UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

March 22, 2000

EA 00-030

Harold B. Ray, Executive Vice President
Southern California Edison Co.
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, California 92674-0128

SUBJECT: NRC ROUTINE INSPECTION REPORT NO. 50-361/00-02; 50-362/00-02

Dear Mr. Ray:

This refers to the inspection conducted on January 23 through February 26, 2000, at the San Onofre Nuclear Generating Station, Units 2 and 3, facility. The enclosed report presents the results of this inspection.

During the 5-week period covered by this inspection, your conduct of activities at the San Onofre facility was generally characterized by safety-conscious operation, sound engineering and maintenance practices, and careful radiological work controls.

Based on the results of this inspection, the NRC has determined that five Severity Level IV violations of NRC requirements occurred. These violations are being treated as noncited violations, consistent with Section VII.B.1.a of the Enforcement Policy and Enforcement Guidance Memorandum 98-002, Revision 2, "Disposition of Violations of Appendix R, Sections III.G and III.L Regarding Circuit Failures." These noncited violations are described in the subject inspection report. If you contest the violation or severity level of these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the San Onofre Nuclear Generating Station, Units 2 and 3, facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be placed in the NRC Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch E
Division of Reactor Projects

Docket Nos.: 50-361
50-362
License Nos.: NPF-10
NPF-15

Enclosure:
NRC Inspection Report No.
50-361/00-02; 50-362/00-02

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-361 50-362
License Nos.:	NPF-10 NPF-15
Report No.:	50-361/00-02 50-362/00-02
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:	January 23 through February 26, 2000
Inspectors:	J. A. Sloan, Senior Resident Inspector J. G. Kramer, Resident Inspector J. J. Russell, Resident Inspector A. B. Earnest, Physical Security Specialist
Approved By:	Linda Joy Smith, Chief, Branch E Division of Reactor Projects

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Units 2 and 3
NRC Inspection Report No. 50-361/00-02; 50-362/00-02

This routine announced inspection included aspects of licensee operations, maintenance, engineering, and plant support. This report covers a 5-week period of resident inspection and the results of a plant support inspection.

Operations

- Operators thoroughly and methodically prepared for and conducted evolutions. Management and supervisors provided close oversight of operational activities. Procedure use and operator communications were generally consistent with written licensee management expectations (Section O1.1).

Maintenance

- Licensee personnel performed maintenance and surveillance activities in a thorough manner with the work package present and in active use. Technicians were knowledgeable and professional. Supervisors and system engineers frequently monitored job progress, and Quality Control personnel were present whenever required by procedure. When applicable, appropriate radiation controls were in place (Sections M1.1 and M1.2).

- A violation of 10 CFR Part 50, Criterion V, occurred because an Operations surveillance procedure used to implement postaccident monitoring instrumentation channel checks contained inappropriate acceptance criteria. Multiple recorders that should have been required for operability were not listed as such. In this respect, the Operations procedure group demonstrated poor attention to the design of the plant. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was in the licensee's corrective action program as Action Request 000101672 (Section M1.3).

Engineering

- A violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," resulted from the failure to electrically separate Class 1E circuits from non-Class 1E circuits. Six components were affected, none of which were required for safe shutdown in the event of a fire. Although electrical separation was not maintained, the components remained operable and capable of performing their intended function. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was in the licensee's corrective action program as Action Request 000101584 (Section E8.1).

Plant Support

- A violation of Technical Specification 5.5.1.1.a occurred when a Maintenance technician removed personal dosimetry while working in a radiologically controlled area. The

potential for a significant unrecorded whole body dose was small due to the low general area radiation levels and the short time the dosimetry was removed. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was in the licensee’s corrective action program as Action Request 000200914 (Section R4.1).

- A violation of paragraph 3.2.4 of the Physical Security Plan occurred when two licensee security officers willfully failed to follow the requirements of Section 6.6.4 of Security Procedure SO123-IV-5.3.3, “Security Processing Facility Search and Inspection,” Revision 6. Specifically, one security officer did not step back and wait for an HP technician to evaluate the cause of a radiation portal monitor alarm and the second officer did not direct the first officer to step back and wait for HP personnel. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a.4 of the NRC Enforcement Policy. The violation was in the licensee's corrective action program as Action Request 990401474 (Section S1.1).

- A violation of 10 CFR Part 50, Appendix R, Section III.G, resulted from the failure of the licensee to ensure that one train of saltwater cooling would remain free of fire damage in the event of a design basis fire because portions of the control circuits did not have the required fire wrapping. This condition resulted from an error in a 1987 design modification. This Severity Level IV violation is being treated as a noncited violation, consistent with Enforcement Guidance Memorandum 98-002, Revision 2, “Disposition of Violations of Appendix R, Sections III.G and III.L Regarding Circuit Failures.” This violation was in the licensee’s corrective action program as Action Request 000101035 (Section F8.1).

Report Details

Summary of Plant Status

Both units operated at essentially 100 percent reactor power during this inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors observed routine and nonroutine operational activities throughout this inspection period. Some of the activities observed included:

- Operator rounds (Units 2 and 3)
- Boric acid makeup tank recirculation (Unit 2)
- Emergency Diesel Generator (EDG) 2G002 fast start prejob briefing (Unit 2)

Operators thoroughly and methodically prepared for and conducted evolutions. Management and supervisors provided close oversight of operational activities. Procedure use and operator communications were generally consistent with written licensee management expectations.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707)

The inspectors observed all or portions of the following work activities:

- Walk-around inspection of EDG 2G003 cylinder liners and fuel injectors (Unit 2)
- EDG 2G003 relief and check valve replacements (Unit 2)
- Charging pump coupling lubrication (Unit 2)
- Partial drainage of reactor coolant pump oil collection Tank 2MT216 (Unit 2)
- Add oil to reactor coolant Pump 2P004 (Unit 2)

b. Observations and Findings

The inspectors found the work performed under these activities to be thorough. All work observed was performed with the work package present and in active use. Technicians were knowledgeable and professional. The inspectors frequently observed supervisors and system engineers monitoring job progress, and Quality Control personnel were present whenever required by procedure. When applicable, appropriate radiation controls were in place.

M1.2 General Comments on Surveillance Activities

a. Inspection Scope (61726)

The inspectors observed portions of the following surveillance activity:

- Weekly start of common diesel fire Pump 2/3P220 (Units 2 and 3)

b. Observations and Findings

The inspectors found the surveillance performed under this activity to be thorough. The surveillance observed was performed with the work package present and in active use. Technicians were knowledgeable and professional. The inspectors observed supervisors and system engineers monitoring job progress, and Quality Control personnel were present whenever required by procedure.

In addition, see the specific discussion under Section M1.3 below.

M1.3 Postaccident Monitoring Instrumentation (PAMI) Channel Checks - Unit 2

a. Inspection Scope (61726)

On February 9, 2000, the inspectors walked down the Unit 2 main control boards. The inspectors reviewed Unit 2 Technical Specification 3.3.11, “Post Accident Monitoring Instrumentation (PAMI)”; the associated Technical Specification bases, and Reference 1 to these bases, “SONGS Units 2 and 3 Regulatory Guide 1.97 Instrumentation Report 90065,” Revision 0, dated October 1, 1992. The inspectors also reviewed the Updated Final Safety Analysis Report (UFSAR), Section 7.5.3.3.2.1, “Qualified Safety Parameter Display System,” Revision 13; Procedure SO23-3-3.35, “PAMI/Safe Shutdown Monthly Checks,” Temporary Change Notice 13-2 and Revision 14, and discussed the condition with Operations and Engineering personnel.

b. Observations and Findings

During the board walkdown, Loop 1 hot leg, wide-range Temperature Recorder TR-0911X1 had an equipment deficiency mode restraint tag dated January 28, 2000, indicating that the recorder was inoperable because it was reading greater than 15 degrees from both associated Loop 1 Temperature Indicator TI-0911X1 and Loop 2 Temperature Indicator TI-0921X2. The inspectors questioned the operators as to why Technical Specification 3.3.11 had not been entered. Technical Specification 3.3.11, Conditions A and B, required that, with one channel of PAMI for hot leg temperature inoperable, the channel be restored within 30 days or a special report be submitted to the NRC within the next 30 days. Because the recorder had only been inoperable for approximately 12 days, the Technical Specification requirement had not been exceeded. The inspectors reviewed the bases for this Technical Specification and noted that, for the Loop 1 channel to be operable, the recorder was required to be operable. Instrumentation Report 90065 listed the recorder as a part of the loop and Technical Specification bases stated that the plant-specific instruments used to fulfil the PAMI requirements were identified in the instrumentation report. Further, Regulatory

Guide 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident,” Revision 2, stated that recording or trending of instrumentation should be provided on at least one channel of PAMI instrumentation (for Type A and/or Category I variables when this trending was needed for operator information or action), and the bases indicated that the PAMI Technical Specification ensured the operability of Regulatory Guide 1.97 Type A and Category I variables. Hot leg temperature is a Category I variable.

On February 11, 2000, the operators changed the status of Recorder TR-0911X1 from an equipment deficiency mode restraint component to a limiting condition for operation action required component and entered the action statement for one channel of hot leg temperature indication being inoperable. The operators had initially used the criterion in Procedure SO23-3-3.35, Temporary Change Notice 13-2, to assess the operability of PAMI with the recorder inoperable. The acceptance criterion in Procedure SO23-3-3.35 was incorrect and caused the error.

10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” states, in part, that activities affecting quality shall be prescribed by procedures appropriate to the circumstances and that these procedures shall include appropriate acceptance criteria. Procedure SO23-3-3.35, Temporary Change Notice 13-2, step 2.2.13, listed the acceptance criterion for PAMI hot leg temperature Channel 0911X1 as the associated recorder or indicator agreeing with the Channel 0921X2 hot leg temperature to within 15 degrees. This acceptance criterion was not appropriate, because the associated recorder was necessary for the channel check to be satisfactory, regardless of the status of the indicator. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (NCV 361; 362/00002-01). This violation was in the licensee’s corrective action program as Action Request (AR) 000101672. This same condition also existed for PAMI channels for cold leg temperature, refueling water tank level, containment temperature, containment water level, containment pressure, and condensate tank level. These channels were Technical Specification 3.3.11 required PAMI channels and also listed recorders in Instrumentation Report 90065 as a part of the applicable loop.

Operation personnel stated that Procedure SO23-3-3.35 had not required operable recorders as listed above for all previous revisions of the procedure that could be verified, including the procedure in effect during the Technical Specification Improvement Program implementation. The inappropriate procedure caused the operators to fail to identify an applicable Technical Specification action, although no action times were identified as being exceeded. As a corrective action, Operations revised the procedure to add a separate table listing all recorders. This table required checking recorders for exhibiting normal trending behavior during performance of the channel check. The licensee also stated that a misunderstanding of which recorders were required for PAMI had caused the error in the procedure.

c. Conclusions

A violation of 10 CFR Part 50, Criterion V, occurred because an Operations surveillance procedure used to implement PAMI channel checks contained inappropriate acceptance

criteria. Multiple recorders that should have been required for operability were not listed as such. In this respect, the Operations procedure group demonstrated poor attention to the design of the plant. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was in the licensee's corrective action program as AR 000101672.

III. Engineering

E8 Miscellaneous Engineering Issues (92700)

E8.1 (Closed) Licensee Event Report 361; 362/2000-002-00: loss of physical independence of electrical system - outside of design basis.

This report involved licensee design engineers' identification that some Class 1E electrical control circuits were not electrically isolated from non-Class 1E circuits. The circuits identified were: the control circuit for Unit 2(3) volume control tank Outlet Valve 2(3)V0227B; the control circuit for Unit 2(3) refueling water storage tank to charging pump Suction Valve 2(3)V0227C; one channel of loss of voltage sensing circuitry for Unit 2(3) Class 1E 4.16 KV Buses 2(3) A04 and 2(3) A06; and the control circuits for Units 2(3) boric acid Makeup Pumps 2(3) P174 and P175. The inspectors walked down various cabinets which contained some of the affected circuits with licensee engineers and Maintenance personnel. All circuits identified as affected were 120 volts ac or 125 volts dc, and those portions of various elementary diagrams used were accurate with respect to the actual wiring and cabinet configuration. The lack of isolation devices identified in the licensee event report was verified. The licensee planned to install electrical isolation devices during the upcoming refueling outages.

10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications and drawings. The Units 2 and 3 UFSAR, Section 8.1.4.3.14.J, states, in part, that non-Class 1E circuits shall be electrically isolated from Class 1E circuits except where analyzed to demonstrate that the associated Class 1E circuits are not degraded below an acceptable level in the event of a failure or fault of the non-Class 1E circuit or device. Contrary to this, electrical elementary diagrams for those Units 2 and 3 components listed above did not correctly translate the electrical isolation description in the UFSAR. These components were not electrically isolated from "X" Train, or non-Class 1E wiring. No analysis had been performed to demonstrate that this degradation was acceptable. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (NCV 361; 362/00002-02). This violation was in the licensee's corrective action program as AR 000101584.

The inspectors assessed the operability of the affected components and determined that the components remained operable but did not conform to standards specified in the UFSAR (Regulatory Guide 1.75, "Physical Independence of Electric Systems," in regard to separation of Class 1E and non-Class 1E components). The components were operable because they remained capable of performing their intended functions and met the guidance for operability contained in NRC Inspection Manual, Part 9900:

Technical Guidance, “Operable/Operability: Ensuring the Functional Capability of a System or Component,” dated October 31, 1991.

IV. Plant Support

R4 Staff Knowledge and Performance in Radiological Protection and Control

R4.1 Dosimetry Removed While Inside a Radiologically Controlled Area (RCA) - Unit 2

a. Inspection Scope (71750)

The inspectors observed a person not wearing personal dosimetry in an RCA. The inspectors reviewed Procedures SO123-VII-20.11, “Access Control Program,” Revision 4, and SO123-XV-24, “Security Responsibilities of Site Personnel,” Revision 5, and AR 000200914. The inspectors discussed the observation with Health Physics (HP), Maintenance, and Security supervision.

b. Observations and Findings

On February 14, 2000, the inspectors observed maintenance activities on a charging pump in a low dose area inside the RCA. Upon entering the space, the inspectors observed the personal dosimetry and security badge for an individual performing coupling lubrication placed to the side of, and a couple feet away from, the individual. The inspectors asked the technician performing the maintenance why the dosimetry and security badge were not being worn. The technician responded that he did not want to get grease on them, and that they were within his line of sight and were only off for a couple of minutes. The technician continued to work on the coupling and after a few more minutes placed the dosimetry and security badge back around his neck. The inspectors estimated the dosimetry had been off of the individual for approximately 5 minutes. The inspectors continued to observe the maintenance activity.

Upon exiting the RCA, the inspectors informed HP technicians about the observation. HP initiated an investigation into the event. HP placed an access hold on the worker and performed an external dose evaluation to ensure that the appropriate dose was assigned to the worker. In addition, the licensee initiated a radiological observation report and AR 000200914.

The inspectors discussed the observation with a Maintenance supervisor. The supervisor indicated that the worker received coaching on proper placement of the dosimetry as a result of the event. An electronic mail message was sent to all machine shop employees on the day of the event that addressed the requirement to wear dosimetry and security badges at all times unless specific direction allowed otherwise. A briefing with all machinists was conducted the following day to discuss the event. The inspectors observed that the Maintenance department corrective actions were prompt.

The inspectors discussed the security aspects of the event with Security supervision. Security supervision indicated that, since the badge was not lost within the protected area such that someone else could have used it, no security event log issue existed.

Procedure SO123-XV-24, step 6.8.5 states, in part, that personnel should affix the security badge to a neck band on the front of the outermost garment. Security indicated that, in this instance, there was not a regulatory issue; however the badge removal was a poor work practice which could result in a lost or stolen badge.

The inspectors had observed two other Maintenance personnel in the immediate vicinity of the worker who had removed his dosimetry. One technician was working on the motor/gear reducer coupling and another technician, standing next to the individual who removed the badge, was providing support to both maintenance activities. The inspectors concluded that both of these individuals had ample opportunity to inform the technician that removal of the dosimetry was in contradiction to the radiation exposure permit (REP).

Unit 2 Technical Specification 5.5.1.1.a requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, recommends procedures for access control to radiation areas, including a radiation work permit system. Procedure SO123-VII-20.11 in a note after step 6.5 states, in part, that on REP sign-up, the individual agrees to comply with the REP instructions. REP 200114 instructions state, in part, that the PD-1 (dosimetry) must be worn on the outermost clothing unless directed by HP. Contrary to the above, a Maintenance technician signed onto REP 200114 and failed to wear his dosimetry on the outermost clothing while performing work inside an RCA. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (NCV 361/00002-03). This violation was in the licensee’s corrective action program as AR 000200914.

c. Conclusions

A violation of Technical Specification 5.5.1.1.a occurred when a Maintenance technician removed personal dosimetry while working in an RCA. The potential for a significant unrecorded whole body dose was small due to the low general area radiation levels and the short time the dosimetry was removed. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was in the licensee’s corrective action program as AR 000200914.

S1 Conduct of Security and Safeguards Activities

S1.1 Access Control - Personnel and Packages

a. Inspection Scope (81700)

The personnel access control program was inspected to determine compliance with the requirements of the Physical Security Plan. The inspectors reviewed Office of Investigation Case 4-1999-022.

b. Observations and Findings

Paragraph 3.2.4 of the Physical Security Plan, Revision 62, stated "SCE has established a management system that provides for the development, revision, implementation, and enforcement of security procedures. Section 6.6.4 of Security Procedure SO123-IV-5.3.3, "Security Processing Facility Search and Inspection," Revision 6 stated, "if the radiation monitoring portal alarms at the protected area exit point, instruct the individual to step back through the alarming monitor and wait for an HPT [Health Physics Technician] to assess the cause of the alarm."

On April 18, 1999, a contaminated cell phone case was discovered by a security officer in the personnel access point exit lanes. The cell phone case was found under a chair on the protected area side of the radiation portal monitors. The security officer who found the cell phone case attempted to leave the protected area with the case, causing the radiation portal monitor to alarm. Instead of stepping back and waiting for an HP technician, the security officer bypassed the portal monitor and tossed the cell phone case to a second security officer. The first security officer told the second officer to throw the cell phone case in the trash. The second officer complied with the first officer's request and placed the case in a trash can outside the protected area. Neither security officer called HP as required by the plant procedures. The NRC Office of Investigation determined that the two plant security officers willfully failed to follow the procedure. The failure to follow the requirements of Section 6.6.4 of Security Procedure SO123-IV-5.3.3, "Security Processing Facility Search and Inspection," Revision 6, was a violation of paragraph 3.2.4 of the Physical Security Plan (NCV 361; 362/00002-04). This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a.4 of the NRC Enforcement Policy. The violation was in the licensee's corrective action program as AR 990401474.

c. Conclusions

A violation of paragraph 3.2.4 of the Physical Security Plan occurred when two licensee security officers willfully failed to follow the requirements of Section 6.6.4 of Security Procedure SO123-IV-5.3.3, "Security Processing Facility Search and Inspection," Revision 6. Specifically, one security officer did not step back and wait for an HP technician to evaluate the cause of a radiation portal monitor alarm and the second officer did not direct the first officer to step back and wait for HP personnel. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a.4 of the NRC Enforcement Policy. The violation was in the licensee's corrective action program as AR 990401474.

F8 Miscellaneous Fire Protection Issues (92700)

F8.1 (Closed) Licensee Event Report 361; 362/2000-001-00: saltwater cooling (SWC) pump control circuits do not meet fire protection design basis.

On January 20, 2000, the licensee identified cables in the SWC pipe tunnel that were missing a portion of their fire wraps required to meet 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,"

Section III.G, “Fire Protection of Safe Shutdown Capability.” The licensee reported the condition to the NRC Operations Center the same day (Log Number 36608).

The licensee determined that a fire protection design change performed in approximately 1987 for the SWC pump control circuits did not provide an isolation fuse as intended. As a result, a fire occurring at the location of the component cooling water heat exchanger discharge valves could cause a ground fault to the power cable of the SWC pump discharge valve solenoid. In addition, the fire could cause an electrical fault at the pump interlock circuit that, combined with the discharge valve cable fault, could trip the dc control power breaker to the pump. This would de-energize the pump start circuit and the pump would not be able to start from the second point of control without additional operator action to first restore the dc control power to the Units 2 and 3 Train B SWC pumps’ breakers.

The licensee performed several corrective actions. The licensee ensured that an hourly fire watch was posted in the SWC pipe tunnel and in each unit’s SWC pump room. The licensee initiated AR 000101035 to evaluate the event. The licensee planned to modify the affected circuits to eliminate the potential vulnerability.

The licensee evaluated the safety consequence of the event. The licensee determined that existing plant procedures were in place to allow operators to manually start an SWC pump if needed. The affected areas were protected by fire suppression equipment and the probability of fire was low. Therefore, the licensee concluded that the safety consequence was minimal.

10 CFR Part 50, Appendix R, Section III.G, requires, in part, that fire protection features shall be provided for components important to safe shutdown and these features shall be capable of limiting fire damage. The failure of the licensee to ensure that one train of SWC would remain free of fire damage in the event of a design basis fire was a violation of Appendix R. This Severity Level IV violation is being treated as a noncited violation, consistent with Enforcement Guidance Memorandum 98-002, Revision 2, “Disposition of Violations of Appendix R, Sections III.G and III.L Regarding Circuit Failures.” (NCV 361; 362/00002-05). This violation was in the licensee’s corrective action program as AR 000101035.

V. **Management Meetings**

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the exit meeting on February 29, 2000. A subsequent exit was held with members of licensee management on March 7, 2000, to discuss the inspection results in Section S1.1. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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G. Cook, Supervisor, Regulatory Compliance
J. Fee, Manager, Maintenance
J. Hirsch, Manager, Chemistry
R. Krieger, Vice President, Nuclear Generation
J. Madigan, Manager, Health Physics
D. Nunn, Vice President, Engineering and Technical Services
A. Scherer, Manager, Nuclear Regulatory Affairs
K. Slagle, Manager, Nuclear Oversight
T. Vogt, Units 2 and 3 Plant Superintendent, Operations
R. Waldo, Manager, Operations

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 81700	Physical Security Program
IP 92700:	On Site Licensee Event Report Review

ITEMS OPENED AND CLOSED

Opened and Closed

361; 362/00002-01	NCV	inappropriate acceptance criteria for instrumentation channel operability (Section M1.3)
361; 362/00002-02	NCV	loss of electrical separation (Section E8.1)
361/00002-03	NCV	individual removed dosimetry while inside an RCA (Section R4.1)
361; 362/00002-04	NCV	failure to follow security requirements (Section S1.1)
361; 362/00002-05	NCV	circuits do not meet fire protection design basis (Section F8.1)

Closed

361;362/2000-002-00	LER	loss of RG 1.75 separation - outside of design basis (Section E8.1)
361; 362/2000-001-00	LER	SWC pump circuit controls do not meet fire protection design basis (Section F8.1)

LIST OF ACRONYMS USED

AR	action request
CFR	Code of Federal Regulations
EDG	emergency diesel generator
HP	Health Physics
NRC	Nuclear Regulatory Commission
PAMI	post accident monitoring instrumentation
RCA	radiologically controlled area
REP	radiation exposure permit
SCE	Southern California Edison
SWC	saltwater cooling
UFSAR	Updated Final Safety Analysis Report