ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-361/95-26
50-362/95-26

Licenses: NPF-10
NPF-15

Licensee: Southern California Edison Co.
P.O. Box 128
San Clemente, California

Facility Name: San Onofre Nuclear Generating Station, Units 2 and 3

Inspection At: San Onofre, San Clemente, California

Inspection Conducted: November 13 through December 1, 1995

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Division of Reactor Safety

Inspection Summary

Areas Inspected (Units 2 and 3): Routine, announced inspection of engineering and technical support and 10 CFR 50.59 safety evaluations.

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Results (Units 2 and 3):

Engineering

- The inspectors identified one violation of procedural requirements, in that a modification was processed as a field change notice and not as a design change package. In addition, a detailed 10 CFR 50.59 safety evaluation was not performed as was required by the field change notice process. A weakness was also identified, in that the licensee's procedure for processing field change notices did not require documenting the basis for the 10 CFR 50.59 applicability screening (Section 5).

- The inspectors were concerned regarding the number of inconsistencies between the updated final safety analysis report and actual plant conditions. Although the individual examples were relatively minor in nature, the number of inconsistencies, in conjunction with the lack of a periodic method to verify the accuracy of the updated final safety analysis report heightened a concern regarding the failure to perform a 10 CFR 50.59 safety evaluation described above (Section 3.2).

- The licensee was effectively implementing the requirements of the nonconformance reporting program. This was based on a review of 64 nonconformance reports (Section 2.2).

- The inspectors concluded that the licensee's corrective actions for the failure of a 16-inch WKM Model D-2 "Pow-R-Seal" shutdown cooling suction isolation valve on July 23, 1995, were not timely. Specifically, the licensee had not yet implemented all the identified corrective actions involving inspections of all similar valves to confirm that the failure was an isolated case (Section 2.2).

- The licensee performed a superior root cause evaluation for the failure of Square-D linestarters during the Unit 2, Cycle 7, refueling outage after several control power fuses blew during attempts to cycle motor-operated valves. The report clearly presented the problem and discussed possible causes by giving both supporting and refuting evidence. The exclusion of alternate root causes was well supported. The report clearly documented the condition of the equipment and source of the problem. The report was of sufficient quality to inform, in a precise and understandable manner, engineers who had no previous experience with the problem (Section 2.3).

- The inspectors reviewed six design calculations and concluded that the calculations were thorough and reflected good engineering practices (Section 3.3).

- The design change package involving a cross connection of the shutdown cooling and spent fuel systems was satisfactory in all respects, and reflected a comprehensive and conservative methodology. The accompanying safety evaluation was broad in scope and clearly presented. The minor modifications performed under the field change notice process were well done. (Section 3.4).

- System engineering personnel performance was determined to be superior. System engineers were very knowledgeable of their systems, both from a configuration standpoint and in reference to current developments affecting the systems. The inspectors noted that system engineers were spending a substantial amount of time in the plant (Section 4.2).

- Design engineering's aggressive pursuit of corrective actions to a previous violation resulted in the discovery that a high energy line break could affect adjacent safety-related areas via the nonsafety-related heating, ventilating, and air conditioning system. This was an original design flaw by the vendor that was very unlikely to be discovered during routine evaluations (Section 6).

- The overall material condition of the accessible portions of the safety injection and component cooling water system was adequate. These areas appeared clean, uncluttered, and well illuminated. A non-cited violation was identified for the storage of a large box near safety-related equipment (Section 2.1).

Summary of Inspection Findings:

- One non-cited violation was identified (Section 2.1).

- Inspection Followup Item 361/9526-01, 362/9526-01 was opened (Section 2.2.1).

- Violation 361/9526-02, 362/9526-02 was identified (Section 5).

- Inspection Followup Item 361/9526-03, 362/9526-03 was opened (Section 6).

- Violation 361/9510-01 was closed (Section 7.1).

- Inspection Followup Item 361/9513-01 was closed (Section 7.2).

- Licensee Event Report 361/95-009 was closed (Section 7.1).

Attachments:

- ATTACHMENT 1 - Persons Contacted and Exit Meeting
- ATTACHMENT 2 - List of Documents Reviewed

DETAILS

1 INTRODUCTION

The engineering and technical support inspection was conducted at the San Onofre Nuclear Generating Station during the weeks of November 13 and 27, 1995. This inspection was conducted to evaluate the licensee's ability to provide effective engineering and technical support to the plant. The inspection activities encompassed the activities of the nuclear engineering design organization and the station technical division (includes system engineers). These objectives were accomplished by:

- Assessing the effectiveness of engineering and technical support by focusing on a vertical slice of the functional aspects of the safety injection and component cooling water systems;
- Assessing the effectiveness of the licensee's design change processes to ensure that plant design and safety were maintained, and to assure conformance with regulations and commitments;
- Assess the licensee's ability to identify and resolve technical issues and problems;
- Assess the licensee's ability to maintain accurate design basis information; and
- Assessing the effectiveness and thoroughness of the licensee's implementation of the 10 CFR Part 50.59 safety evaluation program.

2 SYSTEM REVIEWS (37550)

2.1 <u>System Walkdowns</u>

The inspectors performed several walkdowns of the accessible portions of the component cooling water and safety injections systems. Generally, these areas appeared clean, uncluttered, and well illuminated. The following material discrepancies were identified during the walkdowns:

- The inspectors found a heavy, unrestrained box lying on the floor next to Component Cooling Water Surge Tank S21203MT004 in Unit 2. The box was positioned approximately 2 feet from the surge tank, which was contrary to the requirements of Procedure SO123-I-1.20, "Seismic Controls," Temporary Change Notice 3-1, which required an 8-foot separation. The licensee issued Nonconformance Report 95110040. The licensee moved the box to a point approximately 4 feet from the surge tank, which was still within the exclusion zone. However, the licensee evaluation determined that, even at the as-found, 2-foot separation, the potential for tank damage was unlikely and concluded that the tank was operable throughout this period. The inspectors accepted the licensee's operability determination. This failure constitutes a violation of minor significance and is being treated as a non-cited violation, consistent with Section IV of the NRC Enforcement policy.

- Motor-Operated Valve 3HV6228B (Unit 3) had lubrication leaking from the actuator housing and the motor-to-actuator flange. The licensee issued Maintenance Order 95111115000 to correct this condition.

- White, crystallized powder was present on Motor-Operated Valve 3HV6228A (Unit 3). The licensee issued Maintenance Order 95111114000. The licensee determined that the white powder consisted of crystallized Calgon, a chemical added to the component cooling water system to inhibit corrosion.

- The declutch lever of motor-operated Valve 3HV6551 (Unit 3) was butted up against an electrical conduit. In this configuration, reclutching to engage the motor operator may have been impeded. The licensee issued Maintenance Order 95111116000 to relocate the conduit. The licensee determined that the operability of the motor-operated valve was not in question because the valve could not be returned to an operable status until after it was successfully stroked electrically following manual operation.

- The declutch lever arm on motor-operated Valve 3HV6227 (Unit 3) was bent and wedged tightly against a conduit. The licensee issued Maintenance Order 95120179000 to analyze this configuration.

- An abnormal amount of grease and debris had accumulated on the valve stem of Valve 3HV6551 (Unit 3). The licensee issued Maintenance Order 95111117000 to clean the valve and adjust the packing.

- Condensate from Air Conditioning Unit S21507ME499 (Unit 2) was dripping on the floor. The licensee initiated Maintenance Order 95110994000 to clean the drain line.

- Boric acid was evident on the tailpipe of High Pressure Safety Injection Pump 2P019, suction relief Valve 2PSV8154 (Unit 2). The licensee initiated Maintenance Order 95110945 to investigate for a leak.

The inspectors concluded that the above items had minimal safety significance, and that the licensee had taken appropriate action.

2.2 Review of Nonconformance Reports (37550)

The inspectors reviewed 64 nonconformance reports to assess the adequacy of the licensee's ability to identify and correct nonconforming conditions. Overall, the inspectors concluded that the licensee was effectively implementing the requirements of Procedure SO123-XV-5, "Nonconforming

Material, Parts, or Components." The inspectors concluded that the following nonconformance reports required additional information or review in order to assess licensee performance for these particular examples:

2.2.1 Nonconformance Report 95070067

Nonconformance Report 95070067 described the failure of Valve 3HV9339 shutdown cooling suction isolation, to fully open during a stroke on July 23, 1995. At 70 percent of the full-open position, the torque switch actuated and terminated the stroke. The licensee repeated the stroke and observed identical results.

Valve 3HV9339 was a 16-inch WKM Model D-2 "Pow-R-Seal" valve which consists of a split, parallel-expanding disc that provides wedging action in both the open and closed position. The valve was kinematically restrained by a "Lev-R-Loc" mechanism, consisting of a shoe attached to a pivoting arm mounted on the gate. Guide rails are fastened with cap screws to skirts which fit around each seat. The "Lev-R-Loc" shoe rides between the guide rails during intermediate valve travel, preventing lateral expansion of the gate and segment portions of the disc. In the full-open or full-close positions, the "Lev-R-Loc" shoe was permitted to clear the rails and, thereby, enable the valve to wedge. This was due to the angled gate and segment surfaces sliding relative to each other until mechanically restrained by the seating surface.

Upon examination of the valve internals, the licensee discovered that both gate rail segments had become detached and had fallen into the valve body. The gate rails were originally attached to the skirt with cap screws, all of which had sheared. One of the gate rails had fallen to the bottom of the valve body, which did not impair valve operation. However, the other rail was positioned in the conduit section of the gate/segment assembly perpendicular to the stem. This prevented the valve from opening beyond 70 percent of full open. The upper ends of the gate rails were galled at the point where the "Lev-R-Loc" shoe contacts the gate rail during a closing stroke starting from the full-open position. The licensee's WKM valves were operated in this manner until 1991, at which time the valves were reset to open to a lesser extent such that the "Lev-R-Loc" shoe remained restrained on the gate rail while the valve was open. This revision to the valve's operation was intended to prevent the "Lev-R-Loc" shoe from imparting an impact load on the guide rails during closing strokes. Based on these facts, the licensee speculated that most of the damage to Valve 3HV9339's guide rails occurred prior to 1991 and that repeated strokes since then, though imparting a reduced load, were sufficient over time to caused the rails to detach.

The licensee had 48 WKM valves in safety-related service (24 per unit), as follows: 4 main steam isolation valves, 8 main steam isolation and main steam block valves, 16 component cooling water valves, 12 shutdown cooling valves, and 8 safety injection tank valves.

After failure of WKM "Lev-R-Loc" components in the main steam isolation valves at another nuclear facility, the licensee undertook an inspection program of its main steam isolation valves. Valve inspections conducted in 1988, 1989, and 1991 revealed some broken and loose cap screws, broken skirts, and galling on the upper surfaces of the guide rails. A contractor evaluation concluded that the observed valve damage was caused by wear or cycle related degradation of the shoe/rail interface, which was accentuated by high stem speeds, a small guide rail chamfer angle, and a large number of valve cycles. The licensee re-angled the chamfers of the main steam isolation valves to lessen the impact loading.

At that time, the licensee was confident that the WKM guide rail problems were most likely limited to the high speed main steam isolation valves. These valves had stem velocities ranging from 6 to 8 inches per second. Other WKM valves and their stem speeds were as follows:

- Main feedwater isolation and block valves (2 to 3 inches per second),
- Component cooling water valves (less than 1 inch per second),
- Shutdown cooling valves (less than 1 inch per second), and
- Safety injection tank valves (less than one inch per second).

The failure of Valve 3HV9339's guide rails, in light of the slow stem speed of this valve, was unexpected.

The licensee prepared a justification for continued operation report, dated September 15, 1995, and delivered a presentation at the NRC Region IV office. In the report, the licensee defended its previous corrective actions relative to this failure mode of WKM valves based on the information available at the time. Using a test valve, the licensee demonstrated that WKM valves could operate successfully with the guide rails removed from the valve. The licensee also predicted that no modes of failure associated with detached guide rails could occur that would prevent closure of the valves or preclude opening to any position less than 70 percent of full open. The licensee admitted that leakage functions of a WKM valve could be impaired by broken guide rail components, such as sheered cap screws, if these parts were to become wedged between sealing surfaces. The report assessed the safety significance of the failure mechanism as being very low because of train redundancy and expectation of adequate flow at the assumed worst-case, partially-open position. In recognition of the unexpected failure of Valve 3HV9339, the licensee stated that it would reassess its approach to WKM valve repairs.

In response to this event, the licensee had reviewed diagnostic test traces of its WKM valve population for indications of impact loads associated with shoe/rail interactions. Valve 3HV9339 had not been tested prior to its failure. A preliminary review revealed no obvious problems. At the time of this inspection, the licensee was assembling a team of reviewers to assess the traces for subtle effects that may have gone unnoticed in the original review.

The licensee had not performed any valve internal inspections in response to this event. However, some valves had been inspected previously for other reasons. Other than the main steam isolation valves discussed above, these inspections had not documented any guide rail problems.

The inspectors reviewed Nonconformance Report 95070067 and the supporting justification for continued operation. The inspectors discussed this subject with the licensee and examined a three-dimensional scale model of a WKM valve.

The inspectors concluded that the licensee's overall response to this event was acceptable and that the current operability of the WKM valves was not in question. However, several aspects of the issue were subjects of potential concern. Specifically,

- The valve, that failed, had very slow stem speeds and, according to the licensee's contracted study, should not have been vulnerable to guide rail damage. Therefore, this failure could suggest an alternate causal mechanism, a flaw in the contractor study, or a manufacturing/assembly problem, all of which could suggest the potential for similar problems to exist currently in other WKM valves.

- The licensee did not perform inspections of other WKM valves to confirm its assumption that the detachment of guide rails in Valve 3HV9339 was an isolated event.

- Because of the complexity of the WKM valve and uncertainties associated with the movement of separated guide rails and sheered cap screws, the licensee's judgement that the closing function of the WKM valves could not be affected by this problem, or that the opening function could not be impaired beyond that encountered with Valve 3HV9339, appeared to not be supported by existing information.

- The licensee's review of WKM diagnostic traces for subtle effects had not commenced at the time of the inspection. Given the lack of internal inspections, the complete review of diagnostic traces appeared untimely since it had not been completed 4 months following the event.

The licensee's review of this event was still in progress at the end of the inspection. The concerns identified above will be followed as an inspection followup item (361; 362/9526-01).

2.2.2 Nonconformance Report 94070015

Nonconformance Report 94070015 was prepared to document that a Unit 2 safety injection header bypass relief valve had been set at an incorrect set pressure. The relief valve was required to be set at a pressure of 615 psig minus 50 psig back pressure for a total spring set pressure of 565 psig. The licensee concluded that back pressure had not been considered during bench testing. The licensee concluded that the valve was operable since the lines upstream of the valve were analyzed based on a pressure of 700 psig and the design pressure of the line was 700 psig. In addition to the nonconformance report, the licensee prepared Corrective Action Request CAR-010-94, which addressed a number of other pressure relief valves that had been set without considering back pressure. Long-term corrective actions included improving documentation for pressure relief valves to include back pressure criteria.

The inspector questioned if the licensee had verified that the relief valve's spring was still in its linear range, since the spring set pressure was 50 psig greater than the intended value. The licensee agreed to verify this aspect of the valve's design.

2.3 Review of Root Cause Evaluation

The inspectors reviewed Root Cause Evaluation 95-013, "Failure Analysis of Square-D Linestarters." During the Unit 2 Cycle 7 Refueling Outage, several control power fuses blew during attempts to cycle motor-operated valves. The licensee discovered, during troubleshooting, that the mechanical interlock on the Square-D linestarters had become jammed, preventing the contactor coil from moving as designed. The licensee used a staff of approximately six engineers to perform detailed root cause evaluations along with other activities requiring exceptional technical skills. The inspectors discussed Root Cause Evaluation 95-013 with two of the engineers responsible for its development. All questions were satisfactorily resolved.

The inspectors considered this root cause evaluation to be superior for the following reasons. The report clearly presented the problem and discussed possible causes by giving both supporting and refuting evidence. The root cause, which was identified as the repetitive mechanical challenging of the interlock during maintenance activities, was well supported. The exclusion of alternate root causes was equally well supported. The report contained a large number of photographs that clearly documented the condition of the equipment and source of the problem. The report was of sufficient quality to inform, in a precise and understandable manner, engineers who had no previous experience with the problem.

2.4 Review of Surveillance Tests

The inspectors reviewed the following surveillance tests to determine whether components were being properly tested to assure operability:

2.4.1 Surveillance Test SO23-3.18

The inspectors reviewed Surveillance Test SO23-3.18, "Component Cooling/Saltwater System Monthly Test," dated October 5 and November 2, 1995, for Unit 2, and tests dated October 12 and November 9, 1995 for Unit 3. These tests satisfied the Technical Specification 4.7.3 requirement that at least once per 31 days each valve was to be verified that it was in its correct position. The inspectors concluded that the Technical Specification surveillance requirements had been documented as being demonstrated in the surveillance test results.

2.4.2 Surveillance Test SO23-3-3.31.3

The inspectors reviewed Surveillance Test SO23-3-3.31.3, "Component Cooling Water Check Valves Test," Revision 0, with Technical Change Notice 1. The purpose of the test was to insure that the component cooling water pump discharge check valves opened fully and the pumps passed their rated flow rate. The inspectors reviewed a summary of test data for the six discharge check valves for Units 2 and 3 and found that flow rates were in excess of the minimum requirements. The inspectors also reviewed the Unit 2 pumps ASME Section XI inservice test results. The inspectors found that the system engineer was trending the test results. The inspectors concluded that the trending program was very good and the pumps were performing well.

2.5 Review of System Report Cards

The inspectors reviewed the system report cards (termed "mimic") for the emergency core cooling and component cooling water systems covering the period December 1, 1994, to May 31, 1995. These reports reflected the material condition of the subject systems, the timeliness of the surveillance program, and out-of-service times. The report conveyed generally positive results and did not identify any problems of significant concern. The inspectors' review of these two systems during the inspection substantiated the information in these reports.

3 DESIGN ENGINEERING (37550)

3.1 Overview

The San Onofre Nuclear Generating Station engineering organization was divided under two vice presidents. The nuclear engineering design organization was one of the three organizations that reported directly to the nuclear engineering and construction division. This division then reported directly to the Vice President of Engineering/Technical Services. The station technical division reported to the Vice President of Nuclear Generation. The responsibilities of design engineering were the development and implementation of the design and performance of all design activities, and consolidation and maintenance of the plant design basis.

3.2 Review of Design Input Information

As part of the evaluation of the design basis of the safety injection and component cooling water systems, the inspectors reviewed the updated final safety analysis report and noted the following six minor discrepancies:

3.2.1 Safety Injection System

- Section 6.3.2.2.5.1 stated that the thermal relief valve capacities were 5 gallons per minute. The licensee was unable to find documentation to support this statement.

- Section 6.3.3.3.1 stated that the time delay for the start of the high pressure coolant pumps was assumed as 30 seconds. The actual delay was 31.2 seconds.

- Section 6.3.4.1 stated that the maximum postulated flow velocity approaching the recirculation sumps was 0.22 feet per second. The correct value was 0.23 feet per second.

- Section 6.3.4.1 stated that the maximum head loss through the trashrack, screen, grating cage, and intake, plus two bends, at 3900 gallons per minute was 0.30 feet. The supporting calculation stated that this maximum value was 0.27 feet.

3.2.2 Component Cooling Water System

- Section 9.2.2.3 stated that "The component cooling water system operates continuously during normal plant operation and shutdown, under flow and pressure conditions that approximate accident conditions. Provisions are incorporated in the design to facilitate periodic starting of the component cooling water pumps and verification of the required flow path at pressure conditions approximating the accident conditions. These operations demonstrate the operability, performance, and structural integrity of all component water system components." The licensee could not produce the documentation supporting this statement.

  During the inspection, the licensee initiated and submitted Change Request SAR23-412, dated November 22, 1995, that deleted "and verification of the required flow path at pressure conditions approximating the accident conditions. These operations demonstrate the operability, performance, and structural integrity of all component cooling water system components." In the description of why the change was made in the updated final safety analysis report change request, the licensee stated that the statement did not provide a clear definition for conditions of tests and that other tests would satisfy this requirement.

- Section 9.2.2.4 stated that operability testing was performed to "verify the operation of the component cooling water pumps at various flows determined by valve sequencing." During the inspection, the licensee initiated and submitted Change Request SAR23-412, dated November 22, 1995, which deleted this sentence. In the description of why the change was made, the licensee stated that the normal system operation and inservice testing of components were addressed in another section of the updated final safety analysis report and the deleted sentence had no meaning.

As the inspectors identified these discrepancies, the licensee initiated a updated final safety analysis report change request. The inspectors considered the discrepancies to be minor, but they represented a concern with the overall accuracy of the updated final safety analysis report. This finding was accentuated by the fact that the inspectors performed a very limited review and found at least a minor discrepancy in approximately one out of every three items reviewed. In addition, the inspectors noted that the licensee did not periodically perform a review of the accuracy of the updated final safety analysis report in the absence of a design modification, such as during routine system engineering system walkdowns.

3.3 Review of Design Calculations

The inspectors reviewed six design calculations. The inspectors discussed these calculations with the responsible design engineers. No errors were identified. The inspectors concluded that the calculations were thorough and reflected good engineering practices, including adequate discussion of design inputs, assumptions, references, and methodologies.

3.4 Review of Design Change Process

The inspectors reviewed Design Change Package 2-6863.00SN, "Cross-Connection of the Shutdown Cooling System and Spent Fuel Pool Cooling System to the Containment Spray Pump Suction/Discharge Headers," Revision 0. The extent of the inspectors' review included the modification scope and description, impact reviews, revisions to licensing documents, and the 10 CFR 50.59 safety evaluation. The inspectors concluded that this design modification package was satisfactory in all respects, reflecting a comprehensive and conservative methodology. The safety evaluation was broad in scope and clearly presented.

The inspectors reviewed nine minor modifications performed under the Engineering Design Quality Procedure SO123-XXIV-10.21, "Field Change Notice (FCN) and Field Interim Design Change Notice (FIDCN)," Revision 5. The procedure described the requirements for performing a design change under the field change notice. The field change notice was a stand-alone document used only to implement small scope changes.

The inspectors reviewed Work Request 2056, dated February 2, 1994, and Field Change Notice F10023E, dated July 13, 1994. The documents were prepared to wire contacts from each of the high pressure safety injection, low pressure safety injection, and containment spray pumps in parallel with the associated component cooling water low flow alarm contacts in order to disable nuisance alarms in the control room when the pumps were not in operation. In addition, the inspectors reviewed Field Change Notices F10801M, dated February 3, 1995; F10802M, dated February 3, 1995; F10803M, dated February 3, 1995; F10804M, dated February 3, 1995; F10805M, dated February 3, 1995; F10806M, dated February 3, 1995; and, F10807, dated February 3, 1995. These changes were prepared to replace a check valve and revise flow diagrams, isometric drawings and bills of material for valve tag number changes. The inspectors concluded that all of the changes were minor and the field change notices were well done.

4 STATION TECHNICAL (37550)

4.1 Overview

The station technical division, which included the system engineers, reported directly to the Vice President of Nuclear Generation. The responsibility of this division was to provide engineering support of plant operations and maintenance, plant modifications, computer hardware/software, and compliance.

4.2 System Engineers

During the course of the inspection, the inspectors had numerous interactions with the system engineers assigned to the component cooling water and safety injection systems. These contacts included field walkdowns, discussions concerning technical issues, and a demonstration of computer trending programs. Several observations were considered noteworthy. The system engineers were very knowledgeable of their systems, both from a configuration standpoint and in reference to current developments affecting the systems. This included familiarity with maintenance and modification activities and past and present nonconforming conditions. The system engineers were very active in trending safety-related parameters associated with their systems and appeared to have the capability to identify evolving problems very early in their development. From the walkdowns, it was evident the system engineers were spending a substantial amount of time in the plant. The inspectors determined that the system engineers were technically competent, motivated, and sufficiently empowered by their management to impart a major contribution to the safe and efficient operation of their systems.

5 SAFETY EVALUATIONS (37001)

This inspection included a review of the programs and processes instituted by the licensee to comply with the requirements of 10 CFR 50.59. The primary focus of this inspection was on the licensee's implementation of its 10 CFR 50.59 safety evaluation program. The inspectors reviewed selected 10 CFR 50.59 safety evaluations prepared to support plant design changes and to evaluate the significance of deficiencies identified through the nonconformance report process.

A weakness was identified in the implementation of Procedure SO123-XXIV-10.21, "Field Change Notice (FCN) and Field Interim Design Change Notice (FIDCN)," Revision 5, used by the licensee for field change notices. Field change notices are stand-alone documents used by the licensee to implement small scope changes. The licensee defined small scope changes as those that would not result in major changes to plant function or any changes to design bases described in primary design drawings, or any regulatory design commitment documents. If the proposed change exceeds these criteria, the change must be processed as a design change package. Procedure SO123-XXIV-10.21 included a field change notice decision tree that served several purposes. The procedure provided guidance on how to classify a change as a field change notice or a design change package. The procedure also provided the criteria for assisting the evaluator in determining the need for a detailed 10 CFR 50.59 safety evaluation.

The inspector's limited review of several hundred field change notices found that very few facility change notices had a detailed 10 CFR 50.59 safety evaluation performed. Instead, the licensee used a safety evaluation screening process (i.e. a yes or no checkoff) to indicate whether a 10 CFR 50.59 evaluation was required. However, the individual performing the 10 CFR 50.59 screening was not required to provide a basis for the answers to the screening questions. Thus, a second party reviewer would not know the basis for the preparer answers to the screening questions. The inspectors identified this concern to the licensee as a weakness in their procedure.

The inspectors reviewed several field change notices and found one example where the procedural guidance had not been properly followed. Field Change Notice F09329M was processed to modify the reactor coolant gas venting system by replacing an existing flow-limiting orifice with a gate valve that acts like an orifice when closed. The gate valve had a hole drilled in the disc, which would act as a flow-limiting device only when fully closed. Updated Final Safety Analysis Report, Paragraph 9.3.7.1, "Design Bases," indicated that the design function of the flow-limiting orifice was to limit flow for postulated breaks downstream of the orifice so the mass flow rate of reactor coolant would be less than the makeup capacity of a single charging pump. Figure 9.3-15, "Reactor Coolant Gas Vent System Sketch," depicted the layout of the reactor coolant gas venting system, and included the flow-limiting orifice. This modification also required a revision to Procedure SO23-0-17, "Locking of Safety-Related Critical Valves and Breakers," Revision 10, Temporary Change Notice 10-41 to assure the valve was locked during Modes 1 through 4. The licensee processed this field change notice without performing a detailed 10 CFR 50.59 safety evaluation.

Procedure SO123-XXIV-10.21 stated in the introduction that a change processed as a field design change shall not result in any change to the design bases described in the updated final safety analysis report. If it does, the procedure required that the change be processed as a design change package according to Procedure SO123-XXIV-10.16, "Development, Review, Approval and Release of Conceptual Engineering Packages (CEPs) and Design Change Packages (DCPs) SONGS 1, 2 & 3."

The replacement of a fixed orifice with an orifice gate valve introduced the possibility of this valve being left in a less-than-fully closed position. Since the function of the orifice was to limit mass reactor coolant system flow during accident conditions, the orifice gate valve would not perform this function if in an open position. Thus, the licensee relied on administrative controls to assure that this flow-limiting orifice satisfied its design function.

The inspectors determined that the description in the updated final safety analysis report was of sufficient detail to conclude that the replacement of a flow-limiting orifice with a orifice gate valve constituted a change in the design bases. This would have required that the modification be processed as a design change package, with an accompanying 10 CFR 50.59 safety evaluation. This was the first example of the failure to follow procedures. The inspectors also concluded that, even though the licensee did not determine the modification to be a change to the design bases, a detailed 10 CFR 50.59 safety evaluation was required by Procedure SO123-XXIV-10.21, Section 2.2.3.3. This was required if the change involved a minor change to function that did not affect the design bases. In addition, the licensee had not instituted a change to the updated final safety analysis report as required. This was the second example of the failure to follow procedures. The above two examples constitute a violation of NRC requirements (361; 362/9526-02).

6 Followup on High Energy Line Break Issue (92903)

6.1 Description of Issue

On November 27, 1995, the licensee issued several nonconformance reports describing the potential effects of a high energy line break on safety-related areas. These reports described how adjacent safety-related areas could be affected by steam and moisture when transmitted via the heating, ventilating, and air conditioning system. These problems were identified through the licensee's barrier control program, which was a formal program for the control of plant hazard barriers. The licensee's analysis identified interactions with all buildings except the diesel generator and saltwater pump rooms. Concurrent with the issuance of these reports, design engineering provided a summary of the potential interactions. This summary listed the building, area or system, interaction with a high energy line break, estimated risk (low, medium, or high), basis for operability, and short- and long-term actions. In some areas where the licensee could not prove the affected equipment would be operable, compensatory actions were taken. These actions included realigning the heating, ventilating, and air conditioning system to prevent steam and moisture intrusion. These were done using Procedure SO-0-23, "Control of System Alignments," Revision 0.

6.2 Background

The licensee identified that they did not have a formal program to control plant hazard barriers prior to September 1993. Specifically, barriers such as doors, hatches, and penetration seals could be blocked open to support work activities without an engineering evaluation of the effects on the barrier's design function. Inspection Report 50-362/93-29 documented a resident inspector's finding that the licensee had not considered the impact of the removal of watertight floor plugs and the opening of doors on the design basis flooding accidents. The inspector concluded that an underlying reason for the condition was an inadequate program to control watertight doors and plugs in the context of their role to prevent flooding, and the ensuing lack of procedural guidance. This issue was followed as an unresolved item and, following further evaluation, resulted in the issuance of a violation.

As corrective action for the violation the licensee formed an interim barrier control program. This program required that the design engineering organization perform an operability evaluation of a breeched barrier in accordance with 10 CFR 50.59 requirements. This evaluation would consider the breeched barrier's effect on flooding, fire, security, missiles, steam, and radiation. The licensee estimated that this had resulted in approximately 1500 evaluations performed thus far under the interim barrier program. The licensee was performing, in parallel, the completion of the long-term barrier control program. This would create allowed outage times for each barrier, with a 10 CFR 50.59 safety evaluation required only after the expiration of the allowed outage time. The supporting analysis will be presented to the Office of Nuclear Reactor Regulation in the near future. The inspectors considered this response to an NRC violation to be an outstanding effort.

However, as part of the resolution of this issue, a design engineer noted that the interim barrier control program had only been analyzing the effects of a high energy line break on nearby equipment. The engineer further noted that the effects on safety-related equipment by interactions with nonsafety-related components (heating, ventilating, and air-conditioning systems) had never been considered by the plant's original architect engineer. In June 1995, as part of the licensee's activities to develop the long-term barrier control program, the licensee recognized, during a walkdown, the potential for steam to propagate to the auxiliary feedwater enclosure following a high energy line break in the main steam isolation valve area. This interaction was determined to be possible through the heating, ventilating, and air conditioning system. The main steam isolation valve area was designated as a harsh environmentally qualified area while the auxiliary feedwater enclosure was a mild area. Nonconformance Reports 95060041 and 95060042 were written to address this concern for each unit, respectively.

It was due to this independent discovery that the licensee expanded their program to identify whether any other mild areas could become harsh areas due to steam or moisture propagation via the heating, ventilating, and air conditioning system. This proactive effort resulted in the licensee's determination on November 15, 1995, that several plant areas could be similarly affected. The licensee's immediate corrective actions were based on the fact that operability of the affected equipment could not be assured.

6.3 Status of Issue

At the conclusion of the inspection, the licensee was still in the process of determining the extent of the problem and the effect on equipment operability due to temperature and humidity. A long-term action proposed was to install new back draft dampers which would be able to close against flow to isolate on a high temperature condition. The inspectors will review the completion of the licensee's analysis to determine the effect on equipment operability as an inspection followup item (361; 362/9526-03).

The licensee initially concluded that the problem identified was potentially generic to other plants that used Bechtel Topical Report BN-TOP-2 for high energy line break analysis and NUREG-0588 Category 2 criteria for environmentally qualified program development. The licensee subsequently determined, after discussions with Bechtel, that the error in the licensee's environmentally qualified program was not applicable to other plants. However, the licensee did distribute this information over the industry's network system.

In retrospect, the inspectors noted that design engineering had determined that a potential problem existed on November 15, 1995, but had not initiated nonconformance reports until following the holiday on November 27. The inspectors were concerned that operations personnel should have been informed of this issue prior to November 27. The licensee's engineering managers indicated that the operations department had not been informed prior to this date. Although design engineering management had requested that operations be notified on November 15, this action had not been taken due to an oversight. When questioned by the inspectors on why a nonconformance report was also not written on November 15, the licensee responded that the issue had not been fully validated until November 27. However, the licensee did agree with the inspectors that operations personnel should have been informed earlier. The inspectors concluded that operations personnel should have been informed as early as November 15, and that one way to assure this was to issue a nonconformance report. The licensee agreed with this observation.

7 FOLLOWUP ON RELATED ENGINEERING ISSUES (92903)

7.1 <u>(Closed) Violation 361; 362/9510-01: Design Basis Information Not Correctly Translated Into Specifications, Drawings, Procedures, and Instructions.</u>

<u>Background</u>

This violation concerned the discovery that from initial operation until March 31, and April 14, 1995, for Units 2 and 3, respectively, the design basis of maintaining a minimum 75 percent of nominal voltage at the diesel generator terminals was not correctly translated into specifications and drawings. Specifically, the licensee determined that the voltage would drop below 75 percent for the design basis loading sequence which would occur during a loss of feedwater event with a loss of normal alternating current on a reactor trip, with a high pressure safety injection pump connected at the time of the event. The licensee's investigation resulted in the identification of events with potentially more severe voltage drops than due to the event described above. However, the licensee's analysis concluded that the emergency diesel generators would have been operable during these postulated events. These design problems were determined to be initial design flaws. The licensee submitted voluntary Licensee Event Report 95-009 to document this finding and describe the corrective actions.

The violation also stated that the licensee's design basis required the assurance that electrical system voltage and frequency would be restored in less than 40 percent of each load sequence time interval. However, Calculation E4C-082, "System Dynamic Voltages During Design Basis Accident," Revision 1, incorrectly included acceptance criteria for voltage and frequency restoration of less than 60 percent of each load sequence interval. The licensee determined the cause of this to be a personnel error by the individual performing the calculation in that the incorrect regulatory guide revision was referenced. The licensee also found that an inconsistency as to the correct regulatory guide revision existed in the updated final safety analysis report. The licensee's corrective actions included revising the calculation, issuing an updated final safety analysis report change request, and providing the details of this finding to appropriate personnel.

In addition, the violation stated that the licensee found that the Class-1E 4160V switchgear circuit breakers on Buses 2A04, 2A06, 3A04, and 3A06 were not included in Surveillance Operating Instruction SO23-3-3.23.1, "Diesel Generator Refueling Intervals Tests." The licensee's investigation concluded that this was an isolated personnel error by the procedure writer. The procedure was revised and the individual counseled.

Inspector Followup

The inspectors verified that the licensee had completed the corrective actions for both Units 2 and 3 during the last refueling outages, and included modifications to control circuitry to prevent the voltage dropping below 75 percent. This action was sufficient to close this example of the violation and Licensee Event Report 95-009.

7.2 (Closed) Inspection Followup Item 361; 362/9513-01: Heat Exchanger Performance

Background

At the time of the previous inspection, the licensee was in the process of developing calculations to evaluate heat exchanger performance and had not yet tested the Unit 3 component cooling water heat exchangers using newly installed test equipment (thermowells and temperature detectors). This item was opened to review these items upon completion.

Inspector Followup

The licensee had tested the Unit 3 component cooling water heat exchangers and had completed calculations to analyze test data and correlate test conditions to design basis conditions. The inspectors reviewed Calculation EC383, "CCW HX Performance U3C8," which evaluated test data taken on the Unit 3 component cooling water heat exchangers during plant cooldown for the Unit 3 Cycle 8 refueling outage. This test indicated that the Unit 3 Train A component cooling water heat exchanger could remove the design heat load of 144,210,000 BTU/hr, at the design ocean inlet temperature of 76°F, such that water exiting the heat exchanger would not exceed 95.47°F. The corresponding outlet temperature for the Unit 3, Train B, component cooling water heat exchanger was 94.22°F. Because the design temperature for the heat exchanger outlet was 105°F, the test demonstrated a large heat capacity margin for both Unit 3 component cooling water heat exchangers.

The inspectors discussed the test results with licensee engineers and concluded that the licensee had satisfactorily addressed this issue.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

D. Axline, Licensing Engineer
P. Blakeslee, Senior Engineer, Station Technical
D. Breig, Manager, Station Technical
J. Brower, Manager, Plant Operations Group
D. Bradford, Engineer, Design Engineering
E. David, Senior Engineer, Nuclear Design Engineering
G. Gibson, Manager, Compliance
D. Irvine, Supervisor, Technical Support
G. Johnson, Engineer, Nuclear Design Engineering
W. Marsh, Manager, Nuclear Regulatory Affairs
D. Niebruegge, Supervisor, Motor-Operator Valve Group
G. Plumlee, III, Supervisor, Regulatory Compliance
J. Rainsberry, Manager, Plant Licensing
R. St. Onge, Manager, Nuclear Engineering Projects
P. Schofield, Supervisor, Performance Monitoring
A. Thiel, Manager, Electrical Systems Engineering
J. Thomas, Senior Engineer, Safety Engineering
M. Wharton, Manager, Nuclear Design Engineering
T. Yackle, Manager, Nuclear Mechanical

1.2 NRC Personnel

M. Fields, Project Manager, Nuclear Reactor Regulation
P. Goldberg, Reactor Inspector, Engineering Branch, Region IV
R. Mullikin, Reactor Inspector, Engineering Branch, Region IV
M. Runyan, Reactor Inspector, Engineering Branch, Region IV
J. Russell, Resident Inspector
J. Sloan, Senior Resident Inspector
C. VanDenburgh, Chief, Engineering Branch, Region IV

The above personnel attended the exit meeting on December 1, 1995.
Mr. D. Axline attended the re-exit meeting on January 3, 1996, via telephone.

2 EXIT MEETING

An exit meeting was conducted on December 1, 1995. Another exit meeting was held via telephone on January 3, 1996. During these meetings, the inspectors reviewed the scope and findings of the inspection. During the December 1 meeting, the licensee disagreed with the basis for a proposed non-cited violation concerning the failure to perform a detailed 10 CFR 50.59 safety evaluation. Upon further evaluation, this finding was determined to be a cited violation. During the January 3 meeting, the licensee agreed that their procedures had not been followed, but again expressed the opinion that a violation of 10 CFR 50.59 had not occurred. This item is discussed in Section 5 of the report. The licensee did not identify any information provided to, or reviewed, by the inspectors as proprietary.

ATTACHMENT 2

LIST OF DOCUMENTS REVIEWED

NONCONFORMANCE REPORTS

- 93060044, Revision 2
- 93060045, Revision 3
- 93060046, Revision 2
- 93060047, Revision 0
- 93060048, Revision 2
- 93060049, Revision 0
- 94010033, Revision 0
- 94010045, Revision 1
- 94030014, Revision 0
- 94030015, Revision 1
- 94040036, Revision 0
- 94040045, Revision 0
- 94040046, Revision 0
- 94040047, Revision 0
- 94040056, Revision 0
- 94050004, Revision 1
- 94050040, Revision 0
- 94060020, Revision 2
- 94070015, Revision 0
- 94080009, Revision 0
- 94090029, Revision 0
- 94090038, Revision 0
- 94090039, Revision 0
- 94100004, Revision 0
- 94110006, Revision 0
- 94110043, Revision 1
- 94120002, Revision 0
- 94120005, Revision 0
- 95010037, Revision 0
- 95020069, Revision 0
- 95030027, Revision 0
- 95030029, Revision 0
- 95020045, Revision 2
- 95020063, Revision 1
- 95030065, Revision 0
- 95020067, Revision 0
- 95020070, Revision 0
- 95020102, Revision 0
- 95020103, Revision 1
- 95020104, Revision 1
- 95030040, Revision 0
- 95030056, Revision 0
- 95030065, Revision 0

- 95030077, Revision 0
- 95030083, Revision 2
- 95030105, Revision 0
- 95030148, Revision 0
- 95030195, Revision 1
- 95040017, Revision 1
- 95040020, Revision 0
- 95050058, Revision 1
- 95050088, Revision 0
- 95060011, Revision 0
- 95060017, Revision 0
- 95060041, Revision 0
- 95060042, Revision 0
- 95060083, Revision 1
- 95060092, Revision 1
- 95070067, Revision 0
- 95070080, Revision 0
- 95070092, Revision 0
- 95070103, Revision 0
- 95070109, Revision 0
- 95080018, Revision 1
- 95080076, Revision 0
- 95080128, Revision 0
- 95080178, Revision 0
- 95090002, Revision 0
- 95090044, Revision 0
- 95110001, Revision 0
- 95110006, Revision 0
- 95110064, Revision 0
- 95110065, Revision 0
- 95110066, Revision 0
- 95110067, Revision 0
- 95110068, Revision 0
- 95110069, Revision 0
- 95110070, Revision 0

CALCULATIONS

- M-0012.027, "HPCI and LPCI IST Minimum Performance Requirements," Revision 0

- M-0012.030, "HPCI Mini-Flow Path Flowrate and Volume Calculation Following RAS for Leakage by S2(3) 1204MU010/011," Revision 0

- M0027-023, "CCW/SWC Heat exchanger Operability," Revision 0, Calculation Change 1

- J-EGA-019, "Uncertainty in CCW Heat Exchanger Performance Measurement," Revision 3

- N-4080-027, "Containment P/T Analysis for Design Basis MSLB," Revision 0, and Calculation Changes 1 and 2
- J-EAG-0111, "CCW Surge Tank Local Level Indication, Revision 0

SURVEILLANCE AND MAINTENANCE TESTS

- Surveillance Test SO23-3.18, "Component Cooling/Saltwater System Monthly Test"
- Surveillance Test SO23-3.31.3, "Component Cooling Water Check Valves Test," Revision 0, Temporary Change Notice 1
- Maintenance procedure SI23-I-8.88, "Cold Bench Testing and Calibration of ASME III, ASME VIII, and Non-ASME Safety/Relief Valves," Revision 3, Temporary Change Notice 3-4

DESIGN CHANGES

- Design Change Package 2-6863.00SN, "Cross-Connection of the Shutdown Cooling System and Spent Fuel Pool Cooling System to the Containment Spray Pump Suction/Discharge Headers," Revision 0
- Work request No. 2056, February 2, 1994
- Field Change Notice F09329M
- Field Change Notice F10023E
- Field Change Notice F10801M
- Field Change Notice F10802M
- Field Change Notice F10803M
- Field Change Notice F10804M
- Field Change Notice F10805M
- Field Change Notice F10806M
- Field Change Notice F10807M

PROCEDURES

- SO23-0-17, "Locking of Safety-Related Critical Valves and Breakers," Revision 10, Temporary Change Notice 10-41

- SO123-0-23, "Control of System Alignments," Revision 0, Temporary Change Notice 0-20

- SO123-XV-43, "Site Problem Report (SPR)," Revision 1, Temporary Change Notice 1-5

- SO123-XXIX.2.16, "Retrofit Problem Reporting System," Revision 1, Temporary Change Notice 1-4

- SO123-XV-5, "Nonconforming Material, Parts, or Components," Revision 3, Temporary Change Notice 3-22
- SO123-XXIV-10.21, "Field Change Notice (FCN) and Field Interim Design Change Notice (FIDCN)," Revision 5

MISCELLANEOUS

- Updated Final Safety Analysis Report

- Technical Specifications

- Root Cause Evaluation 95-013, "Failure Analysis of Square-D Linestarters"

- Site Quality Assurance Audit Report SCES-442-94, "Nonconformances," May 1994

- Site Quality Assurance Audit Report SCES-444-94, "Configuration Control," December 1994

- Long-Term Barrier Control Program Overview

- Nuclear Organization Jurisdiction Statement NO-JS-NE&C, "Nuclear Engineering & Construction Division," Revision 0

- SONGS Technical Division Performance Assessment Report - September 1995

- Roles and Responsibilities of STEC Personnel in the System Engineer Program, January 1, 1995