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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
10CFR50.59 Summary Report

Pursuant to 10CFR50.59, the attached report contains a brief description and summary of the safety evaluation of changes, tests, and experiments conducted at the South Texas Project.

Gaps in the numerical sequence of the attached Unreviewed Safety Question Evaluation summaries represent changes, tests, and experiments that have been cancelled, that were not completed prior to six months before this report, or that were summarized in a previous summary report.

If you should have any questions, please contact Mr. A. W. Harrison at (512) 972-7298 or me at (512) 972-8686.

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Attachment: Summary of Unreviewed Safety Question Evaluations

IE47/1

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Summary Of
Unreviewed Safety Question Evaluations

Unreviewed Safety Question Evaluation #92-022

Description: This change revises design assumptions in calculating radiological doses resulting from a Loss of Cooling Accident, and concerns potential leakage of containment sump water into the Refueling Water Storage Tank.

Safety Evaluation Summary: Radiological consequences described by this change are bounded by those set by 10CFR100 and the Safety Evaluation Report.

Unreviewed Safety Question Evaluation #93-021

Description: The acceptable dilution factor is changed to 1:400 from 1:1000 for liquids during post-accident sampling.

Safety Evaluation Summary: Reducing the dilution from 1:1000 to 1:400 for the liquid samples will still have the radiation exposure to the operators within 10CFR19 limits. Reducing the dilution factor does not affect equipment important to safety.

Unreviewed Safety Question Evaluation #93-031

Description: This change deletes reference to a corrosion monitoring system in the UFSAR Section 9.2.1.2.3 discussion of the Essential Cooling Water System.

Safety Evaluation Summary: The corrosion monitoring system is not a safety feature; it was added to the Essential Cooling Water System solely to monitor and quantify corrosion rates. Corrosion and fouling data collection continues to be performed.

Unreviewed Safety Question Evaluation #93-046

Description: Stainless steel wire cloth material was chosen as a replacement for carbon steel wire screen for the Essential Cooling Water System traveling screen baskets.

Safety Evaluation Summary: The replacement screen is of stronger material than the original, with an identical grid size. Consequently, there is no adverse safety impact.

Unreviewed Safety Question Evaluation #93-051

Description: This change deletes reference to the onsite sodium hypochlorite generation system. The system is to be abandoned in favor of purchasing hypochlorite from an offsite production facility. Administrative controls over residual chlorine in the condenser header during chlorine injection periods are also revised from the Environmental Report.

Safety Evaluation Summary: There is no adverse impact on plant operations since sodium hypochlorite will still be used.

Unreviewed Safety Question Evaluation #94-018

Description: This change revises the specific filter descriptions for waste holdup tank filter and spent resin sluice pump filters in Liquid Waste Management Systems in the UFSAR to provide operational flexibility by allowing the proper micron size filter to be used for the waste stream being processed.

Safety Evaluation Summary: Activity levels associated with a filter failure are enveloped by postulated accidents. The existing installed demineralizers will be used to retain particulate activity passed by the filters.

Unreviewed Safety Question Evaluation #94-030

Description: This evaluation addresses a temporary modification to permit testing of an alternate water chemistry pH control reagent for the secondary side of Unit 1. At present, ammonium hydroxide is added to the Feedwater System for pH control. Ethanolamine is to be tested and evaluated for possible use at the South Texas Project.

Safety Evaluation Summary: The alternate chemical additive will not adversely affect the post-accident radioactive release calculations. Data has been obtained by EPRI from their test program which has been reported by EPRI manual TR-102952.

Unreviewed Safety Question Evaluation #94-032

Description: This change allows limited access by members of the public to the Main Cooling Reservoir for the purpose of fishing.

Safety Evaluation Summary: Open access by the general public is specifically prohibited by UFSAR Section 11.2.3.5. No routine radiation exposure pathway is being added for members of the public. The limited access to the reservoir is restricted to areas beyond the exclusion zone, so the assumptions on the accident analyses remain valid. Dose to an individual will not exceed the current LLD of 20 pCi/kg of fish flesh.

Unreviewed Safety Question Evaluation #94-036

Description: Revision of calculations resulted in changes to the Fire Hazards Analysis Report for combustible loading values for three fire zones.

Safety Evaluation Summary: Existing conditions in the Fire Hazards Analysis Report remain valid regarding safe shutdown. Safe shutdown can be achieved following a maximum postulated fire in any of the affected zones.

Unreviewed Safety Question Evaluation #94-040

Description: This change removes the low level alarm from the Boric Acid Tanks. The intent of the change is to reduce the number of nuisance alarms in the Control Room.

Safety Evaluation Summary: The low level alarm is not required for safe operation of the system. Operators will have sufficient time to respond to avoid violating the Technical Specification limit by initiating the batching process on receipt of the low-low level alarm. Removal of the low level alarm will not impact the functional capability of the boric acid tank. The low level alarm for boric acid tanks is not required by Technical Specifications.

Unreviewed Safety Question Evaluation #94-041

Description: Changes to UFSAR Table 9.1-2 will permit use of filter elements for the Spent Fuel Pool without restriction on the filter element mean pore size. The characteristics and concentration of particulate material in the Spent Fuel Pool water will determine the filter elements to be used.

Safety Evaluation Summary: The nominal pore size radius of the filter elements will not have an impact on postulated accidents. The filter element will have no significant impact on the Spent Fuel Pool water chemistry parameters.

Unreviewed Safety Question Evaluation #94-042

Description: This change replaces the existing pair of Target Rock inline solenoid-operated valves located in each of the Bulk Sampling lines from the Steam Generators with one Copes-Vulcan air-operated valve. Additionally, the drain lines upstream and downstream of these valves have had the existing second drain valve removed and the drain line has been capped.

Safety Evaluation Summary: The change in design continues to satisfy GDC 57. No safety feature is changed by the modification. Addition of the new valves has no adverse effect on the piping and supports. These valves are normally closed and fail closed.

Unreviewed Safety Question Evaluation #94-046

Description: This change substitutes ethanolamine for ammonium hydroxide for pH control of the secondary plant.

Safety Evaluation Summary: Use of this additive does not adversely affect the post-accident radioactive release calculations or affect the availability of equipment important to safety. With this additive, pH control is established and reduces erosion/corrosion and iron transport.

Unreviewed Safety Question Evaluation #94-047

Description: This change reclassifies six valves in the Residual Heat Removal system (three valves per unit) as active, rather than passive.

Safety Evaluation Summary: Calculations confirmed that the valves will perform their safety-related active function of going to their fail-safe position and providing a flow path for cooling fluid upon demand under design-basis loading. Review of existing in-service testing requirements, spare part procurement, maintenance part replacement, and safety evaluations indicate no impact from the requested change.

Unreviewed Safety Question Evaluation #94-048

Description: This change to the Operations Quality Assurance Program Description describes the organization resulting from the Nuclear Assurance/Nuclear Licensing merger, and Technical Services realignment to Generation Support.

Safety Evaluation Summary: This organization change does not reduce any elements or responsibilities for implementation of the quality assurance program.

Unreviewed Safety Question Evaluation #94-049

Description: The responsibilities of the Technical Services Department have been reassigned to the General Manager, Generation Support, and Chemical Operations has been reassigned to Plant Operations.

Safety Evaluation Summary: This is an organization change only, with no physical effect on the plant. All required functions are reassigned in the organization.

Unreviewed Safety Question Evaluation #94-050

Description: This change deletes Main Steam Isolation Valve Above Seat drain valves and replaces them with orifices. Due to use of these restricting orifices, flow from the lines will be limited and no operator action is required to close the valves.

Safety Evaluation Summary: The amount of steam released is limited by the flow restriction orifices in the seat drain lines. Radiological consequences of a release are bounded by those given in the Safety Evaluation Report. All doses for this change are less than the acceptance criteria.

Unreviewed Safety Question Evaluation #94-052

Description: This evaluation addresses a procedural requirement to initiate an orderly plant shutdown at least two hours prior to sustained hurricane winds of 73 mph reaching the South Texas Project site. Previous plant guidelines required initiation of plant shutdown when sustained wind speeds reached the design rating of the limiting component in the offsite power supply system (120 mph).

Safety Evaluation Summary: This change will allow for orderly shutdown of the South Texas Project units and minimize the potential for plant system transients resulting from a reactor trip/turbine trip initiated by Loss of Offsite Power.

Unreviewed Safety Question Evaluation #94-053

Description: This change replaces the existing pair of Target Rock inline solenoid-operated valves located in each of the Bulk Sampling lines from the Steam Generators with one Copes-Vulcan air-operated valve. Additionally, the drain lines upstream and downstream of these valves will have the existing second drain valve removed, and the drain line will be capped.

Safety Evaluation Summary: No safety feature is changed by this modification. The valves are part of the Steam Generator Blowdown System, which is not required for safe shutdown of the plant or to mitigate the consequences of an accident. All signals which automatically closed the existing valves will automatically close the new valves.

Unreviewed Safety Question Evaluation #94-054

Description: This evaluation addresses revision of main control panel CP-004 and auxiliary shutdown panel ZLP-100 labels to clarify the letdown system orifice flow rates.

Safety Evaluation Summary: This change only enhances plant labeling. The change is considered editorial in nature and does not alter system reliability or availability.

Unreviewed Safety Question Evaluation #94-057

Description: This change increases the amount of aluminum allowed inside containment given in UFSAR Table 6.2.5-6. The increase is due to aluminum scaffold platforms and other items needed inside containment during power operation to support maintenance.

Safety Evaluation Summary: There will be no impact on safety-related equipment in case of a seismic event. One hydrogen recombiner is more than adequate to decrease the hydrogen concentration and keep it below the 4.0% limit.

Unreviewed Safety Question Evaluation #94-058

Description: This change adds a new section and table to the UFSAR to address components bought and installed in the plant utilizing the guidance in Generic Letter 89-09, but not previously indicated in the UFSAR. These are components not currently available in full compliance with the stamping and documentation requirements of the Code.

Safety Evaluation Summary: Components procured in accordance with Generic Letter 89-09 will meet all requirements of the applicable Section III of the ASME Boiler and Pressure Vessel Code except that the "N" symbol need not be applied. All systems and components are maintained in the same or equivalent configuration within the design limits.

Unreviewed Safety Question Evaluation #94-059

Description: The evaluation addresses a procedure prepared for performing fuel assembly reconstitution at the South Texas Project.

Safety Evaluation Summary: Although not specifically designed to handle and move irradiated fuel, the nuclear fuel elevator is structurally capable. No additional requirements are imposed on fuel handling equipment. The UFSAR Fuel Handling Accident analysis remains bounding.

Unreviewed Safety Question Evaluation #94-060

Description: This change consolidates Fire Areas 23 and 32 into a single fire area. The fire area boundary between the two fire areas has large open penetrations where monorails penetrate the wall.

Safety Evaluation Summary: Previous analyses assume a fire occurs in these fire areas. The same probability is assumed for a fire occurring in the combined fire area. There is no effect on equipment contained in the area.

Unreviewed Safety Question Evaluation #95-001

Description: UFSAR Table 6.2.6-1 lists systems that are not vented during Type A testing. However, draining a percentage of the listed penetrations during the test will place the containment building in more realistic design basis accident configuration.

Safety Evaluation Summary: The Integrated Leak Rate Test (Type A testing) is only performed in plant Mode 5 or when the unit is defueled. Only those trains not required for ensuring plant safety will be drained.

Unreviewed Safety Question Evaluation #95-002

Description: This change to the Operations Quality Assurance Program Description addresses the decentralization and realignment of functional responsibilities of the Engineering Programs department of Nuclear Engineering into other engineering areas.

Safety Evaluation Summary: This is an organizational change which does not reduce any elements of or responsibilities for implementation of the quality assurance program.

Unreviewed Safety Question Evaluation #95-004

Description: This evaluation addresses installation of permanent 13.8 kv switchgear adjacent to the 1k/2k switchgear located near the switchyard, and temporary installation of a diesel generator connected to the new 13.8 kv switchgear.

Safety Evaluation Summary: These changes provide a new onsite power generation source for use during reduced reactor coolant system inventory operations, if all offsite power sources are unavailable and only one standby diesel generator is operable. Associated plant power system changes, plant procedure revisions, and special test methods will have no impact on existing plant safety margins.

Unreviewed Safety Question Evaluation #95-005

Description: This change deletes listings of safe shutdown equipment in individual fire zones given in FHAR Section 3.0.

Safety Evaluation Summary: There is no change made to safe shutdown equipment or circuits. The information being removed from Section 3.0 is already given in FHAR Table 2-2.

Unreviewed Safety Question Evaluation #95-006

Description: This change increases the maximum allowable flow rate for the Auxiliary Feedwater Auto-Recirculation Valve from 120 gpm to 130 gpm.

Safety Evaluation Summary: Increasing the allowable flow rate increases the total volume required from the Auxiliary Feedwater Storage Tank from 470,600 gallons to 479,000 gallons. This is still less than the Technical Specification required minimum volume of 485,000 gallons. Failure of this valve is not an accident initiator.

Unreviewed Safety Question Evaluation # 95-007

Description: This change adds an automatic multiple rod drop test system in each of the two in-containment digital rod position indication data cabinets. The test system automatically gathers, retrieves, processes, and stores rod drop test data.

Safety Evaluation Summary: Rod drop testing is performed only when the reactor is shutdown (subcritical). There is no change in the functional design or operation of the reactor trip switchgear, rod control system, or the dropped rod position indication data cabinets.

Unreviewed Safety Question Evaluation #95-008

Description: This change revises UFSAR Section 9.1.3.2 to remove requirements for draining and filling the Fuel Handling Building fuel transfer canal and transfer them to a procedure.

Safety Evaluation Summary: The criticality analysis of the spent fuel racks assumes that unborated water in the spent fuel pool will maintain K_{eff} <0.95. However, boron concentration is maintained to ensure K_{eff} <0.95. Administrative controls are in place to minimize the potential for a dilution event. Therefore, this change has no impact on the criticality of the spent fuel pool.

Unreviewed Safety Question Evaluation #95-009

Description: This change abandons the existing boron analyzer in place. After modification, the existing ion chromatograph will be used for boron sampling.

Safety Evaluation Summary: The boron analyzer is used only to measure the amount of boron. The only change is in the means by which boron concentration is measured.

Unreviewed Safety Question Evaluation #95-010

Description: This modification replaces and upgrades the Fire Detection, Control, and Annunciation System for several structures on site. This modification is intended as a component upgrade.

Safety Evaluation Summary: The Fire Detection System is a non-safety-related, augmented Quality Class 7 system. There are no concerns related to: Seismic II/I design requirements; creased plant auxiliary bus loading or power cable sizing; emergency operating capability; combustible cable loading, raceway fill, and raceway support loading; or plant environmental qualification requirements.

Unreviewed Safety Question Evaluation #95-011

Description: This change revises the UFSAR to remove reference to 50° F/hr for a normal Reactor Coolant System heatup and cooldown, and as a cooldown limit for the Residual Heat Removal system. The plant cooldown procedure specifies an administrative cooldown limit of 80° F/hr for the Reactor Coolant System and 160° F/hr for the pressurizer.

Safety Evaluation Summary: Analyses developed for the licensing basis assume the reactor vessel is subjected to 200 full-thermal cycles at a 100° F/hr rate of normal heatup/cooldown. An administrative limit is not assumed in any analysis previously evaluated in the Safety Analysis Report. This change is consistent with the cooldown limit in Technical Specifications 3.4.9.1 and 3.4.9.2.

Unreviewed Safety Question Evaluation #95-012

Description: This change removes a statement in UFSAR Section 6.2.6.1 indicating the secondary system isolation valves are checked for leakage in conjunction with Type A 10CFR50 Appendix J testing.

Safety Evaluation Summary: When the Type A test is performed, the steam generator shell and piping act as the pressure boundary. Since this pressure boundary is not vented during the Type A test, there is no pathway from the containment atmosphere to these isolation valves. This change does not affect the ability of the Main Steam Isolation Valves to perform their intended function.

Unreviewed Safety Question Evaluation #95-013

Description: This change revises the UFSAR mass and energy release steam line break analysis for the Isolation Valve Cubicle and the Reactor Containment Building.

Safety Evaluation Summary: The revised safety analysis demonstrates that the acceptance limits of the Isolation Valve Cubicle and Reactor Containment Building are not exceeded.

Unreviewed Safety Question Evaluation #95-014

Description: This change installs a permanent flow element, a flow switch, and a local flow indicator in each Spent Fuel Pool Cooling System pump discharge piping. The flow switches are used to generate a common annunciator alarm in the Control Room upon low system flow.

Safety Evaluation Summary: The equipment being added by this design change is designed and installed in accordance with the same specifications as the interfacing system.

Unreviewed Safety Question Evaluation #95-015

Description: Cracks have been found in rows 7, 8, and 9 stationary vanes while inspecting low pressure turbines 11 and 12. Cracks are assumed to be present in rows 7, 8, and 9 stationary vanes of low pressure turbine 13.

Safety Evaluation Summary: The condition in LP 13 is being accepted without inspection or repair. The maximum credible turbine damage scenario would not affect any safety-related systems, structures or components, and is covered by the Failure Modes and Effects Analysis.

Unreviewed Safety Question Evaluation #95-016

Description: This evaluation is for the Unit 1, Cycle 6 fuel reload, addressing core and fuel changes from Unit 1, Cycle 5.

Safety Evaluation Summary: The proposed change is bounded by revised analyses in the safety analysis report to the total cycle burnup (including coastdown) of 494 effective full power days.

Unreviewed Safety Question Evaluation #95-018

Description: Changes to the Operations Quality Assurance Plan consist of decentralization and realignment of functional responsibilities of the Engineering Support Department.

Safety Evaluation Summary: There is no reduction in any elements of or responsibilities for implementation of the quality assurance program.

Unreviewed Safety Question Evaluation # 95-019

Description: This evaluation addresses a temporary change providing an additional C Train thermocouple into quadrant C. This is accomplished by using jumper to cross-connect a Train A core exit thermocouple to a Train C thermocouple.

Safety Evaluation Summary: Installation of the metal-clad cable jumper will not violate separation criteria or seismic criteria. The required number of thermocouples per quadrant will continue to be met after this change.

Unreviewed Safety Question Evaluation #95-020

Description: This change limits the accumulator water temperature to less than or equal to 90° F, otherwise compensatory measures are taken to ensure that the Peak Clad Temperature is limited to less than 2200° F in the event of a large break loss of coolant accident. The 90° F temperature was used as part of the input to the UFSAR analysis, but was not listed in the UFSAR itself.

Safety Evaluation Summary: An accumulator water temperature of 90° F was used in the input for the large break loss of coolant accident. By imposing an operating restriction that reflects this input, the margin of safety as determined in the safety analysis is maintained.

Unreviewed Safety Question Evaluation #95-022

Description: This change increases the allowable pressure drop across the NSSS filters from 25 psi to 35 psi at design flow rate. Use of a smaller micron-rated filter is also permitted.

Safety Evaluation Summary: Filter elements are designed to withstand a differential pressure of 75 psid without mechanical failure. Consequences of an accident are not increased by changing the filter element pore size nor the allowable pressure drop. Filter elements are remotely handled during replacement.

Unreviewed Safety Question Evaluation #95-025

Description: This change removes the requirement to perform biennial procedure reviews.

Safety Evaluation Summary: The intent of the biennial review is accomplished by other South Texas Project programmatic controls already in place.

Unreviewed Safety Question Evaluation #95-026

Description: This change adds the option of processing wet solid radioactive waste utilizing South Texas Project personnel and procedures based on the vendor's process control program, and the vendor's Topical Report. The requirement to lease the equipment from the vendor is also deleted.

Safety Evaluation Summary: The same equipment as is presently being used will still be utilized to process waste. The potential consequences of an accident are not changed.

Unreviewed Safety Question Evaluation #95-027

Description: This change is to remove references to the Toxic Gas Monitoring System from the Updated Final Safety Analysis Report and the Technical Requirements Manual.

Safety Evaluation Summary: Onsite and Offsite Toxic Gas Analyses show that none of the potentially hazardous chemicals utilized onsite or in the surrounding five-mile radius pose a credible hazard to the South Texas Project.

Unreviewed Safety Question Evaluation #95-028

Description: This change is to revise the UFSAR description of the reactor vessel head stud tensioners. The current description does not match the actual design.

Safety Evaluation Summary: There is no change in the operation of the stud tensioners.

Unreviewed Safety Question Evaluation #95-029

Description: This evaluation is for a revision to the sequential automatic start time delays for the three diesel-driven fire pumps from 10 seconds to 5 seconds to minimize excessive pressure swings due to large pressure transients.

Safety Evaluation Summary: The current low pressure setting may cause large system transients in the event of a significant pressure drop in the Fire Protection System. This change does not affect the current system operating pressure. The Fire Protection Program is not part of the accident basis as described in UFSAR Chapter 15. The diesel-driven fire pumps are not safety-related.

Unreviewed Safety Question Evaluation #95-030

Description: This change revises calculations for Class 1E battery, battery charger, and inverter sizing to clarify the general calculational methodology and battery cable jumping guidelines. The changes are to address loss-of-offsite power and station blackout considerations.

Safety Evaluation Summary: The changes addressed represent conditions bounded by the analyses performed for the original plant design. Changes to the Updated Final Safety Analysis Report are editorial only.

Unreviewed Safety Question Evaluation #95-031

Description: This change is to revise plant procedure 0PGP03-ZF-0031, "Fire Watch Program," to incorporate definitions for hourly and continuous fire watch. The procedure explains how the South Texas Project will administer the fire watch program when fire protection systems are impaired.

Safety Evaluation Summary: Fire watches are not assumed nor are they considered in the bases for accidents. The capability to achieve and maintain safe shutdown is unchanged by use of these definitions.

Unreviewed Safety Question Evaluation #95-033

Description: This evaluation provides additional assessment of changes to the UFSAR description of Containment Emergency Sumps, the Emergency Core Cooling Pumps (Low Head and High Head Safety Injection) and the Containment Spray System Pumps.

Safety Evaluation Summary: The changes were made within the bounds of the established licensing bases and accident analysis. The changes are consistent with the observations and conclusions of the Safety Evaluation Report regarding the emergency sumps. Emergency sump performance is not impacted. Change in the description of potential debris does not impact sump performance.

Unreviewed Safety Question Evaluation #95-034

Description: This evaluation provides additional assessment of changes to net positive suction head parameters for the emergency core cooling pumps (Low Head and High Head Safety Injection) and the Containment Spray System pumps.

Safety Evaluation Summary: The changes made to the UFSAR are within the existing design assumptions and criteria. The function and performance of the pumps is not affected. Calculations utilizing approved criteria and limiting assumptions demonstrate that the performance requirements continue to be met.

Unreviewed Safety Question Evaluation #95-035

Description: This modification adds a key-lock switch to the Standby Diesel Generator auto-start circuitry. The switch function is to block test mode start signals while the Standby Diesel Generator is in standby.

Safety Evaluation Summary: An emergency start signal will energize the same components as originally designed. Failure of any component used in this modification will not prevent emergency mode starting of the Standby Diesel Generator.

Unreviewed Safety Question Evaluation #95-036

Description: This change revises the allowed outage time given in the Technical Requirements Manual for the boron injection flow paths in Modes 1, 2, and 3 from 72 hours to 7 days.

Safety Evaluation Summary: One flow path will remain operable throughout the allowed outage to provide a source of boron to the reactor coolant system. The boration system of the Chemical and Volume Control System is not a part of the primary success path for mitigation of a design basis accident or transient. This function is provided either by the ECCS or through maintenance of the shutdown margin established by the Technical Specifications and not affected by the change.

Unreviewed Safety Question Evaluation #95-037

Description: Technical Requirements Manual Section 3.4.2.1 required one operable pressurizer safety valve in Modes 4 and 5. The change allows the opening created by a removed pressurizer safety valve to be substituted for an operable safety valve.

Safety Evaluation Summary: Independent and redundant overpressure protection is provided via the Cold Overpressure Mitigation System. The opening provides more flow relief capacity than does a pressurizer valve.

Unreviewed Safety Question Evaluation #95-038

Description: This is an evaluation of a condition in which 300-ton and 150-ton Essential Chillers may start simultaneously with the containment spray pump and discharge motor-operated valve. The postulated event requires a loss of coolant accident plus a loss of offsite power as precursors. The condition had not been analyzed in any of the standby diesel generator transient voltage and frequency evaluations performed during the original plant design.

Safety Evaluation Summary: Although the transient voltage limits defined in the Updated Final Safety Analysis Report are exceeded under this condition, this operability analysis confirms that affected systems, structures and components will operate as designed and perform their safety functions. NRC acceptance limits for standby diesel generator voltage and frequency transients (Regulatory Guide 1.9) are not exceeded.

Unreviewed Safety Question Evaluation #95-039

Description: This change revises the supplemental purge flow rate given in Updated Final Safety Analysis Report Section 15.6.5.3.3 and Table 15.6-10 from 88,900 cfm to 83,200 cfm.

Safety Evaluation Summary: Results of the worst-case analysis show that the calculated dose is within the NRC acceptance limit.

Unreviewed Safety Question Evaluation #95-040

Description: This change to Updated Final Safety Analysis Report Section 9.3.4.1.3.1 specifies that there is a sufficient amount of boron added to the Reactor Coolant System through Reactor Coolant Pump Seal injection to counteract xenon decay after shutdown, rather than specify that the required time is 3.5 hours.

Safety Evaluation Summary: The Chemical Volume and Control System will maintain its ability to provide sufficient negative reactivity to accommodate the positive reactivity associated with xenon decay.

Unreviewed Safety Question Evaluation #95-041

Description: This is an evaluation of the Unit 2, Cycle 5 Reload Safety Evaluation against all known reload-related parameters given in the Updated Final Safety Analysis Report and the plant licensing basis.

Safety Evaluation Summary: The proposed change is bounded by revised analyses in the safety analysis report, addressed by other Unreviewed Safety Question Evaluations, or the license was amended and accepted by the NRC.

Unreviewed Safety Question Evaluation #95-042

Description: This change updates the description of the organization to reflect the current configuration. Nuclear Security and Nuclear Fuels Management are removed from the Operations Quality Assurance Plan. "Surveillance" activities now include "performance monitoring".

Safety Evaluation Summary: These changes are not considered to be a reduction of quality assurance program commitments described in the Safety Analysis Report and previously accepted by the Nuclear Regulatory Commission.

Unreviewed Safety Question Evaluation #95-043

Description: This change reduces the frequencies of some of the Main Cooling Reservoir monitoring activities. The embankment crest elevation measurements are reduced from semiannually to annually. Relief well flow measurements are reduced from quarterly to semiannually. Inclinometer measurements are reduced from quarterly to annually.

Safety Evaluation Summary: These monitoring activities are used to monitor gradually changing characteristics of the reservoir. The consequences of a design basis flood, resulting from failure of the reservoir embankment, are not affected by the change in frequency. Meaningful changes in those reservoir characteristics happen very slowly.

Unreviewed Safety Question Evaluation #95-044

Description: This change deletes all requirements pertaining to Environmental Qualification of safety-related mechanical equipment from sections of the UFSAR. Compliance with 10CFR50, Appendix A, GDC4, is maintained through other programs at the South Texas Project.

Safety Evaluation Summary: Elimination of the Mechanical Environmental Qualification program will not impact safety or intended functions of affected equipment.

Unreviewed Safety Question Evaluation #95-046

Description: This evaluation addresses changes to the Emergency Electrical Loading requirements reflecting results of a new design calculation which analyzed the steady-state standby diesel generator loading during automatic load sequencing following a loss of offsite power, both with and without a safety injection signed.

Safety Evaluation Summary: There is no physical change to any plant structures, systems, or components, and the overall standby diesel generator loading reflected by UFSAR Table 8.3-3 is reduced from previous values. Assumptions used in previous analyses bound plant conditions and equipment functionality that may result from this change.

Unreviewed Safety Question Evaluation #95-103

Description: This evaluation addresses an instance when three Rod Cluster control Assemblies failed to fully insert to the rod bottom position.

Safety Evaluation Summary: The Rod Cluster Control Assemblies are still capable of performing their safety function (reactivity control or shutdown).

Unreviewed Safety Question Evaluation #96-001

Description: This change provides an update to the Operators Quality Assurance Plan and incorporates the methodology for implementation of Graded Quality Assurance activities.

Safety Evaluation Summary: These changes are programmatic only and do not affect design basis or operators.

Unreviewed Safety Question Evaluation #96-003

Description: This evaluation addresses use of essential chiller 11C to maintain C train essential chilled water operability when essential chiller 12C is removed from service for compressor seal maintenance. Under accidental conditions involving a LOCA with failure of A or B train HVAC, the temperature for C train cooled rooms will be 15° higher.

Safety Evaluation Summary: The elevated room temperatures, will not initiate nor affect system functions in a manner that would initiate an additional accident. Station battery room temperatures may increase above the allowable room temperature, but within the manufacturer's recommended operating band.

Unreviewed Safety Question Evaluation #96-004

Description: This evaluation addresses a change in the limitation on Residual Heat Removal pump flow during mid-loop operation from 1500 gpm to 3000 gpm.

Safety Evaluation Summary: Adequate Residual Heat Removal suction nozzle submergence is maintained, and mid-loop water level is monitored. Accident analyses presume loss of decay heat removal; therefore, changing the maximum Residual Heat Removal flow rate has no impact on consequences. The new mid-loop maximum flow rate is within the design basis of the Residual Heat Removal system.

Unreviewed Safety Question Evaluation #96-005

Description: Under an approved deviation from APCSB 9.5-1, Appendix A, Section D.1.j, combustible material is precluded from within 50 feet of the non-rated outside walls. This change to the deviation basis provides exceptions to the 50-foot separation criterion based on Generic Letter 86-10 fire hazards evaluations.

Safety Evaluation Summary: A Generic Letter 86-10 evaluation addresses the defense-in-depth concept of fire protection and determined these changes would be equivalent to the approved condition.

Unreviewed Safety Question Evaluation #96-006

Description: This evaluation addresses use of an alternate reactor coolant system vacuum venting system and process instead of normal fill and vent process as described in UFSAR Section 9.3.4.1.2. The procedure also provides an alternate method of drawing a steam bubble in the pressurizer rather than as described in UFSAR Sections 5.4.7.2.3 and 9.3.4.1.2.

Safety Evaluation Summary: Vacuum venting and filling does not affect the design, integrity and capability of the safety systems to perform their safety functions.

Unreviewed Safety Question Evaluation #96-007

Description: The control loop for a modulating damper is demonstrating unreliable control behavior when in the automatic mode following switchover to the backup power supply. Operator actions are being implemented to maintain the operational readiness of the damper and its associated filtration bank.

Safety Evaluation Summary: The damper serves only to function in response to an initiating event, and therefore does not increase the probability of occurrence of an accident. Use of operator action has been a part of the design basis of the subsystem, so that operator action to maintain operational readiness of the filter trains is not considered a reduction or deviation from commitments.

Unreviewed Safety Question Evaluation #96-008

Description: Full core offloads are used as normal refueling practice, rather than one-third core as discussed in the UFSAR.

Safety Evaluation Summary: Full core offloads during refueling will not result in the spent fuel pool temperature exceeding the 145.7°F discussed in the Safety Evaluation Report.

Unreviewed Safety Question Evaluation #96-010

Description: This change adds an electrical jumper on the refueling machine stationary mast pneumatic system which will disable the failed fuel detection system. The jumper bypasses the stationary mast door open unit switches. This is to support stopping an air leak from the pneumatic system.

Safety Evaluation Summary: This system is non-safety related and its use is not required. The function of the refueling machine in safe handling of fuel assemblies is not affected by the jumper installation. Installation does not affect the design, integrity, and capability of any safety system to perform its safety function.

Unreviewed Safety Question Evaluation #96-011

Description: This evaluation addresses a temporary onsite mobile laundry facility installed near the south Unit 1 Radwaste yard to support the Unit 1 Outage (1REO6).

Safety Evaluation Summary: The facility is physically isolated from plant buildings and does not interface with any system important to safety. Discharges are directed via the Solid Waste Processing System to the Liquid Waste Processing System and eventually dispositioned as liquid radioactive waste.

Unreviewed Safety Question Evaluation #96-012

Description: Calculations of LOCA onsite and offsite doses had incorrectly excluded the dose contribution due to residual radioactivity from prior to isolation. Calculations were also revised using a verified computer code for purge blowdown flow rate.

Safety Evaluation Summary: The onsite and offsite consequences of a design basis LOCA are analyzed to still be within the limits of GDC19 and 10CFR100.

Unreviewed Safety Question Evaluation #96-013

Description: This change removes the requirement to submit a report detailing outage dates and causes for emergency core cooling systems for the last five years of operation. It is replaced with a statement that later rules and regulations ensure that the intent of Section II.K.3.17 of NUREG-0737 are met.

Safety Evaluation Summary: Since later rules meet the intent of the section and NUREG-0737, this change is administrative only.

Unreviewed Safety Question Evaluation #96-014

Description: This change is to allow the rod control cluster assemblies to be partially withdrawn from the reactor core during plant cooldown from Mode 3 to Mode 5.

Safety Evaluation Summary: Main Steam Line Break and Boron Dilution Accident analyses presented in the UFSAR remain bounding. Safe shutdown margin requirements will be met.

Unreviewed Safety Question Evaluation #96-015

Description: The UFSAR is being updated to reflect the impact of the correction of a coding error in the NOTRUMP computer code affecting peak cladding temperature for a small break loss of coolant accident.

Safety Evaluation Summary: All of the acceptance criteria specified in 10CFR50.46 for a loss of coolant accident continue to be satisfied. The assumptions for the dose analysis are not affected by the proposed change. The containment pressure/temperature analysis is not impacted by the proposed change.

Unreviewed Safety Question Evaluation #96-016

Description: The electrical overspeed trip of the main turbine generator is inoperable and does not give a trip signal when tested. Trouble-shooting of this circuit could result in a main turbine generator trip. Since the failure mechanism is unknown, a temporary modification to defeat a potential spurious actuation of the electrical overspeed trip was considered prudent.

Safety Evaluation Summary: Failure of the electrical overspeed trip has a minor impact on the probability of destructive overspeed due to failure of control action and overspeed trips. The overall probability of turbine missile generation from destructive overspeed is not significantly impacted. Technical Specification 3.3.4 requires that at least one turbine overspeed protection system shall be operable. Mechanical overspeed protection will continue.

Unreviewed Safety Question Evaluation #96-017

Description: This revision to analytical assumptions used on Reload Core Analyses removes the Fuel Densification Spike penalty. This is warranted due to improvements in the fuel manufacturing process.

Safety Evaluation Summary: The revised Safety Analysis (Reload Specific) shows acceptable results with the fuel densification spike factor removed. The NRC has approved formal removal of the fuel densification spike factor as documented in WCAP-13589-A. "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel."

Unreviewed Safety Question Evaluation #96-018

Description: This evaluation addresses changes to assumptions used in the spent fuel pool boiling dose analysis covered in the UFSAR. The changes are made to adjust the calculation to plant conditions.

Safety Evaluation Summary: The radiological consequences described in this change are bounded by those given in the Safety Evaluation Report.

Unreviewed Safety Question Evaluation #96-019

Description: This change is to allow for conducting refueling operations and reactor startup without the use of secondary source assemblies in the reactor core.

Safety Evaluation Summary: The assumptions on the Boron Dilution Analysis and Uncontrolled RCCA withdrawal from a subcritical condition are retained and remain valid. The respective analyses in the UFSAR remain bounding.

Unreviewed Safety Question Evaluation #96-020

Description: This is an evaluation of the Unit 1 Cycle 7 Reload Safety Evaluation against all reload-related parameters within the UFSAR and licensing basis, as well as miscellaneous procedures and calculations.

Safety Evaluation Summary: There are no changes to plant equipment or procedures, except for fuel changes to the safety analysis addressed by other unreviewed safety question evaluations, bounded by the revised safety analysis, or approved by the NRC. There is no change in the radiological dose resulting from accidents.

Unreviewed Safety Question Evaluation #96-021

Description: Fire Area 06 Zone 019 has been approved for use as a storage area for ladders and electrical load cells while they are not in use.

Safety Evaluation Summary: The increase in combustible loading has been calculated and evaluated in terms of combustible loading and fire severity. There are no safe shutdown components or circuits in this fire area. Therefore, the increase in combustible material will have no effect on the capability to achieve and maintain safe shutdown. The conclusions of the Appendix R analysis remain unchanged and bounding.

Unreviewed Safety Question Evaluation #96-022

Description: This change deletes the UFSAR requirement for the shift supervisor to designate the authority for licensed activities to another senior reactor operator in writing during his absence from the control room. This authority is delegated verbally.

Safety Evaluation Summary: This is an administrative change to reflect South Texas Project practice.

Unreviewed Safety Question Evaluation #96-023

Description: This change revises the requirements of the shift Technical Advisor given in the UFSAR to state that the South Texas Project follows the Commission policy regarding engineering expertise on shift. The training requirements will also be revised.

Safety Evaluation Summary: This is an administrative change to better state the basis for the South Texas Project shift technical advisor program and training requirements in accordance with licensing basis documents.

Unreviewed Safety Question Evaluation #96-024

Description: This change adds weapon vaults in various locations on Category 1 buildings.

Safety Evaluation Summary: These vaults are metal and are secured with access by authorized personnel only. Internal missiles from stored ammunition will not be generated since the bullets will not penetrate the metal walls of the vault.

Unreviewed Safety Question Evaluation #96-025

Description: During Unit 1 Cycle 6, three Rod Control Cluster Assemblies failed to fully insert following reactor trip. This has been evaluated for Cycle 7.

Safety Evaluation Summary: Reasonable precautions have been taken to ensure that this will not occur, or otherwise be limited, for the duration of Cycle 7. However, failure of the Rod Control Cluster Assemblies to insert given the bounding scenarios is acceptable since it does not represent an unreviewed safety question.

Unreviewed Safety Question Evaluation #96-026

Description: This evaluation is for a change to the Operations Quality Assurance Plan to reflect the current management organization.

Safety Evaluation Summary: This change does not reduce any elements of or responsibilities for implementation of the quality assurance program. The change is to the management organization only.

Unreviewed Safety Question Evaluation #96-027

Description: The UFSAR is revised to clarify that Bottom-Mounted Instrumentation flux thimble tubing and fittings are exempt from ASME III code requirements for replacements as provided by ASME XI. The original compression nuts in the Incore Neutron Flux Monitoring System Bottom Mounted Instrumentation thimble guide tubes are damaged, and will be replaced with an alternate split nut design.

Safety Evaluation Summary: The replacement fitting performs the same function and exhibits similar material strengths as does the original fitting. Effect of split nut failure is bounded by the small break loss of coolant accident. The UFSAR revision is based on an exemption under an interpretation from ASME Section XI.

Unreviewed Safety Question Evaluation #96-030

Description: This evaluation confirmed that, if the Rod Control Cluster Assemblies fail to fully insert upon reactor trip, the condition is not an unreviewed safety question.

Safety Evaluation Summary: The condition of stuck rods occurring during Unit 2 Cycle 5 has been evaluated and found to not result in a violation of safety limits. The condition does not violate any licensing requirements.

Unreviewed Safety Question Evaluation #96-033

Description: This evaluation addresses continued operation of the plant with a loose part present in the reactor coolant system or interconnected primary systems. The most likely source is a loose control rod guide tube support pin.

Safety Evaluation Summary: Loss of a control rod guide tube support pin has no adverse effect on the continued safe operation of the unit. Presence of this loose part, another equivalent size loose part or smaller loose part in the reactor coolant system and various other interconnected primary systems, does not affect the safety-related function of the equipment and systems.

Unreviewed Safety Question Evaluation #96-037

Description: This change to UFSAR Section 12.4 uses historical radiation exposure data from years 1990 through 1995, rather than using projections of numbers of plant workers, radiation levels, and occupance times, in estimating dose to site workers.

Safety Evaluation Summary: This is a change in methodology and is not a change to the facility, does not affect equipment operation, and does not change a basis for any technical specification.

Unreviewed Safety Question Evaluation #96-038

Description: These changes to the description of the health physics program update the UFSAR language to more accurately reflect the facility and current management philosophy.

Safety Evaluation Summary: Although some procedures described in the UFSAR are affected, the procedures are strictly administrative in nature and are not safety-related. The thrust of the changes is editorial, removal of detail, and updating in accordance with current practices.

Unreviewed Safety Question Evaluation #96-041

Description: This evaluation provides a basis for concluding that operations, maintenance, and testing that temporarily interrupt Spent Fuel Pool cooling during non-refueling conditions is acceptable as long as the Spent Fuel Pool temperature remains below 150.7°F under licensing basis assumptions.

Safety Evaluation Summary: This procedure change has no impact on the ability of the Spent Fuel Pool Cooling and Cleanup System to provide adequate cooling to the Spent Fuel Pool under normal operating, abnormal operating, or accident conditions. No changes are made to plant equipment.

Unreviewed Safety Question Evaluation #96-042

Description: This change removes the at-power MSIV partial stroke test because it is no longer required to meet the requirements of the Section XI program.

Safety Evaluation Summary: Periodic testing of the MSIVs will continue to be performed as required by the Technical Specifications and the Inservice Testing program. MSIV response times are not affected by this change.

Unreviewed Safety Question Evaluation #96-044

Descriptior.. Portions of the Integrated Computer System are installed to support implementation of a modification. This results in a change in the combustible loads.

Safety Evaluation Summary: There is no increase in the probability of occurrence of a fire, or in the consequences since the entire contents of a given fire area are assumed lost to a fire.

Unreviewed Safety Question Evaluation # 96-101

Description: This evaluation addresses a temporary modification made to maintain operability of the Unit 2 B train Control Room Envelope HVAC Emergency Makeup Damper. The damper was placed in a fixed, post-accident, position.

Safety Evaluation Summary: The subject damper is idle during normal plant operation. The temporary change places the damper in its "safe", post-accident, position, and supports accident mitigation. This "safe" position is within the design basis of the Makeup Filter Units.