## ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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| Docket Nos.: | 50-361<br>50-362 |
| License Nos.: | NPF-10<br>NPF-15 |
| Report No.: | 50-361/98-16<br>50-362/98-16 |
| Licensee: | Southern California Edison Co. |
| Facility: | San Onofre Nuclear Generating Station, Units 2 and 3 |
| Location: | 5000 S. Pacific Coast Hwy.<br>San Clemente, California |
| Dates: | August 23 through October 3, 1998 |
| Inspectors: | J. A. Sloan, Senior Resident Inspector<br>J. G. Kramer, Resident Inspector<br>J. J. Russell, Resident Inspector<br>P. A. Goldberg, Reactor Engineer |
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| ATTACHMENT: | Supplemental Information |

# EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Units 2 and 3
NRC Inspection Report 50-361/98-16; 50-362/98-16

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 includes a routine followup inspection by a reactor inspector.

## Operations

- The response to a steam generator tube leak was excellent. Operators identified the leak before a radiation monitor alarmed, quickly validated the problem, and promptly initiated a reactor shutdown to mitigate the event. The operators manually tripped the reactor 30 minutes later, as directed by the abnormal operating instruction for a reactor coolant leak. Implementation of the steam generator tube rupture emergency operating instruction was effective, with the leak from the steam generator being isolated approximately 31 minutes after the manual reactor trip. The subsequent plant cooldown was well controlled. Technical support to operators was excellent, and Operations oversight of the event was effective (Section O1.2).

- The preparations and conduct of midloop operations were excellent, characterized by effective management oversight, thorough and safety-conscious preparation, and reliable equipment operation (Section O1.3).

- Operations procedures were generally technically accurate and usable. Procedure revisions that included human factors enhancements significantly improved the procedure usability. However, some inconsistencies in the information and requirements in a procedure and among procedures were identified (Section O3.1).

- Personnel displayed weaknesses in attention to detail during the Unit 2 reactor startup. The shift technical advisor and the reactor engineer who reviewed the estimated critical position did not identify that a step had not been initialed, although the requirements of the step were met. A reactor engineering data transmittal contained obsolete pages of a procedure. A reactor operator did not follow procedure guidance when determining startup channel values used in the inverse count rate ratio plot; this failure constitutes a violation of minor significance and is not subject to formal enforcement action (Section O4.2).

- Operators demonstrated excellent overall performance during a salt water cooling system heat treat evolution, as evidenced by well coordinated gate movements, excellent communications between the control room and field operators, and effective management oversight (Section O4.3).

Maintenance

- Good plant material condition was being maintained, although three minor deficiencies were identified. Specifically, inspectors identified: trash inside a security electromagnetic field control box that did not affect the function of the component, oil on three of four reactor coolant pumps that resulted from a past poor practice of overfilling, and leakage from a plug on the discharge of a motor-driven auxiliary feedwater pump (Section M2.1).

Engineering

- An engineering analysis of a leak sealant repair to a charging system check valve seal weld was rigorous. The analysis thoroughly addressed all applicable additional stresses that could result from the weight of the leak sealant box. A nonconformance report accurately assessed ASME Code ramifications of the leak sealant repair, including the amount of allowable liquid sealant and valve operability (Section E2.1).

- The root cause assessment of the plug leak in Steam Generator 2E088 was thorough and provided adequate confidence that the other plugs of the same design would not fail as a result of the same root cause (Section E2.2).

- A noncited violation of Technical Specifications 3.3.1.12 and 3.3.2.3 in accordance with Section VII.B.1 was identified. The licensee determined that a Technical Specification surveillance requirement had not been properly performed for verifying the setpoints for the automatic removal of the log power reactor trip bypass (Section E8.8)

## Report Details

### Summary of Plant Status

Unit 2 began this inspection period operating at essentially 100 percent reactor power. On August 30, 1998, Unit 2 reduced power to 80 percent to perform a heat treat of the circulating water system and returned to essentially full operating power on that same day. On September 18, the Unit 2 reactor was rapidly down powered and manually tripped from 35 percent because of a steam generator (SG) tube leak (Section O1.2). The unit entered Mode 5 on September 19. Following repairs to the SG, a reactor startup began on September 28 and reached Mode 1 on September 29 (Section O4.2). The unit operated at essentially 100 percent power through the end of this inspection period.

Unit 3 began this inspection period operating at essentially 100 percent reactor power. Unit 3 reduced power to 80 percent on September 6, to perform a heat treat of the circulating water system (Section O4.3). The unit returned to essentially full operating power on September 6 and operated at full power through the end of 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:

- Shift turnover (Units 2 and 3)
- Equipment operator rounds (Unit 3)
- Starting boric acid makeup pumps and recirculating boric acid storage tanks (Units 2 and 3)
- Reactor coolant system (RCS) pump sweeps (Unit 2)
- Main turbine startup and grid synchronization (Unit 2)
- Securing a condensate pump (Unit 2)
- Starting Train A high pressure safety injection pump (3P017) for inservice testing (Unit 3)
- Control operator turnover for midshift relief (Unit 3)
- Starting and loading Unit 3 Train A emergency diesel generator (EDG) (3G002) for surveillance testing

Operators were thorough and methodical in preparing for and conducting routine evolutions. Close management and supervisory oversight of operational activities were evident. Procedure use and operator communications were general consistent with documented management expectations. Specific comments on activities are discussed below.

O1.2 Forced Reactor Shutdown and Cooldown - Unit 2

a. Inspection Scope (93702)

The inspectors observed operators perform a rapid shutdown of the Unit 2 reactor on September 18, 1998, and a subsequent plant cooldown, in response to indications of a sudden increase in primary to secondary leakage.

b. Observations and Findings

b.1 Event Description

At approximately 10:05 a.m. (PDT) on September 18 control room operators observed an indication of increasing radiation levels on steam jet air ejector Radiation Monitor RE7817A. The monitor alarmed shortly thereafter, indicating that the leak rate was approximately 30 gallons per day (gpd). Increased radiation levels were also observed by the operators on the SG 2E088 radiation monitor and later on the SG 2E089 radiation monitor. Operators entered Abnormal Operation Instruction SO23-13-14, "Reactor Coolant Leak," Temporary Change Notice 2-2 and initiated a reactor shutdown.

At 10:45 a.m. the licensee calculated the leak rate at 451 gpd, based on the steam jet air ejector radiation monitor indications. However, this value was described as conservative by a factor of at least 2, because of the high energies of the short-lived fission products compared with the isotopic energies used to calibrate the monitor.

At 11:15 a.m. operators manually tripped the reactor from 35 percent power in accordance with Abnormal Operation Instruction SO23-13-14. At 11:29 a.m., after completion of the standard post trip actions (SPTAs), operators diagnosed the event as a SG tube rupture and entered Emergency Operating Instruction (EOI) 12-4, "Steam Generator Tube Rupture," Revision 16. Initially, the licensee estimated the leak rate at 80-90 gpd. Later, the licensee determined that the maximum leak rate was about 170-200 gpd. The Technical Specification (TS) limit for primary to secondary leakage is 720 gpd. Operators cooled down the RCS and isolated SG 2E088 at 11:46 a.m.

At 1:20 p.m., the licensee estimated that the leak rate had decreased to approximately 10 gpd because of the reduced differential pressure between the primary and secondary. Operators were unable to achieve the EOI goal of reducing the differential pressure to less than 50 psid because of the heat output from SG 2E088, but this had no significant operational impact because the leak rate was already very low. Operators were successful in controlling SG level and pressure to within acceptable limits.

Operators exited EOI 12-4 at 5:25 p.m. on September 19 after shutdown cooling entry conditions were met. Operators initiated shutdown cooling and continued to cool down the RCS.

b.2 Licensee Performance

Accomplishment of the SPTAs was methodical, although several operator actions were performed before the SPTAs were completed and reported. However, the control room supervisor methodically ensured that the SPTAs were completed and formally communicated. The cooldown was carefully controlled and accomplished at a steady cooldown rate. The operators demonstrated familiarity with accomplishing the rapid shutdown and cooldown, reflective of frequent practice in the simulator.

Procedure use was excellent. Operators frequently referred to the applicable procedures and successfully accomplished the intent of the procedures. Communications among operators were good. Two- and three-way communications were almost always used, as appropriate. No communications-related weaknesses were identified. Command and control were excellent. Operators closed the control room to unnecessary personnel. The Operations superintendent was in the control room, providing direct oversight and support to the shift manager and Operations staff. Division of responsibilities to accomplish the various portions of the EOI were clearly identified, with suggestions from the Operations superintendent being implemented through the shift manager.

Technical support during the event was excellent. For instance, the radiation monitoring system engineer was present and helped to interpret radiation monitor data. Chemistry and reactor engineering personnel were also present, providing leak rate estimates and other support.

The licensee accurately determined that the event did not exceed any emergency action level for event declaration.

c. Conclusions

The response to an SG tube leak was excellent. Operators identified the leak before a radiation monitor alarmed, quickly validated the problem and promptly initiated a reactor shutdown to mitigate the event. The operators manually tripped the reactor 30 minutes later, as directed by the abnormal operating instruction for a reactor coolant leak. Implementation of the SG tube rupture EOI was effective, with the leak from the SG being isolated approximately 31 minutes after the manual reactor trip. The subsequent plant cooldown was well controlled. Technical support to operators was excellent, and Operations oversight of the event was effective.

O1.3 Midloop Operations - Unit 2

a. Inspection Scope (71707)

The inspectors observed operators drain the RCS to midloop conditions on September 21, 1998, and observed other operational activities while at midloop.

b. Observations and Findings

The inspectors verified the proper alignment of the RCS level monitoring systems in containment prior to the operators commencing the draindown and monitored the successful calibration and correlation of the level indications before and during the draindown. The prejob briefing for the draindown was thorough and methodical. The licensee had prepared a "high risk evolution briefing package" to highlight safety-significant aspects of the evolution. Operations responsibilities were clearly defined, and contingency measures were discussed.

Management oversight of the draindown was excellent. An Operations manager was in the control room continuously during the draindown, providing beneficial insights and suggestions to the shift manager. No significant problems were identified during the draindown or during other operational evolutions observed after stable midloop conditions were achieved. All level monitoring systems functioned reliably.

The inspectors identified some minor inconsistencies in the procedures associated with the draindown and operation in Modes 5 and 6 (refer to Section O3.1).

c. Conclusions

The preparations for and conduct of midloop operations were excellent, characterized by effective management oversight, thorough and safety-conscious preparation, and reliable equipment operation.

**O3 Operations Procedures and Documentation**

O3.1 Operations Procedure Review - Units 2 and 3

a. Inspection Scope (42700)

The inspectors performed a review of Operations procedures to verify that the technical adequacy of the procedures was consistent with desired actions and modes of operation. The inspectors also verified the usability of procedure content and format by determining the degree to which acceptable human factors principles had been incorporated. The inspectors reviewed Procedures SO23-2-17, "Component Cooling Water System Operation," Revision 12; SO23-1-3.1, "Emergency Chilled Water System Operation," Revision 10; SO23-3-2.6, "Shutdown Cooling System Operation," Revision 14; SO23-3-1.6, "Adjusting RCS Level With the RCS Drained," Revision 0; SO23-5-1.8, "Shutdown Operations (Modes 5 and 6)," Revision 8; SO23-3-1.8, "Draining the Reactor Coolant System," Revision 13; and SO23-6-2, "Transferring of 4kV Buses," Revisions 5 and 6.

b. Observations and Findings

b.1 Procedure Inconsistencies

The inspectors identified that Procedure SO23-2-17 varied slightly from the bases for TS 3.7.10, "Emergency Chilled Water," in the interpretation of when an emergency chilled water train was inoperable while transferring the component cooling water supply from one unit to the other. The inspectors discussed the observation with a Station Technical engineer. The engineer initiated Action Request (AR) 980900701 to evaluate and correct the differences. In addition, the engineer identified that Procedure SO23-1-3.1 differed from the above documents in the interpretation of emergency chilled water train operability when transferring electrical power supplies from one unit to the other.

With Unit 2 on shutdown cooling, the inspectors identified that the flow and alarm set points were not consistent in the shutdown cooling flow guidelines table located in Procedures SO23-3-2.6, SO23-3-1.6, and SO23-3-1.8. The table provided the allowable shutdown cooling flow for various pump combinations and RCS levels and provided the high and low flow alarm set points. The licensee initiated AR 980901177 to evaluate and correct the differences.

During a Unit 2 RCS draining evolution, the inspectors identified that Procedure SO23-3-1.8 contained discrepancies in the correlation of the refueling water level instrument level with pressurizer level. Attachment 4, Step 2.3, provided for a numerical calculation of refueling water level instrument level based on pressurizer level. However, when the inspectors used the reactor coolant level correlation chart in Attachment 19, a different value for refueling water level instrument level was obtained when comparing it to pressurizer level. The calculated difference in the methods at 35 percent pressurizer level was 0.5 feet. The licensee initiated AR 980901291 to evaluate and correct the procedure discrepancies.

b.2 Procedure Revisions

The inspectors reviewed recent revisions of several procedures and specifically noted that Procedure SO23-6-2, Revision 6, improved the human factors readability. With the use of bullets, font variation, bolding, rewording of steps, and removal of unnecessary information, the procedure usability improved significantly.

c. Conclusions

Operations procedures were generally technically accurate and usable. Procedure revisions that included human factors enhancements significantly improved the procedure usability. However, some inconsistencies in the information and requirements in a procedure and among procedures were identified.

## O4 Operator Knowledge and Performance

### O4.1 Operator Awareness of Control Board Annunciators - Units 2 and 3 (71707)

On September 15, 1998, the inspectors observed approximately 17 annunciators illuminated in Units 2 and 3. The control room operators for the units demonstrated prompt and accurate knowledge for the reasons for each of the annunciators. The inspectors observed that usually very few annunciators were illuminated, consistent with the effort to maintain a "black board" (no annunciators illuminated because of chronic problems or annunciator failures). The inspectors concluded that operator knowledge and awareness of the annunciators were excellent.

### O4.2 Reactor Startup - Unit 2

a. Inspection Scope (71707)

The inspectors observed operators perform a reactor startup. The inspectors reviewed Procedures SO23-3-1.1, "Reactor Startup," Revision 19, and SO23-5-1.3.1, "Plant Startup from Hot Standby to Minimum Load," Revision 17.

b. Observations and Findings

On September 28, the inspectors observed operators perform a reactor startup. Operations management and Reactor Engineering provided guidance to the crew during the startup. The dayshift crew completed the prerequisites for the startup near the end of the shift. Operations management directed that the startup be delayed until the oncoming night shift arrived. Management did not want to conduct shift turnover during the startup or hold the day shift over to complete the startup.

The inspectors reviewed Procedure SO23-3-1.1 and determined that the procedure allowed for shift turnover during reactor startup. The inspectors found that Operations management displayed conservative decision-making by not allowing a shift turnover to be conducted during the startup, although outage time would have been reduced.

The inspectors reviewed the estimated critical position (ECP) performed for the startup and identified that Step 16 of the ECP had not been initialed. The step checked that the adjusted ECP was between Group 4 at 25 inches and Group 6 at 50 inches. The adjusted ECP was Group 4 at 143 inches, which met the requirement of the step. The final ECP "performed by" and "verified by" steps had been completed. The inspectors concluded that the reactor engineer and shift technical advisor exhibited a weakness in attention to detail by not ensuring that all steps of the ECP were signed off before signing the final signature blanks of the ECP.

The inspectors reviewed the reactor engineering data transmittal that was provided to the operators for use during the startup. The transmittal included pages of Procedure SO23-3-1.1. The transmittal was dated September 25 and included old revision pages of Procedure SO23-3-1.1, which had been revised on September 24. A reactor engineer indicated that the transmittal was created using the current revision at the time. However, the inspectors observed that final approval of the transmittal by Station Technical supervision contained outdated procedure pages. The inspectors reviewed the pages involved and determined that the content of pages in the transmittal were not changed as part of the procedure revision. The inspectors concluded that Station Technical displayed a weakness in attention to detail when providing a transmittal to Operations containing pages from an obsolete procedure revision.

During the startup, a reactor operator and a reactor engineer performed an inverse count rate ratio (ICRR) plot to predict withdrawn control element assembly reactivity to achieve reactor criticality. Procedure SO23-3-1.1, Attachment 5, Step 2.4.1.2 provides direction on how to obtain the startup channel values used in the ICRR plot, and states, in part, to average at least five count rate readings over a 1-minute period. After the first hold point, the inspectors questioned the reactor operator performing the ICRR plot on the method being used to obtain the startup channel values for the plot. The reactor operator indicated that, as with all control room indication, a visual average of the channel was used to obtain the value for the ICRR plot. The inspectors informed the operator of the procedural direction on how to obtain the startup channel values.

The inspectors concluded that the reactor operator displayed a weakness in obtaining startup channel values used in the ICRR plot. The operator's performance had minimal safety consequences because the operator was, in fact, obtaining startup channel values, although not in a method directed by Procedure SO23-3.1.1. In addition, the normal practice was to compare the reactor operator's ICRR plot with the reactor engineer's plot and resolve any discrepancies. However, the inspectors concluded that the failure of the reactor operator to follow procedure was a violation of TS 5.5.1.1.a. This failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

Personnel displayed weaknesses in attention to detail during the Unit 2 reactor startup. The shift technical advisor and the reactor engineer who reviewed the ECP did not identify that a step had not been initialed, although the requirements of the step were met. A reactor engineering data transmittal contained obsolete pages of a procedure. A reactor operator did not follow procedure guidance when determining startup channel values used in the ICRR plot; this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

Operations management demonstrated conservative decision-making throughout the reactor startup. Specifically, management delayed the startup so that shift turnover would not occur during the startup.

O4.3 Heat Treat - Unit 3

a. Inspection Scope (71707)

The inspectors monitored the performance of operators during a heat treat evolution and reviewed Procedure SO23-5-1.1, "Heat Treating the Circulating Water System," Revision 8.

b. Observations and Findings

On September 5, 1998, the inspectors observed the operators perform a heat treat of the Unit 3 circulating water system and the salt water cooling system on Unit 3 Train B and Unit 2 Train A. The inspectors observed good Operations management guidance in the control room. The operators appropriately entered the applicable TS action statements during the heat treat. The communications between the operator at the local control panel and the control room were excellent and well coordinated during the gate movements. Electrical maintenance promptly responded to trouble shoot the failure of one of the gates to move.

c. Conclusions

Operators demonstrated excellent overall performance during a salt water cooling system heat treat evolution, as evidenced by well coordinated gate movements, excellent communications between the control room and field operators, and effective management oversight.

## 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:

- Troubleshoot Valve 3HV4762, Train B motor-driven auxiliary feedwater pump to SG 2 discharge valve bypass, for failure to stroke (Unit 3)
- Adjust Zone 15 security perimeter E-field sensitivity (Units 2 and 3)
- Troubleshoot Train B emergency chiller (ME335) (Units 2 and 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:

- Turbine-driven auxiliary feedwater pump (2P140) quarterly inservice test (Unit 2)
- Valve 2HV4716, auxiliary feedwater turbine steam supply throttle valve, inservice test (Unit 2)
- Valve 2SV4700, auxiliary feedwater turbine speed controller, inservice test (Unit 2)
- Unit 2 Condensate Storage Tank (2T121) sampling
- Unit 2 Train B EDG (2G003) monthly surveillance
- Unit 3 Train A EDG (3G002) monthly surveillance
- Unit 3 Train B EDG (3G003) monthly surveillance
- Isothermal temperature coefficient measurement (Unit 3)
- Verification of charging pump flow (Units 2 and 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.

**M2 Maintenance and Material Condition of Facilities and Equipment**

M2.1 Review of Material Condition During Plant Tours - Units 2 and 3

a. Inspection Scope (71707)

The inspectors conducted routine plant tours and evaluated material condition of the units.

b. Observations and Findings

Most equipment and plant areas appeared to be well maintained.

On August 27, 1998, the inspectors observed the inside of the Zone 15 security perimeter fence electromagnetic field control box while electricians were adjusting the sensitivity of the electromagnetic field. As the electricians were closing the box, the inspectors observed that the control box contained rubbish, including loose fasteners, plastic debris, and paper. Electromagnetic field control boxes were supposed to have debris removed during quarterly preventive maintenance. An electrician removed the debris from this control box. The debris did not appear to affect operability of the circuits. The licensee generated AR 9808011948 to document the debris and determine if further action was warranted. The inspectors found that electricians' attention to detail in ensuring that the control boxes were free of debris was poor.

On September 21 the inspectors observed oil covering the horizontal surfaces interior to the coupling guard on Unit 2 Reactor Coolant Pumps P004 and P003. Unit 2 Reactor Coolant Pump P001 had oil covering portions of these surfaces. All three pumps had oil pooled on the horizontal pump lagging guard exterior to the coupling guard adjacent to the view windows. The oil interior to the guard on each of the three pumps was hardened and flaking off and was covering the upper exterior of the pump case and seal heat exchanger. The inspectors determined that the most probable source of the oil was leakage along the pump shaft adjacent to the lower motor oil reservoir. The inspectors discussed this with Station Technical personnel, who stated that in the past lower oil reservoirs had been overfilled, but that recent oil additions were performed in a more cautious manner. The inspectors found that the excessive oil on three of the four Unit 2 reactor coolant pumps was indicative of past poor practice in adding oil to the pumps.

On September 26 an equipment operator noticed leakage from a threaded plug on the casing for Unit 2 Train B motor-driven auxiliary feedwater pump. The leakage was approximately 240 ml/min, directed to the floor, with the motor-driven auxiliary feedwater pump operating. The plug location had been provided by the vendor for an instrument or drain installation but was not used by the licensee. The licensee inspected the plug and pump casing threads, which evidenced no significant wear. The licensee unsuccessfully attempted to stop the leakage by reinstalling the plug with a "speed alloy" coating. The inspectors reviewed the operability assessment contained in AR 980901779, and the guidance in NRC Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2 and 3 Piping," and found that the Train B motor-driven auxiliary feedwater pump remained operable with the leakage. The inspectors also observed that similar plugs on the auxiliary feedwater pump cases had leaked in the past, indicating that the licensee had not been proactive in preventing the leaks.

c. Conclusions

Good plant material condition was being maintained, although three minor deficiencies were identified. Specifically, inspectors identified: trash inside a security electromagnetic field control box that did not affect the function of the component, oil on three of four reactor coolant pumps that resulted from a past poor practice of overfilling, and leakage from a plug on the discharge of a motor-driven auxiliary feedwater pump.

## III. Engineering

### E2 Engineering Support of Facilities and Equipment

E2.1 Leak Sealant Engineering Analysis - Unit 3

a. Inspection Scope (37551)

The inspectors evaluated an engineering analysis used to justify the use of a leak sealant repair to a leaking seal weld on Check Valve S3MU069, Unit 3 swing charging pump (3P191) discharge check. The inspectors reviewed Nonconformance Report (NCR) 980801578; Calculation M-1208-008-3A, Calculation Change Notice N-3; and visually inspected the leaking weld location both before the leak sealant repair and after the leak sealant box had been installed.

b. Observations and Findings

Check Valve S3MU069 is a 2-inch Kerotest spring-loaded valve installed in an ASME Code Class II system. The licensee had red tagged closed the suction and discharge isolation valves for the swing charging pump in order to replace Check Valve S3MU069. Check Valve S3MU069 was suspected of leak-by in the reverse direction, causing pipe vibration in the charging system. When mechanics began to cut the body-to-bonnet seal weld on Check Valve 3MU069, water began to spray from the cut. The check valve body is screwed into the bonnet, then the valve body is seal welded to the bonnet. The design pressure boundary and the structural support is provided by the threaded connection. The isolation valves leaked by and the threaded connection leaked by, which caused the water leakage after the seal weld was partially cut.

On August 28, 1998, contract mechanics: (1) peened the cut weld and installed a leak sealant box over the seal weld, (2) injected the box with liquid sealant in order to stop the leakage, (3) opened the boundary isolation valves, and (4) placed the swing charging pump in service. The licensee planned to replace Check Valve S3MU069 during the scheduled Cycle 10 refueling outage.

Calculation M-1208-008-3A, Calculation Change Notice N-3, was a stress analysis of the charging system discharge piping with the additional weight of the leak sealant box (75 pounds was used in the calculation, actual box weight was 48 pounds). This engineering analysis calculated the pipe stress incorporating internal pressure, weight, the operational basis earthquake, the design basis earthquake, and stresses because of thermal effects. All stresses in the piping were below ASME Construction Code allowables. The inspectors considered the calculation rigorous because all applicable stress factors were analyzed. NCR 980801578 contained a safety analysis for the use of the leak sealant repair. The ASME Construction Code prohibited the use of liquid sealant for pressure boundary or structural members in Code Class II systems. The NCR demonstrated that the seal weld was not, by design, either a pressure boundary nor a structural support for the valve. Consequently, the use of liquid sealant was allowable. The NCR was rigorous because it incorporated consideration of the ASME Code, as well as limiting the amount of liquid sealant to be used. In addition, the NCR addressed check valve operability with the leak sealant clamp in place.

c. Conclusions

An engineering analysis of a Leak sealant repair to a charging system check valve seal weld was rigorous. The analysis thoroughly addressed all applicable additional stresses that could result from the weight of the leak sealant box. An NCR accurately assessed ASME Code ramifications of the leak sealant repair, including the amount of allowable liquid sealant and valve operability.

E2.2 SG Tube Leak - Unit 2

a. Inspection Scope (37551)

The inspectors reviewed the root cause assessment of a SG tube leak that occurred on September 18, 1998.

b. Observations and Findings

Based on an increase in radiation monitor readings on September 18 the licensee determined that a tube leak had developed in SG 2E088. The licensee promptly shut down the reactor, cooled down the RCS, and drained the RCS to midloop conditions to support inspection and repair of the leaking SG.

The licensee inspected SG 2E088 by means of conducting a low pressure test. The secondary side of SG 2E088 was pressurized to approximately 165 psi while the primary side of the tube sheet was being monitored by a remotely-controlled camera. The licensee determined from this inspection that one plugged tube was leaking from around the edge of the plug. The leaking plug was one of approximately 720 Westinghouse Alloy 690 mechanical plugs that had been installed in Units 2 and 3 during the 1995 refueling outages, primarily to replace older Alloy 600TT plugs. The plugs were expanded by pulling on an internal wedge to form a seal between the plug and the tube. The further the wedge was pulled, the more the plug was expanded. For the tubes in Units 2 and 3, the normal wedge travel was 0.4 - 0.6 inches.

Four plugs installed in 1995 had longer than normal wedge travel. Of these, one leaked during the return to service at the end of the 1995 refueling, and one leaked during the return to service at the end of the 1997 refueling. The licensee, in conjunction with Westinghouse engineering personnel, determined that the longer wedge travel was an indicator of abnormal plug installation, although not necessarily indicating that the plug installation was unacceptable. The licensee evaluated the four plugs and determined that the two that leaked had not sealed properly because of slight tube ovality caused by drill bit movement during the removal of the old plugs. The leaks were not revealed until continued tube wear caused through-wall penetration of the tubes. The licensee could not explain why one of the remaining plugs had longer than normal wedge travel and preventively replaced that plug in 1997.

The licensee had a reasonable explanation for the longer wedge travel on the last of the four plugs. Because of damage caused to the tube during removal of the old plug in 1995, the licensee had machined the inner diameter of the tube to restore the tube surface condition so that it could be plugged. Both the licensee and Westinghouse engineering personnel had determined that the tube was acceptable for this plug design. Because the machining resulted in a slightly larger tube inner diameter, it was expected that the plug would have to be expanded more than normal.

This last, longer wedge travel, plug was the plug that was found to be leaking. The licensee determined that, in retrospect, the efforts to machine the tube to be perfectly round had not been successful. The licensee determined that the plug had never adequately sealed and that tube support wear had ultimately revealed the leak observed on September 18.

The inspectors also observed that, before the cause of the leak was identified, the licensee had developed a list of 12 possible leakage initiators. A poor seal causing leakage around an installed plug was listed as the most likely initiator. The predictive analysis proved to be correct.

c. Conclusions

The root cause assessment of the plug leak in SG 2E088 was thorough and provided adequate confidence that the other plugs of the same design would not fail as a result of the same root cause.

**E8 Miscellaneous Engineering Issues (92700, 92903)**

E8.1 (Closed) Inspection Followup Item 50-361; 362/96010-01: review the diagnostic traces for motor-operated valves following testing

Background

The inspectors observed that during dynamic testing of Valve 2HV9348, safety injection and containment spray minimum flow to the refueling water storage tank, the open force trace exhibited a rapid force increase at approximately midstroke. The licensee noted a change in the seating characteristics, which was observed in the static test that was performed after the dynamic test. Based on these observations, the licensee performed maintenance on the valve internals and then performed a static test on the valve that showed that the seating anomaly had been eliminated. However, the inspectors were concerned that the opening anomaly evidenced during dynamic testing appeared to indicate internal interference during stroking of the valve that was unrelated to the seating anomaly.

The inspectors observed similar anomalous behavior in the opening dynamic trace for Valve 3HV9306, safety injection pump minimum flow isolation valve. The inspectors were concerned that the anomalous behavior of the valves was not well understood by the licensee in order to assure that behavior of the valves was predictable and would not become a challenge to the adequacy of the actuator capability or switch settings. The licensee identified that repeat dynamic testing of the two valves was planned for the next outage.

The inspectors determined that the maintenance procedure directed lubrication of valve internals following valve internal maintenance to aid in fitup and reassembly. The inspectors questioned whether this maintenance practice might temporarily enhance the valve performance immediately following maintenance because of internal lubrication. As a result of the inspectors' questions, the licensee committed to design basis test the two valves during the Cycle 9 refueling outages. Since the licensee planned to retest Valve 2HV9348 during the next outage, the inspectors decided to include review of the test results for lubrication effects as part of the diagnostic trace review.

Inspection Followup

The inspectors reviewed the Cycle 9 refueling outage test packages for Valves 2HV9348 and 3HV9306. The diagnostic test data were evaluated and no anomalies, as identified following the previous test were found. Based on the new diagnostic test data, the inspectors determined that the previous maintenance on the valve internals had solved the problem of the rapid force increase at midstroke during the dynamic test and of anomalous seating characteristics during the static test, for the two valves. In addition, the inspectors determined that the maintenance procedure that directed internal lubrication during assembly did not affect valve performance.

E8.2 (Open) Inspection Followup Item 50-361; 362/96010-02: review of the evaluation of Information Notice 96-48, "Motor-Operated Valve Performance Issues"

Background

The inspectors determined that the licensee used run efficiency in the close direction in analyzing motor-operated valve actuator capability. The inspectors informed the licensee of the issues recently highlighted in Information Notice 96-48, which included concerns regarding the use of run efficiency and problems with motor-operated valve key failures. The licensee was evaluating the information notice for its applicability and planned to incorporate appropriate measures to assure that the motor-operated valve program remained based on the best available data for predicting valve performance.

Inspection Followup

The inspectors reviewed AR 970101858, dated January 31, 1997, which the licensee initiated to address issues identified in Information Notice 96-48. One issue related to performance problems with motor-operated valve keys. The licensee stated that they had replaced all of the 1018 material keys with the recommended 4140 material for all of the valves in the Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," program. Since they had replaced all of the 1018 keys in the Generic Letter 89-10 program, the licensee concluded that the key material issue was resolved. To address the potential for motor keyway distress stemming from the use of the stronger 4140 key material, the licensee examined valves in the Generic Letter 89-10 program that were fast acting, had a high motor start torque rating, and whose diagnostic traces indicated high impact loads. The licensee found that none of the motor-operated valves met all three of the criteria. Based on high impact loads, the licensee selected the Units 2 and 3 emergency cooler valves and inspected their motor keyways. No evidence of cracking was found. The licensee stated that, while keyway cracking was not expected, the valves that were most susceptible were scheduled for inspection.

The second issue that the licensee addressed was the potential for torque output from the motor-operated valve actuators to be less than predicted by the actuator vendor, Limitorque Corporation. The licensee stated that it had received preliminary guidance on actuator efficiencies that was being evaluated. In May 1998 Limitorque issued Technical Update 98-01 to provide updated guidance for determining the output torque capability. Limitorque specified that in the sizing equation the licensee should use nominal motor starting torque, pullout efficiency, overall actuator gear ratio based on the particular actuator, and an application factor. The licensee planned to perform testing starting in Cycle 10, with the more marginal valves being tested first. This testing will determine if the use of run efficiency was appropriate. This inspection followup item remains open pending review of the Cycle 10 test data.

E8.3 (Closed) Violation 50-361; 362/96017-01: review of corrective actions for directing unbracketed steps to be marked N/A without verifying completion of these steps

Background

The inspectors found, during the Unit 3 refueling outage in September 1995, that the RCS had been filled solid, then drained to approximately 50 percent pressurizer level for integrated leak rate testing and integrated engineered safety features testing. The licensee entered Procedure SO23-3-1.4, "Filling and Venting the RCS," Temporary Change Notice 15-2, Procedure Modification Permit 1. Attachment 3 of this procedure included a check that Valve 3MU995, reactor head vent line orifice, was locked closed. During an outage in September 1996, Valve 3MU995 was found in an open position during routine work. The alignment performance guidelines for Procedure SO23-3-1.4, Attachment 3 allowed for unbracketed steps, including the check of Valve 3MU995, to be omitted if verification was made that they had been performed during a previous fill evolution. The inspectors determined that no verification was made during the previous fill evolution and that, therefore, checks were required.

Inspection Followup

The inspectors reviewed the response letter to the violation dated January 15, 1997. The immediate corrective actions included: Operations management provided interactive training to the operating crews on self-checking; Operations manager stressed the expectations regarding good operating practices of Procedure SO23-0-44, "Professional Operator Development and Evaluation Program," with the operators; and operators were assigned required readings on the errors and good operating practices. The inspectors discussed these corrective actions with applicable licensee personnel and verified that the actions were completed.

The inspectors reviewed Quality Engineering Root Cause Report SEA 96-014. The licensee determined that some of the mispositioned components resulted from problems with procedures and processes; however, most of the problems resulted from inadequate implementation of standard work practices. The analysis made a number of recommendations that were addressed in Corrective Action Request 970700839-01, dated May 21, 1998. The licensee determined that there were 48 recorded instances of mispositioned components in 1996. In addition, the corrective action request implemented many of the recommended corrective actions from the root cause analysis.

As part of the corrective actions, the licensee presented an 8-hour training course to the Operations staff which consisted of human error reduction habits training. Training records documented that all of the Operations crews, maintenance personnel, and chemistry personnel had completed the training. The licensee developed a self/cross-checking simulator. The inspectors reviewed Laboratory Evaluation MT-8250-OPS, "MT-8250 Self/Cross Check Trainer Laboratory Evaluation," Revision 0, which provided instructions for operating the self-checking simulator. The inspectors found that the simulator design consisted of a panel that contained many switches with very similar nomenclature to each other. This configuration created the need for the student to perform the correct self-check techniques in order to succeed. The personnel using the simulator were expected to demonstrate proper pre-task review prior to actual execution of the task, and correct physical verification of the panel prior to manipulating the switches. Personnel were given 15 minutes to perform the task. The licensee stated that adherence to the procedure and using correct self-checking techniques would result in a panel fan starting without causing any alarms. The licensee stated that the self/cross-checking simulator was incorporated into the continuing training programs for Operations, Chemistry, and Maintenance personnel. At the time of the inspection, Operations and Maintenance personnel had completed the training.

The licensee stated that Chemistry revised some of its procedures to strengthen control of component manipulations. The inspectors reviewed Procedure SO123-III-2.10.23, "Units 2/3 Secondary Chemical Feed Systems Operation," Revision 18, which was revised to include performer sign-offs for manipulation steps. Procedure SO123-III-2.12, Issue 2, "Units 2/3 Auxiliary Chemical Feed Systems Operation and Condenser Vacuum Drag Chemical Addition," Revision 1, was revised to add independent verification sign-offs for some of the manipulations.

E8.4 (Closed) Inspection Followup Item 50-361; 362/97019-04: review of laboratory testing of the correlation between SG manways hydraulic tensioner preload and actual torque

Background

In NRC Inspection Report 50-361; 362/97-19, the inspectors found that the vendor technical manual for the SGs contained instructions for the installation of the cold leg manway cover, preloading the studs and torquing the nuts. The licensee stated that engineering evaluations and calculations were performed to ascertain the adequacy of the preload applied on the Unit 2 primary manway cover fasteners. The inspectors determined that assumptions of several variables such as fastener thread dimensions, thread friction factor, nut to face friction factor, thread lubricant, fastener temperature, stack up of tolerances, et cetera, would need to be considered. Additionally, variables associated with the hydraulic tensioner would also need to be considered. The licensee stated that laboratory tests were planned to confirm the correlation between the vendor manual required torque value and the preload applied using the hydraulic tensioner. Review of the laboratory tests was identified as an inspection followup item.

Inspection Followup

The inspectors reviewed Failure Analysis Report 97-014, "Failure Analysis of the Steam Generator Primary Manway Gaskets," Revision 1, which addressed both the need to hot torque the SG manway gaskets to stop weepage and the correlation between the tensioner applied stud preload values and the torque wrench applied preload values. The test established a correlation between the stud preload and the tightening torque when a torque wrench was used instead of the hydraulic tensioner. The licensee performed testing on the maintenance training mockup of a SG using both the tensioner and torque wrench to determine the correlation between the tools when installing a manway cover. Hollow studs with proof rods were installed in six of the manway stud holes to allow direct measurement of the stud length and elongation. The licensee performed additional testing at EML Laboratories on a hollow stud to establish the actual stress/deflection relationship of the material. The licensee conducted a second mockup test in order to gather additional test data that would assist in further correlating torque with stud elongation and tensioner pressure. From the test results, the licensee determined that the torque/preload relationship documented in Calculation M-DSC-347, "Evaluation of Steam Generator and Pressurizer Primary Manway for Higher Preload," Revision 0, was not consistent with the experimental evidence and required revision.

The inspectors reviewed Calculation M-DSC-347, Revision 1, which incorporated the test results of the manway testing using the SG mockup reported in Failure Analysis Report 97-014. This calculation addressed the stud loading during installation and the removal of the manways of the SG and the pressurizer.

The inspectors reviewed AR 970501180, dated May 16, 1997, which implemented the corrective actions from Failure Analysis Report 97-014. The inspectors reviewed Procedure SO23-I-6.113, "Removal and Installation of Steam Generator (Primary) and Pressurizer Manway Covers," Revision 11. The inspectors determined that the licensee revised the procedure to reflect the acceptance criteria in Calculation M-DSC-347, Revision 1. The inspectors concluded that the licensee was proactive in performing testing to determine the correlation between the stud preload and the tensioner pressure applied during assembly and the correlation between the stud preload and the tightening torque when a torque wrench was used.

E8.5 (Open) Unresolved Item 50-361; 362/96013-03: review of a design change involving the EDG electrical cross tie

Background

The inspectors noted that a self-assessment team had reviewed Design Change Package 2/3-7048.00SE, which provided a capability to manually cross connect an EDG from one unit to the 4.16 kV Class 1E bus on the other unit. The self-assessment team observed that the 10 CFR 50.59 evaluation for the design did not clearly address the use of the design during a 10 CFR 50.54(x) condition. The self-assessment team considered that additional discussion on the bounds of the 10 CFR 50.59 evaluation related to physical installation and that actual operations needed more analysis. In addition, the self-assessment team questioned whether the impact of installing the 10 CFR 50.54(x) switches into the existing control circuits had been thoroughly addressed in the safety evaluation. The inspectors determined that it would take two deliberate switching actions to cross tie an EDG from one unit to the safety bus of the other unit.

During a review of the 10 CFR 50.59 evaluation, the inspectors found that the design change added new electrical switches, which bypassed the permissive logic for the EDG cross tie circuit breakers to allow electrical alignment of the Unit 2 EDGs to Unit 3. The inspectors also determined that the misoperation of these switches could prevent the automatic sequencing of loads onto the Class 1E bus following a loss of normal onsite AC power. In response to the inspectors' concerns, the licensee installed jumpers and lifted electrical leads as required to isolate the safety-related portions of the cross tie design change and return the system to its premodification status. The licensee issued a letter to the NRC Program Office, dated December 31, 1996, which described the modification. Although the letter indicated that the modification did not represent an unreviewed safety question, the licensee planned to maintain the modification disabled pending further review by the NRC.

Inspection Followup

The inspectors discussed this unresolved item with the project manager from the Office of Nuclear Reactor Regulation. The project manager stated that there was not a regulatory context to approve this electrical cross tie modification since it was intended to be used under 10 CFR 50.54(x). The inspectors determined that use of the cross tie was clearly a contingency action for use beyond the design basis of the plant. If the NRC were to approve the modification, it would place the cross tie modification into the design basis of the plant. This item remains open pending the results of further discussions between the NRC Program Office and the licensee.

E8.6 (Closed) Inspection Followup Item 50-361; 362/97006-01: review of weaknesses in the AR procedure

Background

The inspectors reviewed Procedure SO123-XX-1, Issue 2, "Action Request/Maintenance Order Initiation and Processing," Revision 4, and found procedure weaknesses. The inspectors found that the procedure provided a process for plant personnel to identify conditions adverse to quality; however, the procedure did not specifically require plant personnel to document conditions adverse to quality by means of an AR or other corrective action program process. The procedure described the composition of the AR committee, and specified the committee's function. However, the procedure did not specify a minimum committee composition that would constitute a quorum nor did it specify the minimum required membership for performance of operability evaluations. Other procedure weaknesses included a lack of criteria for ARs selected for review, the staff that would perform the review, and the process for performing the review.

Inspection Followup

The inspectors reviewed Procedure SO123-XX-1 Issue 2, Revision 5 and found that the revised procedure addressed weaknesses previously identified. The inspectors concluded that the revised procedure appropriately addressed the concerns.

E8.7 (Closed) Inspection Followup Item 50-361; 362/97006-02: review of Independent Safety Engineering Group responsibilities

Background

The inspectors reviewed the joint utility management audit concerning the Independent Safety Engineering Group. The audit team interviewed site personnel who stated that the only time that the Independent Safety Engineering Group was seen in the plant was when they were investigating a plant event to perform a root cause evaluation, or to lead or participate in a special project. The audit team recommended that the licensee evaluate whether or not the Independent Safety Engineering Group met the intent of the original Independent Safety Engineering Group charter. The licensee issued Corrective Action Request 002-97 to resolve the audit concerns.

Inspection Followup

The inspectors noted that Corrective Action Request 002-97 concluded that the Independent Safety Engineering Group met the requirements of the original charter. The inspectors reviewed Procedure SO123-XII-2.24, "Independent Safety Engineering Group Functions," Temporary Change Notice 2-1. The inspectors found that surveillance of plant activities was accomplished during root cause investigations. In addition, Quality Engineering reviewed Independent Safety Engineering Group program requirements and concluded that all of the functions were being performed. The inspectors concluded that the licensee was performing all of the required functions of the Independent Safety Engineering Group.

E8.8 (Closed) Licensee Event Report (LER) 50-361/98-011-00 and 98-011-01: missed TS surveillance for log power bypass removal setpoint

In June 1998 the licensee determined that the setpoints for the high log power bypass automatic removal had not been verified as required by TS Surveillance Requirements 3.3.1.12 and 3.3.2.3. The facility design utilized a single bistable (in each protection channel) to accomplish the automatic bypass removal for the low departure from nucleate boiling and high local power density trips (as power is increasing) and for the high log power trip (as power is decreasing). The bypasses for the departure from nucleate boiling and local power density trips are required to be automatically removed when power is greater than or equal to 1E-4 percent, and the high log power trip is required to be automatically removed when power less than or equal to 1E-4 percent. Because a setpoint of exactly 1E-4 percent could not physically and reliably be achieved, the required conditions could not be implemented with a single bistable. The licensee had implemented the surveillance requirement by considering 1E-4 percent as a nominal value; the calibration and surveillance procedures provided a tolerance that implemented the general intent of the TS within the limits of the safety analysis.

The inspectors determined that, following a number of surveillance compliance problems identified in the wake of implementing the improved TS, the licensee had performed an exhaustive review of surveillance requirements to ensure that all requirements were being satisfied and were properly addressed in the implementing procedures. That review had missed this surveillance discrepancy.

Although the actual and potential safety consequence of the reported surveillance omission was minor, the licensee had a previous opportunity to identify the error. This nonrepetitive, licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-361/9816-01; 50-362/9816-01).

The licensee developed a method for starting up and shutting down that would satisfy all TS requirements, documented in NCR 980701034. The method involved resetting the bistable as appropriate for the affected trip function, after power passes through 1E-4 percent. Before reactor startup, the bistable would be set at about 8E-5 percent power. During the startup, the automatic bypass removal would occur (and be verified) before power reached 1E-4 percent, so that the high log power bypass would be automatically removed on the next down power before power decreased below 1E-4 percent. This method, though cumbersome, would ensure that the appropriate bypass would be automatically removed when required. AR 980701034 stated that, as a permanent corrective action, the licensee intended to submit a TS amendment request that would allow an appropriate tolerance for the bistable setting. The inspectors considered this corrective actions acceptable.

## IV. Plant Support

### R8 Miscellaneous Radiological Protection and Chemistry Issues (92700, 92904)

R8.1 (Closed) LERs 361/98-004-00, 361/98-004-01 and IFI 361/98003-05: EDG fuel oil particulates

On February 5, 1998, the licensee sampled the EDG fuel oil system for the Unit 2 Train A EDG and sent the sample to an off-site facility for analysis. On February 11, the licensee received the results, which indicated that the particulate concentration was 31.9 mg/l, which exceeded the TS limit of 10 mg/l. The previous off-site lab results were 8.6 mg/l on November 20, 1997, and the on-site analysis indicated 0.0 mg/l. Based on the February 11 sample results, the licensee filtered the fuel oil with diatomaceous earth. The filtering process removed a varnish-type substance (nonparticulate) related to the aging of the fuel oil. Subsequently, the particulate level, as determined by an off-site lab, was returned to within the TS limit.

The evaluation of the testing methodology, techniques, and test equipment revealed that verbatim compliance with the testing methodology can result in test values for particulates that very significantly. The licensee concluded that a TS surveillance requirement for the EDG fuel oil particulates had not been exceeded, and revised the LER to indicate that it was a voluntary report. The inspectors confirmed the licensees's conclusion.

### P8 Miscellaneous Emergency Preparedness Issues (90712)

P8.1 (Closed) LER 361/98-017-00: manual toxic gas isolation system actuation

This LER was a minor issue and was closed.

## V. Management Meetings

### X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the exit meeting on October 7, 1998. 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

J. Fee, Manager, Maintenance
D. Brieg, Manager, Station Technical
G. Gibson, Manager, Compliance
D. Herbst, Manager, Site Quality Assurance
J. Hirsch, Manager, Chemistry
R. Krieger, Vice President, Nuclear Generation
J. Madigan, Manager, Health Physics
D. Nunn, Vice President, Engineering and Technical Services
T. Vogt, Plant Superintendent, Units 2 and 3
R. Waldo, Manager, Operations

## INSPECTION PROCEDURES USED

| | |
|---|---|
| IP 37551: | Onsite Engineering |
| IP 42700: | Plant Procedures |
| IP 61726: | Surveillance Observations |
| IP 62707: | Maintenance Observations |
| IP 71707: | Plant Operations |
| IP 71750: | Plant Support Activities |
| IP 86700: | Spent Fuel Pool |
| IP 90712: | Inoffice Review of LER |
| IP 92700: | On Site LER Review |
| IP 92901: | Followup - Operations |
| IP 92902: | Followup - Maintenance |
| IP 92903: | Followup - Engineering |
| IP 92904: | Followup - Plant Support |
| IP 93702 | Prompt Onsite Response to Events at Operating Power Reactors |

## ITEMS OPENED AND CLOSED

Opened and Closed

| | | |
|---|---|---|
| 50-361; 362/98016-01 | NCV | Missed TS surveillance (Section E8.8) |

Closed

| | | |
|---|---|---|
| 50-361;362/96010-01 | IFI | Review the diagnostic traces for motor-operated valves following diagnostic testing (Section E8.1) |
| 50-361;362/96010-02 | IFI | Review of the evaluation of Information Notice 96-48 (Section E8.2) |

| | | |
|---|---|---|
| 50-361;362/96017-01 | VIO | Review of corrective actions for directing unbracketed steps to be marked N/A without verifying completion of these steps (Section E8.3) |
| 50-361;362/97019-04 | IFI | Review of laboratory testing of the correlation between SG manways hydraulic tensioner preload and actual torque (Section E8.4) |
| 50-361;362/97006-01 | IFI | Review of weaknesses in the AR procedure (Section E8.6) |
| 50-361; 362/97006-02 | IFI | Review of Independent Safety Engineering Group responsibilities (Section E8.7) |
| 50-361/98-011-00, -01 | LER | Missed TS surveillance for log power bypass removal setpoint (Section E8.8) |
| 50-361/98-004-00, -01 | LER | EDG fuel oil particulates (Section R8.1) |
| 50-361/98003-05 | IFI | EDG fuel oil particulates (Section R8.1) |
| 50-361/98017-00 | LER | Manual toxic gas isolation system actuation (Section P8.1) |

Discussed

| | | |
|---|---|---|
| 50-361;362/96013-03 | URI | Review of a design change involving the EDG electrical cross tie (Section E8.5) |

## LIST OF ACRONYMS USED

| | |
|---|---|
| AR | action request |
| CFR | Code of Federal Regulations |
| ECP | estimated critical position |
| EDG | emergency diesel generator |
| EOI | emergency operating instruction |
| gpd | gallons per day |
| ICRR | inverse count rate ratio |
| LER | licensee event report |
| NCR | nonconformance report |
| NRC | Nuclear Regulatory Commission |
| PDR | Public Document Room |
| RCS | reactor coolant system |
| SG | steam generator |
| SPTA | standard post trip action |
| TS | technical specification |