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

March 13, 2001

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 INTEGRATED INSPECTION REPORT 50-361/01-02; 50-362/01-02

Dear Mr. Ray:

On February 17, 2001, the NRC completed an inspection at your San Onofre Nuclear Generating Station, Units 2 and 3. The enclosed report documents the inspection findings which were discussed on January 19 and February 20, 2001, with Mr. R. Krieger and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Circumstances affecting the financial viability of Southern California Edison Co. have continued to evolve during this inspection period. Actions have been initiated by the state of California and Southern California Edison Co. to address the impacts of these financial challenges. The NRC has exercised communications channels to better understand your planned and implemented actions, especially as they relate to your responsibility to safely operate the San Onofre reactors. NRC inspections, to date, have confirmed that you are operating these reactors safely and that public health and safety is, thus far, assured.

In response to these conditions of economic stress, there will be two differences in how the Region communicates its inspection findings. First, we will continue the 6-week periodicity of our integrated inspection reports (the other reactors in Region IV will be transitioning to a quarterly report frequency, with the exception of Diablo Canyon). Second, the description of the scope of the individual inspection activities will be significantly more detailed. This is being done to keep the public more fully informed of the breadth and depth of the NRC's inspection and oversight activities.

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of

NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

Charles S. Marschall, Chief
Project Branch C
Division of Reactor Projects

Dockets: 50-361
50-362
Licenses: NPF-10
NPF-15

Enclosure:
NRC Inspection Report
50-361/01-02; 50-362/01-02

cc w/enclosure:
Chairman, Board of Supervisors
County of San Diego
1600 Pacific Highway, Room 335
San Diego, California 92101

Alan R. Watts, Esq.
Woodruff, Spradlin & Smart
701 S. Parker St. Suite 7000
Orange, California 92868-4720

Sherwin Harris, Resource Project Manager
Public Utilities Department
City of Riverside
3900 Main Street
Riverside, California 92522

R. W. Krieger, Vice President
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, California 92674-0128

David Spath, Chief
Division of Drinking Water and
Environmental Management
P.O. Box 942732
Sacramento, California 94234-7320

Michael R. Olson
Sr. Energy Administrator
San Diego Gas & Electric Company
P.O. Box 1831
San Diego, California 92112-4150

Ed Bailey, Radiation Program Director
Radiologic Health Branch
State Department of Health Services
P.O. Box 942732 (MS 178)
Sacramento, California 94327-7320

Steve Hsu
Radiologic Health Branch
State Department of Health Services
P.O. Box 942732
Sacramento, California 94327-7320

Mayor
City of San Clemente
100 Avenida Presidio
San Clemente, California 92672

Truman Burns/Robert Kinosian
California Public Utilities Commission
505 Van Ness, Rm. 4102
San Francisco, California 94102

Robert A. Laurie, Commissioner
California Energy Commission
1516 Ninth Street (MS 31)
Sacramento, California 95814

Douglas K. Porter
Southern California Edison Company
2244 Walnut Grove Avenue
Rosemead, California 91770

Dwight E. Nunn, Vice President
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, California 92674-0128

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(JAS7)**
Branch Chief, DRP/C **(CSM)**
Senior Project Engineer, DRP/C **(DPL)**
Section Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**

<u>Only inspection reports</u> to the following:
Scott Morris **(SAM1)**
NRR Event Tracking System **(IPAS)**
SONGS Site Secretary **(SFN1)**
Dale Thatcher **(DFT)**

R:_SO23\SO2001-02RP-JAS.wpd

RIV:SRI:DRP/C	RI:DRP/C	C:DRS/EMB	C:DRP/C	
JASloan	JGKramer	CEJohnson	CSMarschall	
T - CSMarschall	Not available	/RA/	/RA/	
3/12/01		3/12/01	3/13/01	

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-361 50-362
License Nos.:	NPF-10 NPF-15
Report No.:	50-361/01-02 50-362/01-02
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 7 through February 17, 2001
Inspectors:	J. A. Sloan, Senior Resident Inspector J. G. Kramer, Resident Inspector L. E. Ellershaw, Senior Reactor Inspector
Approved By:	Charles S. Marschall, Chief, Project Branch C

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: NRC’s Revised Reactor Oversight Process

SUMMARY OF FINDINGS

San Onofre Nuclear Generating Station
NRC Inspection Report 50-361/01-02; 50-362/01-02

IR05000361-01-02, IR05000362-01-02: 01/07/01-02/17/01, Southern California Edison, San Onofre Nuclear Generating Station, Units 2 & 3; Integrated Resident and Regional Report; maintenance risk assessment and emergent work evaluation.

Resident inspectors and regional reactor inspectors conducted the inspection. This inspection identified one finding of No Color. The significance of the issues is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process.” Findings for which the Significance Determination Process does not apply are indicated by “No Color” or by the severity level of the applicable violation.

Cornerstone: Miscellaneous

- During control element drive mechanism control system diagnostic testing, instrumentation and controls technicians (not a licensed operator or senior operator) used a test card and raised a control element assembly one step and then returned it back to its original position, therefore manipulating the controls of the facility. This was a violation of 10 CFR 50.54(i) which requires, in part, that the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator. This failure constitutes a violation of minor significance and is not subject to formal enforcement action. This violation is in the licensee’s corrective action program as Action Request 001101366.

  This issue had no credible impact on safety and was not evaluated using the Significance Determination Process, because the licensee’s actions caused only a negligible reactivity change, while the reactor was shut down with significant shutdown margin. However, the issue is being documented because the associated technical information relates directly to an issue of potential generic interest (Section 1R13.2).

Report Details

Summary of Plant Status:

Unit 2 operated at essentially 100 percent power throughout this inspection period.

Unit 3 began this inspection period operating in Mode 6. Following the completion of refueling activities, operators performed a reactor startup, entering Mode 2 on January 31, 2001, and Mode 1 on February 2. On February 3, operators synchronized the generator to the grid, ending the Cycle 11 refueling outage. Approximately 11 hours later, at 3:14 p.m. on February 3, an automatic reactor trip occurred from 39 percent power (Section 4OA3.1 and NRC Inspection Report 50-362/01-05). On February 5, the operators completed cooling the unit down to Mode 5. The unit remained in Mode 5 for the remainder of the inspection period.

1. **REACTOR SAFETY**
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed a periodic visual inspection to determine the operational status of seals and barriers, sumps and drains, and level alarms and to identify the existence of other potentially unanalyzed internal flooding hazards. The inspectors reviewed the following area susceptible to internal flooding:

- Emergency core cooling systems pump rooms (Safety Equipment Building Rooms 2, 5, and 15). The inspection included a review of portions of Calculation M-120.015, "Plant Flooding Analysis Review," Revision 6, and a discussion of the calculation with the cognizant Nuclear Design engineer (Unit 3).

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination (NDE) Activities

a. Inspection Scope

The inspectors observed the licensee's NDE contractor personnel perform the inservice inspection program specified examinations listed below.

System	Inservice Inspection Identification Number	Report Number	Examination Method
Auxiliary Feedwater	03-046-110	301-11/MT-002 301-11/UT-004	Magnetic Particle Ultrasonic

Auxiliary Feedwater	03-046-130	301-11/MT-002 301-11/UT-004	Magnetic Particle Ultrasonic
Auxiliary Feedwater	03-046-140	301-11/MT-002 301-11/UT-004	Magnetic Particle Ultrasonic
Auxiliary Feedwater	03-046-150	301-11/MT-002 301-11/UT-004	Magnetic Particle Ultrasonic
Auxiliary Feedwater	03-046-160	301-11/MT-002 301-11/UT-004	Magnetic Particle Ultrasonic

During the performance of each examination, the inspectors verified that the correct NDE procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors reviewed the NDE certification packages of the contractor personnel and verified that they had been properly certified in accordance with ASME Code requirements. The inspectors also verified that indications revealed by the examinations were compared against the ASME Code-specified acceptance standards and appropriately dispositioned.

b. Findings

No findings of significance were identified.

.2 Unit 3 ASME Code Repair and Replacement Activities

a. Inspection Scope

The inspectors reviewed the following repair and replacement work orders for the current Unit 3 outage that were subject to ASME Code Section XI requirements. The inspectors verified that the correct preservice or inservice Section XI-specified examinations were identified and included in the work orders. In addition, the inspectors reviewed the welding procedure specifications and their applicable procedure qualification records and determined them to be in accordance with the requirements of Section IX of the ASME Code. The inspectors reviewed the welding filler material control records on the work orders and verified that the welding filler material had been properly issued and controlled.

Work Order Package	Component Identification
WO 00111753000	Reactor Coolant System Hot Leg (Loop 1) Thermowell
WO 99100779001	Centrifugal Charging Pump Discharge Check Valve
WO 01010907000	High Pressure Safety Injection Isolation Valve

The inspectors reviewed the radiography procedure, technique and reader sheets, and radiographic film for the following acceptable welds.

Component Identification	Radiography Report
Centrifugal Charging Pump Discharge Check Valve S3-1208-MU-067	3RT-011-01 (Weld BH-R1) 3RT-004-01 (Weld BG)
Centrifugal Charging Pump Discharge Check Valve S3-1208-MU-069	3RT-005-01 (Weld BA) 3RT-006-01 (Weld BB)
High Pressure Safety Injection Isolation Valve 3HV-9330	3RT-012-01 (Weld SC) 3RT-013-01 (Weld SB)

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors performed a detailed review of the sample of action requests (ARs) listed in the attachment. The corrective action documents reviewed were all initiated during the year 2000 to identify and correct problems related to the inservice inspection program type issues identified below:

- Dimensional discrepancy between snubber design and as-found condition,
- Discrepancy between pipe support drawing, as-found condition, and applicable stress calculation,
- Failure to follow procedure during postweld heat treatment phase of steam generator tube sleeve installation,
- Steam generator tube end repair was performed on the wrong tube,
- Post-torque calibration checks for plug installations in the steam generator hot leg side were outside the limits specified in the procedure,
- A steam generator tube sleeve was not accessible for welding, and
- Defects were identified on the inlet and outlet flange seating surfaces of spare pressurizer safety valves.

The review was conducted to ascertain that the licensee's corrective action program was identifying performance issues within the inservice inspection program. Further review assessed the effectiveness of cause determination, the appropriateness of applied corrective action, the adequacy of transportability review and identification of generic issues, and the overall corrective action program effectiveness in addressing previously identified administrative issues affecting the inservice inspection program.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the implementation of the requirements of the Maintenance Rule (10 CFR 50.65) for the following system and component:

- 480 volt breaker for Charging Pump 2P192 (AR 001200113). The inspection included a review of how generic breaker issues were being addressed by the licensee (Unit 2).

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

.1 Maintenance Risk Assessments

a. Inspection Scope

The inspectors reviewed the effectiveness of risk assessment and risk management and verified that the licensee entered the appropriate risk category for the following activity:

- Loss of offsite ac power source to Unit 2 vital 4 kV busses as a result of the nonvital 4 kV breaker fire in Unit 3 (Unit 2).

b. Findings

No findings of significance were identified.

.2 Control Element Drive Mechanism Control System (CEDMCS) Troubleshooting - Unit 2

a. Inspection Scope

The inspectors reviewed the emergent maintenance activities associated with CEDMCS in support of Unit 2 control element assembly (CEA) testing in Maintenance Order 00090602. The inspectors reviewed ARs 001101366 and 010102717, initiated by the licensee, that questioned aspects of reactivity manipulation.

b. Findings

During CEDMCS diagnostic testing, instrumentation and controls technicians (not a licensed operator or senior operator) used a test card and raised a CEA one step and then returned it back to its original position, therefore manipulating the controls of the facility. This was a violation of 10 CFR 50.54(i).

The inspectors reviewed AR 001101366 that described a potential reactivity manipulation involving a TE190N test card. The AR documented that when the lift high current switch on the test card was operated, the CEA lifted 0.75 inches and dropped back to its original position when the switch returned to the neutral position. The licensee evaluated this evolution and concluded that this activity was required to be performed by a licensed operator, except when the CEA was fully inserted or fully withdrawn and the differential CEA worth was essentially zero. The licensee concluded that, at the extremes of CEA travel, moving the CEA a single step would not directly affect the reactivity of the reactor and that instrumentation and controls technicians may perform the testing with the knowledge and consent of a licensed operator. The inspectors disagreed with the licensee's evaluation and concluded that a licensed operator would always be required to operate the test card whenever the card moved a CEA, even if the CEA worth was essentially zero.

10 CFR Part 50, Appendix A, Criterion 26, "Reactivity Control System Redundancy and Capability," requires, in part, that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods. The second system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes.

10 CFR 50.54(i) states that, "except as provided in §55.13 of this chapter, the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator . . . ." 10 CFR 50.2 defines controls, when used with respect to nuclear reactors, as apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor. The inspectors concluded that the TE190N test card was an apparatus that directly affects a reactivity control system and has a direct effect on reactivity of the reactor.

10 CFR 50.54(i) states, in part, that the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator. Contrary to the above, on several occasions in November 2000, the licensee permitted instrumentation and controls technicians (not a licensed operator or senior operator) to performed diagnostic testing using a TE190N test card, ultimately manipulating the controls of the facility. This failure to comply with the regulatory requirement constitutes a violation of minor significance and is not subject to formal enforcement action. This violation is in the licensee's corrective action program as AR 001101366.

This issue had no credible impact on safety and was not evaluated using the Significance Determination Process, because the licensee's actions caused only a negligible reactivity change, while the reactor was shut down with significant shutdown margin. However, the issue is being documented because the associated technical information relates directly to an issue of potential generic interest (FIN 361/2001002-01).

The inspectors noted that there are a number of apparatus and mechanisms that can affect the reactivity or power level of the reactor by changing the temperature or pressure of the reactor coolant system (RCS) (feedwater heaters and auxiliary feedwater). Similarly, there are other apparatus and mechanisms that can affect reactivity and power by addition of neutron poisons (high pressure safety injection) or the erroneous tripping of protective bistables. The simple fact that these apparatus and mechanisms may affect reactivity or power under normal or abnormal operating conditions does not make them reactivity control systems. The use of the TE190N test card which caused CEA movement was clearly an apparatus that affected a reactivity control system and has a direct effect on reactivity of the reactor.

The inspectors reviewed AR 010102717. The AR documented a CEDMCS trouble alarm that was received in the control room and the subsequent actions performed by a nonlicensed individual. The nonlicensed individual manipulated a manual transfer switch and transferred the CEA from the lower gripper to the upper gripper and, as part of system design, the CEA moved up approximately 1/16 of an inch. Based on the previous example, the licensee was concerned that another unauthorized manipulation of controls had just occurred. The inspectors concluded that in this case an unauthorized manipulation of controls did not occur since the manual transfer aspect of the CEDMCS was not designed nor intended to change reactivity.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

On February 1, 2001, the inspectors observed operators perform the Unit 3 reactor startup from the Cycle 11 refueling outage. The inspectors reviewed Procedure SO23-3-1.1, "Reactor Startup," Revision 22, Attachment 4, "Calculation of Critical Boron Concentration for Desired Critical CEA Height;" and Attachment 5, "Calculation of Inverse Count Rate Ration (1/M) Plot."

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the operability evaluations documented in the following ARs to ensure that operability was properly justified:

- High Pressure Safety Injection Pump 3P018 inboard bearing oil leak (AR 000901619). The inspectors discussed the operability assessment documented in the AR with a Station Technical supervisor (Unit 3).

- Damaged/missing insulation on Low Pressure Safety Injection Pump 2P015 discharge piping (AR 010102138). The inspection included a review of the licensee's basis for determining that the additional heat load in the pump room was within the capacity of the room cooling systems (Unit 2).

- Error in the CEFLASH computer code for small break loss of coolant accident analysis (AR 010102719). The inspectors reviewed the vendor notification and the licensee's assessment to determine if the error could potentially result in a current inoperable condition, and to determine the basis by which the licensee determined that the error was bounded within the current plant design (Units 2 and 3).

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

On February 8, 2001, the inspectors reviewed operator workarounds to evaluate their cumulative effect on the operators' ability to implement abnormal or emergency operating procedures. The review included the ability of operators to deal with the many additional actions required as a result of the loss of nonsafety-related electrical power in Unit 3 after the reactor trip on February 3. Examples of those workarounds included:

- charging the emergency diesel generator air start system receivers
- pumping various sumps
- using flashlights where plant lighting was not functioning
- monitoring component cooling water pump bearings
- monitoring saltwater cooling pump bearings
- monitoring various transformer temperatures

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and/or reviewed postmaintenance testing for the following activities to verify that the test procedures and activities adequately demonstrated system operability:

- Main Steam Safety Valve 3PSV8412 replacement. The inspection included observation of the actual test performance, using the hydroset test method in accordance with Attachment 10 of Procedure SO23-I-2.5, “Testing of Main Steam Safety Valve Surveillance,” Temporary Change Notice 12-3 and a review of the test results (Unit 3).

- Turbine-Driven Auxiliary Feedwater Pump 3P140 Trip-Throttle Valve 3HV4716 repairs. The inspectors reviewed Maintenance Order 01011965, which documented the repairs, and Procedure SO123-I-6.16, “Valve Inspection, Lubrication, Packing Replacement and Gland Adjustment,” Revision 3, that documented postmaintenance testing (Unit 3).

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Refueling Outage - Unit 3

a. Inspection Scope

The inspectors periodically observed plant conditions to verify that safety systems and support systems, including electrical distribution, were properly aligned, with defense-in-depth commensurate with the outage risk control plan. The inspectors periodically verified that the shutdown cooling system configuration was consistent with Technical Specification requirements and that the RCS inventory was adequately controlled. The inspectors also verified that containment closure requirements were met.

The inspectors observed and verified refueling activities. The inspectors verified that fuel handling operations and containment penetration closure were performed in accordance with Technical Specifications and approved procedures, and verified that the location of the fuel assemblies, including new fuel, was tracked during the core offload and reload.

The inspectors performed a containment cleanliness and material readiness tour prior to the unit's entry into Mode 3.

The inspectors also observed the following evolutions:

- Core offload from the refueling machine on January 9, 2001

- Diverse level monitoring system and refueling water level instrumentation valve lineup (prior to midloop entry)

- Midloop entry on January 24

- Swapping of shutdown cooling pumps and heat exchangers

- Draining the refueling cavity

- Reactor restart on February 1

b. Findings

No findings of significance were identified.

.2 Forced Outage - Unit 3

a. Inspection Scope

In addition to the event response activities (Section 4OA3.1), the inspectors periodically monitored operational status of the shutdown cooling system and the vital and nonvital electrical power distribution systems.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and/or reviewed documentation for the following surveillance tests to verify that the system and its components were capable of performing their intended safety functions and to assess their operational readiness:

- Turbine-Driven Auxiliary Feedwater Pump 3P140 (Unit 3)
- Emergency Diesel Generator 3G002 largest load reject test (Unit 3)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary plant modification to verify that the safety functions of safety systems were not affected:

- CEA 40 position indication system (Unit 2).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors verified the accuracy of data reported by the licensee from the period of September 1 through December 31, 2000, for the following performance indicators to ensure that the performance indicator color was correct:

- BI01 RCS Specific Activity - Unit 2
- BI01 RCS Specific Activity - Unit 3

The inspectors reviewed AR 010200833 associated with the RCS specific activity performance indicator for Unit 2. The inspectors reviewed the chemistry results documented by the licensee in the San Onofre Nuclear Generating Station Test Equipment Management System to verify that the highest activity was reported. The inspectors reviewed NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 Nonvital 4 kV Breaker Fire and Reactor Trip - Unit 3

a. Inspection Scope

The inspectors responded to the Unit 3 reactor trip from 39 percent power that occurred on February 3, 2001, at approximately 3:14 p.m. The event was initiated by a fire in the nonsafety-related 4.16 kV switchgear located in the Unit 3 turbine building. The fire resulted in the Unit 3 reserve auxiliary transformers relaying out, causing the reactor coolant pump (6.9 kV) busses and the safety-related 4.16 kV busses to automatically transfer to the Unit 2 reserve auxiliary transformers. All other nonsafety-related power in Unit 3 was lost. All annunciators were lost in Unit 3 for 14 minutes during the event. The only heat removal path available was by releasing steam through the atmospheric dump valves.

The inspectors maintained continuous onsite coverage of Unit 3 from approximately 4:45 p.m. on February 3 until shutdown cooling was initiated at 1:34 a.m. on February 5. The inspectors monitored operator response to the various problems resulting from not having nonsafety-related power available, including nonfunctioning sump pumps and lighting.

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 362/2000-003-00: inoperable containment dome air circulator. The licensee voluntarily reported that one of two dome air circulating fans in one train had failed and that the fan may have been inoperable during power operations for greater than the 30 days allowed by Technical Specification 3.6.8. The licensee satisfied the actions required by Technical Specification within 30 days from the time of discovery, so no violation occurred. Additionally, the licensee determined that only one fan was required in each train to perform the safety function and revised the Technical Specification Bases accordingly. The issue was in the licensee's corrective action program as AR 001000003. This was a minor issue and was closed.

.3 (Closed) LERs 361; 362/2000-014-00, -01: missed inservice test (IST) for safety injection tank (SIT) nitrogen valves.

a. Inspection Scope

The inspectors reviewed: LERs 361; 362/2000-014-00, -01; AR 001000900; Procedure SO123-0-23, "Control of System Alignments," Revision 9; and Procedure SO23-3-3.31.10, "Miscellaneous Systems Valve Testing - Offline," Revision 6, Attachment 15, "Safety Injection Tank Nitrogen Supply Check Valve Tests."

b. Findings

The licensee failed to include the SIT nitrogen supply check valves in the IST program and therefore never tested the check valves. This was a violation of 10 CFR Part 50, Appendix B, Criterion XI.

On November 3, 2000, the licensee determined that the SIT nitrogen supply check valves were not in the IST program and had not been tested. The licensee performed an operability assessment for the nitrogen supply check valves in Unit 3 and determined that the SITs were capable of performing their intended safety function. Unit 2 was in a refueling outage with SITs not required to be operable. The licensee tested the Unit 2 SIT check valves on November 4, 2000, and the Unit 3 valves on January 2, 2001, satisfactorily.

The licensee documented in the LER that, during a small break loss of coolant accident, the SITs are required to maintain pressure for a period of time until they are needed for injection. Because SIT pressure has an accident mitigation function, SIT pressurization has been assured by designing the SIT nitrogen supply system to ASME Code

Section III, Class 2, requirements, as specified by Regulatory Guide 1.26. The SIT nitrogen supply check valves are a Code boundary for the SITs and are required to be tested.

10 CFR Part 50, Appendix B, Criterion XI, requires, in part, that a test program shall be established to assure that all testing required to demonstrate that components will perform satisfactorily in service is identified and performed in accordance with written procedures. Contrary to the above, the licensee's test program did not include the SIT nitrogen supply check valves and therefore the valves had never been tested. This failure constitutes a violation of minor significance and is not subject to formal enforcement action.

.4 <u>(Closed) LER 361; 362/2000-015-00</u>: missed Technical Specification surveillances on postaccident monitoring instrumentation containment isolation valves. The licensee determined that the affected valves had been successfully tested on a 24-month frequency by the IST program and that the ISTs satisfied the surveillance scope, but were not performed at the required 18-month periodicity. This was a violation of Technical Specification Surveillance Requirement 3.3.11.4. The valves were confirmed to have been tested satisfactorily as of November 9, 2000. This violation was in the licensee's corrective action system as ARs 001100550 and 001100552. This failure constitutes a violation of minor significance and is not subject to formal enforcement action.

.5 <u>(Closed) LER 361/2000-016-00</u>: manual start of a containment emergency cooler, engineered safety features actuation. The licensee reported that operators manually started the containment emergency cooling units after Normal Containment Chiller 2ME201 failed on December 31, 2000. The manual actuation was intended to control containment temperatures until the other normal chiller could be placed into service. The emergency cooling units were not placed in service in order to perform their engineered safety features safety function. The actuation was not a violation of NRC requirements. This was a minor issue and was closed.

4OA5 <u>Other</u>

<u>Financial Status</u>

The NRC has exercised communications channels to better understand the licensee's planned and implemented actions, especially as they relate to safely operating the reactors. The inspectors have specifically reviewed the following on a weekly basis:

- Staffing of on-shift operating personnel and the number of qualified Emergency Response Organization responders
- Corrective maintenance removed from the 12 week rolling schedule and the corrective maintenance backlog
- Reduction in safety or risk important outage activities

- Reduction in planned risk important modifications or enhancements
- Emergency Response Facility and siren availability
- Generator voltage loading
- Impact of rolling blackouts of the grid on offsite power availability

NRC inspections and inspector observations, to date, have confirmed that the licensee operated the units safely and that public health and safety was, thus far, assured.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Krieger and other members of licensee management at exit meetings on January 19 and February 20, 2001. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether or not any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Allen, Supervisor, Reliability Engineering
C. Anderson, Manager, Site Emergency Preparedness
D. Brieg, Manager, Station Technical
J. Fee, Manager, Maintenance
J. Hirsch, Manager, Chemistry
R. Krieger, Vice President, Nuclear Generation
J. Madigan, Manager, Health Physics
A. Mahindrakar, Inservice Inspection Engineer
M. McBrearty, Engineer, Nuclear Oversight and Regulatory Affairs
A. Meichler, Supervisor, Codes and Welding
D. Nunn, Vice President, Engineering and Technical Services
R. Richter, Supervisor, Fire Protection Engineering
A. Scherer, Manager, Nuclear Oversight and Regulatory Affairs
S. Shaw, Supervisor, Nuclear Services
M. Short, Manager, Site Technical Support
T. Vogt, Operations Plant Superintendent, Units 2 and 3
R. Waldo, Manager, Operations

NRC

John Pellet, Chief, Operations Branch, Region IV

Others

C. Thompson, Authorized Nuclear Inservice Inspector, Factory Mutual Insurance Company

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

361/20001002-01	FIN	nonlicensed person manipulated controls of the facility (Section 1R13.2)

Previous Items Closed

362/2000-003-00	LER	inoperable containment dome air circulator (Section 4OA3.2)
361; 362/2000-014-00	LER	missed IST for SIT nitrogen valves (Section 4OA3.3)

361; 362/2000-014-01	LER	missed IST for SIT nitrogen valves (Section 4OA3.3)
361; 362/2000-015-00	LER	missed Technical Specification surveillances on postaccident monitoring instrumentation containment isolation valves (Section 4OA3.4)
361/2000-016-00	LER	manual start of a containment emergency cooler emergency safety features actuation (Section 4OA3.5)

LIST OF ACRONYMS USED

AR	action request
CEDMCS	control element drive mechanism control system
CEA	control element assembly
CFR	Code of Federal Regulations
IST	inservice test
LER	licensee event report
NCV	noncited violation
NRC	Nuclear Regulatory Commission
RCS	reactor coolant system
SIT	safety injection tank

DOCUMENTS REVIEWED

Procedures

SO123-IN-1, “Inservice Inspection Program,” Revision 4
SO123-XVII, “Inservice Inspection Program Implementation,” Revision 8
SO23-XVII-3.1, “Inservice Inspection of Class 1 Components and their Supports,” Revision 3
SO23-XVII-3.2, “Inservice Inspection of Class 2 Components and their Supports,” Revision 2
SO23-XVII-3.3, “Inservice Inspection of Class 3 Components and their Supports,” Revision 2
SO23-XXVII-20.47, ”Magnetic Particle Examination,” Revision 01
SO23-XXVII-30.5, “Ultrasonic Examination of Ferritic Piping Welds,” Revision 0
SO23-XII-9.401, “Radiographic Examination,” Revision 1

Drawings

90064, “Second 10-Year Inservice Inspection Program Plan,” Revision 5

Miscellaneous

Manual, “SONGS RFO-11 ISI, LMT Personnel Certifications”

Welding Procedure Specification WPS 8-GT, “Manual GTAW of P-Number 8 austenitic stainless steel alloys using IN308L/ER308L or IN316L/ER316L filler metals,” Revision 1, and supporting Procedure Qualification Records 5 and 68

Welding Procedure Specification 43-GT-1, “Manual GTAW of P-43 Nickel Base Alloys using Inconel 52 filler,” Revision 0, and supporting Procedure Qualification Records 33 and 36

Action Requests

AR 001002408
AR 001002523
AR 001100377
AR 001100482
AR 001100565
AR 001100590
AR 001100634
AR 001201010

ATTACHMENT 2

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.