

The Light company

Houston Lighting & Power

South Texas Project Electric Generating Station P. O. Box 289 Wadsworth, Texas 77483

December 21, 1994
ST-HL-AE-4942
File No.: G20.01
G21.01
10CFR50.59

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

Reference: Correspondence from J. F. Groth to Document Control Desk dated
December 21, 1994 (ST-HL-AE-4943)

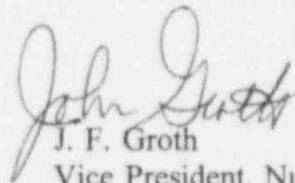
Pursuant to 10CFR50.59, Houston Lighting & Power Company submits the attached report containing a brief description of changes, tests, and experiments including a summary of the safety evaluation of each. This report is being submitted in conjunction with the update of the South Texas Project Updated Final Safety Analysis Report (see referenced letter).

The gaps in the numerical sequence of the attached Unreviewed Safety Question Evaluation summaries represent changes, tests or experiments that have been cancelled, had not been completed within six months of this report, or were submitted with a previous summary report.

If you should have any questions, please contact Mr. K. J. Taplett at (512) 972-8416 or me at (512) 972-8664.

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DNB/pas

Attachment: Summary of Unreviewed Safety Question Evaluations

JEH 11

Houston Lighting & Power Company
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Page 2

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Unreviewed Safety Question Evaluation #90-202

Description: The Fire Hazards Analysis Report has been changed to reflect recalculation of combustible loading in various fire areas using a new computer program.

Safety Evaluation Summary: The new computer program uses existing cable load and the as-built configuration of the plant. The changes in combustible loading do not affect the ability to safely shutdown following a maximum postulated fire in any one fire area.

Unreviewed Safety Question Evaluation #91-051

Description: The operating logic for the Feedwater Isolation Valve solenoids and Heating, Ventilation and Air Conditioning system has been modified for better cooling of the solenoids.

Safety Evaluation Summary: The logic changes do not affect the closing time or ability of the Feedwater Isolation Valves to close when required. The increased cooling decreases the probability of Feedwater Isolation Valve failure.

Unreviewed Safety Question Evaluation #91-057

Description: Installation of new toxic gas analyzers has increased the combustible loading in various fire zones due to the addition of new cables.

Safety Evaluation Summary: The combustible loading and new materials are within the margins and deviations listed in the Fire Hazards Analysis Report. The type and severity of fire that could be created by the new material is not different from that previously analyzed.

Unreviewed Safety Question Evaluation #91-060

Description: The circuit cards used for a variable steam generator water level setpoint are being removed because a fixed (constant) setpoint is used and the variable setpoint cards are not required.

Safety Evaluation Summary: The cards and the control cabinets in which they are mounted are not safety related and the removal of the cards will not create an adverse interaction with safety related components. The cards are being removed to reduce the potential for a common point circuit failure.

Unreviewed Safety Question Evaluation #91-061

Description: Delete the level indication for the Essential Cooling Pond (ultimate heat sink) because it is erratic and unreliable.

Safety Evaluation Summary: The Essential Cooling Pond level indication is not safety related and is not directly related to initiation or mitigation of an accident or malfunction. The Essential Cooling Pond level is monitored by a permanent staff gage which ensures that adequate water is available for the mitigation of accidents or malfunctions.

Unreviewed Safety Question Evaluation #92-016

Description: The Updated Final Safety Analysis Report has been changed to reflect that the Essential Cooling Pond primary makeup supply is well water rather than the Main Cooling Reservoir, that the Essential Cooling Water system free chlorine is monitored by grab sample or by free chlorine analyzer, and to clarify that the side stream corrosion monitoring system monitors corrosion rates rather than corrosion products.

Safety Evaluation Summary: The use of well water for normal makeup will keep critical Essential Cooling Water chemistry parameters further below the maximum evaluated levels than makeup from the Main Cooling Reservoir. The system reliability should be enhanced by keeping the potential for system scaling lower than before. System chemistry will still be maintained within evaluated parameters.

Unreviewed Safety Question Evaluation #92-020

Description: Twelve 55 gallon drums of noncombustible compounds were added to a location inside the plant for easy and prompt access. The 55 gallon drums are plastic, which increases the combustible loading in that area.

Safety Evaluation Summary: The additional combustible loading is bounded by the previous analysis of a fire in this area.

Unreviewed Safety Question Evaluation #92-028

Description: This modification provides an alternate means of cooling the Spent Fuel Pool when in the abnormal maximum heat load condition (full core offload) while the entire Train "C" electrical system is deenergized for maintenance.

Safety Evaluation Summary: The modification ensures that an alternate means of Spent Fuel Pool cooling is available at all times. The proposed temporary modification does not adversely impact equipment important to safety. The proposed operator actions will not disable equipment assumed to be available in the safety analysis.

Unreviewed Safety Question Evaluation #92-029

Description: A 10-inch ethylene pipeline was installed offsite to the Bay City Celanese plant. The new pipeline run is within 5 miles of the site.

Safety Evaluation Summary: The effects of an ethylene pipeline explosion are bounded by a previous analysis of a natural gas line explosion.

Unreviewed Safety Question Evaluation #92-030

Description: Procedure 0PSP03-HE-0001 was modified by Field Change 920624 to allow surveillance testing of the control room Heating, Ventilation and Air Conditioning system with the toxic gas monitoring system inoperable during the defueled mode.

Safety Evaluation Summary: Realignment the control room Heating, Ventilation and Air Conditioning system to conduct the surveillance test is an expected configuration and operating condition. No plant transient is introduced by realigning the Heating, Ventilation and Air Conditioning system.

Unreviewed Safety Question Evaluation #92-031

Description: The Standby Diesel Generator fuel oil is cleaned by recirculation through a non-safety related system. This practice was evaluated for impact on Standby Diesel Generator operability.

Safety Evaluation Summary: A calculation was performed to show that the cleanup process does not affect the ability of the Standby Diesel Generator to run for the required seven days after a loss of coolant accident coincident with a seismic event and loss of offsite power.

Unreviewed Safety Question Evaluation #92-033

Description: During core offload, it was observed that a number of fuel assemblies removed from the core had one or two fuel assembly top nozzle holddown springs that had not returned to their original position. A process was developed to repair the springs while the fuel assembly was secured in the new fuel elevator.

Safety Evaluation Summary: All fuel handling equipment has administrative and physical controls to ensure that spent fuel is not raised above the elevation where safe water shielding would be lost. If the new fuel elevator should fail and drop the spent fuel to the floor of the spent fuel transfer canal, the accident consequences are bounded by the design basis fuel handling accident.

Unreviewed Safety Question Evaluation #92-034

Description: The Emergency Plan has been changed to reflect the Nuclear Utility Management and Resource Council methodology for classifying emergencies, which uses a logical progression of classification based on the hazards to the plant and the public.

Safety Evaluation Summary: With respect to classification of an emergency, the actions directed by the Emergency Plan do not directly affect equipment, plant operations or maintenance practices.

Unreviewed Safety Question Evaluation #92-035

Description: The post-accident radiation monitors for the Operational Support Center have been deleted from the Regulatory Guide 1.97 Type "E" variable list in the Updated Final Safety Analysis Report because NUREG-0696 and NUREG-0737 do not require the Operational Support Center to have access area radiation monitors.

Safety Evaluation Summary: The radiation monitors provide only a monitoring function and have no impact on other equipment or plant operations. Habitability and evacuation decisions are based on other data.

Unreviewed Safety Question Evaluation #92-036

Description: In line review of engineering design documents by Quality Assurance was deleted. Quality Assurance will continue to overview the design engineering process through surveillances and audits.

Safety Evaluation Summary: The review by Quality Assurance was performed to check the seismic and quality group class, quality standards and identify any deviations from quality standards. The deletion of this review would not affect the initiators or actions taken to mitigate the consequences of an accident or malfunction.

Unreviewed Safety Question Evaluation #92-037

Description: An evaluation was performed to confirm that safety analyses and parameters specified in the Updated Final Safety Analysis Report and Technical Specifications will be bounded during Cycle 5 of Unit 1.

Safety Evaluation Summary: The existing safety analyses remain bounding for the entire cycle.

Unreviewed Safety Question Evaluation #92-038

Description: The specification and associated documents were revised to allow the use of Kwik-Bolt II stud-type expansion anchors instead of the currently approved Kwik-Bolts.

Safety Evaluation Summary: The Hilti Kwik-Bolt II (KB-II) is a comparable anchor to the original Kwik-Bolt. All the requirements for the Kwik-Bolt apply to the KB-II as well and existing standard designs are adequate for KB-II installations. Structural integrity of supports utilizing KB-II are fully maintained.

Unreviewed Safety Question Evaluation #92-039

Description: The cross-tied mode of operation has been abandoned and the cross-tie valves have been locked closed (USQE 88-146). This current modification removes control instrumentation and main control panel wiring for the deleted electro-hydraulically operated Essential Cooling Water chiller outlet valves.

Safety Evaluation Summary: The modification eliminates control instrumentation made obsolete by the initial modification that locked closed the Essential Cooling Water cross-tie valves. The removal of instrumentation improves the safety considerations and control room human factors concerns by eliminating the non-functional panel mounted components.

Unreviewed Safety Question Evaluation #93-001

Description: The requirement to lock thirteen manual, normally closed valves that were part of vent or drain double-valve pairs was deleted.

Safety Evaluation Summary: Each valve remains physically in place and in its normal, closed position. Each valve was part of a double-valve protection pair and the second valve (inboard) in each pair still remains locked. System flow rates and isolation requirements remain as before.

Unreviewed Safety Question Evaluation #93-003

Description: The Updated Final Safety Analysis Report was changed to allow low power calorimetrics to be based on primary delta-T measurements instead of a secondary heat balance.

Safety Evaluation Summary: The change improves the overall accuracy of nuclear instrumentation calibrations by reducing the process error at low power levels. The change only affects the nuclear instrumentation calibration accuracy, not the function of the nuclear instrumentation itself.

Unreviewed Safety Question Evaluation #93-004

Description: The change increased the maximum clearance between the bottom of a fire door and the floor from 3/4" to 1" and the maximum clearance between the door and its frame from 1/8" to 1/4".

Safety Evaluation Summary: Although the National Fire Protection Association Code allows only 3/4" and 1/8", respectively, fire tests with gaps of 1" and 1/4" have shown the doors are still acceptable barriers according to American Nuclear Insurers.

Unreviewed Safety Question Evaluation #93-006

Description: Beginning with the fresh fuel loaded into Unit 2 Cycle 4 and Unit 1 Cycle 6, the fuel for South Texas Project may be manufactured using Integral Fuel Burnable Absorber. The other significant design change associated with the V5H upgrade is the replacement of the eight Inconel mid-grids with eight Zircaloy mid-grids.

Safety Evaluation Summary: The accident parameters which have an impact on dose, fuel damage and offsite releases are not adversely affected by the fuel upgrade. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the fuel change. No new failure scenarios are created as a result of this fuel transition. Based on the analyses performed for the topical report, the new fuel design does not impact the Technical Specifications, and the operating envelope continues to be bounded by the analytical bases.

Unreviewed Safety Question Evaluation #93-007

Description: A temporary modification in Unit 1 substituted a flange without isokinetic nozzles for the flanged isokinetic nozzle assembly to allow partial operation of the Unit monitors in the plant main exhaust during work under a permanent modification.

Safety Evaluation Summary: The isokinetic nozzles provide sampling for two effluent monitors set to alarm before Offsite Dose Calculation Manual values are exceeded. The temporary flanges allow continued operation of a portion of the unit vent radiation monitors, i. e., the noble gas activity monitor. Any increase in gaseous radioactivity will be detected by the noble gas activity monitor which is not affected by loss of isokinetic sampling.

Unreviewed Safety Question Evaluation #93-009

Description: The number of manual valves required to be locked in the Feedwater, Containment Spray, Steam Generator Blowdown, Component Cooling Water, High Head and Low Head Safety Injection systems has been reduced to eliminate redundant locking requirements.

Safety Evaluation Summary: Removing the locking requirements does not affect the material strength, normal position or performance characteristics from the affected valves. A redundant valve in the same line will remain locked closed to meet regulatory requirements.

Unreviewed Safety Question Evaluation #93-011

Description: The change revised the experience requirements for the General Manager, Nuclear Assurance.

Safety Evaluation Summary: The change is administrative in nature and did not affect any accident or equipment malfunction.

Unreviewed Safety Question Evaluation #93-012

Description: The modifications added a stand-alone N-16 primary-to-secondary leak monitoring system to provide supervisory monitoring and main control room indication of critical steam generator tube leak data. A remote N-16 gamma detector is installed adjacent to each main steam line in the Turbine Generator Buildings.

Safety Evaluation Summary: This system's only interface with other plant systems is with the Proteus plant computer through the Proteus I/O subsystem. There are no interactions with other plant equipment.

Unreviewed Safety Question Evaluation #93-014

Description: This change deleted the requirement for Quality Assurance to perform in-line review of procurement documents.

Safety Evaluation Summary: Quality Assurance will still approve changes to quality requirements and will maintain overview of the site procurement process through the Quality Assurance Audit Program.

Unreviewed Safety Question Evaluation #93-015

Description: Added descriptions for the Director of Quality Assurance and Manager of Nuclear Quality Control and Material Testing. Also, changed the reporting chain for Manager of Nuclear Fuels.

Safety Evaluation Summary: The organizational changes were administrative in nature and did not affect any accident or malfunction of equipment.

Unreviewed Safety Question Evaluation #93-016

Description: The site flooding analysis was revised to account for the completion of S.W. Freese dam (located on Colorado River about 600 miles from the plant site).

Safety Evaluation Summary: The S. W. Freese dam did not affect South Texas Project because the site flood level from the Colorado River remains below the design basis flood level created by a breach of the Main Cooling Reservoir embankment.

Unreviewed Safety Question Evaluation #93-017

Description: Added a description of the new Vice President, Support and Manager, Human Resources positions.

Safety Evaluation Summary: The organizational changes were administrative in nature and did not affect any accident or malfunction of equipment.

Unreviewed Safety Question Evaluation #93-018

Description: The Component Cooling Water system water chloride limit during normal plant operation was changed from 0.15 ppm to 0.500 ppm.

Safety Evaluation Summary: The increased chloride limit did not affect the nitrite protective layer and would not promote stainless steel stress corrosion cracking. The proposed change did not affect the cooling capacity of the Component Cooling Water system.

Unreviewed Safety Question Evaluation #93-019

Description: The motor of Essential Chiller #1 in Unit 2 has been replaced during startup with an equivalent replacement.

Safety Evaluation Summary: The change did not impact the operation of the Essential Chiller or require alteration of any equipment operating procedures. The change does not affect the ability of the chiller to operate as designed, and does not change the ability of the Standby Diesel Generator to start on demand and accept loads as sequenced.

Unreviewed Safety Question Evaluation #93-020

Description: The heat conduction model used in the analysis modeled the reactor vessel as stainless steel rather than carbon steel with a stainless steel clad. Correction of the model resulted in a 25°F benefit in peak clad temperature.

Safety Evaluation Summary: The peak clad temperature remains below the acceptance criteria of 10CFR50.46 and the results of the loss of coolant accident dose analysis remain bounding.

Unreviewed Safety Question Evaluation #93-022

Description: The Emergency Plan has been changed to reflect the Nuclear Utility Management and Resource Council methodology for classifying emergencies, which uses a logical progression of classification based on the hazards to the plant and the public.

Safety Evaluation Summary: With respect to classification of an emergency, the actions directed by the Emergency Plan do not directly affect equipment, plant operations or maintenance practices.

Unreviewed Safety Question Evaluation #93-024

Description: Procedure OPRP04-ZR-0015 was written to incorporate revised 10CFR20 requirements and language. It includes new criteria for posting very high radiation areas and changes criteria for posting airborne radioactivity areas.

Safety Evaluation Summary: Accessibility to equipment important to safety is not decreased by the required posting. There is no change in either the radiation environment of such equipment or the maintenance to be performed on it.

Unreviewed Safety Question Evaluation #93-025

Description: Temporary piping has been installed to bypass Steam Generator Bulkwater Containment Isolation valves to provide an alternate sampling flow path.

Safety Evaluation Summary: The temporary piping is connected to non-class piping and has been designed to not have any adverse interactions with equipment important to safety. The function of the safety-related portion of the Steam Generator Bulk Water system and the normal system configuration are unaffected by this change. Administrative controls have been implemented to ensure that the normally locked closed containment isolation valves would be closed within the four-hour limit stated in Technical Specification 3.6.3 if containment integrity is required.

Unreviewed Safety Question Evaluation #93-026

Description: The changes to 10CFR20 had a mandatory effective date of January 1, 1994. Many documents, including the Updated Final Safety Analysis Report and procedures, required revision to comply with the changes.

Safety Evaluation Summary: 10CFR20.1001-2401 consists of administrative radiation protection requirements. Neither 10CFR20.1001-2401 nor the documents that implement these regulations change the physical or operating characteristics of any equipment or system or impact any physical parameter associated with the plant.

Unreviewed Safety Question Evaluation #93-027

Description: The definition of raining waivers and exemptions has been changed to be consistent with current Institute of Nuclear Power Operations recommendations.

Safety Evaluation Summary: The definition changes are administrative in nature and did not affect any accident or equipment malfunction.

Unreviewed Safety Question Evaluation #93-029

Description: The comparison of the Containment Isolation Signal Override to Comanche Peak has been changed to convey the as-built South Texas configuration.

Safety Evaluation Summary: This change does not physically alter the plant and does not alter the bases for the accident analyses.

Unreviewed Safety Question Evaluation #93-030

Description: The maximum closure time for the Main Steam Isolation Bypass valves was changed.

Safety Evaluation Summary: The required closure time is shorter, which is more conservative.

Unreviewed Safety Question Evaluation #93-032

Description: During emergencies food and water will be supplied to the Control Room instead of being stored inside the Control Room envelope

Safety Evaluation Summary: Changing the storage requirement for Control Room food and water supplies does not affect any accident or malfunction of equipment.

Unreviewed Safety Question Evaluation #93-033

Description: The Standby Diesel Generator control circuits (non-1E portion) were modified.

Safety Evaluation Summary: Technical Specification 3/4.8 addresses on-site emergency power requirements. These requirements are achieved by the class 1E control circuits. The scope of this change is limited to the non-1E circuits.

Unreviewed Safety Question Evaluation #93-034

Description: Compensatory measures were implemented to prevent misoperation of the Airlock Interlock mechanism

Safety Evaluation Summary: The compensatory actions ensure that auxiliary airlock integrity is maintained.

Unreviewed Safety Question Evaluation #93-035

Description: The nonshrink grout specification was revised.

Safety Evaluation Summary: The structural capability of grout supplied under the new specification meets or exceeds the requirements of the old specification and does not create a new failure mechanism.

Unreviewed Safety Question Evaluation #93-036

Description: Modifications have been implemented to improve chiller operation.

Safety Evaluation Summary: The modified components are designed to the same Quality Class, codes and standards, and seismic requirements specified for the balance of the Essential Cooling Water system. The modifications increase the systems' reliability and stability of operation. The analyzed line breaks and single failures envelope all credible accidents or malfunctions of equipment.

Unreviewed Safety Question Evaluation #93-038

Description: Nuclear Group organizational changes were made.

Safety Evaluation Summary: The change describes organizational restructuring. Organizational functions and personnel qualifications are not changed.

Unreviewed Safety Question Evaluation #93-040

Description: Digital Rod Position Feeder Breakers were relabeled.

Safety Evaluation Summary: No change was made to the cabinets, device location, cable routing, or circuit protection.

Unreviewed Safety Question Evaluation #93-041

Description: The education and training requirements for Shift Technical Advisors was changed.

Safety Evaluation Summary: The Shift Technical Advisor education and training requirements of Technical Specification 6.2.4 are satisfied.

Unreviewed Safety Question Evaluation #93-042

Description: Two fuel assemblies were discovered to have newly damaged grid straps during mid-cycle core offload.

Safety Evaluation Summary: The probability of occurrence of an accident previously evaluated in the Safety Analysis Report is independent of the grid strap damage. The fuel design, operation, and basic structural configuration are not changed. The mechanical damage to the grid strap would not affect the fuel/reactor parameters specified in the Technical Specifications.

Unreviewed Safety Question Evaluation #93-043

Description: The motor pinion and worm shaft gear were replaced on three motor operated valves in the Safety Injection system. The new valve stroke times exceeded the previous specified maximum value.

Safety Evaluation Summary: The valves are closed by operator action and the closing time is not specified in any accident analysis.

Unreviewed Safety Question Evaluation #93-044

Description: Two new divisions, Engineering Support and Engineering Programs, have been created in the Nuclear Engineering Department to more effectively support plant operations and maintenance.

Safety Evaluation Summary: The change describes organizational restructuring. Organizational functions and personnel qualifications are not changed.

Unreviewed Safety Question Evaluation #93-045

Description: Training programs have been changed to reflect the Systematic Approach to Training for certain job positions, as required by 10CFR50.120. Also, pre-operating license training commitments have been updated to reflect completion.

Safety Evaluation Summary: The changes to the site training program do not degrade safety system performance, create challenges to safety, or increase the dose levels to the public.

Unreviewed Safety Question Evaluation #93-047

Description: The itemized lists of combustible materials for each fire area have been removed from the Fire Hazards Analysis Report. The list of combustible materials will be maintained in a plant calculation, which will ensure that the total loading value is acceptable.

Safety Evaluation Summary: The combustible loads have not changed. The ability to safely shutdown following a maximum postulated fire in any one fire area is not affected.

Unreviewed Safety Question Evaluation #93-049

Description: Specific valves are listed in the Updated Final Safety Analysis Report that are to be locked closed during shutdown modes to prevent a boron dilution event. Wording has been added to allow use of the Reactor Coolant Pump Standpipe Supply Isolation valves under certain conditions during mode 4.

Safety Evaluation Summary: The Reactor Coolant Pump Standpipe Supply Isolation valves will be locked closed when a boron dilution event can occur.

Unreviewed Safety Question Evaluation #93-050

Description: The procedure used for classifying emergencies has been changed to use the Nuclear Utility Management and Resource Council methodology, which uses a logical progression of classification based on the hazards to the plant and the public.

Safety Evaluation Summary: The emergency classification procedure establishes criteria for classifying radiological and other plant emergencies, and actions taken in the procedure will not directly affect equipment, equipment operating procedures plant operations or maintenance practices.

Unreviewed Safety Question Evaluation #93-052

Description: License Amendment 56/45 deleted audit frequencies from Technical Specification 6.5.2.8. Quality Assurance program effectiveness will be increased by allocating resources and scheduling audits on activities in need of evaluation.

Safety Evaluation Summary: Quality Assurance audit activities do not affect any accident or equipment malfunctions.

Unreviewed Safety Question Evaluation #94-001

Description: The Operations Quality Assurance Plan has been changed to reflect unitization of the Nuclear Generation organization, reorganization of the Nuclear Engineering organization, and formation of the Plant Services organization.

Safety Evaluation Summary: The change describes organizational restructuring. Organizational functions and personnel qualifications are not changed.

Unreviewed Safety Question Evaluation #94-002

Description: The Updated Final Safety Analysis Report has been changed to accurately reflect the design features/parameters of debris interceptor structures installed in the Reactor Containment Building sumps.

Safety Evaluation Summary: The structural capability of the containment sumps and debris interceptor structures has not been altered. The containment sumps and debris interceptors will perform their design function.

Unreviewed Safety Question Evaluation #94-003

Description: The peak clad temperature for the small break loss of cooling accident has changed from 1490°F to 1627°F because of drift flux flow regime errors and safety injection flow was found to have worse effect when modelled as flowing into the core rather than spilling out of the broken loop.

Safety Evaluation Summary: The peak clad temperature changes do not result in a peak clad temperature exceeding the 10CFR50.46 acceptance criteria.

Unreviewed Safety Question Evaluation #94-004

Description: Operability requirements for the Feedwater Isolation Bypass Valves and Steam Generator Preheater Bypass Valves have been changed from Technical Specification 3.7.1.7 to Technical Specification 3.6.3 because the Steam Generator tubes are a second isolation barrier for these containment isolation valves.

Safety Evaluation Summary: The effect of four stuck open Feedwater Isolation Bypass Valves was evaluated for applicable accidents and determined to be bounded by existing analyses. The same conclusion is applicable for stuck open Steam Generator Preheater Bypass Valves.

Unreviewed Safety Question Evaluation #94-005

Description: An evaluation was performed for the use of a diesel driven compressor to supply compressed air to the Instrument Air system during maintenance.

Safety Evaluation Summary: The Instrument Air system is not relied on for a design basis accident and is not required for safe shutdown of the plant. Upon loss of instrument air, safety related valves fail in or to their required position.

Unreviewed Safety Question Evaluation #94-006

Description: The Updated Final Safety Analysis Report has been changed to exempt the Accumulator Discharge Isolation Valves from testing while the plant is operating at power.

Safety Evaluation Summary: Because the valves are open (with electrical power removed) when the plant is at power, the valves are in their required position for an accident. Therefore, eliminating the requirement for testing when the plant is at power would leave the valves in their required position.

Unreviewed Safety Question Evaluation #94-007

Description: The temperature and humidity setpoints and sensitivities for the Control Room Envelope Computer Room and the Technical Support Center Computer Room were changed to reflect current operating parameters.

Safety Evaluation Summary: The change does not affect the intended function of the components located in these computer rooms because the new temperature/humidity ranges are enveloped by the normal environmental requirements for the components in these rooms. The new ranges ensure the proper environment for computer protection.

Unreviewed Safety Question Evaluation #94-008

Description: The Operations Quality Assurance Plan has been changed to address Nuclear Regulatory Commission Generic Letter 89-09, which discusses procurement of replacement components and parts from vendors who do not have a current Certificate of Authorization.

Safety Evaluation Summary: The change authorizes use of the provisions of Generic Letter 89-09; no plant components or parts have been replaced by this change. A separate evaluation will be required for plant components and parts that are changed.

Unreviewed Safety Question Evaluation #94-009

Description: Correction of a calculational error in the loss of coolant analysis has changed the time to initiate transfer to hot leg recirculation following a loss of coolant accident from 10.5 to 6.5 hours.

Safety Evaluation Summary: By changing the initiation time from 10.5 to 6.5 hours, plateout of boron precipitate is prevented and sufficient flow of water through the core is provided to maintain core recovery and a coolable geometry in accordance with 10CFR50.46(b)(4). As a result, the loss of coolant accident analysis is not impacted and the acceptance criteria specified in 10CFR50.46 continue to be satisfied.

Unreviewed Safety Question Evaluation #94-010

Description: The frequency of obtaining grab samples from the unit vent for tritium analysis during outages has been changed from at least once per 24 hours when the refueling canal is flooded to at least once per week when the Primary Coolant or the Refueling Water Storage Tank contains water with the tritium concentrations in excess of $5\mu\text{Ci/gm}$.

Safety Evaluation Summary: The controls specified for tritium in the Offsite Dose Calculation Manual are intended to maintain dose rates below regulatory limits at the site boundary under normal operating conditions. This change in sampling frequency would not increase the dose rate at the site boundary due to released gaseous effluents.

Unreviewed Safety Question Evaluation #94-011

Description: Temporary instrumentation was installed on Standby Diesel Generator 21 to determine the cause of inadvertent starts.

Safety Evaluation Summary: The installation of the temporary instrumentation considered electrical separation and seismic II/I criteria. In addition, the double isolation fuses were sized such that the safety function of the starting circuitry for Standby Diesel Generator was not affected.

Unreviewed Safety Question Evaluation #94-012

Description: An evaluation was performed to confirm that the results of the large break loss of coolant accident were acceptable when modeled with Vantage 5H hybrid fuel.

Safety Evaluation Summary: Although the zircaloy/water reaction increased to reflect use of Vantage 5H hybrid fuel, the acceptance criteria of 10CFR50.46 are satisfied.

Unreviewed Safety Question Evaluation #94-013

Description: During installation of new instruments to monitor the performance of Turbine Driven Auxiliary Feedwater Pump 24, fire, flood and high energy line break barriers were temporarily breached.

Safety Evaluation Summary: The temporary breaching of the fire barrier was controlled under fire protection procedures and the breaching of the flooding and high energy line break barriers are enveloped by existing analyses for flooding and spray. Additionally, the new installations met electrical separation and seismic II/I criteria.

Unreviewed Safety Question Evaluation #94-016

Description: The "light disc" turbine low pressure rotors have been replaced with "medium" and "heavy" rotors. The weights, exit velocities of disc fragments, etc., are not the same for the "medium" and "heavy" disc rotors.

Safety Evaluation Summary: The consequences of a turbine missile are determined by location of safety-related targets relative to the possible ejection angles, and the protection provided by barriers. There are no changes to the design of those discs that could credibly strike a safety-related structure.

Unreviewed Safety Question Evaluation #94-017

Description: The administrative functions of the Nuclear Safety Review Board have been deleted from the Updated Final Safety Analysis Report as the responsibilities of the Nuclear Safety Review Board are clearly delineated in the Technical Specifications.

Safety Evaluation Summary: Deleting the administrative functions of Nuclear Safety Review Board from the Updated Final Safety Analysis Report does not affect plant operations or configuration, and would not have an effect on any accidents or equipment malfunctions.

Unreviewed Safety Question Evaluation #94-019

Description: An evaluation was performed to confirm that safety analyses and parameters specified in the Updated Final Safety Analysis Report and Technical Specifications will be bounded during Cycle 4 of Unit 2 for Modes 2 and 1.

Safety Evaluation Summary: The existing safety analyses and parameters remain bounding for the entire cycle.

Unreviewed Safety Question Evaluation #94-020

Description: The hollow mounting studs on the Standby Diesel Generators have been replaced with solid studs to increase the allowable stress.

Safety Evaluation Summary: The increased allowable stress would decrease the probability that a stud failure would occur. If fuel line blockage or fuel injection pump seizure were to occur, the cam lobe would slip and prevent engine damage or fuel pump ejection.

Unreviewed Safety Question Evaluation #94-021

Description: The withdrawal schedule for irradiation specimens from the Unit 2 reactor vessel has been adjusted to reflect the inadvertent withdrawal of specimen "V" instead of specimen "U" at the first refueling outage for Unit 2.

Safety Evaluation Summary: The operability, reliability, and integrity of the Unit 2 reactor vessel are not affected by examination of the irradiation specimens on a different schedule.

Unreviewed Safety Question Evaluation #94-023

Description: Nuclear Purchasing and Material Management has been moved from the Engineering Department to the Plant Services Department

Safety Evaluation Summary: The change describes organizational restructuring. Organizational functions and personnel qualifications are not changed.

Unreviewed Safety Question Evaluation #94-025

Description: The valve trim was replaced on the Unit 2 Train "A" Auxiliary Feedwater flow control valve to reduce vibration at low-flow conditions. A test was performed to confirm the effectiveness of the modification.

Safety Evaluation Summary: Failure or loss of Auxiliary Feedwater flow with the plant in Mode 4 is backed up by the Residual Heat Removal system. Also, a pipe rupture in one of the four independent trains of the Auxiliary Feedwater system would not prevent safe shutdown of the plant.

Unreviewed Safety Question Evaluation #94-026

Description: The Emergency Plan was changed to incorporate terminology from the new 10CFR20, make minor changes to training, and correct editorial errors.

Safety Evaluation Summary: The changes do not affect equipment, equipment operating procedures, plant operations or maintenance and the effectiveness of the Emergency Plan has not been decreased.

Unreviewed Safety Question Evaluation #94-028

Description: In the Operations Quality Assurance Plan, the responsibility for review of "commercial requirements" has been removed as it does not relate to quality. Responsibility for nuclear fuel procurement has been clarified with respect to the relocation of Nuclear Purchasing and Material Management from the Engineering Department to the Plant Services Department

Safety Evaluation Summary: The change relates to organizational restructuring. Organizational functions and personnel qualifications are not changed.

Unreviewed Safety Question Evaluation #94-029

Description: The ability of the plant to achieve and maintain safe shutdown in the event of a load drop and/or crane collapse was evaluated because installation of new chillers required lifting heavy loads over the Mechanical Electrical Auxiliary Building.

Safety Evaluation Summary: The installation instructions restricted the height of the load lift and restricted movement over and around buildings to bound the effect of a load drop and to prevent the collapse of the crane on safety-related targets. These restrictions maintained the integrity of the Mechanical Electrical Auxiliary Building and other safety-related components.

Unreviewed Safety Question Evaluation #94-035

Description: The Updated Final Safety Analysis Report has been updated to reflect changes associated with the new 10CFR20, Standards for Protection Against Radiation, which was implemented on January 1, 1994 via a Nuclear Regulatory Commission approved Technical Specification change. In addition, operational practices were updated and unnecessary detail was removed from chapters 11 and 12.

Safety Evaluation Summary: The changes deal with radiation protection and waste management details that are not associated with initiation of an accident or malfunction of equipment important to safety.

Unreviewed Safety Question Evaluation #94-037

Description: The Updated Final Safety Analysis Report description of reactor vessel material properties, pressure-temperature operating limits, and pressurized thermal shock was updated to reflect the results of current reactor vessel surveillance capsule analyses and to correct design basis information.

Safety Evaluation Summary: The reactor vessel surveillance capsule analyses demonstrate that the structural integrity of the reactor vessel beltline has not changed. The pressure-temperature limits specified in Technical Specification 3/4.4.9 remain bounding.

Unreviewed Safety Question Evaluation #94-038

Description: The potential for weld cracking in the Essential Cooling Water system was evaluated and a description of the compensatory inspections and monitoring performed has been added to Updated Final Safety Analysis Report Section 9.2.

Safety Evaluation Summary: The compensatory inspections and monitoring ensure that cracks in the Essential Cooling Water system would be identified and repaired prior to the cracks reaching a size where failure could be credibly postulated. Even with small flaws, the Essential Cooling Water system is able to perform its design function.

Unreviewed Safety Question Evaluation #94-043

Description: Several "key" management positions were filled with new personnel. Also, Shift Technical Advisors now report to the Plant Operations department.

Safety Evaluation Summary: The new managers exceed the education and experience requirements of Regulatory Guide 1.8, "Personnel Selection and Training". Also, the Shift Technical Advisors are performing the same duties and responsibilities as before.

Unreviewed Safety Question Evaluation #94-044

Description: The Updated Final Safety Analysis Report description of the low lube oil trip has been corrected to match the design. The Standby Diesel Generator is stripped from the electrical bus if low lube oil pressure is sensed and the engine is stopped using air to isolate the fuel supply.

Safety Evaluation Summary: The Standby Diesel Generator does not impact the initiation sequence of any accident previously evaluated. A malfunction of a Standby Diesel Generator is considered in the accident analyses, but the probability and consequences of that malfunction are not affected by the method by which the low lube oil trip functions.

Unreviewed Safety Question Evaluation #94-045

Description: Change the description of the Reactor Coolant Vacuum Degassing system to reflect that the system is not used during every plant shutdown.

Safety Evaluation Summary: The Reactor Coolant Vacuum Degassing system is intended to reduce primary coolant activity prior to removing the reactor vessel head. Not using the system during every shutdown would be an ALARA consideration and does not affect the initiators or consequences of an accident and would not affect equipment important to safety.

Unreviewed Safety Question Evaluation #94-101

Description: A valve in a Turbine Overspeed Control system was gagged open to reduce the hydraulic oil pressure to zero for personnel safety while a leak was being repaired in an unisolable hydraulic line.

Safety Evaluation Summary: The probability and consequences of a turbine overspeed event were not increased because only one of five overspeed control/trips were rendered inoperable for the repair. In addition, an operator was stationed at the turbine control panel to trip the turbine on a loss of turbine load and a six-hour time limit was established for the repair, which is consistent with the Technical Specifications allowed outage time.