U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No.: 50-361/88-10, 50-362/88-10

Docket No.: 50-361, 50-362

License No.: NPF-10, NPF-15

Licensee: Southern California Edison Company
P. O. Box 800
2244 Walnut Grove Avenue
Rosemead, California 91770

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

Inspection at: San Clemente and Rosemead, California

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S. A. Richards, Chief, Engineering Section — Date Signed: 8-2-88

Summary:

Inspection conducted on May 2-6, May 16-27, and June 6-10, 1988 (Report Nos. 50-361/88-10, and 50-362/88-10)

Areas Inspected: This special, announced team Safety System Functional Inspection (SSFI) involved the areas of Engineering, Maintenance, Surveillance Testing, Operations, Health Physics, Quality Assurance and Administration. During this inspection, inspection modules 30703, 92700, 92701, 92702, 37700, 37701, 37702, 62700, 62702, 62704, 73051, 73055, 71707, 41400, 41701, 83728 83729, 84723, 90701 and 90702 were used.

Results: Three violations and one deviation were identified. The team concluded that the licensee does not fully understand the design of the systems reviewed; that the licensee does not have ready access to accurate design information; and that technical work is not always performed in a complete, technically correct manner. Deficiencies were also identified with testing, maintenance, and operation aspects of the systems reviewed.

DETAILS

1. Persons Contacted

Southern California Edison (SCE)

*D. Fogarty, Executive Vice President
*K. Baskin, Vice President, Nuclear Engineering, Safety, and Licensing
*C. McCarthy, Vice President, Site Manager
*R. Dietch, Vice President, E&C
*H. Morgan, Station Manager
*R. Rosenblum, QA Manager
*F. Briggs, Supervising Engineer, NSSS Electrical
*T. Herring, NES&L Site Representative
*J. Reilly, Manager, Station Technical
*R. Krieger, Operations Manager
*M. Medford, Manager, Nuclear Engineering & Licensing
*J. McMahan, Assistant Maintenance Manager
*J. Wambold, Project Manager
*M. Short, Manager, Nuclear Training
*B. Katz, OMS Manager
*J. Curran, Nuclear Safety Manager
*M. Wharton, Assistant Technical Manager
*D. Cox, Nuclear Licensing Supervisor
*D. Fellows, Nuclear Generation Engineering Manager
*G. Shelton, Salt Water Cooling Cognizant Engineer
*M. Schwaebe, Component Cooling Water Cognizant Engineer
*D. Nunn, Chief Engineer - TF Leader
*C. Couser, Compliance Engineer
*M. Zenker, Compliance Engineer
*J. McGaw, Licensing Engineer
*A. Sistos, Mechanical Engineer, E&C
*T. Vogt, Assistant Plant Superintendent - Operations
*V. Fisher, Plant Superintendent
*B. Carlisle, Project Engineer
*J. Yann, Project Engineer
*D. Eastman, Engineering Manager, E&C
*R. Plappert, Compliance Engineer

USNRC Personnel

In addition to the inspection team, the following NRC Regional Office and Headquarters personnel participated in the inspection activity and/or attended the associated management meetings:

*T. Murley, Director, Office of Nuclear Reactor Regulation
*J. Martin, Regional Administrator, RV
*D. Kirsch, Director, Division of Reactor Safety and Projects, RV
*B. Grimes, Director, Division of Reactor Inspection and Safeguards, NRR
*D. Hickman, SONGS 2/3 Project Manager, PD5, NRR
*J. Tatum, SONGS 2/3 Resident Inspector

*R. Huey, SONGS Senior Resident Inspector
*M. Johnson, Engineer, Office of the EDO

Other Personnel Attending Exit Meeting

*R. Lacy, SDG&E Nuclear Manager
*M. Hug, PG&E Regulatory Compliance Engineer
*G. Edwards, Power Resources Project Manager, City of Anaheim
*S. Harris, Power Resources Engineer, City of Riverside

* Denotes those personnel in attendance at an exit meeting on June 10, 1988.

In addition to those individuals listed above, the team interviewed numerous other licensee personnel, including engineers, operators, maintenance technicians, and administrative personnel.

2. Inspection Objectives

This team inspection had three main objectives, as follows:

- To assess the licensee's access to accurate and complete design information for Units 2 and 3, and to assess the licensee's understanding of the design.

- To assess the licensee's engineering capability and their performance of technical work at both the corporate general office (GO) and at the San Onofre site.

- To assess whether all design requirements are maintained and system operability is assured through appropriate operations, maintenance, and testing activities.

3. Basis and Scope of Inspection

In defining the scope of this inspection, the NRC considered generic probablistic risk assessment data to identify systems of high safety significance, which interfaced with multiple components of several safety systems. Recent NRC findings relating to engineering and quality assurance problems (e.g., root cause assessment) were also considered, in addition to equipment failure data from the Nuclear Plant Reliability Data System (NPRDS). This review process resulted in the selection of the component cooling water (CCW) system and the salt water cooling (SWC) system, as the focus of this inspection. Interfacing electrical, mechanical, and instrumentation/control components and systems were also reviewed in a less focused manner.

4. Inspection Approach

A Safety System Functional Inspection (SSFI) was performed on the selected safety systems. An SSFI type inspection is a design based inspection process. One of the chief advantages of this type of inspection is that it concentrates a comprehensive, multi-discipline review into a relatively narrowly defined area. The intent of the

inspection is to define an inspection area that is not only important to plant safety or accident mitigation, but to select an area involving a broad cross-section of activities by the various disciplines in the licensee organization. This type of approach allows the inspectors to develop a fairly accurate perspective of how well the licensee organization integrates the various aspects of plant design, engineering, operation, maintenance, etc. Recent experience with this type of inspection has shown it to be effective at pointing out deficiencies both in the area of licensee understanding of the design basis for the plant and in the area of control of the design process.

An Engineering Sub-Team was organized, consisting of an Assistant Team Leader, several NRC inspectors, and Design Consultants experienced in functional analysis of design changes. The Consultants were contracted to select appropriate design changes, and then to provide an independent engineering analysis of their impact on safety functions. This included identification of aspects of the modifications which may be particularly sensitive to proper implementation, so that those aspects could be verified by the Site Sub-Team. Such aspects included specifics of modification quality verification, as-built verification, post modification testing, and changes in methods and procedures for operations, maintenance, and surveillance testing.

The Site Sub-Team was organized with the Team Leader and several experienced NRC inspectors, who were assigned to examine on-site implementation of selected design changes and to review, in detail, the site activities related to the selected systems.

5. Component Cooling Water System Reevaluation

During the team's review of the component cooling water (CCW) system, it became apparent that the system has created a number of problems for the licensee over the plant life, basically due to several poor or incorrect design decisions made during the construction of Units 2 and 3. The CCW system was originally designed to be a "zero leakage" system. Because of the zero leakage design basis, the makeup system to CCW, which is the nuclear service water system, was not designed to be seismically qualified or safety related, and is not considered to be available for most accident analysis scenarios. During the plant licensing process, in the Question and Response section of the FSAR, the NRC questioned whether the makeup water source shouldn't be seismically qualified; or requested that a temporary makeup connection be provided and be capable of being activated within 7 days, in conjunction with an analysis that demonstrates that the CCW system can operate without makeup for 7 days. The licensee's response in the FSAR was that the calculated system leakage rate was only 0.008 gpm, based on pump seal and valve seat and stem leakage, and that the system could therefore operate 122 days without a makeup supply. The licensee further stated in the FSAR that no special provisions for additional makeup were required.

The team observed that the licensee's statements in the FSAR apparently did not consider the response of the system under certain accident conditions, such as a Safe Shutdown Earthquake (SSE) or a Main Steam Line Break (MSLB) in the vicinity of a CCW line. During these events, the

failure of a CCW line, due to the event itself, would cause an initial system leakage far in excess of 0.008 gpm. When considered under accident conditions, the need for a seismic, class IE makeup capability is much more apparent. With Units 2 and 3 licensed and operating in 1983, the licensee began to experience problems with the CCW heat exchangers. Introduction of marine debris and marine life into the Salt Water Cooling (SWC) side of the CCW heat exchanger (HX) apparently began to cause HX tube leakage and tube fouling. The licensee recognized at this time that the actual CCW system leakage through the HX tubes was at odds with their statement in the FSAR concerning calculated system leakage. The licensee then calculated that the system could leak at a rate of 0.142 gpm and still operate for 7 days without makeup, however, in a licensee letter to the Bechtel Corporation dated July 18, 1983, the licensee noted that the 0.142 gpm was often exceeded. This same letter authorized Bechtel to design an emergency fire hose connection for the CCW surge tanks to allow for the establishment of a temporary makeup system using the site fire trucks.

The modifications which installed the fire hose connections to the surge tanks took about a year to complete. Apparently, neither the operability of the system nor the reportability of the situation were considered in the interim, and the units continued to operate. Again, the system response during accident scenarios was apparently not considered.

In late 1987, the licensee began to consider moving spent fuel from Unit 1 to Unit 2/3. To accomplish this, the fire truck located in the Unit 2/3 Fuel Handling Building would need to be relocated. The fire truck location is important because when not in use, the truck must be seismically secured. While reviewing the function of the fire truck, the licensee's corporate office engineers were reportedly surprised to learn that the fire truck was required to provide a backup makeup water source for the CCW system, a fact that appears to have been well recognized at the site. In December 1987, the corporate engineers apparently recognized that the effect of CCW system leakage on the system performance during accident conditions could be of significance, and therefore embarked on a task to review this issue. The review basically considered the system operation under the following accident conditions: A Critical Crack Occurrence; a Safe Shutdown Earthquake (SSE); a Loss of Coolant Accident (LOCA); and a Main Steam Line Break (MSLB). The results of the licensee's reevaluation were documented in a memorandum dated March 30, 1988. The reevaluation reached the following conclusions:

- Minor system leakage is not a factor for the Critical Crack Occurrence or a LOCA because one train of the system always remains operable and makeup to the system is not immediately required.

- For a HELB event, the reevaluation calculated a 4550 gpm leak rate from a CCW non-critical loop (NCL) pipe which was postulated to be impacted by the HELB. For this event, the CCW surge tank would isolate on low level if the NCL isolation valves do not close in 15 seconds or less. The technical specifications allow the NCL isolation valves to close as slow as 20.9 seconds. If the surge tank isolates, the associated train of CCW is then in a solid water condition, a condition in which it is not designed to operate nor

had the system's operation in a solid water condition been analyzed. Assuming the NCL isolation valves go closed fast enough to keep the surge tank from isolating, the CCW system external leakage and cross train leakage become critical, in that a relatively small amount of leakage (1-2 gpm) could then cause the surge tank to isolate prior to the operators taking action to hookup the temporary fire hose connection, which the analysis assumed would occur after 60 minutes. It should be noted that the CCW train which is not affected by the HELB is assumed to be the single failure (failure of its associated emergency diesel generator) and the remaining train must remain in continuous service to mitigate the consequences of the HELB.

- For the SSE, the reevaluation concluded that due to a postulated failure of a pipe in the NCL, the critical loop servi ing the non-critical loop will lose water inventory until the surge tank low-low level is reached and both the NCL isolation valves and the surge tank isolation valve automatically go closed due to the low-low level condition. The critical loop is then assumed to be non-functional. Considering the occurrence of a single failure in the opposite train, the result is that neither train of CCW is then available. The reevaluation went on to determine that the reactor coolant system could be maintained stable for up to four hours until the temporary hose connections could be established to refill the surge tank from the fire truck and return one train of CCW to service. The event scenario is complicated by the assumption that a fire occurs on site following the SSE, and therefore the fire truck is unavailable until the fire is put out and the fire truck water tank is replenished.

The corporate office reevaluation was transmitted to the site and addressed at the site via NCR G-852, dated April 28, 1988. The NCR interim disposition accepted the system as operable, however a number of interim actions were required by the NCR. These actions included prestaging of fire hoses to ensure the capability to establish makeup to the surge tanks within 1 hour; operator training on the results of the reevaluation; delineation of actions that must be accomplished following an SSE with indication of a NCL failure to ensure that room cooling is adequate; direction to quantify system leakage; and revision of the inservice testing (IST) program allowable stroke time for the NCL isolation valves from 19.7 seconds to 14.5 seconds. The NCR final disposition was to provide a permanent, seismic category I makeup capability for the CCW system.

Following the issuance of the March 30, 1988 reevaluation results, the licensee continued to refine the reevaluation. The Bechtel Corporation, who had originally designed the system, had been contracted by the licensee to perform calculational work to support the reevaluation and Bechtel continued to reperform calculations related to the CCW system during the inspection.

On April 29, 1988, the licensee submitted Licensee Event Report (LER) 88-008 for Unit 2. This LER only identified that, due to the very low allowable leakage rates stated in the FSAR Question and Responses section, Units 2 and 3 may have operated outside the design basis of the

CCW system prior to the implementation of the design change to allow a temporary hose connection to the surge tanks. This LER did not report any other of the conclusions of the reevaluation of the CCW system. The licensee did note in the LER, that the LER was approximately five years late. The stated cause for the LER being late was that an NCR was erroneously not written on the issue at the time (1983), and that the NCR is the mechanism which initiates a reportability determination.

Members of both the site and engineering inspection sub-teams reviewed the reevaluation in detail, interviewed numerous personnel involved in the reevaluation, and conducted several group meetings with licensee personnel to discuss the licensee's efforts.

Personnel involved in the preparation of LER 88-008 and personnel who may have had knowledge of the system problems in 1983 were also interviewed. The team had the following observations and findings based on this review:

- Although during the plant initial licensing process, the licensee had stated in the FSAR that the system needed no provisions for a seismic makeup system due to the very low system design leakage rate (.008 gpm), the licensee had no provisions in place to ensure that the leakage rate remained below that stated. No system leakage tests were conducted during the commercial life of the facility until March, 1988.

- The original calculation which determined the expected leakage rate to be 0.008 gpm simply added the vendor listed leakage values for pump seals, and valve stems and seats. The calculation did not take into account system component degradation, such as heat exchanger tube erosion and leakage, excessive pump seal leakoff, valve seat or disk erosion, and excessive valve stem leakage. These conditions can normally be called routine and dealt with simply for systems such as CCW, however in light of the restrictive system leakage assumed, these types of minor problems can be very significant and should have been considered.

- When CCW system leakage apparently became excessive in 1983, and was recognized as such, the licensee took action to modify the system to address the leakage, however system operability and reportability were apparently not considered. Team interviews with personnel involved with the issue at the time indicate that they simply were unaware of the importance of leakage. By not thoroughly reviewing the issue at that time (1983), the licensee missed an important opportunity to resolve the issue early in plant life.

- When the corporate engineers began their reevaluation, the various accident scenarios that were considered during initial plant licensing were unclear or unavailable to the point that the analysis was basically completely redone. This is illustrative of the licensee's lack of ready access to basic design information.

- A CCW system single active failure was erroneously not assumed for an SSE event, by the facility's architect engineer. The licensee

apparently was unaware of this fact. In performing the reevaluation, the licensee did assume a single failure and only later determined from the architect engineer what assumptions were originally made.

- During the inspection, the licensee's position on whether a single active CCW failure is required to be considered for an SSE was confused. By the close of the inspection, the licensee indicated their agreement that a single failure was required to be considered.

- With regard to the CCW response under a HELB event inside containment, an analysis to consider this event was apparently not performed during original licensing. The complete analysis was first performed in early 1988.

- A Bechtel calculation which was performed to support the reevaluation, concluded that with a CCW train in a solid water condition, a total leakage of 5 gallons would render the system inoperable. Although the licensee's engineers repeatedly expressed doubt concerning this conclusion, and the conclusion had a significant effect on the overall analysis, the calculation had not been revised as late as four months after it was performed. The licensee's engineers were apparently very hesitant to revise the calculation themselves and chose rather to convince Bechtel that the result was overly conservative.

The team specifically discussed with the licensee, in detail, what reportability and operability considerations were made during the reevaluation. While it is difficult to reconstruct past considerations made by the licensee, it was clear based on a letter from Bechtel to the licensee that as early as January the need to document a Justification of Continued Operation was being considered, and yet the issue was not formally addressed until the NCR was finally written in late April. It appears that operability and reportability determination responsibility lies solely at the site and that the site programmatically need not consider engineering issues until the final corporate office evaluation is forwarded to the site. The licensee's staff stated several times that corporate office engineers do consider these issues, however when challenged by the team as to whether the corporate office ever considered the plant to potentially be in an unanalyzed condition while performing the reevaluation, due to the MSLB event and the SSE event apparently not having been previously correctly analyzed, a licensee manager responded that the corporate office only provides the evaluations they are directed to provide and that basically, reportability is not their line of work. The team found this to be a very narrow and potentially hazardous point of view. The team concluded that the corporate office responsibilities in this regard should be made clear and that the responsibility should not rest solely at the site. The team also concluded that the licensee should document their operability considerations while analytical work is in progress under conditions such as this, so that it is clear that the continued operation of the plant is based on sound thought and logic.

With regard to specific reportable events, the team concluded that the licensee failed to make a required report for the following conditions:

- HELB event not having been properly analyzed - 50.73(a)(2)(ii) - Any condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly compromised plant safety.

- The combination of CCW leakage and an erroneously high allowable NCL valve closure time, which could have prevented the CCW system from functioning during a HELB event, according to the licensee's reevaluation. - 50.73(a)(2)(v) - Any condition that alone could have prevented the safety function of a system needed to mitigate the consequences of an accident.

- The late reporting of LER 88-008.

Failure to make these reports to the NRC within the time allowed is an apparent violation (50-361/88-10-01).

With regard to the licensee's failure to ensure that a single active failure was assumed with an SSE, the licensee maintained that in spite of this error, their reevaluation found that the CCW system would fulfill its safety function during an SSE when a single active failure is considered. The team questioned whether the licensee's procedures and training had been adequate in the past to address an SSE with a single failure. The licensee maintained that the changes required to procedures by NCR G-852 were enhancements, and that their past procedures were adequate. The significance of not having considered a single active failure is an unresolved item (50-361/88-10-02) pending further NRC review of the licensee's past capability to sustain an SSE with a single active failure and a more detailed review of the licensee's original licensing submittals regarding this issue.

6. Engineering and Design Activities

A. General Description

The design portion of the inspection focused on a review of design change packages and original design basis documentation related to the selected safety systems (component cooling water, saltwater cooling and 1E electric systems that interface with CCW and SWC).

Based on the results of the design inspection, the team determined the saltwater cooling system was operable. With regard to the CCW system, the team found the system to be operable to the degree the system was reviewed, however due to the large number of significant issues which have recently surfaced regarding the CCW system, and which have apparently been dealt with by the licensee, the team concluded that a complete review of the system operation and design, on a priority basis, would be prudent. As a result, SCE management at the inspection exit meeting committed to expeditiously assess the CCW system in a thorough manner.

Based on the engineering and design review, the team concluded SCE engineering does not fully understand the systems' basic design. It appears SCE engineering and construction (corporate) engineers do not, in most cases, have detailed knowledge of original design or

design calculations. Design changes appear to be a lower quality product due to a lack of understanding the basic design. The FSAR has not in all cases been maintained current. The team identified instances where technical work was not performed in a complete, technically correct manner. In many cases, design calculations had not been maintained current nor had design assumptions been verified. To SCE's credit, the design review that was conducted in preparation for this safety system functional inspection identified the same types of concerns as did the NRC SSFI team inspection. The team concluded it is essential for SCE to review the basic design to assure a complete and accurate design basis is available for use in future plant design modifications.

Specific examples of these observations are described in Section B below.

B. Design Activities

The team reviewed design documentation related to the CCW and SWC systems, including design change packages (DCP's) and their associated proposed facility change packages (PFC's), system descriptions, P&IDs, and design analyses (calculations), to assess the capability of these systems to perform their safety functions. Additionally, the team reviewed the CCW and SWC design reports and state of the system reports which were recently developed to prepare for the NRC SSFI. For the electrical ancillary support systems, the team reviewed design analyses associated with the Class 1E 4160 VAC, 480 VAC and 125 VDC systems, which included AC and DC voltage calculations and protective device settings and coordination. The review focused on the adequacy of the methodology, the accuracy of the design data, and the validity of the design assumptions utilized in the calculations.

The team identified a number of observations which indicated weaknesses in several areas of engineering activities. These weaknesses included inadequate design analyses and inadequacies in the verification of design analyses. The team noted a general lack of awareness by SCE engineers of the specific design assumptions utilized in calculations.

(1) Inadequate Calculations

a. Adequacy of AC Voltage While Power Is Supplied by Station Emergency Diesel Generators

The Class 1E AC auxiliary power systems for each of Units 2 and 3 consist of two redundant 4.16 KV and 480V systems. The redundant systems of each unit are completely separate and independent. The normal or preferred power source for these systems of each unit is the unit's reserve auxiliary transformer. The reserve auxiliary transformers are fed from the Southern California Edison 230 KV System. An alternate power source for each unit's Class 1E auxiliary power system can be obtained from the other unit's 4.16 KV

Class 1E system or from its own unit auxiliary transformer. Connection to the latter requires removing the isolated phase bus disconnect links to the unit's main generator so that power can be obtained from the 230 KV system by back feeding through the unit's main step-up transformer. The onsite standby power source for each unit is from two redundant diesel generator sets. The diesel generator sets are not shared between the two units.

The team reviewed the available calculations which were intended to determine the anticipated starting and steady state running voltages for the various 4.16 KV and 480 V Class 1E motors. These calculations are E4C-011, Rev 2 - "Medium Voltage Regulation" and E4C-012, Rev 5 - "Low Voltage Regulation". The methodology used in the calculations was standard industry practice at the time Units 2 and 3 were designed and is considered acceptable by the team.

The 4.16 KV system calculation assumed a condition with power supplied from the preferred source (230 KV system) and with that source assumed to have infinite capacity. The 480 V system calculation assumed that the 4.16 KV system, its power source, had a 350 MVA fault capability, which is equal to the 4.16 KV switchgear's fault duty rating. Using these assumptions, the calculations yielded results indicating that motor terminal voltages would be above 75% of nominal during starting and above 90% of nominal during steady state running conditions. These values are consistent with the specified characteristics of the Class 1E motors. However these calculations did not consider the worst case conditions of limited power supply, i.e., a degraded 230 KV system or the system aligned to the standby power supply (EDG). Acceptable performance of Class 1E motors based on adequate starting and running voltages under the worst case conditions of power supply and loading has not been demonstrated by the design calculations.

SCE stated that they have obtained a computer program (ASDOP) which will be used to analyze the 4.16KV and 480 VAC Class 1E auxiliary electrical power systems. The program will address auxiliary electrical system loading, voltage regulation and fault levels.

This issue is considered unresolved pending implementation of the ASDOP computer program and the obtained results. This is identified as unresolved item 50-361/88-10-03.

b. Adequacy of DC Voltage Supplied to Class 1E DC Motor Operated Valves

As a result of INPO SER 80-83, which reported a potentially generic deficiency where voltage drops could be unacceptable when class 1E batteries are near "end-of-life", the licensee performed calculation DC 2642, Revision 0. Calculation DC-2642 was performed to verify the operability of the Class 1E 125 VDC loads when supplied from the batteries operating at "end-of-life" conditions during the 90 minute period following a design basis event. The results determined that less than the minimum specified starting voltage would be available for several Class 1E DC motor operated valves under these conditions. The evaluation criteria of the calculation states that the minimum starting voltage shall be 75% of nominal (125 VDC is the nameplate rating), as specified by the manufacturer. The motor operated valves of concern are in the auxiliary feedwater system of both Unit 2 and 3. The valves are control valve 3HV-4705, isolation valves 2HV-4715, 3HV-4715, 2HV-4730, 3HV-4730, and turbine stop valves 2HV-4716 and 3HV-4716. Voltage supplied to valve 2HV-4705 was demonstrated to be acceptable. The calculation demonstrated that the worst case condition involved motor operated valve 3HV-4730 with 109.59 volts at its starter terminals resulting in 43.91 volts across its motor armature. The licensee included in their calculation, test results performed on the worst case motor operated valve, 3HV-4730, which demonstrated that with no flow through and/or differential pressure across the valve, the valve will open and close with 93.45 volts applied to its starter. Based on the calculation method used, the licensee stated that the motor armature voltage in this test case would have been approximately 36 volts. MOVATS reports for this valve are referenced in calculation DC-2642, which show that flow and differential pressure have little impact on motor starting current magnitude. The calculation goes on to conclude that, "Since 3HV-4730 is the worst case, by logic all the other (motor operated) valves should start successfully". The team does not agree with this reasoning since the motor operated valve assemblies of concern are basically of three different categories or types, having different valve sizes, operator sizes and motor sizes. The extrapolation of the results of a test on one valve of only one type would not necessarily be valid for the other types.

This issue is considered open pending performance of a detailed safety evaluation that analyzes all variables of the different valve applications which do not see adequate armature voltage. This is identified as unresolved item 50-361/88-10-04.

c. CCW Surge Tank Relief Valve Sizing

Calculation M26.3, Revision 0, "CCW Surge Tank Pressure," does not include a postulated "failed open" nitrogen supply valve in its analysis to assure that Surge Tank relief valves 2PSV-6356 and 2PSV-6359 capacities are adequate. The calculation states the capacity of the relief valve as 201 scfm at 10% accumulation, while SCE identified the flow through the postulated "failed open" nitrogen supply valve at a significantly higher flow rate (approximately 600 scfm).

The team review of the ASME Section III, Code NV-1, Code Data Report for these relief valves identified the capacity as 226 scfm at 39 psig setpoint pressure.

DCP 970.0-J re-rated the relief valves at a 45.5 psig setpoint and a capacity (by Manufacturer's Test Certification) of 226 scfm at 60°F, with 10% accumulation. The re-rating was performed under the SCE ASME Section XI Program and was properly documented on an ASME NIS-2 form. However, no revision had been made to the calculation to assure that the relief valve can accommodate the higher flow (i.e. "failed-open" nitrogen supply valve) without exceeding the Surge Tank Design Pressure of 150 psig. In addition, the Safety Evaluation for DCP 970.0-J did not include an evaluation of such a postulated failure.

This issue is considered open pending an analysis that verifies re-rating of the relief valves could not over pressurize the CCW surge tank. This is identified as open item 50-361/88 10-05.

d. Single Intake Structure Supply of Three Saltwater Cooling Pumps

During cold shutdown of one unit, one SWC system train is required for shutdown operation, while two SWC trains are normally in operation for the other unit, if the other unit is in power operation. When these modes of operation occur and one unit intake is dewatered, three pumps, minimum, are required for the two units, with the SWC pumps supplied only from the operating unit's intake structure. The water supply required for three pumps at 17,000 gpm each is 51,000 gpm total.

FSAR Sections 9.2.1.2 and 9.2.5.2 identify a Design Basis intake flow of 34,000 gpm available for all modes of operation for each unit, including provisions to ensure this flow during a seismic event. No provision is made in the FSAR, nor has any analysis been performed to assure three pump operation (51,000 gpm) during or following a seismic event, when one unit is dewatered and all saltwater cooling is from one intake structure. For this

condition, FSAR 9.2.1.1F presently identified that "4% of normal circulating water flow is available" (i.e. 34,000 gpm).

Calculation M27.2 provides an analysis for the SWC pumps with a postulated seismic event causing collapse of the offshore intake and discharge conduits. This calculation identifies that 4% of normal circulating flow or 34,000 gpm will still be available (the calculation assumption was not revised when seismically designed intakes and emergency discharge were later provided).

The calculation determined a minimum submergence of 2.5 feet for the Saltwater Cooling pumps, with a drawdown of the intake such that the Circulating Water Pumps would lose all suction and, therefore, no longer cause further drawdown. However, the calculation does not analyze the three pump drawdown (51,000 gpm vs. 34,000 gpm), which would result in reduced submergence. Such reduction could cause all three pumps to become inoperable. In addition, if the normal operation of four pumps (two per Unit) is to be continued in this mode, the calculation and Design Basis must accommodate this quantity of flow.

This issue will remain unresolved pending an analysis that verifies the Seismic Category I intake can supply the three pumps at a flow of 51,000 gpm. A cursory review by the team indicates that a three pump supply should be capable of being maintained. This is identified as unresolved item 50-361/88-10-06.

e. Incomplete Analysis for Design Change Package 970.0-J

DCP 970.0-J enabled manual opening of the redundant CCW critical loop / non-critical loop isolation valves, by electrically bypassing an interlock such that during this transition time, both critical loops are cross connected. The DCP was implemented to allow the transfer of the non-critical loop from one critical loop to the other without interrupting CCW flow to the reactor coolant pumps.

Neither DCP 970.0-J, nor the safety evaluation in Proposed Facility Change (PFC) 2/3-83-242 addressed a Critical Crack Occurrence in the Loop A/Loop B supply crossties or return crossties, when both supply and/or both return valves were opened during transfer of non-critical loop operation (valves 2HV-6212, -6213, -6218, and -6219 are open at the same time). Prior to the DCP, the interlock prevented connection of the two critical loops by preventing opening of one set of supply/return valves while the other set was open.

The referenced PFC Safety Evaluation, Section III.B. states: "All analysis of failure modes, as described in FSAR Table 9.2-3 remain as stated and unchanged by this modification". However, Table 9.2-3 does not include a piping Critical Crack Occurrence when critical loops are cross-connected, nor does the safety evaluation refer to the proper FSAR section, Section 3.6 for the Critical Crack analysis.

Therefore, no assurance was provided that a postulated critical crack in the crosstie between critical loops has been enveloped by the CCW Critical Crack analysis previously performed or by design bases included in FSAR Section 3.6, as such a failure is common mode and does not allow exclusion by redundant trains (loops).

SCE stated that they had considered the consequences of a critical crack during the shifting of the CCW NCL between critical loops; however, it was not documented. SCE committed to provide an analysis for the postulated Critical Crack when transferring the CCW non-critical loop between critical loops. This issue will remain open pending an analysis that verifies system operability will be retained throughout the transfer of the non-critical loop between the critical loops. The team recognizes the the probability of occurrence of the above event is very small. This is identified as open item 50-361/88-10-07.

f. <u>Inadequate CCW/SWC System Preoperational Heat Rejection Test Due to Inadequate Instrumentation and Engineering Methodology</u>

Calculation M201 was performed in part to verify CCW system heat rejection design. The team identified the following concerns that appear to invalidate the CCW system heat rejection test that was performed as part of the original testing program following completion of construction activities:

- The basis for defining cumulative error (accuracy) of <u>each</u> test instrument was not identified. Therefore, instruments of insufficient accuracy were used during the test.

- The basis for determining <u>total</u> error contributed by all instruments was not defined. Consequently, the test may have been invalid without it being recognized.

- A test exception was incorrectly dispositioned which contributed to not recognizing that the test was invalid.

While it is clear to the team that the results of this test were insufficient to verify the heat rejection design of the CCW/SWC system, the team believes the conservatism of the design of the SWC system prevents this issue from being an immediate operational concern. At the exit interview SCE committed to examine the operability of the CCW system.

The following is a detailed technical discussion of the teams concerns with the CCW/SWC Preoperational Heat Rejection Test.

The basis for the instrument accuracies used in the calculation was not evident, and appeared to be limited to the manufacturer's typical specifications for the individual instruments. Consideration of process effects (such as temperature stratification and flow approach conditions) and other sources of uncertainty were not evident in the calculation. A history of inaccurate measurements also existed during and after the tests. Some specific examples of considerations that did not seem to have been addressed were as follows:

- The scale precision for the 50 - 300°F range CCW bimetallic thermometers would be expected to be +/- 2°F; the 25 - 125°F SWC bimetallic thermometers would likely have a +/- 1°F precision. Errors due to scale reading should therefore have been included.

- For the venturi flowmeters used in measuring shell side flow, it was not evident that the +/- 1% overall error cited in the calculation included considerations such as flow element errors (considering flow approach conditions); differential pressure transmitter errors; square root extractor errors; and indicator errors.

- For the ultrasonic flowmeters used in measuring tube side flow, it was not evident that the cited +/- 1% overall error included considerations such as uncertainties in velocity profile correction factors (characteristic of the instrument with its installed flow approach conditions); errors due to acoustic short circuit and signal/noise ratio; and errors due to changes in fluid composition or mechanical changes that might affect the length of the beam path. It was also determined that the original flowmeters were portable clamp-on types, and that permanently installed flowmeters were later installed, but subsequently replaced due to insufficient accuracy.

The team noted that the calculation was based on a sample calculation presented in Appendix 10G of SCE Procedure 2HA-299-02, CCW System Heat Rejection Test, Rev 0, and

uses the same initial values. To evaluate the impact on overall measurement of the acceptance value (CCW outlet temperature), successive independent calculations were made using values for tube side flow, CCW inlet temperature, CCW outlet temperature, SWC inlet temperature, and SWC outlet temperature. For each calculation, one of these values was decreased by its assumed instrument error (the other values remain the same) and an overall result is compared to the result in the example, and an isolated error contributed by the degraded value is determined. After this is done for all of the values, a square-root-of-the sum-of-the-square (SRSS) method is used to determine the overall error to be assumed for CCW outlet temperature in establishing the acceptance criterion (i.e., the requirement that measured/extrapolated CCW outlet temperature cannot be greater than 102°F).

The concern is that this method of determining overall error does not adequately bound the result of combining all instrument errors. Compliance to the acceptance criterion is sensitive to uncertainties in measured values of the four terminal temperatures of the heat exchanger; this is because the computation involves terminal temperature differentials and logarithmic computation of the log mean temperature difference (LMTD) before determining the extrapolated value (based on worst case design conditions) for judging acceptable CCW outlet temperature. Therefore, there is a potential for adding instrument errors as well as increasing the error effects logarithmically.

Essentially, small variations in the large or small terminal temperature differences have a much larger effect on LMTD, due to the logarithmic relationship of the values. This is particularly true for the terminal differential values in the region measured (for example, "large" terminal differential values around 10°F). The team's primary concern is that the simplified method cited above did not properly account for this interactive sensitivity to temperature measurement errors.

Other problems were noted with the referenced calculation. The method used in calculating large and small terminal temperature differences was not correct (incorrect terminal temperatures were subtracted). In addition, the team noted that the errors determined for service water inlet temperature and CCW inlet temperature were determined to be 0°F, which appears unrealistic, particularly since the acceptance test evaluation (Rev 2) attributed significant sources of error to the installation configuration of the saltwater temperature measurements; these errors were reported to occur due to inadequate immersion length of the temperature element in

the saltwater cooling system flow stream, and a DCP was developed to correct the problem.

Also, if values different from those in the Appendix 10G sample were used, different overall tolerances could result. This reinforces the concern that the method does not adequately bound all reasonable error contributions.

The team has an overall impression that the critical nature of these temperature measurements was overlooked in the design and application of the instruments; for example, it appears that standard range and accuracy bimetallic thermometers were provided, and the resulting data was accepted unless obvious anomalies were reported during the test.

For the reasons cited above, it appears to the team that SCE's acceptance of the CCW heat rejection test results relied too heavily on measurements that were insufficiently accurate to demonstrate heat rejection with adequate margin. We also found the select substitution of calculated values for measured values to be unsupportable, based on the information reviewed.

This item will remain open pending licensee action that confirms the SWC/CCW heat exchanger design capacity. This is identified as a followup inspection item (50-361/88-10-08).

(g) The licensee has developed a graph of SWC flow versus water temperature, to determine SWC system operability. The team questioned whether instrument errors were considered when the graph was calculated. The licensee determined that instrument error was not considered. This remains an unresolved item (50-361/88-10-09) until the licensee revises the curves and assesses past operations. Due to the large amount of conservatism in the actual SWC flow versus that flow required, it appears that this is not a direct safety problem.

(2) Outdated Calculations or Calculations with Unverified Assumptions

a. Emergency Diesel Generator Acceleration and Loading

The standby power source for Unit 2 and Unit 3 consists of two redundant tandem diesel engine driven generator sets. Each set is rated at 4700 KW, 0.8 pf, 60 hertz for continuous operation. The team reviewed calculation E4C-014, Rev 6 - "Generator Sizing, Diesel Generator", E4C-016, Rev 5 - "ESF Sequencing" and E4C-026, Rev 1 - "Motor Acceleration," which address the diesel generator unit sizing and loading sequence. The calculations are based on unverified assumptions. Calculation E4C-026 had

been performed to aid in the selection of the diesel generators and the results were to be used to evaluate compliance with USNRC Regulatory Guide 1.9 which had been committed to in the FSAR. Also the calculation results were intended to evaluate whether the engines would be sized based on the continuous load rating or for transient load acceptance and load acceleration. The conclusion to the calculation states, "When all motor data has been received from the vendors, an evaluation must be made by the selected D.G. vendor for frequency and voltage response to satisfy Regulatory Guide 1.9".

Calculation E4C-026 has not been updated based on data received from vendors for the various ESF drive motors nor have later calculations performed by the licensee, his design agent, or diesel generator supplier been identified and made available for review. However, the supplier's (Stewart & Stevenson Services, Inc.) Factory Test Report, Rev E, was made available for review. The acceptance criterion given in the test report shows that the purchased units have voltage and frequency transient characteristics that do not exceed the limits imposed by USNRC Regulatory Guide 1.9 when subjected to the loading sequence specified in the test procedure. The first load block in that test sequence is 1900 KW applied at 20 seconds after the engine start signal is initiated, followed by an additional 1910 KW at 30 seconds and finally, at 40 seconds after the initiation of the start signal, a load equivalent to an 800 HP motor is to be applied. The team feels that the test sequence does not represent the loading sequence in FSAR Table 8.3-1. The test sequence loading has not been shown to be conservative when considering the required loading and thus compliance with Regulatory Guide 1.9 has not been demonstrated.

This is a potential deviation of the FSAR commitment to Regulatory Guide 1.9 and is identified as unresolved item 50-361/88-10-10.

b. Incorrect 4160/480 VAC Transformer Tap Setting

When performing the voltage regulation calculations, one of the design outputs developed is the optimum primary winding connections to be used with each 480 V load center transformer. Calculation E4C-012 assumed incorrectly that winding taps on these transformers are on the secondary side rather than on the primary windings. Thus, the wrong connections using the "plus" 2 1/2% taps rather than the "minus" 2 1/2% taps were recommended for the Class 1E load center transformers. The team was informed by the licensee that the error had been recognized during construction and that the "minus" 2 1/2% taps had been used, as verified by revision 2 of calculation E4C-011,

dated 3/26/84. The intended purpose of the revision included the verification of transformer tap connections. The "minus" 2 1/2% tap connections were also indicated in start-up test procedure 2PE-401-01, Rev 0 - "Transformer Voltage Tap Verification". The team noted that it is recorded in "Start-up Test Exception Report", TER #2E-401-01/5, dated 10/16/81, (issued prior to the date of E4C-011, Rev 2) that the connections to the Class 1E load center transformers were found to be connected to the 0% taps. An evaluation of the "0%" tap connections had been requested by the "field" in the test exception report. Disposition by the design agent of the request has been indicated on the TER, reviewed by the team, to be "accept as is". The team views this anomaly as an isolated case of failure of the design verification effort by the licensee and/or his design agent and a possible communication failure within the design agent's organization.

The team has a concern that a setting of 0%, with minimum bus loading, that the bus may exceed maximum voltages for the motors (max allowed 506 V) and the battery charger. The test results for Buses B04 and B06 were 540V and 532V respectively. This is an unresolved item (50-361/88-10-11).

The Licensee committed to revise calculation E4C-012 to clarify the Class 1E load center tap transformer connection recommendation. However, as a consequence of not updating the calculation when the error was recognized, Revision 2 (3/26/84) of Calculation E4C-011 again indicated the taps were set a - 2 1/2% vice 0%. Only through good fortune did this error not impact the analysis. This, however, exemplifies the importance of maintaining design calculations current.

(3) Final Safety Analysis Report not Updated

a. Critical Crack Leakage Rate Not Updated in FSAR

Calculation M26.4 establishes the maximum Critical Crack for a 28 inch diameter CCW pipe as 898 gpm, while the FSAR Section 9.2.2.3H identifies this value as 42 gpm, with makeup sized for 100 gpm and thus sufficient.

FSAR Q&R 10.29 indicated 42 gpm Critical Crack with 200 gpm makeup. This was later amended in response to Q&R 10.48 to "approximately 700 gpm" for the 28 inch CCW main header.

The SSFI Team observed that neither the response to Q&R 10.48, dated 2/79, nor the Calculation M26.4, Rev 2, dated 6/83 were included in the FSAR Update for this applicable section, dated 2/86. The period of time between the Q&R

or the calculation and the FSAR update, exceeds any normal update period for proper identification of the critical crack leakage.

The licensee's cognizant engineer had identified the same FSAR deficiencies, in addition to many others, in the "State of the System" report.

In addition, neither the FSAR, the response to the Q&R, nor the calculation clearly identify whether the critical loop would remain operable. With the critical crack flow of 898 gpm, alarmed at low level, no operator action for 30 minutes (Q&R 10.48), and no makeup water assumed available (nonseismic source: FSAR 3.6.1.3D, FSAR 3.6B.2.B, and Q&R 10.49), the Surge Tank inventory will drawdown to the low-low level and result in surge tank isolation closure, water-solid condition in the CCW System critical loop, and potential loss of one critical loop. The team noted that for a critical crack occurrence, no other failures are required to be assumed, unless the crack itself causes additional failures. Therefore, one critical loop of CCW would apparently remain operable.

The team concluded that the FSAR Safety Evaluation is presently inaccurate and untimely in its update. In addition, design basis calculations and other documents do not presently include identification of loss of a critical loop when a critical crack is postulated. No enforcement action is proposed because the licensee identified the same concerns prior to the inspection. The actions taken by the licensee to restore accuracy to the FSAR is an open item (50-361/88-10-12).

b. Routine Operation of both CCW Critical Loops

During the entire commercial operation of Units 2/3, the CCW system has been operated in a manner contrary to the FSAR. Paragraph 10 of this report discusses this issue in detail.

(4) Design Change Package Adequacy

a. Potential Common Mode Failure of Both CCW Surge Tank Isolation Valves during a Seismic Event

Because the control circuits for the surge tank isolation valves are not safety class or seismically qualified, the surge tank isolation valve should be postulated to fail during a seismic event causing spurious simultaneous closure of the valves for each surge tank. If this were to happen during a seismically induced break in the non-critical loop (NCL) piping, each surge tank would be isolated from its respective loop and each independent loop would be starved of makeup water. In addition, the

automatic NCL isolation on low level could be disabled, since the surge tanks would be isolated early by the seismically induced spurious actuation. Thus, the CCW system is vulnerable to a single failure during a seismic event.

This is an apparent violation of 10CFR50 Appendix A, General Design Criteria 2 and 44, which require that systems and components important to safety be designed to withstand earthquakes and that the cooling water safety function not be precluded by a single failure (50-361/88-10-13).

The licensee documented this item on NCR 3-2034 dated June 15, 1988. As an interim action, the licensee has removed the thermal overload devices from the control circuit.

b. Unreliable CCW Surge Tank Level Indication During High Transient Level

The CCW Surge Tank is a closed pressurized tank using a differential pressure transmitter to measure level hydrostatically. For this type of application, one connection of the differential pressure (DP) cell is to the bottom of the tank, and the other equalizing connection is near the top of the tank via a reference leg. The reference leg in the SONGS installation uses a dry reference leg.

The team noted that the dry leg installation could result in erroneous and potentially misleading indication to the operators for events leading to large transient surges in level that could flood and fill the equalizing reference leg. Subsequent level decreases will also result in confusing indication, since the reference leg will remain full during the level decrease in the tank. This could lead to incorrect operator actions and responses. Both the analog indication and the hi/lo alarms could be affected.

Because the licensee does not have controlled instrument loop diagrams that present basic and functional information about the instrument channels in a complete and integrated fashion, it was not possible for the licensee or the team to determine from design drawings whether or not the reference leg was wet or dry. When the team inquired about the type of reference leg, the licensee was only able to provide an answer by consulting the instrument calibration procedures at the site.

This issue is open pending licensee review of the issue. This is identified as open item 50-361/88-10-14.

c. Adequacy of Design Change to Incorporate Jogging Control for RCP Seal Cooling Water Return Isolation Valve (DCP 793.01J)

Each reactor coolant pump (RCP) is cooled by CCW cooling water. In the original design, a high temperature interlock was provided such that the return line would close on high temperature. Because of the risk of inadvertent isolation of the seal cooling water and consequential damage to the RCP seals, the licensee decided to eliminate the automatic isolation feature, retain the high temperature alarm, and rely on operator isolation of the seal cooling return lines in response to the alarm. In addition, jogging control was added to these valves with the intent of controlling CCW flow in such a way as to reduce the potential for thermal shock. Jogging control eliminated the original "seal-in" circuit feature that had required the valve to continue operating through its full stroke before stopping. With the new design, the operator would have the ability to "Jog" the valve to intermediate positions, and presumably obtain a throttling capability.

The licensee was unable to retrieve documentation from their files that the Limitorque valve actuators were designed and rated for the jogging duty required for this new service condition. The team was specifically concerned that the different thermal duty cycle that may now be imposed on the motor, and the mid-travel operation of any "hammer blow" features provided with the actuator, may not have been considered when the change was made; no such considerations are evident in the DCP. During the inspection, the licensee requested and received a letter from the actuator vendor stating that the MOVs covered by the purchase order for these valves were acceptable for jogging duty provided that no more than 20 starts were made for the 15 minute duty rating. THe team requested that the licensee confirm that operating procedures were in place to assure that operators were aware of this restriction. The licensee retrieved Procedure S0123-0-23.1, which includes a precaution that no more than 5 starts be made in one minute for any MOV, and that if 5 starts are made in a minute, a fifteen minute cooling period must be allowed before attempting another start.

Further clarification by the vendor was requested by the team. If the clarification is favorable, the team can conclude that the precautions in the licensee's procedure were sufficiently conservative regarding thermal cycling. An additional concern exists on the vulnerability of the control circuit and motor to mid-travel torque switch bounce and consequent intermittent/erratic operation of the motor that might result in excessive thermal cycling. The seal-in circuit provided in the original design

precluded this concern. THe licensee committed to contact the vendor and verify the acceptability of the present valve operation.

This issue is considered unresolved. This is identified as unresolved item 50-361/88-10-15.

d. Adequacy of Seismic Qualification of Devices Added to Electrical/Instrumentation Panels

Addition of control components to a safety-related panel must be evaluated to verify that the original seismic design bases for the panel and the additional components are not unduly compromised by the modification.

Therefore, engineering controls must be established and maintained for any modifications to safety related panels involving the location of additional components. A seismic evaluation must be provided for all additions to safety related panels, and the location and installation of the components must be under engineering controls.

In our review of DCP's, the team noted inconsistent evidence of adequate seismic evaluations in several cases where equipment was being added to a panel.

DCP 793.3J added a CIAS close signal to the CCW non-critical loop (NCL) supply and return valves located inside containment. The original design basis had not included credit for automatic containment isolation using these valves. Providing this automatic feature required addition of a safety related control relay to each valve control circuit, and verification that the entire circuit met safety criteria for containment isolation valves.

In review of this DCP and in discussions with the licensee, the team noted that SCE was unable to retrieve documentation demonstrating that a specific seismic evaluation for the addition of the control relay had been performed, and that engineering controls had been in effect regarding the specific location of this relay. Consequently, the seismic qualification of this circuit was not evident from the DCP. During the inspection, the licensee retrieved design information showing that the location of the relay and its qualification appeared to be bounded by previous qualification of similar relays and the panel.

In the review of DCP 2-6345.0TJ, which added an indicator to the main control board in an area at the top of the panel where no other instrumentation was evident, the DCP Plant Hazards Requirements form indicated, "No special requirements" for seismic effects. Again, adequate seismic evaluation was not evident. Subsequent discussion

with the licensee indicates that there is a reasonable assurance that the installation is seismically qualified for reasons similar to those cited in the previous example. However, the team is concerned that in both cases there was no evidence of a thorough seismic evaluation. Although the design in these cases did not appear deficient, better engineering control is needed on the location and seismic evaluation of these types of modifications.

Regarding the licensee's program for assuring that adequate seismic evaluations are performed for these types of modifications, the licensee explained that the procedures in effect at the time of the DCP were limited to a simple check-off of the item, "Tornado Missile Effects," on the DCP Plant Hazards Requirements form; according to the licensee, this item was understood to include a seismic evaluation. No justification of the check-off or reference to supporting analysis was provided. The team concluded that this practice is inadequate to assure maintenance of seismic qualification.

Regarding current practice, the licensee explained that the DCP Plant Hazards Requirements form has been revised to include a separate check-off for a seismic evaluation. However, the team noted that no justification is required if "no" is checked on the list. The team concluded that this is also a poor practice. The licensee indicated that a recent licensee QA Audit had reached a similar conclusion, and issued a CAR requesting that further documentation be provided for seismic and other engineering evaluations. The team noted that this CAR committed to providing a summary of the rationale for "yes/no" checkoffs on the Hazards Requirements Form. The team agrees that this is an acceptable corrective action.

e. <u>Discrepancy Between the Instrument Isometric and Calibration Documents for CCW Surge Tank Level Instrumentation</u>

Level setting diagrams establish the calibration and installation requirements for the CCW surge tank level transmitters (alarm and indication) and the level switches (control of outlet isolation and NCL isolation valves). These diagrams provide a datum to which all level instruments for the surge tank can be referenced with respect to both tank elevation and plant elevation. The level setting diagrams are controlled design inputs to the instrument/tubing isometric drawings with the latter providing the installation details.

In examining the level setting diagram and the instrument/tubing isometrics, a discrepancy of 1" between the drawings was noted. The licensee investigated the

discrepancy and subsequently determined that the discrepancy resulted from failing to account for the correct grout thickness upon which the pad is supported on elevation 8' -0". As a result, the LSD 56364-4 had an incorrect 1" offset.

SCE also verified conformance of the isometrics to the other physical drawings, and confirmed that the level instruments would function correctly since they were correctly referenced to the tank in all cases.

This appeared to be an isolated problem with minor specific consequences in this case. Since there was no effect on the level measurement, no functional safety significance is directly evident. However, incorrect information on the level setting diagram could result in a more significant error on a future modification, if the future instruments were improperly located based on the incorrect plant elevation provided on the diagram. As noted during the licensee's revaluation of the CCW system response to various accident scenarios, the different of only 1" of water in the CCW surge tanks could have a significant impact due to the apparent sensitivity of the system to leakage.

At a pre-exit meeting, the licensee committed to correct and revise the level setting diagrams to show the correct elevations.

7. Operations Activities

The inspector assessed the implementation of selected design parameters and design parameter changes in the operations area as well as normal operation activities. The specific operations attributes examined included an examination of operator and auxiliary operator training for the selected systems and design changes to the systems, valve line-up controls, knowledge of operating procedures, adequacy of the normal and abnormal procedures and annunciator response guide lines, auxiliary operator round sheet adequacy, and adequacy of observing and reporting plant deficiencies by operations personnel.

Findings in the areas examined are described below:

(1) Normal Operation of the Component Cooling Water System Contrary to the FSAR Description.

The Component Cooling Water (CCW) system consists of two independent, full capacity, critical cooling loops, and one non-critical cooling loop. Cooling water for the non-critical loop can be supplied from either Critical loop. A process radiation monitor is installed to detect leakage of radioactive fluid into the CCW system. The radiation monitor receives flow from the supply header of the non-critical loop, monitors the radiation level of the cooling water, and returns the flow to the return header of the

non-critical loop. During the inspection, it was determined that the normal mode of operation of this system by the licensee involves the circulation of cooling water through both critical loops, with the non-critical loop aligned to one critical loop of the system and the letdown heat exchanger aligned to the other critical loop. The licensee had, prior to this inspection, identified the fact that in the description of the system's normal mode of operation contained in the facility FSAR, one critical loop was identified as being maintained in a wet layup status while the other critical loop was in operation supplying all non-critical components including the letdown heat exchanger. The significance of this discrepancy was however, apparently not recognized by the licensee. The current mode of operation with both critical loops operating results in a potential leakage path of radioactive fluid from the letdown heat exchanger into the CCW system not being continuously monitored. This condition exists because the system has been operated in a manner not originally intended by the system design and is considered a deviation from commitments contained in the licensee's FSAR. This issue is discussed in detail in paragraph 9 of this report.

(2) CCW Surge Tank Pressure Not Being Monitored or Maintained.

In the review of calculations supporting operation of the CCW system, it was found that in some cases it had been assumed that the CCW surge tank pressure would be maintained at a pressure of 33 plus or minus 2 psig. During initial system walkdowns by the inspection team, the surge tanks in Unit 2 were found to be at a indicated pressure of 28 psig and 30 psig. It was also later determined that the tank pressure is neither alarmed in the control room nor otherwise monitored by operations personnel during operator tours. Although the licensee's site engineering group determined that the identified condition did not affect operability of the system, the lack of a clearly specified acceptance criteria for tank pressure in the normal operating procedure or on auxiliary operator round sheets is considered a weakness in assuring operability of the CCW system.

(3) Abnormal Operating Procedure Deficiencies.

In the course of a review of abnormal operating procedures, the inspector found that the abnormal operating procedure for the CCW system did not contain instructions for connecting the emergency makeup water supply, which entails the running of hoses from the seismic fire main or fire truck to the CCW surge tanks, nor when such action should be taken. Instead, detailed instructions for performing these connections were found in an Appendix to the system's normal operating procedure. Additionally, the normal operating procedure was found to contain a precaution that the seismic fire truck should be flushed first if it previously was filled with saltwater. A recent licensee reanalysis of several accident scenerios determined, in some cases, the need to provide emergency makeup capability to the CCW surge tanks in less than one hour. It appeared that the licensee's procedures were written considering that 7 days were available to establish the emergency

makeup supply. In light of the licensee's analysis, it appears the licensee's current abnormal operating procedures for the CCW system are outdated. The licensee committed to issuing an update to the plant's abnormal operating procedures in response to this concern.

(4) Results of Detailed System Walkdowns - Need For Better Labeling of System Vents, Drains and Instrument Root Valves.

The inspector performed a hand over hand walkdown of portions of both the Saltwater Cooling and CCW systems, using as references both the system P&ID's and the piping isometrics. The inspector found during the course of these walkdowns a number of instances in which system vent and drain valves and some instrument root valves were poorly identified. Although none were found to lack identification tags as such, in some instances the markings on the tags had deteriorated to the extent they were illegible or located in such a manner as to make reading them very difficult. Instances were found where plant personnel had found it either necessary or more convenient to identify some valves by marking the identification numbers on the piping systems themselves with marking pens. No discrepancies between piping isometerics and the actual installed configuration of the piping systems were found. Some discrepancies however, were found with the system P&ID's. In one instance, a drain valve in the CCW system was not identified at all on the P&ID's. Since the presence of vents and drains provide likely leakage paths from these systems and the CCW system has been found particularly sensitive to leakage by the licensee's own analysis, adequate labeling of these valves appears all that much more important. In addition, the misoperation of certain instrument root valves could render important monitoring instrumentation, interlocks, or permissives inoperable. Thus, it would appear prudent for the licensee to further assess the need for improved labeling of these particular kinds of valves.

(5) Operator Training.

The inspector reviewed current training given to operators regarding operations of the systems reviewed by the team. This included both initial qualification and requalification training, and the methodology used to update operators on current system operating characteristics or design changes. The inspector found no problems with the initial training given to operators. The inspector found that most updates or design changes were passed on to the operations staff through shift briefings or required on-shift reading materials. The licensee has also included in their requalification training, updated plant system classroom overviews for a selected number of plant systems during each requalification cycle. The inspector considered this a positive aspect of the requalification program. In particular, during the past requalification cycle, an updated classroom lecture on the Saltwater Cooling system was conducted. The inspector noted however, that an overview of the CCW system had not been included in any of the past three requalification cycles nor since the licensee began to include specific system reviews in their program. The inspector recommended

that the licensee consider including the CCW system as one of the selected systems to be reviewed during the next requalification cycle, particularly in view of the recent reanalysis of the system response under various accident conditions.

8. Control of Drawings

The team identified several weaknesses relating to drawing control during evaluation of the administrative controls. These concerned the licensee's general process of revising drawings and the use and control of operational schematics. Additionally, the team was informed that SCE had decided to void the logic diagrams. These diagrams are use to conceptualize the control circuits and form the basis for detailed design output documents such as elementary drawings and wiring diagrams. The licensee indicated that the latter drawings were being maintained.

A. Operational Controlled Drawing Stick File Drawings

The station identifies those drawings which are needed to operate the plant and designate these as "operational controlled drawing stick file drawings." These typically include piping and instrumentation drawings (P&IDs), electrical one lines, electric elementaries, etc.

Upon turnover of a design change, interim drawing change notices are placed in front of the applicable drawing by the station CDM group. These drawings are stamped as "interim as-built" drawings. E&C Department QA procedure 24-8-7 defines the time period which is allowed for revising these drawings following receipt of a notification that the design change package (DCP) was turned over. This notification nominally takes several days following system turnover to operations.

Thirty calendar days are allowed to revise all "operations" drawings, except P&IDs, which must be revised in 14 days. If the change requires a new drawing, these time periods are changed to 60 and 28 days, respectively. Other drawings are required to be revised once ten DCNs have accumulated. Selected drawings, such as equipment lists, setpoint and annunciator lists, require revision in 90 days.

The team was concerned with SCE's method of control for these drawings. Although controlled drawings are provided to the operators, the need to compile the interim changes mentally onto the drawings may create a problem if the step is either missed or improperly completed. Accepted industry practice is to provide composite operational drawings at the time of turnover. These can be either temporarily drawn, or marked up drawings. For major changes, common practice is to issue revised drawings which are pre-drafted for the applicable change. This is in contrast to the licensee's practice, which allows twice the time for complex changes compared with that allowed for simple redrafting.

B. Operational Schematics

Operational Schematics have been prepared for operator use and are available in the control room. These drawings typically show the system on a single sheet, and identify operational information such as valve numbers, setpoints, and control logics. These drawings are not controlled in that facility changes are not incorporated on a real time basis, nor were the drawings prepared as design disclosure documents.

The stick files containing those drawings contain a disclaimer in the front of the file stating:

> Operational Schematics are Operational Aids and are for Reference only
>
> For Current Configuration See Design Document on Control Room Drawing Stick File

The team's cursory review of Operational Schematic 50.127B, Component Cooling Water System No. 1203, identified several errors. Control Logics 2 and 9 both refer to safety injection actuation signals which operate selected valves. This signal has been changed to a containment isolation actuation signal. Control Logic 2 does not reflect changes for DCP 970-0-J, which allowed non critical loop isolation valves to be open simultaneously while switching the non critical loop from one critical loop to the other. Other changes from this DCP, such as revised surge tank float and relief pressures, were reflected on the drawings.

The team considered that uncontrolled drawings should not be used to operate the plant. The team was informed that the drawings are used for reference only, and that plant configuration is manipulated using only controlled drawings. The team remained concerned regarding potential operator use of schematics to operate the plant.

The team subsequently was informed that widespread operator use of the schematics, in deference to controlled P&IDs, was observed during the recently completed limited scope SSFI. Apparently in response to this finding, an update of schematics is planned in the near future. The licensee stated that the new revision of these drawings will contain a printed message stating "For Information Only, Not a Design Disclosure Document, Shall Not be Used to Operate the Plant". The licensee is also considering semi-annual updates for the schematics in the future.

The team remained somewhat concerned with this approach. Accepted industry practice is to not allow uncontrolled drawings such as these schematics to be present in the control room. More fundamentally, the operational schematic drawings appear to fulfill an operator need for convenient, composite information regarding the various systems, as opposed to the P&IDs which contain extraneous (to operators) information, lack operational input, and typically span several sheets. The team considers that the licensee should seriously consider upgrading the schematics and their control to allow their use for plant operation.

9. Maintenance Activities

The inspection team assessed the licensees' corrective, preventative, and repetitive maintenance activities for the Component Cooling Water (CCW) and Salt Water Cooling (SWC) systems. This assessment included a review of the maintenance program, technical manuals, maintenance orders and maintenance requirements. Work in progress was also observed during the inspection.

The licensees' maintenance program is described in the San Onofre maintenance policy guideline, SO123-S-6, "Preventative Maintenance Program Objectives and Responsibilities." The licensee's Preventative Maintenance (PM) program includes predictive, periodic, and planned maintenance activities. The predictive maintenance activities that are used to detect abnormalities in equipment performance include In-Service Test (IST), vibration and performance monitoring, lube oil analysis to detect wear, and Motor Operated Valve (MOV) testing and sampling. Some additions to the PM program were being considered at the time of the inspection and included infrared surveys (to detect hot spots on electrical equipment), and equipment that can be used to determine wire degradation. The generation of work plans and maintenance orders (MOs) is governed by licensee procedure SO123-I-1.9, "Repetitive Maintenance Implementation and Scheduling," and SO123-I-1.7, "Maintenance Order Preparation, Use, and Scheduling.

The inspector reviewed the Nuclear Plant Reliability Data System (NPRDS) data base for SONGS 2 and 3 for the CCW and SWC cooling systems to determine historical problems/trends with the CCW and SWC equipment. This data base documented problems with salt water pump bowl and bearing degradation and with fouling of the CCW heat exchanger on the saltwater side. The inspector also reviewed the Piping and Instrument Diagrams (P&IDs) to identify several components with which to perform a PM program evaluation. These components included the SWC pump, cyclone separators, several pressure regulators, and certain valves. The inspector determined that the licensee was taking action to address the pump bowl erosion and that a modification had been made to the SWC system to address the fouling of the CCW heat exchanger. The other components identified were on the PM program and no problems were noted with these components. In discussing the PM program with the licensee, no concerns were noted by the inspector with the program.

The licensee's repetitive maintenance program incorporates periodic calibrations of instruments. The inspector asked for the instrument calibration records for several flow and temperature elements of the CCW and SWC systems. The instruments were found to be in calibration or within the grace period (25% of the time interval for recalibration) allowed for recalibration. The instruments that were in the 25% time extension were calibrated during the inspection. Several of the instruments have recently had their recalibration interval shortened to improve plant reliability.

The inspector also observed maintenance work in progress during the inspection. This work included cleaning and inspection of the unit 3 CCW/SWC heat exchanger and the recalibration of level switches for the

CCW surge tank. The inspector noted that the work was done with the appropriate procedures and maintenance orders and had the appropriate signoffs. For the work on the heat exchanger, Quality Control (QC) checks were observed being performed, and foreign material exclusion and material control was maintained. The inspector also observed work on the CCW level switches which isolate the surge tank on low level (2 feet). The calibration of the level switches was performed with a calibrated test instrument and in accordance with the procedure. No problems were identified with the observation of maintenance activities or the procedures used.

10. Radiological Control

The radiological and chemical aspects of the operation, maintenance and surveillance of the CCW and SWC systems were reviewed. System walkdowns were performed outside containment at both units 2 and 3 and inside containment at unit 3. Numerous interviews were held with the cognizant system engineers, effluent engineers, I & C technicians, operators and other licensee personnel to determine the status of these systems.

The Unit 2 and 3 CCW systems each have an inline process monitor, 2 & 3RT-7819, which is calibrated after repair, adjustment or component replacement, and at least every 18 months. Channel functional tests are required after calibration and at least every 92 days. Select records of channel calibrations in accordance with procedure SO23-II-4.56 and select channel functional tests in accordance with procedure SO23-II-4.57 were reviewed for the period 1986 to present and appeared complete. The monitors were also sighted during facility tours and their control room readouts observed. It was noted that the monitors provide no positive indication of flow, either at the monitor locations or at the control room readouts. This was brought to the attention of the cognizant system engineer and it was noted that this particular instrument model has previously experienced flow blockage problems in other systems at SONGS, which have rendered the monitors inoperable for extended periods of time.

Chemical and radiological sampling of the CCW systems is performed in accordance with chemistry procedure SO123-III-1.1.23, "Units 2/3 Chemical Control of Primary Plant and Related Systems." This procedure as well as sampling records from March 1987 and 1988 were reviewed. The procedure specifies broad normal ranges for pH, conductivity, nitrite, degassed gross beta activity and a recently added parameter for microbiological organisms. Most out of range results require no specific corrective action and the weekly degassed activity is performed on a 2 milliliter evaporated sample, which has a significantly higher "lower limit of detectability" (LLD) than many liquid radioactivity sampling methods currently in use at SONGS. In the reviewed records, the degassed activity results indicated "less-than" values in the E-5 microcurie/cc range. For comparison, one liter liquid gamma spectral analyses typically achieve LLDs in the E-7 to E-8 microcurie/cc range.

The SWC systems require radiological sampling and analyses in accordance with Technical Specification (TS) Table 4.11-1 and chemical sampling in accordance with the Unit 2 and 3 Environmental Protection Plan, as required by the SONGS National Pollutant Discharge Elimination System

(NPDES) permits with the State of California, and as specified in procedure SO123-III-2.2.23, "Units 2/3 Support Systems Chemistry Control and Sampling Frequencies." Select records of the TS radiological sampling and analyses were reviewed from September 1987 through March 1988. The records were complete and the TS indicated LLDs were achieved for all analyses. No sample provided a positive indication of plant related activity for the period reviewed and the cognizant effluent engineer stated that he could recall no instance where activity has ever been detected in the samples, even though plant radiological discharges are routinely made through this system. The SWC system provides approximately 50 million gallons per day of dilution to any activity released through this pathway.

Select records of the sampling and analyses of marine debris in accordance with procedure SO123-VII-8.2.11, "Release of Potentially Contaminated Liquids, Sludges, Slurries, and Sands to Unrestricted Areas," collected from the SWC system associated "fish baskets" were also reviewed for the period of January 1988 to date. The analyses occassionally indicated plant related activity. In these cases the material was held for decay at the instruction of health physics engineering, resampled to verify the absence of activity and then released.

Sodium Hypochlorite is added to the SWC system by an automatic system to reduce slime fouling in the heat exchangers. Procedure SO23-4-1, "Sodium Hypochlorinator Operation," and SD-SO23-340, "Chlorinator Design Procedure," were reviewed, system operation was discussed with the cognizant engineer, and system components were sighted.

Procedures SO23-2-8, "Saltwater Cooling System Operation," and SO23-5.1.1, "Heat Treating the Circulating Water System," were also reviewed. It was noted that these procedures allow an alternate discharge path for the SWC system effluent through the seawall across the beach adjacent to the normal outfall discharge path. The status of the SONGS application to the State of California for the use of this path was reviewed. It was found that an application had been made in November 1987 and had not yet been finally accepted. The alternate discharge path has, however, been used on three occasions this year after appropriate verbal and followup written notification to the State. The cognizant system engineer stated that this alternate discharge path is normally used when heat treating the Circulating Water system or when maintenance or inspection is being performed on the outfall structures, and that it would be used in emergency situations should the outfall become unavailable. SO23-5.1.1 was noted to prohibit radiological discharges and hypochlorite addition during heat treatment, in accordance with the NPDES application, but no such prohibition was contained in SO23-2-8 to restrict such activities during maintenance and inspection activities when the alternate discharge path is in use. The cognizant engineer acknowledged this observation and stated that action would be taken to revise their procedures to add an appropriate prohibition to avoid possible noncompliance with the NPDES permit application.

The program for ALARA reviews of Design Change Packages (DCPs) was examined and the ALARA review of DCP 768.5N, "Relocation of Radwaste

Discharge Line," associated with the SWC system, was examined. Work on the SWC system and the CCW heat exchangers was observed during the inspection and involved maintenance and engineering personnel were interviewed.

Procedures SO23-2-17, "Component Cooling Water Pump and System Operations," and SD-SO23-400, "Component Cooling Water System" (system description), were reviewed as well as system P&I diagrams. It was noted that the process monitors, 2 & 3RT-7819, are installed across the noncritical loop supply and return headers, that the monitors operate due to the differential pressure between the two and that they sample the noncritical loop supply flow. The system is normally operated as specified in paragraph 6.1 of SO23-2-17, which states:

> "The CCW system should be lined up with both trains running, the third of a kind pump lined up, in standby to the loop supplying the noncritical loop, and the letdown heat exchanger being supplied by the opposite critical loop...."

The noncritical loop is isolated upon CIAS actuation thus removing the monitor from the system. SD-SO23-400 states in part:

> "1.2 The Component Cooling Water System has the following additional function:
>
> 1.2.1 To provide a radioactively monitored intermediate barrier between the reactor auxiliary fluids and the Saltwater Cooling System."

The updated SONGS 2&3 FSAR, in section 9.2.2, "Component Cooling Water System," paragraph 9.2.2.1, "Design Bases," states in part:

> "N. The component cooling water system is designed to provide a radiation monitored intermediate barrier between the reactor auxiliary systems fluid and the saltwater cooling system during nonaccident conditions."

Paragraph 9.2.2.2.1, "General Description," states in part:

> "The system is continuously monitored for radioactivity and all components can be isolated."
>
> and
>
> "Radioactivity levels in the noncritical loop <u>return</u> header are continuously monitored in the control room to indicate any leakage of radioactive fluid into the component cooling water system."

Paragraph 9.2.2.2.3.2, "Normal Operation," states in part:

> "During normal system operation, one redundant loop consisting of one component cooling water pump, one component cooling water heat exchanger, and one saltwater pump is in service supplying cooling

water to the various components in the noncritical loop and to critical loop A. Critical loop B is on wet standby...."

The FSAR was reviewed in accordance with NUREG-0800, "Standard Review Plan" (SRP), which in section 9.2.1, "Station Service Water System," part III, "Review Procedures," paragraph 3.d. states in part:

"Provisions are made in the system to detect and control leakage of radioactive contamination into and out of the system. It will be acceptable if the system P&IDs show radiation monitors located on the system discharge and at components susceptible to leakage, and these components can be isolated by one automatic and one manual valve in series."

The inspection revealed that the CCW process monitors were not installed in accordance with the statements of the FSAR, in that they sample the supply flow of the system rather than the return flow, and that this is not consistent with the description in the SRP. Also, the normal operation of the CCW system in accordance with SO23-2-17, with the noncritical loop supplied from one loop and the letdown heat exchanger supplied from the other, is contrary to the mode of operation specified in the UFSAR. This is an apparent deviation (50-361/88-10-16).

Additionally, this mode of operation could provide an unmonitored leakage path from the letdown heat exchange through a failure in the CCW system to the SWC system.

This matter was discussed with representatives of the SCE corporate engineering office during the course of the inspection. Based on this discussion, the engineers did not appear to understand the operation of the system in that they believed that the monitors drew their sampling flow from the return headers of the noncritical loop and were unaware that dual loop operation removed the letdown heat exchanger from the monitored portion of the system. They also believed that any unmonitored release to the SWC system would be identified by routine sampling of this system, thus not recognizing the lack of sensitivity of both the routine CCW and SWC system sampling procedures as noted above. At the end of the discussion, the corporate engineering representatives acknowledged that it appeared that the CCW system was being operated outside its design bases and that corrective action to either implement single loop operation or install an additional process monitor would be considered.

A review of licensee records indicated that the initial edition of SO23-2-17, dated May 23, 1978, specified single loop operation in accordance with the requirements of the FSAR. However, revision 2 of SO23-2-17, dated February 11, 1982, clearly specifies that the normal mode of operation is with both trains running as indicated above. It was stated by representatives of the Operations department that the CCW system has essentially always operated with both loops running, in that the implementation of revision 2 was very shortly before the initial startup of Unit 2. The Procedures Review Committee, meeting number 82-014 of February 11, 1982, found that this document did not involve an unreviewed safety question as defined in 10 CFR 50.59. This review appeared to be inadequate in that it failed to recognize the as-built

discrepancy between the installed monitor location and the FSAR description and the need to update the FSAR due to the change in normal system operating configuration as required by 10 CFR 50.71 (e).

The licensee has documented this issue on NCR G-0867 dated June 15, 1988. Interim action has been taken to align the letdown heat exchanger to the critical loop supplying the non-critical loop.

In this area, one deviation was identified.

11. Testing/Surveillance Activities

A. The inspector observed the performance of the quarterly component cooling water (CCW) pump and check valve in-service test (IST) conducted under the licensee's procedure S023-V-3.4.2. The inspector found that the procedure was performed thoroughly by the system cognizant engineer and included both performance testing of the equipment and vibration monitoring for trending. The testing involved operation of the A train CCW pump and verification that its discharge check valve was functioning open and that the parallel CCW pump discharge check valve was functioning closed. The testing was alternately repeated for the parallel CCW pump aligned to the A train.

During the testing, the inspector observed that proper check valve position was verified for the various pump operating configurations, but that active repositioning or exercising of the check valve was not an explicit requirement of the test procedure. The inspector discussed his observation with licensee representatives responsible for IST.

The licensee acknowledged that the check valves were only implicitly exercised per their procedure when performed in combination with swing pump testing. The licensee stated that they would review and change their IST procedures to clarify the intent of the surveillance in accordance with ASME Section XI check valve testing requirements.

B. The inspector reviewed records of the post-modification/ pre-operational testing performed per licensee procedure 2/3 PE-232-02 following modification of the CCW heat exchangers to allow backflush operation under DCP 6204.2SM.

(1) The inspector found that the testing which had been performed in 1985 consisted solely of flow testing to verify acceptable specified SWC flow rates. The modification involved reversing the SWC flow direction through the tube side of the heat exchanger to dislodge debris from the tubes.

However, when in backflush mode, SWC water flows in the same direction inside the heat exchanger tubes as the CCW flow outside the tubes. This reduces the overall heat transfer coefficient of the heat exchanger requiring higher calculated SWC flow rates in the back flush mode than in the normal SWC

flow direction to transfer an equivalent amount of heat. However, no testing to verify the adequacy of the resulting heat transfer capacity was performed as part of the post modification testing. As discussed in paragraph 6.B.(1)f. of this report, initial system testing did attempt to verify heat transfer capacity, indicating that verification of heat transfer capacity in the backflush mode following the modification would have been appropriate.

(2) The inspector observed that the flow rates required per the acceptance criteria of the test were satisfied based on a calculated flow derived from the measured SWC pump differential pressure (dp) and the original SWC pump head curve. Redundant direct measuring flow instrumentation (controlitron ultrasonic flow detectors) were installed at the time but were considered to have unreliable accuracy.

The inspector observed that based on the calculated flow, the test acceptance criteria was satisfied. However, based on the Controlitron indicated flow, the test would not have met the acceptance criteria. However, no NCR had been initiated based on this condition due to the opinion at the time that the controlitron data was less accurate than the data calculated from the pump head curve.

The inspector noted that the licensee later determined the SWC pump curves to be inaccurate due to pump degradation, therefore, indicating that the controlitron data may have been correct and the test acceptance criteria not met.

Based on his review of the preoperational test results, the inspector found that the limited flow testing which was performed appeared to be adequate due to the excess design margin available in the original design. However, the inspector noted that the pre-operational performance testing lacked essential baseline data to verify the heat transfer capacity during backflush mode of operation.

C. In discussing the IST program, the inspector noted that the butterfly valves which isolate the CCW non-critical loop from the critical loops are classified as category B valves by the licensee's ASME Section XI IST program. Category B valves are valves in which seat leakage in the closed position is inconsequential for fulfillment of their design function. However, when the valves were originally procured, the system design required the valves to be "zero leakage" valves, and the vendor was required to provide test documentation to this affect. During initial plant operation, the licensee apparently lost sight of this design requirement, however the maximum acceptable leakage of water out of the CCW system including past these isolation valves, has recently surfaced as a concern for system operability. Therefore these valves may have to be leak tested to verify operability. At the time of the inspection, these interface valves were required to close in 15.3 seconds by analysis to assure system operability. The recent IST

test on unit 3 showed that one of these valves closed with a time greater than was allowed by analysis (16.8 seconds) and another valve closed in 15.0 seconds. The possible incorrect designation of these interface valves and the long stroke time is an unresolved item (50-361/88-10-17).

D. The inspector questioned whether SWC valves HV6494 and HV6496 were contained in the licensee's Inservice Testing (IST) program. These valves allow lineup of the SWC system directly to the beach in the event that the circulating water return becomes unavailable. These valves were not part of the IST program. This is considered an apparent violation of Technical Specification 4.0.5 (50-361/88-10-18). The team noted that these valves have been occasionally cycled during the past year to aid in control of marine growth in the circulating water system.

E. The licensee reportedly has an exemption from performing quarterly IST on the CCW non-critical loop isolation valves due to the potential to interrupt cooling flow to the Reactor Coolant Pumps (RCPs). This exemption was apparently requested prior to the modification which now allows those valves to be cycled without flow interruption to the RCPs. The team questioned whether the exemption remains valid in view of the modification. This is open item (50-361/88-10-19).

F. The inspector noted that the CCW surge tank is not periodically checked for nitrogen leakage with the non-safety nitrogen supply secured. Additionally, it was not clear if maintenance of some minimum level of nitrogen pressure is important. This is open item (50-361/88-10-20).

12. General Oversight Activities

The team examined licensee oversight group activities to assess their involvement in significant plant problems. The review attempted to determine the extent of these groups' pro-active initiatives and involvement in activities that would contribute to enhanced plant safety. Oversight group records were examined to identify the extent of their review of issues and adequacy of corrective actions.

A. Nuclear Safety Group

The team reviewed the activities and functions of the Nuclear Safety Group (NSG). The off-site safety review committee functions normally established by the plant technical specifications are implemented by the staff of the Nuclear Safety Group (Technical Specification 6.5.3). Because this is a staff-level group, more time is available to perform reviews of the technical details associated with the review items than if the these reviews were performed by senior facility management. The effects of this difference in approach were evident to the team through the team's review of the documentation of selected reviews conducted by the NSG.

NSG activities are defined in E&C Department Quality Assurance Procedure 40-9-21, Nuclear Safety Group Review, Evaluation and Audit Responsibilities for SONGS. This procedure contained appropriate requirements and guidance to implement technical specification requirements. NSG activities were summarized in monthly reports and various review checklists. The following monthly Nuclear Safety Reports were reviewed:

Period	Report Date
March 1988	4/29/88
February 1988	3/31/88
January 1988	3/15/88
December 1987	1/29/88
November 1987	12/15/87
October 1987	11/30/87
September 1987	10/30/87
August 1987	9/30/87
July 1987	8/31/87

The NSG has initiated probable risk assessment (PRA) analysis on a limited basis. One pro-active feature of this effort includes the monthly tracking of probability of core-melt which is graphically included in the report attachments, allowing assessment by others, including the Nuclear Control Board. Calendar year 1987 showed a dramatic decrease in core-melt probability due to the installation of a back-flush capability for the sea water side of the component cooling water heat exchangers. This modification reduced the amount of time that the heat exchangers were required to be out of service for mechanical cleaning.

The following documents were also reviewed by the team:

Document	Title	NSG Checklist
Audit report SCES-043-87	SONGS 1,2,&3 Corrective Action	1/12/88
PFC 2-87-6554.12 Rev. 1	Fire Isolation Switch Rewiring	11/23/87
PFC 2/3-87-6554.21, Rev. 0	Train A Backup Power Source for Control Room Emergency Lights	1/5/88
PFC 2-87-6554.13, Rev. 0	Saltwater Cooling Valve Platform	9/24/87
PFC 2/3-87-042, Rev. 0	CCW Chemical Addition Platform/ AFW Pump Bearing Leakoff Drain Piping	3/11/88
PFC 2-87-6554.10 Rev. 0	Saltwater Cooling Pump Logic	12/15/87
PFC 2/3-87-6679, Rev. 0	Chlorinating System Supply to CCW HX	10/28/87
PFC 3-87-6554.13, Rev. 0	Saltwater Cooling Valve Access Platform	10/15/87
PFC 2-86-6621.0, Rev. 0	Modification of MSIV Dump Valves	12/4/86

The team also reviewed the current (May 1988) month's set of procedure review checklists, which document NSG review of the changes to plant procedures to verify that unreviewed safety questions were not created. Twenty-four checklists were completed without comments. Four checklists contained comments ranging from

suggestions (eg., battery sizing and loading calculations should be referenced since these determine values used in procedure attachment); to safety improvements (eg., to prevent testing to wrong criteria, performance test criteria should be checked against latest revision of applicable calculations); to safety concerns (eg., procedure fails to verify that an equalizing charge has been conducted between three to seven days prior to the start of the test per IEEE 450-1980 - examples from procedure review checklist for SO123-I-2.6, Rev. 0, Battery Performance Test). Safety improvements and concerns required response from the responsible organization and were tracked by the NSG.

The Nuclear Safety Group conducts monthly meetings to discuss the previous month's activities among the members of the group. Persons in attendance include NSG members and typically a representative from the Quality Assurance organization. Occasionally other interested parties attend, such as ISEG members and the Manager of Nuclear Safety. The team sat in on the June 7, 1988 meeting that discussed the activities reviewed during May 1988. These meetings are typically held at the plant site, which affords NSG members a periodic "forced" opportunity to examine specific plant issues firsthand and conduct facility tours. The supervisor of the NSG indicated that this approach was specifically designed to provide an opportunity to his staff to pursue issues at the site and to allow for frequent oversight tours of the units.

Based on these reviews, the team considered that the Nuclear Safety Group was conducting adequate reviews and that adequate administrative procedures had been implemented to control NSG activities so that technical specification requirements were being met.

B. Nuclear Control Board

A Nuclear Control Board (NCB) has been established to provide for management oversight of the engineering and operations activities. This committee functions to verify that administration, maintenance, and operations are consistent with company policy, rules, approved procedures and operating license provisions, as related to safety and environmental efforts. NCB functions include review of significant safety issues identified by the NSG, OSRC, ISEG or an NCB member. The following documents were reviewed by the team to assess the activities of the NCB:

(1) SONGS 2&3 NCB Charter, dated January 25, 1988.

(2) Minutes and Agenda for NCB Meetings held on April 12, 1988; October 6, 1987; July 7, 1987; April 1, 1987; and January 6, 1987.

Items reviewed included retrofit, OSRC, QA, and nuclear licensing activities, and reports from the ISEG and NSG. In addition, special activities were reviewed, such as the results of a limited scope SSFI and a special presentation on the Chernobyl nuclear accident.

The team noted that the NCB did not appear to initiate many pro-active efforts, as evidenced by no new NCB open items for all of 1987 and 1988 to-date. The team did verify that a NCB recommendation, that site QA forward draft audit plans to the NSG for prior review, was being implemented.

Although the NCB is not required by the SONGS Units 2 and 3 Technical Specifications, this board complements NSG activities by providing a vehicle for senior management oversight of nuclear safety matters in a collegial environment separate from the line management role. The NCB is a commitment established in the FSAR and does satisfy selected requirements for offsite safety review activities specified by ANSI N18.7-1976. The combination of detailed technical reviews by the NSG, complemented by general oversight by the NCB, appeared to be an effective method of implementation of the offsite safety review requirements.

C. Quality Assurance Activities

The team examined recent quality assurance activities that were designed to examine aspects of the licensee's design and modification control programs. In June 1987, audit responsibilities were rearranged such that the site quality assurance organization conducts all audits of engineering for SONGS, including examination of the Engineering and Construction (E&C) efforts. Prior to June, 1987, this activity was a shared responsibility of the site and general office quality assurance groups.

The site quality assurance group recently initiated a program that is intended to conduct detailed technical audits. The program is designed to perform integrated assessments of functions focused on performance. The initial effort in this area was a design control audit, SCES-036-87, initiated during September 1987 and completed on April 8, 1988. This audit was a major undertaking, consisting of over 5,500 man-hours of effort involving three full-time and eight part-time auditors. The audit consisted of a detailed vertical audit of the design change/modification process; from examination of design development, design procedures and training of engineers (both SCE and the principal engineering contractors - Fluor and Bechtel) through the design change implementation process at the site; including construction, turnover and testing, and final documentation. The audit initially planned to examine 3 DCPs, however the sample size was expanded to twenty due to the nature of problems identified during the audit.

The results of this audit had not been formalized during the conduct of the team inspection. However, the team did review available documentation detailing specific findings and the audit approach. As an example, the inspection plan detailed a thorough check for compliance with the design controls required by ANSI N45.2.11. However the inspectors' completed checklists were not yet available. In addition the specific findings of the audit had been translated into 36 corrective action reports, 32 problem review reports, and 3 nonconformance reports. The team examined the subject forms and the

licensee's proposed corrective actions (if determined). The findings of the audit appeared to be consistent with those of the SSFI team for those areas that were common to both efforts.

The licensee stated that they plan to continue conducting integrated assessments similar to SCES-036-87, with at least one such audit scheduled per year. The licensee indicated that the next assessment would examine the maintenance program in depth.

The team also observed that a memorandum addressing the review process for nuclear safety concerns, from the SCE chairman of the board to all personnel, was posted throughout the general offices. This memorandum established a three tier process for raising concerns:

(1) through the immediate supervisor
(2) through the head of the appropriate safety review organization
(3) through the Nuclear Safety Concerns Program administered by the quality assurance organization

For general office personnel, the Manager of Nuclear Safety was designated head of the safety review organization. The team interviewed this manager regarding activity by employees in raising concerns. The manager indicated that no such concerns had been raised through him. He indicated that the normal vehicle for raising such concerns was the Nuclear Safety Concerns Program, which affords an opportunity to raise concerns anonymously.

D. <u>On-Site Review Committee (OSRC) Activities</u>

Activities of the OSRC were inspected to determine the OSRC involvement in chronic or significant plant problems and plant modifications, and to assess the performance of its mission as described in Section 6.5.1 of the Technical Specifications for Units 2 and 3. In the course of the inspection, the inspector reviewed recent reports, reviews, meeting summaries, recommendations for actions, and other documents associated with or resulting from OSRC action. The inspector attended a regular monthly meeting of OSRC, and afterward interviewed some participants and presenters, including the OSRC chairman (who is the Station Manager). During the meeting, the inspector noted the manner in which OSRC addressed various agenda items, including a formal review of units operations to detect "potential nuclear safety hazards", a review of all reportable events, technical specifications violations that had been investigated and reported to the nuclear safety group (NSG), licensee event reports, notice of violation responses, and other such matters of the past month. The inspector observed during the meeting that, although there were numerous items considered on the agenda, the Committee chairman pressed vigorously to ensure clarity and completeness on all substantive points. Of particular note was sharp questioning by the Committee chairman regarding LER-1-88-007 (an improperly discarded sample) and a response to NOV in IR

50-206/88-3, dated 4/15/88 (a N_2 cylinder leakage and safety implications).

In reviewing OSRC documents, the inspector observed that the OSRC was much inclined to apply the "50.59 process criteria" to a wide range of items of possible safety concern (as LERs, NOV responses).

In addition to regular monthly OSRC meetings, there are special called meetings where requested by the Station Manager or the NSG Supervisor. One such recent meeting on March 10, 1988 was to consider a detailed review of the unreviewed safety question (USQ) and potential nuclear safety hazard (PNSH) aspects of proposed transshipment of spent fuel from Unit 1 to Unit 2. (The application of the 10 CFR 50.59 criteria to parts of this proposed action is addressed elsewhere in this report).

The inspector discussed the role and activities of OSRC with other plant personnel (other than OSRC Chairman and members of the Committee), including the resident inspector, other inspection team members, and the NRR project manager.

The broad range of OSRC responsibilities makes it difficult for OSRC to carefully review all issues that come before it. This situation is largely overcome by the strong direction given by the OSRC chairman. OSRC appears on the whole to do a good job of fulfilling its responsibilities as defined in Section 6.5.1 of the Unit 2/3 Technical Specifications.

E. Independent Safety Engineering Group (ISEG) Activities

Activities of ISEG were inspected to assess the Group's involvement in plant problems and its effectiveness in carrying out its mission as described in Section 6.2.3 of the Unit 2/3 Technical Specifications (TS). Although ISEG was originally a subgroup of the Nuclear Safety Group (NSG), an offsite group also described in the TS, ISEG is an independent group, formally reporting (as does NSG) to the Manager, Nuclear Safety.

The inspector conducted two interviews with the ISEG supervisor with another ISEG member present, reviewed samples of different kinds of reports prepared by ISEG, discussed the activities of ISEG with the ISEG supervisor and with other plant personnel including the Station Manager, and with a resident inspector, the senior resident inspector, other team members, and the NRR project manager.

Because of the role of ISEG as described by plant TS, an assessment of its impact on plant safety is more subjective than usual. ISEG is tasked to review NRC issuances, industry advances, and other sources of plant design and operating information that might indicate areas for improving performance and safety at SONGS, and to make recommendations based on that information. ISEG has ready access to early information (as INPO notepad and NRC morning call) regarding global current plant operating experience. It digests and disseminates this information regularly for possible application of

safety or operational significance for SONGS. The ISEG supervisor stated that a primary function of ISEG is to "get the word out". This mission is accomplished by regular issuance of ISEG Operating Experience Summaries, an "electronic bulletin board" - type publication that summarizes events and causes, with lessons learned for SONGS. ISEG also issues Action Requests that result from an ISEG Field Evaluation of a plant event, and performs other special studies directed toward betterment of plant operation and safety. The ISEG supervisor stated that he believes ISEG drives a lot of work done at the site. The inspector reviewed samples of such ISEG documents and interviewed some of those to whom certain ISEG-generated reports were directed.

From the above-described inspection activities, including observations, interviews, and reviews of documents information, it appears that ISEG is effective in fulfilling its functions as described in the TS, and in fact exercises some proactive influence for the betterment of plant operation and safety by early identification of problem areas.

13. Additional Pro-Active Initiatives and Activities

A. Design Control Task Force

The Engineering and Construction Department established a design control task force in 1988 to examine the licensee's design change process and recommend process improvements. The task force, headed by a project manager within the Nuclear Generation Engineering group, was established following a series of events and reviews which indicated weaknesses in the licensee's current methods. These included a notice of violation relating to Unit 1 batteries, an INPO review in December, 1987, the design change QA audit, and general reviews of the results of NRC inspections, such as SSFIs recently performed at other facilities. The team reviewed the objectives of this task force and its accomplishments to date.

The task force broke down the design modification process into more than 70 discrete steps, in addition to several general aspects related to the design process. These steps were then prioritized regarding the task force members' perception of implementation problems associated with each step. The steps that were identified as in need of improvement or as problem areas were then sorted and assigned to one of five separate task groups which deal with general areas of the design change process. These groups included design basis, interoffice communications, work process, document retrieval, and training.

An example of a work product emerging from this task force was the design basis task group's development of a request for proposal for a design basis document and an associated design roadmap document. This proposal was in response to areas which were considered weak within the steps of the design/modification process. The request, which was issued during the inspection, defines the design basis as that body of information which identifies the specific functions to

be performed by a structure, system, or component; and the specific values or ranges which constitute controlling parameters as reference bounds for detailed plant design necessary to ensure public health and safety. These values are established by plant safety analysis, regulatory or other pre-approved criteria, codes and standards, or other commitments which define acceptable safety margins against which the plant was reviewed and approved by the NRC. The design roadmap document was envisioned as a tool to aid the engineer in identifying, locating, and using appropriate documents to be considered in the performance of design activities. A final decision regarding whether or not to develop a design basis document and/or a design roadmap had not been made by the licensee.

These efforts, and similar areas being examined by the design control task force, appeared to represent a significant pro-active undertaking by the licensee, independent of the initiative of the normal oversight groups.

B. San Onofre Project Review Meetings

The team discussed the conduct of San Onofre Project review meetings with staff members that participate in these meetings and in addition reviewed the agenda of meetings held on July 16, 1987; January 20, 1988; and March 18, 1988. These meetings are well attended by line management, including the Vice Presidents of NES&L, E&C, and NGS. These meetings provide an additional level of facility oversight, and in addition form an avenue for the initiation of pro-active initiatives. For example, the limited scope SSFI performed for NES&L was initiated following the January 1988 meeting. Results of this effort were reviewed by the NCB during the March meeting.

C. Limited Scope Safety System Functional Inspection

The licensee had been monitoring NRC and industry experience relating to design inspections. As a result, the licensing department had been investigating contracting an organization to perform an internal SSFI late in 1987. Following notification that the NRC planned to conduct an SSFI, SCE proceeded with a limited scope SSFI on an expedited basis to probe for weaknesses in this area. The shutdown cooling system was selected for review by a contractor with experience in conducting SSFIs.

The team was provided the results of the initial report of this limited scope SSFI, dated May 12, 1988. (The licensee indicated that the report was still under review and would be reissued.) The effort resulted in 30 requests for information for specific findings and, in addition, 4 programmatic concerns:

(1) Design criteria and inputs were not rigorously specified in the design change packages.

(2) Design change packages did not contain supporting calculations.

(3) Post-modification testing requirements were not specified by engineering.

(4) Appropriate documentation was not revised following modifications, with a specific example being numerous FSAR errors.

Corrective action to address these findings had been slowed due to the press of activity related to outages for Units 1 and 3 and the NRC's SSFI.

Although the team did not perform a detailed review of the limited scope SSFI, it appeared that the results of the review were valid and consistent with the findings of the NRC team.

D. State of the System Reports

The licensee initiated reviews of selected systems following discovery that the NRC planned to conduct an SSFI in 1988. The reports were compiled in anticipation of the SSFI by the respective system cognizant engineers. The emphasis of these reviews was on the functionality of the systems. Specific areas examined included design/licensing basis, operations and maintenance. Recently drafted design reports (performed under contract by Bechtel, the architect/engineer of record) provided a basis for the review.

The team reviewed these reports (issued in May 1988) for the salt water cooling system and the component cooling water system. The reports identified functional problems with the selected systems typical of those identified during NRC SSFIs. In particular, the findings appeared consistent with the team's examination of the selected systems.

The design reports reviewed were draft documents that had not received quality assurance and design verification reviews; nonetheless they appeared to represent an enhanced compilation of design basis information, although there were a number of obvious errors noted, apparently due to the information being drawn from sources which had not been maintained up-to-date.

The team considered initiation of these reviews to be a significant pro-active initiative by the licensee that had identified problems with safety significance for resolution. The licensee was strongly encouraged to continue particularly the effort of the cognizant engineers, in that their system reviews appeared to provide significant benefit in gaining a full understanding of the systems and in identifying problems.

14. 10 CFR 50.59 Activities

The quality of safety evaluations performed in accordance with 10 CFR 50.59 was inspected. The inspection was not confined to the application of 10 CFR 50.59 to perform safety evaluations for design changes, but included uses by the licensee in other areas as well, as in the

disposition of some nonconformance reports (NCR), and assessing the safety implications of technical specification (TS) violations and reportable plant events (LERs), and responses to notices of violations (NOVs).

Samples of specific plant design change items and events (PFCs, NOVs, NCRs, LERs, TS violations) were chosen to assess the quality of 10 CFR 50.59 reviews performed. Some design change items and events included in the sample are recent and some are not. The purpose here is not to shed new light on these design change items or events, but to assess the use of the 10 CFR 50.59 process in evaluating the safety implications of proposed plant design changes and plant events. Of the large number of such items reviewed, the several cited below are meant to be representative.

PFC No. 2/3-83-242 - This design change was to provide an interlock between closed cooling water cross tie isolation valves to allow the operator to keep both CCW trains operating during changeover from operating train to standby train. This would assure that there would be no interruption of CCW to RPC seals. To accomplish this it would be necessary to change the setpoint for the CCW surge tank safety relief valves and the nitrogen regulator valves so as to increase the capacity of the system.

This PFC was completed, implemented, and tested. However, the safety evaluation performed under 10 CFR 50.59 consisted of checking "No" to the three criteria questions for an USQ and a one-sentence justification as a basis for the "No" findings.

Spent Fuel Transshipment from Unit 1 to Units 2/3 - This proposed action was the subject of a Special Meeting of the Onsite Review Committee (OSRC) (88-03) on March 10, 1988, reconvened on March 15, 1988. The Special Meeting of OSRC was called "to discuss the Unreviewed Safety Questions (USQ) determination on the proposed fuel transshipment."

The review had considered the criteria from an USQ as given in 10 CFR 50.59 and applied the criteria to an evaluation of the Unit 1 turbine gantry crane, Unit 1 transshipment activities, and Units 2/3 transshipment activities. The conclusion reached for each criteria, for each activity to which it was applied was "No", so the overall conclusion was that no USQ was involved. It appeared to the inspector that the provisions of 10 CFR 50.59 had been excessively narrowly applied in reaching a "no unreviewed safety question" conclusion.

NCR 2-2404 dated 5/9/88 - A pipe and safety valves 2PSV-8404/8409 were experiencing excessive vibration. An interim repair (shims under pipe) and permanent repair (modify pipe supports) were proposed for disposition of this NCR.

A 10 CFR 50.59 safety evaluation was performed that described the proposed fix in some detail, but did not [illegible] to answer the questions regarding the probability and consequen[illegible] [illegible]ident[illegible], aths. previously analyzed or different from [illegible] [illegible] analyzed in the

FSAR. However, a failure analysis of the SRV vibration and a root cause determination were performed that were complete and acceptable.

NCR-3-1960 dated 5/16/88 - Indicator for containment isolation valve for penetration 47 indicated "intermediate" when value is fully closed. Containment was unable to reach test pressure during LLRT in accordance with 5023-V-3.13. The cognizant engineer's "Disposition/Comments (including 50.59 aspects)" recommended a disposition of "rework" according to detailed disposition instruction. A 10 CFR 50.59 evaluation is not needed for this choice of disposition, but a root cause analysis will be performed.

NCR 2-2417 dated 6/7/78 - A SWC pump motor mounting was flawed. This was found earlier by NDE and reported on W088050191 dated 6/6/88. the disposition recommended by the cognizant engineer is "accept-as-is". The inspector questioned the cognizant engineer in detail about the basis for this disposition. After some clarifying discussion, the inspector agreed that the "accept-as-is" disposition is appropriate. The 10 CFR 50.59 evaluation required for an "accept-as-is" disposition had been performed, but did not seem to address the questions of increased consequences of a previously evaluated accident or the possibility of creating a different type of accident.

NCR 2-1912 dated 9/24/86 - A mechanical snubber on a 30-inch salt-water pipe in a tunnel was found during a 1986 inspection walkdown to have heavy corrosion on the outer casing. The snubber was functional-tested and found to be frozen. The disposition by the cognizant engineer was to replace it and send the defective snubber to Station Technical for evaluation. A written engineering evaluation was performed, which concluded that the component would have performed its safety function had a DBE occurred. However, the corrosive environment to potentially (or actually) degrade pipe supports, snubbers, other structural members, a flow meter, and possibly other equipment/components important to safety had been known since December 1983 and not corrected. Apparently no safety evaluation had been performed to evaluate the degradation of important-to-safety equipment in this corrosive environment.

While there is extensive use of the three criteria given in 10 CFR 50.59 to determine USQs associated with plant design changes and to evaluate plant events, there is a need for improved training of station technical personnel in the use of 10 CFR 50.59 in performing safety evaluations and a need for improved documentation of the considerations made to reach 50.59 conclusions. The team recognizes that extensive documentation is normally not necessary, however, a brief statement of considerations made would aid in ensuring that all potential problems were considered.

15. Training Programs for SCE Corporate Engineers

Formal training for SCE Corporate engineers is described in Engineering and Construction (E&C) Procedure 41-1-2. This program is essentially an annual review of important E&C internal procedures. The review of procedures includes, but is not limited to training on safety evaluation and design change control. A formal tracking system has been implemented to ensure all corporate engineers' training remains current. The

training records were reviewed and generally, all the engineers training was current.

SCE had not instituted a formal training program for newly hired corporate engineers other than the program addressed. Additionally corporate engineers receive no specific training on the nuclear plants' design basis, systems or integrated system operation. SCE corporate engineering relies heavily on the engineer's previous work experience and education. Additionally, SCE engineering management noted the turnover rate of corporate engineers is low.

Based on this review and the findings of the team during the SSFI team inspection, the team concluded that the licensee does not have an adequate training program for the corporate engineers. SCE's training program does not evaluate the individual engineer's knowledge nor does the program establish a standard knowledge requirement. The absence of training on the licensee's nuclear units' design bases and systems is a fundamental weakness.

16. Management Meetings

The Team Leader met routinely with licensee management to discuss the progress of the inspection, details of preliminary issues, and requests for additional data. On June 6, 1988 the inspection team and NRC managers met with the licensee staff to summarize the inspection findings and their relationship to license conditions, performance indicators and current issues.