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

Subject: **Docket Nos. 50-361 and 50-362**
Facility Change Report
San Onofre Nuclear Generating Station Units 2 and 3

Dear Sir or Madam:

This letter transmits the Facility Change Report required by 10 CFR 50.59(b)(2) for San Onofre Units 2 and 3 for the period from February 4, 2001 through April 30, 2003. The report (Enclosure 1) provides a summary of the facility changes, procedure changes, and any tests and experiments, including a summary of the safety evaluations performed for each change. The scope of this report is based on an extensive review of plant records and all 50.59 evaluations identified for the time period above are included in this report. The report also includes five 50.59 evaluation summaries omitted from the previous Facility Change Report. Complete facility change documentation is available onsite.

Enclosure 2 provides a report on commitment changes made per NEI "Guidelines for Managing NRC Commitments."

If you would like any additional information, please contact Mr. J. L. Rainsberry at (949) 368-7420.

Sincerely,

Enclosures

cc: T. P. Gwynn, Acting Regional Administrator, NRC Region IV
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ENCLOSURE 1

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

FACILITY CHANGE REPORT (FCR)

FOR THE PERIOD
FROM FEBRUARY 4, 2001 THROUGH APRIL 30, 2003

AR 971101489-10: Height of Units 2 and 3 Fire Hose Connection Standpipes

Description:

The hand wheels for some fire hose connection standpipe valves located in the Unit 2 and Unit 3 Control Building cable riser galleries are located above the National Fire Protection Association (NFPA) code requirement maximum height of sixty inches above the floor surface.

Evaluation Summary:

A 10 CFR 50.59 Evaluation was prepared for Updated Fire Hazards Analysis (UFHA) Change Request ACN 18 to Appendix D. The UFHA was updated to address standpipe hose connections and hose connection valves located in excess of the required NFPA distance from the floor. Adequate provisions were made in site procedures to access these particular valves to provide a greater degree of access for fire brigade personnel. This will ensure that the fire protection features required to protect the plant from the hazards of fires are accessible for the fire brigade members. In addition, no personnel hazards are introduced. Based on this evaluation it was concluded, that the UFHA document update that will address this exception to compliance does not require prior NRC approval.

AR 980100417-17: Removal of Units 2 and 3 Technical Specification Bases Automatic Boration Requirement

Description:

This evaluation examined Technical Specification Bases change B00-029. This change revised the Bases for the Limited Condition for Operation (LCO) "Boration System - Operating," to delete the requirement for automatic boration by the Chemical and Volume Control System (CVCS) Charging pumps from a Safety Injection Actuation Signal (SIAS). This change was performed under 10 CFR 50.59 to ensure that no changes outside the scope of Amendments 163 and 154 were made without due consideration. No physical change was made to the plant design, the safety analysis, or to the plant operating procedures. No current Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis credits automatic boration to mitigate the consequences of a Main Steam Line Break (MSLB) event.

Evaluation Summary:

In previous plant operating cycles the MSLB post-trip return to power accident used automatic boration upon Safety Injection Actuation Signal (SIAS) to mitigate the consequences of this event. However, in Cycle 11, specific credit for automatic boration into the Reactor Coolant System (RCS) was not credited to mitigate the consequences of this event. The results of this MSLB event remained within the NRC approved criteria.

The proposed changes were consistent with the design basis information contained in the UFSAR and with NRC approved removal of reference to a SIAS start signal for CVCS by Technical Specification Amendment.

AR 980600386-03: Units 2 and 3 Diesel Generator Head Loss and Back Pressure Values Calculation M-0015-006

Description:

During the Design Bases Documentation (DBD) research for the Diesel Generator, calculation M-0015-006 was revised to evaluate the effects that missing diesel exhaust enclosure boxes (or panels) had on the surrounding concrete plenum and structures, and evaluated engine backpressure effects on the diesel engine. The calculation used 58,000 cubic feet per minute (cfm) as the assumed airflow per Heating and Venting Air Conditioning (HVAC) exhaust duct. The airflow rates used were based on old data rather than the updated HVAC sizing calculation. The "exhaust" air volume flow rate was actually higher than the "supply" airflow rate, as documented in Calculation M-0076-041. The results in calculation M-0015-006 (head loss and back pressure values) were expected to be slightly higher than currently indicated.

The Configuration Change Notice (CCN) to calculation M-0015-006 revised it by using the following inputs:

- Preoperational test data for the emergency supply fans per calculation M-0076-040, Revision 2,
- Updated exhaust temperatures for the 16-cylinder and 20-cylinder engine exhaust piping/plenum based on input provided by the current diesel engine vendor,
- Diesel engine exhaust piping elbow terminations with all 90-degree fittings oriented vertically upwards and centered at the bottom of the plenum, and
- Zero entrainment of building ventilation air in the diesel engine exhaust air plenum.

Evaluation Summary:

Incorporation of the above inputs in calculation M-0015-006, Rev. 0, CCN N-1 provided the following results:

- Lower plenum hot side and lower cold side concrete surface temperatures compared to the acceptable temperature value established in Rev. 0,
- Lower pressure drop across the 16-cylinder engine, but higher pressure drop across the 20-cylinder exhaust piping/plenum arrangements compared to the values in Rev. 0, and
- Re-confirmed that fuel consumption per train was within the allowable value.

This activity was determined to not require prior NRC approval.

AR 980603194-03: Units 2 and 3 Fire Detection Instrumentation Required Action Completion Times and Terminal Actions

Description:

Proposed Licensee Controlled Specifications (LCS) change L01-006 deleted the LCS requirement to provide compensatory fire hose within 24 hours if an inoperable hose station outside containment is not the primary means of fire suppression. This resulted in a requirement to provide a compensatory hose within 1 hour. Credit was allowed for the fire hose provided by the fire department in responding to a fire.

Evaluation Summary:

The proposed change made the LCS requirement more stringent since a fire hose must be provided within 1 hour versus within 24 hours. Therefore, prior NRC approval is not required.

AR 980701536-31: Evaluation of the Impact on Units 2 and 3 from Removal of Guard Towers and Installation of a Closed Circuit Television (CCTV) System

Description:

This evaluation was for Facility Change Notice (FCN)-24824E associated with removing Guard Towers from the Protected Area (PA) Boundary, installing a CCTV System, with the associated lighting, cameras, and alarm panels, and eliminating the PA Boundary around Unit 1. The installation includes new lighting poles, fixtures, cameras, console workstations, and other related security changes.

The relocation of the Unit 1 PA Boundary and installation of CCTV to replace the guard towers is addressed by the 10CFR50.54p change process. This evaluation is intended to address the impact of the changes on Units 2 and 3 plant systems.

Evaluation Summary:

Changes to the PA Boundary are implemented through several design packages that include: CCTV system installation, new lighting poles, replacement of existing light fixtures, installation of supporting equipment and security consoles, and the connection of a reliable power source to the system. The CCTV system replaces the existing guard towers with cameras mounted along the PA perimeter. NRC approval of exemption requests for the Unit 1 PA was received in advance of the proposed modification. Security compensatory measures were in place for construction along the new PA Boundary in accordance with the site-specific security plan. The CCTV cameras and equipment are powered from the San Onofre Nuclear Generating Station (SONGS) electrical systems. Power supply loadings have been analyzed and determined to be within acceptable limits. The installation of the lighting will not interfere with Units 2 and 3 system operations, or impact any Unit 2 or Unit 3 safety systems. The electrical supply for the lights and cameras is not credited for accident mitigation and is within the allowable capacity of the SONGS electrical system. The proposed changes to the security system affecting Units 2 and 3 have been determined to not require prior NRC approval and do not require a change to the Technical Specifications.

AR 990402371-28: Unit 3 Reactor Coolant Pump (RCP) Anti Reverse Rotation Device (ARRD)

Description:

This evaluation was for a Non-Conformance Report (NCR) accept-as-is disposition for configuration of a Unit 3 RCP ARRD. The ARRD is important to safety and credited in the safety analysis. The ARRD's temperature indication, functional test during the recent shutdown, and oil testing support that the ARRD was not degraded.

Evaluation Summary:

This activity does not affect the operation or design basis of the reactor coolant system and/or reactor coolant pump. The ARRD is credited in the safety analysis and remains operable and within its design basis. Although the proposed change will render the RCP 3P002 Lube Oil Flow Low Annunciator inoperable, it does not monitor a key variable as defined in Regulatory Guide (R.G.) 1.97, it is not relied upon to maintain the plant in a safe shutdown condition, nor is it required to prevent, or mitigate an accident. This NCR installed a blank orifice plate in the instrument loops of the motor, which will direct additional oil flow to the ARRD to increase the oil supply to the ARRD. The design bases assumed in the Updated Final Safety Analysis Report (UFSAR) remain unchanged. No change has been made to the plant function or design bases.

AR 990800890-02 and –03: Units 2 and 3 Fire Detection Instrumentation Required Action Completion Times and Terminal Actions

Description:

These evaluations were for Licensee controlled Specification (LCS) change L01-007 to LCS 3.3.106 to restore periodic completion times for containment inspection and temperature monitoring, upon inoperability of one or more early warning detectors for a containment fire area zone (AR 990800890-2), and also to provide terminal actions when certain required actions or completion times are not met (AR 990800890-3).

Evaluation Summary:

This change makes the LCS 3.3.106 required actions for containment inspection and temperature monitoring more conservative by requiring periodic actions, rather than one-time action, and also makes LCS 3.3.106 more conservative by requiring terminal actions where none previously existed.

AR 991000624-03: Units 2 and 3 Salt Water Cooling (SWC) Pump Column Bearings

Description:

This change replaced solid tin-bronze bearings with composite bearings consisting of a tin-bronze inner bearing and a non-metallic "sleeve," such that there is no direct contact of tin-bronze material with stainless steel material in seawater. This eliminated the galvanic couple at the inner column/bearing interface. "Micarta Grade 97, NIMA Grade LE" was selected for the bearing casing material. This change eliminates degradation of the SWC pump inner columns and column bearings by erosion/corrosion.

Evaluation Summary:

The modification replaced the original pump bearings with functionally equivalent bearings that are less sensitive to degradation by erosion/corrosion. The change improved the equipment's response; therefore, prior NRC approval was not required. To validate mechanical properties of the casing material, an actual size prototype bearing was fabricated and successfully tested in seawater. The objective of the test was to determine the maximum load at which the non-metallic casing threads would strip. The SWC pump Original Equipment Manufacturer (OEM) reviewed and approved the proposed material and the design of the new composite bearing. The tests confirmed that the new composite bearings will perform satisfactorily in the intended application, and the failure of the new bearing is no more likely than the failure of the bearing being replaced. The new bearings were manufactured and installed as Quality Class (QC) II components, consistent with the quality classification of the bearings being replaced. The new bearings are considered functionally and structurally equivalent to the bearings being replaced.

AR 991000874-22: Test to Connect the Unit 2 Fire Protection System to the Screen Wash System

Description:

This activity tested a connection of the Unit 2 fire protection system to the screen wash system to demonstrate the ability of the fire protection system to provide a temporary water supply for screen wash operation. The temporary supply needed to be maintained while maintaining the design basis of the fire protection water supply. The temporary supply of water to the screen wash system is required to permit clearing the screen wash pumps for electrical modifications.

The fire protection system was connected to the Unit 2 screen wash system via two hoses from fire hydrant #6 to a specially configured flange on the screen wash system header. The flange was fitted with a temporary screen wash header supply valve. The screen wash pumps were taken out of service and the two electric motor driven fire pumps started. The diesel driven fire pump remained in standby. Both valves at the hydrant were opened as well as the temporary screen wash header supply valve. Screen and rake sets were started and the screen washing operation was observed. When it was determined that the objectives of the test were met, both systems were restored to their original configuration and the screen wash system returned to service.

Evaluation Summary:

The design basis for the fire protection water supply system is contained in the Updated Final Safety Analysis Report (UFSAR), section 9.5.1 and also in the Updated Fire Hazards Analysis (UFHA). In order to avoid more than minimal adverse effects on the design basis of the fire protection water supply system, San Onofre Nuclear Generating Station provided certain compensatory actions while conducting the test. Fire hydrant #6 was to be continuously manned so that in the event of a fire, the two valves on the hydrant could be closed. The service water storage tanks were to be monitored for draw down to ensure tank level did not go below Licensee Controlled Specification requirements (83% of the tank's capacities). The screen wash pumps were de-energized before opening the temporary screen wash header supply valve to ensure that no salt water could be introduced into the fire protection system water supply. All three fire water pumps were required to be operable prior to the test to ensure an adequate supply of water for both screen wash operations and fire protection water supply system design bases. The diesel engine driven fire water pump by itself is capable of supplying water to meet the requirements of the design basis for the fire protection water supply system. The fire protection system loop header is sized to accommodate the flow from both screen wash water supply and fire protection water supply system design basis. Based on this data, prior NRC approval was determined to not be required.

AR 991000874-24: Change to Enable Connection of the Unit 2 Fire Protection System to the Screen Wash System During Motor Starter Replacement

Description:
This activity uses the fire water system for an emergency backup water supply for the screen wash system during installation of motor starters for the screen wash pumps. The motor starter installation takes one pump out of service at a time, leaving only one pump in service. If the pump in service fails during a heavy influx of debris in the intake, the fire water system will be used to supply screen wash water until the failed pump can be repaired or replaced. Use of the fire water system for emergency backup was controlled as described in the evaluation below. A previously performed test (AR 991000874-22) demonstrated the fire protection system's capability to provide a temporary water supply for screen wash operation, while maintaining the design basis of the fire protection water supply. The fire protection system was connected to the Unit 2 screen wash system via hoses from fire hydrant #6 to a specially configured flange on the screen wash system header. The flange was fitted with a temporary screen wash header supply valve. If the fire protection system is put in service to supply screen wash water, the screen wash pumps will be verified to have been taken out of service and the two electric motor driven fire pumps will be started. The diesel driven fire pump will remain in standby. Both valves at the hydrant will be opened as well as the temporary screen wash header supply valve. The screen wash cycle will commence by manually washing one screen and one rake at a time until the cycle is complete. Manual screen washing will continue until the failed screen wash pump is operable. Once the work to install the screen wash pump motor starters is completed, both systems will be restored to their original configuration and the screen wash system will be returned to service.

Evaluation Summary:
The design basis for the fire protection water supply system is contained in the Updated Final Safety Analysis Report (UFSAR), section 9.5.1 and also in the Updated Fire Hazards Analysis (UFHA). In order to avoid more than minimal adverse effects on the design basis of the fire protection water supply system, San Onofre Nuclear Generating Station used certain compensatory actions. These same actions were used in the feasibility test performed as a part of this activity. Fire hydrant #6 was to be continuously manned so that in the event of a fire, the two valves on the hydrant could be closed. The service water storage tanks were to be monitored for draw down to ensure tank level did not go below requirements. The screen wash pumps were de-energized before opening the temporary screen wash header supply valve to ensure that no salt water could be introduced into the fire protection system water supply. All three fire water pumps were required to be operable to ensure an adequate supply of water for both screen wash operations and fire protection water supply system design bases. The diesel engine driven fire water pump by itself is capable of supplying water to meet the requirements of the design basis for the fire protection water supply system. The fire protection system loop header is sized to accommodate the flow from both screen wash water supply and fire protection water supply system design basis. Based on this data, prior NRC approval was determined to not be required.

AR 991001327-8: Reanalysis of Units 2 and 3 Post Accident Containment Transient Pressure and Temperature Response.

Description:

Southern California Edison performed a recalculation of the containment transient pressure and temperature response to a spectrum of design basis large main steam line breaks (MSLBs) documented in Revision 1 to calculation N-4080-027. The reanalysis was originally undertaken as a part of the reduction in Tcold project for San Onofre Nuclear Generating Station Units 2 and 3 and included updated plant design and performance input data, consideration of instrumentation total loop uncertainties (TLUs), and updated passive heat sink data. Mass-energy release calculations by the Nuclear Steam Supply System (NSSS) vendor, ABB-Combustion Engineering (CE), confirmed that the original Tcold, still permitted by Technical Specification Limiting Condition for Operation (LCO) 3.4.1, was more limiting for containment Pressure-Temperature (P-T) response; therefore, the new analysis continues to be based on a maximum Tcold of 560F.

Evaluation Summary:

The revision of the containment post-MSLB pressure-temperature response calculation resulted in a change in the worst case MSLB event from the previously identified 102% power MSLB with cooling train failure to the 102% MSLB with Main Steam Isolation Valve (MSIV) failure. The peak containment post-MSLB pressure was reduced from 56.6 pounds per square inch gauge (psig) to 56.5 psig, and the peak vapor temperature was reduced from 428 degrees Fahrenheit (F) to 409F. The peak vapor temperature for in-containment equipment qualification was reduced from 407F to 405F. The lower peak containment pressure and temperature reduce the consequences of the design basis MSLB event, and increase the margin of safety. The radiological consequences of design basis MSLB events in containment remain unchanged from those based on the previous analysis of record.

AR 991101109-30: Addition of a Backdraft Damper (BDD) to Units 2 and 3

Description:

Southern California Edison added a Quality Class (QC) II BDD on the discharge side of the Control Room Emergency Fans A206 and A207 to minimize reverse rotation of the associated fan, which was causing air leak by between the normal HVAC system and these standby fans.

Evaluation Summary:

The function of the added BDD is to:

(1) Close on the back flow from normal operating AC supply system to minimize the reverse rotation of the associated fan, and

(2) Open when the associated fan (A207 or A206) is in operation to route supply air to the control room.

Fan A207 (or A206) does have a safety function credited in the event/accident analyses, as discussed in Sections 9.4.2, 9.5.3 and 15 of the Updated Final Safety Analysis Report (UFSAR). This evaluation confirmed that there is no safety function impacted by adding this BDD. Similar to supported component (fan A207 or A206), the BDD was procured as Quality Class II, Seismic Category 1 to meet and retain the same functional and design requirements. The weight of the added BDD in train A and train B is heavier than the existing duct sections of the same length. Structural analysis demonstrates that all seismic requirements are satisfied, and the existing duct supports are adequate. While the pressure drop of the added BDD is higher than that of the existing duct section of the same length, its pressure drop in the open position is insignificant compared to the available pressure drop margin. In the unlikely event that the BDD fails, its failure would cause the impact identical to the failure of the fan itself, which was evaluated previously in the UFSAR. Therefore, prior NRC approval was not required for this proposed activity.

AR 000300835-8: Units 2 and 3 Backup Control Element Drive Mechanism Control System (CEDMCS) Cabinet Cooling Unit

Description:

Southern California Edison added a temporary 5-ton backup CEDMCS cabinet cooling unit to maintain the CEDMCS cabinet at reasonable temperature levels when installed air conditioner equipment is being serviced/maintained, or is out of service. The temporary unit is staged at a near by location until needed to support maintenance activities. At that time, the 5-ton unit will be wheeled to its specified location, anchored in place, and connected to the system.

Evaluation Summary:

The temporary backup cooling unit was sized to maintain the CEDMCS cabinet at acceptable temperature levels, when required, to operate together with one permanently installed air conditioning unit during maintenance activities. Cooling air from the temporary back-up cooling unit, as well as the permanently installed air conditioning units, is supplied through the computer floor cavity and directed to the CEDMCS cabinet to pick up the heat generated in the cabinet. Hot air is extracted from the top of the cabinet by means of several fans and directed to the backup cooling unit and to the permanently installed air conditioning unit. An evaluation to determine the temperature in the Unit 3 CEDMCS cabinet during this scenario provided values that are within the acceptable limits of 75° Fahrenheit (F), +/- 10° F normal operating temperature and 104° F maximum for 8 hours. This evaluation was based on actual temperatures of adjacent areas during normal plant operation. This is not considered a design basis scenario since the temperature values used in the evaluation were actual values that were lower than those used in the design basis calculation (generally 104° F). When the backup cooling unit is required to be in service, it will be wheeled to its designated staging location and anchored in place. A calculation evaluated the structural adequacy of the seismic restraint assembly for the loads resulting from the backup cooling unit skid and its panel assembly. When the backup cooling unit is stored or staged (Unit 2 or Unit 3) its restraints and concrete anchors satisfy requirements. This activity does not alter the design, function, or method of performing the function of structures, system or components described in the Updated Final Safety Analysis Report (UFSAR). This activity provides better operating conditions in the CEDMCS cabinets/rooms when normal plant equipment (air conditioning unit) is being serviced/maintained, thus minimizing degradation of CEDMCS components. It is concluded that this activity does not require prior NRC approval.

AR 000401086-66 and –67: Removal and Addition of Insulation on Units 2 and 3 Pipes

Description:

This Facility Change Notice (FCN) package covers the work necessary to remove and install insulation on pipe sections in Units 2 and 3 Pump Rooms 005. The work scope included removal of unprotected and degraded Cera blanket insulation from Low Pressure Safety Injection (LPSI) suction piping in both Units. This removal is approximately an eight-foot run of suction piping in each unit. Also included was the installation of insulation in the tunnel approaching Room 5.

Evaluation Summary:

This design change involves removing and installing insulation on pipes. It does not change the piping system's design function. The proposed change meets the design, material, and construction standards applicable to the affected piping system. The modified configurations have been verified by piping stress calculations in accordance with NRC approved criteria and the applicable American Society of Mechanical Engineers (ASME) code requirements. Based on a thermal analysis, the proposed change will not affect the pump room temperatures for which the equipment has been qualified to operate. It is concluded that this activity does not require prior NRC approval.

AR 000801574-03: Additions to the Barrier Procedure for Units 2 and 3

Description:

Procedure SO23-XIII-57, Barrier Inspection, provides criteria for performing inspections of fire rated barriers separating fire areas with redundant equipment required for safe shutdown, and/or fire barriers separating defined areas of the plant, which have specific fire rating requirements. These barriers include walls, floors and ceilings constructed of concrete and/or plaster. This evaluation is for additions to the procedure, which specifically address all concrete and plaster barrier assembly degradations that do not adversely affect the fire rating of the barrier assemblies.

Evaluation Summary:

The operability of a fire barrier ensures that fire damage will be confined to a single fire area. Many concrete fire barriers also provide structural support functions. As such, the concrete walls, floors, and ceilings generally exceed the minimum thickness required to meet their respective fire rating. During initial construction and periodic maintenance activities, unused holes in concrete barriers are left abandoned in place. These holes are not through penetrations. Rather, they are generated to accommodate an anchor bolt or other mounting devices. These degradations are usually shallow, up to a few inches in depth. Plaster barrier openings are allowed on one side of a barrier for electrical boxes, fire sprinkler heads, etc. These membrane openings are limited to 16 square inches in area and the aggregate area of the boxes is not to exceed 100 square inches for any 100 square feet of wall area. Larger openings are required to be protected with suitable fire stops or be installed per manufacturer's instructions for such use. Critical areas bounded by fire rated plaster assemblies have low in-situ combustible loading, early warning detection, and suppression systems located either area wide or directly over the hazard. Surface degradations and membrane penetrations in plaster walls and partitions discovered during a barrier inspection are primarily due to maintenance activities where a cart or tool has scraped or gouged the plaster. These degradations are generally at a low elevation on the wall, far from any potential hot gas layer that would be present during a fire event. Thus, there is reasonable assurance that any potential fire event would be detected during its incipient stages, and confined to a single area/zone and extinguished promptly by the on-site fire department. No adverse impact to safe shutdown is introduced due to minor degradations in plaster wall or partition assemblies and compensatory measures are required for inoperable fire barrier assemblies. Discovered degradations and membrane openings in plaster wall assemblies will continue to be reworked or repaired. Barrier protection to support other hazard protection, including missiles, flooding, radiation, toxic gases, security, high energy line breaks (HELB), and Equipment Qualification (EQ) were evaluated to be acceptable. It was concluded that these modifications to the revised SO23-XIII-57 procedure did not compromise the fire protection system's ability to ensure that adequate fire rated assemblies remain operable to separate areas containing safe shutdown and/or safety related equipment. It is concluded that this activity does not require prior NRC approval.

AR 000900187-02: Removal of Units 2 and 3 Component Cooling Water (CCW) Pumps Low Discharge Pressure Start Circuits and Addition of Safety Injection Actuation Signal (SIAS) Indicating lights

Description:

This design change removes the low discharge pressure start circuit from the control circuits of the CCW pumps and adds a SIAS indicating light on each of the 4KV switchgear cubicle doors for the CCW pumps. Since one pressure instrument loop is common to two pumps of the same train providing low pressure interlock, a sequence of order of Facility Change Notice (FCN) implementation was followed to maintain operability of the swing CCW pump P025 when the FCNs for pumps P024 or P026 were implemented. An indicating light was installed on the switchgear cubicle door for each pump to indicate SIAS actuation. Bypassing the auto-start feature of the swing CCW pumps allows more even run times.

Evaluation Summary:

There is no benefit from starting the opposite loop pumps on a loss of pressure since it does not restore flow in the loop experiencing low pressure. Also, the low-pressure auto start feature is not required to mitigate the consequences of an accident and the pumps do receive a SIAS. During normal operation, one critical loop is in operation and the non-critical loop is aligned to the operating critical loop. Operator action is required to align and transfer the non-critical loop to the standby loop in case of low pressure in the operating loop. Therefore, auto start of the standby loop pump and the swing pump do not serve any purpose during normal plant operation. Manual start and stop of the CCW pump at operator's discretion to meet the plant needs will allow equal run time of the pumps and thereby reduce wear and tear of the pump. Local indication of actuated SIAS is beneficial to plant operation and maintenance. Deletion of low discharge pressure auto-start feature and addition of a local SIAS indicating light for the CCW pumps do not require prior NRC approval because the auto-start feature does not perform a safety function and the indicating light does not impact the safety function of the CCW system.

AR 000900535-15: Units 2 and 3 Licensee Controlled Specifications (LCS) Fire Suppression Water System Changes

Description:

This evaluation analyzed the acceptability of modifying the Unit 2 and Unit 3 Licensee Controlled Specifications (LCS) 3.7.105, "Fire Suppression Water System." Many Required Actions of this LCS are being removed and referred to site procedures for expanded clarification, equipment lists, and engineering evaluations. As a result, sections of this LCS are being modified to ensure that adherence to San Onofre licensing commitments are procedurally accomplished in the current working environment.

Evaluation Summary:

These modifications to LCS 3.7.105 do not compromise the fire protection system's ability to control and extinguish fires in areas containing safe shutdown and/or safety related equipment. Southern California Edison (SCE) is permitted to make changes to the approved fire protection program without prior NRC approval if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire; this criterion is satisfied by the evaluation.

AR 000901044-06: Units 2 and 3 Updated Final Safety Analysis (UFSAR) changes to Section 9.5.2 "Communication System"

Description:

This is an evaluation of UFSAR change 9.5-14 to revise section 9.5.2 "Communication Systems" with regard to the security force communications sections and the fire related communications section. These UFSAR revisions reflect recent changes made to Unit 1 under the "cold and dark" decommissioning activities.

Evaluation Summary:

The proposed activity corrects UFSAR descriptions of the security communications section and fire related communications section to reflect the described changes. The proposed activity revises the UFSAR description of the security communications section to be consistent with the Physical Security Plan. The fire related communications section remains consistent with the Updated Fire Hazard Analysis. The proposed changes do not impact the Technical Specifications or impact any accident basis scenarios. It is concluded that this activity does not require prior NRC approval.

AR 001201167-08: Units 2 and 3 Updated Final Safety Analysis (UFSAR) changes to Steam Generator Sections 10.4.8.2.3 and UFSAR Table 5.4-4

Description:

This change revises UFSAR sections 10.4.8.2.3 "Steam Generator Blowdown Processing System Operation" and UFSAR Table 5.4-4 "Steam Generator Parameters" to be consistent with the current normal operation of the system with blowdown flows up to 250 gallons per minute (gpm) in place of 150 gpm. The change also increases the maximum blowdown flow to 275 gpm versus the previously identified maximum operating flow of 200 gpm. The new values support proper chemistry control in the steam generators and are consistent with current operation of the blowdown system.

Evaluation Summary:

Per UFSAR section 10.4.8.1 Design Bases, the blowdown system has three design bases:

A. The system provides isolation valving on lines leaving the containment
B. The system from the steam generators up to and including the isolation valving is designed to remain functional after a design basis earthquake.
C. "The steam generator blowdown system, in conjunction with the condensate and feedwater chemical injection system, as described in section 10.4.10, and the Full Flow Condensate Polisher Demineralizer (FFCPD), as described in section 10.4.6, is capable of maintaining the chemical composition of the steam generator secondary water within the limits specified in table 10.4-5. While no significant primary to secondary leakage is assumed in normal operation, the blowdown demineralizer can remove radioisotopes from a small amount of primary -to-secondary leakage. Additionally, while no small amount of primary -to secondary leakage is assumed in normal operation, a moderate amount of condenser in leakage can be handled by the FFCPD."

The proposed change in flow rates only impacts design bases "C"

The UFSAR Safety Evaluation 10.4.8.3 C states: "Design basis C is unrelated to nuclear safety and evaluation is, accordingly, not presented." Blowdown system operation is unrelated to nuclear safety. However, blowdown does have an important to safety design bases to be capable of being isolated upon receipt of an Emergency Feedwater Actuation Signal (EFAS) or Main Steam Isolation Signal (MSIS). The new flow rates are consistent with the Nuclear Steam Supply System (NSSS) vendor's (Combustion Engineering [CE]) analysis and with the normal operating range for Core Operating Limit Supervisory System (COLSS). Further, although the increased flow increases the amount of potential diversion of condensate for cool down, should blowdown fail to isolate upon receipt of an EFAS or MSIS, this scenario has been reviewed against existing calculations for impact on condensate storage requirements to support cool down. The applicable calculations demonstrate that the condensate storage tank continues to retain sufficient water to meet its design bases for plant cool down. It is concluded that this activity does not require prior NRC approval.

AR 980900701-05: Units 2 and 3 Emergency Chillers Operability

Description:

This evaluation supported Technical Specification Bases change B01-003, which clarified that an emergency chiller may be considered operable during the time period when it is being transferred between operable Component Cooling Water (CCW) critical loops, provided that the transfer operation is completed in less than 2 hours. The previous Technical Specification Bases wording required that an emergency chiller must be aligned to an operating or standby operable CCW critical loop for operability. This clarification allows flexibility for transferring between CCW critical loops.

This change likewise clarified that an emergency chiller may be considered operable during the time period when it is being transferred between energized 4 KV buses, provided that the transfer operation is completed in less than 2 hours. The Bases was previously silent on emergency chiller operability during power supply transfers

Evaluation Summary:

Engineering provided a detailed basis for concluding that equipment supported by the Emergency Chilled Water (ECW) system will remain functional without an ECW cooling supply for at least 2 hours. Providing the supported equipment remains capable of performing its specified safety function, the emergency chiller may be considered operable. It is concluded that this activity does not require prior NRC approval.

AR 990201303-06: Units 2 and 3 Containment Penetration Conductor Over current Protection Devices

Description:

This licensee Controlled Specification (LCS) change, number L00-013, revised LCS 3.8.100, "Containment Penetration Conductor Over current Protective Devices" for clarification and to provide operational flexibility. The following changes were made.

1. Clarified that the operable device must be used to de-energize the affected circuit with one or more of the required containment penetration conductor over current protective device(s) inoperable.

2. Clarified that the inoperable device, rather than the backup device, must be verified tripped, removed, or racked out.

3. Revised an action completion time from 7 days to once per 7 days, to make it an ongoing, rather than a one-time verification.

4. Clarified completion time for additional 10% surveillance samples which was previously open-ended.

5. Permitted temporary realignment of protective devices under administrative control without revising the LCS Over current Protective Devices Table, and

6. Deleted metal case circuit breakers from the Bases of Surveillance Requirements.

Evaluation Summary:

It was determined that the above described changes 1 through 4 were more restrictive, that change 5 had no effect on the level of protection afforded by the containment penetration conductor over current protective devices, and that change 6 was editorial. It is concluded that this activity does not require prior NRC approval.

AR 990801012-02: Units 2 and 3 Fuel Handling Isolation Signal (FHIS)

Description:

Licensee Controlled Specification (LCS) change L00-011 deleted section 3.3.112 in its entirety. LCS 3.3.112 required one FHIS channel to be operable whenever irradiated fuel was stored in the spent fuel pool (SFP) and required no movement of irradiated fuel assemblies. This change allows no operable FHIS channels during periods when there is spent fuel in the SFP and no spent fuel movement is taking place. During spent fuel movement, Technical Specification 3.3.10 will continue to apply to satisfy this requirement.

Evaluation Summary:

Deleting the LCS requirement for one operable FHIS channel during times when no spent fuel is being moved does not require prior NRC approval because the relevant accidents evaluated in the safety analysis report do not take credit for the post accident cleanup units actuated by FHIS.

AR 991200426-49 and 97: Units 2 and 3 Closed Circuit Television (CCTV)

Description:

These Facility Change Notices (FCN F22710E and F24506E) installed CCTV equipment into the Central Alarm System (CAS), Secondary Alarm System (SAS), Post 39, and the security storage area of the Service Building near the South Security Processing Facility (SSPF) to monitor Protected Area (PA) boundary cameras that will be installed under a separate design change. This change involved removing the old CAS/SAS/SSPF consoles and incorporating some of the old equipment into the new consoles. The CAS, SAS, Post 39 and SSPF service building was updated with new more efficient monitoring consoles. The additional CCTV equipment is necessary to provide adequate viewing of the security boundary once it is relocated per the security boundary relocation project. The impact of this change on security requirements as defined in the Physical Security Plan (PSP), Safeguards Contingency Plan (SCP), and the Security Force Training and Qualification Plan (T&Q) was addressed through the 10CFR50.54P process. This safety evaluation focuses on the power, seismic, and Heating and Venting Air Conditioning (HVAC) considerations involved with this design change.

Evaluation Summary:

The CCTV draws power from existing non-1E Uninterruptible Power Supply (UPS) bus. It was verified that there was margin remaining on the non-1E UPS, which is isolated from class 1E bus 2A04 via a qualified breaker. The additional heat load to the CAS was calculated to be acceptable. The addition of the new equipment was determined to not exceed the total allowable weight for the seismically mounted panel. Addition of the materials to the Fire Areas/Zones for the CAS, SAS, Post 39, security storage area, and U3 Diesel Generator (DG) building were within steel enclosures or conduits. The combustible loading created by cable insulation was not required to be included in the in-situ combustible loading calculations in the Updated Fire Hazards Analysis (UFHA). Thus fire loading remains below the maximum permissible for the applicable zones, and no fire hazards were being created. It is concluded that this activity does not require prior NRC approval.

AR 000100837-02: Non-Installation of South Yard Facility (SYF) Beta Booths

Description:

Design Change Package (DCP) A-7022.00SC reduced the scope of the original SYF DCP by not installing the wireless area radiation monitoring system, a Data Acquisition System (DAS) printer, and two beta booths. It was determined that the South Yard Facility's (SYF) Wireless Area Radiation Monitoring System could be removed or abandoned in place. These monitors were not required for effluent engineering. They were intended to provide workers in the South Yard an indication that area dose rates had increased and to notify Health Physics (HP) of the increase. Health Physic's procedures require continuous HP coverage for contaminated or high radiation work in the SYF, therefore these monitors are not needed for local area warning.

Evaluation Summary:

This equipment has never been installed in the plant. The evaluation determined that the continuous HP coverage obviates the need for installed equipment and that all applicable regulatory requirements continue to be met. It is concluded that this activity does not require prior NRC approval.

AR 000801415-01: Units 2 and 3 Operation's Procedure for Calculating and Assessing Shutdown Margin (SDM)

Description:

This evaluation supports a SDM calculation performed for the Unit 2 End of Cycle (EOC)-10 shutdown to determine the extent of permissible cool down allowed immediately after the reactor was to be shutdown for the refueling outage. The calculation incorporates all of the same considerations as outlined in the Bases for Limiting Condition for Operation (LCO) 3.1.1.1 (i.e., Reactor Coolant System (RCS) boron concentration, Control Element Assembly (CEA) positions, RCS average temperature, fuel burnup based on gross thermal energy generation, Xenon concentration, Samarium concentration, and Isothermal Temperature Coefficient (ITC)). The analysis uses specific cool-down temperature limits (415 and 290F) and very specific restrictions on the trip condition and xenon worth history following reactor shutdown. This determination of SDM specifies cool down temperature limits and used specific restrictions on the trip condition and xenon worth history following reactor shutdown.

Evaluation Summary:

The proposed SDM evaluation used typical considerations for determining the SDM except that this SDM was performed with judiciously selected initial conditions, and identified very unique trip requirements and temperature limits for use. Once the requirements were verified, the SDM allowed expeditious cool down to as low as 415 degrees Fahrenheit (F) by crediting xenon at the time of trip (but none of the peak xenon after the trip) and then to as low as 290 F by crediting xenon (as above) with a prior minimum required boration. Conservative assumptions and methodologies were used throughout the evaluation, and at all times, the minimum SDM was maintained above the minimum Technical Specification required value of 5.15% delta rho. It is concluded that this activity does not require prior NRC approval.

AR 010100068-02: Units 2 and 3 Replacement Main Feedwater Flow Measurement Instrumentation

Description:

Design Change Package (DCP), 2&3-2084.00SJ, replaced the instrumentation to measure feedwater and steam generator blowdown flow rates and feedwater temperature with more accurate ultrasonic flow and temperature measuring sensors. The addition of ultrasonic flow and temperature instrumentation is necessary to provide high accuracy secondary calorimetric calculations by the Core Operating Limits Supervisory System (COLSS). This equipment is needed to support increased reactor power level. Increasing power level is covered separately outside of the 50.59 process with NRC approved Amendments. [Reference: Proposed Change Number (PCN)-514 "Power Uprate."]

Evaluation Summary:

Installation of this equipment will not impact the main feedwater or blowdown piping and does not have direct control or safety functions. This system will serve as an input to COLSS and will enhance the ability of COLSS to provide accurate information as an operator aid. This equipment is provided to assist in the COLSS calculation of secondary calorimetrics, by providing constant calibration to the existing calorimetric inputs to COLSS. The equipment is used to increase the accuracy of the existing inputs to COLSS for determination of secondary calorimetric power. Since the basic COLSS functions remained unchanged, and the inputs for secondary calorimetric power became more accurate it is concluded that this activity does not require prior NRC approval.

AR 010100068-37: Lowering Units 2 and 3 License Power Limit (LPL) on Extended Loss of Ultrasonic Main Feedwater Flow Measurement Instrumentation

Description:

The SONGS Units 2 and 3 Technical Specifications require a power reduction from 3438 MWt to 3390 MWt if the crossflow feedwater/steam flow instrumentation is out of service in excess of 31 days. After 31 days, the Core Operating Limit Supervisory System (COLSS) artificially increases the feedwater and steam mass flows to the calorimetric power calculation which results in an overpower alarm if reactor power is above 3390 MWt. It was noted that the methodology used to generate the COLSS alarm would not work under the rare condition when calorimetric power fails and power indication is determined by reactor core differential temperature or turbine power. Therefore, a new method of generating an alarm was developed using the plant computer rather than COLSS. This modification is a software change to COLSS such that the artificial increase to the feedwater and steam mass flows does not occur; the alarm function that this coding provided is now provided by the plant computer.

Evaluation Summary:

COLSS is a passive monitoring system with an alarm function. This change removes an alarm function, which is not a Technical Specification or regulatory requirement (the alarm function will now be provided by the plant computer), therefore, this change does not require prior NRC approval.

AR 010100604-05: Units 2 and 3 Updated Fire Hazards Analysis (UFHA) Review Project

Description:

UFHA Change #1 - Section 1.0 was the lead document for a UFHA Review Project. The UFHA review was performed under the requirements of the 10 CFR 50.54(f) letter as a "non-safety significant system." This UFHA review ensured the accuracy of the information within the UFHA, ensured that the 10 CFR 50.54(f) review requirements were met, and made minor enhancements to the UFHA to improve fire protection program effectiveness and efficiency. Information pertinent to operation of the Unit 1 fire protection system was moved to the Unit 1 Defueled Safety Analysis Report (DSAR) and other Unit 1 information not related to Unit 2/3 was removed.

Evaluation Summary:

The UFHA Review project and the resultant changes were enhancements. The changes made also ensured that the content of the UFHA was accurate with respect to the as-designed, as-built, and as-operated condition of the plants. The content of the UFHA is essentially a summary level description of information that is controlled via the design control and other administrative processes. There are no physical modifications being performed as a result of this UFHA Review Project. Therefore, these changes to the UFHA do not directly affect the design and operation of the plant. It is concluded that this activity does not require prior NRC approval.

AR 010100615-11: Sampling Changes to the Units 2 and 3 Offsite Dose Calculation Manual (ODCM)

Description:

Representative sampling of radioactive releases from the site is required by 10 CFR Part 50.36a and Part 50 Appendix I to accurately quantify discharges to the unrestricted area and the resultant dose to a member of the public. Sample compositors have been used since plant start-up but were being added to the ODCM for certain continuous flow streams per the guidance in NUREG 0472 and NUREG 1301.

Evaluation Summary:

The proposed change formally documents the use of composite samplers but will not modify operation of the instrumentation. It was determined that there would be no increase in radioactive liquid releases from the site nor any increase in dose to a member of the public, as a result of this change. It is concluded that this activity does not require prior NRC approval.

AR 010100615-12: Units 2 and 3 Offsite Dose Calculation Manual (ODCM) Effluent Evaluations

Description:

As part of the implementation of the revised 10 CFR 50.59, San Onofre put a separate procedure into practice for effluent evaluations. The procedure included a checklist of related regulations, regulatory guidance, and licensing basis documents to ensure a comprehensive review. Effluent evaluations will be approved by qualified personnel for changes to the ODCM, as well as design changes. Other regulatory guidance documents also refer to performing a 10 CFR 50.59 evaluation for situations that could affect the control of radioactive effluents. Wherever the regulatory guidance refers to a 10 CFR 50.59 evaluation that addresses the potential for creating or modifying the control of radioactive effluents, an effluent evaluation using the new procedure will be performed.

Evaluation Summary:

The purpose of the ODCM is to ensure compliance with regulations regarding dose and curies released, setpoint calculations, sampling, and monitoring of effluent pathways, and control and maintenance of radiation monitors. This new method of change evaluation more directly addresses these issues through a set of review questions developed as a method for evaluating changes to the ODCM. These questions will replace the 50.59 evaluation methodology and will be used to assess the effects related to ODCM changes. Personnel qualified in accordance with site procedures will approve effluent evaluations to ensure compliance with all applicable regulations and regulatory guidance. These evaluations will ensure that there will be no increase in radioactive effluents released to the environment and no increase in dose to a member of the public. It is concluded that this activity does not require prior NRC approval.

AR 010100770-11: Reduced Surveillance Frequency for Units 2 and 3 Low Pressure Turbine Stop and Control Valves

Description:

Licensee Controlled Specification (LCS) change L01-001 revised Surveillance Requirement (SR) 3.3.109.1 to decrease surveillance frequency for the low pressure turbine stop and control valves from once every 31 days to once every 6 months.

Evaluation Summary:

This change reduced the frequency of transient conditions that this test imposed on the secondary plant. This change satisfies the licensing basis for the turbine, including the overspeed protection system and the turbine stop and control valves, to maintain the probability of missile generation at less than 1E-5 per reactor year. It is concluded that this activity does not require prior NRC approval.

AR 010101990-37: Units 2 and 3 Core Protection Calculator (CPC) and Core Operating Limit Supervisory System (COLSS) Decrease in Indicated Power Value

Description:

CPC and COLSS addressable constants were changed to implement the NRC approved power uprate provided through, license amendments 180 (Unit 2) and 171 (Unit 3). To accommodate the new Reactor Thermal Power increase of approximately 1.4%, Indicated Power was decreased by approximately 1.4% with no actual change in power.

Evaluation Summary:

This evaluation justified the CPC and COLSS addressable constants changes that were required to implement the power uprate. The multipliers of power in the Departure from Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD) calculations in CPCs and COLSS were increased by 2% - 3%, which more than compensated for the 1.4% power uprate. Per the supporting evaluation of the power uprate Amendment request (Proposed Change Number [PCN]-514 section 4.1.1.4), the current transient analyses that credit the CPC Variable Over Power Trip add 2% power uncertainty to the trip setpoint, which bounds the change. It is concluded that this activity does not require prior NRC approval.

AR 010200212-71: Addition of Fire Proofing on Cable Trays In the Unit 3 Turbine Building Switchgear Room

Description:

After a fire in the Unit 3 Turbine Building switchgear room, San Onofre personnel observed that cables located in the cable trays directly above the switchgear cubicles which ignited, became damaged from the heat generated by the breaker cubicle fire. To mitigate the consequences of another switchgear fire, Facility Change Notice (FCN) F25037E modified the cable trays located directly above all the switchgear cubicles in this room by wrapping the cable trays with 3M Type 7 fireproofing mats. To support the change, a non-fire rated detail, Type 7 - Exposure Fire Barrier Radiant Energy Shield, has been added to Construction Specification CS-E06. Type 7 fire wrap does not completely wrap around the cable trays but does provide a layer of protection on the bottom and sides of the cable tray. This fire wrap is a non-tested, non-fire rated configuration, and only provides a level of protection from radiant heat from a breaker fire below the cable trays. This Type 7 firewrap will only be installed on cable trays containing non-power cables (e.g., instrument and control cables). Since power cables are not involved, installation of this firewrap does not affect cable ampacity derating calculations. Addition of one layer of this fire proofing material adds approximately 5 pounds per foot to the cable trays. This additional weight is within the limits for the cable tray supports per structural calculations C-270-01.02, Seismic Class I Cable Tray Supports.

Evaluation Summary:

The firewrap is being installed on non-1E cables. The installation of this firewrap does not impact the ability of the cables to perform their intended function. Addition of one layer of this fire proofing material adds approximately 5 pounds per foot to the cable trays. This additional weight is within the limits for the cable tray supports per structural calculations. This firewrap provides a level of protection from heat generated from a switchgear fire below the cable trays. It is concluded that this activity does not require prior NRC approval.

AR 010200212-76: Addition of Fire Water Drains in the Unit 3 Turbine Building Switchgear Room

Description:

During the Unit 3 Turbine Building switchgear room fire, fire suppression personnel observed that this room lacked floor drains to evacuate the water used to suppress the fire in the switchgear cubicle. As a result, water pooled in this room, which created a personnel hazard for firefighting and equipment inspection personnel, since the room still contained energized equipment. In response to this event, floor drains were installed in several locations in this room to facilitate draining water. Facility Change Notice (FCN) F25305C added drains along the West and North walls of the room. These drains discharge water to the 7-foot turbine deck where it then drains to the normal turbine-building sump.

Evaluation Summary:

There is no equipment below these drain locations (e.g., electrical switchgear, pumps, motors, etc.), which could be impacted by water discharging from the drain locations. There is no impact to the plant flooding analysis, as this analysis takes credit for fire suppression activities. The drains were located to not interface with any fire barriers that could impact safe shutdown features. Installation of these floor drains did not compromise the Unit's ability to achieve or maintain safe shutdown. Functionally redundant components, which will be used for safe shutdown, remain protected from fire damage, and there was no other impact to safe shutdown equipment or systems. It is concluded that this activity does not require prior NRC approval.

AR 010200572-07: Addition of a Unit 3 Air Supply Valve

Description:

This modification installed a solenoid valve to isolate instrument air supply to the pneumatic control components at panels 3L-528, 3LY-3293 and 3LIC-3293 upon loss of electrical power at breaker 3QO19-08. The new valve prevents inadvertent overfilling of the Condensate Storage Tank (CST) and flooding of the CST building.

Evaluation Summary:

This change did not have any effect on the Condensate Storage Tanks or the Auxiliary Feedwater System to perform their design functions to provide condensate to the auxiliary feedwater pumps during normal startup and during loss of offsite power conditions, as well as during Chapter 6 and 15 accidents scenarios. The installation of this solenoid valve in automatic level control system does not preclude the use of the tank's inventory for accident mitigation. The preferred makeup from the high flow Make Up Demineralizer (MUD) tanks is still available manually without utilizing the automatic makeup system. The installation and use of the new solenoid valve does not prevent this manual makeup. The design basis for the Condensate Tanks will not be changed by this modification. The level in the tanks can be controlled automatically or manually in order to maintain the required levels per the design basis. Manually controlling the makeup valves will not be altered by this proposed change. The addition of the circuits and components for this design modification will not increase the failure rate of any important to safety component due to isolation and the installation of the design. This change does not affect the tank level instrumentation at the Remote Shutdown Panel or affect the required contained volume of water in the condensate tanks. It is concluded that this activity does not require prior NRC approval.

AR 010200572-25: Alternate Drainage Paths for Units 2 and 3 Condensate Storage Tanks

Description:

Facility Change Notices (FCN) F25166C (Unit 2) and F25592C (Unit 3) added enclosure drains to the tank building compartments containing the safety related, Seismic Category I condensate storage tank T-121 and the non-safety related, Seismic Category II condensate storage tank T-120 system. This design change provided alternate drainage paths for flood sources in the enclosures.

Evaluation Summary:

The proposed change provided enclosure drains for the condensate storage tanks to ensure that credited manual valve positioning operator actions can be performed. The locations of the enclosure drains were designed so that there is no impact on the amount of condensate inventory that must be retained for safe shutdown. Condensate storage tank T-121 is reclassified as an unprotected outdoor tank. Administrative controls are required to limit the total radioactive inventory of T-121 so that an uncontrolled release would not exceed 10 CFR 20 limits. Administrative controls are an approved method for providing assurance against unauthorized or excessive releases of radioactive liquids. Therefore, prior NRC approval is not required.

AR 010200572-31: Valve Addition to Unit 3 Condensate Storage Tank's (CST) Transfer Line

Description:

This modification installed a valve outside of the Unit 3 CST T121 vault in the condensate transfer line from CST T120 to CST T121 via the condensate transfer pump P049. This addition relieved some of the 90 minutes time pressure on Operations to enter a potentially hazardous confined space (the CST T121 vault) to isolate T120 following a design basis earthquake.

Evaluation Summary:

This addition of an isolation valve outside CST T121 vault makes Unit 3 more consistent with the Unit 2 configuration. The modification does not pose a safety concern with respect to requiring an increased missile protection, reduction in safety, reduction in design function, or corrosion concerns. It is concluded that this activity does not require prior NRC approval.

AR 010200708-01: Unit 3 Temporary Power

Description:

During a planned or forced outage the plant requires many different requests for Temporary Power. The following safety evaluation addressed installation and configuration of Temporary Power installed during the Unit 3 A07 Breaker Trip/Fire Forced outage.

Evaluation Summary:

None of the temporary power cables were connected to safety related loads. The cables were routed at least six inches away from either train cable trays or were wrapped with a fire blanket. The temporary power connection was confirmed to be in conformance with established design bases. There was no increased risk to personnel, equipment safety, nor did the evaluated temporary power configuration introduce adverse conditions to any safety related structures, systems, or components. It is concluded that this activity does not require prior NRC approval.

AR 010200890-68: Unit 3 Main Turbine Modifications

Description:

On February 3, 2001, San Onofre Nuclear Generating Station Unit 3 reactor power was at 39 percent and increasing at three percent per hour following a just completed refueling outage when a circuit breaker fault caused a fire, a partial loss of AC off-site power, and a reactor shutdown. The failure of a DC breaker to function properly resulted in the unavailability of the turbine emergency DC lubricating oil system causing extensive damage to the turbine generator. Most turbine damage was resolved by parts replacement or restoration. However, some damage to the turbine required modification of the original design. This safety evaluation addressed the following such modifications:

1. Rotor journals #1 thru # 11 were machined to remove cracks and or hardened material. Journals #3 thru # 9 were also heat treated to restore original material properties.
2. Optiflow inner cylinder casing studs were replaced with studs with an upgraded material to reduce the potential for cracking following testing that identified indications in the thread roots of these studs.
3. Several bearing housings (# 4, 7, & 9) had cracks identified that were repaired by stop drilling. All housings were also heat treated to remove residual stresses and machined to repair distortions, as required.
4. The bearings were refitted with new bearing liners to fit the smaller diameter journals
5. New thrust bearing housing, cages, pads, thrust collar and sleeve were required.
6. Miscellaneous machining and or grinding: Nominal diameter reductions on Gland and Oil Wiper areas, Coupling Heads and Faces and Instrument Track areas

Evaluation Summary:

The major concern with respect to safety for the turbines is the potential of generating a missile that could strike and damage systems important to safety. Acceptable probability of damage from a turbine missile was maintained. The changes maintained the P1 probability evaluated by GEC as less than 1X 10E-5. P2 and P3 are still assumed to be 1 X 10E-2. Therefore, the missile damage probability continues to be less than or equal to 1X 10E-7 as required by Reg. Guide 1.115. A value of 1 X 10E-7 is considered an acceptable risk rate for an essential system. It is concluded that this activity does not require prior NRC approval.

AR 010200925-22: Units 2 and 3 Shut Down Cooling (SDC) Isolation Valve's Pressure Locking Evaluations

Description:

Shutdown cooling suction isolation valves 2(3)HV-9337, 2(3)HV-9339, 2(3)HV-9377 and 2(3)HV-9378 provide isolation between the reactor coolant system and the SDC system. They are normally closed, and are opened during outages for SDC operation. The subject valves are Category "A" active Motor Operated Valves (MOVs) subject to periodic testing in accordance with the In-Service Testing (IST) program at San Onofre. They are double disc gate valves manufactured by WKM. An evaluation of these valves was required to ensure that pressure locking (PL) would not impact their opening safety function.

Evaluation Summary:

This activity documented an Updated Final Safety Analysis Report (UFSAR) change which added "no pressure locking" as a design basis for valves 2(3)HV9337, 2(3)HV9339, 2(3)HV9377, and 2(3)HV9378. This change reflects the results of an evaluation performed to assess the susceptibility of the subject valves to PL. The evaluation determined the leak rate required to dissipate any excess pressure in the valve bonnet, which could result in a pressure locking condition. It then showed that the actual leak rate found during the IST program exceeded the leak rate required to eliminate the potential for PL. Based on the results of the evaluation, it was concluded that PL would not impact the safety function of these valves. It is concluded that this activity does not require prior NRC approval.

AR 010201241-02: Evaluation of Impact on Units 2 and 3 from Removal of a Unit 1 Security Perimeter Light Pole Power Supply

Description:

Facility Change Notice (FCN) F24956E removed Unit 1 Security Perimeter Light Pole #1 and made splicing of perimeter lighting circuit #1 from Motor Control Center (MCC) Cubicle DM7C in an alternative location, instead of at the base of the light pole. The splicing was to be done before light pole removal to minimize the power loop outage time.

Evaluation Summary:

This change had no impact on the Unit 2 and 3 UFSAR, the Physical Security Plan (PSP), or the Safeguards Contingency Plan. Compensatory measures ensured that an equivalent level of protection was provided in the event of lost power supply to the lights. Removal of Lighting Pole #1 from the security perimeter lights did not alter the minimum required 0.2 foot candle level for the monitoring and observation of isolation zones within the protected area. An electrical calculation concluded that the reduction of load to the Unit 2 and 3 MCC by this proposed modification to the perimeter lighting circuits was acceptable and satisfactory. It is concluded that this activity does not require prior NRC approval.

AR 010201292-13: Unit 2 Reduced Containment Wall Penetration Air Flow Rates

Description:

This activity addressed the acceptability of reduced supply air flows identified during a startup Test Exception Report (TER) review of the as-tested (lower than design values) air flow rates through containment wall penetrations for the Unit 2 supply air fans serving the main steam, feed water, and the secondary water sample penetrations through the containment wall.

Evaluation Summary:

A revision to plant design calculations determined that the maximum allowable containment wall penetration temperature would not be exceeded with the reduced supply airflow to the affected containment wall penetrations. A UFSAR change notice was issued to reflect the reduced airflow. This activity does not require prior NRC approval.

AR 010300182-01: Units 2 and 3 Core Protection Calculator (CPC) Channel Check Criteria

Description:

This activity was to widen the CPC Local Power Density (LPD) & Departure from Nucleate Boiling Ratio (DNBR) channel check criteria. The change to the channel check only applied during the unique time frame in which the CPC's transition from utilizing a pre-calculated axial power distribution to a measured axial power distribution (and vice versa). This "Unique Time Frame" was when Reactor Power is near 17% on power ascension or near 15% on a power decrease.

Evaluation Summary:

The proposed activity changes the channel check criteria that are applied to the CPC calculated value of LPD & DNBR during the transition from utilizing a pre-calculated axial power distribution to a measured axial power distribution (and vice versa). Technical Specifications (TS) require that channel checks be performed on certain parameters. However, the TS do not specify the numerical and technical details of implementation. It is the Licensee's discretion to promulgate the details to ensure that the channel checks perform their intended safety function. Implicit in this responsibility is to choose channel check criteria incorporating the expected behavior of the parameter. This change accommodated the expected channel behavior during the transition and thus improved the fidelity of the check to the expected behavior of the equipment. Consequently, prior NRC approval is not required.

AR 010400196-01: Units 2 and 3 Remote Venting Station Outside the Spent Fuel Pool (SFP) Heat Exchanger Room

Description:

Facility Change Notice (FCN) F24510M provided a remote venting station outside the SFP heat exchanger room to be accessed and operated after a design basis Loss Of Coolant Accident (LOCA). With this modification installed, to perform venting, the operator will access the Primary Plant Make-up Storage Tank (PPMST) room, rather than the SFP room. This change results in much lower operator doses (although a respirator will still be required). The vent station for both SFP heat exchangers will be located in the PPMST room. Venting progress will be observed in a sight glass with a flow indicator.

Evaluation Summary:

Extending the vent line and adding this remote venting station does not affect any systems important to safety nor does it change the facility as described in the Updated Final Safety Analysis Report (UFSAR). As such, this modification does not require prior NRC approval.

AR 010400217-01: Weld Repair of Units 2 and 3 Saltwater Cooling Pumps

Description:

The activity addressed by this Safety Evaluation is weld repair of bolt-hole corrosion on saltwater cooling pump outer column flanges and saltwater cooling pump impeller cases. An engineered weld repair was performed on the corroded areas restoring the components to their original design configuration.

Evaluation Summary:

This weld repair of corrosion areas restored the components to their original design configuration by replacing the material lost to corrosion with an engineered weld repair. Functionally, the replacement material's performance is the same as the material lost to corrosion. It is concluded that this activity does not require prior NRC approval.

AR 010400566-07/-09: Unit 2 Full Flow Condensate Polishing Demineralizer (FFCPD) Alarm

Description:

Facility Change Notice (FCN) F25743E electrically disabled the FFCPD control room alarm contacts from alarm instrument 2CPCSH571 (condensate polisher effluent sample condition). This instrument (monitor) would generate a continuous high straight conductivity alarm signal due to ethanol amine (ETA) injected into the FFCPD effluent to control water chemistry for reduced corrosion. Once the ETA is present in the FFCPD effluent at the monitor, the straight conductivity meter would loose the resolution required to discern the changes for which it was originally designed and installed. Disconnection of this alarm signal from SH571 was necessary to enable immediate injection of ETA into the system.

Evaluation Summary:

Disconnection of the FFCPD high straight conductivity alarm had no effect on the bases of the Technical Specifications. This activity does not require NRC approval because there is no effect on important-to safety systems, and has no impact or effect on accidents and incidents analyzed in the Updated Final Safety Analysis Report (UFSAR).

AR 010501009-51: Units 2 and 3 Emergency Diesel Generator (EDG) Fuel Change

Description:

This activity changed the type of EDG diesel fuel oil to California Air Resources Board (CARB)-formulated diesel fuel oil to reduce particulate, nitrogen oxide, and sulfur oxide emissions. This fuel oil change was approved by NRC issuance of License Amendments 183 and 174 for Units 2 and 3 respectively. The diesel fuel used in the EDGs during plant startup had a 28 API gravity where a typical batch of CARB diesel fuel has 35-36 API gravity. As American Petroleum Institute API gravity increases, the British Thermal Units (BTU) per gallon of diesel fuel decreases. This evaluation confirmed acceptability the resultant decrease in BTU per gallon as well as changes to other physical and chemical properties, not addressed in the approved License Amendments.

Evaluation Summary:

The resultant drop in the specific heat (BTU per gallon) of diesel fuel oil, as well as changes to other bulk physical and chemical properties were determined to be acceptable by this evaluation. CARB fuel meets the standard set by American Society of Testing and Materials (ASTM) D975-81, the licensing commitment. A later version of the standard, ASTM D975-01A, recommended a fuel lubricity value not consistently met by samples of CARB fuel. However, the guidance in the standard provided a basis for acceptance of the CARB fuel lubricity. When combined with similar service reports, and an evaluation of the lubrication needs of the EDGs, it was evident that there would not be adverse lubricity impacts. This evaluation reviewed relevant parameters and supporting calculations affecting EDG operability and concluded that the change to CARB fuel would not result in changes to EDG performance outside the design basis established in the Final Safety Analysis Report (FSAR). It is concluded that this activity does not require prior NRC approval.

AR 010501433-04: Unit 3 Reactor Coolant Pump (RCP) Oil Level Detection System Camera

Description:

During power, the oil level detection system for a Unit 2 RCP motor lower reservoir was making step changes (10-20 %). Because the oil level sight glass was not accessible during power it could not be determined if the oil actually changed level or if there was a problem with the detection system. Temporary Facility Modification (TFM) 3-01-BBA-001 installed a temporary video camera near the lower site glass for RCP motor (S31201MM001). The installation of the camera was to allow Engineering to observe the sight glass during plant operation.

Evaluation Summary:

Both the video camera and its stand were anchored with stainless steel wire to steel components to avoid the potential of migrating to the sump. This TFM installed one camera and approximately 15 pounds of cable in containment. The containment cleanliness guidelines were used to ensure that the containment sump operability was not impacted. The additional fire load from the cable was evaluated to be acceptable. It is concluded that this activity does not require prior NRC approval.

AR 010501444-01: Units 2 and 3 Loss of Control Room Annunciators Procedure Change

Description:

The activity provided an addition to procedure SO23-13-22, "Loss of Control Room Annunciators," to provide for the case when Core Operating Limits Supervisory System (COLSS) is still functioning - calculating and displaying the Kilowatt (KW)/Foot (Ft) and Departure From Nucleate Boiling (DNBR) Power Operating Limits (POLs), power indication(s), and power distribution values for Axial Shape Index and Azimuthal Tilt - when the Main Control Board (MCB) COLSS alarm, 50A02, (only) is out of service. For this case, COLSS is still operable, but the continuous monitoring function is impaired in that the Main Control Board (MCB) COLSS alarm which will not visually or audibly annunciate if either POL becomes less than plant power, or if the Axial Shape Index (ASI) or Azimuthal Tilt limit is reached.

Evaluation Summary:

The structure/system/component function in this case is to calculate licensed power, ASI, and Azimuthal Tilt, and compare them to limitations, including calculated POLs; with a functional COLSS. This function had not changed. The method of performing the function is to continuously compare the calculated values against a limitation and provide an alarm if the limitation is exceeded; this method has not changed. The alarm is changed from the audible and visual prompt of the main control board COLSS alarm to the visual (only) Plant Monitoring System (PMS)/COLSS Backup Computer System (CBCS) System Event Summary page on one of the plant computer screens. With an operator assigned to continuously (at least every 5 minutes) monitor the System Event Summary page, this method of alerting operators when the COLSS parameters exceed their limits was determined equivalent to the method provided by the audible and visual MCB COLSS alarm. It is concluded that this activity does not require prior NRC approval.

AR 010600258-02: Unit 2 Turbine Stop and Control Valve's Surveillance Frequency

Description:

Licensee Controlled Specification (LCS) change L01-014 revised the Unit 2 LCS 3.3-109 "Turbine Overspeed Protection System" to decrease the Surveillance frequency for the low pressure turbine stop and control valves in the first four years of a ten-year inspection interval; from once every 6 months to once every 24 months. The Surveillance testing frequency at all other times continues to be once every six months.

Evaluation Summary:

Prior to this LCS change the licensing basis for the turbine, including the overspeed protection system and, in turn, the turbine stop and control valves, was to maintain the probability of missile generation to be less than 1E-5 per reactor year. Following this change, the total probability of missile generation remained less than 1E-5. The change in valve test frequency was determined to be bounded by the probability acceptance criteria previously approved by the NRC. The probability, and consequences, of a turbine missile striking and damaging equipment important to safety remains unchanged by this change in turbine stop and control valve Surveillance testing frequency. It is concluded that this activity does not require prior NRC approval.

AR 010600694-01: Units 2 and 3 Feedwater Flow Instrumentation Change

Description:

This activity changed the feedwater venturi constants in the Core Operating Limits Supervisory System (COLSS) and backup COLSS by converting to use of Ultra-sonic Flow Measurement (UFM) of the feedwater to each steam generator. COLSS monitors power, and power distribution, and provides alarm indication to the control room when parameters are found out of limits. The parameter of interest with this evaluation is the Feedwater Based Secondary Calorimetric (FWBSCAL). FWBSCAL calculates feedwater mass flow from venturi transmitter data. A venturi constant is necessary for this calculation and there is a COLSS venturi constant for each steam generator that is typically established by a calibration laboratory. Prior to secondary plant physical cleaning, to accommodate associated changes in the venturi, these two constants were increased causing FWBSCAL, feedwater mass flow, and power indication to increase, adding conservatism. With secondary plant physical cleaning completed data was collected, which shows that FWBSCAL mass flow remained high, compared to UFM, which is highly accurate. This evaluation addresses lowering the venturi constants such that COLSS feedwater mass flow will match UFM.

Evaluation Summary:

The use of UFM to measure feedwater mass flow and calibrate MSBSCAL power was analyzed with the addition of MSBSCAL algorithm to COLSS. Usually MSBSCAL is the selected plant power indicator. The specifics, regarding feedwater mass flow measurement with the UFM instruments, are described by an approved plant procedure. The constant changes allowed by this evaluation removed some of conservatism installed in the recent past. These conservatisms were added beyond the required conservatism to accommodate the anticipated effect of system cleaning. Data collection after cleaning demonstrated instrumentation accuracy, which meant that the additional conservatisms were not required. The conservatism being removed is not credited in any safety evaluation. It is concluded that this activity does not require prior NRC approval.

AR 010601031-04: Units 2 and 3 Control Panel Shelves Seismic Classification

Description:

This activity was a Non Conformance Report (NCR) accept-as-is disposition for Foxboro control panel shelves. Contrary to their current classification as Seismic Category (SC) I components, the shelves do not meet SC I design requirements. However, none of the instruments supported by the shelves are SC I. Consequently, the shelves do not actually perform a SC I design function. The NCR disposition changes the seismic classification of the shelves to Seismic Interaction II/I; this classification is consistent with their actual design function. The changes in classification made by the NCR disposition were necessary because the existing design records for the shelves indicated that they would be able to perform a SC I function. Without the changes in design records, SC I components for possible plant modifications could be inappropriately installed in the shelves.

Evaluation Summary:

No Technical Specification changes were required to implement the NCR disposition. The disposition did not affect the facility or any procedures described in the Updated Final Safety Analysis Report (UFSAR). Instrument uncertainties and protection of seismic category I equipment design bases are unaffected. Therefore, prior NRC approval is not required.

AR 010800321-10: Unit 3 Facility Modifications for Dry Cask Storage

Description:

In preparation for the dry storage and transfer of spent fuel to the future Independent Spent Fuel Storage Installation (ISFSI), the following modifications were required in the Fuel Handling Building (FHB):

1. Installation of a seismic restraint for the transfer cask in the cask wash down area at elevation 63'-6" of the FHB. The transfer cask is not permitted to tip over during a seismic event when loaded with spent fuel. The restraint for the cask is two "cables" around the cask with attachment plates on the FHB wall. The wall of the FHB is adequate to support the transfer cask seismic loads. There is also a removable seismic restraint on the cask wash down floor to prevent the cask from sliding towards the wall.

2. Installation of gas bottles and a rack near the cask wash down area at elevation 63'-6" for preparation of the dry fuel canister. The bottle rack will be attached to the FHB wall, which is not safety-related and designed to Seismic Interaction II/I requirements.

The modifications are new installations onto the existing FHB walls.

A 10 CFR 50.59 evaluation was required only for item 1 above - the addition of the seismic restraints, because of the additional loads on the FHB wall beyond those considered in their initial design.

The addition of the gas bottles and rack was screened out because their incremental loads were within the general design loads that were already considered on the FHB wall and floor.

Evaluation Summary:

Although the seismic restraints for the dry fuel storage transfer cask will add additional loads to the FHB wall, the additional design loads on the FHB concrete wall are within the allowable capacity of the wall and thus there would be no failure of the wall. An Updated Final Safety Analysis Report (UFSAR) change to Table 3.8-10 shows that the design moment of 224 kilogram-feet/foot (k-ft/ft) is less than the moment capacity of 332 k-ft/ft. There was no departure from the methods of evaluation described in UFSAR Sections 3.7 and 3.8 for the Fuel Handling Building. It is concluded that this activity does not require prior NRC approval.

AR 010800825-02: Unit 2 Containment Cooling Abnormal Alignment

Description:

An abnormal alignment resulted from running of only three of the five normal coolers in the Unit 2 containment. This alignment tested the normal cooler configuration during a planned bus B08 electrical outage. An emergency cooling unit was also started during the evolution. While the Updated Final Safety Analysis Report (UFSAR) section 9.4.1.1.2.3 describes four of five normal cooling units operating during normal power operation, the running of three normal cooling units was determined to maintain the ambient temperature below the design 120° Fahrenheit (F). On June 12, 2001 Unit 2 was run for 1 1/2 hours with three normal cooling units and one emergency unit (2ME402) operating. The containment calculated temperature changed from 83°F to 85°F. The normal secured units were available during the planned bus outage duration, if required, and the two remaining Emergency Cooling Units (ECU) were also available.

Evaluation Summary:

The ECUs were not required, but were operated anyway to ensure a more even temperature distribution inside the bioshield. The UFSAR has no requirement for the normal containment system to have any nuclear safety design bases. Its only function, related to nuclear safety, is to maintain the containment below 120°F during normal operation. This assures that all equipment operates below its design limits and is the starting point for all accident analyses performed for containment. Technical Specification 3.6.5 requires that the containment average air temperature shall be less than or equal to 120°F. This abnormal alignment maintained the temperature below 100°F. This evaluation demonstrated that this activity does not require prior NRC approval.

AR 010801543-09: Temporary Compressor for Units 2 and 3 Respiratory System

Description:

This activity provided a temporary backup air compressor for the Respiratory System Air Supply (RSAS) Air Compressor SA2423MC445, until a new air compressor could be installed. A temporary electrically driven air compressor was connected to the header to provide an adequate alternate air supply.

Evaluation Summary:

Abandoning this RSAS Air Compressor and providing a temporary backup air compressor until the new air compressor could be installed, did not adversely impact any safety system component. The RSAS compressors have no nuclear safety design basis. The temporary electrically driven air compressor connected to the header provided an adequate alternate respiratory air supply. Additionally. a backup diesel driven air compressor was also made available. It is concluded that this activity does not require prior NRC approval.

AR 011000765-04: Backup Water Supply Connection at Units 2 and 3 for the Unit 1 Spent Fuel Pool (SFP) Makeup

Description:

This activity provided a procedurally controlled, remote, removable makeup water source from Units 2 and 3 for the Unit 1 SFP. The makeup source was a fire hose connection onto the primary make-up (PMU) pump. This hose connection was to be normally removed and available, but could be installed for routine testing or in the event of an emergency situation when no other sources were available. The preferred source to use in an emergency, or for routine testing of this flow path, was the Units 2 and 3 de-mineralized water Hill Tanks. Because of their relative elevation and line sizes, these tanks provide more pressure and flow than is provided by gravity feed from the present Unit 1 seismic qualified and approved Aux Feed Water (AFW) storage tank.

Evaluation Summary:

The emergency connection is between the Units 2 and 3 de-mineralized water Hill Tank supply and the Unit 1 PMU system. The PMU pumps will be procedurally disabled to prevent any potential for back-feeding water from the Unit 1 PMU tanks to Unit 2/3 systems. The PMU supply to the SFP is through a check valve and a normally locked closed safety related manual isolation valve. The supply water pressure from the proposed source is greater then SFP pressure at this connection and therefore 3 barriers (manual valve, check valve, and higher system pressure) exist to prevent water from flowing out of the SFP from this connection. If the Safety Related locked closed manual isolation valve were to fail open and could not be closed, a backup check-valve would prevent water from flowing out of the SFP system, plus several other valves are in-line between this hose and the SFP cooling system, which could be closed to stop any leakage out of the system. It is concluded that this activity does not require prior NRC approval.

AR 011200343-07: Units 2 and 3 Component Cooling (CCW) Water Make-up

Description:

This activity revised the description of the operation of the normal make-up for the CCW system in the Updated Final Safety Analysis Report (UFSAR) and in the Design Basis Document (DBD) to identify potential failure modes and effects of spurious opening/failure to re-close, and the credited operator actions. An associated calculation was also revised to address issues related to potential spurious opening of the normal makeup valves due to non-1E raceway interactions.

Evaluation Summary:

Normal makeup to the CCW system is provided from the nuclear service water (NSW) system through a NSW make-up valve. The valve is closed prior to the water level in the Surge Tank reaching high level. Failure of the operators to close the make-up valve would result in a control room alarm on high water level, which would alert the operators that the nuclear service water had not yet been isolated. When approaching the high level, there are many potential options available for stopping the makeup flow. The control circuits are not Quality Class (QC) II, and therefore, it was conservatively assumed that the Control Room switch fails to close the valve upon demand. The available action time of at least 30 minutes was determined sufficient to terminate flow prior to overflowing the Surge Tank. In addition to the normal fill evolution, the potential for spurious opening of the fill valves was identified and resolved, based on acceptable Non-Class 1E circuits design. Neither the breakers nor the Surge Tank high-level alarm are seismic Category I. Consequently, when seismic activity is felt, which may cause observable effects to plant processes, the operators enter Abnormal Operating Instruction (AOI) SO23-13-3 (Earthquake). When this occurs, the operators have a higher level of alertness to changes in Control Room instrumentation readouts, such as the Surge Tank level indication, and the operators would notice the initiation of a spurious fill of the Surge Tanks shortly after this AOI is entered. For non-seismic events, if one of the breakers for other non-1E circuits sharing a raceway with the NSW makeup valves is assumed to be the postulated single failure during the accident, the high level alarm and Surge Tank level indication would remain available to alert operators of the spurious overfill. Once the operators are aware that spurious filling is taking place (either due to the alarm, or due to entry into SO23-13-3, depending on the scenario), operator action would prevent an overflow. As discussed above, there would be more than 30 minutes available (after reaching the high level) to terminate the fill. Alarm response Instruction SO23-15-64.A lists corrective actions to be taken in response to a CCW Surge Tank high-level alarm. To ensure that appropriate action is taken, additional guidance was added. There will be no adverse affect on the design functions of the CCW system as described in the UFSAR as a result of these proposed changes to the procedures. Likewise, the additional operator actions needed for CCW normal makeup isolation will not impact the remaining operators ability to respond as credited for design basis event/accident mitigation. It is concluded that this activity does not require prior NRC approval.

AR 020200745-03: Units 2 and 3 Toxic Gas Isolation Signal (TGIS) Allowed Outage Time (AOT)

Description:

Licensee Controlled Specification (LCS) change L02-008 extended the AOT for one train of the TGIS, in LCS 3.3.101, from seven to fourteen days.

Evaluation Summary:

Probabilistic Risk Assessment (PRA) evaluation PRA-02-07 was performed that verified that the impact of this TGIS AOT extension was bounded by a previously performed and approved PRA. Consequently, this AOT extension was found to have an insignificant increase in risk. The increase in the unavailability of TGIS was similar to the increase in the likelihood of occurrence of a malfunction. However, it was determined that this increase was not more than minimal. There was no increase in the probability of occurrence of a previously evaluated accident, nor increase in the consequences of a previously evaluated accident or malfunction. There was no new accident or malfunction created by this AOT extension and there was no effect on a design basis limit for a fission product barrier. It is concluded that this activity does not require prior NRC approval.

AR 020301482-25: Leak Rate Testing Exemption for Units 2 and 3 Refueling Water Storage Tank (RWST) Check Valves

Description:

The RWST discharge check valves were excluded from leak-rate testing based on a single failure exemption. A concern was raised to determine if this exemption was valid. In parallel with this exemption investigation, the Updated Final Safety Analysis Report (UFSAR) Chapter 15 post-Loss Of Coolant Accident (LOCA) dose analyses were revised to increase the modeled leakage past these check valves from a small nominal value to a more substantial value that could be used for a leak-rate test acceptance criterion for testing.

Evaluation Summary:

To avoid excessively increasing the post-LOCA dose consequences, analysis was performed to recapture dose margin lost with the increased check valve leakage by reducing the allowable leakage past the mini-flow isolation valves which normally isolate the Safety Injection and Containment Spray pumps minimum flow return paths to the RWST. Although post-LOCA dose consequences have increased as a result of this activity, the revised doses meet the acceptance criterion for being considered a minimal increase in consequences. It is concluded that this activity does not require prior NRC approval.

AR 030102193-11: Unit 3 Reactor Vessel Nozzle and Core Barrel Markings

Description:

This activity was the corrective action implemented to resolve marking reported on the reactor vessel outlet nozzle faces and the core support barrel snubber blocks. Loose material and raised areas on sealing surfaces were removed. The corrective action included surface preparation, inspection, evaluation, and any required analysis.

Evaluation Summary:

The marks on the hot leg bosses and the core support barrel snubber will remain. Thus, the final corrective action results in a modification to the affected reactor internals. There was no adverse effect on structural integrity or core bypass flow. The corrective actions of removing loose material and raised material, then subsequently completing detailed visual inspections prior to service, along with any required evaluations or analyses, or subsequent examinations, ensured that the corrosion protection provided by the cladding was not significantly degraded. It was determined that impact from the existence of these marks was minimal. It is concluded that this activity does not require prior NRC approval.

AR 991200910-01*: Addition of Units 2 and 3 Turbine Building High Energy Line Break (HELB) Detection Instrumentation and Implementation of an Associated Licensee Controlled Specifications (LCS) Change
(* Referenced AR originated task. This 50.59 evaluation was in hard copy form)

Description:

A change to the plant added temperature instrumentation in the Turbine Building to mitigate the effects of a Turbine Building HELB. LCS L01-013 created a new LCS Section, 3.3.110 "HELB Temperature Instrumentation," which identified this instrumentation for operability and Surveillance requirements (SR).

Evaluation Summary:

Addition of this instrumentation to the Turbine Building provided increased protection from the consequences of a HELB. The new LCS section ensures appropriate Operability and Surveillance is maintained for this instrumentation. The performance characteristics of this instrumentation will not affect any plant safety system. It is concluded that this activity does not require prior NRC approval.

AR 980900571-10*: Units 2 and 3 Technical Specification (TS) Bases Change to Revise "Boron System – Operating" Requirements
(* Referenced AR originated task. This 50.59 evaluation was in hard copy form)

Description:

Bases Change B00-029 revised the Bases for Technical Specification 3.1.9 "Boron System – Operating," to identify suction and discharge paths for charging pumps to inject borated water into the Reactor Coolant System. The revision also clarified use of the alternate flow path for Chemical and Volume Control System boration through the High Pressure Safety Injection path.

Evaluation Summary:

There was no physical change to the plant design, the safety analysis, or the plant operating procedures. This change to the TS Bases made it consistent with its use in the safety analysis. In previous fuel cycle's plant operation the Main Steam Line Break (MSLB) accident did use automatic boration upon Safety Injection Actuation Signal to mitigate the post-trip consequences of this event. The results of the MSLB event remained within the NRC approved criteria for this event. It is concluded that this activity does not require prior NRC approval.

<u>Note</u>:
The following five 50.59 evaluation summaries were inadvertently omitted from the previous San Onofre Units 2 and 3 Facility Change Report (FCR), submitted to the NRC on August 2, 2001, and are included in this FCR for completeness.

AR 990701529-03: Units 2 and 3 Installation of Cables for the Comprehensive Application for Reduced Exposure System (CARES) / Health Physics (HP) Instrumentation

Description:

Cables were added inside containment to operate video and audio equipment vital to ensuring As Low As Reasonably Achievable (ALARA) radiation goals are achieved during outages, to support the CARES / HP instrumentation. Alternative methods were utilized for cable routing and separation, which were evaluated for nuclear safety and electrical safety. The cable was not to be secured to pipes or other equipment that exceeded the cables thermal rating. HP personnel are required to visually inspected the cables each outage, to ensure cable integrity is maintained. This Safety Evaluation was prepared to document Updated Fire Hazards Analysis (UFHA) change requests Nos. 11 and 12 to section 7.0, 2(3)-CO for addition of cable insulation materials in fire area/zones 2(3)-CO-15-1A, 1B, 1C and 2(3)-CO-63-1D.

Evaluation Summary:

The CARES cables have no safety related function. An evaluation was performed to ensure that there was no unacceptable consequence to applicable codes and standards. Further, the evaluation determined no impact from Loss of Coolant Accidents (LOCA) or High Energy Line Breaks (HELB). The cabling was not routed with any other cabling (either safety related or non safety related). The addition of cable combustible loading and the type of cable being added has been evaluated to ensure that the bases and results of previously performed fire hazards analyses are not invalidated. In addition, this evaluation confirmed that the safety related/safe shutdown components/cables located in the area/zone are not adversely impacted due to the increase of cable combustible material. This change was determined to not require prior NRC approval.

AR 000201260-14: Secondary Pressure Evaluation and Acceptance Criteria for Units 2 and 3 Feedwater System Pipe Break Accident

Description:

Modification of Updated Final Safety Analysis Report (UFSAR) Section 15 and the Accident Analysis Design basis Documentation (DBD) to change the acceptance criteria for secondary pressure from 120% of design pressure to the NRC approved acceptance criteria of 110% of design pressure for the Feedwater System Pipe Break accident.

Evaluation Summary:

This activity changed the UFSAR and Design basis Documentation DBD-SO23-TR-AA to implement the NRC approved criteria on peak secondary pressure for the Feedwater System Pipe Break accident. No changes to the plant, or the way it is operated, were made. The Feedwater System Pipe Break accident met the modified secondary pressure acceptance criterion. It is concluded that this activity does not require prior NRC approval.

AR 991201443-02*: Backup Pumping and Water Source for Units 2 and 3 Firewater Pumps and/or Water Supply

(* Referenced AR originated task. This 50.59 evaluation was in hard copy form)

Description:

This evaluation was prepared to assesses the safety significance of Facility Change Notice (FCN) F21740M for modifications to Units 2 and 3 implementing equipment to provide backup pumping by connecting to the station fire truck and water source from the High Flow Make Up Demineralizer (HFMUD), for alternate pump and/or water supply requirements, in the event of an impairment to the normal Unit 2 and 3 firewater pumps and/or water supply.

Evaluation Summary:

Installation of connections from the HFMUD storage tanks to the Unit 2 and 3 firewater main and connecting to the station fire truck will not compromise the ability of either Unit to achieve safe shutdown. When employed with compensatory measures for limited durations, this alternative provides acceptable backup protection in the event of impairment to the normal Unit 2 and 3 firewater pumps or water supply. No new fire hazards are being introduced that have not already been analyzed in the Updated Fire Hazards Analysis (UFHA). All modifications were implemented in accordance with the applicable design codes and standards. This modification does not impact either the firewater system or the condensate system from their design functions. It is concluded that this activity does not require prior NRC approval.

AR 000400399-02*: Minor Quantities of Combustibles In Units 2 and 3 Updated Fire Hazards Analysis (UFHA)
(* Referenced AR originated task. This 50.59 evaluation was in hard copy form)

Description:

UFHA section 2/3-4.0 change numbers 6 and Appendix change number 13 documented minor quantities of various combustibles, including Styrofoam inside Unistruts and Poly Vinyl Chloride (PVC) material and PVC insulated / jacketed / coated cables in various plant locations. An evaluation was performed of the effects of these combustible materials that had been identified in Units 2 and 3.

Evaluation Summary:

The limited amount of Styrofoam is located inside plant Unistruts and is not exposed to conditions where its ignition would represent a significant fire hazard. The UFSAR compliance statements recognize that there are limited amounts of non-Institute of Electrical and Electronic Engineers (IEEE) 383 qualified PVC cabling located in certain plant areas. These combustibles did not increase the consequences of a fire since the combustible loading contribution was determined to be insignificant and inconsequential, in terms of the overall impact on the minimum fire protection features. It is concluded that this activity does not require prior NRC approval.

AR 980602450-05*: Units 2 and 3 Reactor Coolant Pump (RCP) Vibration Monitoring Instrument Cable Combustible Load
(* Referenced AR originated task. This 50.59 evaluation was in hard copy form)

Description:

Updated Fire Hazards Analysis (UFHA) section 2/3-7.0-2CO, change number 9 incorporates the increased combustible loading values resulting from RCP vibration monitoring instrument cables. The fire zones affected by the addition of combustible loading from the non-qualified RCP vibration monitoring equipment cable are 2-1A, 2-1B, 3-1A, and 3-1B. This instrumentation cable is located near each RCP inside the bioshield next to the vibration monitoring equipment and represents approximately five pounds of plastic material.

Evaluation Summary:

The limited amount of added combustible material does not conflict with the guidelines for allowable combustible materials as defined in the UFHA since the amount being added by this UFHA change will not cause the maximum allowable to be exceeded. There is no impact on any equipment important to safety by the additional cables. It is concluded that this activity does not require prior NRC approval.

ENCLOSURE 2

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 2 AND 3

REPORT ON THE COMMITMENT CHANGES MADE PER NEI "GUIDELINES FOR MANAGING NRC COMMITMENTS"

FOR THE PERIOD FROM JULY 1, 2001 UNTIL JUNE 30, 2003

REPORT ON COMMITMENT CHANGES MADE PER NEI "GUIDELINES FOR MANAGING NRC COMMITMENTS"

NRC letter to Mr. Ralph Beedle, Senior Vice President and Chief Nuclear Officer, Nuclear Energy Institute (NEI), dated March 31, 2000 and SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," both state that the NEI 99-04 "Guidelines for Managing NRC Commitments Changes" Revision 0, dated August 2, 1999, was an acceptable guide for licensees to follow for managing and changing their commitments to the NRC. Part of the commitment change process, given in the NEI guidelines, identifies that various commitments can be changed with the notification to the NRC made in a report submitted annually or along with the FSAR updates as required by 10CFR50.71(e). The intent of this report would be to provide a brief summary of the commitments changed since the last report in lieu of filing individual notifications as commitments are revised.

The following summarizes the commitment changes for San Onofre Units 2 and 3 that are to be reported and have occurred from July 1, 2001 until June 30, 2003.

1. Verification of Scanned Image Pages for Quality Affecting Records

This commitment was made in response to Generic Letter 88-18, Plant Record Storage on Optical Disks. On August 6, 1996 SCE sent a letter to the NRC stating our intention to store quality-affecting records on optical media in accordance with GL 88-18. This letter stated that after the quality affecting record had been scanned and saved, the image would be visually verified. This has resulted in 100% verification of each image after scanning is completed. To maintain compliance with GL 88-18 guidance, regarding verification of all images, 100% image verification for legibility will now occur during the scanning operation using the image file which is displayed, instead of after the document has been scanned.

This commitment change is being reported to the NRC in the Refueling interval summary report per the NEI Guidelines. This is a change made in response to a Generic letter that already has been implemented.

2. Component Cooling Water Operability

The original commitment was based upon an interpretation of General Design Criteria 2. The requirement to, "withstand the effects of ...earthquakes...without loss of capability to perform their safety function" was interpreted by SCE to mean that the Component Cooling Water system must remain functional following a combination of Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident. This interpretation was submitted to the NRC by letter dated February 14, 1989, and SONGS has been committed to this interpretation since that time. The NRC Inspection Manual Part 9900, Technical Guidance, states that: "Safety related structures, systems and components are designed to (1) remain functional during an SSE, (2) ensure the integrity of the reactor coolant pressure boundary, and (3) to have the capability to shut down the reactor and maintain it in a safe condition, or the capability to mitigate the consequences of accidents. However, as a design-basis event, the SSE is not assumed to occur simultaneously with postulated accidents." Based on the guidance provided in the Inspection Manual, it is concluded that design basis accidents concurrent with a SSE do not have to be considered in a safety analysis.

This commitment change is being reported to the NRC in the Refueling interval summary report per the NEI Guidelines. This is a change to a commitment necessary for compliance with an Obligation that has been implemented.

3. Requirement for the NuPac PAS-2 Cask

This commitment was a result of not having the capability to analyze an undiluted grab sample, with design source terms, onsite. As a result the NuPac PAS-2 Cask was credited and maintained for this use. There is no longer a requirement to have the ability to take and analyze an undiluted grab sample. With the requirement for maintaining a Post Accident Sampling System being eliminated, all that is needed is the ability to take and analyze a diluted grab sample during the recover phase. The capability to analyze a diluted grab sample does exist onsite. Because of this, the need to maintain the NuPac PAS-2 Cask is not longer required.

This commitment change is being reported to the NRC in the Refueling interval summary report per the NEI Guidelines. This is a change to a commitment that was credited as a basis for a safety decision in an NRC SER. The original commitment had been implemented.