ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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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:	June 27 through August 7, 1999
Inspectors:	J. A. Sloan, Senior Resident Inspector J. G. Kramer, Resident Inspector J. J. Russell, Resident Inspector G. A. Pick, Senior Project Engineer
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ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Units 2 and 3
NRC Inspection Report No. 50-361/99-09; 50-362/99-09

This routine announced inspection included aspects of licensee operations, maintenance, engineering, and plant support. This report covers a 6-week period of resident inspection and inspection of the corrective action program.

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).

- Operators in the simulator effectively used the emergency operating procedures to mitigate plant damage and radioactive release during an emergency preparedness drill. In one instance, control board awareness of a partially stuck open steam generator atmospheric dump valve was not timely; however, the licensee adequately addressed this untimely response during a postdrill critique (Section O5.1).

- The corrective action program was effectively identifying, resolving, and preventing problems. Specific strengths identified were the low threshold of reporting problems into the corrective action program and the priority of problem resolution (Section O7.1).

Maintenance

- Maintenance and surveillance activities were performed in a manner consistent with licensee procedures and work control documents. All work observed was performed 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).

- Maintenance personnel did not proactively initiate a procedure change for a procedure inconsistency identified during a containment hydrogen monitor calibration. A lower-tier calibration procedure did not agree with a higher-tier work control procedure. The lower-tier procedure did not allow personnel to adjust as-found amplifier gain, when the as-found amplifier gain was within acceptance limits but not centered. Maintenance personnel adjusted the amplifier gain, as allowed by the work control procedure; however, they took no action to change the calibration procedure to avoid confusion in the future (Section M1.3).

- Equipment in safety-significant areas of Units 2 and 3 was free from excessive leaks, and no significant material discrepancies were observed. One 6-foot stepladder was not properly secured, which the licensee promptly corrected. Minor lighting discrepancies were also identified and promptly corrected (Section M2.1).

- The material condition of the Unit 3 emergency diesel generator cooling water systems was generally good, except for components located under deck plate grating, which were externally corroded. Cooling water system drawings and valve alignments were generally accurate, except that startup strainers were not installed in the plant and one valve was incorrectly identified in the valve alignment procedure (Section M2.2).

Engineering

- A Station Technical engineer demonstrated good attention to the saltwater cooling system in identifying abnormal accumulator air pressure during a routine system walkdown. The operability assessment and the initial declaration of Technical Specification 3.0.3 entry were conservative, and the immediate corrective actions were prompt and thorough (Section E2.1).

- Although appropriately addressing the technical issues associated with equipment deficiencies, personnel demonstrated weaknesses in documenting evaluation results in action requests or following procedural guidance for the content of evaluations for cause in all five action requests randomly sampled out of a population of approximately 400 (Section E7.1).

- A violation of Technical Specification 3.3.3.9 (pre-1996) and Technical Specification 5.5.1.1.a (post-1996) resulted from incorrect waste gas system oxygen monitor setpoints. This Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The instrument total loop uncertainty had not been calculated until 1994, and the calculation misapplied the total loop uncertainty such that the setpoints were not adjusted to account for the uncertainty. The licensee determined that the analytical (safe) limit for oxygen concentration was 5 percent and that at no time could this value have been reached without an alarm. This violation was in the licensee's corrective action program as Action Requests 980900715 and 990700022 (Section E8.1).

- A violation of Technical Specification 3.7.7 was identified as the result of a design error that rendered the component cooling water noncritical loop isolation valves inoperable under limited conditions. The licensee determined that the increases in the core damage frequency and large early release frequency were both 4.6E-7/year. The low probability of occurrence of both the initiating condition (seismic event) and the specific failures required to create the inoperability rendered the safety significance low. This Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. This violation was in the licensee's corrective action program as Action Request 990200839 (EA 99-187) (Section E8.2).

- The NRC found the licensee's design basis assessment of the refueling water storage tank outlet valves to be incorrect. The NRC concluded that the valves, in their degraded condition, would not perform all the Updated Final Safety Analysis Report design basis requirements. Although the valves did not meet their design basis, the plant was not outside its design basis and the event was not reportable (Section E8.4).

Report Details

Summary of Plant Status

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

Unit 3 operated at essentially 100 percent reactor power throughout this inspection period, except for a power reduction to 90 percent on July 10-12 to repair a heater drain pump discharge valve.

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:

- Shift turnover (Units 2 and 3)
- Align primary makeup water to supply both units (Unit 3)

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.

O5 Operator Training and Qualification

O5.1 Simulator Observation During Emergency Preparedness Drill

a. Inspection Scope (71707, 71750)

On July 28, 1999, the inspectors observed control room operators in the simulator during an emergency preparedness drill and observed a postexercise critique held with the operators and licensee evaluators.

b. Observations and Findings

The scenario was as follows: A reactor coolant pump sheared shaft resulted in an automatic reactor trip. After the trip, a Steam Generator (SG) 2E089 tube rupture developed, and later a loss of offsite power occurred. Emergency Diesel Generator (EDG) 2G003 failed after starting. The SG 2E088 atmospheric dump valve failed shut, and the SG 2E089 atmospheric dump valve failed partially open. Reactor fuel failure resulted in a radioactive release through the failed open SG 2E089 atmospheric dump valve, necessitating a General Emergency classification. Operators were forced to use the ruptured, faulted SG 2E089 to cool down the reactor coolant system. The drill terminated when personnel were able to restore EDG 2G003 to operation and to shut the SG 2E089 atmospheric dump valve.

The operators effectively used the emergency operating instructions to mitigate the scenario. The operators took approximately 12 minutes to diagnose the SG tube rupture and an additional 23 minutes to isolate the ruptured SG. These times were relatively short and indicated good diagnostic skills and control board manipulation of components on the part of the operating crew, given the multiple equipment failures that occurred. However, it took the operators 14 minutes, from the time they noted decreasing pressure in SG 2E089, to observe from the control board indication that the SG 2E089 atmospheric dump valve was partially stuck open, causing the pressure decrease. This time was relatively long. Consequently, in this instance, control board awareness was not timely. The operating crew was aware of the relatively long time they took in recognizing the partially stuck open atmospheric dump valve and discussed this during the postdrill critique, which demonstrated effective crew training.

c. Conclusions

Operators in the simulator effectively used the emergency operating procedures to mitigate plant damage and radioactive release during an emergency preparedness drill. In one instance, control board awareness of a partially stuck open SG atmospheric dump valve was not timely; however, the licensee adequately addressed this untimely response during a postdrill critique.

O7 Quality Assurance in Operations

O7.1 Corrective Action Program

a. Inspection Scope (40500)

The inspectors reviewed the licensee programs used to identify and correct problems at the facility. The review focused on the following areas: (1) threshold of reporting adverse conditions, (2) priority of problem resolution, (3) monitoring to ensure program effectiveness, (4) program measurement or trending, (5) personnel understanding of the program, (6) repetitive problem identification and resolution, and (7) followup of cited and noncited violations.

b. Observations and Findings

b.1 Threshold of Reporting Adverse Conditions

The inspectors reviewed the threshold for documenting issues in the action request (AR) system (corrective action process). The licensee initiated 15,513 ARs during the 7-month period from November 1, 1998, through May 31, 1999, of which 5,190 were designated as important to safety. On July 9, 1999, the licensee distributed a memorandum to all nuclear organization workers explaining the purpose of the AR system and emphasizing management's expectation that there was no threshold for creating an AR. In addition, the resident inspectors reviewed the recently generated ARs on a daily basis and concluded that personnel had a low reporting threshold.

The inspectors reviewed Procedures SO123-OR-1, "Operational Experience Review Program," Revision 5, and SO123-XIV-5.5, "Operating Experience Report: Sharing Industry Information," Revision 0, and discussed the operating experience review program (OERP) with the nuclear network coordinator. The purpose of the OERP is to review events and information gained during the operation of the San Onofre units and other utilities' units. The OERP places industry data with the appropriate organizations for evaluation and implementation of lessons learned.

The inspectors attended operational experience review committee meetings. The committee addressed industry events and component failures by assigning the issues to specific organizations to further evaluate them for applicability or initiated ARs to address known potential generic issues. The inspectors reviewed the ARs generated from the OERP and concluded that the threshold for entering issues into the AR process was low. The inspectors reviewed five of the ARs and noted that the issues were being appropriately addressed.

b.2 Priority of Problem Resolution

The inspectors attended several AR committee meetings to observe the disposition of ARs and reviewed Procedure SO123-XX-1, "Action Request/Maintenance Order Initiation and Processing," Revision 10. The committee reviewed ARs to determine the required action to address the issues by generating assignments in the AR. The assignments were used to identify, evaluate, resolve, and track the issues. For equipment deficiencies, the committee assigned an urgency code based on a work priority matrix. The inspectors determined that the AR committee appropriately generated assignments in the ARs to prioritize work and resolve problems.

b.3 Monitoring to Ensure Program Effectiveness

The effectiveness of corrective actions was verified through several programs: the AR corrective action followup assignment, site experience reviews during significant event evaluations, Nuclear Oversite audits, corrective action program metrics, On-Site Review Committee activities, and Nuclear Safety Group reviews. The inspectors reviewed: (1) the biannual effectiveness of corrective actions audit reports dated November 23, 1998, and June 2, 1999; (2) quarterly divisional self-assessment reports; and (3) quarterly station performance reports. The inspectors concluded that licensee processes adequately monitored the implementation of the corrective action program.

b.4 Program Measurement or Trending

The licensee used performance indicators (metrics) on a divisional and site level as a means of tracking performance. The metrics measured: the timeliness of issue identification, the rate of self-identified (line organization) issues, the timeliness of issue resolution, and the effectiveness of issue resolution. In addition, the licensee tracked the corrective action program issues backlog. The inspectors reviewed the backlog and determined that the licensee was appropriately resolving the issues.

b.5 Program Understanding

The inspectors assessed the corrective action program understanding of licensee personnel. The inspectors found that the personnel interviewed were knowledgeable regarding the AR process. Based on review of the ARs, the inspectors observed that ARs were being generated by various disciplines and levels within the organization. The inspectors concluded that licensee personnel had an appropriate understanding of the corrective action program.

b.6 Repetitive Problem Identification and Resolution

Procedure SO123-XV-50.39, "Root Cause Program Standards and Methods," Revision 1, required that the evaluators review industry and site experience and determine if the current event is a repeat event and, if so, evaluate the adequacy and timeliness of the previous corrective actions. This process was used when addressing significant events as required by Levels 1 and 2 event reports and root cause evaluations. In addition, the licensee generated corrective action followup assignments for significant adverse conditions to determine if the corrective actions taken had effectively prevented recurrence. The process implemented to prevent repetitive issues was appropriate.

b.7 Followup of Cited and Noncited Violations

The licensee had received approximately 50 violations (cited and noncited) from previous NRC inspections covering the period of January 1998 through June 1999. The inspectors reviewed seven violations, not previously reviewed by the NRC, to determine if the violations had been entered into the corrective action program and if they had been resolved, or were being resolved, in a timely manner commensurate with their significance. The inspectors determined that all the violations reviewed had been entered into the corrective action program and were being appropriately addressed.

c. Conclusions

The corrective action program was effectively identifying, resolving, and preventing problems. Specific strengths identified were the low threshold of reporting problems into the corrective action program and the priority of problem resolution.

O8 Miscellaneous Operations Issues (90712)

O8.1 (Closed) Licensee Event Report (LER) 362/1999-003-00: manual reactor trip because of loss of main feedwater.

This issue was discussed in NRC Inspection Report 50-361; 362/99-06, Section O1.4. The licensee planned to implement appropriate corrective actions. The inspectors did not identify any additional issues.

O8.2 (Closed) LER 362/1999-004-00: manual reactor trip (emergency safety features actuation) because of feedwater control valve opening.

This issue was discussed in NRC inspection Report 50-361; 362/99-06, Section O1.5. The licensee planned to implement appropriate corrective actions. The inspectors did not identify any additional issues.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance Activities

a. Inspection Scope (62707)

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

- Add oil to reactor coolant Pump 2P004 (Unit 2)
- Preserve Train B Chiller ME335 chilled water outlet valve (Units 2 and 3)
- Refurbish motor-driven Firewater Pump 2/3P222 (Units 2 and 3)
- Troubleshoot EDG 3G003 Automatic Voltage Regulator B (Unit 3)
- Replace EDG 3G003 diesel fuel oil storage tank Level Detector 3LI5906 (Unit 3)

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 all or portions of the following surveillance activities:

- Monthly test of turbine-driven Auxiliary Feedwater Pump 2P140 (Unit 2)
- High pressure turbine stop and governor valve testing (Unit 2)
- Testing of high and low pressure stop governor valves (Unit 3)
- EDG 3G003 monthly start (Unit 3)

b. Observations and Findings

The inspectors found all surveillances performed under these activities to be thorough. All surveillances observed were 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.

In addition, see the specific discussions of surveillances observed under Section M1.3.

M1.3 Containment Hydrogen Monitor Calibration - Unit 2

a. Inspection Scope (61726)

On July 6, 1999, the inspectors observed Instrument and Control technicians performing portions of Procedure SO23-II-1.12, "Surveillance Requirement Containment Post LOCA Hydrogen Monitoring System Channel B Channel Calibration," Revision 12.

b. Observations and Findings

Procedure SO23-II-1.12, Section 6.4, provided guidance for calibration of the hydrogen monitor pressure signal conditioning card. The hydrogen monitor uses an input of containment pressure when deriving hydrogen concentration. Step 6.4.2 directs the technicians to record the voltage at Bistable 13 of the Plant Protection System (PPS). This bistable provides containment pressure input for the PPS and is separate from the hydrogen monitor pressure sensing circuitry. Step 6.4.1 directs the technicians to record the voltage at the hydrogen monitor pressure card output. The PPS voltage is then normalized to the hydrogen analyzer voltage, using an equation, and the two voltages are compared. If the voltages do not agree within 0.05 volts direct current, the procedure directs that a calibration check be performed on the hydrogen monitor pressure signal card. If this calibration check is not within the acceptance criteria, then the technicians would adjust the gain on the hydrogen monitor pressure signal card within acceptance criteria and to within tolerance of the PPS voltage.

During the surveillance on July 6, 1999, the PPS voltage and the hydrogen monitor voltage were not within tolerance. The technicians checked the calibration of the hydrogen monitor pressure card and found the gain within the acceptance criteria. Since the procedure did not allow the hydrogen monitor pressure signal card gain to be adjusted if it was found within tolerance, the technicians contacted Maintenance supervision for guidance (the procedure directed steps to change the gain be marked N/A if the as-found voltage readings were within tolerance). Maintenance supervision determined Procedure SO123-I-1.3, "Work Activity Guidelines," Temporary Change Notice 6-2, specified that, if as-found values were within the specified acceptance criteria, adjustments to improve accuracy were allowed. The technicians adjusted the hydrogen monitor pressure card gain to bring the PPS voltage and the hydrogen monitor voltage closer.

After performing the surveillance, Maintenance personnel failed to initiate a change to reconcile Procedure SO23-II-1.12 with the higher-tier work control procedure and avoid confusion in the future. The inspectors found that Maintenance personnel were not proactive in this regard. In response to this finding, the licensee generated AR 990701308 in order to change Procedure SO23-II-1.12.

c. Conclusions

Maintenance personnel did not proactively initiate a procedure change for a procedure inconsistency identified during a containment hydrogen monitor calibration. A lower-tier calibration procedure did not agree with a higher-tier work control procedure. The lower-tier procedure did not allow personnel to adjust as-found amplifier gain, when the as-found amplifier gain was within acceptance limits but not centered. Maintenance personnel adjusted the amplifier gain, as allowed by the work control procedure; however, they took no action to change the calibration procedure to avoid confusion in the future.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 General Plant Walkdown - Units 2 and 3 (71707)

The inspectors walked down accessible safety-significant areas of Units 2 and 3. Equipment in these areas was free from excessive leaks and no significant material discrepancies were observed. One 6-foot stepladder was not properly secured, which the licensee promptly corrected and documented in AR 9907001352. Minor lighting discrepancies were also identified and promptly corrected.

M2.2 EDG Cooling Water Systems Walkdown - Unit 3

a. Inspection Scope (37551, 62707, 71707)

The inspectors walked down the Unit 3 EDGs 3G002 and 3G003 cooling water systems. The inspectors reviewed portions of Piping and Instrument Drawings (P&IDs) 40110A through 40110D, Revisions 16, 24, 21, and 20, respectively; Isometric Drawings S3-2420-ML-055, Revision 9, and S3-2420-ML-042, Revision 9; pipe stress Calculation M2420-055-A, dated April 16, 1980, and the associated pipe hanger drawings; Updated Final Safety Analysis Report (UFSAR), Section 9.5.5, "Diesel Generator Cooling Water System;" and Procedures SO23-2-13, "Diesel Generator Operation," Revision 18, and SO23-3-3.23, "Diesel Generator Monthly Test," Revision 15.

b. Observations and Findings

b.1 Material Condition

The cooling water piping and valves located between the radiators and diesel engines, and associated with the expansion tanks, were free of surface rust; however, the pipe, flanges, and drain valves located under the deck grating adjacent to the diesel engines had significant surface corrosion, although wall thickness and structural integrity were not affected. The radiator fan motor casings and power cabling conduit had significant exterior surface corrosion. In response to these observations, the licensee initiated AR 990701169 to evaluate preservation plans. On July 7, 1999, during EDG 3G003 operation for monthly surveillance, the inspectors noted that cooling water Temperature Gauge 3TI5969C was loose and slowly spinning around. The licensee generated AR 990700885 to tighten the gauge. The inspectors found that the material condition of the cooling water system was generally good, except for components located under deck plate grating. In addition, the inspectors noted good housekeeping in the emergency diesel building.

b.2 Drawings

The P&IDs and isometric drawings indicated that temporary strainers were installed in the return and supply piping from the EDGs. The inspectors identified that the as-built plant configuration had orifice spacer plates or spacer plates (the inspectors were unable to determine which from visual observation) installed in place of strainers. The licensee initiated AR 990700603 to correct the applicable drawings. The inspectors then asked Station Technical personnel about the size of the orifices, if orifices were installed, because the inspectors found no mention of these orifices in the UFSAR or on applicable drawings. The licensee was unable to rigorously determine the size of all the orifices installed and generated AR 990701613 to research orifice size. Metal tags attached to some of the orifice tabs indicated 4.9-inch openings, while the applicable isometric drawings indicated 8-inch spacers. The inspectors noted that cooling water flow rate had been sufficient and had not caused excessive engine temperatures to develop during monthly EDG tests. The licensee planned on positively confirming the dimensions of each spacer plate.

P&ID 40110ASO3 showed Drain Valve D-096 located on a strainer casing on the pipe between Radiator E546 and EDG 3G002. Licensee personnel verified that the nameplate for this valve, which was located under deck grating adjacent to the diesel, was D-96. The inspectors observed that Procedure SO23-2-13, Attachment 3, described this valve as MR086; however, the designation should have been MR096. In response, the licensee ensured that the valve was closed (its normal position), confirmed that personnel had checked the position of the valve during the most recent valve alignment remotely without verifying the valve nameplate, and determined that no one noted any discrepancy between the valve identification in the valve alignment and the valve nameplate.

Personnel did not properly implement Procedure SO23-2-13, step 2.1.1.50 because personnel had checked Valve S22420MR086 as closed; however, no valve with this designation existed. This failure resulted in a violation of Technical Specification 5.5.1.1.a; however, this deficiency constituted a violation of minor significance and was not subject to formal enforcement action. This failure is being documented, however, because failing to positively identify valves while performing valve alignments could result in improper valve alignments of risk-significant systems going unnoticed. This violation was in the corrective action program as AR 990700951.

b.3 Design Consideration

The inspectors noted the following instances in which design limitations or descriptions were not fully translated into applicable procedures.

UFSAR Section 9.5.5.2.2 specified, with the EDG in standby, the cooling water temperature was maintained between 100 and 157°F. However, Procedure SO23-3-3.23, Attachment 2, specified a range from 100 to 170°F. Actual cooling water temperatures, with the Unit 3 EDGs in standby on July 19, 1999, were between 140 and 152°F. In response to this concern, engineers stated that the elevated temperature permissible in Procedure SO23-3-3.23 allowed for the higher temperatures expected immediately after EDG shutdown. The inspectors still considered that, since the procedure guidance could be applied to the EDGs after the engines had cooled, an abnormally hot, standby EDG might go unnoticed.

UFSAR Section 9.5.5.2.1.5 stated that expansion tank level was maintained such that free volume in the tank allowed room for cooling water to expand after heating up during EDG operation. Procedure SO23-3-3.23, Attachment 2, specified greater than or equal to ½ of full level with no upper limit established. On July 19, 1999, actual Unit 3 EDG expansion tank levels were between ½ and ¾ full. The inspectors noted that Procedure SO23-2-13 did prescribe that expansion tanks be filled to approximately ½ full. The inspectors considered that, without an upper limit on level, routine monitoring of the expansion tank level would not be an effective barrier to prevent an abnormally high level.

The licensee continued to evaluate these design considerations at the end of this inspection report period. Because actual EDG temperature and expansion tank level were within UFSAR assumptions, the inspectors had no immediate safety concerns.

c. Conclusions

The material condition of the Unit 3 EDG cooling water systems was generally good, except for components located under deck plate grating, which were externally corroded. Cooling water system drawings and valve alignments were generally accurate, except that startup strainers were not installed in the plant and one valve was incorrectly identified in the valve alignment procedure.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Saltwater Cooling (SWC) Valve Operability - Unit 2

a. Inspection Scope (37551, 71707)

The inspectors reviewed the circumstances surrounding an entry into Technical Specification 3.0.3 after operators declared Valve 2HV6203, SWC Pump 2P114 discharge isolation, inoperable.

b. Observations and Findings

On July 15, 1999, the Station Technical engineer for the SWC system observed low air pressure in the emergency air accumulator for Valve 2HV6203. The pressure was 50 psig and the nominal pressure was 80 psig. Valve 2HV9303, Train A containment emergency sump outlet, had been removed from service for breaker maintenance. Operators entered Technical Specification 3.0.3 at 3:45 p.m., as a result of both trains of the emergency core cooling system being inoperable. The Technical Specification 3.0.3 condition was exited at 4:15 p.m., when Valve 2HV9303 was returned to service. Operators placed the alternate Train B SWC pump into service and then exited Technical Specification 3.7.8 (one train of SWC inoperable).

The system engineer determined that the low pressure resulted from an internal fault in the 4-way solenoid valve that supplied air to the actuator. The licensee repaired the actuator and initiated a root cause evaluation. Station Technical performed an operability assessment, documented in AR 990700734, that concluded that Valve 2HV6203 was inoperable in the as-found condition. The valve was open, which is its safety position; however, because of the degraded condition of the 4-way solenoid valve, the licensee could not provide assurance that Valve 2HV6203 would remain open during a seismic event.

Further review performed during the reportability assessment, also documented in AR 990700734, determined that the valve would remain open with instrument air failed and the accumulator tank depressurized. This condition had been specifically evaluated and incorporated into the design basis document for the SWC system. The licensee therefore concluded that the Train B SWC system had remained operable and that a Technical Specification 3.0.3 condition was not actually entered.

c. Conclusions

A Station Technical engineer demonstrated good attention to the SWC system in identifying abnormal accumulator air pressure during a routine system walkdown. The operability assessment and the declaration of Technical Specification 3.0.3 entry were conservative, and the immediate corrective actions were prompt and thorough.

E7 Quality Assurance in Engineering Activities

E7.1 Engineering Performance During Resolution of Corrective Action Program Issues

a. Inspection Scope (40500)

The inspectors reviewed ARs 981100216, 990101913, 990201844, 990300184, and 990302156, and Procedure SO123-XX-1, "Action Request/Maintenance Order Initiation and Processing," Revision 10.

b. Observations and Findings

The inspectors reviewed ARs 981100216 and 990101913 associated with component cooling water (CCW) heat exchanger issues and identified documentation deficiencies. To address the documentation weaknesses, the licensee initiated additional assignments, a nonconformance report, and an event report to AR 990101913.

The inspectors reviewed AR 990302156, which addressed questions concerning an actuator spring pack for a containment isolation valve. The AR included an evaluation for cause (EEC) assignment. The inspectors concluded that the EEC did not include the recommendations of Procedure SO123-XX-1, such as documenting corrective actions to prevent recurrence and documenting previous occurrences to determine if an adverse trend exists that should be monitored. In addition, the inspectors identified that the EEC documented an incorrect cause of the problem. The licensee corrected the inaccurate EEC documentation.

The inspectors reviewed AR 990201884, which addressed auxiliary feedwater check valve testing. The inspector identified weaknesses associated with the performance of the unreviewed safety question screening criteria. The licensee corrected the screening criteria deficiencies and concluded that a formal safety evaluation was not required.

The inspectors reviewed AR 990300184 associated with high vibrations of a SWC pump. The EEC addressed the cause of the high vibration and the corrective actions, but did not document previous occurrences to determine if an adverse trend exists that should be monitored as recommended by Procedure SO123-XX-1. The licensee acknowledged the deficiency.

c. Conclusions

Although appropriately addressing the technical issues associated with equipment deficiencies, personnel demonstrated weaknesses in documenting evaluation results in ARs or following procedural guidance for the content of evaluations for cause in all five ARs randomly sampled out of a population of approximately 400.

E8 Miscellaneous Engineering Issues (92700, 92903)

E8.1 (Closed) LER 361; 362/1998-019-00: oxygen monitor exceeds Technical Specifications limits.

a. Inspection Scope (92700)

The inspectors reviewed the LER, Nonconformance Report 980900715, and Calculation J-SLA-012, "Waste Gas System Hydrogen and Oxygen High Concentration Alarm Setpoints," Revision 1, dated October 3, 1996. The inspectors discussed the calculation and the setpoint calculation program with Nuclear Engineering Design personnel.

b. Observations and Findings

In the LER, the licensee reported that the waste gas system oxygen monitor setpoint had exceeded the limits of Technical Specifications 3.11.2.5 and 3.3.3.9 (pre-1996). The setpoint also exceeded the limits of Licensee Controlled Specifications 3.3.107 and 3.7.111 (post-1996), contrary to Technical Specification 5.5.1.1.a. The cause of the condition was the failure to include total loop uncertainty (TLU) in establishing the setpoints until 1994 and applying the TLU incorrectly in calculations since 1994.

The licensee was required to maintain the waste gas system oxygen concentration at less than 2 percent (by volume) whenever the hydrogen concentration exceeded 4 percent. The licensee determined that the hydrogen concentration had essentially always exceeded 4 percent. However, the oxygen monitor setpoint had been 2 percent since initial plant licensing, and oxygen concentrations could have been as high as 2.82 percent without an alarm annunciating.

The licensee determined that the analytical (safe) limit for oxygen concentration was 5 percent and that at no time could this value have been reached without an alarm. The licensee did not determine whether or not the oxygen concentration had actually exceeded 2 percent. However, the licensee subsequently adjusted the oxygen monitor setpoints to ensure that an actual oxygen concentration of 2 percent by volume would not be exceeded.

Technical Specification 3.3.3.9 required that the oxygen monitors shall be operable with their alarm/trip setpoints set to ensure that the limits of Technical Specification 3.11.2.5 were not exceeded. Even though the licensee did not determine whether or not the limit was actually exceeded, the licensee determined that the setpoint was not set to ensure that the limit would not be exceeded. Likewise, Licensee Controlled Specification 3.3.107 requires that the setpoint be set to ensure that the limits of Licensee Controlled Specification 3.7.111 are not exceeded. Failure to meet this requirement was a violation of Technical Specification 5.5.1.1.a (post-1996) and Technical Specification 3.3.3.9 (pre-1996). This Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy (NCV 361; 362/99009-01). This violation was in the licensee's corrective action program as ARs 980900715 and 990700022.

The standard setpoint calculation format applied the TLU such that the actual setpoint value would not exceed the analytical limit. In response to the inspectors' questions, Nuclear Engineering Design personnel determined that the setpoint calculation program did not include specific controls to ensure that the TLU was properly applied in a manner that was consistent with the Technical Specifications context. Although the licensee had recently performed six audits of setpoint calculations and performed extensive reviews of design information as part of its response to a 10 CFR 50.54(f) letter from the NRC, these activities did not specifically address the proper application of TLU to the setpoint calculations. While the Nuclear Engineering Design personnel believed that the oxygen monitor setpoint error was an isolated occurrence, they had not sampled other similar calculations to confirm that the programmatic vagueness had not resulted in other errors. The licensee planned to upgrade the setpoint calculation program to provide specific guidance on proper application of the TLU and to review a sample of other calculations to confirm that this was an isolated error.

c. Conclusions

A violation of Technical Specification 3.3.3.9 (pre-1996) and Technical Specification 5.5.1.1.a (post-1996) resulted from incorrect waste gas system oxygen monitor setpoints. This Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The instrument TLU had not been calculated until 1994, and the calculation misapplied the TLU such that the setpoints were not adjusted to account for the uncertainty. The licensee determined that the analytical (safe) limit for oxygen concentration was 5 percent and that at no time could this value have been reached without an alarm. This violation was in the licensee's corrective action program as ARs 980900715 and 990700022.

E8.2 (Closed) LER 361; 362/1999-003-00: CCW noncritical loop isolation valves inoperable because of previously unrecognized instrument air configuration anomaly (EA 99-187).

The licensee identified and corrected a design error that rendered the CCW noncritical loop isolation valves inoperable under limited conditions, as discussed in NRC Inspection Report 50-361; 362/99-04. Specifically, a seismic event could depressurize the instrument air and nitrogen supply lines to these air-operated valves, and the check valves in these supply lines would prevent venting air off the pilot valves, preventing air from the local emergency air receivers from actuating the isolation valves. This condition had existed since 1995 when the licensee installed the check valves in conjunction with adding the nitrogen supply lines.

In the LER, the licensee identified that both trains of CCW could be lost if one train was already out of service at the time of the seismic event. With one train out of service, if the noncritical loop isolation valves failed to close to isolate the operable CCW train from the nonseismically qualified noncritical loop, and the noncritical loop ruptured during the event, the operable train would also become disabled. The licensee reported that there were periods of time when one train of CCW was inoperable for other reasons and that the second train could have been also rendered inoperable by a seismic event. This could have caused a complete loss of safety function, in that the CCW system is a required part of the decay heat removal systems.

The licensee did not establish specific data that one CCW train was out of service for corrective or preventive maintenance (or for other reasons) during these periods of operation, but stated simply that there had been such times. The inspectors determined that the duration that both trains were vulnerable was very short (approximately 10 days since 1996). The licensee performed a risk assessment of the condition, documented in Report NSG-99-03, "Risk Significance of CCW NCL Isolation Valves Failing to Close," dated, March 25, 1999. The report concluded that the increases in the core damage frequency and the large early release frequency were both 4.6E-7/year. The inspectors determined that the low probability of occurrence of both the initiating condition (seismic event) and the specific failures required to create the inoperability rendered the safey significance low.

The failure to have either train of CCW operable during Modes 1 through 4 is a violation of Technical Specification 3.7.7. The failure to have one train of CCW operable during Modes 1 through 4 beyond the 72 hours allowed by the associated action statements is also a violation of Technical Specification 3.7.7. This Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. This violation was in the licensee's corrective action program as AR 990200839 (NCV 361; 362/99009-02).

E8.3 (Closed) Unresolved Item 361; 362/99004-04: assessment of licensee's evaluation of whether to submit an LER for refueling water storage tank (RWST) valve out of design basis.

The unresolved item was opened to give the licensee an opportunity to provide its perspective on the reportability of Valve 3HV9300, RWST T005 outlet, failing to properly change position during a 1995 surveillance activity, because of corrosion and a lack of valve preventive maintenance. Valve 3HV9301, RWST T006 outlet, was also found degraded for similar reasons. The valve preventive maintenance aspects of this issue were addressed in NRC Inspection Report 50-361; 362/96-10, Section E8.2. This unresolved item was closed based upon the discussion in Section E8.4.

E8.4 (Closed) LER 362/1999-002-00: RWST isolation valve design basis.

The licensee indicated that the UFSAR does not describe the design basis of the plant as having the RWST outlet valves closed following receipt of a recirculation actuation signal. Since failing to close these valves was not a condition outside the design basis for the plant, the licensee concluded the occurrence was not reportable. The licensee indicated that the RWST outlet valves provided a redundant feature to enable the emergency core cooling system to perform its safety-related function. Specifically, the valves were not required to maintain the principle safety barriers of fuel cladding and reactor coolant system integrity; check valves located on the RWST outlet lines ensured that containment integrity would be maintained. The licensee concluded that the UFSAR indicates that the RWST outlet valves do not have a design function to close and that the valves failing to close was a maintenance concern.

The inspectors further discussed the design basis requirement of the RWST outlet valves with the NRC Office of Nuclear Reactor Regulation. The inspectors disagreed with the design basis assessment and concluded that the RWST outlet valves performed an important redundant containment boundary function when the emergency core cooling system pumps were not running during a loss-of-coolant accident and in their degraded condition would not perform all the UFSAR design basis requirements. However, the inspectors acknowledged that the failure of the valves to meet their design basis did not necessarily mean that the plant was outside its design basis. Therefore, the inspectors concluded that the plant was not outside its design basis and the event was not reportable.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the exit meeting on August 10, 1999. 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

D. Brieg, Manager, Station Technical
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 40500:	Effectiveness of Licensee Process to Identify, Resolve, and Prevent Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 90712:	Inoffice Review of LER - in use
IP 92700:	On Site LER Review
IP 92903	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened and Closed

361; 362/99009-01	NCV	oxygen monitor setpoint limits did not ensure limit would not be exceeded (Section E8.1)
361; 362/99009-02	NCV	CCW noncritical loop isolation valves inoperable due to previously unrecognized instrument air configuration anomaly (EA 99-187) (Section E8.2)

Closed

362/1999-003-00	LER	manual reactor trip because of loss of main feedwater (Section O8.1)

362/1999-004-00	LER	manual reactor trip (emergency safety features) actuation because of feedwater control valve opening (Section O8.2)
361; 362/1998-019-00	LER	oxygen monitor exceeds Technical Specification limits (Section E8.1)
361; 362/1999-003-00	LER	CCW noncritical loop isolation valves inoperable due to previously unrecognized instrument air configuration anomaly (EA 99-187) (Section E8.2)
362/1999-002-00	LER	RWST isolation valve design basis (Section E8.4)
362/99004-04	URI	assessment of licensee's evaluation of whether to submit an LER for RWST valve out of design basis (Section E8.3)

LIST OF ACRONYMS USED

AR	action request
CCW	component cooling water
CFR	Code of Federal Regulations
EDG	emergency diesel generator
EEC	evaluation for cause
LER	licensee event report
NCV	noncited violation
NRC	Nuclear Regulatory Commission
OERP	operating experience review program
P&ID	piping and instrument diagram
PPS	plant protection system
RWST	refueling water storage tank
SG	steam generator
SWC	saltwater cooling
TLU	total loop uncertainty
UFSAR	Updated Final Safety Analysis Report