Log # TXX-95010
File # 10112
Ref. # 10CFR50.59(b)(2)

TU ELECTRIC

February 1, 1995

C. Lance Terry
Group Vice President, Nuclear

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) - UNITS 1 AND 2
DOCKET NOS. 50-445 AND 50-446
ANNUAL 10CFR50.59 SUMMARY REPORT FOR 1994

Gentlemen:

Attached is the CPSES Units 1 and 2 annual report required by 10CFR50.59(b)(2) for 1994. This report contains a brief description of the changes, tests and/or experiments implemented or performed pursuant to 10CFR50.59(a), including a summary of each of the safety evaluations. Items in this report are referenced by their 50.59 Safety Evaluation numbers. This report includes those activities which were completed in 1994 and were not reported to the NRC in previous annual reports. This report covers the period from January 1, 1994, through December 31, 1994.

If you have any questions, please contact Mr. Jacob M. Kulangara at (214) 812-8818.

Sincerely,

C. L. Terry

By: D R Woodlan
D. R. Woodlan
Docket Licensing Manager

JMK/jmk
Attachment

c - Mr. L. J. Callan, Region IV
Resident Inspector, CPSES
Mr. T. J. Polich, NRR
Mr. D. D. Chamberlain, Region IV

IE47
1/1

Attachment to TXX-95010

COMANCHE PEAK STEAM ELECTRIC STATION

ANNUAL 10CFR50.59 REPORT
1994

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK UNITS 1 AND 2
ANNUAL 10CFR50.59 REPORT

TABLE OF CONTENTS

This report contains a description and a summary of the following 10CFR50.59 Evaluations:

Evaluation Number
SE-90-041
Revision 4

Activity Title:

LDCR SA-93-141; Allowance of 4000 Square Feet Each of Additional Aluminum and Zinc Inside Units 1 and 2

Description of Change(s):

This activity revises pertinent analyses regarding allowable aluminum and zinc inventories inside containment during Modes 1-4 to support maintenance activities. This revised evaluation takes into account the effects of aluminum and zinc in solutions on Containment radiation levels.

Summary of Evaluation:

The possibility of increased radiation levels due to additional aluminum or zinc inside Containment depends primarily on the possibility of parts or materials containing these elements being exposed to an intense neutron flux during power operation, thereby becoming irradiated. It is not expected that such parts or materials could inadverently be allowed to enter the Reactor Coolant system, thus passing through the core neutron flux region, while the Reactor is at power.

Any potential increase in dose rates due to Al-28 would be of short duration after reactor shutdown and have insignificant radiological impact. Even an unreasonable large quantity of irradiated zince assumed to be released to the Containment would yield post-accident dose rates which are negligible in comparison to those from the fission products postulated to be released per Regulatory Guide 1.4.

Assuming that all of the Al and Zn in Containment dissolves into the aqueous phase, this would produce a very dilute solution of metal cations in the water. A reactor would then be required to deposit the Al or Zn in the stainless steel grain boundaries (the chance of metal ion undergoing the electrochemical reduction directly on a grain boundary are very small). Since the RCS is cool (below 200oF) and depressurized, the effect of embrittling would not be a problem. The temperature needed to "soften" the grain boundary is not present and the stress needed to propagate a crack is not present.

The direct effect to the Containment atmospheric pressure will be a net increase of less than 1.0% following LOCA; therefore, negligible.

Evaluation Number
SE-91-093
Revision 1

Activity Title:

Removal of Feedwater pump suction strainers and associated differential pressure instrumentation.

Description of Change(s):

This activity removes the Unit 2 Feedwater Pump suction strainers and their associated differential pressure switches. The Unit 1 strainers were removed under a previous evaluation. The strainers are prone to leakage following plant trips. The relatively large mesh size strainers were originally installed to prevent possible construction debris from entering the pumps or the S/Gs. It is common industry practice to remove these strainers once the system has been cleaned up. Inspection of these strainers have found them to be clear of debris.

Summary of Evaluation:

All components affected by this change are non-safety related. Since the Feedwater system is clean, removal of the strainers does not affect the system performance (except to reduce condensate leakage following plant trips). It is not expected that this modification will have any impact on the frequency or consequences of any analyzed accident/malfunction or create the potential of a new accident/malfunction.

Evaluation Number
SE-91-127

Activity Title:

DM 90-504 R1; Partial Removal of Reverse Osmosis Equip. from Rm. 165 of Aux. Bldg. & Conversion to I&C Hot Tool Rm. and M&TE Hot Calib. Lab

Description of Change(s):

This design modification (DM) removed selected components of the unused and abandoned Laundary Reverse Osmosis System and associated piping and electrical from Auxiliary Building Room 165 and converted part of the room into a M&TE Hot Calibration Laboratory and I&C Hot Tool Room. A partitioning fence was built inside Room 165 to seperate the hot tool room and laboratory from other commodities in the room which are mostly non-safety related. Installation of the I&C Hot Tool Room and M&TE Hot Laboratory in Room 165 was implemented to more permanetly locate an area for these facilities inside the CPSES Radiation Controlled Area (RCA) and thus more effectively support plant outages.

Summary of Evaluation:

The Laundry Reverse Osmosis (R.O.)System, which was partially removed from Room 165 by this activity, was not utilized for liquid waste processing at CPSES; this system had been previously abandoned in place. Associated piping,instrument air lines and electrical cable related to reverse osmosis system operation were partially removed and determinated/isolated as approriate so as not to impact any other system. The laundry reverse osmosis system is non-safety related; however,the hot tool room and laboratory area being setup in Room 165 contain work tables,tools ,equipment and partition fencing which are also non-safety related and not seismically installed. The existing commodities in Room 165 are mostly all non-safety related except for some class IE electrical conduits and a 24' diameter high energy line on one side wall of the room. To avoid any potential of interaction of the non- safety related commodities with these targets, the partition nearest to the targets was placed 6'-4" away and this portion of the partition was designed to be seismic Category II. Also, restrictions were imposed on the use of the tool room area just inside of this part of the partition. Similar restrictions were stipulated during construction. Based on review of the DM prior to construction/implementation, Safe Zones were established in Room 165 to address satisfactorily the above seismic/nonseismic concerns.

This DM does not introduce any significant additional fire load in Room 165, and the subject R.O. system has not been a part of the accident analysis of the CPSES Liquid Waste Processing System; therefore, this DM does not impact on the existing accident analyses.

Evaluation Number
SE-92-110

Activity Title:

DM 92-063 R0,LDCR SA-92-709; Modification of 480V Common Motor Control Center Transfer Switches and FSAR Figure Update to Reflect Changes

Description of Change(s):

The equipment affected by this change are the automatic transfer switches associated with safety related motor control centers (MCCs), XEB2-1, XEB2-2, XEB3-2, XEB4-2, XEB1-1, and XEB1-2, that can be electrically powered from either Unit 1 or Unit 2 (i.e., common). The transfer switches are designed to transfer power to the available power supply under low voltage conditions of the aligned power source. In the original design, the transfer switch connected the selector switch on the panel door directly to a 480VAC supply without any circuit protection. Since the selector switch is manually operated, the original design could have resulted in a potential personnel hazard. This change involved changing the power supply to the selector switch to 120VAC with fuse protection so that operation of the selector switch will no longer be a potential personnel hazard.

Summary of Evaluation:

The change in power supplies from 480VAC to 120VAC does not affect the function of the circuit. On a low voltage condition, the circuit will still automatically transfer to the alternate power supply. The mechanical linkages and electrical interlocks will still prevent the common MCCs from being powered from both the preferred and alternate power sources. The circuit modifications were all internal to the transfer switch and therefore does not create an electrical separation or fire safe shutdown combustible loading concerns. In addition, the short circuit capability of the transfer switch was reevaluated and the results indicate that the maximum available short circuit current at the transfer switch is well below the amperage rating of the upstream breaker.

There is no unreviewed safety question associated with these activities.

Evaluation Number
SE-92-158
Revision 1

Activity Title:

DCN -5648;LDCRs SA-93-025,TR-92-017; Revision to Safeguards Sequencer and Response Time Requirements

Description of Change(s):

The Technical Requirements Manual, the FSAR and Design Basis Documents are revised to reflect the one second delay in the sequencer self test circuit and to change the containment spray pump response time test to include the pump response assumed in the limiting Containment Analysis. (References DCN-5648; LDCR TR-92-017, Revision 1; and, LDCR SA-93-025.)

Specifically, LDCR# TR-92-017, Revision 1, revises Technical Requirement 1.1, Table 1.2.1, Engineered Safety Features Response Times for Containment Pressure - Hi-1. The response times for the containment spray pumps without Diesel Generator delays is deleted and the response time with DG delays is changed from 27 to 32 seconds to include pump response time. Though it had been indicated that the pumps would be response time tested in the ASME XI IST Program, this is not the case. Note 7 to the revised table is revised to provide clarification. This change brings the TRM into agreement with the Containment Analysis assumptions for pumps response times.

LDCR# TR-92-017, Revision 1, also changes Note 4 to reflect the sequencer design. LDCR# SA-93-025 revises FSAR Sections 6.3 and 8.3 to reflect the one second sequencer delay that is part of the self test circuit. This reset delay affects the performance of sequenced equipment for cases where offsite power is available. DCN-5648 updates the affected Design Basis Documents.

These changes are proposed to disposition engineering resolution of ONE Form 93-0071, which documented failure of containment spray pumps to pass response time tests which included only ESFAS and sequencer response times. The failure revealed that the sequencers do not comply with original design requirements as stated in the FSAR and Westinghouse interface requirements.

Summary of Evaluation:

This change brings the TRM into conformance with the bases and acceptance limits assumed in the accident analyses. Failure values are not affected. No credible failure modes are introduced. The one second delay did not meet original requirements; however, it does not affect the safety analysis assumptions. Although design margins are reduced, there is no affect on the margin of safety.

Evaluation Number
SE-93-014

Activity Title:

DM 91-054, Rev 2; LDCR SA-93-036; Waste Water Management System Modification,Phase IV;Revision to Associated FSAR Figures in Sect. 9.2

Description of Change(s):

DM 91-054 Rev. 2 implements Phase IV of the Waste Water Management System modification. LDCR-SA-92-0667 was written as a result of Phase III and was incorporated, except for the figure changes, into the FSAR as a part of Amendment 87. The figure changes have already been addressed in Safety Evaluation SE No. 92-72, this activity will add a Condensate Polisher Decant Basin and Clarifier Sludge Decant Basin to the figures originally submitted under LDCR SA-92-0667. (NOTE: The figures originally submitted under LDCR SA-92-72 will be resubmitted as a result of this DM revision).

Summary of Evaluation:

The system is non-safety related and the addition of the basins will not change Safety Evaluation SE-No. 92-72; therefore, SE No. 92-72 is applicable to this activity and an additional evaluation is not required.

Evaluation Number
SE-93-036

Activity Title:

DM 93-029 R0, LDCR SA-93-078; Ion Chromatograph Installation

Description of Change(s):

This modification consist of installing new enclosure CPX-SSMESS-01 in the Unit 1 Turbine Building on floor elevation 778', routing new sample lines from the secondary sample panel CP1-SSPASS-01 to the new enclosure and installing larger replacement impellers in the Hotwell Sample Pumps A and B. The new enclosure is equipped with necessary support services consisting of conditioned and unconditioned electrical power, air conditioning, lighting, demineralized water, and nitrogen gas. The Ion Chromatograph system is common to both Unit 1 and Unit 2 is installed inside the new enclosure. A new fire protection sprinkler head is routed to the enclosure.

Summary of Evaluation:

There are no credible potential failure modes involving the structures, systems or components that are affected by this modification. The only parameters affected by implementation of this modification are the electrical loadings imposed by the addition of the new enclosure and the new Ion Chromatograph and the samples' flow rates imposed by the addition of the new sample lines and the new Ion Chromatograph. The electrical loadings have been analyzed and found not to cause source overload conditions and the samples' flow availabilities have been analyzed and found to have a negligible impact on transport time or Ion Chromatograph operation. The modification is designated non-safety related. Addition of a new sprinkler head meets NFPA criteria and will have a negligible effect on Turbine Building wet pipe sprinkler coverage.

Evaluation Number
SE-93-042

Activity Title:

LDCR SA-93-082; Revision to FSAR to Reflect Acceptance of Less Than Minim. Separ. Betw. Class 1E Cabls and Non 1E Difrnt. Prot. Relay Cabls

Description of Change(s):

The FSAR is revised, to identify that the minimum physical separation required per IEEE-384, is not maintained between Class 1E cables and non Class 1E differential protection relay (87/ST1 & 87/ST2) cables for startup transformers inside the Class 1E switchgear cubicles. Additionally FSAR section 8.3 is revised to provide the analysis for the acceptability of this as-installed condition for the differential protection relay cables.

Summary of Evaluation:

The differential protection relay cables from 6.9kv Class 1E preferred/alternate source breaker CT's are designed such that they will function adequately and will not fail due to circuit voltage and current conditions to which, these might be exposed, due to open circuit & short circuit conditions. As such there is no potential credible failure mode for these cables. The cables will maintain their integrity and will not adversely affect the Class 1E cables in the vicinity of these cables inside the cubicles. Therefore, the above identified non Class 1E cables used in differential relay protection for startup transformers are not required to meet the minimum separation requirement of IEEE-384 inside the 6.9kv Class 1E switchgear.

The Licensing Document Change Request (LDCR SA-93-082) revises the FSAR sections 1A(B) and 8.3 to reflect these conditions and to provide justification/analysis for not requiring the minimum physical separation for startup transformer non Class 1E differential protection relay cables inside the Class 1E switchgear cubicles.

The implementation of this activity therefore does not involve an unreviewed safety question.

Evaluation Number
SE-93-079

Activity Title:

LDCRs SA-93-87, 88, 89, 90, 91, & 92; MMs 93-438, 93-439, 93-440, 93-566, 93-442 & 93-443; Removal of Internals of Eleven Check Valves

Description of Change(s):

These minor modifications delete the internals of eleven (11) check valves located in Auxiliary Feedwater (AF), Component Cooling (CC), and Station Service Water (SW) Systems.

Summary of Evaluation:

The implementation of these minor modifications do not introduce any new failure modes to the systems affected. Potential system interaction effects such as high and moderate energy lines, flooding, shutdown logic, etc. are also found to be acceptable with no impact. These check valves are either duplicating a function already being performed by other check valves in the system or are not required to perform an isolation function under any system operating modes.

Evaluation Number
SE-93-081

Activity Title:

DM 92-066,LDCR SA-93-144; Interim Low Level Radwaste Storage Facility

Description of Change(s):

This activity pertains to a plant design modification for preparation and use of an Interim Low Level Radwaste Storage Facility (ILLRSF) at CPSES. Interim low level radioactive waste storage at CPSES is required during the time in which currently licensed radioactive waste disposal facilities are unavailiable and before the Texas disposal facility becomes availiable. The Barnwell, South Carolina radioactive waste disposal site became unavailiable to CPSES as of June 30, 1994; the earliest expected availability of the Texas radioactive waste disposal facility is 1996 or 1997. Warehouse C and areas adjacent to Warehouse C are used for the ILLRSF at CPSES.

Summary of Evaluation:

The design and operation of the ILLRSF at CPSES was reviewed using appropriate NRC guidance documents including GL 81-38 to assure that radiological consequences of design basis events are within acceptable limits. The ILLRSF is physically removed from the primary plant. Radwaste (i.e., processed dry active waste (DAW), dewatered spent resins and filters) is packaged in a form suitable for transportation and/or disposal. DAW metal boxes are stored inside the warehouse; containers with dewatered resins and filters are placed in concrete vaults in a fenced area adjacent to Warehouse C. The ILLRSF uses support services interconnected to the plant (telephone, gaitronics, fire protection alarms & water,electrical power). The first two are not related to safe operation of CPSES. Fire protection, alarm and water are supplied by the site fire protection system which notifies/alarms the central fire alarm and Control Room. Electrical power is supplied from off-site non-safety related sources.

Evaluation considered 3 design basis events which may present an interaction between the ILLRSF and the primary plant. These are tornado, flooding and fire. For tornado, no significant unbounded interaction between the ILLRSF, or the housed materials, and the primary plant is expected to occur. For externally generated flood scenario, no release of radioactive effluent is anticipated. The amount of water ingress to the ILLRSF is expected to be bounded by the internally generated flooding scenario, upon activation of the fire suppression system. For fire, the combustion of boxed DAW is not deemed to be a credible event. Combustible material loading and ignition sources for the ILLRSF are less than those for the current Warehouse C use. A fire at the ILLRSF would not be an initiating event for the primary plant and would not introduce new failure modes. The failure of the support services would not degrade the safety margin for CPSES.

Evaluation Number
SE-93-081

The evaluation concludes that there is no impact on any of the existing safety analyses reported in the CPSES licensing basis documents; however, there are credible accident and design basis events associated with the ILLRSF which could result in the potential release of radioactive materials to the environment. These releases are of a type not explicitly documented in the licensing basis documents; however, the potential radiological consequences have previously been evaluated in Safety Evaluation SE-91-062 (Rev 5) and reported to NRC with the 1993 annual summary report.

Evaluation Number
SE-93-082

Activity Title:

TM 93-2-17 R0; Addition of Nitrogen Gas to Reactor Makeup Water Pump (RMUWP) Mini-flow Header Downstream of the Flow Orifice

Description of Change(s):

The level of Dissolved Oxygen (DO) in the Reactor Makeup Water Storage Tank (RMUWST) for Unit 2 is excessively high. The condition has occurred several times during the startup phase for Unit 2. When the level of DO exceeds 100 ppb, the water is typically dumped to the drain and low DO water is then placed in the tank. The present location that nitrogen gas is added to the RMUWST to maintain an inert atmosphere is above the water level. This modification would add nitrogen gas to the Reactor Makeup Water Pump (RMUWP) mini-flow header downstream of the flow orifice.

This should allow the DO content to be reduced without having to replace the water in the RMUWST. This will result in reducing the amount of water that will have to be processed by waste processing as radioactive waste and provide data that may be used to modify the nitrogen gas blanketing system.

Summary of Evaluation:

The introduction of nitrogen gas bottles in the Auxiliary Building will not have a significant effect on the seismic qualification. An engineering evaluation of the structure and restraints used has determined that there is no safety concern that would cause a structural failure.

The accidents evaluated in the licensing basis documents do not utilize the equipment that could malfunction as a result of two potential malfunctions of equipment, gas binding a RMUWP or over-pressurization of the RMUWST. Neither event would cause a loss or failure of equipment required to shutdown the plant or affect radiological consequences.

Evaluation Number
SE-93-093

Activity Title:

LDCR TR-93-012; Revision to TRM Tables 4.1.1a and 4.1.1b to Delete 8 Circuit Breakers from Technical Specification Surveillance Requirement

Description of Change(s):

This activity is for revising the Technical Requirements Manual (TRM) tables 4.1.1a and 4.1.1b for deletion of eight 480 VAC Motor Control Center circuit breakers from the testing requirements of Technical Specification 3/4.8.4 for containment penetration conductor overcurrent protection devices.

Summary of Evaluation:

TRM identifies circuit breakers required to meet the surveillance testing requirements of Technical Specification 3/4.8.4 (for containment penetration conductor overcurrent protection devices).

Eight circuit breakers identified in the TRM, provide overcurrent protection to Motor Operated Valves (MOVs) located inside valve isolation tanks. The isolation tanks (located outside the reactor containment) have electrical penetrations. The associated MCC circuit breakers (MOVs/ isolation tanks) are 1EB3-2/9RF (1-HV-4782/ CP1-CTATVT-01), 1EB4-2/8RF (1-HV-4783/ CP1-CTATVT-02), 1EB3-2/9RM (1-8811A/ CP1-RHATVT-01), 1EB4-2/8RM (1-8811B/ CP1-RHATVT-02), 2EB3-2/9RF (2-HV-4782/ CP2-CTATVT-01), 2EB4-2/8RF (2-HV-4783/ CP2-CTATVT-02), 2EB3-2/9RM (2-8811A/ CP2-RHATVT-01) and 2EB4-2/8RM (2-8811B/ CP2-RHATVT-02). The valve isolation tanks were originally part of the containment barrier under the original design basis. Design and licensing documents were revised prior to receiving the Units 1 and 2 operating licenses to exclude these tanks from being a part of the containment barrier. The valve isolation tanks are open to the atmosphere and used for leak detection only as described in the FSAR. Because these isolation tanks are no longer considered as part of the containment, the tank's electrical penetrations are not containment penetrations and therefore the associated circuit breakers no longer require the surveillance testing of Technical Specifications 3/4.8.4.

LDCR TR 93-012 revises the TRM to delete the eight circuit breakers from the specified tables inorder to delete from the above stated surveillance requirements. This revision to the TRM also makes the document correct and consistent with the FSAR as the TRM changes were inadvertently omitted earlier.

The activity will not create the possibility of an accident different from any accident evaluated in the Licensing Basis Documents because the activity will not affect the design, performance, or process parameters of the circuit breakers, MOVs and valve isolation tanks. The implementation of this activity will not introduce new, credible failure modes for the associated components and does not involve an unreviewed safety question.

Evaluation Number
SE-93-094

Activity Title:

DM 93-006 R1, LDCR SA-93-121; HVAC Drain Line Modifications

Description of Change(s):

This modification consists of rerouting the condensate drainage piping from Auxiliary Building (AB) HVAC Demister CPX-VAMEDM-03, and reworking loop seals on drain lines from Primary Plant Ventilation Supply units. The Demister drain line will terminate at a floor drain on Auxiliary Building Elevation 852'-6" which leads to the Component Cooling Water Drain Tank.

Summary of Evaluation:

This modification involves no safety-related structures, systems, or components, and does not involve an unreviewed safety question.

Evaluation Number
SE-93-101

Activity Title:

LDCR SA-93-130;Deltn.of Chlorides as a Continuously Measured Parameter from the Main Steam after MSR Sample Stream, from FSAR Table 10.4-20

Description of Change(s):

Deletes chlorides as a parameter continuously measured from the Main Steam after MSR Sample Source, from FSAR Table 10.4-20 "Secondary Plant Sampling System Measured Parameters".

Summary of Evaluation:

The conclusion of the Evaluation is that no unreviewed question exists, and an amendment to the Technical Specification is not required as a result of this activity. This activity does not impact any accident/ malfunction analysis nor any likely hood of any existing analyzed accident or malfunction increased as the monitoring of chlorides (and other anions) within the secondary cycle is continuously performed via inline cation conductivity analyzers located in the Hotwell, Condensate Pump Discharge, Heat Drain Pump Discharge, Polisher Outlet,Final Feedwater,Steam Generator Blowdown and Main Steam after MSR Sample Streams. This activity thereby eliminates redundant monitoring and unwarranted expenses incurred to maintain and calibrate analyzer.

Evaluation Number
SE-93-103

Activity Title:

MM 93-567,-568,-569;LDCR SA-93-140;Removl of Unit 2 Chk Vlve Internals for 2SW-0014,-00116,-0017,0048,-0084,-0085;2CC-0317,-0602/Update FSAR

Description of Change(s):

Minor Modifications MM93-567, 93-568 and 93-569: Remove the internals of following Unit 2 check valves; 2SW-0014, 2SW-0048, 2SW-0084, 2SW-0085, 2SW-0016, 2SW-0017, 2CC-0317, and 2CC-0602.

Summary of Evaluation:

This activity proposes the removal of internals for eight (8) Unit 2 check valves. There are no credible operating scenarios under which these valves are required to perform a "closed" safety function.

This proposed activity is similar to activity described in MM93-438 through 93-440, MM93-442, 93-443, and 93-566 which cover the removal of internals for the corresponding Unit 1 check valves.

Evaluation Number
SE-93-104

Activity Title:

LDCR SA-93-142; Rev to FSAR Sect. 1AB to Take Exception to RG 1.84 Limitation on Service/Shelf Life for Elastomer Diaphragms at CPSES

Description of Change(s):

The change is being issued to document the exception taken to the additional requirements regarding service/shelf life included in Regulatory Guide 1.84 for the contingent approval of ASME Code Case N31-1 for elastomer diaphragms used in ASME Class 2 and 3 applications. The change imposes the following conditions: (i) the service life of the elastomer diaphragms should be the lower of that determined based on (a) 1/2 of the average number of cycles from a minimum of three tests per the Code Case and (b) total anticipated radiation exposure during applicable operating modes determined based on valve operability requirements; and (ii) the shelf life of elastomer diaphragm should be determined in accordance with CPSES procedures for shelf life determination.

Summary of Evaluation:

The Regulatory Guide imposes generic service life and shelf life limitations for elastomer diaphragms, regardless of the material, in addition to the basic Code Case Requirements. CPSES uses diaphragms made from a material which has been extensively tested in accordance with the requirements of Code Case N31-1. The evaluation of diaphragm test results in conjunction with CPSES-specific system parameters resulted in a calculated diaphragm service life in excess of five years for most valves. While the Regulatory Guide 1.84 limitation on service life may be appropriate with regard to a generic reference to "elastomer diaphragms", it is overly restrictive for the material specified for use at CPSES. Diaphragm material aging tests, as part of the testing required by the Code Case, show no consequential degradation of the EPDM diaphragms. The CPSES procedurally controlled shelf life program is based on industry (e.g., EPRI NP-6804 Guidelines for Establishing, Maintaining, and Extending the Shelf LIfe Capability of Limited Life Items) likewise support the conclusion that EPDM is relatively insensitive to aging degradation.

Evaluation Number
SE-93-105

Activity Title:

LDCR OD-93-001; Revise Offsite Dose Calculation Manual (ODCM), Monitor Alarm Setpoint Methodology for Liquid Waste Monitor XRE-5253

Description of Change(s):

The monitor alarm setpoint methodology for primary liquid effluent monitor XRE-5253 (final monitor in primary liquid waste stream discharge) is revised to use only the gamma emitting radionuclides in the liquid effluent stream instead of a combination of gamma emitting radionuclides plus non-gamma emitting radionuclides (e.g., tritium, alpha emitters, Fe-55, Sr-90, Sr-90, etc.). This change is necessary because in certain instances the current methodology calculates the alarm setpoint to be overly conservative; the overly conservative alarm setpoint then unnecessarily alarms and/or terminates approved and permitted routine liquid releases.

Summary of Evaluation:

Under the current alarm setpoint methodology for liquid effluent monitor XRE-5253, when release permitting liquid effluent with a very low gamma radionuclide concentration, the resultant calculated alarm setpoint may be overly conservative. The calculational methodology is being influenced by one major factor that leads to the overconservatism. The problem is that monitor XRE-5253 is a gamma only sensitive monitor and, at low gamma concentrations, a factor related to the presence of tritium and composite concentrations for alpha emitters, Fe-55, Sr-89 and Sr-90 contributes disproportionately to the calculated setpoint value. Since the monitor does not dynamically respond to the liquid effluent stream's non-gamma emitting radionuclides, the monitor's setpoint alarm value should not be biased by these radionuclides. The liquid monitor alarm setpoint is typically used to indicate an operational problem with a permitted, ongoing release; secondarily, it indicates compliance with 10CFR20 liquid concentration limits. The actual 10CFR20 compliance verification for a specific batch liquid release is performed prior to release using other station and ODCM procedures. The proposed new alarm setpoint methodology relies only on gamma emitting radionuclides in the liquid effluent stream, maintains the 10CFR20 compliance verification of other plant procedures, and will eliminate operational problems as described above. The new monitor alarm setpoint methodology is more appropriate to the application; relative concentrations of the non-gamma emitting radionuclides should not change unexpectedly. If they do change significantly, gamma radionuclides are also expected to change and indicate a release problem.

The monitor setpoint being determined by this calculational methodology change has no impact on the safe operation of CPSES. The automatic alarm/termination of the liquid effluent release does not affect systems important to reactor safety.

Evaluation Number
SE-93-107

Activity Title:

LDCR SA-95-005; Revise Hot Leg Recirculation Switchover Time from 15 Hours to 6 Hours; Revision to Associated Procedures

Description of Change(s):

This activity revises the hot leg recirculation switchover time from 15 hours to 6 hours. This reduction in the hot leg recirculation switchover time is necessary because of the increase in boron concentrations for the RWST and ECCS accumulators to accommodate shutdown margin and safety analysis requirements associated with extended reload cycles.

Summary of Evaluation:

Hot leg recirculation is initiated to prevent boron precipitation in the core following a postulated LOCA. The concern of boron precipitation is due to boron build-up as a result of boiling in a stagnant core region. Boron precipitation will cause the degradation of heat transfer from the fuel rod, and thus may result in cladding heatup and potential fuel damage. The boron builds up more rapidly as the ECCS boron concentrations are increased. Thus, this concern is addressed by decreasing the hot leg recirculation switchover time. The revised switchover time is being incorporated into appropriate plant procedures and licensing documents.

Evaluation Number
SE-93-108

Activity Title:

LDCR TR-93-015; Revision to TRM Tables 4.1.1.a and 4.1.1.b to Delete Relay 51M2 as Primary Electrical Penetration Protection

Description of Change(s):

The Technical Requirements Manual (TRM) is revised to delete the containment penetration conductor overcurrent protective relay 51M2 from the TRM tables 4.1.1.1a and 4.1.1.1b.

Summary of Evaluation:

TRM tables 4.1.1.a and 4.1.1.b require relays 50M1-51 and 51M2 as primary protection for Reactor Coolant Pump (RCP) electrical penetration conductors. This requirement is applicable for modes 1 through 4. Relay 50M1-51 provides adequate primary protection for RCP electrical penetration conductor protection against short circuit and overcurrent faults during modes 1 through 4. However relay 51M2 provides RCP motor protection against overcurrent fault during hot loop load and is inadvertently listed as RCP primary electrical penetration protection relay.

Since relay 50M1-51 provides adequate RCP primary electrical penetration protection, relay 51M2 is being deleted from TRM requirement.

Therefore the deletion of 51M2 relay from the TRM tables will have no adverse effect on protection of RCP electrical penetration conductors. This activity does not involve any credible potential failure, will not create probability/possibility of new accident/malfunction of equipment important to safety of the plant and does not involve unreviewed safety question. Also, it will not affect margin of safety.

Evaluation Number
SE-93-109

Activity Title:

LDCR SA-93-126; Revise FSAR Chapter 11 to reflect updated radiological analyses for CPSES operation with an extended (18 month) fuel cycle

Description of Change(s):

CPSES operation with an extended fuel cycle (18 months vs. 12 months) will increase the fission product inventory in the reactor cores and result in changes to the normal operation radioactive gaseous and liquid effluents released from the plant. This activity revises the FSAR to reflect updated evaluations of the estimated radionuclide concentration in liquid and gaseous effluents and the maximum predicted offsite dose expected from radioactive effluent release pathways. Also, this FSAR revision deletes the reference to 10 CFR 50, Appendix I, Docket RM-50-2 design objective criteria and in place specifies operational objectives of 10 CFR 50, Appendix I.

Summary of Evaluation:

Updated evaluations resulted in slightly higher expected equilibrium concentrations of some radioruculides in the reactor coolant and in various CPSES liquid wastes streams; however, CPSES gaseous waste streams were not significantly affected and therefore related FSAR Chapter 11 information for gaseous radionuclide effluents was not changed in this revision.

Although the analyses are not safety related, the estimated offsite radiation doses will increase and, with conservatism, exceed 10CFR50,Appendix I, Docket RM-50-2 design objectives but remain within 10CFR50, Appendix I, site criteria. The commitment to RM-50-2 was met by the Liquid Waste Processing System (LWPS) as described in the FSAR during construction which eliminated a requirement to perform a cost-benefit analysis for the LWPS at that time. Subsequently, during operations, the design objective limits of 10CFR50, Appendix I are applicable. Calculated offsite doses after considering the extended fuel cycle operations remain below the applicable numerical limits of 10 CFR 50, Appendix I.

Evaluation Number
SE-93-118

Activity Title:

LDCR SA-93-147; Revision to FSAR Section 1AB to delete RG 1.33 requirement to perform a biennial review of routine plant procedure

Description of Change(s):

Currently the FSAR, by endorsing Regulatory Guide 1.33 Revision 2, 1978, requires that plant procedures be reviewed no less frequently than every two years. This activity replaces the biennial review requirement with CPSES programmatic controls already in place. These controls initiate procedure reviews and changes upon identification of new or revised source material or other information to ensure procedural adequacy.

Summary of Evaluation:

The intent of the biennial review is accomplished by the following CPSES programmatic controls already in place:

- Site Modification Process
- Corrective Action Program
- Off-Normal Occurrence
- User Feedback and Procedure Compliance
- Operating Experience Review
- Vendor Technical Information
- Licensed Document Change/50.59 Evaluation
- Commitment Tracking System (CTS)
- Trending
- Infrequently Performed Evolutions
- Requalification Training
- Quality Assurance Activities

These controls ensure equivalent or better procedure adequacy than was provided by the biennial review and thus the change does not result in an unreviewed safety question.

Evaluation Number
SE-93-119

Activity Title:

PCN STA-202-23-05;Removal of Nuclear Overview Department from in line reviews of quality related procedures not subject to FSAR Jurisdiction

Description of Change(s):

The FSAR is the primary source for Nuclear Overview Department in-line reviews of station procedures/instructions. The NOD in-line review of safety-related instructions used to perform detailed work activities which are unique to a particular department duplicates the NOD evaluation activities (ie audits, assessments, surveillances, task teams and industry reviews) on the same instructions and is therefore not required.

Summary of Evaluation:

The review functions with respect to procedures of all organizations are detailed in administrative procedures. The procdure change was reviewed and it was determined that the change does not result in a change to a test/experiment, the facility, or to the procedures as described in the Licensing Basis Documents, nor does it involve a change to the Technical Specifications.

Evaluation Number
SE-93-121

Activity Title:

Replacement of the CPSES Units 1 & 2 Containment Analyses

Description of Change(s):

This activity replaces the Stone and Webster Engineering Corporation (SWEC) containment pressure/temperature (P/T) analysis methodology, the LOCTIC code and the resulting containment P/T analyses for postulated LOCA and MSLB scenarios (FSAR Section 6.2), with TU Electric methodology, CONTEMPT-LT/028 code (as modified by TU Electric), and the corresponding analyses.

Summary of Evaluation:

The Unit 1 and 2 containment P/T response to postulated LOCA and MSLB scenerios were reanalyzed to integrate the effects of design modifications performed on the Component Cooling Water system (see SE-94-015). The results of the reanalysis were:

- the containment P/T design limits were met,
- the P/T envelopes for the LOCA scenarios remained valid, and,
- the equipment qualification (EQ) P/T envelopes for MSLB scenarios were revised to bound the P/T envelopes.

Separate evaluations were performed to ensure that all EQ acceptance limits remained valid and no credible failure modes were associated with the implementation of the activity.

Evaluation Number
SE-93-122

Activity Title:

LDCR FP 93-006;Revsn.to Sect. III of FPR,the Fire Safe Shutdwn Equip. List to Add SI Accum.Isol.Vlves for Mnual Opr.to Vent SI Accum.

Description of Change(s):

The ICNs I002710 and I002711 input valves 1SI-8950A,B,C,D for Unit 1 and 2SI-8950A,B,C,D for Unit 2 into the INDMS database for application in running the Fire Safe Shutdown Analysis (FSSA). These valves are added to the FSSA equipment list for manual operation to vent the SI accumulators. Operation of these valves to vent the nitrogen from the SI accumulators reduce the pressure to prevent injection as Reactor Coolant System pressure is reduced. This accomplishes a function equivalent to closing the accumulator isolation valves.

Summary of Evaluation:

In response to IN 92-18 an alternate method of achieving cold shutdown was evaluated. The "hot short" of the SI accumulator isolation valves as a result of a control room or cable spread room fire, prevents the manual operation of these valves to isolate the accumulators, when depressurizing to cold shutdown. An alternate method of preventing injection is to vent the accumulator. The ICNs being evaluated document these valves in the FSSA database and the Licensing Document Change Request (LDCR) adds these valves to the Fire Protection Report (FPR) Fire Safe Shutdown equipment list.

Evaluation Number
SE-93-123

Activity Title:

DM 93-056R0;LDCRS TR-94-001,SA-94-002; Provide Plant Support Power for Unit 2 and Common Safety/Nonsafety Rltd.Equpmnt. Needed During Outages

Description of Change(s):

Certain Unit 2 and common equipment is provided with an outage power supply. Most non-Class 1E equipment is permanently relocated to Plant Support Power. For Class 1E equipment, a key-locked manual transfer switch is added to allow continued use of the following equipment during Modes 5 and 6 when the original 480 Volt power supply is out of service. This Class 1E equipment includes Bypass Transformers, Battery Chargers, Battery Room Exhaust Fans and Lighting Transformers.

Summary of Evaluation:

The design modification will provide a Class 1E manual transfer switch and outage power supply from the Plant Support Power System to certain Class 1E equipment. The original power source is occasionally required to be out of service during Modes 5 and 6 for maintenance and testing. During these outage periods, the affected loads are required to maintain desirable plant operation.

This change will provide a non-Class 1E outage power source to certain Class 1E equipment that will be available during the outages of the original power source. Operation of the transfer switch is limited to plant Modes 5 and 6. Prior to operation of the transfer switch, the associated equipment must be declared inoperable and the circuit breakers for 1E power and Plant Support Power will both be open. After the switch is aligned to the desired source, the appropriate circuit breaker is closed. Thus, the affected equipment is not credited for being operable and equipment from the other train satisfies any minimum operability requirements. The use of the outage source will make equipment available above the minimum required even though no credit is taken for their availability.

When aligned to Plant Support Power, adequate protection to the 1E power cable and other cables which may share raceways, is ensured through the use of either a circuit breaker in series with a fuse or two circuit breakers in series. In either case, both protective devices are selected to prevent cable damage during fault conditions.

When fed from Plant Support Power, the cables are protected by Square D circuit breakers of the same ratings as the original equipment. Loads located inside containment have two circuit breakers in series to protect the penetration assembly. There is no unreviewed safety question associated with this activity.

Evaluation Number
SE-93-124

Activity Title:

LDCR EP-93-008; Revise CPSES Emergency Plan to Reflect Implementation of Federal Protective Action Guidance EPA-400-R-92-001

Description of Change(s):

This activity revises the CPSES Emergency Plan to reflect new federal guidance (EPA-400-R-92-001,"Manual of Protective Action Guides and Protective Actions for Nuclear Incidents") related to the development of plans and procedures for protective measures. 10CFR50.47(b)(10) requires that the CPSES Emergency Plan guidelines for the choice of protective actions during an emergency be consistent with federal guidelines. NRC Information Notice 92-38 states that the implementation date of the new EPA guidance should be consistent with implementation of the revised 10 CFR Part 20 regulations; the revised 10 CFR Part 20 regulations and new EPA guidance were jointly implemented by the State of Texas and TU Electric beginning January 1, 1994.

Summary of Evaluation:

This change to the Emergency Plan reflects that Protective Action Guides (PAGs) are now specified in units of Total Effective Dose Equivalent (TEDE) and Adult Thyroid Committed Dose Equivalent (Thyroid CDE) instead of Whole Body exposure and Child Thyroid exposure as previouly based on guidance from EPA 520/1-78-016. The change also revises Emergency Plan guidance on dose limits for emergency response workers with respect to actions for "lifesaving" or "operating/saving critical equipment/property" and adds a clarifying statement that decisions regarding evacuation, sheltering or relocation are the responsibility of the local county government.

This change does not decrease the effectiveness of the CPSES Emergency Plan, is administrative and programmatic in nature and does not involve any plant primary or secondary system equipment. There is no impact on plant safety or existing accident analyses described in the licensing basis documents.

Evaluation Number
SE-93-125

Activity Title

LDCR SA-93-167, PCN CHM-503-10 R7: Revision to FSAR Section 10.3.5, Water Chemistry, to Update the Secondary Chemistry Program Description

Description of Change(s):

1. Deletes reference to the Westinghouse manual as a controlling document for the secondary chemistry program.

2. Revises the requirement to adhere to the EPRI secondary guidelines except where industry practice and/or CPSES technical evaluations exceed and improve those specifications to allow exception or differences justified by CPSES technical evaluations according to the EPRI guidelines format and philosophy of revision 3.

3. Deletes FSAR Table 10.3-10. Removes parameters and values for the secondary chemistry program.

4. Adds wording to clarify the secondary water chemistry program with the philosophy of the current revision (3) of the EPRI secondary water chemistry guidelines.

Summary of Evaluation:

This change removes detail duplicated in chemistry procedures from the FSAR, relegates the detail of the secondary chemistry program to station procedures, and recognizes the program will progress with the development of the EPRI secondary chemistry guidelines, with which Westinghouse concurs, and meets the Branch Technical Position MTEB 5-3, Revision 1 commitments. This modificaiton does not represent any change or affect any structure, system, or components and/or system parameters as related to safety in any fashion as no current secondary chemistry specification or limit is changed as described by the EPRI seondary chemistry guidelines and CPSES Technical Specifications.

Evaluation Number
SE-93-126

Activity Title:

LDCR EP-93-009; Revise CPSES Emergency Plan to Delete Requirement for Manual Backup Dose Assessment Method & Add Multiple Computer Locations

Description of Change(s):

This acitivity revises Emergency Plan Section 7.2, "Calculation Of Offsite Doses" and Appendix K to delete description of a backup manual dose assessment method. Section 7.2 is revised to indicate that backup dose assessment capability is provided by existing multiple dose assessment computer locations at CPSES.

Summary of Evaluation:

The manual dose assessment method described in procedure EPP-300, "Manual Calculation of Offsite Dose Rates" has in the past been slow and prone to errors. Changes to the CPSES Emergency Plan required by implementing the federal guidance of EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", would make manual calculation even more cumbersome due to the need to account for ground deposition and ingestion doses for the Total Effective Dose Equivalent (TEDE). A "backup" method of performing offsite dose assessment is required under declared emergency condtions; however, a manual calculational method is not the only "backup" option/capability. This revision deletes the manual method of EPP-300 and adds that backup dose assessment is now provided by the seperate and independent computer systems (personal computers) availiable in each emergency response facility. These computer systems have uninterruptable power supplies and incorporate new dose terminology related to EPA-400-R-92-001 and new 10 CFR Part 20.

These changes do not decrease the effectiveness of the CPSES Emergency Plan, are administrative and programmatic in nature, and do not involve any plant primary or secondary system equipment. The subject dose assessment computer systems are existing "stand-alone" personal computers and are not linked to plant operation computers. There is no impact on plant safety or existing accident analyses described in the licensing basis documents.

Evaluation Number
SE-94-001

Activity Title:

LDCR SA-94-003; Revision to FSAR to Identify the Separation Requrmnts. for Redundant Train Cables and Thermo-Lag Enclosures

Description of Change(s):

FSAR section 1A(B) (pertaining to Regulatory Guide 1.75 regulatory position C.6) is revised to identify that there is no separation requirement between Thermo-lag protected cable/ conduit/tray and it's counterpart in the redundant train when these are protected by One-Hour Fire rated Thermo-lag installed to satisfy requirements described in FSAR section 9.5.1.2. Also FSAR section 8.3 is revised to provide analysis which conclude to not requiring separation under these conditions.

Summary of Evaluation:

At CPSES, one inch separation is maintained between Thermo-lag fire barriers which enclose and protect cables needed to safely shutdown the plant under fire conditions (FSSD cables) and redundant train cables routed adjacent to the enclosures. This is the same separation requirement used to satisfy minimum spacing requirements of Regulatory Guide 1.75 where separation barriers (i.e., metal tray covers) are utilized to maintain minimum fire separation spacing between various redundant train cables not associated with FSSD.

Where Thermo-lag fire barrier enclosures are used to protect FSSD cables, the fire barrier qualification requirements are much more stringent than the requirements of RG 1.75 and because of these qualifications, an additional one inch air gap is not needed to ensure adequate protection of cables internal to an enclosure from a cable fault (fire) in a redundant cable outside the enclosure. Based on a Generic Letter 86-10 analysis performed as part of this safety evaluation, Thermo-lag enclosure is verified to adequately protect all external, redundant train cables from a cable fault (fire) inside the enclosure.

Therefore, where Thermo-lag enclosures are used to protect FSSD cables, an additional one-inch air gap is not required to ensure adequate fire separation of redundant train cables outside an enclosure from cables inside an enclosure. There is no unreviewed safety question associated with this activity.

Evaluation Number
SE-94-003

Activity Title:

LDCR SA-94-008; Elimination of Annual and Conditional Whole Body Count Requirements for Radiation Workers from FSAR

Description of Change(s):

Revise the FSAR, Sect on 12.5.3.5, to delete the requirement to perform routine annu.l and conditional whole body counts on CPSES radiation workers.

Summary of Evaluation:

Revised 10CFR20, implemented at CPSES on January 1,1993, does not require monitoring of internal occupational intake of radioactive materials (by whole body count or any other means) for individuals unless the individual is likely to receive in one year an intake in excess of 10% of the applicable Annual Limit on Intake. Routine annual whole body counts have been performed on CPSES radiation workers for the first three years of commercial plant operation yielding no positive whole body counts above action levels. Furthermore, results from a technical evaluation of the plants' Personnel Contamination Monitors (PCMs) capability for detecting internal contamination indicates the ability to detect 1-3% of the Annual Limit on Intake for the major radionuclides present in the plant environment.

Based on the new 10CFR20 regulation, results from the PCM technical evaluation and whole body count history, elimination of the annual and conditional whole body counts has been determined to be acceptable. Whole body counts performed on a case-by-case basis (i.e., as indicated by PCM) will satisfy the requirements of NRC Regulatory Guide 8.9.

Annual and conditional whole body counts are used to evaluate internal dose of personnel and have no other purpose with regard to plant structures, systems, or components. Therefore, no structures, systems or components and/or system parameters could be affected by elimination of the requirement for these whole body counts.

Evaluation Number
SE-94-004

Activity Title:

LDCR SA-94-014; Revision to Section 17 of the FSAR to Reflect the Reorganized Nuclear Overview Department

Description of Change(s):

This change reflects the reorganization of the Nuclear Overview Department to support an operational 2 unit facility.

Summary of Evaluation:

This change reflects the restructuring of the Nuclear Overview Department into functional area responsibilities such that resources are aligned along the same lines as the organization of plant programs and processes. This will facilitate focused overview of these programs and provide clearly defined ownership and accountability. These organizational changes will result in a more effective overview function and allow for better utilization of manpower.

This is an administrative change only and results in no functions described in the FSAR being deleted or reduced. These changes do not represent a reduction in commitment to the QA Program and therefore no impact exists on the margin of safety or any unanalyzed events.

Evaluation Number
SE-94-007

Activity Title:

DM 93-049,-050;LDCR SA-94-025,-076; Replacement of Condensate Polshing Syst Powdex Bttrfly Vlvs, Upgrade Cntrl Pnl, Vessel Inlet/Outlet Vlvs

Description of Change(s):

Theses modifications consist of replacement of the butterfly valves found in the Condensate Polishing Powdex System with current technology ball valves. For enhanced protection of the Powdex vessels, both inlet and outlet valves will be equipped with motor operators. The manual control panel in each unit is to be replaced with an upgraded dual Programmable Logic Controller (PLC) and field instrumentation (e.g., limit switches, position indicators, transmitters) will be added or upgraded to interface with the new control system. Power to the MOV's and new PLC will be provided by a new 480 V electrical power distribution system which will be supplied by existing switchgear XB1 to be fed by the reconfigured MCC XB1-4.

Summary of Evaluation:

The replacement of the valves and control panels will enhance the overall effectiveness and reliability of the Condensate Polishing System, reduce the amount of maintenance and minimize the operator interface required for reliable system operation. The implementation of this activity does not affect any equipment important to safety as described by the License Basis Documents, therefore there are no applicable accidents or malfunctions. Based on the results of this evaluation, implementation of this activity does not involve an unreviewed safety question.

Evaluation Number
SE-94-009

Activity Title:

Revision of CPSES Commitment 25084 to Eliminate Recording the LCOAR Number on Each Clearance Tag Associated with the LCOAR

Description of Change(s):

Revise commitment #25084 to eliminate recording the LCOAR number on each associated clearance tag.

Summary of Evaluation:

This safety evaluation allows elimination of the requirement to record the LCOAR number on each associated clearance tag. In-lieu of recording the number on each tag, the LCOAR number will be recorded on the associated clearance report, which is the plant configuration controlling document. This change is consistent with current regulatory requirements and docketed commitments. The addition of compensatory measures in combination with existing plant configuration controls, ensure the plant will not be placed in an unanalyzed condition, introduced to an unreviewed safety question, or reduce its margin of safety.

Evaluation Number
SE-94-010

Activity Title:

LDCR SA-93-096; Updates calculated large break LOCA PCT for Unit 2 and small break LOCA PCT for Unit 1.

Description of Change(s):

This activity only affects the large break Loss Of Coolant Accident (LOCA) analyses of record for CPSES Unit 2 and small break LOCA analyses of record for CPSES Units 1 and 2 as documented in the Final Safety Analysis Report Section 15.6.5.

Summary of Evaluation:

This activity is the result of evaluation / analyses performed by Westinghouse using the NRC-approved Evaluation Models (EM) and updates the calculated Peak Cladding Temperatures (PCTs) for CPSES Unit 2 and replaces the small break LOCA analysis of record for CPSES Unit 1 as documented in the CPSES FSAR.

The calculated PCTs resulting from the implementation of this activity remain well below the limit of 2200 degree F. Based upon the evaluation results, the implementation of the activity does not involve an unreviewed safet question.

Evaluation Number
SE-94-012
Revision 1

Activity Title:

DM 94-012; LDCRs SA-94-062 & SA-94-074; Piping Modifications Required to Resolve Containment Spray System Piping Failures.

Description of Change(s):

High vibration in the Containment Spray piping in the vicinity of the pumps has resulted in some fatigue cracks at connections of small bore piping to headers. A number of approaches are being employed to reinforce the connecting piping or delete the piping connection entirely. The subject of this safety evaluation is the deletion of selected piping connections and the reinforcement of the remaining piping connections. The piping connections being deleted are in the categories of Vent and Drain Connections, Instrument Root Valves and Pump Suction Relief Valves.

Summary of Evaluation:

The criterion applied to determine which pipe connections could be deleted was that the deleted connection would not impact system function, operability, maintenance or testing in such a way as to: adversely impact the CSS in performing its design function; jeopardize the safe operation of the unit; or significantly impact ALARA.

Two of the drain valves deleted based on the above criterion are containment isolation valves which are not required for normal system operation, but are used during leakrate testing of the valves associated containment penetrations. The valves may be deleted without affecting containment integrity, however, the function must be restored prior to the next scheduled containment penetration leakrate test. The remaining drain valves deleted were determined not to significantly affect system operation.

The vent valves deleted were piping high point vents and pump suction casing vents. The high point vents deleted were installed during the unit startup to facilitate system hydro-testing and are no longer required. The CSS pump casing vents that were deleted were reviewed for the potential impact on pump performance due to potential air entrapment. Based on a review of the containment spray actuation time calculations and the MSLB and LOCA analysis calculations, it was concluded that the small quantity of air that may be trapped in the CSS would not significantly impact the system operation. Further, based on a review of the pump manufacturer's operation manual, venting of the pump casing is performed via the casing high point vent, which is not being deleted.

The root valve connections for two local pressure indicators were deleted by tieing into the tube runs for the adjacent transmitters. Four other pressure indicators associated with the eductors from the chemical additive tank were deleted since they are not required for normal system operation. However, since these instruments function to provide local indication during the five year interval surveillance testing, the function must be restored prior to the next scheduled

Evaluation Number
SE-94-012
Revision 1

surveillance test.

The CSS pump suction relief valves were also deleted. While it is common to protect the pump suction lines of high differential pressure pumps from an inadvertent over pressurization during testing or upset condition, it is not specifically required by the ASME Code if it can be shown that the over pressurization condition could only occur during non-design basis operating conditions. The pump suction piping is designed for pressures significantly higher than the maximum pressure resulting from the specified upset condition. Thus, the relief valves are no longer required.

Based upon the results of this evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE-94-014

Activity Title:

LDCR SA-94-026; Revise FSAR section 13.1 to designate the Shift Ops. Mgr. to meet the "Operation Mgr" requirements of ANSI N18.1-1971.

Description of Change(s):

These changes designate the CPSES Shift Operations Manager as the position which meets the ANSI N18.1-1971 functional and experience requirements for "Operations Manager", and eliminates the requirement for the CPSES Operations Manager to maintain an active senior reactor operators license.

Summary of Evaluation:

The elimination of the requirement for the CPSES Operations Manager to maintain an active SRO license will allow the Operations Manager to focus more attention on plant administrative control, thereby enhancing plant oversight and safety. The designation of Shift Operations Manager as the position which must meet the ANSI N18.1-1971 qualification requirements increases the experience requirements for the individual filling the position to ensure the individual responsible for plant day-to-day operations has adequate experience to ensure the plant is operated within its licensing conditions and technical specifications limitations. These changes are consistent with CPSES Supplemental Safety Evaluation Report no.22 (NUREG- 0797). There still exists adequate regulatory, administrative, and plant configuration controls to ensure that no action will be taken to place the plant in an unanalysed condition, introduce an unreviewed safety question, or impact the plant margin of safety.

Evaluation Number
SE-94-015

Activity Title:

DM 93-042,-043;LDCR SA-94-047; Modification of the CCW flow to the RHR & Containment Spray Heat Exchangers Post-LOCA.

Description of Change(s):

This activity implements a modification to throttle the Component Cooling Water (CCW) valves to the Residual Heat Removal (RHR) and Containment Spray System (CSS) heat exchangers to an intermediate position during P-Signal operation. Previous to this modification the CCW valves were full open during P-Signal operation.

Summary of Evaluation:

The CCW heat exchangers have experienced fouling factors that are very restrictive and presented the possibility of forcing a two unit shutdown during extreme summer weather. An evaluation was performed to determine whether the CCW valves to the RHR and CSS heat exchangers could be throttled post-LOCA (P-Signal operation) and thereby allow higher fouling factors. As a result of this modification a new containment pressure-temperature (P/T) analysis was performed for LOCA and MSLB. The safety evaluation of the new containment analysis is addressed in SE-93-121. The results of the safety evaluation indicated that all containment P/T design limits are met and that equipment qualification envelopes for LOCA and MSLB (as revised) are met with the proposed throttled flow and higher fouling factors.

Evaluation Number
SE-94-016

Activity Title:

DCN7760 R0;LDCR SA-94-039;Replce 7A TEC Brkers in MCCS 1EB3-1 & 1EB4-1 with 15A THED Breakers with 25,000A short Circuit Ratings

Description of Change(s):

Existing 7A TEC breakers in MCCs 1EB3-1 and 1EB4-1 for Battery Room Exhaust fans CP1-VAFNID-08 and CP1-VAFNID-10 are being replaced with 15A THED breakers with individual short circuit ratings of 25,000A. FSAR Figure 8.3-9 sheets 2 and 4 which depict the Unit 1 One Line Diagrams are being updated to reflect the new breaker size. This safety evaluation serves to demonstrate that the new circuit breakers will not impact the operation of CP1-VAFNID-08 and CP1-VAFNID-10 or the safety of Unit 1.

Summary of Evaluation:

The replacement circuit breakers will provide adequate short circuit and overload protection for the connected cables and provide short circuit protection for the loads. The loads are protected from overload by the existing thermal overload relays in the MCCs. Based on a review of calculation 16345-EE(B)-008 and the GE time/current curve for a 15A THED circuit breaker, the motor starting current for CP1-VAFNID-08 and CP1-VAFNID-10 is 15A and will not cause nuisance tripping of the breaker during motor starting conditions. Also, THED circuit breakers are rated for 25,000A short circuit current; therefore, the short circuit current ratings of the MCCs are returned to the 25,000A level which is greater than available short circuit current.

Evaluation Number
SE-94-018

Activity Title:

LDCR OD-94-002; Revise the ODCM to Delete the Specific Duration Value (24 hours) for a Containment Purge Batch Release.

Description of Change(s):

The Offsite Dose Calculation Manual (ODCM) is revised to change the specified value of release duration in accounting for radioactive materials released during a containment purge. Currently the ODCM specifies that the first 24 hours of a containment purge is treated as a batch release with the remaining duration of the purge accounted for under the routine continuous plant vent release. This ODCM revision replaces the specified 24-hour batch release period with more general guidance that the initial portion of a containment purge, during which most of the radioactivity is discharged, should be treated as a batch release. This change will allow the specific batch release duration to be established in plant procedures.

Summary of Evaluation:

This change improves the accuracy of recorded values of radioactive materials released in each part of a containment purge release event (batch and continuous parts). There is no change in the general purge release procedure and the total quantity of radioactivity released per purge. The change only affects the administrative procedures for accounting for radioactive materials released and does not directly affect any plant equipment or systems. This change maintains the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a and Appendix I to 10 CFR 50, and does not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations.

Evaluation Number
SE-94-020

Activity Title:

TM 2-94-004 R1; Temporary Barrier At Containment Air Lock

Description of Change(s):

During 2MC01, with fuel removed from vessel, a temporary wooden door shall be erected at the containment air lock. This is done to allow both hatches to remain open while maintaining an envelope capable of maintaining negative pressure in the Unit 2 safeguard building to support Unit 2 operational requirements.

Summary of Evaluation:

This Safety Evaluation considers all aspects previously considered by SE 93-114 in addition to the affects of continued usage of this door during Unit 2 Mode 5 and 6, except during periods of core alteration or irradiated fuel movement in the containment.

Based upon the results of this evaluation, implementation of the proposed activity neither represents an unreviewed safety question, no requires a change to the plant technical specifications.

Evaluation Number
SE-94-021

Activity Title:

DM 93-063,-064;LDCR SA-94-045;Expanding CCW Drain System to Include Other Non-radioactive Drain Sources/ Revision to FSAR Sections 9.2/9.3

Description of Change(s):

Expand the CCW dra system to include other non-radioactive drain sources (DM-93-0 M-93-064 and LDCR SA-94-045)

Summary of Evaluatio

The purpose of this modification is to reduce the normally non-radioactive drain sources which are chemically treated that are routed to the floor drain tanks and are subsequently processed by the radioactive waste system. This is being accomplished by redirecting riser 1 in the safeguards building of both units to each unit's CCW drain tank. There are no credible failure modes identified and no accidents or malfunctions of equipment affected as a result of this modification. Additionally, there are no Technical Specifications which are identified as being associated with this modification.

Evaluation Number
SE-94-022

Activity Title:

DM 92-093; Addition of a Unit 2 Boric Acid Blender Discharge Sampling Point(LDCR 94-46)

Description of Change(s):

DM 92-093, Addiiton of a Unit 2 Boric Acid Blender Discharge Sampling Point

Summary of Evaluation:

Current sampling of the Boric Acid Blender Discharge in Unit 2 is being done via a drain valve not intended for sampling. Installation a sample sink, sampling lines, isolation valves, and a drain line will provide a means for safe and adequate sampling while minimizing the risks associated with spills, leakage and contamination. Refer to Safety Evaluation 92-083 which evaluated this activity and it's implementation in Unit 1 and completely encompasses the current activity and its implementation in Unit 2.

Evaluation Number
SE-94-023

Activity Title:

Design Modification (DM) 94-004; Relocation of the spent fuel storage racks (low density) from spent fuel pool 2 to spent fuel pool 1.

Description of Change(s):

DM 94-004 includes the relocation of the spent fuel storage racks from spent fuel pool 2 to spent fuel pool 1. This is the first phase in the long term effort to provide adequate spent fuel storage capacity for CPSES until a high level radioactive waste depository is established.

Summary of Evaluation:

This activity includes the relocation of the spent fuel racks from spent fuel pool 2 to spent fuel pool 1 to allow the preparation of spent fuel pool 2 for installation of high density storage racks. A Seismic Category II, single-failure-proof lift system will be temporarily installed on the Fuel Handling Bridge Crane rails and utilized to accomplish this rack transfer. Spent fuel is currently stored in Pool 1. This phase of the DM activity does not affect the design or the design basis of the plant because the design basis consists of both spent fuel pool 1 and 2 filled with fuel storage racks and this configuration has been evaluated in the system design. The crane that will be installed for the transfer of the spent fuel racks satisfy NUREG-0554 for single- failure-proof requirements, NUREG-0612 for handling heavy loads at nuclear power plants, ASME NOG-1 for construction of overhead and gantry cranes, and ANSI 14.6 for design of special lifting devices for radioactive materials. The crane is also designed to meet the Quality Assurance requirements of 10CFR50 Appendix B and Regulatory Guide 1.33 revision 2 for critical components, and meets Seismic Category II requirements. As such, the temporary rack handling crane can retain the maximum deign load during a Safe Shutdown Earthquake and remain in place under all postulated seismic loadings. No loads will be transported over racks storing spent fuel. A load drop analysis has been performed and found to be acceptable for this activity. Based on this design approach, the rack relocation activities associated with this phase of the Design Modification will have adequate design safety features to prevent or mitigate the consequences of postulated accidental load drops in accordance with the requirements of NUREG-0612.

Evaluation Number
SE-94-024
Revision 1

Activity Title:

DM-93-082; LDCR FP-94-002; Implementation of Adequate Fire Detection Capability in Unit 2

Description of Change(s):

The design modification enhances the fire detection capability of smoke type fire detectors in specific rooms to support commitments in CPSES letter to the NRC (TXX-93034), to perform corrective actions for specific smoke detector locations. TU Electric committed to perform these activities by the end of the first Unit 2 refueling outage, in response to NRC Inspection Report 50-445;446/92-49, paragraph 5.2. ONE form 93-073 was written to identify the Unit 2 rooms where HVAC airflows could affect the performance of smoke type fire detectors. The design modification optimize the detector locations by adding and/or relocating smoke detectors, so their performance is not potentially degraded by ventilation air-streams associated with HVAC or chiller/ventilation systems in these rooms. The design modification adds a detector in Unit 2 containment spray pump rooms 2-051 and 2-054 as well as in safety injection pump rooms 2-060 and 2-062. Also, existing smoke detectors are relocated in Unit 2 feedwater penetration area rooms 2-100B and 2-100C and in rooms 2-051, 2-054 and 2-062. The Fire Protection Report table that identifies the quantities of smoke detectors by room is affected by this change.

Summary of Evaluation:

The Generic Letter 86-10 analysis included in the safety evaluation concludes that there is no impact to the fire protection program at CPSES other than the presented improvements to the detection system for the affected rooms. Therefore, this modification does not adversely affect the ability to achieve and maintain fire safe shutdown per condition 2G of the Operating License. Based upon the results of this evaluation, implementation of the activity does not involve an unreviewed safety question.

Evaluation Number
SE-94-025

Activity Title:

DMs 93-065,-066;LDCRs SA-94-044,OD-94-007;Replcmnt of Offline Rad.Mont 1/2-RE-5100 with Adjacent-to-line monitors and revise FSAR/ODCM

Description of Change(s):

Debris and oil in the Turbine Building sump has caused Turbine Building drain radiation monitors 1-RE-5100 and 2-RE-5100 to frequently be out of service due to flow switch clogging when obtaining an offline liquid sample from the sumps prior to discharge. This activity involves a plant modification to replace the offline monitors with adjacent-to-line monitors which will eliminate the need for a flow switch, pump, and temperature controller. The new type monitors will eliminate sampling problems but continue to perform the same function (i.e. monitoring liquid effluent releases) by providing continuous monitoring of the process fluid through the process pipe line.

Summary of Evaluation:

These monitors are required for accident monitoring, are used for effluent monitoring, and are considered non-1E and non-seismic. The new monitors are equivalent to the existing monitors in that both perform the same function albeit by different method. Credible potential failure modes would be failure of the new mounting for the detector/shield assembly or the introduction of the detector/shield assembly as missiles during a seismic event or tornado; however, a seismic event or tornado that may cause the detector/shield assembly to break away from the pipe, become a missile, or affect the Turbine Building drain line or other nearby components, systems or structures in the Turbine Building would not affect plant safety since these components, systems and Turbine Building structure have been classified as non-safety and non-seismic. Due to their remote location from plant safety-related strutures, systems and components, failure of the mounting for the new detector/shield assembly would not affect the ability of the plants' safety-related equipment to perform its safety-related functions.

Evaluation Number
SE-94-026

Activity Title:

Revision to NUC-203, "Incore Excore Detector Calibration", for Implementation of a Uniform Approach to Intercept Current Alignment

Description of Change(s):

TU Electric developed a methodology to support the calibration of the power range nuclear instrumentation system using the results from a single incore flux map. A Calibration Standard was developed based upon previously obtained multipoint incore/excore calibrations. This normalized relationship describes the change in the calibration current of the excore detectors as function of axial offset, and is proven to be independent of unit, cycle and burnup.

Summary of Evaluation:

The use of a uniform approach to intercept current alignment does not change the margins of safety provided by the current Technical Specifications. Based on this evaluation, the implementation of the activity does not involve an unreviewed safety question.

Evaluation Number
SE-94-027

Activity Title:

LDCR EP-94-001 & Rev. 9 to EPP-201; Revise CPSES Emergency Plan and EPP-201 for Initiating Conditions (ICs)/Emergency Action Levels (EALs)

Description of Change(s):

This activity involves a revision to the CPSES Emergency Plan (Revision 19) and Emergency Plan Procedure EPP-201, "Assessment of Emergency Action Levels, Classification and Plan Activation" (Revision 9). The CPSES Emergency Plan, Section 2.0, "Emergency Classification System" was revised to delete specific details of Initiating Conditions(ICs)/Emergency Action Levels(EALs) and reflect only general event categories based on guidance of both NUREG-0654 and NUMARC/NESP-007; specific details of the ICs/EALs are referenced to the EPPs. EPP-201 was revised to update organizational titles, delete unnecessary information and delete or modify ICs/EALs including deletion of several ICs/EALs for declaring a Notification of Unusual Event (NOUE) and downgrading the emergency classification of selected other ICs/EALs (e.g., weather related hazards).

Summary of Evaluation:

The CPSES Emergency Classifcation System was revised to incorporate changes based in part on new guidance provided by NUMARC/NESP-007 (Rev 2), NRC Memorandum from R.L. Emch, NRR, to NRC Regional Offices dated July 11, 1994, "Branch Postion on Acceptable Deviations to Appendix 1 to NUREG-0654/FEMA-1" and TU Electric letter to NRC logged TXX-94230 "Request for NRC Review and Approval of Revisions to CPSES Emergency Classification System" dated September 1, 1994. NRC acceptance of the proposed changes was obtained by letter from NRC Region IV dated October 6, 1994. Information in the CPSES Emergency Plan (Section 2.0 and Table 2.2) and EPP-201 was revised per the above documents.

These changes are procedural in nature and eliminate emergency classifcation of some events which have been determined to be either inside the operating safety envelope for CPSES as defined by the Technical Specifications, including Limiting Conditions of Operation and associated Action Statement times, or events not recoginized as precursors of more serious events which have potential for impact on plant safety related structures and systems. Events external to CPSES operations (e.g., weather related hazards) are limited in level of declared emergency classification when based solely on the external event itself without regard to affects on plant equipment and systems.

The revised ICs/EALs are based on the most recent industry and NRC guidance documents for ICs/EALs and were approved by NRC prior to being made effective. These changes do not involve any plant primary or secondsary system equipment, change the CPSES Technical Specifications or impact on plant safety or existing accident analyses described in licensing basis documents.

Evaluation Number
SE-94-028

Activity Title:

DM 93-051;LDCR SA-94-057; Upsize of Instrument Air Compressor Motor and Update the FSAR Blackout Diesel Generator Blackout Loading Table

Description of Change(s):

This modification replaces existing 100HP instrument air compressor CP1-CICACO-02 motor with a 125HP motor and the Licensing Document Change Request (LDCR) updates the Final Safety Analysis Report (FSAR) Table 8.3-2 for the Blackout Diesel Generator Loading, to account for the increased load on the diesel generator.

Summary of Evaluation:

Due to increased loads on the instrument air system, compressor CP1-CICACO-02 routinely ran at the end of its 1.25 service factor for extended periods. This condition caused numerous circuit trips due to the motor overload. Replacing with a larger motor alleviate this condition while ensuring reliable operation of the system. Replacement of the 100HP instrument air compressor motor with a 125HP motor, meet the instrument air system requirements and provide for more efficient compressor operation.

The instantaneous trip setting of the compressor feeder breaker in MCC 1EB4-1 has been increased to account for the increased inrush current during motor starting and the new setting's coordination with upstream protective devices verified. The overload relay has been changed and heaters sized in accordance with DBD-EE-051 to protect the compressor motor and feeder cable against overloads. Compressor feeder cable size acceptability for ampacity and voltage drop was also verified. 480V switchgear 1EB4 to MCC 1EB4-1 feeder cable size and breaker settings were also verified as acceptable for the increased load.

The increase in compressor horsepower translates to a negligible increase in blackout loading on diesel generator CP1-MEDGEE-02 at the 90 second step. The increased inrush load on the diesel generator at the 90 second load step is still well below the largest diesel generator factory test report calculated values. In addition, this increase results in total diesel generator load less than the 6300KW limit.

As discussed, based on this evaluation, implementation of these activities do not involve an unreviewed safety question.

Evaluation Number
SE-94-030

Activity Title:

LDCR SA-94-059; Revision to FSAR To Describe Actual Process of Performing Reactor Coolant Pump Underfrequency Relay Testing

Description of Change(s):

This activity is a revision to section 7.2 of the FSAR to describe theactual process for performing testing of the Reactor Coolant Pumps underfrequency relay and associated time delay relay.

Summary of Evaluation:

Currently the FSAR states that there are no bypass to test situations for trips sharing a two-out-of-four logic and that the interruption of a monitored signal will annunciate in the Main Control Room (MCR).

The test circuitary has no provisions for inserting a trip for this parameter alone into the Solid State Protection System (SSPS). The test circuitary also has no provisions for annunciating in the MCR when the channel is placed in test. The purpose of the UF trip is to protect agasinst a loss of reactor coolant flow due to grid underfrequency.

Neither changing the FSAR to reflect actual test conditions nor performing the actual UF test under the as-built conditions will increase the probability of having a low grid under-frequency and an accompanying loss of flow. The test of the UF trip provides a test frequency which causes the UF relay and the time-delay relay to set and reset. The test does not increase the probability of a malfunction of equipment important to safety.

Testing of the UF trip under the as-built conditions does not create or change the probability for an accident or malfunction previouusly evaluated.

Testing of the UF relays under the as-built conditions and changing the FSAR to reflect those conditions will not change the margin of safety for any technical specification. The requirement of Technical Specification 3/4.3.1 is three operable channels. With one channel temporarily in test, the other three channels will be monitoring grid frequency.

The proposed change does not involve an unreviewed safety question.

Evaluation Number
SE-94-031

Activity Title:

MM 93-481,-482;LDCR SA-94-058;Replcmnt of Obslte Contnmnt Dew Pt.Measurmnt Snsrs 1-ME-5460-3 & 2-ME-5460-1 and FSAR Revsn to Rsolv Dscrpncy

Description of Change(s):

MM 93-482 replaces obsolete containment dew point sensor 2-ME-5460-1 and monitor 2-MQY-5460-1. MM 93-481 replaces 1-ME-5460-3 and monitor 1-MQY-5460-3. Revises the FSAR to show the appropriate detail associated with the dewpoint sensors. Unit 2 modification is complete. Unit 1 is scheduled for refueling outage 1RF04.

Summary of Evaluation:

The Minor Modification Activity implements design changes required to support replacement of an obsolete unit 2 containment dew point sensor. The replacement equipment meets or exceeds the required specifications for the dew point sensors. The activity revises the FSAR to reflect the appropriate level of detail required for the dew point sensor. The implementation of this activity does not affect any equipment important to safety as described in the License Basis Documents, therefore there are no applicable accidents or malfunctions. Based on the results of this evaluation, implementation of this activity does not involve an unreviewed safety question.

Evaluation Number
SE-94-034

Activity Title:

DM 94-028,LDCR SA-94-093;Change of Gear Ratio for RHR mini-flow valves FCV-610 & 611 and to increase the allowable valve stroke time

Description of Change(s):

This activity analyzes an increase in the stroke time for the RHR mini-flow valves, 2-FCV-610 and 2-FCV-611 from 10 seconds to 15 seconds, to allow for a revision to the gear ratio of their valve actuators.

Summary of Evaluation:

This activity only impacts the Large Break LOCA analysis section of the FSAR. An evaluation of the Unit 2 Large Break LOCA was performed, and all event acceptance criteria were shown to be satisfied with the 15 second valve stroke time.

Evaluation Number
SE-94-042

Activity Title:

LDCR SA-93-132; Revision to FSAR Section 10.2 to Identify Alternate Test Method for Turbine Overspeed Trip Test

Description of Change(s):

This evaluation reviews the use of alternative testing of the turbine mechanical overspeed trip mechanism bolts.

Summary of Evaluation:

The evaluation reviews the impact of removing the stub shaft from the high pressure turbine and testing it on a test stand to verify the trip bolts function to dump trip fluid pressure. Once the test is completed satisfactorily, the shaft is then reassembled and no further actual mechanical overspeed tests need be conducted. The automatic turbine tester will continue to test the overspeed trip assembly every two weeks as required by technical specification 3/4.3.4. The evaluation concluded that the activity does not present an unreviewed safety question, a new accident not previously evaluated or reduction in the margin of safety.

Evaluation Number
SE-94-044

Activity Title:

TM-2-94-007; Blind Flange for Relief Valve 2-8853B while performing valve repair.

Description of Change(s):

Temporary Modification (TM) 2-94-007 was performed to remove leaking relief valve 2-8853B. The connection was blind flanged while the subject valve was repaired. The inlet and outlet connections were blind flanged to prevent system leakage and allow required system operation. Except for limited periods, valves 2-8802B and 2-8821B, which are in the piping protected from overpressure by relief valve 2-8853B, can not both be closed, thus preventing overpressure for the associated piping.

Summary of Evaluation:

The TM removes relief valve 2-8853B from the SI Train B hot leg injection pathway. The relief valve is provided for overpressure protection of the associated lines during the long-term recirculation phase of operation. In accordance with the TM the subject relief valve is to be removed and the associated piping blind flanged while the valve is repaired or a new valve is procured. A review of the system design and operation indicates that the overpressure protection explicitly provided by 2-8853B is not required because the associated piping is not isolated from other pressure relieving devices. The only way such isolation could occur would be as the result of multiple procedural violations of EOS1.4. The TM explicitly noted that valves 2-8802B and 2-8821B can not both be closed for extended periods (greater than 2 minutes) without defeating the overpressure protection requirements. No new failure mechanisms were identified.

The relief capabilities of the valve were temporarily not needed while relief valve repair was undertaken, due to the combination of the system design characteristics and the operational practices.

Evaluation Number
SE-94-047

Activity Title:

TM-1-94-0002; Providing of Alternate Source of Compressed Air for the Instrument Air Systm when the Air Compressor CP1-CICACO-02 is Reworked

Description of Change(s):

The Temporary Modification provides an alternate as well as redundant source of compressed air to the instrument air system while the instrument air compressor CP1-CICACO-02 is being reworked. A temporary air compressor is connected to the instrument air system, but it is isolated by a manually operated isolation valve.

Summary of Evaluation:

If failure of instrument air compressor CPX-CICACO-02 occurs while CP1-CICACO-02 is down for repair, the temporary air compressor will provide an alternate source of compressed air. The equipment will provide equipment redundancy to the instrument air system. The instrument air system serves no safety function and it is not required to achieve safe shutdown or to mitigate the consequences of a design based accident. Safety related accumulators are provided to maintain control of selected equipment during a design basis accident. The accumulators will not be affected by this activity. The evaluation concludes that the activity does not present an unreviewed safety question, new accident not previously evaluated or reduction in the margin of safety.

Evaluation Number
SE-94-049

Activity Title:

LDCR SA-94-086;Revision to FSAR Sec. 9.5.1.6.2 to Reflect Acceptable Alternate to IEEE 383-1974 Flame Retardant Test for Non-Class 1E Cable

Description of Change(s):

This change adds acceptable alternate to IEEE 383-1974 flame retardant test for non-Class 1E cables and identifies that fire test standards IEEE 1202, UL 1581, UL 1685, ICES T-29-520, CSA FT-4 and UL 1666 have been evaluated to be equal or better than the IEEE 383-1974 test for fire retardant characteristics of electrical cables.

Summary of Evaluation:

Although IEEE-383-1974 contains a vertical cable tray fire test, the standard is often viewed by the industry as solely nuclear safety related. As a result of this perception it is often difficult to obtain cables for non-safety related applications, cables tested to the flame test requirements of IEEE 383-1974. However there are several industry fire tests that have been developed to endorse a test methodology similar to IEEE 383-1974, make improvements to this standard or demonstrate flame retardant characteristics for cabling installed in environments susceptible to a fire more severe than simulated by this standard. This change allows the use of a selected set of fire tests considered equivalent to or better than IEEE 383-1974, to demonstrate flame retardancy for non-Class 1E cables.

This activity has no impact on the Fire Safe Shutdown Analysis (FSSA) or the Combustible Loading Analysis, since all cables located within areas governed by Regulatory Guide 1.75 and Appendix A, remain flame retardant or be contained within a non-combustible covering. This change does not adversely affect the ability to achieve and maintain safe shutdown of the plant under fire conditions.

Evaluation Number
SE-94-050

Activity Title:

DM 94-020 R0,LDCR SA-94-094; Re-installation of Vent/Drain Valves and and addition of instrument flex hoses in the Containment Spray System

Description of Change(s):

This activity involves the re-istallation of select vent and drain valves, and the installation of two instrument flex hose connectors in the Unit 2 Containment Spray System (CSS). The valves were originally removed as part of a modification to correct vibration cracking in the CSS. The valves were determined to be necessary for optimum venting and draining of the CSS. To reduce the potential for vibration cracking, a vibration resistant valve design was used.

Summary of Evaluation:

The installation of the new vent/drain valve design and the flex connectors was evaluated to determine its impact on the ability of the Containment Spray System to perform its safety function and to maintain containment integrity following a postulated LOCA or MSLB. It was conclude that the modifications would not have an adverse impact on the system safety function or on containment integrity.

Evaluation Number
SE-94-052

Activity Title:

DM 93-046; Hardware Changes Supporting Deletion of the Boron Dilution Mitigation System from the CPSES Technical Specifications for Unit 2

Description of Change(s):

This activity implements design changes required to support the revised safety analysis for the inadvertent boron dilution event in Modes 3, 4 and 5. The modifications listed below were previously described in a License Amendment Request (TXX-93098 dated April 30, 1993) which was subsequently approved in License Amendment 20 for Unit 1 and License Amendment 6 for Unit 2. The modifications described are for Unit 2. The Unit 1 modifications were included in the 1993 annual summary. The modifications include:

1. An alarm added to the Volume Control Tank (VCT) to alarm at 70% span.

2. An alarm added to annunciate when the VCT divert valve is not in the "VCT" position.

3. An alarm added to annunciate when the VCT divert valve is diverting flow to the Hold-up Tank.

Summary of Evaluation:

The safety impact of the installed boron dilution alarms was discussed in the license amendment submittal in which it was concluded that there was no adverse safety impact. Installation of the equipment was also found not to have safety impact.

Evaluation Number
SE-94-054

Activity Title:

LDCR OD-94-005; Revise Offsite Dose Calculation Manual(ODCM),Tab.3.3-7 Action Statement 32, Allow Unit 2 CCW Monitors to Backup SSW Monitors

Description of Change(s):

This activity involves a revision to the ODCM, Table 3.3-7, "Radioactive Liquid Effluent Monitoring Instrumentation", Action Statement 32a to allow use of the Unit 2 Component Cooling Water (CCW) monitors to backup Station Service Water (SSW) monitors. This revision corrects the ODCM Action Statement to allow additional system monitoring flexibilty and, in some operational circumstances, preclude performing manual grab sampling.

Summary of Evaluation:

The revision procedurally changes ODCM Table 3.3-7 Action Statement 32a for the SSW effluent lines. The revision changes the Action Statement wording to replace designators for CCW monitors (1RE-4509, 1RE-4510 and 1RE-4511) with the letter designation "u", e.g., (uRE-4509, uRE-4510 and uRE-4511). The "u" designation signifies either unit and thus allows Unit 2 CCW monitors to be used as a backup to the SSW monitors (in addition to Unit 1 CCW monitors). The Unit 1 CCW monitors were already being used correctly; this change allows the extra operational flexibility of using the Unit 2 CCW monitors as necessary to backup SSW monitors if the SSW monitors become inoperable and system lineups are such that they support the operation of the units. The fuctions of the Unit 2 CCW monitors are the same as the Unit 1 CCW monitors.

This change is procedural in nature. There are no changes in existing plant design. The change only impacts the ODCM program for monitoring the potential release pathway of radioactive materials from the CCW system. The change does not affect the actual concentrations of radioactive materials and therefore does not create the possibility of an accident different from any accident evaluated in the licensing basis documents. The ODCM, as revised, will continue to provide assurance that the acceptance limits for radioactive effluent release are achieved. This change does not change or modify the acceptance limits; implementation of the change does not impact the margin of safety.

Evaluation Number
SE-94-056

Activity Title:

LDCR SA-94-099;Deletion of Unnecessary details such as Cable/MCC Compartment Nos,Motor Heater/Starter Sizes and Load Data from FSAR Figures

Description of Change(s):

FSAR figures 8.3-9 sheets 1,2,3&4; 8.3-10 sheets 1&2; 8.3-11 sheets 1,2&3; 8.3-12 sheets 1&2; 8.3-14 sheets 1&2; 8.3-14A sheets 1&2; 8.3-15 sheets 1,2&3; 8.3-15A sheets 1&2; 8.3-15B sheets 1&2 and 8.3-15C, are revised to delete unnecessary details such as cable number, MCC compartment number, motor space heaters, motor starter size, drawing references, cable splices, junction boxes, fuse type and load data from the 480V MCC, 118/120V AC, 24/48V DC and 125/250V DC figures. Additionally, notes in these figures are rearranged and irrelevant notes/information are deleted.

Summary of Evaluation:

Deletion of the above information from the FSAR figures, does not compromise the intent of Regulatory Guide 1.7 Rev. 3 and the NRC Standard Review Plan. The information which is being deleted from the FSAR figures, is available on Project Electrical Drawings. Also since the above activity is a paper change activity and does not require that any change be made to plant structure, system or component, its implementation does not involve an unreviewed safety question.

Evaluation Number
SE-94-063

Activity Title:

LDCRs SA-94-112,-130; Revision of FSAR Diesel Generator Loading Tables 8.3-1A,1B and 2/Update of Design Basis for Primary Plant Ventilation

Description of Change(s):

FSAR diesel generator load tables are being revised as a result of new diesel generator loading calculations EE-CA-0007-3378, Rev. 0, EE-CA-0007-3377, Rev. 0 and EE-CA-00073376, Rev. 0. Also involves a change to the diesel generator loading basis for Primary Plant Ventilation (PPV) fans during loss of offsite power. FSAR section and the design basis for Primary Plant Ventilation (PPV) is being revised to reflect that the ESF filtration Heaters are not loaded on the diesel generators during a loss of offsite power.

Summary of Evaluation:

This activity is an administrative task which updated the calculations to incorporate outstanding CCNs using a PC based computer software program. The resulting diesel generator load is within the existing technical specification limit of 6300 kw.The revision of these calculations is not the result of any design modification activities and does not result in a change to the design basis of CPSES. This activity has no effect on any systems, components or structure. Therefore this activity has no effect on any accidents or malfunctions described in the Licensing Basis Documents.

The PPV ESF heaters are not required to perform any required functions during a loss of offsite power. There is no effect on any accidents or malfunctions described in the Licensing Basis Documents. The PPV function during a loss of offsite power is satisfied by Non-ESF Filtration units. There is no unreviewed safety question associated with this activity.

Evaluation Number
SE-94-065

Activity Title:

DM 94-028,LDCR SA-94-093;Change of Gear Ratio for RHR mini-flow valves FCV-610 & 611 and increase the allowable valve stroke time

Description of Change(s):

This activity analyzes an increase in the stroke time for the Unit 1 RHR mini-flow valves, 1-FCV-610 and 1-FCV-611 from 10 seconds to 15 seconds, to allow for a revision to the gear ratio of their valve actuators.

Summary of Evaluation:

This activity only impacts the Large Break LOCA analysis section of the FSAR. An evaluation of the Unit 1 Large Break LOCA was performed, and all event acceptance criteria were shown to be satisfied with the 15 second valve stroke time.

Evaluation Number
SE-94-066
Revision 1

Activity Title:

MM 94-292, LDCR SA-94-113; Modification to Increase Fuel Bridge Crane Speed and Update of FSAR Section to Reflect the Associated Changes

Description of Change(s):

The existing 2 bridge drive assemblies each consisting of a 2 speed motor, a fluid coupling, a brake and right angle gear box were replaced with 2 new drive assemblies consisting of a single speed motor, a brake and right angle drive. The fluid couplings were eliminated. These units are mechanically directly interchangeable with the existing units. An enhancement to the bridge is a variable frequency invertor with dynamic braking and the associated electrical wiring and control circuits which provide variable speed control to the bridge. The completed installation provided the capability to drive the bridge at speeds up to 33 feet per minute. Before the modification, 33 feet per minute was the maximum safe allowable speed for the bridge itself; the speed is electrically limited to this speed. Since the trolley hoist can be moved in a direction perpendicular to the bridge movement, using the hand driven chain mechanism, the maximum speed of the fuel assembly is slightly higher than 33 feet per minute.

Summary of Evaluation:

The addition of the control boxes to the bridge resulted in stresses well below the allowable and the addition of the control boxes is therefore acceptable.

This increase in speed did not introduce any significant increase in forces experienced by fuel assemblies during handling operations which could lead to fuel drop accidents. In addition, the severity of any such accident is bounded by accident conditions described in the FSAR. The design basis fuel handling accident is defined as the dropping of a spent fuel assembly in the containment building or spent fuel storage area floor resulting in the rupture of the cladding of all fuel rods in the assembly, which remains limiting regardless of bridge speed.

It has been verified that any additional forces on the fuel assemblies and the bridge crane due to the increased crane speed are within acceptable limits.

Evaluation Number
SE-94-067

Activity Title:

TM 1-94-004;Installation of Wooden Framed Cover Bolted to Security Grating Material on the Penthouse Discharrge End of the Exhaust System

Description of Change(s):

This TM is required in order to allow maintenance personnel to perform the damper inspection/repair on gravity dampers in the control room exhaust system while ensuring the control room can be maintained positive. TM-1-94-004 installs a wooden framed cover bolted to the security grating material on the penthouse discharge end of the exhaust system. The TM will be suitable to retain the expected maximum control room positive pressure of 0.50 inches of water (two pressurization units on at the same time) and satisfy plant design requirements for the various postulated accident scenarios.

Summary of Evaluation:

This TM will permit the inspection of the dampers and will not have an adverse effect on the Control Room HVAC system or its ability to protect the operators in a radiological accident. No other plant system or structure will be affected by the installation of this Temporary Modification.

The safety evaluation looked at the various possible failure modes. In each case the design basis of the plant is maintained. The plant is not designed for radiological accidents coincident with either a tornado, a seismic event or winds in excess of 80 miles per hour. In each of these events the Temp Mod can fail, but there is no impact on safety because no release of radiation is postulated to occur at the same time. The temp mod is designed to hold the maximum expected positive pressure expected to be generated by the control room emergency pressurization system during the designed base LOCA and fuel handling accident. Operator safety is maintained.

Evaluation Number
SE-94-068

Activity Title:

LDCR SA-93-150;FSAR Section 9.1/DBD Revisions to Reflect the As-built Spent Fuel Pool Cooling & Cleanup System/Current Refueling Practices

Description of Change(s):

LDCR-SA-93-150 updates Section 1.2, 1.3 and 9.1 to reflect the as-built spent fuel storage capability and to clarify the description of the plant. Section 9.1.3 is also revised to reflect new design bases for 18 month fuel cycles and refueling procedures and practices. The changes document the design bases for spent fuel pool cooling and cleanup which satisfy ANSI N18.2 and ANSI N210 and the intent of SRP 9.1.3. Tables 9.1-1 and 9.1-4 are updated to show the projected spent fuel storage through 1RF04 noting that the emergency full core offload (ECO) reserve will not be available in spent fuel pool no. 1 during or after 1RF04. LDCR SA-94-091 updates section II.B.2 to reflect the design and licensing bases and to document the as-built status, including calculations recently performed to implement the design bases.

Summary of Evaluation:

The updates show that CPSES satisfies the licensing basis in the SER 9.1.3 and SSER 22 which establish the NRC acceptance of the cooling and cleanup system based on SRP 9.1.3. The evaluation concludes that the spent fuel pool cooling and cleanup system is in conformance with BTP ASB 9-2 and satisfies the acceptance criteria of SRP 9.1.3 except that the 140 F pool limit is applied at the supply to the demineralizers downstream of the heat exchanger and the normal maximum pool temperature is 150 F. This deviation from the SRP is acceptable based on the NRC Safety Evaluation for Crystal River Unit 3 dated April 16, 1991, which states that the SRP "guideline specifies this temperature in order to protect demineralizer resins in spent fuel pool cleanup systems which may be affected by temperatures in excess of 140 F." Thus, it is concluded that the intent of the SRP and the acceptance criteria in the SER are satisfied and no unreviewed safety question is created by these changes to the FSAR and DBDs.

An ECO is an extremely unlikely event. The ECO provisions in the spent fuel storage capacity emanate from ANSI N18.2 and ANSI N210 and are not based on 10CFR50 requirements. Furthermore, there is no regulatory commitments that prevent the use of the design capacity for normal storage. Although the ECO capacity will not be available in spent fuel pool no. 1, the capability for an ECO will be retained by the following provisions:

1) If Unit 2 required an ECO during 1RF04 core offload, the Unit 1 fuel could be replaced in the reactor to make room for Unit 2 fuel. After the core is reloaded, pool no. 1 will have room for at least 165 spent fuel assemblies from either unit should the need arise.

Evaluation Number
SE-94-068

2) ECO storage for 28 spent fuel assemblies exist outside pool no. 1. Twenty five storage locations in each containment are available for an emergency.

3) A high density 12 x 14 rack could be set in the wet cask pit by the Fuel Building Overhead Crane and used to store 42 spent fuel assemblies (1 out of 4 checkerboard configuration) in an emergency under the provisions of 10CFR50.59.

Evaluation Number
SE-94-069

Activity Title:

Cycle 2 Core Configuration for CPSES Unit 2

Description of Change(s):

For the CPSES Unit 2, Cycle 2 core configuration, 88 Westinghouse Optimized Fuel Assemblies (OFA) replaced 65 Region 1 and 23 Region 2 fuel assemblies which were used in the Cycle 1 configuration.

Summary of Evaluation:

This core configuration has been evaluated for mechanical and thermal-hydraulic characteristics by Westinghouse. All applicable design criteria were determined to be satisfied. The neutronic characteristics of the Cycle 2 core configuration have been evaluated for their effect on accident analyses. In all cases, it was determined that the applicable event acceptance criteria were satisfied. Because all mechanical design criteria continue to be satisfied, there is no reduction in any failure point introduced by the Cycle 2 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. It is concluded that the Cycle 2 core configuration does not reduce any margin of safety as defined by the plant technical specifications; and therefore, the changes do not involve an unreviewed safety question.

Evaluation Number
SE-94-071

Activity Title:

MM 94-324,94-325; LDCR SA-94-121; Use of Soft Run Cables in the Containment for Reactor Coolant Pump Motor Stator Temp. Monitoring

Description of Change(s):

Minor modifications pertain to installation of soft run cables (i.e., cables not installed in raceway) from the Reactor Coolant Pump (RCP) rooms to outside the containment bioshield wall, to provide additional capability to monitor RCP motor stator temperature during plant operation. FSAR Section 8.3 is also updated to reflect the use of soft run cables for the data acquisition from RCP stator temperature sensors.

Summary of Evaluation:

The soft run cable routing from the Reactor Coolant Pump (RCP) motor stator to outside the bioshield wall in CPSES units 1 & 2 provides alternate stator temperature monitoring capabilities. Connection of the cables outside the bioshield wall is required only during data acquisition activities. The cables carry low level signals and are qualified to IEEE-383 and installed to seismic category II requirements. Also the cables meet Regulatory Guide 1.75 separation requirements. The potential failure modes of the soft run cables are the same as of those of enclosed raceways.

The cable installation in this manner for the above application does not adversely impact operation of the units and there is no unreviewed safety question associated with these activities.

Evaluation Number
SE-94-072

Activity Title:

MM 93-326;LDCR SA-94-119;Replacmnt of MR-4 Meter Relays with Sentry-34 Relays for Unit 2 Generator,and Main, Auxiliary & Startup Transformers

Description of Change(s):

The activities involve replacement of existing model MR-4 type relays 77/2G, 77/2MT, 77/2UT and 77/2ST with Sentry-34 type relays 77-1/2G,77-2/2G; 77-1/2MT,77-2/2MT; 77-1/2UT,77-2/2UT; 77-1/2ST,77-2/2ST (2 for each MR-4 type one for spare output) for the Unit 2 power metering circuits for the Unit 2 generator and the main, auxiliary & startup transformers. FSAR Figure 8.3-2 sheets 1 and 3 are also revised to reflect the modification to the relay type for Unit 2.

Summary of Evaluation:

The existing model MR-4 type relays are causing problems resulting in frequent repairs and replacements and also do not provide output compatible for Unit 2 computer. These relays are replaced by the Sentry-34 type relays to reduce the repairs/replacements and to provide additional spare outputs compatible to the Unit 2 plant computer. These relays used for power metering for the Unit 2 generator and main, auxiliary & startup transformer are non-safety relays and do not associate or affect the safety-related systems or components.

Evaluation Number
SE-94-074

Activity Title:

Increase of Doppler Reactivity Coefficient Uncertainty Factor from 10% to 15%.

Description of Change(s):

This activity involves increasing the Doppler (fuel temperature) reactivity coefficient uncertainty factor for safety analysis applications from 10% to 15% in association with the transition to the use of the CASMO-3 Version 4.7 computer code for nuclear cross-section calculations. An uncertainty factor is applied to ensure appropriate conservatism when the calculated Doppler coefficient is used in the safety analyses. CASMO-3 Version 4.7 includes a change in the U-238 cross-section library which affects fuel temperature reactivity effects. The change was incorporated by the code supplier to remove some observed inconsistencies in the data which resulted in non-linear behavior of Doppler coefficients versus temperature. As a result of this change, the calculated Doppler fuel temperature coefficients are affected.

Summary of Evaluation:

The Doppler coefficient uncertainty factor is determined, in accordance with NRC-approved methodology, by statistical comparisons of calculations with both experimental data and independent calculations with alternate computer codes. The current NRC-approved Doppler uncertainty factor, which is based on evaluations for earlier CASMO-3 code versions, is 10%. For CASMO-3 Version 4.7, the Doppler coefficient uncertainty factor was evaluated using the current NRC-approved methodology. The results of the uncertainty analysis indicate that a 15% Doppler uncertainty factor will maintain the NRC-approved level of conservatism. This change does not affect the uncertainty evaluation for previous versions of the CASMO-3 code. A 10% Doppler uncertainty factor remains valid for the earlier code versions.

The use of a 15 % Doppler uncertainty factor with CASMO-3 Version 4.7 ensures that the required level of conservatism is maintained in the safety analyses, consistent with NRC approved methodology. The Doppler uncertainty factor is an analytical allowance that is applied to maintain the required level of conservatism, but does not in itself affect the probability of occurrence or consequences of any accident or malfunction, or reduce the margin of safety. Therefore, the proposed change does not involve an unreviewed safety question.

Evaluation Number
SE-94-076

Activity Title:

MM 94-372; Removal of Diaphragm from Boric Acid Tank CPX-CSATBA-02

Description of Change(s):

This activity will remove the diaphragm from Boric Acid Tank CPX-CSATBA-02. The diaphragm has deteriorated and must be removed. Since the diaphragm serves no function, it will be removed permanently.

Summary of Evaluation:

Removal of the diaphragm from the boric acid tank will not impact the safety of plants structures, systems, and components because the boric acid solution stored in the tank contains equilibrium levels of dissolved oxygen due to the solution's preparation process. Oxygen concentration will not increase if the diaphragm is removed. Operating experience has confirmed that the use of this boric acid solution in the RCS has not resulted in the RCS dissolved oxygen concentration to be greater than 2 parts per billion, well below the Technical Specification limit of less than 0.1 part per million. Since the activity has no effect on the systems, there are no failure modes created by it.

Evaluation Number
SE-94-078

Activity Title:

LDCR FP-94-001;Revision to FPR Sectns. II and V App. C Deviation 1b to Change Criteria of Thermolag Enclosed Non-essential Cables in Rm 115A

Description of Change(s):

This activity involves revision to the Deviation 1b in the Fire Protection Report. The deviation describes electrical cables in room 115A, including cables that are non-essential to Fire Safe Shutdown (FSSD), as "enveloped in a one-hour rated barrier system." Thermo-Lag enclosures protecting non-essential cables in 115A are not upgraded to current one-hour rated configurations. The Licensing Document Change Request (FPR) revises the description of these enclosures from "one-hour rated barrier system" to "fire resistive material," and provides distinction between the subject enclosures and the up-graded, properly qualified one-hour enclosures protecting essential cables in the room.

Summary of Evaluation:

Non-essential electrical cables span across the two partial fire-barrier walls which separates redundant safety-related chiller units and their respective circulation pumps in room 115A. The purpose of enclosing the non-essential cables in Thermo-Lag is to ensure that the cables do not provide a potential flame propagation path across the partial fire barrier walls separating redundant safe shutdown equipment, thereby, defeating the purpose of the walls. Therefore, these Thermo-Lag enclosures actually serve as fire stops, and there is no specific or implied requirement to enclose the non-essential cables in a rated fire barrier system. Testing has shown Thermo-Lag material to be combustible when exposed to a sufficient external heat source such as fire. However, the material will not support combustion in the absence of external heat, and its flame propagation properties are low enough to prevent flame propagation across the partial barriers. Therefore, this change does not degrade the existing protection in room 115A and does not adversely affect the ability to safely shut down the plant during fire conditions.

Evaluation Number
SE-94-080

Activity Title:

MM 94-376;LDCR SA-94-140; Removal of Remote Operators from CVCS Valves 2CS-8387A&B and Revision to FSAR Section 9.3 to incorporate the change

Description of Change(s):

This activity will remove the remote operators for alternate seal injection isolation valves 2CS-8387A and 2CS-8387B.

Summary of Evaluation:

The valves in question are the train A and train B alternate seal injection isolation valves. The alternate seal injection line was provided as an alternative means of providing seal flow to the reactor coolant pumps in the event that 2-FCV-121 was unavailable. However, per FSAR Table 9.3-9 (Sheet 27), that scenario will be mitigated by using the positive displacement charging pump to control charging and seal injection flow. No where in any plant design document, licensing basis document, or procedure is credit taken for the availability or use of the alternate seal injection line.

The valves in question were recently changed from first generation Rockwell Edwards valves to third generation Rockwell Edwards valves. Upon completion of installation of the new valves, the remote operators could not be reattached without significant modification. Since the alternate seal injection line is superfluous and the valves will never be operated, the remote operators are not needed and will be removed permanently.

Evaluation Number
SE-94-082

Activity Title:

LDCR SA-93-098;FSAR Section 6.2 Update to Replace the Licensing Basis for Containment Pressure/Temperature Analysis of Record/Rev. of Cmtmnt

Description of Change(s):

The replacement of the licensing basis containment Pressure/Temperature analyses with TU Electric's in-house analyses (SE-93-121), in CPSES Units 1 and 2 FSAR Chapter 6.2, necessitated the revision of information which were based on the previous analyses. This safety evaluation supplements SE-93-121. The information, along with the related FSAR section numbers which are affected by this activity are as follows:

1. FSAR Table 6.2.1-5 - analysis initial conditions (revised minimum usable volume for the Refueling Water Storage Tank).

2. FSAR Section 6.2.1.1.3.7 9414) - use of natural convection heat transfer coefficient, thermal conductances for interface between steel liner and concrete, and painted structural heat sinks used in the analysis.

3. FSAR Section 6.2.1.1.3.7 - selection of mesh spacing in the structural heat sinks.

4. FSAR Section 6.2.1.4.9, Tables 6.2.1.4, 6.2.1.4A, 6.2.1.4B, and Section 6.2.1.1.3.11 - Revised mass and energy releases for postulated Main Steam Line Break Scenarios and the assumptions and calculation method for the mass and energy releases to the containment following the equilibration phase of a postulated Main Steam Line Break (MSLB) in the loops with faulted and intact steam generators.

5. FSAR Section 6.2.1.1.3.11 - assumptions regarding the spillage of Emergency Core Cooling System (ECCS) water during the blowdown, refill/reflood, and post-reflood phases of the postulated LOCA scenarios.

6. FSAR Section 6.2.1.4.8 - Containment Spray System (CSS) actuation delay times.

7. FSAR Section 6.2.1.4.8 - Main Feedwater (MFW) valve closure time, at CPSES Units 1 and 2, following postulated Main Steam Line Break scenarios.

8. FSAR Section 6.2.1.1.3.11 - Use of LOCTIC computer code for containment pressure and temperature analysis at CPSES Units 1 and 2.

Summary of Evaluation:

All changes, as described above, to CPSES Units 1 and 2 FSAR Chapter 6.2 have been evaluated as part of TU Electric's

Evaluation Number
SE-94-082

containment P/T analyses for postulated LOCA and MSLB scenarios. The assumptions, results, and conclusions of the in-house containment P/T analyses remain valid and continue to meet the design limits, and are unaffected by this activity.

Evaluation Number
SE-94-084

Activity Title:

LDCR SA-93-152; Revision to FSAR Section 1A(B) and Table 8.3-10 to Add RCP Circuit PT Cabinets for Not Requiring Internal Cable Separation

Description of Change(s):

The Licensing Document Change Request (LDCR) is to update the FSAR Table 8.3-10 and Section 1A(B) by adding the RCP circuit PT cabinets to the list of equipment not requiring internal cable separation and it also provides that physical separation and an isolation device is not required between the RCP circuit Non-Class 1E cables and Class 1E PT input terminals.

Summary of Evaluation:

All possible voltage and frequency degraded conditions as well as short circuit condition have no adverse effect on the performance of the Class 1E potential transformers(PTs). The degraded conditions are adequately reflected in the PT secondary voltage. Under voltage, open circuit, short circuit and ground fault conditions in the Non-Class 1E portion of the circuit, result in the degraded voltage on the PT secondary side which in turn result in the Reactor Trip initiation signal by this channel. Similarly the degraded frequency condition also result in the Reactor Trip initiation signal.

The maximum possible over voltage condition at CPSES, have no adverse effect on the performance of Class 1E PTs.

The cables connecting the RCP circuit to the Class 1E PTs, are fire retardant and routed in dedicated conduits, which precludes any external Non-Class 1E cable related fire damage to the PT cabinets.

Based on the this evaluation, separation or an isolation device between the Non-Class 1E cables and PT input fuse terminals, is not required and implementation of the above activity does not involve an unreviewed safety question.

Evaluation Number
SE-94-085

Activity Title:

DM 93-072, 94-019; LDCR SA-94-144; Replacement of Containment Spray Pump Impellers and FSAR Section 6.2 Revision to Reflect Changes

Description of Change(s):

'n order to reduce Containment Spray System vibration (see SE-012), this activity replaces the Unit 2 Containment Spray pump impellers with impellers of a different design. As a result of this modification, the pump characteristics, the Containment Spray system performance and Net Positve Suction Head requirements have also changed.

Summary of Evaluation:

The Containment spray performance and the system parameters were reviewed. The review identified that while the pump NPSH and system characteristics have changed the resulting system performance was determined to be acceptable (i.e., adequate NPSH is available and minimum and maximum required system flowrates are satisfied).

Evaluation Number
SE-94-087

Activity Title:

Change in Committed Schedule for Performance of Deferred Preop Testing of Unit 2 Reactor Cavity Skimmer Pump from 2RFO1 to 2RFO2.

Description of Change(s):

CPSES previously committed to performing the deferred preoperational testing of the Unit 2 Reactor Cavity Skimmer Pump and vibrational testing of the associated piping during 2RFO1. However, the Unit 2 reactor cavity skimmer pump was in need of repair and the preoperational testing could not be performed during 2RFO1. The required preoperational testing of the Unit 2 Reactor Skimmer Pump and vibrational testing of the associated piping has been rescheduled for 2RFO2.

Summary of Evaluation:

During the licensing of CPSES Unit 2, TU Electric deferred the preoperational testing of the Unit 2 Reactor Cavity Skimmer Pump to 2RFO1. However, during 2RFO1, the Unit 2 Reactor Skimmer Pump was in need of repair and the preoperational testing could not be performed during the refueling outage. Since the performance of preoperational testing of the Unit 2 Reactor Cavity Skimmer Pump and associated piping requires that the reactor cavity be flooded (which only occurs during refueling outages), the required preopeational testing has been rescheduled for 2RFO2.

The surface skimming function performed by the Reactor Cavity Skimmer Pump is a non-safety related function. The skimmer system is only placed into operation during refueling outages when the cavity is filled and additional clarity is needed beyond what the refueling water purification system can provide. The required testing will still be performed prior to the system being required to performed its design function.

The rescheduling of the preoperational testing of the Unit 2 Reactor Cavity Skimmer Pump and associated piping does not create an unreviewed safety question.

Evaluation Number
SE-94-088

Activity Title:

LDCR SA-94-148;Revision to FSAR Sect. 3.6B to update the environmental analysis in the Main Steam and Feedwater penetration areas.

Description of Change(s):

This activity involves a revision to FSAR Section 3.6B as applicable to the environmental analyses resulting from a one square foot non-mechanistic crack outside of containment in the main steam steam and feedwater piping, located in the safeguards penetration area (superpipe). The revision updates the FSAR to be consistent with the current CPSES environmental analysis, which incorporated revised Westinghouse supplied mass and energy releases.

Summary of Evaluation:

A bounding environmental analysis was performed using the revised mass and energy releases. The environment in compartments containing equipment required for the safe shutdown of the plant remained unchanged and did not result in any changes to equipment qualification parameters or compartment pressures and temperatures.

Evaluation Number
SE-94-089

Activity Title:

DM 93-034; LDCRs SA-94-042,-104; Modification of RCS Reduced Inventory Measurement Systems and Revision to FSAR Fig.5.1-1 to Reflect Changes

Description of Change(s):

Transmitter piping and valve equipment added and connected to the CRDM's on the Reactor Vessel to provide compensated level indication for when the Reactor Vessel head is removed or on the reactor vessel.

Summary of Evaluation:

DM 93-34 compensates the existing measurement system in the event the reactor vessel pressure differs from the containment pressure. Additionally, an extended wide range transmitter will be connected to extend the range of measurement available when entering reduced inventory conditions. Evaluations have been performed to assess the effects of the modified reduced inventory measurement system on the RCS boundary and the operability enhancement of the existing system. Structural analysis was performed on the tubing, supports, valves and the CRDM end cap modification to ensure that the design was within the acceptance criteria for these systems. All systems were found to satisfy the codes and standards requirements. All equipment was determined to meet the equipment qualification standard Section 4.4 of IEEE 279-1971 and determined that it would not become a missile hazard during a design base earthquake.
Based on the results of this evaluation, implementation of the proposed design modification does not involve an unreviewed safety question.