this requirement. RCS Pressure Isolation Valve (PIV) leakage is governed by TS 3.4.14. Leakage through PIVs is also covered by TS 3.4.13. The leakage rate through the flow restricting tubing can be monitored to ensure it is accounted for as part of the identified RCS leakage. FSAR Section 5.2.5 is being revised to reflect this function. The post-accident SI leakage rate could be increased by up to 0.2 gpm. When the alternate relief path is in use and unattended, a 0.2 gpm penalty will be taken against the 1 gpm RSLI limit. This will ensure the radiological consequences of a LOCA as described in the FSAR remain bounding.

Summary of Evaluation:

The reduction of SI flow and RWST inventory are below the significant figures in ECCS design and analysis and, therefore, insignificant with respect to ECCS performance. Although the quantity of leakage of primary coolant outside containment post-LOCA is increased, the design limits the leakage to a fraction of that previously approved in the FSAR and, as discussed above, this leakage will not increase the radiological consequences as calculated and documented in the FSAR.

Evaluation Number SE-00-01 Unit: 1X2
Revision 0

Activity Title: MODIFY THE HYDROGEN PURGE EXHAUST SYSTEM TO ENSURE 700 CFM FLOWRATE THROUGH THE FILTERS

Description of Change(s):

Install a flow control valve in the Hydrogen Purge System exhaust to modulate flow through the filtration units and return the system to "as designed" conditions.

Summary of Evaluation:

This activity is to bring the system into conformance with the design and licensing basis. This change involves the modification of the HPS to meet its design functional requirements and assures the capability to provide a backup alternative to the Containment Electric Hydrogen Recombiners in the event of their failure post-LOCA. The HPS also provides the hydrogen control function if both Containment Electric Hydrogen Recombiners are inoperable as allowed by Technical Specification 3.6.8. The HPS also provides the capability for a controlled purge of the containment atmosphere to aid in cleanup of an accident as described in the FSAR Chapter 15. The HPS's only nuclear Safety-Related function is Containment Isolation which is not adversely affected by this change. The change involves opening the containment isolation valves at 5.8 psig in lieu of 5.0 psig; however, the capability of the Containment Isolation Valves to perform their safety function was evaluated and found acceptable. No new failure modes are created by the addition of the flow control valve. This modification brings the HPS into conformance with the FSAR commitment to the radiological control parameter in Reg. Guide 1.143. Therefore there is no increase in the radiological consequences as described in the FSAR for the HPS.

Evaluation Number SE-00-002 Unit: NN2
Revision 0

Activity Title: INSTALL A GAG ON VALVE 2BS-0025 AND REMOVE LIMIT SWITCHES 2-LSI & LS2 (2-ZS-AL036 & AL037)

Description of Change(s):

This temporary modification (TM) removes the limit switch housing and places a gag on valve 2BS-0025. This valve gag uses the mounting for the limit switch to hold the valve in place.

Summary of Evaluation:

The TM will gag valve 2BS-0025 and prevent movement of the valve. The removal of the limit switches from the valve although not able to provide indication of the actual valve position will only invoke interlocks preventing the operation of the interior door if they are mis-positioned. (Conservative condition). The limit switch that is removed will be restrained seismically to prevent damage to Safety Related Components.

Gagging the equalization valve will require that the exterior door only be operated manually if it is required to be used. The interior door can be used either manually or electrically.

Power to the electrically driven hydraulic pump is typically isolated except when entries are made. There is a TM limitation that the power can not be restored unless it is assured that limit switch 2-LSI (2-ZS-AL036) is positioned correctly.

Evaluation Number SE-00-005 Unit: NXN
Revision 0

Activity Title: FUEL BUILDING BRIDGE CRANE ANTICRABBING MODIFICATION

Description of Change(s):

DMA/FDA-1999-002177-01 is a design modification installing an anti-skew control system to the Fuel Building Handling Bridge Crane. This control system prevents the skewing (crabbing) of the crane which has resulted in breaking of drive wheel bearings. Elimination of this problem will prevent the possibility of materials from broken wheel bearings dropping into the spent fuel pool. The Design Mod makes changes to FSAR Section 9.5 describing effects of PVC cable additions resulting from the hardware changes.

Summary of Evaluation:

Installation of this mod. will not result in any changes to current crane control philosophy and failure modes remain unchanged. Weight added to the Fuel Building Bridge Crane assembly has been considered and the stresses remain within the allowable limits. The existing circuit breaker, cable size, and distribution panel remain adequate for this application. The FSAR is being revised to document the acceptability of the small amount of PVC cable being added.

Evaluation Number SE-00-007 Unit: NN2
Revision 0

Activity Title: RETURN BATTERY BT2ED1 TO A 60 CELL BATTERY AND INCREASE DESIGN MARGIN TO 15%

Description of Change(s):

This change per FDA-1999-000863-01-00 removes a jumper around cell 19 of battery BT2ED1. The jumper was installed by DCN-12931 after cell 19 failed. The failed cell has been replaced and is now ready to be reconnected. This change will return BT2ED1 to a 60 cell battery and remove a unit difference. Also, the FSAR description of BT2ED1 provided by Table 8.3-4 and Figure E2-0020 (8.3-14) will be revised to reflect the change from 59 cells to 60 cells and the increase in design margin to 15%.

Summary of Evaluation:

The implementation of this activity does not introduce any new failure modes or Licensing Basis Accidents. The probability of a malfunction of equipment important to safety and the probability of a Licensing Basis Accident will be unchanged by implementation of this activity. There are no radiological consequences associated with this activity. The activity returns the battery to its original design configuration. No unreviewed safety questions are created by implementation of this activity.

Evaluation Number SE-00-008 Unit: NN2
Revision 0

Activity Title: INSTALLATION OF THROTTLING VALVES IN EACH OF THE RCP SEAL #3 STANDPIPE FILL LINES & ALSO IN THE RMUW SUPPLY HEADER

Description of Change(s):

DM-1999-003418-01-00 will install two 3/4" socket welded throttling valves in the Reactor Coolant System (2RC-8155 and 2RC-8156). One valve will be installed in each of the Reactor Coolant Pump Seal #3 standpipe fill lines (3/4-RC-2-125-151R-5 and 3/4-RC-2-126-151R-5 respectively). During normal operation, level control valves 2-LCV-178, 2-LCV-179, 2-LCV-180 and 2-LCV-181 cycle to allow the Reactor Coolant Pump seal #3 standpipes to fill. Extensive testing has determined that the cycling of these LCVs causes a pressure transient to occur which lifts relief valves 2RC-0036 and 2DD-0600. Installing the new valves in the LCV supply headers and throttling the valves to the flow required to maintain the RCP #3 seal should minimize any pressure transients, helping to eliminate the spurious lifting problems currently experienced by the relief valves. Also, this design modification will install a 3" throttling valve in the 3" reactor makeup water header that supplies the 3/4" standpipe fill lines. The installation of the 3" throttling valve is required to throttle the flow in the 3" supply header in order to reduce the pressure in the line, thereby helping to prevent the inadvertent lifting of relief valve 2DD-0600 during standpipe fill activities.

This modification has already been performed on Unit 1 during 1RF07 (with the exception of the 3" isolation valve). The associated Unit 1 Safety Evaluation which evaluated the Unit 1 activity is SE-99-030.

Summary of Evaluation:

Installation of the throttling valves in the 3" supply header and the 3/4" fill lines to the RCP Seal #3 standpipes will not impact the system's ability to make-up sufficient seal water inventory to the seal standpipes in a timely manner, nor will it impact the ability to makeup sufficient flow (150 gpm @ 65 psig) to the PRT spray line. Evaluation indicates that installation of the throttling valves will not adversely affect the operation of the safety related Reactor Coolant System (nor the chemical and volume control or demineralized and reactor makeup water systems). As a result of this activity, there is no increase in the probability of a Licensing Basis Accident or safety related equipment malfunction, nor is there a potential for creating a previously unanalyzed event. Also, there is no increase in the consequences of a previously analyzed event. As a result of this activity, there is no change to the Technical Specifications.

Evaluation Number SE-00-009 Unit: 1NN
Revision 0

Activity Title: ADDITION OF STEAM GENERATOR LEAK RATE MONITORS (UNIT 1)

Description of Change(s):

A plant design modification installed new Steam Generator Leak Rate Monitors (SGLRMs) on each Unit 1 Main Steam Line (MSL). These SGLRMs were installed to aid the Operator's in early detection of a slowly-propagating steam generator tube leak. The SGLRMs are strap-on Adjacent-To-Line Nitrogen (N-16) gamma scintillation detectors that are mounted on each MSL just upstream of the Main Steam Isolation Valves (MSIVs). This system of radiation monitors is more sensitive than currently available secondary-side radiation monitors with respect to detecting a slowly-propagating steam generator tube leak. TXU Electric committed to installing these radiation monitors via docked correspondence [TXX-99139] in conjunction with seeking a License Amendment and implementing voltage-based alternate repair criteria [Generic Letter GL-95-05, Attachment 1] for Unit 1 steam generator tubes affected by outer diameter stress corrosion cracking.

The FSAR update addresses the plant changes (additions) made by this activity.

Summary of Evaluation:

This activity installed monitoring instrumentation which is used for the purpose of leak detection only. There is no safety or control functions involved with this instrumentation. No new failure modes are introduced by the activity. Installation was made using applicable criteria for non-safety related equipment such as RG 1.75 electrical separation and RG 1.29 seismic criteria.

The new SGLRM and existing MSL monitor assemblies are nonsafety-related and powered from nonsafety-related power sources. Changes to loading to the plant electrical system, maintenance of seismic category II qualification, and impact on heat load in the rooms in which installation occurred are justified by calculation updates. The equipment was manufactured and installed in accordance with an approved vendor quality assurance program.

Computer equipment changes made to support this activity utilized existing types of software and firmware currently used for these systems, so no changes were made to equipment originally analyzed and tested. Software and firmware changes and associated test procedures were provided by the manufacturer. Vendor testing was performed to ensure that software and firmware changes perform in accordance with design. An evaluation determined that components installed under this activity do not impact other plant systems with EMI/RFI, and are not themselves impacted by EMI/RFI from other plant systems.

Evaluation Number SE-00-010 Unit: 1N2
Revision 0

Activity Title: Revise Technical Requirements Manual to define DRPI OPERABILITY surveillance requirements independent of control rod drop timing.

Description of Change(s):

This change to the Technical Requirements Manual incorporates an enhancement identified during implementation of the Improved Technical Specifications (ITS), Amendment 64, per EVAL-1999-001661-18. The changes are applicable to TR 13.1.39 and TRB 13.1.39. The main objective is to define DRPI OPERABILITY requirements independent of control rod crop timing. The current TRM requires DRPI OPERABILITY to be determined during the performance of individual shutdown and control rod drop time measurements. Another objective is to allow Control Rod Drive Mechanism (CRDM) step traces to be performed independent of control rod drop timing or DRPI testing. The emphasis of the proposed new TR LCO 13.1.39 wording is to allow for control rod movement in MODEs 3, 4, or 5 for any reason as long as Keff <= 0.95 and no more than one shutdown or control bank is withdrawn from the fully inserted position. The new wording allows DRPI Operability testing and CRDM step traces to be performed in MODE 5 without concurrent cold rod drop time measurements. This activity changes the procedure as described in the current TR 13.1.39.

Summary of Evaluation:

This change to the Technical Requirements Manual proposes changes to TR 13.1.39 and TRB 13.1.39. The main intent of the changes is to specify rod drop timing and DRPI OPERABILITY testing as independent, but related activities. Allowance is also made for CRDM step traces, although step traces are not required by Technical Specifications. The modified TR requirements allow for early identification of Rod Control and DRPI problems during post-outage recovery. The testing methods that are retained do not deviate from the requirements described in the Licensing Basis documents. Rod drop timing in MODE 3 has traditionally been the first time to pull the rods after an outage, and there a convenient time to checkout DRPI. The intended plant sequencing schedule to allow DRPI and control rod operability testing to be performed in MODE 5 without concurrent rod drop time measurements does involve a change in the procedure currently described in TR 13.1.39. Changing this sequencing, with proper reactivity controls, allows for optimum scheduling of control rod and DRPI testing while maintaining adequate subcritical margin.

Evaluation Number SE-00-011 Unit: 1N2
Revision 0

Activity Title: Add a two second time delay to the "low oil pressure" turbine trip circuitry

Description of Change(s):

Evaluate FDA 1999-001535-01-00 (for Unit 2) & FDA 1999-001535-02-00 (for Unit 1), to add a two second time delay to the main turbine low lube oil pressure trip circuitry. Evaluate adding the time delay and revise the FSAR by removing turbine trip logic FSAR Figure 10.2-1, and replacing it with logic drawings E1 & E2-0022, 22A, 22B & 22C.

Summary of Evaluation:

The main turbine lube oil systems function is to lubricate the turbine bearings, lift the shaft and rotate the shaft on turning gear. This is a non safety function and the loss of this function is not credited in an accident analysis. At above 50% power, with a low lube oil pressure trip will come an "anticipatory" reactor trip, but adding a two second delay is within the current design basis and licensing basis. This lubricant low pressure delay trip does not increase the chances of a missile or affect the primary system. The failure of the time delay relay would not affect the response time of the reactor protection functions or disable a reactor trip.

The differences between FSAR figure 10.2-1 (created from E1-0022, 22A, B & C) and E1/E2-0022 drawings are editorial, additional detail, clarification, or minor technical changes.

Evaluation Number SE-00-012 Unit: 1N2
Revision 0

Activity Title: Procedure change to cross connect Unit 1 and Unit 2 Train A Safety Chilled Water - Unit 1 Supplying

Description of Change(s):

Crossties between Unit 1 and 2 Safety Chilled Water at the Spent Fuel Pool Cooling pump room emergency fan coils units (EFCUs) are considered to be GDC-5 isolation boundaries required to ensure that in the event of an accident in one unit, the other can be safely shut down. The procedure change will only be used in the event of a loss of Train A decay heat removal on Unit 2 and there is a need for long term cooling of Unit 2 Train B. Opening the crosstie from Unit 1 Train B results in that Train being inoperable for the mitigation of DBAs and the procedure change requires the Unit 1 LCO to be entered. The procedure change ensures that Unit 1 Train A supported equipment is OPERABLE before making Train B inoperable.

Calculation shows that Unit 1 Train B CHS can maintain Unit 2 RHR and CCW pump rooms below normal maximum temperatures via the crosstie if the flow to certain non-required EFCUs are isolated. The procedure change requires the above isolation and directs additional isolation if required to maintain Unit 2 Train B pump room temperatures. The implementation of the cross-connect via this procedure change will ensure the long term operability of decay heat removal via Unit 2 Train B.

Summary of Evaluation:

There is no effect on the probability of malfunctions, events, or accidents as described in the current licensing basis (CLB). The consequences of accidents as described in the CLB are not affected by this change because the Unit 1 LCO would be entered as is allowed.

Evaluation Number SE-00-013 Unit: NN2
Revision 0

Activity Title: REVISE UNIT 2 TURBINE DISC INSPECTION BASIS

Description of Change(s):

LP Turbine disk inspection interval of 50,000 hours is based upon vendor evaluations referenced in the FSAR. Vendor reports for the licensing basis are for Unit 1 LP turbine rotor disk inspection intervals and are not directly applicable to the Unit 2 rotors as implied by the FSAR. These reports were developed from specific metallurgical samples from the Unit 1 turbines. The Unit 2 LP turbine disk and rotor samples exhibit different attributes. Inspection data from Unit 2 outage reports could be less conservative than that data previously approved for Unit 1. Ref. FSAR 10.2.3.6 for Inservice Inspection, ER-8402. This activity provides for the revision of the FSAR to document the Unit 2 disk inspection basis. It will differentiate between the Unit 1 and Unit 2 disk inspection basis.

Summary of Evaluation:

The LP turbine disk inspection is discussed in the FSAR sections 3.5.1 and 10.2.3.6 for turbine missile generation and inservice inspections of the turbine generator. The LP turbine disk inspection interval is derived from the vendors analysis of the turbine properties and estimated probability of disk burst failure. The vendors estimated probability of failure is an input to the CPSES analysis. The missile analysis was performed to ensure the safety and integrity of systems, structures, and components. The original CPSES turbine missile analysis used the vendor engineering reports to determine the probability of missile generation from LP turbine disk burst. The vendor has submitted a new engineering report which uses improved calculation techniques and actual inspection data from previous CPSES turbine inspections.

This evaluation reviews the data available to determine that the Unit 2 LP turbine basis is unique and that the inspection intervals are still valid. The normal inspection intervals of 50,000 hours between inspections are not affected by this evaluation.

LP turbine 2-02 was inspected in 2RF03 and LP turbine 2-01 was inspected in 2RF04. Inspection results indicate that the Unit 2 LP turbines can continue to be operated until the next scheduled disk inspection.

Evaluation Number SE-00-014 Unit: 1N2
Revision 0

Activity Title: REVISE SR 3.1.3.2 BASES TO INCLUDE ADDITIONAL INFORMATION WHEN THE SR SHOULD BE PERFORMED AND DISCUSS APPROPRIATE ADJUSTMENTS

Description of Change(s):

The main objective of change to the Technical Specification Bases is to clarify in the BASES when SR 3.1.3.2 should be performed with respect to 300 ppm RCS boron concentration. Clarification is also needed to describe the adjustments that are made to the measured RCS boron concentration for comparison to the nominal conditions at which the predicted boron concentration was determined. The adjusted boron concentration is used to determine when 300 ppm has been obtained for purposes of scheduling and performing the surveillance activity. This activity changes the procedure as described in the BASES portion of Technical Specifications for SR 3.1.3.2.

Summary of Evaluation:

This evaluation proposes changes to the BASES portion of Technical Specifications for SR 3.1.3.2. Measuring the end-of-life moderator temperature coefficient (EOL MTC) is required by Technical Specifications SR 3.1.3.2 to be performed at 300 ppm RCS boron concentration. SR 3.1.3.2 contains three Notes, the first of which states, "The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARFO) boron concentration of 300 ppm". The SR does not indicate that the surveillance cannot be done earlier in core life than 300 ppm, but it definitely indicates it must be performed prior to exceeding 7 EFPDs after reaching 300 ppm, without any extension granted per SR 3.0.2. This LDCR proposal makes it clear the SR should not be performed prior to the adjusted concentration of 300 ppm. This LDCR also provides for the use of a Boron-10 (B-10) isotopic depletion model to adjust the measured boron concentration to be more consistent with the calculational basis of the SR values.

Evaluation Number SE-00-015 Unit: 1N2
Revision 0

Activity Title: SWIS CHLORINATION BUILDING REFURBISHING - DMA 1999-1929

Description of Change(s):

In the SWIS Chlorination building, there are two tanks (one 500 gallon tank filled with sodium bromide and one 1500 gallon tank filled with sodium hypochlorite) solutions. These tanks are sized such that there is a new truck load of chemicals required to fill both tanks once a week or more often in the summer time when more chlorination is required.

This activity proposes that the existing tanks and pump skid, be replaced with 6 larger (1900 gallons each) tanks with a much larger volume capacity, and a new pump skid with 4 new pumps. This activity will be worked in conjunction with DMA 1998-2070 which improves the distribution system for the chemicals into the SWIS intake bay.

The new tanks will reduce the number of fill times required annually, thereby reducing the potential for an inadvertent chemical spill with respective accident ramifications.

Summary of Evaluation:

The results of the SE indicate that the proposed activities will not impact normal plant operations and will not affect any licensing basis accidents. This activity will not create the possibility of a malfunction of equipment important to safety different from any previously evaluated in the Licensing Basis Documents.

Evaluation Number SE-00-020 Unit: NN2
Revision 0

Activity Title: UNIT 2, CYCLE 6 CORE CONFIGURATION

Description of Change(s):

During the next refueling outage for Unit 2 (2RF05), prior to operation of Cycle 6, 88 fresh Region 8 fuel assemblies, 4 twice-burned Region 5 fuel assemblies, and one once-burned Region 2 fuel assembly will replace 68 Region 6 assemblies, 24 Region 5 assemblies, and one Region 2 assembly. For the Unit 2 Cycle 6 core configuration, 88 fresh fuel assemblies manufactured by Siemens Power Corporation (SPC) will be co-resident with 103 partially burned fuel assemblies manufactured by SPC and 2 partially burned optimized fuel assemblies (OFA) manufactured by Westinghouse. In addition, provisions for increasing the power level from 3445 MW to 3458 MW with regard to FSAR Chapters 6 and 15 safety analyses have been made.

Summary of Evaluation:

The CPSES U2C6 mixed core configuration has been evaluated for mechanical and thermal-hydraulic compatibility between the different SPC and W fuel Assemblies. All applicable design criteria were determined to be satisfied at both the current and the increased power levels. The neutronic characteristics of the Cycle 6 core configuration have been evaluated for their effect on the accident analyses. In all cases, it was determined that the applicable event acceptance criteria are satisfied. Because all mechanical design criteria continue to be satisfied, there is no reduction in any failure point introduced by the Cycle 6 core configuration. All acceptance criteria of the accident analyses continue to be satisfied; therefore, there is no increase in the consequences of any accident previously analyzed. Based on the foregoing, it is concluded that the Unit 2 Cycle 6 core configuration does not reduce any margin of safety as defined by the plant Technical Specifications.

Evaluation Number SE-00-021 Unit: NN2
Revision 0

Activity Title: COVERAGE TEST OF WIRELESS LAN SYSTEM TO BE PERFORMED DURING THE PLANT OUTAGE 2RF05

Description of Change(s):

This activity is a radio coverage test of the proposed wireless LAN system. This test will address the activities to be performed to determine how many Access Points (radio transcievers/LAN interface devices) will be required to be installed in the Unit 2 Containment Building, Unit 2 Cable Spreading Room and other radio sensitive areas noted in Operations Standing Order, OSO-004 Rev. 1, dated 8/5/98 during a proposed subsequent plant modification. The test involves activation (operation) of low power output 2.4 GHZ radio transceivers. The test will be performed while the Unit 2 Reactor Vessel's core is off-loaded (no fuel). The test will encompass the Unit 2 Containment Building, Reactor Vessel's core is off-loaded (no fuel). The test will encompass the Unit 2 Containment Building, Unit 2 Cable Spread Room, and other radio sensitive areas of the facility. No other testing, in any other areas of the plant, is proposed for this test. A test plan will be developed for the test and will contain the appropriate precautions to limit the location of the Access Point radio transceivers as well as hand held transceivers to ensure the minimum exclusion distances are maintained.

Summary of Evaluation:

Based on industry guidance, testing performed at other facilities, industry expert experience, and calculations based on the manufacturer's data for the equipment used during the test, it has been determined that this activity (test) will not cause an adverse plant condition to exist during the performance of the test.

Evaluation Number SE-00-022 Unit: NN2
Revision 0

Activity Title: ARV SOUND LEVEL TEST (PROCEDURE ETP-TP-00B-1, REVISION 0)

Description of Change(s):

Procedure ETP-TP-00B-1, Revision 0, describes a proposed test to be performed to gather sound level and/or vibration data for the Unit 2 steam generator atmospheric relief valves (ARVs). The proposed test would be performed at power levels between 25% RTP and 95% RTP and would allow concurrent steam relief through the test ARV, the steam dump valves (in pressure control mode) and the main turbine.

Summary of Evaluation:

Procedure ETP-TP-00B-1, Revision 0, describes a proposed test to be performed to gather sound level and/or vibration data for the Unit 2 steam generator atmospheric relief valves (ARVs). The proposed test would be performed at power levels between 25% RTP and 95% RTP and would allow concurrent steam relief through the test ARV, the steam dump valves (in pressure control mode) and the main turbine. This alignment (concurrent flow through the ARV, Steam Dump System, and main turbine) at elevated power levels is an off-normal system configuration. However, throughout the proposed test, all systems and valves would remain capable of performing their design functions; i.e., the ARVs remain available for plant cooldown, the turbine capable of automatically tripping, and the Steam Dump System capable of normal post-trip dissipation of the RCS heat to the condenser, if available. The proposed activity will exercise the ARV, the Steam Dump System, and the main steam/feedwater controls in manners consistent with their designs. The proposed activity does not affect the capabilities of any SSCs required to mitigate a licensing basis accident. The accident analyses presented in FSAR Chapters 6 and 15 remain bounding and are unaffected by the proposed activity. Therefore, the radiological consequences of the accident analyses are unaffected. Because of all accident analyses remain valid and are unaffected by the proposed activity, the bases for all Technical Specifications remain valid. In addition, because the accident analyses are unaffected and because the proposed activity does not affect the failure point of any fission product barrier, the margin of safety is unaffected by the proposed activity.

Evaluation Number SE-00-024 Unit: 1N2
Revision 0

Activity Title: TEMPORARY CHANGE TO CCW THROTTLE VALVE SETPOINTS

Description of Change(s):

FDA-99-1397-06 will temporarily increase the flow set point for the S-signal and P-signal flow rates through the RHR/CT HX CCW Return Valves 1-HV-4572, 4573, 4574, 4575 and 2-HV-4572, 4573, 4574, 4575. This design change is only a temporary compensatory action for a degraded condition (valve cavitation at the current setpoint) to be implemented during the cooler months of 2000-2001 to take advantage of cooler SSI and CCW temperatures which in effect allow the higher flow rates through the RHR and CT heat exchangers. Increasing the flow setpoint will temporarily result in a fouling monitoring criteria change. A minor increase in CCW HX maintenance could result but is not expected. The temporary flow setpoints will reduce the pressure drop across the valves moving the operation to a less severe cavitation regime.

Summary of Evaluation:

Increasing the flow setpoint constitutes a compensatory action for a degraded condition which will reduce the likelihood of failure of CCW or any supported function. Throttling of the subject valves is the result of the Reactor Protection System response to an accident or to slave relay testing. The reduction in the severity of throttling during an accident will also reduce the severity during surveillance testing. Therefore, this change will also reduce the probability of a loss of CCW during normal plant conditions. There are no adverse impacts on any design basis accidents or events. Containment Analyses and Ultimate Heat Sink Analyses remain bounding. No new failure modes are introduced.

Commitment Management Change Evaluations

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CMCE: 97-05
Commitment Number: 23486
Change Type: Revision

Source Document:

TXX-89596
TXX-89744
SDAR-CP-89-019
SDAR-CP-89-015

Original Commitment Description:

Shift log information TU Electric will take actions to improve the documentation of equipment problems in the shift log: 1) problems causing initiation of a PIR will be referenced in the station log with its PIR number, 2) Technical Specification LCOs will be tracked in the Unit log and will be discussed during the shift turnover process.

Revised Commitment Description:

Shift log information TU Electric will take actions to improve the documentation of equipment problems in the shift log: 1) Technical Specification LCOs, will be tracked in the Unit log and will be discussed during the shift turnover process.

Justification for Change:

This change removed the actions associated with a Plant Incident Report (PIR). In most cases, the determination whether an event meets the PIR criteria is made in the corrective action documentation meetings (quorum) which may occur days after the initial event is recorded in the station log. Because of this, in most cases, the Control Room would have to be notified after PIR determination has been made. This was a cumbersome process with little value added.

As noted above, Operations is notified of PIR events initially when the determination is made, during the Plan-of-the-Day meetings and upon completion of the PIR report via distribution to the Operations Manager.

CMCE: 97-07
Commitment Number: 24792
Change Type: Revision

Source Document:

TXX-90031
GL-89-13

Original Commitment Description:

Perform the maintenance activity for the inspection of the Service Water Intake Structure using a SCUBA diver at least once per fuel cycle. Supporting documentation for this commitment shall be available in your file for NRC review. Any significant accumulation of silt and debris (such as clams) encountered will be removed.

Revised Commitment Description:

Perform the maintenance activity for the inspection of the Service Water Intake Structure using a diver at least once every five years. Supporting documentation for this commitment shall be available in your file for NRC review. Any significant accumulation of silt and debris (such as clams) encountered will be removed.

Justification for Change:

The identification of SCUBA equipment would (via literal compliance) preclude other types of diving equipment which may be more available or more applicable to the diving situation (such as hard hat and umbilical cord tether systems).

The change to the five year inspection is based upon several factors:

- Chemical treatment of the water has yielded good results in the control of algae and clam infestations,
- Historical diving inspections have noted little buildup of sediment on the floor of the Service Water Intake Structure (SWIS),
- Water intake capability of the SWIS has not degraded,
- Extending inspection interval will reduce personnel safety risks encountered by the diver and,
- Results in cost savings and burden reductions.

CMCE: 98-07
Commitment Number: 02877
Change Type: Revision

Source Document:

TXX-2833
INSRPT-445/7808

Original Commitment Description:

TUGCO indicated weekly calls to USGS office in Ft. Worth to report Squaw Creek gauge height readings and to verify conversion of the reading to cfs. This telephone verification assures that the most recent rating curve shifts are utilized and qualified USGS personnel interpret the rating curve.

Upon request, USGS personnel began monthly velocity profile checks of Squaw Creek at Hwy. 144 gaging station to verify flow conditions and to increase accuracy of shift determinations of the rating curve.

Revised Commitment Description:

Flow in Squaw Creek downstream of Squaw Creek Reservoir Dam will be monitored weekly by either taking local gauge height readings or by accessing USGS Web site for published data.

Justification for Change:

Based upon technological advances in stream gauge instrumentation, it is no longer necessary to obtain the information by telephone. USGS personnel have agreed to inform CPSES whenever shift changes are made which will also be reflected in the WEB site published data. This commitment does not pertain to safety related aspects of CPSES

CMCE: 98-12
Commitment Number: 02977
Change Type: Deletion

Source Document:

TXX-4055
INSRPT-445/9517, 9341, 9333

Original Commitment Description:

To reaffirm total commitment to an effective, independent QA/QC program, TUGCO will develop an audiovisual program to re-emphasize commitment to quality at CPSES, that QA/QC personnel are required to report non-conformance and that no interferences with QA/QC functions will be tolerated. All individuals involved with quality related work, current employees, and future employees will be required to view this program.

Revised Commitment Description:

N/A

Justification for Change:

This commitment was generated from Enforcement Action 83-64 from the Atchison QA/QC allegations. As part of the response to the enforcement conference, TU Electric (then Texas Utilities Generating Company (TUGCO)) identified eight different steps that would be taken to assure (a) unfettered access to individuals or organizations to identify potential safety problems (b) development of an audiovisual program to reemphasize the commitment to quality and CPSES, (C) development and implementation of programs and organizations to allow personnel to identify any concerns and (d) post documentation and notices to remind personnel of the need to perform quality work.

Of the eight points identified in the TXX-4055 answering the enforcement action, the current TXU Electric program(s) and processes address, control, track and implement methods for identifying potential safety concerns, investigating the concerns and resolving the issues.

All programs are identified during initial hiring practices through training (as exampled in PAT) and literature and through continuing processes during the individuals employment at CPSES. The video "The Peak of Quality" was a second-generation film which was formerly shown as "Quality, It's Your Job" in order to stress quality awareness and problem identification requirements and processes during construction and startup activities.

TXU Electric still implements the 'Safeteam' program, has a Corporate Security 'Hot Line', Quality Assurance 'Hot Line' and emphasizes the commitment to quality in PAT. Other processes such as the use of electronic bulletin boards and CPTV (closed circuit television specific to CPSES) keep personnel informed of not only current quality problems which may exist, but remind personnel of their part to assure a safe and quality work environment. All these activities and actions still implement the intent of the letter.

The removal of the film "The Peak of Quality" would not adversely impact the overall commitment for quality and safety that TUE has made to CPSES.

CMCE: 98-14
Commitment Number: 26419
Change Type: Deletion

Source Document:

TXX-93117
INSRPT-445/9262

Original Commitment Description:

Nuclear Overview will implement a routine review of Nuclear Overview findings open greater than 90 days and report the results to Senior Management.

Revised Commitment Description:

N/A

Justification for Change:

The task team that assessed the Plant Incident Report (PIR) concluded that Nuclear Overview findings [Quality Assurance Deficiencies - QADs] over 90 days old should be periodically reviewed in an effort to determine if resolution of the finding(s) is receiving appropriate priority/resource allocation. Discussions with Nuclear Overview management indicates that findings such as these are typically noted as Quality Assurance Deficiencies.

As stated in the original response, corrective action procedures contain the controls/guidance for processing QADs and ONE/SmartForms (corrective action documents). The section on QADs requires the establishment of disposition due dates and accomplishment dates. If extension of either dates is necessary, the Responsible Manager contacts Nuclear Overview and reaches concurrence.

Since an appropriate due date is established upon issuance of the QAD and internal tracking and reminding systems are in place via the software for the SmartForm, a '90 day clock' for revisiting the subject is not required. Further, other commitments are active which direct investigation for any adverse audit findings, scheduling and implementation of significant failure analysis as necessary. These processes support the justification of deleting this commitment.

CMCE: 99-01
Commitment Number: 08328
Change Type: Revision

Source Document:

SSER-12
TXX-4412
Original Commitment Description:

In a letter dated 2/7/85, the applicant stated that the staff's position has been incorporated as follows: Rev. 2 to maintenance procedure MSM-C0-7307, "Safety Injection Pump Inspection", provides for both visual and surface examinations of the diffuser shroud to be performed in conjunction with the 9th refueling outage inspection of pump intervals.

Revised Commitment Description:

Visual and surface examinations of the Unit 1 & 2 Safety Injection (SI) Pump diffuser shrouds will be performed during normal or emergency maintenance at approximately 10 year intervals.

Justification for Change:

TXU Electric's intent was to examine the diffuser shrouds in the Unit 1 and 2 SI pumps during the 9th refueling outage for each Unit. However, since 1 of the Unit 1 SI pumps was dissembled for maintenance during the 7th refueling outage, we indent to perform the visual and surface exams on the other pump shroud during 1RF08. Performing these inspections during normal or emergency maintenance at approximate 10 year intervals was originally recommended by Pacific Pumps and was accepted by the staff. Revising the commitment description as noted will make this commitment consistent with the CPSES Inservice Inspection (ISI) Plan.

Maintenance procedure MMI-307, "Safety Injection Pump Inspection: replaced the earlier procedure listed in the original commitment. Further, a section has been added to the ISI 10 year plan addressing the examinations described above for the SI Pumps.

CMCE:	99-05
Commitment Number:	02978
Change Type:	Deletion

Source Document:

TXX-4055

Original Commitment Description:

To reaffirm total commitment to an effective, independent QA/QC program, TUGCO will conduct meetings between management and selected personnel to emphasize the commitment to quality and an effective QA/QC program.

Revised Commitment Description:

N/A

Justification for Change:

This commitment was generated in response to an NRC Notice of Violation. The response was developed during a construction era that was less integrated than today's quality management system. Further, a new Operations Review Committee (ORC) subcommittee has assumed the role of independent review of the QA program. Thus, the commitment is not necessary today.

This commitment does not affected any codified documents and process and organizational evolution/development (such as the ORC) make this commitment out-dated.

CMCE: 99-09
Commitment Number: 22350
Change Type: Deletion

Source Document:

SSER-20

Original Commitment Description:

The [Commitment Tracking System] CTS is routinely audited by the NEO Quality Assurance Department.

Revised Commitment Description:

N/A

Justification for Change:

This commitment was an abbreviated part of section 4.2.3, SSER-20. NTOL and early startup/operations corrective actions have been completed in most cases and incorporated into station quality programs. The current Nuclear Overview evaluation program is technically oriented and performance based and routinely assess commitments and the CTS by multiple avenues and inspection activities.

The FSAR (17.2.18.1) specifies that Regulatory Affairs (owner of the CTS) is subject to evaluations by Nuclear Overview. These licensing basis requirements ensure that the CTS is evaluated. These actions and commitments, embedded within the Nuclear Overview process, obviate the need for a commitment to specifically evaluate (audit) the CTS. Specialized evaluations in given disciplines/functions, which include the review of commitments, ensure that the CTS is periodically evaluated.

CMCE: 99-12
Commitment Number: 23115
Change Type: Deletion

Source Document:

INSRPT-445/8924
INSRPT-445/8937
TXX-89430

Original Commitment Description:

In the future, should a One Form {Operations Notification & Evaluation (ONE) Form*} operator error or procedure violation occur, the Operations Shift Manager will assure that a critique with the responsible Operations department individual will be held and documented.

*Corrective action document.

Revised Commitment Description:

N/A

Justification for Change:

This commitment was issued as part of an action plan in 1989 to reduce the number of operator errors. Evolution of internal programs and vigilance in identifying and reducing error conducive conditions, generation of lessons-learned and management overview have given growth to a culture of error-free operation. Operator errors have been well under control for a number of years obviating the need for the continuance of the commitment.

Specified actions in this commitment have accomplished the intended objective to reduce the number of operator errors. Other commitments are currently in place which make these actions unnecessary. Further, internal personnel error programs are in place to assure evaluation of problems stemming from personnel error.

CMCE: 00-03
Commitment Number: 26576
Change Type: Deletion

Source Document:

LDCR-SA-95-0077
SOER-92-01
TXX-92364

Original Commitment Description:

Performance Enhancement Review Committee (PERC), which includes Plant Manager, Manager ISEG and Training Manager, as a minimum, will review all significant personnel errors (as categorized by the ONE Form Committee), within five (5) working days of the event. The committee will discuss the event and make recommendations to enhance the human performance.

Revised Commitment Description:

N/A

Justification for Change:

Improved Operations performance over the last 8 years indicates that there is no longer a need for this formal program. Currently, the corrective action program as identified in station procedures give guidance when an assessment / evaluation should be performed associated with personnel errors.

As examples, corrective action procedures identify that human performance issues impacting plant operation such as, human error with any plant or compliance consequence, significant personnel performance events (HPES), wrong equipment events not impacting plant operations, repetitive human performance issues and mis-positioned equipment should be evaluated.

Concurrently, the procedures also address human performance issues not impacting plant operations. These errors that ultimately do not result in any adverse consequences but could have had adverse consequences if they had not been caught or that could result in adverse consequences if their root causes are not identified and corrected (non-consequential performance problems). Further, repetitive human performance conditions, including events for which previous corrective actions have proven ineffective and wrong equipment events impacting plant operation are also evaluated. STA-513, "Human Performance Enhancement System" directs many of these aspects that were previously captured via the commitment.

CMCE: 00-06
Commitment Number: 23098
Change Type: Deletion

Source Document:

IEB-83-005
TXX-89050
TXX-89250

Original Commitment Description:

Performance test of the spare parts for Station Service Water (SSW) pumps (supplied between 1977 through 1981 [by the Hayward Tyler Pump Company]:

Category 1 Spare Parts: Perform operational test of not less than one hour duration. Record the test data for maintenance performed and for replacement of parts, which should meets the intent of the requirements of the [Hayward Tyler Pump Company] HTPC Operational Maintenance Instructional Manual.

Category 2 Spare Parts: Perform the test as per the Category 1 parts testing and an additional 48 hour performance test per the bulletin requirement, which will assure the intended safety function of the SSW pump.

Revised Commitment Description:

N/A

Justification for Change:

The commitment was for spare parts manufactured during a specific time period. During this time period, procedures and processes were in place to assure compliance with the IEB requirements and assure pump performance would meed licensing and design basis requirements.

Based on engineering review , the spare parts have been exhausted. The identified parts are no longer available for spares from the manufacturer (Hayward Tyler Pump Company), and others procured from different approved vendors are in use. Therefore, the commitment does not continue to have merit.

CMCE: 00-07
Commitment Number: 02963
Change Type: Deletion

Source Document:

IEB-83-005
CAR-89-006, Item B.3
ONE-97-1386
TXX-4024
TXX-89250

Original Commitment Description:

Performance test of the spare parts for Station Service Water (SSW) pumps (supplied between 1977 through 1981 [by the Hayward Tyler Pump Company]:

Category 1 Spare Parts: Perform operational test of not less than one hour duration. Record the test data for maintenance performed and for replacement of parts, which should meets the intent of the requirements of the [Hayward Tyler Pump Company] HTPC Operational Maintenance Instructional Manual.

Revised Commitment Description:

N/A

Justification for Change:

The commitment was for spare parts manufactured during a specific time period. During this time period, procedures and processes were in place to assure compliance with the IEB requirements and assure pump performance would meed licensing and design basis requirements.

Based on engineering review, the identified spare parts have been exhausted via attrition from on-site stores and are not available from the manufacturer (Hayward Tyler Pump Company). Other spares are procured from different approved vendors and are in use. Therefore, the commitment does not continue to have merit.

CMCE:	00-08
Commitment Number:	05963
Change Type:	Deletion

Source Document:

IEB-83-005
TXX-89250

Original Commitment Description:

The procedure for disassembly, inspection, repair and reassembly for the service water pumps has been revised to include the provisions of I.E.Bulletin 83-05, Attachment 3.

Revised Commitment Description:

N/A

Justification for Change:

The commitment was for spare parts manufactured during a specific time period. During this time period, procedures and processes were in place to assure compliance with the IEB requirements and assure pump performance would meed licensing and design basis requirements.

Based on engineering review, the identified spare parts have been exhausted via attrition from on-site stores and are not available from the manufacturer (Hayward Tyler Pump Company). Other spares are procured from different approved vendors and are in use. Therefore, the commitment does not continue to have merit.

CMCE: 00-09
Commitment Number: 02964
Change Type: Deletion

Source Document:

IEB-83-005
TXX-89250

Original Commitment Description:

Hayward Tyler category 2 spare parts testing will include an additional 48 hour pump performance test in addition to the category 1 testing (operational test of no less than one hour duration).

Revised Commitment Description:

N/A

Justification for Change:

The commitment was for spare parts manufactured during a specific time period. During this time period, procedures and processes were in place to assure compliance with the IEB requirements and assure pump performance would meed licensing and design basis requirements.

Based on engineering review, the identified spare parts have been exhausted via attrition from on-site stores and are not available from the manufacturer (Hayward Tyler Pump Company). Other spares are procured from different approved vendors and are in use. Therefore, the commitment does not continue to have merit.