COMANCHE PEAK STEAM ELECTRIC STATION

10CFR50.59 EVALUATION SUMMARY REPORT 0006
JANUARY 1, 1995 - FEBRUARY 1, 1996

TEXAS UTILITIES ELECTRIC COMPANY

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COMANCHE PEAK UNITS 1 AND 2
10CFR50.59 EVALUATION SUMMARY REPORT 0006

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

Evaluation Number
SE 91-062
Revision 6

Activity Title:
Radioactive Material and Radioactive Waste Handling and Staging in Areas Outside of the Plant

Description of Change(s):
Due to insufficient space inside the plant, designated areas outside the plant are required for radioactive material handling and staging. The fenced area east of the Fuel Building and areas in and adjacent to Wharehouse C will be used for radioactive material and radioactive waste handling and staging. This activity involves the following: a. Storage of radioactive materials (e.g., contaminated scaffolding and outage equipment, etc.); b. Staging of radioactive waste (e.g., resins, filters, Dry Active Waste (DAW), waste oil, etc.) on a short term basis pending shipment to an intermediate waste processor or waste disposal site; c. Handling and sorting of DAW; d. Staging/storage of contaminated fluids including the use of the onsite vendor laundry; and e. Storage of mixed waste in Warehouse C (This evaluation is concerned with only the radiological aspects of mixed waste; the hazardous waste aspects are covered in other documents).

Summary of Evaluation
It was concluded that this activity does not affect safety related structures, systems, components and/or system parameters. This evaluation considered normal operations in the subject areas as well as the following credible failures: a. Dropping a High Integrity Container (HIC) and the subsequent spillage of high activity resin in an area outside the plant (worst case scenario); b. Tornado winds damaging DAW boxes; c. Fire water "washing" DAW or contaminated equipment and causing a release of contaminated water; and d. Leakage of containers with contaminated process fluids or leakage of laundry water. It was determined that this activity does not involve an unreviewed safety question because the impacts of the credible mishaps are enveloped by existing analyses that are within 10CFR100 siting criteria limits.

Evaluation Number
SE 91-062
Revision 7

Activity Title:

Radioactive Material and Radicactive Waste Handling and Staging in Areas Outside of the Plant

Description of Change(s):

Due to insufficient space inside the plant, disignated areas outside the plant are required for radioactive material handling and staging. The fenced area east of the Fuel Building and areas in and adjacent to Wharehouse C will be used for radioactive material and radioactive waste handling and staging. This activity involves the following: a. Storage of radioactive materials (e.g., contaminated scaffolding and outage equipment, etc.); b. Staging of radioactive waste (e.g., resins, filters, Dry Active Waste (DAW), waste oil, etc.) on a short term basis pending shipment to an intermediate waste processor or waste disposal site; c. Handling and sorting of DAW; d. Staging/storage of contaminated fluids including the use of the onsite vendor laundry; e. Storage of mixed waste in Wharehouse C (this evaluation is concerned with only the radiological aspects of mixed waste; the hazardous waste aspects are covered in other documents); f. Storing of special nuclear material, such as detectors or sealed sources, containing less than 1 gram of material each. In addition, new fuel remaining in the DOT shipping containers, may be brought into Wharehouse C for staging purposes; g. Operation of a self-contained CO-2 decontamination facility for decontamination of tools, hardware, and other equipment in Wharehouse C.

Summary of Evaluation

It was concluded that this activity does not affect safety related structures, systems, components and/or system parameters. This evaluation considered normal operations in the subject areas as well as the following credible failures: a. Dropping a High Integrity Container (HIC) and the subsequent spillage of high activity resin in an area outside the plant (worst case scenario); b. tornado winds damaging a DAW box, a Rad Vault, a sealed source container, or other containers; c. fire water "washing" DAW or contaminated equipment and causing a release of contaminated water; d. Leakage of containers with contaminated process fluids or leakage of laundry water; e. Failure of HEPA filter for CO-2 decontaminiation system. It was determined that this activity does not involve an unreviewed safety question becasuse the impacts of the credible mishaps are enveloped by existing analyses that are within 10CFR100 siting criteria limits.

Evaluation Number
SE 93-019
Revision 2

Activity Title:

PCN SOP-507-R6-6 to utilize the Unit 1 Reactor Makeup Water Storage Tank to supply water to Unit 1 & 2 during Normal Operation

Description of Change(s):

The proposed activity would allow Unit 1 and Unit 2 to utilize either Reactor Makeup Water Storage Tank (RMWST) as a source of water for both units in any normal mode of operation. In the event that a single RMWST is unavailable for any reason, both units stillrequire a clean source of Reactor Makeup Water from the remaining RMWST until the affected tank can be restored to service. It is expected that by observing the operating configurations described in I.3, licensing basis requirements, specifically GDC-5, for both RMWSTs can be met using onlyone tank to supple water for both units. While not considered a normal operating condition, reliance on only one RMWST would allow both units to remain operational during periods of a single tank outage.

Summary of Evaluation

The safety related volume requirement, on which SSER 22 is based, is identified in FSAR Section 9.2.3 as a seismic Category 1 supply of makeup water for the CVCS,SCH,CCW, and SF systems. This safety related volume is calculated in ME-CA-0241-4003 as 47000 gallons for both units. Additionally, a non-safety related requirement of 31000 gallons of Reactor Makeup Water per unit is identified in FSAR Section 9.2.3 and DBD-ME-0241. Therefore, reliance on one tank to support both units would be bounded by a total volume requirement of 62000 gallons (RMWST Level of 836'-1"). This volume is not generated by Tech Specs nor is there a commitment to maintain this volume at all times. No TS LCO is affected if this volume is not available. In the event that water level is allowed to fall below 836'-1" during single tank operation, however, makeup should be initiated as soon as practical to restore RMWST volume.

Evaluation Number
SE 93-088
Revision 0

Activity Title:
LDCR SA-93-100; MM-93-241, R.0; Provide Additional Post LOCA Cooling for Auxiliary Building Rooms 244 and 246

Description of Change(s):
ESF filtration units CPX-VAFUPK-01 & 15, and CPX- VAFUPK-02 & 16, are located in Auxiliary Building Rooms 244 and 246, respectively. Filtration units 01 and 02 are supplied with inlet registers which pull in 3000 scfm air flow to each room. This is sufficient to maintain the rooms at or below 140 degrees F during post LOCA operations. All four ESF filtration units are provided with electric heating coils whose control panels are qualified for mild environment (131 degrees F maximum). This activity provides additional inlet registers to filtration units 15 and 16, which will increase air flow to each room by 3000 scfm and decrease post LOCA maximum temperatures to below 131 degrees F.

Summary of Evaluation
This activity adds inlet registers to ESF filtration units. The registers are Safety Class 3 and siesmic Category I. They are non-active components and do not perform any active function during post accident scenerios. Following installation the systems are tested to ensure that negative pressure requirements for the Auxiliary, Safeguards and Fuel Building negative pressure envelope are maintained. Based upon the results of this evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 94-025
Revision 1

Activity Title:

DMs 93-065,-066 & 95-079; LDCRs SA 94-044 & 95-119; OD 94-007 & 95-006
Replacement of Radiation Detectors for 1/2 RE-5100 and X-RE-5251A

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. Also, debris in the sample line to radiation monitor X-RE-5251A, which monitors discharge from Auxiliary Building Sumps 3 & 11, Unit 1 and 2 Diesel Generator Sump pumps and component cooling water drain tank pumps, has caused sample flow problems that result in frequent, spurious diversion of process flow from the Low Volume Waste (LVW) pond to the Waste Water Holdup Tank (WWHT). Once aligned to the WWHT, realignment to the LVW is difficult due to repeated tripping of the sample pump.

These design modification activities replace the radiation detector of the subject monitors. For each monitor, the current "offline" type detector is replaced with an "adjacent-to-line" detector which eliminates the need for a flow switch, pump and temperature controller. The new "adjacent-to-line" 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 radiation monitors are required for accident monitoring, are routinely used for liquid effluent monitoring, and are considered non-1E and non-seismic. The new "adjacent-to-line" radiation monitors are equivalent to the "offline" monitors in that they perform the same function albeit by a different detector design. Credible potential failue modes are failure of the mounting for the new detector/shield assembly or the introduction of the detector /shield assembly as missiles during a seismic event or tornado. An additional failure mode associated with detector X-RE-5251A is introduction of the related access platform/structure as a missle during a seismic event or tornado. Although this platform is located near some feedwater piping, the affected portion of the feedwater system is non-safety and non seismic. A seismic event or tornado that may cause the X-RE-5251A access platform or any new detector/shield assembly to break away from the pipe, become a missle, or affect the Turbine Building drain line, feedwater piping, 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 structures, 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 94-033
Revision 0

Activity Title:
DM 92-064, LDCR SA-92-065: Upgrade of Reactor Coolant Pump Cartridge Seal and the Changes to FSAR Figure 9.3-10 to Reflect the Upgrade

Description of Change(s):
This activity involves upgrading the Reactor Coolant Pump Seals with cartridge type seals.

Summary of Evaluation
The proposed Design Modification includes the conversion of Model 93 AS Reactor Coolant Pumps with standard seals to accept cartridge seals. These cartridge seals are similar in function and performances to the original equipment, and, as a unit, provides the same sealing function as the original equipment. Changes to the RCP pressure boundary include substitution of a new lower seal housing incorporating a gasketed joint and new bolting associated with the lower seal housing and cartridge seal.

Evaluation Number
SE 94-039
Revision 0

Activity Title:
DBD-EE-041R5;CALC ME-CA-0-1093R2;LDCR SA-94-084;Revision to Credit for Adequacy of Available Volt Other than 80% of MOV Volts at its Terminal

Description of Change(s):
ONE form 94-869 identified that certain Class 1E Motor Operated Valves (MOVs) need more than 80% rated voltage at the motor terminals to meet torque requirements. DBD-EE-041 R5 is being revised by DCN-8133 R0 to include the minimum voltage requirement for the affected Class 1E MOV motors. Calculation ME-CA-0000- 1093 R2 is being revised to evaluate the adequacy of available voltage at the MOV motor terminals. FSAR is being revised by LDCR SA-94-084 to take credit for the adequacy of available voltage, other than generic value 80% of the rated voltage at the MOV motor terminals.

Summary of Evaluation
According to the present design requirements, all safety related Class 1E motors are designed to accelerate their driven load with 80% of motor rated voltage available at the terminals. A review of MOV performance based on Limitorque 10CFR21 notification (VL- 16556) and Generic Letter 89-10, has revealed that some Class 1E MOVs need other than 80% of rated voltage at the motor terminals to meet their torque requirements. It has been determined per calculation ME-CA-0000-1093 R2 (with CCN 1 thru 16) that the respective MCC buses provide adequate voltage at the motor terminals of the affected MOVs to perform their functions. For all Unit 1 and 2 MOVs for which credit is taken for the presence of voltages greater than 80% of the motor nameplate voltage, the voltages are not greater than the least of the voltages calculated as being available during sequencing or other operating conditions. Also for MOV 2-8808D with a motor nameplate voltage of 480V AC, the motor starts and operates at 368V AC (ie. 76% of 480V AC). Therefore, requiring other than 80% of rated voltage at the motor terminals of the affected MOVs is not a concern. The above evaluation does not indicate a potential credible failure mode for the MOVs due to this requirement for higher terminal voltage. The respective MCCs provide adequate voltage at the terminals of the affected MOVs and they perform the required functions as well as do not affect the safety or function of the Class 1E system. The activities do not have any effects on accidents, malfunction of equipment or margin of safety. Based on the results of this evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 94-045
Revision 0

Activity Title:

MM 94-106,-107,-108: Dual Seal Retrofit to Refueling Lift Gates

Description of Change(s):

The three minor modifications change the inflatable seal design for the three refueling lift gates from a single seal to a dual seal. These gates are used in the fuel building transfer canal and in each containment fuel transfer area. Minor modification (MM) 94-106 (CP2-SFHAGT-01), MM-94-107 (CP1-SFHAGT-01), and MM-94-108 (CPX-SFHAGT-03) replace the existing reinforced rubber seal with two smaller seals in series. Either seal will perform the design function independent of the other. The design change will enable the lift gates to be operated similar to the pool swing gates which were orininally supplied with dual seals. The modifications seal weld the existing roll pin holes for the large seal to prevent a crud trap. A non-structural metal spacer is welded in to the existing channel to provide the correct channel depth for the smaller seals. New center retaining clips are installed in the center of the channel and new holes for seal retainer pins are drilled in the channel sides to hold the new seals. In lieu of roll pins, socket head cap screws are being used to aid maintenance replacemnt. The new seals are similar to the origninal seals in design, except they have a smaller diameter and they are reinforced with kevlar instead of nylon. Kevlar is superior to nylon because it has a longer shelf life and is more radiation resistant. The new design will ensure that no single failure of a seal or a pneumatic supply will result in loss of gate function or a drain-down event

Summary of Evaluation

The modification to the gate seal design to provide dual seals will significantly reduce the probability of a loss of gate function. A number of drain-down events have occurred at various nuclear power plants which were caused by the loss of a connected air supply. The pool gates have dual seals which are operated with one seal inflated and isolated by a quick disconnect so that no air supply failure can affect it. The other seal is connected to a continuous pneumatic supply which has a backup available if required.

The modifications have been designed in accordance with the applicable codes and standards and licensing commitments. The seismic Category I design of the gates has been maintained. The seismic Category I design of the storage rack has been assured. The materials have been evaluated to be suitable for their application and have been procured appropriately. The spacer plate and retainers are non-safety related parts. The seals are nuclear safety related. Therefore, it was concluded that these modifications improve safety and are in conformance with the current Licensing Basis.

Evaluation Number
SE 94-051
Revision 0

Activity Title:
DM 93-077,-078;LDCR SA-94-096;Addition of Secondary Corrosion Monitors & Local Dissolved Oxygen Analyzers and Related FSAR Section 10 Update

Description of Change(s):
The activity will permanently install (4) corrosion product monitors to routinely sample the Feedwater, Heater Drains, Steam Generator Blowdown, and Condensate systems for evidence of erosion/corrosion product transport. To meet the sampling media requirements, three of the four samples will require cooling to reduce sample temperatures. Cooling water will be supplied by Turbine Plant Cooling Water (TPCW) to two Monitors while Demineralized Water will be used to provide cooling water to the third Monitor. Two Dissolved Oxygen Analyzers will be installed to provide continuous sampling at the Feedwater and Condensate Discharge Header sample points. Data from these analyzers will be feed to the Plant Computer via Chemistry Digital Acquisition System. Equipment drains will be routed to a suitable floor drain. Two piping penetrations through Turbine Building 803' Elevation floor will be added for TPCW lines to the Corrosion Product Monitor coolers for the Feedwater sample.

Summary of Evaluation
This activity is an enhancement to the currently installed secondary side chemistry monitoring and control program. This activity and it's implementation involves secondary side, non- safety related equipment. Class 5 lines to be added will be seismically supported as required. There is no impact on the operation, function, or response of any of the affected systems. The evaluation concluded that there is no potential impact on safety as a result of this activity or its implementation. There is no impact on fire suppression systems, fire barriers or the Fire Protection Program. Therefore, this activity and its implementation does not represent an unreviewed safety question, a new accident not previously evaluated or a reduction in the margin of safety.

Evaluation Number
SE 94-053
Revision 0

Activity Title:

DM 90-509, LDCR SA-94-095; Upgrade of Unit 1 Loose Parts Monitoring Syst. and Revision to FSAR Sect. 4.4.6.4 to Reflect Associated Changes

Description of Change(s):

The existing Unit 1 Loose Parts onitoring System (LPMS) was upgraded to a newer LPMS because the existing LPMS is obsolete and spare parts cannot be obtained to support future operation. The change required removal/replacement of the existing LPMS cabinet and installing the new system cabinet. LPMS sensors, and their locations, are not affected by this change.

Summary of Evaluation

The Unit 1 LPMS was upgraded with a new model system by replacing the existing cabinet and interior sub-components with a new cabinet and sub-components. The change does not affect the installed sensors and their locations.

The new LPMS, like the existing system, is designed and performs to satisfy Regulatory Guide 1.133 requirements for detection of loose parts in the reactor coolant system (RCS) to avoid or mitigate safety related damage to or malfunctions of the RCS. The new system has been seismically qualified to function following an operating basis earthquake. The cabinet is also mounted seismic category II. The cabinet and its support/anchorage are designed to sustain loads generated due to a safe shutdown earthquake, and maintain structural integrity. The new system presents no new failure modes for the plant, plant systems, or for the LPMS. This activity does not involve an unreviewed safety question.

Evaluation Number
SE 94-061
Revision 0

Activity Title:
DM 93-033; LDCR-SA-94-104; Modification of RCS Reduced Inventory Measurement Systems & Rev to FSAR Fig. 5.1-1 Sh1 to Reflect Changes

Description of Change(s):
Transmitter piping and valve equipment are added and connected to the CRDM's on Unit 1 Reactor Vessel to provide compensated level indication when the Reactor Vessel head is removed or is on the reactor vessel.

Summary of Evaluation
DM 93-33 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.

Evaluation Number
SE 94-062
Revision 0

Activity Title:
Modification of Maintenance and Surveillance Program Requirements for the Emergency Diesel Generators

Description of Change(s):
This activity modifies the current maintenance and surveillance (M/S) program requirements for the Emergency Diesel Generators (EDGs). These requirements were originally specified in NUREG-1216 as a result of the Transamerica Delaval (TDI) diesel generator Owner's Group Design Review/Quality Revalidation (DR/QR) program. Subsequently, substantial operational data and inspection results have demonstrated that the special concerns of NUREG-1216 are no longer warranted. This activity results in the EDGs being treated on a par with other EDGs within the nuclear industry and subjected to the same standard regulations, without the special requirements of NUREG -1216. Maintenance activities in the future will be in accordance with the Cooper-Enterprise Clearinghouse Preventative Maintenance Plan. This issue has been reviewed and approved by the NRC, as documented in TDI Owner's Group Generic Topical Report, TDI-EDG-001-A, March 17, 1994.

Summary of Evaluation
This activity results in a future maintenance program in accordance with the Cooper-Enterprise Clearinghouse Preventative Maintenance Plan and allows increased flexibility in determining when and how maintenance activities are scheduled for the EDGs. This action will improve availability of the engines for service, especially during outages, while maintaining current reliability levels. Future revisions of the M/S program will be subjected to the provisions of 10CFR50.59. This activity superseded the commitments related to the EDG M/S program made in SSERs 6, 12, 22, and 25.

Evaluation Number
SE 94-077
Revision 0

Activity Title:
DM 94-004: Installation of High Density Fuel Storage Racks in Spent Fuel Pool 2

Description of Change(s):
Design Modification (DM) 94-004 includes the relocation of the low density racks from Spent Fuel Pool 2 (SFP2) to Spent Fuel Pool 1 (SFP1) and the installation of high density racks in SPF2. The first phase involving removal of the low density racks from SFP2 is complete. This safety evaluation concerns only the second phase of the activity which is the installation of high density racks in SFP2. Use of the high density racks to store fuel is contingent upon issuance of a License Amendment by the NRC and is not included in this Safety Evaluation (LDCR-SA-94-135).

Summary of Evaluation
This phase 2 of the activity includes the preparation for, and the installation of, high density storage racks in SFP2. SFP2 preparation activities include draining the pool, removing existing bolting at the pool floor, and installing bridge plates and shims at the pool floor.

A safety related,, single-failure- proof heavy lift system will be temporarily installed on the Fuel Handling Bridge Crane rails and utilized to accomplish this installation. The temporary rack handling crane that will be installed for the installation of the high density 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, ANSI 14.6 for design of special lifting devices for radioactive materials, and is designed to meet the Quality Assurance requirements of 10CFR50 Appendix B and RG 1.33 revision 2 for critical components. As such, the temporary rack handling crane can retain the maximum design load during a Safe Shutdown Earthquake and remain in place under all postulated seismic loadings.

No loads will 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 installation 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.

All rack installation activities will occur away from the SFP2 swing gate and the Fuel Building lift gate. Since the swing gate and lift gate seals are protected from any direct rack contact, no adverse affects on the seals will occur. The effects associated with rack-to-rack or rack-to- wall interation due to rack sliding or tipping during a seismic event have been analyzed and found to be acceptable.

There are no credible failure modes or accidents that are introduced by this activity.

Evaluation Number
SE 94-078
Revision 1

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):
Evaluation Number 94-078 Revision 0 was submitted to the NRC as part of the TU Electric annual summary submittal for the year 1994 attached to letter TXX-95010 dated February 1, 1995. Revision 1 of the evaluation is pertaining to the same activity described in Revision 0. Also the two revisions have the same conclusion and justification. However, revision 1 is issued to delete reference to Thermal Science Corporation test, to resolve NRC concern during inspection.

Summary of Evaluation
Refer to the description section above.

Evaluation Number
SE 94-083
Revision 2

Activity Title:
DM 93-068; Reactor Protection System Testing In Bypass Modification for Unit 2

Description of Change(s):
The Reactor Protection System Testing Bypass design modification, introduce bypass testing circuitry (BTC) for the NIS power range reactor trip functions and the 7300 Process Protection System reactor trip functions and ESF functions. With the implementation of the modification, spurious reactor trip or safeguards actuation will be avoided since the partial trip conditions are eliminated. No modification to coincident logic, trip setpoint, or surveillance requirements are associated with this change.

Summary of Evaluation
The evaluation concludes that the implementation of the subject modification does not result in a failure mode which increases the consequence of a malfunction of equipment important to safety. Surveillance testing in bypass does not affect accident initiation sequences or response scenarios as modelled in the safety analyses. The proposed change does not alter the manner in which acceptance limits, limiting safety system setpoints, or limiting conditions of operation are determined. The change does not reduce the margin of safety. Testing in bypass is expected to result in an overall improvement in safety by reducing unnecessary transients and challenges to the protective system by minimizing partial trip which may lead to inadvertent trips. Based upon the results of this evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 94-091
Revision 0

Activity Title:
DM 94-018;LDCR SA-94-159;Replacement of Unit 1 Containment Spray Pump Impellers;FSAR Revision to Reflect Change of Net Positive Suction Head

Description of Change(s):
The Unit 1 Containment Spray pump impellers will be replaced under DM 94-018. The new impellers are of a different design and thus the characteristics of the pump will change. Due to the change in pump characteristics, the Containment Spray piping system performance characteristics will also change. LDCR-SA-94-159 updates the FSAR to reflect the changes in the NPSH.

Summary of Evaluation
Both the Containment Spray pump and system characteristics were reviewed. The review identified the areas of pump Net positive suction head (NPSH) and the system flow performance as potentially changing. While the Containment Spray pump characteristics could change, the resulting system performance is expected to perform similarly to the Containment Spray pumps in Unit 2 and therefore will be acceptable. The minimum and maximum required flow rate rates were satisfied. The impeller change does not introduce new system failure modes, new material issues, component failure modes or new failure modes for the pump. There were no impacts to the accident analysis created by this change nor was the probability of occurrence of a licensing basis accident impacted nor was a new type of unanalyzed event created. Therefore, based on this evaluation, no unreviewed safety question exists or is created with the implementation of this design modification.

Evaluation Number
SE 94-092
Revision 0

Activity Title:
LDCR SA-94-158; PCN STA-113; Deletion of Requirement to Perform a Written Annual Assessment of Adequacy and Effectiveness of QA Program

Description of Change(s):
The CPSES FSAR, Chapter 17.2.2, requires the Senior Management QA Overview Committee to provide an annual written assessment of the adequacy and effectiveness of the QA Program to the Group Vice President, NP (c.f., TXX-88495, TXX-90060, Commitment # 23129). STA-113, Rev. 1 (PCN 01 through 03), implements this requirement in Paragraph 6.2.1. The activity involved is to delete the requirement for an annual written assessment. Therefore, the proposed change when implemented, will revise the QA Program, resulting in a deleted commitment, a revision to the FSAR, and a revision to STA-113.

Summary of Evaluation
The written annual assessment on the overall adequacy and effectiveness of the QA Program provides a post-fact static evaluation. Technology enhancements at CPSES and established processes, such as the Plant Performance Overview Report and the NOD Monthly Integrated Report, provide more dynamic, on-line capabilities for Senior Management to be involved in QA Program implementations. Because of technological development and other processes, the "written annual assessment" has become a process that adds no value to the overall quality and effectivemess of the QA Program. The following statement satisfies the requirements of 10CFR 50.54(a). The proposed change revises the QA Program at CPSES but does not reduce the commitments in the Program description previously accepted by the NRC. The proposed change does not decrease the effectivenss of the QA Program at CPSES.

Evaluation Number
SE 94-093
Revision 0

Activity Title:
DM 94-022,-023; LDCRs SA-94-161,-96-052; Replacement of the Existing Unit 1 Lead and Backup Instrument Air Compressors/Associated Dryers

Description of Change(s):
The design modifications replace the existing Unit 1 lead and backup instrument air compressors and associated dryers with larger more efficient units. The new compressors and dryers are supplied electrical power from the 480 V Class 1E switchgear and the cooling water from the non-safeguards loop of the component cooling water systems. FSAR sections 8.3 and 9.3 are also revised to reflect the associated changes.

Summary of Evaluation
These modifications are to replace the existing Unit 1 lead and backup compressors CP1-CICACO-01 and CP1-CICACO-02 and the associated dryers CP1-CIDYIA-01 and CP1-CIDYIA-02 with more efficient and reliable larger capacity units. As a result, a source of high grade clean dry air is available to support current plant operation and allow for future growth. The new compressors and dryers have a higher power demand than the existing units and are powered from the 480 V Class 1E switchgear and sequence on the emergency diesel generator following blackout signal as presently designed. The increased electrical loading is analyzed in the diesel generator loading calculations and is acceptable. All portions of the design are within the prescribed design and licensing basis criteria. Based on the results of this evaluation, implementation of the proposed activities do not involve an unreviewed safety question.

Evaluation Number
SE 94-094
Revision 0

Activity Title:

DM 94-025:Modification of Plant Security System to Replace Existing Card Reader Cipher Pads with Hand Reader Biometric Identifier System

Description of Change(s):

This activity involves a plant design modification to replace card reader cipher pads at the Protected Area access points with the Hand Geometry Biometric Identifier System (Biometrics System). Biometrics System hand readers were installed in front of access turnstiles to identify each individual who is authorized unescorted access to the CPSES Protected Area. The individual's identification is verified by placing his/her badge in the access card reader and then placing their hand on the measuring surface of the Biometrics System hand reader. The Biometrics System compares the individual's "hand scan" with the unique physical characteristics of the individual's hand (hand geometry) which has previously been registered with their badge number in the access control system. No one can use a badge to gain access except the individual whose hand geometry has been registered to that badge. The Biometrics System replaces the previous system of individual identification,i.e., individual identification by Security officer at the time of picture badge issue at the Protected Area access point followed by the individual's input of a unique code into a card reader cipher pad.

Summary of Evaluation

The Biometrics System is a superior personnel identification process because it provides a nontransferable means of identifying individuals, unlike a picture on a badge or entering a code in a cipher pad. This activity is an enhancement to Protected Area access control and will eliminate the requirement for security officers to issue, retrieve and store picture identification badges at the entrance stations to the CPSES Protected Area. The installation and use of the Biometrics System is supported by approved changes to the CPSES Security Plan and an NRC approved exemption to certain requirements of 10CFR73.55(d)(5).

This activity installs the Biometics System in the Primary Access Point (PAP) and Alternant Access Point (AAP) which are classified as non-safety related and non-seismic structures. This activity is non-safety related and installation is in an area where no safety related equipment is located. The evaluation concludes that the activity has no effect on the accidents and malfunctions evaluated in the licensing basis documents, has no potential for creation of a new type of unanalyzed event, and affects no technical specifications; therefore, there is no unreviewed safety question.

Evaluation Number
SE 94-096
Revision 0

Activity Title:
DM 94-036; Installation of Door in the Unit 1 Containment Access Hallway

Description of Change(s):
This activity installs a removable hollow, metal door along the removable block wall in the Unit 1 Containment access hallway. Additionally, other openings along the top and sides are blocked with gypsum board or other suitable material. This modification provides an alternative boundary to attain a negative pressure in the Safequards building, as required by Technical Specifications. This alternative boundary may be used during refueling outages when both doors of the Personal Airlock (PAL) are open.

Summary of Evaluation
This evaluation looked at the scenarios in which this new wall might be used to provide an alternative boundary to attain a negative pressure in the Safequards building during refueling outages when both doors of the Personal Airlock (PAL) are open. It determined that the primary scenario of concern was the large break LOCA, where explicit credit is taken for ESF filtration; however, limitiations were also identified which required the new door to be open during Modes 1 through 4 due to the current Systems Interaction Program calculations (e.g., environmental, flooding, HELB, etc.). Further evaluation determined that no new credible failure modes could be introduced and that there was no effect on accidents or malfunctions evaluated in the Licensing Basis documents; In addition, there was no potential for the creation of a new type of unanalyzed event and there was no change in the margin of safety as required by Technical Specifications.

Evaluation Number
SE 94-097
Revision 0

Activity Title:
MM-94-398/399/400/401: Modification of the Comp. Cooling Water (CCW) flow Residual Heat Removal (RH) and Cont. Spray (CT) Heat Exchangers.

Description of Change(s):
DM 93-042 and DM 93-043 implemented hardware changes that throttle CCW valves to the RHR and CT heat exchangers to an intermediate position during P-Signal operation. The control circuit opened the normally closed valves to an open intermediate limit switch or closed an open valve to a closed intermediate limit switch on receipt of the signal. The tolerances of the setting of the two switches did not meet the design flow range requirements.

This SE is for a change to the control circuit design to control flow using one limit switch in lieu of two. The normally closed valves will go full open and return to the closed intermediate limit switch on receipt of the signal. An open valve would close to the closed intermediate limit switch on receipt of the signal. This will enable the valves to satisfy the flow range design requirements.

Summary of Evaluation
The impact of the change, including implementation, on all affected systems and components were evaluated and found to be conservative with respect to safety. The change will provide the required design margin for the operation of the systems supported by the CCW heat exchangers. This will ensure their capability to mitigate a DBA. It was concluded that this activity does not involve an unreviewed safety question.

Evaluation Number
SE 95-001
Revision 0

Activity Title:

DCN 8705 R0; LDCRs SA-91-091,-94-145; Update Ventilation Damper Classifications for Unit 1

Description of Change(s):

This updates the flow diagrams and FSAR figures to reflect the as- built NNS classification of the Unit 1 and Common pump room exhaust dampers and their supports were declassified prior to Unit 1 OL. They do not perform any Nuclear Safety functions. The safety injection actuated equipment list is also updated to show that, althou these dampers and their inter-wired NNS room supply counterparts receive S signals, there is no safety function. The electrical train changed to associated to denote the lass of Class 1E functionality requirements.

Summary of Evaluation

The change is a paper change update to correct and clarify the functio of the pump room supply and exhaust dampers. The seismic Category of the damper supports is Cat II in accordance with the FSAR and DBD criteria. The change makes the classification of the exhaust dampers identical to the supply dampers and corrects a document discrepancy since these dampers were declassified on both units during construction. The electrical appurtenances were also declassified on Unit 2. TXX-92065, dated April 6, 1992 (SDAR CP-84-27) advised the NRC that the exhaust dampers were downgraded to NNS; but, the electrical appurtenances were maintained as Class 1E on Unit 1. Unit 2 electrical appurtenances were provided with electrical isolation and declassified to Non-1E. TXX-89760 dated October 9, 1989 (SDAR CP-84-27) stated that the Class 1E limit switches on the NNS dampers were originally supported as Seismic Category 1, but had been downgraded to Seismic Category II during design validation. The change was deemed acceptable based on Seismic Category II design to ensure structural integrity.

Evaluation Number
SE 95-002
Revision 1

Activity Title:

DM 94-037;LDCR SA-95-012; Installation of Electr. Pwr. Feeders Between Alt. Pwr. Supply DGrs. Transfer Switch and 6.9KV Class 1E SWGR

Description of Change(s):

The modification activities involve the installation of electrical power feeder cables between the alternate power supply diesel generators transfer switch and the train A and B Class 1E 6.9 KV switchgear. New feeder cables from alternate feeder supply diesel generators are landed on the load side of existing spare breakers in the train A and B Class 1E 6.9 KV switchgear for providing power to the bus in the event of loss of all AC power (blackout) during Modes 5 and 6. The breakers are normally racked out or disengaged during all plant operating modes. The new feeder cables classified as "associated" are de- energized during all plant operating modes and are run in the existing, safety related Level 1 cable tray inside the building. These cables are routed outside the Unit 1 switchgear rooms 1-83 and 1-103 through new penetrations in the east wall column H-S of the safeguards building and terminate at a train A/B transfer switch.

Summary of Evaluation

The impact of the concrete blockout work in addition to the cable installation was reviewed for impact to the safeguards building east exterior wall barrier function (ie fire, missile, radiation controlled area and security) and determined that the barrier breach for the time necessary to install the conduit sleeves, is acceptable. The installation of the cables in the existing safety related raceways and switchgear is acceptable with respect to the cable ampacity, combustible loading and the existing seismic qualification for the cable tray envelops the additional tray loading resulting from the new cables. Installation of the cable connections into the 6.9 KV switchgear has been evaluated and determined to not compromise electrical separation or operability of either bus. This evaluation concludes that it is acceptable to install the alternate power feed tie-ins to the 6.9 KV switchgear. Based on the results of the evaluation, implementation of the proposed activity does not result in an unreviewed safety question.

Evaluation Number
SE 95-003
Revision 0

Activity Title:

DM 94-035; Installation of Support Structure in Unit 1 Containment to Facilitate Removal/Reinstallation of RCP Motor During Refueling Outage

Description of Change(s):

This activity installs removable structural framework, rails, trolleys etc. in Unit 1 Containment in the vicinity of equipment hatch at EL. 832'-6". The modification also provides temporary supports for Reactor Building slab at EL. 832'-6".

Summary of Evaluation

Above described activity will be performed during refueling outage and while the plant is in defueled mode (with all fuel in the Fuel Building).

In consideration of plant status when the subject activity is performed, this evaluation establishes that no credible failure mode (during MODES 1-6) is introduced for affected structure (Reactor Building slab at EL. 832'-6") and no system or component safety function during the defueled mode is impacted.

The structural framework is designed to sustain gravity loads (dead loads and impact load) and lateral loads generated by transporting of the Reactor Coolant Pump (RCP) motor in and out of Containment. The support columns (shoring of slab at EL. 832'-6") provided to transfer the gravity loads of RCP motor to slab on grade at EL. 808'-0", will be decoupled at the splice once RCP motor removal and reinstallation activity is complete. The support column configuration (decoupled at splice) during plant MODES 1 thru 6 ensures that Reactor Building structural framework is not altered and there is no introduction of new load transfers between floors as well as within the building structural members.

Another consideration is the interactions caused by possible overturning of trolley structure and RCP motor with surrounding equipment/components due to a seismic event. Since the plant is shutdown and is dufueled while the subject activity is being performed, the equipment/components in the area of this activity are not required to operate and are also in shut down mode. Therefore, the safety function of system and components in the area of subject activity is not impacted.

Evaluation Number
SE 95-006
Revision 0

Activity Title:
MM 94-404.LDCR SA-95-017:Extraneous Vent/Suction-Relief Valves Removal to Minimize Unit 1 Cntainmnt Spray Pipe Failures:FSAR Sec 6.2 Revision

Description of Change(s):
MM-94-404 has been issued to remove extraneous vent valves and the four suction relief valves from the containment spray system in Unit 1.

Summary of Evaluation
The deletion of the pump suction casing vents was determined to be acceptable based on a review of the pump manufacturer operations manual which describes the filling and venting requirements.

The Containment Spray system suction relief valves were also deleted. While it is common to protect the pump 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 higher that the maximum pressure resulting from the design basis upset conditions. Thus, the relief valves are no longer required.

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

Evaluation Number
SE 95-007
Revision 0

Activity Title:

MM 94-388; LDCR SA-94-032; Rebalancing Loadings/ Changing Description to Regulating Transformer for Transformers CPX-EPTRNT-42,-43,-44 & -45

Description of Change(s):

Calculation 16345-EE(B)-044 R5 (CCN-001 & -004) has been issued to balance the transformer loads between the phases by rearranging the circuits. The activities associated with this evaluation involve plant modification, revisions to design basis documents DBD-EE-057 & DBD-ME-011 and engineering specification 2323-ES-100 by DCN 7926 R0, and updating FSAR section 8.3 to reflect these loading changes. Additionally, FSAR section 8.3 is revised to change the description of these transformers CPX- EPTRNT-42,-43,-44, and -45 from "isolating transformers to "regulating transformers".

Summary of Evaluation

Some of these transformers were unbalanced between their phases. Therefore the loads in panels fed from these transformers are being rearranged to achieve balanced loads between the phases of the transformers. The rearrangements of the loading have no adverse impact on Class 1E system. These transformers are non Class 1E regulating transformers per DBD-EE-041 Revision 5 and are not used for isolating purposes. The isolation of Class 1E bus is provided by two Class 1E breakers in series at Motor Control Center feeding these transformers and therefore the transformer description is being changed to "regulating transformers". These activities do not have any effect on accidents, margin of safety or malfunction as described in licensing basis documents. Based on the results of this evaluation, implementation of the proposed activities do not involve an unreviewed safety question.

Evaluation Number
SE 95-008
Revision 0

Activity Title:

LDCRs SA-95-020; TR-92-019; Change of Undervoltage RCP Response Trip Time; Addition of Clarification Note

Description of Change(s):

This changes the Undervoltage (UV) - Reactor Coolant Pump (RCP) response time from "1.5" to "1.1" seconds and adds a clarification note. The note states that an additinal of 0.4 seconds maximum calculated voltage decay time from the opening of RCP breaker until voltage reaches the UV set-point provides an overall time less than or equal to 1.5 seconds. (TRM Table 1.1.1 - LDCR-TR-92-019, FSAR Table 15.0.4 - LDCR-SA-95-020, DCN 9000 R0 & Calc. #EE-CA-0008-4022 R0.)

Summary of Evaluation

The reactor trip system instrumentation response time for the "Undervoltage - Reactor Coolant Pumps" is the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. accordingly when the supply bus voltage of the RCP is lost instantaneously, the monitored parameter has exceeded it's trip setpoint, but the channel sensor (U. V. Relay) is unable to detect this due to the effect of back EMF.

Voltage decay is a process of time which falls outside the definitiion of Reactor Trip System Response Time (which starts when the monitored parameter excceds its trip setpoint). The monitored parameter is clearly voltage which does not instantaneously decay; voltage on the bus is not different than flow in the loop. the response time of 1.5 seoncds currently given in TRM Table 1.1.1 and FSAR Table 15.0.4 is not consistent with this definition. The response time has been shortened appropriately to exclude voltage decay time based on calculated and/or tested time for EMF decay. The UV- RCP response time includes 1) time delay for reactor trip breaker to open an RCCA gripper to release, 2) the intentional time delay set into the UV Relay to prevent spurious trips, and 3) inherent UV sensing circuitry time delay for the UV trip setpoint is reached until the UV trip signal is generated.

The respnse time has been changed to 1.1 seconds with a footnote stating that an additional 0.4 second maximum calculated voltage decay time provides an overall response time of less than or equal to 1.5 seconds.

Evaluation Number
SE 95-009
Revision 0

Activity Title:

DCN 6684.R0 for DBD-EE-057; LDCR SA-95-022: Revision to DBD and FSAR Section 8.3.1.2 to Include Frequency Transducer as an Isolation Device

Description of Change(s):

The activities involve revising the design basis and the licensing basis documents to include the frequency transducer as an isolation device between Class 1E circuit and the non Class 1E frequency indicator. FSAR section 8.3.1.2.1 is revised to include frequency transducer in the list of isolation devices.

Summary of Evaluation

The frequency transducer can be used as an isolation device between the Class 1E input circuit and the non Class 1E output circuit because the transducer at the input side provides isolation between the input and the output circuit. If there is fault or short on the secondary metering circuit no excessive load or change would reflect back to the input signal in normal functioning of frequency transducer. This is confirmed by the vendor. This is a paper change only and meets the requirements of the DBD-EE-057. The activity does not result in change to the facility, test or experiments or change to procedures as described in the DBDs or Technical specifications. Based on the results of the evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 95-010
Revision 0

Activity Title:

MM 95-005: LDCR SA-95-024: Removal of Diaphragm from Boric Acid Tank CPX-CSATBA-01 and Revision of FSAR Section 9.3 to Reflect the Removal

Description of Change(s):

This activity will remove the diaphragm from Boric Acid Tank CPX- CSATBA-01. It is known that five small pinhole performations exist in the diaphragm. Because of this, and because the evaluation determined that the diaphargm serves no function (i.e. the diaphargm has no effect on dissolved oxygen concentration in the tank), then it is determined that removal of the diaphargm would be prudent and will have no adverse impact.

Summary of Evaluation

Removal of the diaphargm from the boric acid tank will not impact the safety of the 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 diaphargm is removed. Operating expereience has confirmed that the use of the boric acid solution in the RCS has not resulted in the RCS disolved oxygen concentration to be grater 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 95-011
Revision 0

Activity Title:
LDCR SA-95-027, -96-006: Cycle 5 Core Configuration for CPSES Unit 1

Description of Change(s):
Revise FSAR appendix 4A to provide an up-to-date description of the Unit 1 core configuration for Cycle 5.

Summary of Evaluation
The Cycle 5 core configuration does not introduce any fuel assembly or burnable absorber designs significantly different from those used for the Cycle 4 core configuration. The analyses of the mechanical characteristics of the Siemens Power Corporation (SPC) fuel assemblies and burnable absorbers are still performed by SPC using SPC methodology, and Westinghouse methodology is still used to confoirm the acceptability of the mechanical design characteristics of the Westinghouse fuel assemblies.

The cycle-specific reactor physics, core thermal-hydraulics, and Large Break LOCA safety analyses are performed using the same TU Electric analysis methodologies that were used for the Unit 1 Cycle 4 analyses. The Small Break LOCA safety analysis continues to be based on the Westinghouse NOTRUMP analysis, the same as for Cycle 4. The non-LOCA system transient analyses, performed by Westinghouse using Westinghouse methodologies for Cycle 4, are performed using TU Electric analytical methodologies for Cycle 5. All of the analytical methodologies used to support the Cycle 5 core configuration have been approved by the NRC.

The Cycle 5 core configuration has been evaluated for its safety impact on the plant and TU Electric has determined that all FSAR Chapter 15 acceptance criteria are satisfied and the conclusions of FSAR Chapter 15 remain valid.

Evaluation Number
SE 95-012
Revision 0

Activity Title:

LDCR SA-95-029; DBD ME-250 DCN 9053; Acceptability of Larger than Originally Assumed Mechanical Design Flows for the RCS

Description of Change(s):

This activity is limited to revising the DBD and FSAR to increase the RCS mechanical design flow limitations at 100% power. The single loop flow (output of one RCP) is raised from 105,000 gpm to 109,000 gpm and the total mechanical design flow of the RCS is raised from 420,000 gpm to 424,000 gpm. The original mechanical design flow was exceeded after completion of 2RF01. Furthermore, the original design flow for one RCP is 105,000 gpm but Unit 2 has measured values of approximately 108,000 gpm in one loop. The subject documents were initiated to provide a basis for the acceptability of the existing flowrates and to increase the maximum flows as noted above. The observed Unit 2 flows are the reason this activity is necessry. Unit 1 is changed to add margin against exceeding the mechanical design flows and to avoid creating a unit difference.

Summary of Evaluation

The evaluation concluded that increasing the maximum design flow limits to the noted values does not have an adverse impact on the plant. The review indicated no adverse impact on Westinghouse fuel, Siemens fuel, reactor internals, steam generators, reactor coolant system components subject to flow erosion, reactor coolant pump operation, and accident analysis. This activity provides a basis for operation at 100% with the new mechanical design flow limits as described above. No commitments to the NRC are changed by this activity. Based on the evaluation, larger than originally assumed mechanical design flows in the RCS does not involve an unreviewed safety question.

Evaluation Number
SE 95-013
Revision 0

Activity Title:
LDCR SA-95-030; MM 94-237; Removal of Steam Generator Channel Head Drain Lines Unit 2 Steam Generators

Description of Change(s):
This activity involves removing valves 2RC-8079A,B,C,D as well as their associated tubing and fittings and replacing these parts with a plug. These valves were originally installed to allow a simple way to drain any residual water from the steam generator primary channel head prior to opening the primary manway. However, this method of draining has not been reliable due to drain line clogging and the residual water in the primary channel head being drained through the primary manway. The decision to remove the lines and valves rather than leave them in place is based on two concerns. First of all, these lines have become a crud trap and present significant ALARA challenges when working around them. Secondly, the drain line configuration extends vertically down from the steam generator and is not supported. This configuration can oscillate much like a pendulum during operation and result in accelerated fatigue of the weld which attaches the tubing to the steam generator. For these reasons it was determined to be beneficial to eliminate these lines and plug the penetration at the bottom of the steam generator primary manway.

Summary of Evaluation
The effects this activity has on the plant are not adverse. Among the benefits of this modification are an ALARA savings due to the elimination of a crud trap and the elimination of an identified accelerated fatigue mechanism for the steam generator channel head drain line sleeve penetration. No plant commitments are affected by this activity. This activity has no impact on the daily plant operations or outage manintenance operations/activities. All original design and safety criteria are met. Based on the evaluation, the implementation of the proposed activity does not involve any unreviewed safety questions.

Evaluation Number
SE 95-016
Revision 0

Activity Title:

MM 95-046; LDCR SA-95-032; Modification to Condensate Storage Tank CP1-AFATCS-01 and Revision to FSAR Section 10.4 to Reflect the Changes

Description of Change(s):

Modify Condensate Storage Tank (CST), CP1-AFATCS-01, to include 1) isolation of nitrogen blanket supply lines, 2) modification of pipe supports for line AF-1-920 to eliminate sharp edges that could potentially damage tank diaphragm, 3) addition of a stainless steel screen cover to the CST overflow connections to prevent entrance of birds into the tank and 4)the addition of standoff strips over the penetration for the condensate makeup and reject line to support the diaphragm and prevent the diaphragm from potentially blocking the penetration.

Summary of Evaluation

The removal of the nitrogen blanket capability has no affect on the ability of the aux. feed system to perform its safety functions and introduces no credible potential failure modes which could affect any SSC. The nitrogen blanket is used to aid in control of oxygen in the CST inventory and will no longer be required when the new diaphragm is installed. The support modifications involve the elimination of sharp corners and edges to prevent damage to the diaphragm. The bird screens have twice the cross-sectional area for flow through than the "overflow" vents. The "overflow" vents vent the air above the diaphragm as water level increases. They do not vent water. Installation of the bird screens will not inhibit the venting of the tank. The installation of the standoff strips does not reduce the flow area at the penetration and has no adverse affect on the line, the diaphragm, or tank.

Evaluation Number
SE 95-019
Revision 0

Activity Title:
TM 2-95-004 R0;Installation/Continued Operation of Unit 2 with One of Two Mechanical Turb. Overspeed Prot. Devices Mechanically Restrained

Description of Change(s):
On April 8, 1995, during the testing of the Unit 2 turbine mechanical overspeed protective devices pursuant to Technical Specification 4.3.4.2b, one of the two devices failed to relatch following successful testing. In this condition, an automatic turbine overspeed trip would have occured had the trip device been unblocked. In order to prevent an unwarranted turbine trip/reactor trip and corresponding undesirable plant transient, a mechanical restraint was installed on an accessible portion of the device shaft, holding it in the reset position. As this retrains movement of the device, is renders it nonfunctional. The restrained configuration is documented under a clearance and will be further documented in Temporary Modification 2-95-004 Rev. 0.

Summary of Evaluation
Temporary Modification 2-95-004 Rev. 0 describes the mchanical restraining of one of two Unit 2 turbine mechanical overspeed protection devices. This safety evaluation examines that activity. The conclusion is that the activity does not involve an ureviewed safety question because it does not introduce any new failure modes, does not increase the probability of a turbine overspeed event or it consequences, and compliance with the existing technical specification basis is maintained. This is due principally to the multiple levels of turbine overspeed protection available and functional, and for which the NRC gives credit in the CPSES Safety Evaluation Report.

Evaluation Number
SE 95-023
Revision 0

Activity Title:
TM 2-95-2 Rev. 0: Temporary By-pass of Inoperable Inverter Which Normally Supplies Power to Unit 2 Radio Repeaters.

Description of Change(s):
This temporary modification (TM) bypasses the inoperable inverter which normally supplies power to Unit 2 radio repeaters. This restores functionality to the repeaters during normal operations until replacement inverters are purchased and installed per Design Modification DM 95-048.

Summary of Evaluation
Temporarily bypassing the inoperable inverters, restore power to the radio repeaters resulting in improved radio communications reliability during normal operation. This TM does not add any additional fire loading. Also circuit protection is maintained. No additional failures can result from this activity, since the inverter has already failed and the power supplied to the radio repeaters is within their rating. This TM does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Based on the results of evaluation, the implementation of the activity does not involve an unreviewed safety question.

Evaluation Number
SE 95-024
Revision 0

Activity Title:
LDCR SA-95-054: FSAR Update Of Steam Dump Valve Actuation Characteristics

Description of Change(s):
Update the FSAR to reflect the acceptability of Steam Dump valve stroke time based on achieving required flow rates within the required times, rather than full valve travel, and the greater than indicated maximum flow capacity of the valves.

Summary of Evaluation
The FSAR describes the Steam Dump Valves as being capable of going from full closed to full open within 3 seconds, and capable of being modulated with a maximum full stroke time of 20 seconds. Steam Dump operators use a volume booster to provide for rapid opening; however, when the valves reach about 80 to 90 percent open, the volume booster cuts out and subsequent valve travel time is much slower, often resulting in the valves failing to reach the 'full open' travel stop within the specified time. Calculations indicate that the valves reach the minimum required flow when they are 57% open; therefore they are still able to perform their intended function.

Additionally, the maximum steam flow through the valves is assumed to be less than 269 pounds per second for the 'Inadvertent opening of a Steam Generator Relief or Safety Valve' analysis in FSAR 15.1.4. Contrary to this assumption, calculation indicate that the actual full open steam flow is 308 pounds per second. Nevertheless, operability is unaffected because this condition remains bounded by the double-ended main steam line break evaluated in FSAR Section 15.1.5.

Therefore, based on the results of this Safety Evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 95-025
Revision 0

Activity Title:

TMs 2-95-005 R1 & -006 R0: Use of Fibrous Insulation In-Lieu of Reflective Metal Insulation for Check Valves 2-8378A and/or 2-8378B

Description of Change(s):

Temporary leak repair of the body-to-bonnet joint of check valve 2-8378A and maybe 2-8378B will require installatin of a large "clamp-like" device around the valve body-to-bonnet joint. This will not allow the reinstallation of the normal metallic reflective insulation. For the duration of the temporary modification, an alternate insulating material will be used to maintain local temperatures within acceptable limits. The choosen material is fibrous insulation (fiberglass)

Summary of Evaluation

The insulation has no impact on the valve and will not contribute to any credible valve failure modes. However, as a result of postulated line breaks, the insulation could become debris due to jet impingement. Debris in containment following a LOCA is of concern if it is transportable to the containment recirculation sumps and then only if unacceptable blockage occurs such that the net positive suction head (HPSH) for the ECCS pumps becomes inadequate. No new credible failure modes are introduced as this type insulation already exists within the steam generator compartments. This temporary modification will deviate from descriptions in the LBD's that indicate fibrous insulation is only used on lines for anti-sweat purposes. The existence of fibrous insulation materials in the steam generator compartments is discussed in NUREG-0797, Supplement 9 on pages L-5 and L-6. This discussion concludes that due to the low calculated local flow rates of water durnig the recirculation phase of accident mitigation, "...transport of insulation debris initially deposited in the steam generator compartment is highly unlikely. Therefore, the staff has concluded that insulation debris will not contribute to sump blockage."

The subject valve is in the cone of influence of at least one postulated pipe break but jet impingement is also an unlikely transport mechanism for this material to the sump. There is no direct path from its installation location to the sump and furthermore, there is a screen door installed at the entrance to all four steam generator compartments. This door will provide a very effective additional barrier to bulk transport of this material out of the compartment. In the unlikely event that the material did find its way to the sumps it would reasonably be expected to matt on the floor and lower portion of the sump screen. The paricular fibrous insulation material to be used has no integral backing material that could present a solid surface to currents in the water. Therfore, due to its "porus" nature, it would not be expected to be easily carried by local currents to spread out on the screens and cover any signifcant screen area.

Evaluation Number
SE 95-027
Revision 0

Activity Title:

LDCR SA-95-057; Delete the Storage of Respiratory Protection Equipment (RPE) from Emergeny Response Facilities (except SCBAs from CR & EOF)

Description of Change(s):

This activity changes FSAR Sections 12.5.2.4, 15.6.5.4(4) and III.A.1.2(II(F)) to reflect the removal (no storage) of respiratory protection equipment (RPE) from the CPSES emergency response facilities except for Self Contained Breathing Apparatus (SCBA) for the Control Room and Emergency Operations Facility (EOF).

Summary of Evaluation

With respect to emergency response, full-face respirators, using either HEPA or sorbent cartridges, are no longer intended to be worn by personnel inside the emergency response facilities or on offsite monitoring teams during a radiological emergency at CPSES. Habitability monitoring, including radiological air sampling, is continusously performed in the emergency response facilities. The results of this monitoring are used by senior RP staff and the facilty manager to determine the necessary protective measures to be taken to keep personnel exposures ALARA. Protective measures that may be taken include:

- DAC-hr tracking
- Ingestion of KI
- Facility personnel relocation, and
- Respiratory protective equipment.

RPE remains available onsite for issue if needed, and use is specified in station procedures; however, RPE use is a last resort in protecting individuals from exposure to an airborne hazard. This is because the use of RPE can result in an increase in the overall exposure to an individaul who is not normally required to wear RPE. In additon, increased personal safety considerations must be appraised for those individuals due to the hazards of RPE use (physical and mental stess, fogging, limited vision, limited commumications, etc.). By using a combination of administrative controls, RPE use by facility and offsite team personnel will not be required.

This change to the FSAR presents no new accident types or equipment failure modes at CPSES. This activity proposes no changes in the radiation dose resulting from a design basis accident. No credit is taken in the existing accident analyses for the use of respirators by TSC personnel for the reduction of dose. The protective clothing and respirators (SCBAs) specifed in design basis documents for Control Room personnel are unaffected by this acitiviey. RPE from the TSC is not required by response personnel to enable vital area access for the purpose of accident mitigation. This acitivity has no affect on safety-related systems and equipment, and the radiological consequences of any credible accident are unchanged.

Evaluation Number
SE 95-028
Revision 0

Activity Title:

LDCR TR-95-003: Revisicn to the Required ESF Actuation Time for Feedwater/Steam Line Isolation Valves Closure Upon a SI Actuation Signal

Description of Change(s):

This change revises the required Engineered Safety Features Response Times in Technical Requirements Manual Table 1.2.1 (from less than or equal to 6.5 seconds to less than or equal to 7 seconds) for feedwater isolation (item 2.c: Containment Pressure--High-1 & item 4.c: Steam Line Pressure--Low) and main steam isolation (item 4.n: Steam Line Pressure--Low & item 6: Containment Pressure--High-2). This activity involves the relaxation of the required closure time for isolation valves following an Safety Injection Actuation Signal.

Summary of Evaluation

Although main feedwater and main steamline isolation functions are used in many of the FSAR Chapters 6 and 15 accident analyses, the limiting analysis was the Unit 1 mass and energy release calculation done for the Unit 1 containment response to a steamline break. That calculation has been superseded by an analysis in which main feedwater and main steam line isolation is completed in greater than 7 seconds. Thus, it is permissible to relax the isolation time requirement without introducing any unreviewed safety questions. The revised Unit 1 containment response analysis was evaluated (SE-93-121 and SE-94-082) and found to be acceptable.

Evaluation Number
SE 95-029
Revision 0

Activity Title:

Use of AS-Found T-Ref Program for Unit 1 Cycle 5 and Unit 2 Cycle 2 (ONE Form 93-029)

Description of Change(s):

The turbine first stage pressure was found to differ from the design value. The turbine first stage pressure is used to provide the programmed average temperature (T-Ref) which is used to control the plant's operating average temperature. The difference in turbine first stage pressures results in a T-Ref program which is higher that the design program and reaches its maximum value at approximately 94% of rated thermal power rathan than 100%.

Summary of Evaluation

The accident analyses in the FSAR Chapter 15 were reviewed for impact. Those events which are not always most limiting, with respect to the event acceptance criteria, at full power were evaluated for potential effects. The as-found T-Ref program increases the core inlet temperature from which the event is initiated, potentially resulting in a DNB penalty. However, for all cases potentially effected, it was determined that adequate DNB margin existed to absorb the penalty. Therefore, the as-found turbine first stage pressure is acceptable.

Evaluation Number
SE 95-030
Revision 0

Activity Title:
LDCRs TR-93-016, SA-95-063; DCN 9400 R0; Changes to Containment Isolation Valve Closure Time Requirements

Description of Change(s):
Valve stroke times have been baselined under the ASME IST Program. It was noted that several valves had stroke times which provided minimal margin for expected normal degradation in valve peformance before the maximum design allowable closure times would be exceeded. As a result, the conatinment isolation design commitments for several valves were examined for possible increases of the allowed closure times. The review indicated that there was design margin available for some of these valves. By this activity the design documents are revised to indicate, and technically justify, the changes to the closing time requirements for specific valves. The TRM is revised to document the chages inthe closing times for these valves. The FSAR is updated to reflect the chages to the TRM and the design basis documentation.

Summary of Evaluation
The containment isolation valve closing time commitments, as contained in the FSAR and the TRM, were reviewed against the guidance provided in BTP CSB 6-4 and ANSI N271. For fluid systems penetrating contaiment it was determined that the current closing times are very conservative with respect to the quidance provided in the noted documents. Further review indicated that the closing times for these valves could be relaxed to times which were still conservative with respect to ANSI N 271, while providing sufficeint margin for the expected normal degradation in valve stroke perfomrance. Such a relaxation was found to have no impact upon system performance, the ability of the valves to perform their intended safety function nor result in a significant increase to the calculated, limiting, radiological consequences of an accident. For the two valves that connect to the containment atmosphere, changing the closing time requirement of three (3) seconds to an isolation time of five (5) seconds is also consistent with the quidance of ANSI N271 and BTP CSB 6-4. The licensing basis radiological conseqences anayses were performed using a 5 second isolation time (i.e. including instrumentation delays) for these subject valves. The evaluation of the increase in the closing time for these valves indicates no adverse impact upon the performance of either the valve or the associated system and no impact upon the basis for the Technical Specifications.

Evaluation Number
SE 95-031
Revision 0

Activity Title:
PPT-TP-95B-3; SI Accumulator 2-03 Discharge Check Valve Open Test
Unit 2

Description of Change(s):
The Inservice Testing (IST) Plan requires that the Safety Injection (SI) Accumulator discharge chaeck valves (8956A,B,C,D) be periodically disassebled and inspected. This activity is a special test performed to obtain information regarding the acceptability of implementing this test methodology for satisfying the underlying IST requirement (check valve operability). This one-time test will be conducted with Unit 2 defuelled, reactor vessel head removed, the upper internals in place and the refueling cavity mostly filled. The associated SI Accumulator will be isolated and then filled (or verified filled) with 2400ppm to 2600ppm borated water and pressurized (or depressurized) to approximately 35 psig. The motor operated accumulator discharge isolation valve will be opened allowing forward flow from the accumulator to the Reactor Coolant system, opening the subject check valve and depressuring the accumulator. The check valve will be monitored to confirm proper opening.

Summary of Evaluation
The evaluation determined that several pressure/temperature transients will occur as a result of the test. The components affected by the transients are the SI Accumulator, the associated SI piping, RCS/SI nozzle and the RCS piping and components. These transients were found to either be bounded by transients already in the design basis or were evaluated and found, when examined in conjuction with the design basis transients, to be within the design requirements. Some routine precautions were identified as required for securing various equipment during the test. The radiological consequences of the test were examined and found to be localized and minimal. Since the test is not described in the FSAR, the associated snubbers will be visually inspected per TRM TR 3.1d.

Evaluation Number
SE 95-032
Revision 0

Activity Title:

Discretionary Enforcement of Tech.Spec. Act. 3.7.1.2a for Unit 1 to be in MODE 3 for Extra 72 Hrs while TDAFW Pump Return to Operable Status

Description of Change(s):

On 6/11/95 Unit 1 was manually tripped from 100% power. Upon automatic start of the TDAFW pump, the turbine driver tripped due to overspeed. The turbine driver is being repaired and will be tested for return to OPERABLE status. Tech Spec 3.7.1.2a states "With one auxiliary feedwater pump or associated flow path inoperable, restore the required auxiliary feedwater pump or associated flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours." It is anticipated that the repairs and testing of the TDAFW may not be completed within the 72 hour LCOAR. In order to test the TDAFW pump, Unit 1 must be in MODE 3. This safety evaluation is to provide justification for discretionary enforcement of Tech Spec Section 3.7.1.2, Action a, to allow Unit 1 to remain in Mode 3 for an additional 72 while the TDAFW pump is returned to OPERABLE status. This will avoid an unnecessary plant mode change and its associated thermal transients.

Summary of Evaluation

This Safety Evaluation has determined that the described activity does not involve an Unreviewed Safety Question or constitute a significant hazard consideration. The activity does not create the potential for an accident not previously analyzed in the Licensing Basis Documents, or reduce any safety margins existing the the Tech Spec bases. The additional time with the TDAFW pump INOERABLE is mitigated by the demonstrated operability of the moter driven AFW pumps and the availability of all onsite and at least two offsite ac power sources.

Evaluation Number
SE 95-034
Revision 0

Activity Title:
LDCR SA-95-075; STA-TP-95-2, Rev. 0; Plant Change Implementation by Interim Procedure /Removal of Requirement for Prior SORC Approval

Description of Change(s):
Approve and implement Rev. 0 of STA-TP-95-2 Rev.0, "Plant Modification Process". This procedure describes the new modification process that was developed by a process re-engineering team, and should be implemented to enable a new organization to begin a prototype period during which the process will be tested, evaluated, and improved. At the end of that period, that organization plans to change STA-TP-95-2 to a permanent procedure, transition all Design Modifications and Minor Modifications to the new process, and retire STA-716.

Summary of Evaluation
The new design modification process brings the design modification program in line with the actual regulatory requirements as previously described within NUREG 797 supplement 22, Regulatory Guide 1.33 revision 2, and the Technical Specifications Section 6 for CPSES. No changes to safety related equipment, facilities, components, systems or structures can take place without a review against 10CFR50.59 to determine if a "safety evaluation" per 10CFR50.59 (b)(1) must be performed. The "safety evaluation" must be concluded satisfactorily before the design modification can be implemented. Therefore, this safety evaluation has concluded that implementation of this activity will not produce any new unreviewed safety question nor does any unreviewed safety question exist.

Evaluation Number
SE 95-035
Revision 0

Activity Title:

MM 95-047, DCN -9468; LDCR SA-95-073; Modification to the Unit 2 Condensate Storage Tank

Description of Change(s):

Modify the Condensate Storage Tank (CST), CP2-AFATCS-01, to include 1) isolation of the nitrogen blanket supply lines and removal of the nitrogen bleed off line, 2) removal of the pipe supports for line AF-2-920 to eliminate sharp edges that could potentially damage the tank diaphragm, 3) addtion of a stainless steel screen cover to the CST vents to prevent entrance of birds into the tank, 4) addition of standoff strips over the penetration for the condensate makeup and reject line to support the diaphragm and prevent the diaphragm from potentially blocking the penetration, and 5) modification of the manway cover to provide connection for a blower and a viewing port.

Summary of Evaluation

The removal of the nitrogen blanket capability has no effect on the ability of the auxiliary feedwater system to perform its safety function and introduces no credible failure modes which could affect any system structure or component. The nitrogen blanket was used to aid in the control of oxygen in the CST inventory and will no longer be required when the new diaphragm is installed. The removal of the pipe and supports inside the tank above the diaphragm will eliminate sharp edges that could damage the diaphragm. The addition of the bird screens will not impact the ability of the vents to properly vent the tank during inflow and outflow conditions. The addition of the standoff strips will not adversely impact the flow characteristics of these lines. The addition of the connections and viewing port to the manway cover are designed to be missile protected. The conclusion of the evaluation is that this modification does not involve an unreviewed safety question.

Evaluation Number
SE 95-036
Revision 0

Activity Title:
MM 94-313(DCN 8341),-314 (DCN 8342), -315 (DCN 8343),-316 (DCN 8344); Interrupting Air Supply to EDG Governor Booster

Description of Change(s):
This activity provides the modification information necessary to install an isolation valve upstream of the air supply to the EDG governor booster servomotor. Isolating the booster servomotor will keep the fuel rack from going to the "full fuel" position during an engine start for the first five seconds. Industry experience has shown that by limiting this rack travel, significant benefits can be realized in reduced turbocharger thrust bearing damage, and cylinder liner and piston ring damage.

Summary of Evaluation
This activity results in the physical ability to start the EDGs in a "soft" (non full rack) manner. This action improves reliability as the decrease of damage to the turbocharger and the cylinders reduces the probability of EDG failure due to loss of turbocharger from bearing degradation and major cylinder damage from loss of lubrication/overheating/crankcase explosion. This activity also improves availability of the engines especially during outages by reducing the amount and severity of teardown maintenance required. Based upon the results of this evaluation, implementation of the proposed activity does not involve an Unreviewed Safety Question.

Evaluation Number
SE 95-037
Revision 0

Activity Title:
LDCR SA-95-076; PPT-GO-1014 R 0; Procedure to Govern Grain Thief Sampling of Activated Carbon from the Carbon Adsorber Banks

Description of Change(s):
This activity is the creation of a procedure to govern grain thief sampling of activated carbon from the carbon adsorber banks in the Control Room HVAC, Primary Plant HVAC, and Containment H2 Purge and Containment Preaccess systems. CPSES currently uses sample cannisters to obtain carbon samples for testing ESF filtration units required by Technical Specification surveillances 4.7.7.1b(2) & c and 4.7.8b(2) & c, as well as testing of activated carbon installed in nonsafety filtration systems. The number of sample cannisters installed in each filtration unit is limited. When testing is required, the cannister is removed from the filtration unit, and the activated carbon is removed from the cannister and shipped to a vendor for analysis. If no "Representative" samples are available, all activated carbon in the filtration unit must be removed and replaced. The old carbon must be disposed of as radioactive waste.
To mimimize the unnecessary generation of radioactive waste a procedure (PPT-GO-1014), will obtain "Representative" carbon samples when most sample cannisters have been expended. The new procedure will provide instructions for sampling the carbon adsorber using a grain thief sampler for laboratory tests and re-filling the sample test canister. LDCR No. SA-95-076 will be submitted to to describe obtaining representative samples with the grain thief for refilling the cannisters and laboratory testing.

Summary of Evaluation
FSAR section 1A(B) - Discussion of Regulatory Guides 1.52 & 1.140 states that laboratory testing will be done in accordance with ANSI N509-1980 and N510-1980 instead of the criteria in the Reg. Guide 1.52 & 1.140 section C.6.
For filtration units (like ours) that existed prior to the issuance of N509-1980 on Oct. 30 1980, slotted tube grain thief sampling is allowed by Appendix A of N509-1980. The BASES for both Control Room HVAC (3/4.7.7) and Primary Plant Ventilation (3/4.7.8) Tech. Specs already have the statement "ANSI N510-1980 &N509-1980 will be used as a procedural guide for surveillance testing of charcoal samples". This safety evaluation found that the National Equipment grain sampler is equivalent to the slotted tube sampler described in Appendix A of N509-1980. Equal sampling across the 4" charcoal bed with the grain thief sampler will provide 'Representative" samples for laboratory testing. In doing so, the existintg canisters may be refilled and utilized until results from the lab testing require charcoal replacement.
The activity will not create the possibility of a different type of accident not previously evaluated in the LBD's. The safety margin will not be reduced by sampling charcoal in accordance with the new porcedure PPT-GO-1014.

Evaluation Number
SE 95-038
Revision 0

Activity Title:
DM 95-009 R1;LDCR SA 95-079: Modification Related to the Spent Fuel Pool 2 Suction Piping and Spent Fuel Pool 1 and 2 Exhaust Registers

Description of Change(s):
Design Modification 94-004, Revision 1, includes the installation of high density storage racks in Spent Fuel Pool 2 (SPF2). When both spent fuel pools 1 and 2 are used to store spent fuel, the ability for the Spent Fuel Pool Cooling and Cleanup System (SF) to cool both pools with a single train cooling while the pools are isolated from each other is required under certain operating scenarios. Under these operating conditions, the potential exists to cause an imbalance in the levels of the spent fuel pools due to differences in piping friction losses. The following modifications are included in Design Modification 95-009, Revision 0, to account for this effect.

Item 1) Orifice plates are to be added to the spent fuel pool no. 2 cooling pump suction piping. The orifices are sized to balance the friction losses in the piping lines between spent fuel pool nos. 1 and 2. This will ensure that the volumes removed and returned to the spent fuel pools during the cooling process will remain equivalent for the two isolated pools and the levels will therefore remain constant. (FSAR Figure 9.1-13, Sheet 13 was revised by LDCR-SA-95-79)

Item 2) The exhaust registers are to be modified such that the eight registers on the east and west sides of each spent fuel pool are to be blanked off and drywells are to be provided for the twelve registers on the north and south sides of each spent fuel pool. This activity ensures that pool water does not overflow into the exhaust registers due to a high level in a pool and that the registers modifications will not interfere with the movement of spent fuel within the pools. (FSAR Section 9.4.2.1 was revised by LDCR-SA-95-79)

Summary of Evaluation
The impact to the piping due to the weight added by the orifice plates has been evaluated and found acceptable. The function of the ventilation system will not be adversely impacted due to the exhaust register modifications. The exhaust airflow will be balanced to provide the same total air flow as before. This will assure that Tritium airborne concentrations, if present, will be adequately exhausted to the primary plant ventilation system. Also the Fuel Building room temperature has been evaluated to ensure the requirements are satisfied.

Installation activities were evaluated to assure that the requirements for 1) heavy loads (NUREG 0612 and Tech Spec Surveillance 4.9.7), 2) potential load drops (Tech Spec 3/4.9.7 Bases), 3) spent fuel pool water level (Tech Spec 3/4.9.10), and ALARA guidelines/procedures remain satisfied.

The evaluation has determined that the activity does not involve an Unreviewed Safety Question, does not result in a test or experiment not described in the Licensing Basis documents, does not change the procedures as described in the

Evaluation Number
SE 95-038 (continued)
Revision 0

Activity Title:
DM 95-009 R1:LDCR SA 95-079: Modification Related to the Spent Fuel Pool 2 Suction Piping and Spent Fuel Pool 1 and 2 Exhaust Registers

Description of Change(s): (continued)
Licensing Basis Documents, and does not involve a change to the Technical Specifications. The Activity does not increase the probability or consequences of accidents evaluated in the Licensing Basis Documents or create the Potential for an accident not previously analyzed in the Licensing Basis Documents, or reduce any safety margins existing in Technical Specification bases. This activity does not affect any system used for accident mitigation, will not affect plant impact or response to a system failure, and will meet all system design basis requirements.

Evaluation Number
SE 95-039
Revision 0

Activity Title:
TM 1-95-009:Temp. Mod. to Connect Hose from Valve XFP-078 at Discharge of the Chem. Recirc. Pump to Keep Fire Protection System Full of Water

Description of Change(s):
This temporarary modification (TM) is for connecting a hose from valve XFP-0788 at the discharge of the Chemical Recirculation Pump CPX-FPAPCR-01 to valve XFP-0791 in the Jockey Pump CPX-FPAPFP-07 discharge line. This allows the recirculation pump to supply partial pressure of about 80 psig to the fire protection system to keep the system filled with water. This modification became necessary because the fire pumps and jockey pump have to be out of service for maintenance on electric fire pump discharge relief valve XFP-0329. Also valve XFP-0329 is not isolatable from the discharge relief valves of other fire pumps and jockey pump. This temporary modification ensures maintaining system water pressure to keep the system filled.

For compensatory measure backup fire pumps are the Emergency Refill Pump CPX-FPAPFP-01 located in the Service Water Intake Structure (SWIS) and two fire trucks connected to the truck connection near the SWIS or the Administration Annex Building taking suction from the service water discharge canal. These fire pumps can be manually started and connected to the system if needed. Also this temporary modification is in service with the valves open for only short duration when valve XFP-0329 is being removed and replaced. A blind flange is also installed while valve XFP-0329 is removed, so that the fire pumps and jockey pump can be restored to service. Additionally personnel are made available for response to operate pump CPX-FPAPFP-01 and the fire trucks during the time the TM is in service.

Summary of Evaluation
The TM allows the fire protection system to remain functional during a time when the permanent pressure maintenance (jockey) pump and the permanent fire pumps are not available. Compensatory measures in the way of alternate fire pump capability is provided. Personnel are available to start the alternate fire pumps if needed to ensure a quick response to a fire condition. This response is expected to be quick enough to prevent any significant drain down of the system, should a sprinkler system or deluge system operate and cause loss of system pressure, thus preventing water hammer damage when the fire pumps come on line. The loss of permanent pump capacity and the TM in-service time is expected to be of short duration. The compensatory measures to provide backup fire pump capability and pressure maintenance during the short time the pumps are out of service and the absence of any adverse impact on fire safe shutdown capability, support the conclusion that an unreviewed safety question does not result from this temporary modification.

Evaluation Number
SE 95-041
Revision 0

Activity Title:
DM 95-052, -053: Installation of Level Instrumentation on Steam Supply and Exhaust Drain Lines of Turbine Driven AFW Pump turbine

Description of Change(s):
Installs level sensing instrumentation on the steam supply and exhaust lines for the Turbine Driven Auxiliary Feedwater Pump turbine. This will consist of level switches which will alarm in the control room if excessive condensation is detected in either the supply or the exhaust lines. The switches inititate no automatic actions except the alarm. The level switches are not safety-related and are not required for TDAFW Pump turbine operability. The supply side switch will be connected to existing instrument taps on the 1/2 MS-27 drain pot. The exhaust side switch will require penetration into the exhaust drain line.

Summary of Evaluation
The level switches will enhance the operability and reliability of the system by alerting the operator to a potential problem if excessive condensate accumulates in either of the drain lines. The switches will not affect the normal operation of the Auxiliary Feedwater Pump turbine or Main Steam System because they do not initiate any actions other than the alarm. The Safety Evauation concludes that the implementation of these activities does not involve an Unreviewed Safety Question.

Evaluation Number
SE 95-043
Revision 0

Activity Title:
DM 91-094 Rev. 0;LDCRs SA-95-095. OD-95-004; Pump Disch. Flow Capacity & Flow Indication Accuracy Increase; Deletion of Inoperable I&C

Description of Change(s):
Discharge capacity of Waste Water Holdup Tank Transfer Pumps are increased by addition of new pump impeller and motor. Electrical components are changed due to a larger motor size. New flow measurement equipment is installed to provide accurate measurement of flow being discharged. A ball valve is installed to allow isolation of the Reagent Bulk Feeder. Unsed pH and Oil-in-Water instrumentation, which has not functioned correctly, is removed from the plant and permissive interlocks in control logic of Waste Water Holdup Tank Transfer Pump Discharge Valve and Oil/Water Separator Outlet Valve.

Summary of Evaluation
Discharge flow indications will provide an accurate indication of flow, required for ODCM calculations. Increase in impeller and motor size will aid in increasing flow rates from the pumps. Flow rates will increase a minimum of 15% to 25% therby reducing required run time. Addition of isloation valve at the base of the eductor aids in controlling chemical feed to the eductor.

Evaluation Number
SE 95-045
Revision 0

Activity Title:
DM 95-012: LDCR SA 95-090;Electric. Pwr Cable Installation Between Alt .Pwr.Transfer Switch and Unit 2 trains A/B of 6.9 KV Class 1E SWGR

Description of Change(s):
The modification activities involve (1) the installation of electrical power cables between the alternate power transfer switch and the train A and train B Class 1E. 6.9 KV sitchgear. (2) the replacement of 150/5 A current transformers with 1000/5 A Class 1E current transformers in the train A and train B spare 6.9 KV switchgear cubicles to be used for the alternate power diesel generator (APDG) and (3) the shorting of over current and ground sensor current transformers in these switchgear cubicles.

The new feeder cables from the alternate power transfer switch are landed on the load side of the existing spare circuit breaker in the train A and train B of 6.9 KV Class 1E switchgear. These cables are used for providing power to the Class 1E buses in the event of loss of all onsite and offsite AC power during MODES 5 and 6 of plant operation. These circuit breakers are normally racked out and disengaged during MODES 1 through 4. The new feeder cables are classified as "Associated" and remain de-energized in MODES 1 through 4 and are to be installed in new safety related seismically installed conduits and existing safety related seismically installed cable trays inside the building. These cables are installed outside the Unit 2 Safeguards building in the non-seismically installed conduits and terminate at the alternate power transfer switch.

Summary of Evaluation
The impact of concrete block out work on the barrier function (fire. missile. security. HVAC. radiation) was reviewed for Safeguards building north exterior wall (15 S). Safeguards building room 103 roof and the wall (BS) between the train A switchgear room 83 and hallway room 82. It was determined that the barrier breach for the time necessary to install the conduit sleeve and cables is acceptable. The impact of installing the seismic scaffolding over the train B switchgear for the time necessary to install the conduit was also reviewed and found acceptable. The installation of the cables in the existing safety related cable trays is acceptable with respect to ampacity. combustible loading, and the existing seismic qualification of the cable trays envelope the additional loading resulting from the new cables. Installation of the cable connection into the 6.9 KV switchgears has been evaluated and determined not to compromise electrical separation or operability of either train busses.

The impact of disabling the protection in the spare circuit breaker cubicles (to be used for APDG) was reviewed and it was found acceptable to solely use the APDG protection and manually load limit the APDG to 3500 KW. Since the APDG's short circuit current capability is less than the continuous rating of the 6.9 KV switchgear. it makes the 6.9 KV switchgears inherently protected from APDG fault currents. The impact of the temporary installation of the alternate power diesel generators outside the unit 2 Safeguards building (west side). their testing and operation in MODES 5 and 6 was also reviewed and determined to be acceptable.

Evaluation Number
SE 95-045 **(continued)**
Revision 0

Activity Title:

DM 95-012: LDCR SA 95-090:Electric Pwr Cable Installation Between Alt Pwr Transfer Switch and Unit 2 trains A/B of 6.9 KV Class 1E SWGR

Description of Change(s): (continued)

Since the APDGs are non-Class 1E equipment, plant security is not required to protect them as vital equipment.

The evaluation concludes that it is acceptable to install the alternate power feed tie-ins to the 6.9 KV switchgears. Based on the evaluation, implementation of the activity does not result in an unreviewed safety question.

Evaluation Number
SE 95-046
Revision 0

Activity Title:

DM 94-047, LDCR SA-95-094:Modification of the Fire Suppression Systems to Filteration Units to Preclude Water from Spuriously Entering Filter

Description of Change(s):

Spurious water discharge through the automatic deluge valves into the atmospheric cleanup units (ACUs) during normal maintenance activities, has repeatedly damaged the charcoal adsorber beds of the units, causing excessive interference with the normal operation of the units and added economic burdens of charcoal replacement. The proposed activities involve modifying the existing water-spray fire suppression systems protecting the primary plant exhaust ventilation units, hydrogen purge exhaust ventilation units, and control room ventilation and pressurization units from automatic to manual operation and revising the associated changes in system descriptions in the Final Safety Analysis Report (FSAR).

Summary of Evaluation

Buildup of heat in the ACU charcoal adsorber beds from gamma radiation is a gradual, long term process compared to operator response times to initiate manual fire suppression. Worst case radioactive iodine adsorbtion in a charcoal bed, does not result in a temperature rise above the auto-ignition temperature of the charcoal. The ACU units are robust in construction and are capapable of containing a fire, should it occur. Fire suppression is for mitigation of damage inside the unit, and it does help mitigate release of radioisotopes from the charcoal during fire conditions. Advance warning of excessive temperature in a unit is provided to the operators prior to a potential fire condition. Therefore, manual operation of the fire suppression systems for the ACUs is an acceptable alternative to automatic actuation. The proposed activities do not adversely affect the ability to safely shut down the plant under fire conditions or increase the potential for, or consequences of, a radiological release to the environment. The General Design Criteria (GDC) 3 and BTP APCSB 9.5.1 are satisfied by this modification. Based on the evaluation, the implementation of the proposed activities do not involve unreviewed safety question.

Evaluation Number
SE 95-047
Revision 0

Activity Title:
LDCR SA 95-086; DCN-9727; Clarification of Post Accident Sampling System (PASS) Functions of Certain Containment Isolation Valves

Description of Change(s):
FSAR Tables 6.2.4-3 and 6 are revised to show the instrument air supply to containment may be opened to support PASS functions, that the reactor coolant sample from the hot legs is a PASS function, that the PASS reactor coolant sample return to containment and the containment air PASS sample and return penetrations are "nonessential" in accordance with the criteria in Section 6.2.4 and NUREG-0737. FSAR Section II.B.3.3.3 is revised to show the instrument air function, to show isolation signal override is per Section 7.3.1.1.4 and that the function does not constitute opening of an isolated auxiliary system. The DBD-ME-013 and DBD-ME-228-01 changes coincide with these clarifications and document the resolution of ONE 95-000756.

Summary of Evaluation
PASS is a non-safety related system function mandated by NUREG-0737 and RG 1.97 to ensure post-accident sampling can be performed without exceeding GDC-19 radiological exposures to the operator. The functional and design requirements are contained in NUREG-0737, Section II.B.3 and in RG 1.97, Rev. 2. The PASS function qualification requirement for Category 3 (nonessential) from RG 1.97 is for each valve to be "selected to withstand its service environment." The safety related containment isolation function requires full NUREG-0588 environmental qualification as required by 10CFR50.49. Therefore, a DCN to the design basis documents has been initiated in which the PASS function has been evaluated and documented. With the provisions for air to the valve actuators, the PASS reactor coolant sample and return valves have been evaluated and determined that NUREG-0737 and RG 1.97 requirements are satisfied, the Licensing Basis in the SER/SSER is not changed, and no unreviewed safety question exists.

Evaluation Number
SE 95-048
Revision 0

Activity Title:

PCN SOP-509A-R9-5: Use of Temporary Compressed Air Source as an Alternate For Instrument Air System During Compressor Modification

Description of Change(s):

The temporary compressed air source(s) installed in the Unit 1 Turbine Building Crane Bay will provide an alternate source of compressed air for the Instrument Air System while Design Modifications 94-0022 and 94-0023 are being installed. Design Modifications 94-0022 and 94-0023 will remove the existing Unit 1 Ingersoll Rand, 366 SCFM Compressor (CP1-CICACO-01), and its respective Air Dryer (CP1-CIDYIA-01); and Unit 1 Atlas Copco, 435 SCFM Compressor (CP1-CICACO-02), and its respective Air Dryer (CP1-CIDYIA-02). Each will be replaced with a new Atlas-Copco, 650 SCFM Compressors and Pneumatic Air Products Dryer.

Summary of Evaluation

The addition and use of the temporary compressed air source (s) during the implementation of Design Modifications 94-0022 and 94-0023 is within the prescribed CPSES design and licensing bases criteria. Between CPX - CICACO - 01, CPX - CICACO - 02 and the Temporary Compressed Air Source (s), the continued operation of CPSES Unit 1 (and Common loads) can be supported with no impact to plant safety.

Evaluation Number
SE 95-048
Revision 1

Activity Title:

PCN SOP-509A-R9-2: Use of Temporary Compressed Air Source as an Alternate For Instrument Air System During Compressor Modification

Description of Change(s):

The temporary compressed air source (s) installed in the Unit 1 Turbine Building Crane Bay will provide an alternate source of compressed air for the Instrument Air System while Design Modifications 94-0022 and 94-0023 are being installed. Design Modifications 94-0022 and 94-0023 will remove the existing Unit 1 Ingersoll Rand. 366 SCFM Compressor (CP1-CICACO-01). and its respective Air Dryer (CP1-CIDYIA-01): and Unit 1 Atlas Copco. 435 SCFM Compressor (CP1-CICACO-02). and its respective Air Dryer (CP1-CIDYIA-02). Each will be replaced with a new Atlas-Copco. 650 SCFM Compressors and Pneumatic Air Products Dryer.

Summary of Evaluation

The addition and use of the temporary compressed air source (s) during the implementation of Design Modifications 94-0022 and 94 - 0023 is within the prescribed CPSES design and licensing bases criteria. Between CPX-CICACO-01. CPX-CICACO-02 and the Temporary Compressed Air Source (s). the continued operation of CPSES Unit 1 (and Common loads) can be supported with no impact to plant safety.

Evaluation Number
SE 95-049
Revision 0

Activity Title:

LDCR SA-95-103: Deletion of "DIESEL GENERATOR BREAKER CR-HSP SELECTOR SWITCH IN HSP POSITION" from FSAR Section 8.3.1.2.1.12

Description of Change(s):

This activity involves revising the Final Safety Analysis Report (FSAR) to delete "Diesel Generator Breaker CR-HSP Selector Switch in HSP Position" (ietm 6), as one of the conditions for which Diesel Generator (DG) power window on the SSII panel is to be activated.

Summary of Evaluation

The DG breaker CR-HSP selector switch is only utilized in Control Room fire scenario to transfer breaker control to HSP and isolate its control circuits from Control Room. In this scenario, Control Room is vaccated and therefore, by-pass indication on SSII panel is of no consequence. The probability of fire scenario in the Control Room is negligible. Alignment of this switch to "CR POSITION" is always maintained. During the monthly surveillance per operating procedure OPT-108A & B, the operator(s) visually verify the position of this selector switch and do not physically manipulate it. Based on this, it is concluded that this input to SSII is not required and this activity does not create any safety concern for plant operation. Also the FSAR change does not create any Licensing Basis Document (LBD) accident not previously analysed and does not decrease the margin of safety as defined in the Technical Specifications.

Based on this evaluation, there is no unreviewed safety question associated with this activity.

Evaluation Number
SE 95-050
Revision 0

Activity Title:

DM 95-046: Installation of CVCS Demineralizer Bypass Line & Isolation Valve to Provide Additional Flexibility for CVCS Letdown Flow Cleanup

Description of Change(s):

This DM installs a bypass line and isolation valve around the Mixed Bed Demineralizers TCX-CSDMMB-01 and -02. This bypass line will provide a direct full flow pathway to the Boron Thermal Regeneration System (BTRS) Demeineralizers TCX-TRDMTH-01, -02, -03, -04, and -05. This modification will provide additional flexibility for cleanup of CVCS letdown flow during periods of Reactor Coolant System (RCS) contamination. Flexibility will exist to allow alignment through combinations of mixed bed, cation bed, and BTRS demineralizers. The BTRS demineralizers may be loaded specifically to perform cleanup of CVCS letdown flow. The CVCS and BTRS functional requirement and capability to provide boration and dilution of the RCS will not be affected by this modification.

Summary of Evaluation

This modification will provide additional flexibility for cleanup of CVCS letdown flow. The function of the CVCS to maintain the reactor subcritical through boration to compensate for xenon decay and burnout is not affected. Boration in conjunction with the BTRS to accomodate design plant load follow will continue to be available. This modification does not introduce or increase the potential for inadvertent dilution of the RCS. There are no potential failure modes introduced by implementation of the modification. The modification has no effect on accidents and malfunctions of equipment evaluated in Licensing Basis Documents and does not create potential for a new type of unanalyzed event. This activity does not affect any system used for accident mitigation, will not affect plant impact or response to a system failure, and will meet all system design basis requirements. There are no applicable Technical Specifications for the affected components. No plant commitments are affected by this activity. This activity has no adverse impact on the daily plant operations or outage activities. All original design and safety criteria are met. Therefore, this modification does not involve an Unreviewed Safety Question.

Evaluation Number
SE 95-052
Revision 0

Activity Title:

LDCR SA- 95-110; DCN 5482: FSAR/DBD/Plant Drawing Revision for Accepting the Lack of Separat. Between 1E and Non 1E EDG RTD Circuits

Description of Change(s):

Non-Class 1E RTDs located in the Emergency Diesel Generator (EDG) stator are connected to Class 1E circuits without an isolation device. The affected plant drawings are revised to clarify the safety classification of the RTDs. Design Basis Document DBD-EE-057 is revised to provide an analysis indicating the acceptability of the installation.

Summary of Evaluation

The EDG stator RTDs do not perform a safety related function, and therefore are classified as non-Class 1E. The diesel generator vendor has evaluated the various failure modes and determined that these RTDs have no effect on the safety function of the generator.

RTDs are passive devices. The only credible failure modes for the RTDs are an open circuit or a short circuit. These failure modes can cause false temperature indication. However, the temperature indications derived from these RTDs do not have a safety function. Consequently, any incorrect temperature indications derived will will not have an adverse effect on safety.

Since the RTDs are passive devices, an open circuit does not result in any unusual voltage or current in the circuit. Consequently, an open circuit at an RTD does not adversely affect the Class 1E circuit components.

Because the RTDs are connected to high impedence electronic devices, short circuits at the RTDs produce very small short circuit currents. These small currents do not adversely affect the performance of the Class 1E circuit. Based upon the results of this evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 95-054
Revision 0

Activity Title:

DM 95-010; LDCRs SA 95-108, 95-117; Providing Spent Fuel Pool Cooling Pump Protection on Low-Low Spent Fuel Pool Level

Description of Change(s):

The existing level instrumentation for the spent fuel pools provide a high and low pool level alarm. These instruments will be replaced with new level instruments that will retain the function of high and low level alarms and add the safety related function of providing pump protection for low-low pool level. This will provide pump protection if a failure or a malfunction of equipment in the SFP caused a loss of pool inventory. In addition, if a single pool is aligned to both trains of SF cooling, as may occur under some operating scenarios when both pools are used to store spent fuel, the low-low pool level trip will eliminate the common mode failure of both trains of SFP cooling. (Design Modification 95-10, LDCRs SA-95-108, SA-95-117)

Summary of Evaluation

The new instrumentation will be in two separate devices. A sensor which will be installed in the same location as the existing level switch and an electronics housing which contains the terminals and relays for the switch functions and will be installed in the Spent Fuel Pool cooling pump rooms. Class 1E and non 1E cables terminate on the same terminal block inside the electronic boxes. Non 1E cables are rated at 600 V ac and are fire retardant and the associated terminal block is heavy duty type with adequate rating. The devices inside the electronic boxes are seismically installed. The cables in the vicinity of annunciator cable terminations are trained to terminate on particular terminals. These cables are not expected to get loose and cause hot short at the annunciator cable terminals. These changes will not degrade the operation of the Class 1E boxes.

The Hi and Lo alarm setpoints have not changed. Only a pump trip is added for equipment protection and this will not jeopardize the 23' requirement for pool inventory. This modification provides additional assurance that failures or malfunctions of equipment will not prevent the ability to recover and provide at least one train of cooling to both spent fuel pools.

This modification does not introduce or increase the potential for inadvertent loss of Spent Fuel Pool cooling. There are no potential failure modes introduced by implementation of the modification. The modification has no effect on accidents and malfunctions of equipment evaluated in Licensing Basis Documents and does not create the potential for a new type of unanalyzed eevent. This activity does not affect any system used for accident mitigation, will not affect the plant or response to a system failure, and will meet all system design basis requirements. There are no affected Technical specificatons for the activty. This activity has no adverse impact on the daily plant opertions or outage acitivities. All orignal design and safety criteria are met. Therfore, this modification does not involve an Unreviewed Safety Question.

Evaluation Number
SE 95-055
Revision 0

Activity Title:
TM 2-95-011; Temp. Mod. to Support the Installation of RPS Testing in Bypass/ Power Down of All RPS Racks & N-16 Cabinets in MODES 5 & 6

Description of Change(s):
Temporary modification to monitor certain instrumentation required to perform surveillances required by tech specs during modes 5 & 6 for unit 2 while DM 93-096 is implemented.

Summary of Evaluation
Temp mod provides capability to monitor para ters during outage. it allows compliance w/ tech specs. This does not result in failure which increases consequences of malfunction of system or equipment inmportant to safety. Margin of safety is maintained.

Evaluation Number
SE 95-056
Revision 0

Activity Title:
MM 95-216. DCN 9761; LDCR SA-95-122: Increase of "fast" speed of Fuel Handling Bridge Crane from Controller Programmed Current Max. of 33FPM

Description of Change(s):
Increase the "fast" speed of the Fuel Handling Bridge Crane from controller programmed current maximum of 33 FPM to new programmed maximum speed of 40 FPM and evaluate any impact this speed increase may have on safety related equipment. (LDCR-SA-95-122: MM 95-216: DCN 9761)

Summary of Evaluation
The Fuel Handling Bridge Crane is used for handling fuel assemblies within the spent fuel pools, refueling canal, and wet cask pit. Increasing the "fast" speed from the current programmed maximum of 33 FPM to the new programmed maximum of 40 FPM will cause the fuel assembly to travel with more tilt and will put more force on the top nozzle and fuel assembly. 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 will be slightly higher than 40 FPM. A review was conducted and the conclusion reached that the increased tilt (and resultant swing if the fuel assembly is brought to rest) will not pose any threat to fuel integrity. Review also concluded that the increased hydraulic loading on the fuel assembly components will be negligible in comparison to fuel assembly handling load allowances. It was concluded that the increased speed will not introduce any significant increase in forces experienced by the top nozzles, fuel assemblies, or the fuel handling tool which could lead to fuel drop accidents. The design basis fuel handling accident as described in FSAR Section 15.7.4 remains bounding regardless of fuel assembly speed. It was also concluded that any moment force experienced by the fuel assembly structure or top nozzle would be insignificant and thus would not be expected to have an adverse effect on the fuel assembly. The bridge stops are modified to stop the bridge crane at 40 FPM as required. The increase in speed does not represent an unreviewed safety question nor result in a change to the Technical Specifications. The evaluations and conclusions reached are applicable for both Westinghouse and Siemens supplied fuel assemblies.

Evaluation Number
SE 95-057
Revision 0

Activity Title:

LDCR SA-95-121; Revision of Criticality Safety Evaluation for New Fuel Storage Mechanical Damage Incidents

Description of Change(s):

The FSAR 9.1.1.3 criticality safety evaluation for mechanical damage incidents involving the new fuel storage racks is to be revised. The specific licensing basis limit for maximum fuel assembly drop height onto the new fuel storage racks (3.5 feet) is to be removed, because the revised criticality evaluation does not rely on this limit.

Summary of Evaluation

Because new fuel is normally stored in the new fuel storage racks under dry conditions, it is not necessary to evaluate mechanical damage incidents coincident with wet, moderated conditions. Fuel with an enrichment of 5 w/o U235 or less can not be made critical under dry conditions, even considering any possible accidental damage or deformation. Thus it can be concluded that mechanical damage incidents can not result in an unsafe new fuel storage geometry with respect to criticality considerations.

The proposed FSAR change removes the description of specific limits for maximum fuel assembly drop height, maximum fuel handling bridge crane uplift forces, and administrative restrictions for heavy load handling from Section 9.1.1.3, since these specific limits are not required for the conclusion that accidental criticality is precluded.

It should be noted that separate descriptions of the design features which prevent mechanical damage to fuel assemblies in the new fuel storage racks are provided in FSAR Section 9.1.1.1 and 9.1.1.2. This activity does not modify the descriptions provided in those sections. Specifically, the maximum crane uplift force limit (5000 lbs) and administrative restrictions for heavy load handling that are to be removed from FSAR 9.1.1.3 are still described in FSAR 9.1.1.1 and 9.1.1.2. The maximum fuel assembly drop height limit (3.5 ft above the racks) is not described elsewhere in the FSAR. Thus, this activity (FSAR 9.1.1.3 change) effectively removes only the fuel assembly drop height limit from the CPSES licensing basis.

Because the criticality design basis for fuel storage and handling remains satisfied, there is no increase in the radiological consequences of any accident or malfunction, nor is there any decrease in the margin of safety.

Evaluation Number
SE 96-001
Revision 0

Activity Title:

Justification for Continued Operation while Wide Range RCS Hot Leg Temperature for LOOP 1 is Inoperable per Tech. Spec. 3.3.3.2.1

Description of Change(s):

Proposed activity allows continued operation of Unit 2 while the wide range RCS hot leg temperature indication for RCS Loop 1 at the remote shutdown panel is inoperable.

Summary of Evaluation

Unavailability of Loop 1 Thot WR at the remote shutdown panel cannot be an initiating event for any of the accidents evaluated in the SAR thus the probability of occurence of these accidents is not increased, nor is the possibility of an accident not previously analyzed in the SAR increased. The Remote shutdown panel has adequate instrumentation (without Loop 1 Thot WR) to affect a remote shutdown and cooldown, the consequences of an accident evaluated in the SAR are not increased. Loop 1 Thot WR at the remote shutdown panel has no effect on acceptance limits for operation of the plant thus the margin of safety is unaffected.

Evaluation Number
SE 96-002
Revision 0

Activity Title:

LDCR SA-96-005,DBD-EE-043 DCN-9903;Design Basis Document Change to Allow Removal of a Mech. Interlock for the Instr. Pnl. Incoming Brkrs

Description of Change(s):

Design Change Notice (DCN)-9903 is initiated to describe the removal of the mechanical interlock (for the incoming breakers) with administrative controls for instrument panels 1EC1, 2EC1, 1EC2, 2EC2, 1EC5, 2EC5, 1EC6 and 2EC6. Also the Licensing Document Change Request is initiated to make similar description in the Final Safety Analysis Report (FSAR). This activity facilitate maintaining power to the loads served by these panels during periods when the panels' inverter power supply is not available.

Summary of Evaluation

The described activity does not introduce a credible failure mode. There is no effect on accidents or malfunctions or potential for creating a new type of event. The margin of safety is not changed.

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

Evaluation Number
SE 96-003
Revision 0

Activity Title:
DCN 9920 R/0 DBD ME-250; SOP-112A/B; Revision of Documents to Incorp. Use of Vaccum Refill System for the Reactor Coolant System

Description of Change(s):
In order to reduce the effort it takes to fill and vent the RCS, it is desired to remove most of the air from the RCS with a vacuum process prior to filling. This removes most of the air from the steam generator tubes which have no convenient vent path. By reducing the amount of air in the RCS, the venting requirements are reduced and the fill process is streamlined. This activity establishes a design basis and a procedure for evacuating the air from the RCS prior to filling.

Summary of Evaluation
This activity evaluates the effects of a vacuum on the components which are required to maintain the RCS pressure boundary during RCS fill and vent process. This activity specifically addresses the structural integrity of affected components, instrument reliability and operation, and equipment operation and physical parameters. This evaluation found that all affected components, with the exception of some instrumentation, can withstand the vacuum pressures involved with this activity. The evaluation determined that the limitations with the instrumentation will not affect the ability of the instrument to maintain the RCS pressure boundary. However, the performance of the instrument may be affected, unless the instrument is isolated, and will require calibration and other maintenance procedures to be performed to ensure proper operability of the instrument. No credible failure modes were identified by the evaluation. Based upon the results of this evaluation, implementation of the proposed activity does not involve an unreviewed safety question.

Evaluation Number
SE 96-004
Revision 0

Activity Title:

LDCR SA-96-006; CPSES Unit 2, Cycle 3 Core Configuration

Description of Change(s):

Revise FSAR appendix 4B to provide an up-to-date description of the Unit 2 core configuration for Cycle 3.

Summary of Evaluation

The Cycle 3 core configuration introduces 96 Siemens fuel assemblies to co-reside with the Westinghouse Optimized Fuel Assemblies. The Siemens fuel assemblies are of the same general mechanical design as that used for the Cycle 2 core configuration. The analyses of the mechanical characteristics of the Siemens Power Corporation (SPC) fuel assemblies and burnable absorbers are performed by SPC using SPC methodology, and Westinghouse methodology is still used to confirm the acceptability of the mechanical design characteristics of the Westinghouse fuel assemblies.

The cycle-specific reactor physics, core thermal-hydraulics, and Large Break LOCA safety analyses are performed using the same TU Electric analysis methodologies that are used for the Unit 1 analyses. The Small Break LOCA safety analysis continues to be based on the Westinghouse NOTRUMP analysis, the same as for Cycle 2.

The non-LOCA system transient analyses, performed by Westinghouse using Westinghouse methodologies for Cycle 2, are performed using TU Electric analytical methodologies for Cycle 3. All of the analytical methodologies used to support the Cycle 3 core configuration have been approved by the NRC.

The Cycle 3 core configuration has been evaluated for its safety impact on the plant and TU Electric has determined that all FSAR Chapter 15 acceptance criteria are satisfied and the conclusions of FSAR Chapter 15 remain valid.

Evaluation Number
SE 96-005
Revision 0

Activity Title:

DM 96-001; LDCRs SA-96-003, -009: Main Feed Pump Turbine Governor Digital Upgrade

Description of Change(s):

This modification replaces the existing electronic governor for each pump with a shared digital governor. The replacement is an acceptable substitute; however, it is not functionally or physcially interchangable with the original and is a modification to the plant. This modification consists of adding an air conditioned room, removing the existing electronic governor from the Steam Generator Feedwater Pump Turbine (SGFPT) skid and replacing the existing electronic governor for each pump with a shared digital governor Mark V system located in the new room. Additional sensors are added to provide a 2/3 logic for the new digital governor. SGFPT sensors will have time delays provided by the Mark V. This modificaiton to the SGFPT is desirable for trip reduction. SGFPT sensors will be routed through the Mark V with exception of the trip solenoid. The trip solenoid and overspeed trip features are unaffected. The Mark V will have a different power supply than the original electronic governors. The existing manual control in the main control room will be replaced with a 5 position switch. The operator will now answer control board trouble alarms using one of the two interface computers located in the new room in the turbine building and the Unit 2 main control room computer room. The design addressed applicable concerns specific to the new digital electronics that could result in failure modes and system malfunctions that were not considered during the initial plant design. These include potential common mode failures due to (1) the use of common software in redundant channels and (2) increased sensitivity to the effects of electromagnetic interference.

Summary of Evaluation

This modification is being made to reduce the probability of a malfunction of the feedwater pumps. Therefore, the probability of a malfunction or an accident is decreased. The modification does not create the possibiltiy of a new failure mode, since the accident analysis bounds the failure modes of the new design. The accident analyses bounds the failure modes of the new design. The accident analyses remain bounding for this change to the facility. There are no Technical Specifications affected and there is no reduction in the margin of safety. Therefore, the modificaiton does not constitute an unreviewed safety question.

Evaluation Number
SE 96-008
Revision 0

Activity Title:

TM 2-96-003:Containment Penetration 2-MII-0009 Modification for Local Leak Rate Test(LLRT) Air-lines & Telecommunication Cables for 2RF02

Description of Change(s):

Blind flanges from two of three spare pipes in penetration 2-MII-0009 will be removed and LLRT tubing and telecom cables will be passed through while in MODE 5. Prior to MODE 6 applicability of Tech Spec 3.9.4, the pipes will be sealed with a suitable penetration seal which will ensure the containment atmosphere is isolated from the outside atmosphere. The tubing will be isolated in accordance with TS 3.9.4 during core alteration and fuel movement inside containment.

Summary of Evaluation

There are no credible failures introduced by this temporary modification to the containment penetration as described in the FSAR. Technical Specification 3.9.4 is to be satisfied while this mod is in place. There is no reduction in the margin of safety because the containment closure function is to be maintained during core alterations and movement of irradiated fuel during Mode 6. Based on the evaluation, implementation of the activity does not result in an unreviewed safety question.

Evaluation Number
SE 96-009
Revision 0

Activity Title:
PCN RFO-207-R4-1; Visual Fuel Inspection in Spent Fuel Pool 1

Description of Change(s):
Visual inspection of discharged fuel assemblies is to be conducted in Spent Fuel Pool 1, with the use of an underwater camera. During the inspection process, fuel assemblies will be lowered and then raised in an area between the outside edge of the spent fuel pool 1 rack array and the pool wall. There is horizontal clearance between the pool wall and the adjacent racks in the inspection area of approximately 18 inches. The inspection procedure will require that no spent fuel assemblies are located in the storage cells near the inspection area; this restriction ensures that the currently analyzed fuel spacing requirements for spent fuel criticality remain bounding. However, the fuel assemblies being inspected may be temporarily positioned closer to the spent fuel pool wall than the distance previously analyzed in the wall heating evaluation.

Summary of Evaluation
The spent fuel pool wall heating analysis was evaluated for the anticipated fuel inspection configuration. Calculations have demonstrated that all applicable heating limits remain satisfied, even condidering a single fuel assembly located near the pool wall for the period to time required for the inspection. The criticality design basis for fuel storage and handling also remains satisfied because procedural controls ensure that the currently analyzed spacing requirements remain satisfied. The probability of a fuel handling accident is not increased, based on a comparison to current CPSES fuel handling operations, which includes insertion and removal of fuel from the reactor core and the spent fuel storage racks.

Because all relevant design analysis limits remain satisfied, there is no increase in the radiological consequences of any accident or malfunction, nor is there any decrease in the margin of safety.

Evaluation Number
SE 96-010
Revision 0

Activity Title:
SA 96-012; Procedure RWS-VN-111 R. 0; Use of Drum Dryer for Conditioning or Volume Reduction of Misce. Liquid Radioactive Waste

Description of Change(s):
A drum dryer system was approved for use in the Fuel Building; a brief system description is added to FSAR Section 11.2.

The drum dryer is a self-contained, skid-mounted system that may be used for conditioning or volume reduction of some miscellaneous liquid radioactive waste or sludge generated from tank cleaning or decontamination activities. The system consists of a separation tank, air-cooled heat exchanger, three pumps (supply, vacuum and recirculation), and associated piping, valves and instrumentation.

The subject miscellaneous liquid wastes may not be suited for Filter Demineralizer System (FDS) or floor drain evaporator processing. The liquid could chemically deteriorate the FDS resins. Assuming the floor drain waste evaporator could be used, the liquid would produce additional evaporator bottoms for disposal. Condensed steam from the dryer skid will be processed with other liquid radioactive wastes and the residual solids handled as Dry Active Waste.

Summary of Evaluation
Use of the drum dryer system allows preconditioning or volume reduction of the input liquid radioactive waste without increasing normal operation released radioactivity. Process Control Program requirements do not apply since the system is not intended to process liquid radioactive waste in a form that meets shipping and burial ground requirements.

During normal operation of the drum dryer, the contents of the separation tank may be drained to an appropriate floor drain or tank for routine processing. Since the potential for vapor exists in the separation tank, there is a small potential for airborne radioactivity to be released when the tank is drained; however, any release would be monitored by the plant vent stack radiation monitors per the ODCM requirements. If airborne radioactivity is released during the draining process, the release is expected to be minimal and 10CFR50, Appendix I dose calculations should not be affected.

The credible failure modes, i.e., fire, rupture of the drum or seperation tank, and equipment leak or failure were evaluated for the specified Fuel Building use location(s). The impact of operating tempertures on the concrete floor and its coatings will not affect the structural integrity of the concrete or the adequacy of the floor coating. The CPSES Liquid Radwaste Storage Tank rupture envelopes the radiological consequences that could be introduced by this activity. The ability of safety systems to perform their safety functions is not affected by this activity.

Evaluation Number
SE 96-011
Revision 0

Activity Title:
Potential Loose Parts for Unit 2 Cycle 3

Description of Change(s):
Evaluate the potential adverse effects of a loose part (a 1/2" OD carbon steel ball bearing) introduced into the RCS, or connecting systems, on the operation of Unit 2 Cycle 3.

Summary of Evaluation
The potential loose part could affect components used for accident mitigation. However, potential effects have; 1) been considered in the plant design through redundant isolation devices, 2) would be detected through alarms or Technical Specification required surveillances, or 3) have been explicitly considered in the accident analyses. Therefore, an unreviewed safety question does not exist.

Evaluation Number
SE 96-012
Revision 0

Activity Title:

LDCR PC-96-001; Revise Process Control Program (PCP) for Use of Vendors in Performing Spent Resin Dewatering

Description of Change(s):

This activity revises the Process Control Program (PCP) to add a new vendor for spent resin dewatering and provides a generic evaluation for using vendor supplied equipment in the CPSES Fuel Building for performing radioactive waste processing ,i.e., spent resin dewatering. The dewatering system consists of a means to remove water from spent resin to meet disposal limits of <1% free standing water. Resin dewatering may be provided at CPSES by Scientific Ecology Group (SEG)(includes the former Westinghouse Hittman Nuclear Incorporated), Diversified Technologies, or other vendors who provide this service and remain within the criteria specified in this safety evaluation.

Summary of Evaluation

Spent resin handling (dewatering) activities are to be performed in Fuel Building room locations where the equipment does not impact or affect the performance of plant safety systems. The impacts of fire, rupture of a High Integrity Container (HIC) liner, rupture of a spent resin transfer hose, rupture of a receiver tank (or catch drum), processing equipment failure, potential release of gases, media exothermic reactions during dewatering, and overfill of the HIC were reviewed. No potential for fire or explosion is postulated because ambient temperatures are used in the dewatering process. A generic accident analysis was performed that eveloped the spent resin handling activities. The Existing Liquid Waste Storage Tank accident was determined to envelope the radiological consequences that could be introduced by this activity as the liquid volume and radioactive concentrations in the Liquid Waste Storage Tank rupture accident are much greater than those expected in the dewatering receiver tank or catch drum. Vendor dewatering sytems are designed to comply with applicable codes, regulations and standards and therefore do not increase the probability of a Liquid Waste Storage Tank rupture or Liquid Waste System leak or failure. During normal system operation, any released gases from the dewatering system are directed to the Plant Primary Ventiliation System via a HEPA filter. Any release would be monitored by the plant vent stack radiation monitors per the ODCM requirements. If radioactive gases are released during the dewatering process, the release is expected to be minimal and 10CFR50, Appendix I dose calculations should not be affected.

Evaluation Number
SE 96-014
Revision 0

Activity Title:

LDCR SA-96-006; Reanalysis of FSAR Chapter 15.1.1 Accident Analysis to Evaluate Larger Reduction in Feedwater Temperature

Description of Change(s):

Chapter 15.1.1 of the CPSES FSAR addresses the consequences of feedwater malfunctions which result in a decrease in feedwater temperature. Currently, a decrease in feedwater temperature of 35 degrees F is assumed in the analysis performed to evaluate this event. Prior to shutdown of Unit 2 for its second refueling outage, a plant transient occurred which was initiated by the opening of the low pressure feedwater heater bypass valve followed by isolation of the high pressure extraction steam. During this transient, the feedwater temperature decreased by more that 200 degrees F. Until now, the loss of high pressure extraction steam had not been considered to be a direct consequence of the low pressure feedwater heater bypass valve opening. However, it must now be considered in the accident analysis. The FSAR Chapter 15.1.1 accident analysis has been revised to evaluate a larger feedwater temperature reduction.

Summary of Evaluation

The decrease in feedwater temperature scenario described in Chapter 15.1.1 has been reanalyzed using NRC-approved methods to consider a decrease in feedwater temperature of 246 degrees F upon opening the low pressure feedwater heater bypass valve. All applicable event acceptance criteria have been demonstrated to remain satisfied. Therefore, there is no change to the consequences of the accident as previously analyzed nor is there any reduction in the margin of safety as defined by the plant Technical Specifications.

Evaluation Number
SE 96-015
Revision 0

Activity Title:

PCN SOP-303B-R4-4; Change of Method for Controlling Gland Steam Cond. Bypass Flow Control Vlv. 2-FV-2243 from Air Operator to Manual

Description of Change(s):

Revises SOP-303B to revise method of operating valve 2-FV-2243 from using the air operator to using the manual hand wheel.

Summary of Evaluation

This acitivity revises the method of operating the Gland Steam (GS) Condenser Bypass Valve (2-FV-2243). Valve 2-FV-2243 shall be throttled into position manually and left in this position in lieu of being modulated based on flow through the GS Condenser.

Valve 2-FV-2243 is a 24-inch butterfly valve equipped with a combination air and manual operator. The actuator is an air to open/air to close piston operator equipped with a manual handwheel which may be used to take local control of the valve. Currently, the valve is controlled via a Manual /Automatic station in the Control Room (2-FK-2243) in the automatic mode to maintain 1000 GPM through the GS Condenser as read on 2-FI-2243. The valve modulates to maintain this flow through the condenser, during periods of reduced flow the valve closes to force additional flow through the condenser and in the process creates a larger pressure drop across the system. The design intent was to limit flow-induced-erosion of the GS condenser tubes and since the valve will be throttled (manually) into position the design intent of limiting the the maximum flow through the GS Condenser will still be achieved. In event of some type of valve failure, the high differential pressure alarm features will remain as originally design. This change does not involve an unreviewed Safey Question.

Evaluation Number
SE 96-017
Revision 0

Activity Title:
PCN SOP-907A-R6-2; PCN SOP-907B-R1-6; Compensatory Measures for Personnel Airlock to Address ONE Form 96-190

Description of Change(s):
Procedures were temporarily changed to restore OPERABILITY to the airlock control system by use of manual action in place of automatic action in accordance with Generic Letter 91-18. This is a change to the procedures for the airlock control as described in FSAR 6.2.4. The valve lineup is modified to add closure of hydraulic line valves in the return from the equalization valves which secures them in place.

Summary of Evaluation
Procedural steps were revised to control the position of these hydraulic valves during containment ingress and egress to ensure at least one door is both closed and OPERABLE. These measures will allow containment building ingress and egress using the existing electro-hydraulic system while ensuring the airlock can perform its safety function in the event of a LOCA or in the event of a high energy line break.

The compensatory measures in the airlock operating procedures restores the capability of each airlock to perform its safety function during containment ingress and egress. For Unit 1, securing the inner and outer equilization valves when its respective door is required to be closed and OPERABLE maintains the reliability of the airlock to that assumed in the safety analyses. For Unit 2, securing the outer equalization valve when the outer door is required closed and OPERABLE maintains the reliability of the outer airlock to that assumed in the safety analyses. The inner door reliability is not changed since its pump is tripped on s-signal and this evaluation considers electrical interlocks as backup to the s-signal in the short periods when the outer door may be open. There is no reduction in the margin of safety for the performance of this safety function.

Evaluation Number
SE 96-018
Revision 0

Activity Title:

DM 96-014;LDCR SA- 96-021;Revision to Condensate Polishing Sys. Logic and Instrumentation; Disconnection of HI D/P Across Valve 2-FV--2243

Description of Change(s):

The activity being implemented by DM 96-014 and LDCR SA-96-021 includes revising the instrumentation and the software logic to facilitate the opening of the Condensate Polishing Bypass Valve (2-PV-2242) as well as adding a signal (with 2 out of 3 logic) from the suction of the feedwater pumps to open this condensate bypass valve. The other change in this modification consists of removing the signal for opening the Emergency Low Pressure Heater Bypass Valve (2-PV-2286) from a HI differential pressure (D/P) signal across the Gland Steam Condenser Bypass Valve (2-FV-2243). The original intent for a HI D/P across valve 2-FV-2243 to open the valve 2-PV-2286 was to ensure feedwater to the feedwater pumps. It has been determined that opening of valve 2-FV-2286 causes a serious transient to the plant and it is not benefical to the plant.

Summary of Evaluation

The FSAR change has been generated to update the FSAR regarding the removal of the HI differential pressure (D/P) signal, measured across the Steam Gland Condenser Bypass Valve (2-FV-2243), and being sent to open the Emergency Low Pressure Heater Bypass Valve (2-PV-2286). The opening of valve 2-PV-2286 from an alternate signal was intended to ensure that the feedwater pumps would not trip on low suction. Valve 2-PV-2286 also gets a signal to open when there is low suction at the feedwater pumps. The previous change that was made to add the 2 out of 3 logic on this signal (HI D/P across valve 2-FV-2243) was intended to mitigate a spurious actuation of this valve and prevent an unnecessary transient. However, removing the signal in its entirety will prevent the transient from occurring without impacting the flow of water to the feedwater pumps. Part of this design modification includes opening of the Condensate Polishing bypass Valve (2-PV-2242). This valve will recieve a signal to open prior to opening valve 2-PV-2286. The opening of valve 2-PV-2242 from a 2/3 signal whenever there is low suction pressure at the feedwater pumps will reduce the chances of having valve 2-PV-2286 open as well. Thus, increasing the reliability of the feedwater system. This is an additional measure of preventing a severe transient from occurring in the secondary side of the plant. These activities do not involve an unreviewed safety question.

Evaluation Number
SE 96-019
Revision 0

Activity Title:
DM 95-073: Installation of Support Structure in Unit 2 Containment to Facilitate Removal/Reinstallation of RCP Motor in Modes 5&6/Defuelled

Description of Change(s):
This activity installs removable structural framework, rails, trolley, etc. in Unit 2 containtment in the vicinity of equipment hatch at EL 832'-6". The modification also provides temporary supports for Reactor Building slab at EL 832'-6".

Summary of Evaluation
The above described activity will be performed while the Plant is in MODES 5 or 6 or defueled. In consideration of plant status when the subject activity is perfromed, this evaluatino establishes that no credible failure mode (during MODE 5, MODE 6 or defueled) is introduced for the affected structure and systems/components required to perform their safety functino during MODE 5, MODE 6, or defueled.

The structural framework is designed to sustain gravity loads and lateral loads generated by transportation of the RCP motor. The support columns (shoring of slab at EL. 832'-6") provided to transfer loads of RCP motor to slab at EL. 808' will be decoupled at a splice once the RCP motor removal and reinstallation activity is complete and prior to the Plant entering MODE 4. The support columns configuration (decoupled at splice) during plant MODES 1 thru 4 ensures that the Reactor Building structural framework is not altered and there is no introduction of new load transfers between floors as well as within the building structural members.

The evaluation also addressed the possibiity of overturning or RCP motor under a seismic event (SSE). Based on RCP motor configuraiton (size, location of center of gravity, etc.), weight and amount of seismic input acceleration at SSE it was established that RCP motor overturning is not a credible event. Also the existing concrete beam and slab structure together with support structure being installed (column beam, rails) will no structurally fail (collapse) under seismic loads (SSE) durning RCP motor transportation. Therefore interactions with systems/components in the area at EL. 832'-6" and EL. 808'-0" are precluded.

Evaluation Number
SE 96-023
Revision 0

Activity Title:

LDCR SA-96-024; DCN 10125 R0; Revision of Cable Color Coding Requirement Within 5 Feet Intervals of Entire Cable Length

Description of Change(s):

FSAR sections 8.3.1.3-4c & -4d and 1A(B) for Regulatory Guide 1.75 position C 10 as well as the Electrical Installation & Modification Specification 2323-ES-100 are changed to indicate that the field cables requiring color coding are color coded at each end and at intervals not exceeding five feet for exposed portions.

Summary of Evaluation

The Regulatory Guide (RG) 1.75 indicates that a five feet marking distance is necessary for visual verification that cable installation meets separation criteria. The cable is routed in raceway that is labelled and checked for separation acceptability per RG 1.75. Labelling of cable at visible portions, the ends and any exposed portions is considered adequate for installation purposes. Therefore, there is no additional benefit in color coding cable enclosed in raceway every five feet. This activity makes a change to the color code label interval of the cable and does not change the physical plant or the requirements used to ensure cable separation adequacy. There are no new failure modes associated with this activity and does not adversely affect operability of safety related equipment. Based on the results of this evaluation, implementation of the activity does not involve an unreviewed safety question.

Evaluation Number
SE 96-028
Revision 0

Activity Title:
LDCR SA 96-035; DCN-10228 R0 (DBD-ME-250): Use of Steam Generator Nozzle Dams for Performing SG Maintenance During Refueling Operations

Description of Change(s):
The Design Change Notice (DCN) and the associated Licensing Document Change Request (LDCR) change CPSES as described in the Licensing Basis Documents to allow the use of steam generator nozzle dams for performing maintenance concurrent with refueling operations in MODE 6. This change in the facility and in the refueling procedures was not previously described in the Licensing Basis nor had a 50.59 evaluation been performed. ONE Form 96-158 corrective action requires that this evaluation be performed to ensure no unreviewed safety question exists.

Summary of Evaluation
Nozzle dams serve as temporary reactor coolant loop isolation valves in order to perform steam generator maintenance while refueling is proceeding concurrently. They are to be classified as Nuclear Safety Related, seismically qualified, and subject to a 10CFR50 Appendix B QA program. These measures in conjunction with the nozzle dam design characteristics provide reasonable assurance that catastrophic failure during spent fuel handling is not credible. Failures during the cavity fill mode would not affect the safety related function of any component. Their use does not create a new failure mode or increase the probability of an accident analyzed in the FSAR nor does it reduce the margin of safety.

Evaluation Number
SE 96-031
Revision 0

Activity Title:

LDCR SA-96-052;Revision of DG Load Tabls 8.3-1A,-1B & -2/Fig 8.3-8 &-9 to Reflect Compressor Modification and DG load Table Format

Description of Change(s):

This activity revises the FSAR diesel generator loading tables and associated figures to incorporate (a) the result of the new diesel generator calculation methodology which uses actual demand in lieu of name plate rating for certain loads; (b) the increase in load due to the new larger capacity Instrument Air Compressor replacement and the new configuration;
(c) consolidation of tables 8.3-1A and 8.3-1B into new table 8.3-1 and deletion of "BHP, Demand KVA, Demand KW" & individual time block summary information ; and
(d) correction of errors of tag numbers, equipment description and load group assignments including addition of Control Room Radiation monitor inadvertently omitted in the tables.

Summary of Evaluation

(a) The change in calculation methodolgy results in a more accurate representation of the actual load on the diesel generator during a given event.

(b) The new Instrument Air Compressor is larger capacity and ensures sufficient quantity of high grade clean air is available to support current plant operation and allow future growth. The increased electrical loading due to the new compressor is analyzed in the diesel generator loading calculations and is determined acceptable and is within the prescribed design licensing basis criteria (Reference SE-94-093 for DM 94-22).

(c) The reformat of diesel generator loading tables is consistent with the guidance of Regulatory Guide 1.70. The individual time block loads for all required scenarios are part of controlled calculations.

(d) The appropriate design basis documents and associated calculations have been reviewed for correction of inconsistencies or errors in the tables and is determined not to adversely impact plant safety, operation and equipment field installation.

The activity has no adverse impact on the emergency power system or the plant systems. The applicable margins of safety are not decreased and the ability of the systems to perform their intended safety function is not compromised by this activity. Based upon the results of the evaluation, implementation of the activity does not involve an unreviewed safety question.

Evaluation Number
SE 96-033
Revision 0

Activity Title:

LDCR SA-96-058: Revision of FSAR to Correct Discrepancies Regarding CCW Pump Not Being Watertight

Description of Change(s):

The FSAR Section 9.2.2.3 currently states that the CCW compartments are watertight. The evaluation concludes that there is no requirement for these compartments to be watertight, therfore reference in the FSAR Section 9.2.2.3 regarding these compartments being watertight is deleted.

Summary of Evaluation

The flooding analyses determined the maximum flood levels by making enveloping assumptions which do not require the CCW Pump rooms to have watertight doors. The ability of the required equipment to perform their safety function was evaluated with these flood levels. Therfore, the ability of safety systems to perform their safety functions is not affeted by this activity.