ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Report No.:	50-361/96-13 50-362/96-13
Licensee:	Southern California Edison Co.
Facility:	San Onofre Nuclear Generating Station, Units 2 and 3
Location:	5000 S. Pacific Coast Hwy. San Clemente, California
Dates:	September 16-20, December 9-13, 1996, and January 14, 1997
Inspectors:	W. P. Ang, Senior Reactor Inspector C. J. Myers, Reactor Inspector D. B. Pereira, Reactor Inspector
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Attachment:	Supplemental Information

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EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Units 2 and 3
NRC Inspection Report 50-361/96-13; 50-362/96-13

This inspection evaluated the licensee's self-assessment effort that was performed as an alternative to an NRC engineering and fire protection team inspection.

Engineering

- The licensee's self-assessment team was composed of well-qualified individuals, who were sufficiently independent and objective. The team was capable of performing good assessments (Section E1.2).

- The scope and depth of the engineering self-assessment team was sufficient to satisfy the requirements of NRC Inspection Procedure 37550 (Section E1.3).

- The licensee performed a good self-assessment of the implementation of its fire protection program and generally addressed the inspection objectives of NRC Inspection Procedure 64704. However, the fire protection self-assessment did not address some important inspection attributes such as the implementation of the corrective action program, observation of fire watch activities, and the adequacy of certain fire suppression features. Therefore, additional NRC inspection of this area is required (Section E1.3).

- The licensee's engineering self-assessment team concluded, overall, that San Onofre Nuclear Generating Station engineering was meeting program requirements. The team noted the following strengths in the engineering areas reviewed (Section E1.4):

  - San Onofre Nuclear Generating Station had strong engineering departments. The engineering staff was knowledgeable, experienced, and exhibited strong analytical capabilities.

  - Engineering was responsive to plant safety/operability issues.

  - The Nuclear Engineering Design Organization generated fundamentally sound designs that worked well.

  - Engineering computer tools and trending were comprehensive.

- The licensee's engineering self-assessment team noted the following weaknesses in the engineering areas reviewed (section E1.4):

  - Little or no action was being taken on certain low priority engineering tasks.

- There was incomplete task coordination between engineering groups. Support from one engineering group to another was incomplete.

- Some engineering mistakes indicated a lack of attention to detail.

- Some interface weaknesses in integrating vendor equipment with field equipment were identified.

- The licensee's engineering self-assessment team concluded, overall, that the San Onofre Nuclear Generating Station fire protection program was effective in its ability to prevent, detect, and respond to a fire emergency. The team observed the following strengths of the program (Section E1.4):

  - Good awareness and ownership of the program was demonstrated by personnel associated with the fire protection program.

  - Good material condition of the fire protection systems were observed.

  - Good fire department readiness and response was observed.

  - Good industrial safety practices were observed during the assessment.

- The licensee's engineering self-assessment team noted the following fire protection program implementation weaknesses (Section E1.4):

  - Material condition and storage practices for flammable material storage cabinets did not meet procedural requirements.

  - Fire extinguishers in the plant did not meet procedural acceptance criteria subsequent to performance of surveillances that accepted the conditions identified.

  - There was a lack of operations involvement in fire drills.

- The NRC inspectors generally agreed with the conclusions of the self-assessment team (Section E1.5).

- The NRC inspectors determined that seven self-assessment team observations represented five examples of a failure to correctly translate applicable regulatory requirements and the design basis into specifications, drawings, procedures, and instructions. Taken together, these five design control discrepancies are a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." This licensee-identified violation is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (Section E1.5.b.(1)).

- The NRC inspectors questioned the results of the licensee's 10 CFR 50.59 review of a design change involving the Diesel Generator electrical cross tie. The licensee had determined that the design change would not result in an unreviewed safety

question; however, the inspectors questioned the licensee's basis for determining that the modification did not increase the likelihood or consequences of an accident. Although the licensee had begun installation of the modification in Unit 2, in response to this concern, the licensee electrically isolated the modification and submitted the design change to the Office of Nuclear Reactor Regulation for further review. This issue will be followed as an unresolved item pending this evaluation (Section E1.5.b.(2)).

- The inspectors determined that three of the licensee's engineering self-assessment team observations indicated a weakness in the implementation of the corrective action program (Section E1.5.b.(3)).

- The NRC inspectors determined that three engineering self-assessment team observations represented three examples of a failure to comply with fire protection procedures. This licensee-identified procedure violation is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (Section E1.5.b.(4)).

Report Details

III. Engineering

E1 Engineering Self-assessment

E1.1 Introduction

An NRC Region IV team inspection was planned to accomplish the core inspection program requirements of NRC Inspection Procedures 37550, "Engineering," and 64704, "Fire Protection Program," in September 1996, at the San Onofre Nuclear Generating Station. In a letter dated June 24, 1996, Southern California Edison, the licensee, proposed to perform a self-assessment of San Onofre Nuclear Generating Station engineering activities and programs, and the fire protection program, in lieu of the planned NRC engineering and fire protection team inspection. The licensee included its engineering and fire protection self-assessment plan as an attachment to the letter. The plan provided the details regarding the schedule, objectives, scope, methodology, and team qualifications for the self-assessment.

In a letter dated August 16, 1996, the NRC informed the licensee that their request was reviewed in accordance with the guidelines contained in NRC Inspection Procedure 40501, "Licensee Self-Assessments Related to Team Inspections," and accepted the proposed timing, scope of effort, and the credentials and experience of the assessment team. The option of permitting licensees to conduct a self-assessment in lieu of an NRC inspection was an NRC program aimed at minimizing regulatory impact and utilizing NRC resources more efficiently. Consequently, the NRC informed the licensee that it planned to reduce the level of inspection effort planned for Inspection Procedures 37550, "Engineering," and 64704, "Fire Protection Program," at the San Onofre Nuclear Generating Station. The NRC further informed the licensee that it planned to perform an inspection of the self-assessment effort in accordance with Inspection Procedure 40501.

The licensee performed an engineering and fire protection self-assessment at the San Onofre Nuclear Generating Station on September 9-20, 1996. The results of the self-assessment were contained in a self-assessment report that was submitted to the NRC on December 4, 1996. The NRC inspectors monitored the performance of the engineering and fire protection self-assessment on September 16-20, 1996, reviewed the self-assessment results, and independently inspected the areas assessed by the licensee on December 9-13, 1996. The NRC inspectors continued the review of documents in office, and discussed by telephone the self-assessment results with the licensee, on January 9, 1997. The purpose of the NRC reviews and inspections was to determine the quality and results of the licensee self-assessment.

E1.2 Self-Assessment Team Qualifications, Objectivity, and Independence

a. Inspection Scope (40501)

The purpose of this inspection was to determine the qualifications, objectivity, and independence of the licensee's self-assessment team in accordance with NRC Inspection Procedure 40501, "Licensee Self-Assessments Related to Team Inspections." The inspectors determined the qualifications, objectivity, and independence of the licensee's engineering self-assessment team and individual team members by performing the following:

- The inspectors reviewed the licensee's engineering self-assessment plan prior to, and during, the onsite inspection. The inspectors were informed by the licensee of changes to the team composition due to availability of personnel during the onsite inspection.

- The inspectors interviewed selected team members.

- The inspector accompanied and observed selected team members during their assessment activities. At the time of the onsite inspection, the ongoing team assessment activities were predominantly interviews of licensee personnel and reviews of engineering work products.

- The inspector observed daily team meetings and the team's management exit briefing.

b. Observations and Findings

The inspectors found that the engineering self-assessment team consisted of seven Southern California Edison engineering personnel, one utility manager from Diablo Canyon Nuclear Power Plant, and one consultant. The inspectors determined that the team was composed of diverse and well-qualified members who were sufficiently objective and independent to allow meaningful assessments. Specifically, the inspectors noted the following:

- All engineering team members were degreed engineers. The major engineering disciplines of mechanical, electrical, civil, and nuclear engineering were represented on the team.

- All team members had 10 or more years of industry experience in design, systems, or plant support engineering functions.

- Seven of the nine engineering team members were registered professional engineers.

- Team member assessment activities observed by the inspectors were independent and self-critical. Team meetings observed by the inspectors included critical discussions of team member observations. The team's management exit briefing provided a critical discussion of the team's findings.

Two additional team members performed the fire protection program self-assessment. One of the individuals was a Nuclear Oversight Division auditor, and the other was a fire protection supervisor from the Palo Verde Nuclear Generating Station. The inspectors determined that the fire protection self-assessment team members were qualified and performed an independent and objective assessment of the fire protection program.

c. Conclusion

The inspectors concluded that the team was composed of well-qualified individuals who were sufficiently independent and objective. The team was capable of performing good assessments.

E1.3 Scope and Depth of Self-Assessment

a. Inspection Scope (40501)

The inspectors reviewed the scope and depth of the licensee's self-assessment to determine if it was equivalent to those requirements specified in NRC Inspection Procedures 37550, "Engineering," and 64704, "Fire Protection Program." The inspectors reviewed the self-assessment plan, observed in-process assessment activities, interviewed licensee personnel, and reviewed the licensee's self-assessment report.

b. Observations and Findings

Engineering Self-Assessment

The inspectors determined that the licensee's engineering self-assessment consisted of both a horizontal and a vertical review. The licensee chose to review the emergency chilled water system and the normal chilled water system for the vertical review of engineering activities for those systems. The adequacy of engineering activities regarding design and configuration control, surveillance and testing, and identification/resolution of problems associated with those systems were reviewed.

The licensee chose six engineering subject areas for the horizontal review of engineering activities. General engineering capabilities, design change review, controls for identifying, resolving and preventing technical problems, maintenance and operations support, independent review and operational experience feedback, and system engineering functions were the subject areas reviewed.

The team prepared and completed a matrix of team assessment activities as compared to the inspection requirement attributes of NRC Inspection Procedure 37550.

The inspectors found that the licensee had addressed all portions of Inspection Procedure 37550. The inspectors noted that the licensee had not reviewed any modifications required by paragraph 2.08 of NRC Inspection Procedure 37550, because modifications of the type had not been performed within the last 2 years. This inspection requirement pertained to design changes and modifications installed as part of NRC regulations 10 CFR 50.62, 10 CFR 50.63, and Supplement 1 to NUREG-0737 for Regulatory Guide 1.97 instruments and the safety parameter display system.

Fire Protection Self-assessment

The licensee chose to evaluate the adequacy of its fire protection program by (1) reviewing the approved program, (2) verifying the adequacy of implementing procedures, (3) verifying procedure compliance, (4) verifying installation, operability and maintenance of fire protection systems and equipment, (5) assessing fire prevention readiness, (6) reviewing the implementation of the quality assurance program, and (7) evaluating the effectiveness of fire protection corrective actions.

The licensee team also prepared and completed a matrix of team assessment activities as compared to the inspection attributes of NRC Inspection Procedure 64704. The NRC inspectors found that the licensee addressed a majority of the attributes of Inspection Procedure 64704; however, the following attributes of the inspection procedure were not addressed:

- Classroom and practical fire training was not observed; however, the team did observe the performance of a fire drill.

- Fire watch activities were not observed.

- The adequacy of corrective actions related to fire protection program audits was not reviewed.

- Licensed operator understanding and training in the use of safe shutdown procedures and the pre-fire strategy manual was not reviewed.

- Standpipe fire hose length was not assessed.

- The program used to detect corrosion, erosion, protective coating failure, silting and biofouling in fire system piping and components was not reviewed.

c. Conclusion

The NRC inspectors concluded that the scope and depth of the engineering self-assessment was sufficient to satisfy the requirements of NRC Inspection Procedure 37550. The licensee also performed a good self-assessment of the implementation of its fire protection program and generally addressed the inspection objectives of NRC Inspection Procedure 64704. However, the fire protection self-assessment did not address some important inspection attributes, such as the implementation of the corrective action program, observation of fire watch activities, and the adequacy of certain fire suppression features. These items will be followed as an open item and reviewed during a future NRC inspection (50-361;362/9613-01).

E1.4 Significant Self-assessment Team Conclusions

a. Inspection Scope (40501)

The NRC inspectors reviewed the licensee's self-assessment report. The NRC inspectors discussed the results of the self-assessment with licensee managers, the team leader, and team members. The inspectors summarized the licensee's conclusions.

b. Observations and Findings

Engineering Self-assessment

The licensee team concluded, overall, that San Onofre Nuclear Generating Station engineering was meeting program requirements.

The assessment team noted the following strengths in the engineering areas reviewed:

- San Onofre Nuclear Generating Station had strong engineering departments. The engineering staff was knowledgeable, experienced, and exhibited strong analytical capabilities.

- Engineering was responsive to plant safety/operability issues.

- The Nuclear Engineering Design Organization generated fundamentally sound designs that worked well.

- Engineering computer tools and trending were comprehensive.

The assessment team noted the following weaknesses in the engineering areas reviewed:

- Little or no action was being taken on certain low priority engineering tasks.
- There was incomplete task coordination between engineering groups. Support from one engineering group to another was incomplete.
- Some engineering mistakes that were indicative of lack of attention to detail.
- Some interface weaknesses in integrating vendor equipment with field equipment.

The assessment team noted that the material condition of the plant was well maintained and plant equipment received adequate engineering attention in the areas that they reviewed.

The assessment team identified the following specific "significant findings":

- Some emergency chilled water and heating, ventilating and air conditioning instrumentation, and control circuits were not powered in accordance with the Updated Final Safety Analysis Report requirements for shared systems. Design bases documentation reviews did not pick up the power source deficiencies for the auxiliary building emergency chilled water system.
- The emergency chilled water system design incorporated a number of equipment protective trips that were not adequately specified in either the system design bases document or the setpoint calculations. Some of the trips were redundant to each other, were not required for system operation, and unnecessarily challenged system operability.
- Some actions for open item reports for the emergency chilled water system design bases document were deferred for an excessive period of time (5 years). This was an example of the backlog of low priority engineering tasks.
- A mechanical heat load calculation for the control room complex incorporated interim design inputs (not an as-built source). Engineering did not validate the source.
- Some interim design change notices were not properly converted to as-built documentation after design change turnover, resulting in inadequate configuration control.
- A significant portion of low priority NEDOTRAK items typically took more than 3 months past the requested due date to complete.

- An engineering evaluation for a Nonconformance Report concluded that emergency chilled water instrumentation was qualified to perform at elevated temperatures, but did not address operability of all affected equipment.

- The 10 CFR 50.59 evaluation for a diesel generator cross-tie design change did not clearly address use of the design during a 10 CFR 50.54(x) condition.

- The engineering ammonia limit used in the toxic gas isolation system setpoint calculations was not incorporated properly into Technical Specifications/Licensee Controlled Specifications. This issue was originally identified by the Nuclear Oversight Division in a previous assessment. Also, the ammonia setpoint in the design calculations was not updated into the loop drawings.

- The number of construction problem reports generated against a toxic gas isolation system design change package indicated a weakness in the integration of vendor equipment with field equipment.

- The action request process was generating more action requests than current resources were equipped to handle.

- Actions assigned as a result of Independent Safety Engineering Group recommendations from Industry Experience Evaluations were not scheduled and implemented in a timely manner. Five examples were noted of actions assigned without forecast dates or were past due.

- There was an incomplete understanding of Maintenance Rule impact on station technical function by station technical cognizant engineers (incomplete training).

Fire Protection Program Self-Assessment

The licensee team concluded, overall, that the San Onofre Nuclear Generating Station fire protection program was effective in its ability to prevent, detect, and respond to a fire emergency. Personnel associated with the fire protection program demonstrated strong ownership of the program and good working relationships between departments.

The assessment team observed the following strengths of the program:

- Good awareness and ownership of the program was demonstrated by personnel associated with the fire protection program.

- Good material condition of the fire protection systems was observed.

- Good fire department readiness and response was observed.

- Good industrial safety practices were observed during the assessment.

The assessment team noted the following weaknesses:

- Material condition and storage practices for flammable material storage cabinets did not meet procedural requirements.

- Fire extinguishers in the plant did not meet procedural acceptance criteria subsequent to performance of surveillances that accepted the conditions identified.

- There was a lack of Operations involvement in fire drills.

E1.5 Independent NRC Inspection

a. Inspection Scope (40501)

The NRC inspectors observed licensee performance of the self-assessment, and reviewed the detailed self-assessment plan implementation matrix, the team's assessment observation reports, the licensee's self-assessment report, and the associated action requests to develop an understanding of the basis for the self-assessment team's conclusions.

The NRC inspector walked down accessible portions of the instrument air system and the emergency chilled water system to confirm the as-built configuration of the systems.

The NRC inspectors observed the licensee team's walkdowns of fire protection facilities and equipment. The inspectors observed fire system surveillances and walkdowns of the fire protection facilities and equipment that were performed by the licensee team. The NRC inspectors also observed the fire drill that was also observed by the team.

The NRC inspectors interviewed self-assessment team members and other cognizant licensee personnel. The inspectors also attended self-assessment team meetings and the self-assessment team exit briefing.

b. Observations and Findings

The NRC inspectors generally agreed with the conclusions of the licensee's self-assessment team. The inspection activities, which resulted in a divergent or amplifying view, are described below.

The NRC inspectors noted that the licensee's self-assessment resulted in 37 engineering assessment observations and 11 fire protection assessment observations. The NRC inspectors determined that 7 of the engineering assessment observations indicated a weakness in design control and that 3 of the engineering assessment observations indicated a weakness in the implementation of the corrective action program. The NRC inspectors also determined that two of the engineering observations related to weak performance of a 10 CFR 50.59

evaluation. The inspectors noted that 3 of the fire protection assessment observations were indicative of a weakness in complying with applicable procedures. Each of these four weaknesses are discussed below.

(1) Design Control

(a) **Instrumentation for Shared Emergency Chilled Water System Powered Only from Unit 2 - (Licensee Self-Assessment Observations 005 and 021)** - The licensee team noted that the emergency chilled water system is shared between Units 2 and 3. Two 100 percent redundant trains were provided. The Updated Final Safety Analysis Report, Sections 8.3 and 9.4.2, states that individual electrical loads can be powered by either unit through manual transfer switches. The licensee team determined that the power supplies for Emergency Chillers E335 and E336, and their associated chilled water pumps, can be aligned to either unit. However, the team also noted that the instrument loop power supplies for Chilled Water Flow Transmitters, 2/3 FT 9894-2 and 2/3 9874-1, and control room indicators associated with the chillers come from Unit 2 only. Further review by the licensee team identified 16 additional control room emergency ventilation shared instrumentation and interlocks that were also powered only from Unit 2.

The licensee issued Action Requests 960900515 and 960900750 to address the team's observations. The licensee performed an operability evaluation for the noted conditions, as part of the action requests, and determined that the emergency chilled water system and associated components remained operable due to system design capabilities and redundant Unit 2 power supplies. An evaluation to determine the root cause of the noted conditions and the corrective actions for the conditions was scheduled to be completed by April 1, 1997.

(b) **Incorrect Emergency Chilled Water System Pump Suction Pressure in Design Documents - (Licensee Self-Assessment Observation 025)** - The licensee team noted that Design Basis Document DBD-SO23-800, "Auxiliary Building Emergency Chilled Water System," Revision 3, did not adequately recognize system design requirements necessary to prevent voiding in the emergency chilled water system. The team noted that page 43 of the design basis document specified "minimum 4 psig" for the emergency chilled water pump net positive suction head; page 73 of the design basis document specified 10 feet; and, the pump characteristic curve on page 82 of the design basis document required 3.7 feet. The licensee team considered the difference in the numbers to be of lesser significance than the interpretation of whether psig or psia was intended. The team felt that net-positive suction head was the amount of head needed above the liquid vapor pressure. Consequently, the licensee team considered that the minimum net positive suction head should be 4 psia, not 4 psig. The team further noted that Design Calculation J-GJA-085, "ECW Pump Suction Pressure Analytical Limit," also used a 4 psig pump suction pressure limit as the criteria for the emergency chilled water system low pressure trip.

The licensee issued Action Request 960900743 to address the assessment team's observations. The licensee performed an operability evaluation for the noted conditions, as part of the action requests, and determined that the emergency chilled water system and associated components remained operable because an adequate net positive suction pressure was being provided to the pumps. Analytical/calculation corrective actions for the conditions was scheduled to be completed by January 31, 1997.

(c) **Inconsistencies Between Calculations and Design Basis Document - (Licensee Self-assessment Observation 004)** - The licensee team noted that Design Basis Document DBD-SO23-800, "Auxiliary Building Chilled Water System," Revision 3, Table 1-2, listed the chilled water coil capacity versus heat load. The table identified several discrepancies between the calculated heat load and coil capacities. Engineering evaluations performed at the time the design basis document was completed in 1991 concluded that the coil capacities were adequate for the expected heat load, but the calculations had to be revised to reflect the correct heat loads. Two open item reports (91-092 and 91-093) had been generated in 1991 to document these calculation discrepancies. Revision 3 of Design Basis Document DBD-SO23-800, issued on August 30, 1996, still listed these two open item reports as remaining unresolved. In addition, the following note dating back from the original issue of the design basis document in 1991 was still included in the latest revision of the design basis document: "The revision to Calculation M-0073-043 is in Revision "A" review. Preliminary results indicate that the cooling coils have adequate additional capacity."

The licensee team discussed the discrepancies with the system design engineer and the Nuclear Engineering Design Organization supervisor, and was informed that heat loads for several areas within the auxiliary building had been updated. However, the heat loads in peripheral areas still needed to be updated. The licensee team was also informed that resolution of open item reports had been looked at programmatically. Those with safety impact were scheduled to be completed immediately. Those which did not have safety impact were to be scheduled to be completed within a 3-year period ending December 1997. However, engineering management decided in early 1996 to accelerate that schedule and complete the outstanding open item reports by December 1996.

The licensee issued Action Requests 961000057 to address the assessment team's programmatic observations regarding timely completion of low priority engineering work. The licensee's corrective actions for the observed conditions consisted of (1) providing additional resources to reduce the backlog, (2) review and develop a screening process to eliminate action requests that had little or no value, and (3) periodic future screening of the backlog to ensure that it is managed in a timely fashion.

(d) **Inadequate Setpoint Calculation for Emergency Chilled Water Control Switches - (Licensee Self-Assessment Observation 006)** - The licensee team noted that Setpoint Calculation J-GJA-015, "Setpoints For System Low Flow Ind. Control Switches 2/3FICL6402-1 and 2/3FICL6408-2," did not provide adequate margin to assure the emergency chillers were operable during certain plant events that involve a component cooling water pump start.

Calculation J-GJA-015 calculated the setpoint requirements for an emergency chilled water system chiller trip. The trip was based on low component cooling water flow to the chiller condenser. The licensee team noted that the calculation specified a design flow condition of 5.42 psid and a trip of 2.4 psid. The assessment team further noted that the calculation specified a trip reset value of 3.9 psid with a total loop uncertainty of +/- 2.24 psid. The assessment team determined that the specified maximum reset value could be as high as 6.14 psid when the total loop uncertainty was taken into account. The assessment team concluded that the reset value (6.14 psid) could preclude resetting of the trip and a desired component cooling water pump start under normal design flow conditions (5.42 psid).

The licensee issued Action Request 960900453 to address the team's observations. The licensee performed an operability evaluation for the noted conditions, as part of the action requests, and determined on an interim basis that the emergency chillers remained operable because (1) the total loop uncertainty was overly conservative, and (2) testing had demonstrated that the "chillers never failed to start" because of failure to overcome the reset setpoint and actual total loop uncertainty. The licensee initiated action to determine if the trip function was necessary. The licensee also initiated action to review other calculations to determine if other total loop uncertainty conditions could result in a trip reset problem. Licensee actions are scheduled to be completed by February 1, 1997.

(e) **Inconsistency Between Calculation and Technical Specifications Control Room Atmosphere Ammonia Limit - (Licensee Self-Assessment Observation 008 and 009)** - The licensee team noted that a 98 parts per million ammonia concentration was specified as a control room toxic gas isolation limit in the new licensee-controlled specifications. The team noted that Calculation J-SAA-001, "Toxic Gas Isolation Setpoints," CCN 5, and a letter from engineering to licensing, dated March 28, 1996, designated an allowable limit of 94 parts per million. The assessment team also noted that the earlier Technical Specification limit which was converted to the licensee controlled specification limit was 100 parts per million. The team also noted that Calculation J-SAA-001 determined that the alarm and the control room toxic gas isolation for ammonia concentration should be set at 70 parts per million. The assessment team noted that the toxic gas isolation system loop setpoint drawing, 2/3AR9782, ammonia setpoint was 75 parts per million. The licensee team reviewed earlier Technical Specification Section 3/4.3, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," to

identify any additional similar differences between earlier technical specifications and the new licensee controlled specifications. No additional issues were found.

The licensee issued Action Requests 960900497 and 960901034 to address the team's observations. The licensee performed an operability evaluation for the noted conditions, as part of the action requests, and determined that the toxic gas isolation system was operable based on available design margins resulting from conservative assumptions in the setpoint calculation. A maintenance order was issued and the ammonia concentration setpoint for toxic gas isolation and alarm was reset to 70 parts per million. The licensee controlled specification was being changed to specify a 94 parts per million allowable ammonia concentration for the toxic gas isolation signal.

NRC Inspectors Observations and Findings - The NRC inspectors determined that the seven licensee self-assessment team observations discussed in the section on design control above represented five examples of a failure to correctly translate applicable regulatory requirements and the design basis into specifications, drawings, procedures, and instructions.

The inspectors noted that none of the design control discrepancies discussed above resulted in a determination that a structure, system, or component was inoperable. The inspectors determined that the licensee identified and evaluated the discrepancies, and initiated corrective actions for the discrepancies. The inspectors determined that the licensee's evaluations and corrective actions were reasonable. The inspectors noted that the aggregate design control failure represented by the five discrepancies discussed above, was not a violation that could reasonably have been prevented by corrective actions for a previous violation or licensee finding that occurred within the last 2 years.

The five design control discrepancies discussed above were identified as a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." This licensee-identified violation is being treated as a non-cited violation (50-361,362/9613-02),consistent with Section VII.B.1 of the NRC Enforcement Policy.

(2) 10 CFR 50.59 Evaluations

(a) **Diesel Generator Cross Tie Modification 10 CFR 50.59 Evaluation - (Licensee Self-Assessment Observations 011 and 012)** - The licensee team noted that Design Change Package 2/3-7048.00SE had been issued to provide the capability to manually cross-connect a diesel generator from one unit to the other unit 4.16 kV Class-1E bus. At the time of the licensee's self-assessment, this change had not yet been constructed. The licensee team observed that the 10 CFR 50.59 evaluation for the design did not clearly analyze the condition of a unit in which the design change had been constructed, while waiting for an outage of the second unit for completion of the design change in that unit. In addition, the licensee team observed that the 10 CFR 50.59 evaluation did not clearly address the use of the design

during a 10 CFR 50.54(x) condition. The assessment team considered that additional discussion on the bounds of the 10 CFR 50.59 related to physical installation and actual operations needed more analysis or clarity. In addition, the licensee team questioned whether the impact of installing the 10 CFR 50.54(x) switches into the existing control circuits had been thoroughly addressed in the 10 CFR 50.59 safety evaluation.

The licensee team independently reviewed the elementary diagrams and determined that it would take two deliberate switching actions to cross tie a diesel generator from one unit to the other. In addition, the assessment team determined that normally closed 10 CFR 50.54(x) switch contacts inserted into the existing control circuits used two contacts in parallel. For normally open 10 CFR 50.54(x) switch contacts inserted into the existing control circuits, two contacts in series were used. The licensee team noted that the change in position of any 10 CFR 50.54(x) switch was alarmed and monitored at the standby power system panel. Based on this review, the licensee team concluded that the revised control circuits included adequate design elements to monitor and prevent inadvertent disarming of existing safety features. In addition, the team noted that the design was consistent with other systems in the plant that had override or bypass features.

The licensee issued Action Request 960900875 to address the assessment team's observations. In addition, the licensee issued a letter, dated December 31, 1996, to the NRC that informed the NRC of the proposed modification. However, the licensee did not consider the installation of the modification to be an unreviewed safety question. Rather, the licensee considered the modification to be unique and felt that the intended use of the modification may not be allowed. The letter was being prepared to request NRC approval for the final turnover of the modification for use in a beyond design basis condition.

NRC Inspectors Observations and Findings - The licensee informed the NRC inspectors during this inspection that Design Change 2/3-7048, "Diesel Generator Cross Tie," Revision 0, was currently being installed in Unit 2 during the ongoing Unit 2 outage. The licensee further informed the NRC inspectors that the Unit 3 portion of the design change was planned to be performed during the next Unit 3 refueling outage. The NRC inspectors were also informed by licensee design engineers that mispositioned diesel generator cross tie permissive switches that were being installed in Unit 2 could prevent the automatic load sequencing feature of the Class-1E bus. However, the design engineers also indicated that redundant switches, control room annunciators for switch positions, and administrative controls for use of the cross tie would result in a very low probability that the load sequencing feature of the Class-1E bus would malfunction when needed.

The inspectors reviewed the licensee's 10 CFR 50.59 evaluation for Design Change 2/3-7048 and discussed the evaluation with licensee personnel. The design change added new electrical switches, which bypassed the permissive logic for the diesel generator cross-tie circuit breakers to allow electrical alignment of the Unit 2

diesel generator to Unit 3. The inspectors noted that the misoperation of these switches could prevent the automatic sequencing of loads onto the Class-1E bus, following a loss of normal onsite ac power. The licensee's safety evaluation reasoned that the redundancy of the switches, the control room alarms on the switches, and the administrative controls for the diesel generator cross-tie, precluded the possibility of creating a new accident or increasing the likelihood or consequences of an already reviewed accident from occurring. The automatic sequencing of loads is described in Section 8.3.1.1.3.10 of the San Onofre Unit 2 Updated Final Safety Analysis Report.

In response to the inspectors' concern, the licensee issued Action Requests 961200998 and 961201400 for Units 2 and 3, respectively, to install jumpers and lift electrical leads as required to isolate the safety related portions of the cross-tie design change which could negatively affect plant performance. The license issued a letter to the Office of Nuclear Reactor Regulation dated December 31, 1996, describing the modification. Although the letter indicated that the modification did not represent an unreviewed safety question, the licensee planned to maintain the modification disabled pending further NRC review of the modification. This issue will be followed as an unresolved item pending review by the Office of Nuclear Reactor Regulation (50-361/9613-03).

(3) Corrective Actions

The inspectors determined that the following self-assessment team observations indicated a weakness in the implementation of the corrective actions program.

Self-Assessment Observation 014 - The licensee team reviewed 27 industry experience evaluations performed by the Independent Safety Engineering Group and determined that the evaluations were thorough and complete, and adequately addressed the conditions that were reviewed. However, the assessment team noted that Independent Safety Engineering Group recommended actions for five of the industry experience evaluations either had no established completion dates or had overdue completion dates. The licensee issued Action Request 960900505 to initiate actions to complete the Independent Safety Group recommendations.

Self-Assessment Observations 004 and 022 - The licensee team identified one open item report that was 5 years old, and 99 open low priority action requests that had been assigned to engineering that were over 28 days old. The oldest action request was 255 days old. The team also identified that 47 percent of Nuclear Engineering Department action items reviewed were completed 90, or more, days past the requested date. Twenty-eight percent of the action items reviewed took more than 12 months after the requested due date to complete. The licensee issued Action Request 961000057 to review and initiate action to correct the engineering backlog.

(4) Fire Protection Procedure Compliance

The inspectors determined that the self-assessment team identified the following procedure compliance violations:

- The licensee team determined that Procedure SO23-V-57, "Fire Damper Inspection, Testing and Maintenance," Revision 0, TCN 3, required the surveillance coordinator to maintain a master tracking system to document each fire damper visual inspection, functional test and preventative maintenance. The licensee team determined that the surveillance coordinator was not fulfilling this procedural requirement, but that the cognizant engineer was maintaining the necessary records. The licensee issued Action Request 960901051 to evaluate and correct the noted condition.

- The team determined that Procedure SO123-XIII-21, "Fire Department/ Emergency Services Officers Drills," Revision 3, required drills to be pre-planned to establish training objectives. A fire drill observed by the licensee team had not been reviewed, approved or pre-planned. In addition, the assessment team noted that there was a lack of operations involvement in the drill. The licensee issued Action Requests 960901072 and 960901135 to provide corrective actions for future drills.

- The licensee team determined that Procedure SO123-XIII-52, "Monthly Portable Fire Extinguisher Inspection," required that fire extinguishers be verified to be properly mounted. The team observed that a Unit 2 fire extinguisher located in the intake structure northwest area was detached and "on the ground." The assessment team determined that the discrepancy was originally identified and documented in the procedure on September 21, 1995. However, the team also noted that the surveillance data sheets for the fire extinguisher had documented the condition as satisfactory on three of the last five monthly surveillances. The licensee supplemented an existing and still open Action Request 960401257 to perform a root-cause investigation for the surveillance discrepancies.

  The investigation determined that 10 of 12 of the last surveillances of the fire extinguisher documented the condition as being satisfactory. The investigation determined that nine different inspectors had accepted the condition, but had identified numerous other fire extinguisher discrepancies during the same inspections. The investigation determined that the involved inspectors erroneously concluded that the condition had been accepted, or already had corrective actions planned, and, as such, was satisfactory. The licensee revised the procedure to further clarify the acceptance criteria. The licensee retrained counseled the involved individuals.

The NRC inspectors determined that the licensee identified procedure compliance violations, discussed in the paragraphs above, could not have been reasonably prevented by licensee corrective actions for a previous violation or licensee finding. The NRC inspectors also determined that the procedure compliance violations did

not appear to be willful and were being corrected by the licensee within a reasonable time. This licensee identified procedure compliance violation is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-361/362/9613-04).

c. Conclusion

The inspectors concluded that the licensee's self-assessment team identified examples of weaknesses in design control, 10 CFR 50.59 evaluation, corrective actions, and fire protection procedure compliance. An unresolved item was identified regarding the adequacy of a 10 CFR 50.59 evaluation. Two non-cited violations were identified for design control discrepancies and for fire protection procedure compliance violations.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on December 13, 1996. The licensee acknowledged the findings presented. The licensee did not identify as proprietary any of the information presented to the inspectors during the inspection. The final results of this inspection were discussed with Mr. G. Gibson on January 9 and 14, 1997.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Axline, Compliance Engineer, Licensing
D. Breig, Manager, Station Technical
D. Chiang, Engineer, Station Technical
R. Clark, Manager, Quality Engineering
D. Crozier, Supervisor, Fire Protection at Palo Verde
J. Curran, Project Manager
G. Gibson, Manager, Compliance
J. Huges, Engineer, Station Technical
D. Irvine, Manager, Technical Support
K. Johnston, Manager, Electrical and I/C Engineering
A. Kaneko, Engineer
R. Krieger, Vice President, Nuclear Generation
R. Lee, Supervisor, Nuclear Safety Group
D. Nunn, Vice President, Engineering and Technical Services
D. Pilmer, Project Manager
G. Plumlee III, Supervisor, Regulatory Compliance
V. Powers, Engineer, Quality
J. Rainsberry, Manager, Licensing
S. Root, Supervisor, Integrated Plant Review Engineering
P. Shaffer, Supervisor, Plant Maintenance
A. Sistos, Senior Engineer
K. Slagle, Manager, Nuclear Oversight
D. Stickney, Supervising Engineer
R. St. Onge, Manager
J. Thomas, Senior Engineer, Nuclear Oversight Division
M. Tolson, Supervisor, Fire Protection
D. Tuttle, Supervisor, Station Technical
M. Wharton, Manager, Engineering Design
P. Wohld, Engineer

NRC

J. Sloan, Senior Resident Inspector

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 40501	Licensee Self-Assessments Related to Team Inspections
IP 64704	Fire Protection Program

ITEMS OPENED

Opened

50-361,362/9613-01	IFI	Licensee Self-Assessment of Fire Protection Program Missed Several Attributes.
50-361, 362/ 9613-02	NCV	Five Licensee Identified Design Control Discrepancies, E1.5.b.(1)
50-361/9613-03	URI	Modification to EDG Cross-Tie Permissive Circuitry May Prevent Proper Loading of EDG in an Accident, E.1.5.b.(2)
50-361, 362/ 9613-04	NCV	Licensee Identified Failure to Follow Fire Protection Procedure Requirements, E1.5.b.(4)

ITEMS CLOSED

Closed

50-361, 362/ 9613-02	NCV	Five Licensee Identified Design Control Discrepancies, E1.5 b.(1)
50-361, 362/ 9613-04	NCV	Licensee Identified Failure to Follow Fire Protection Procedure Requirements, E1.5.b.(4)

LIST OF DOCUMENTS REVIEWED

Licensee Procedures

Work Process Procedure S0123-XX-1 ISS 2, "Action Request/Maintenance Order Initiation and Processing," Revision 3

General Procedure S0123-XV-5, "Nonconforming Material, Parts, or Components," Revision 6

Licensee Report

SEA 96-007, "Self-Assessment of Engineering and Fire Protection," November 7, 1996

Action Requests

960900515	960900750	960900743	961000057	960900453	960900497
960901034	960900875	960900505	960901072	960901135	960401257

Note: Portions of numerous licensee procedures, drawings, and calculations referenced in the above noted action requests were also partially used for reference by the inspectors.