

U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos. 50-206/93-11, 50-361/93-11, 50-362/93-11

Docket Nos. 50-206, 50-361, 50-362

License Nos. DPR-13, NPF-10, NPF-15

Licensee: Southern California Edison Company
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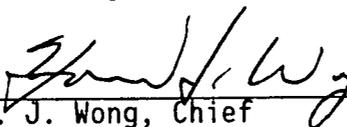
Facility Name: San Onofre Units 1, 2 and 3

Inspection At: San Onofre, San Clemente, California

Inspection Conducted: May 13 through June 23, 1993

Inspectors: C. W. Caldwell, Senior Resident Inspector
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7/21/93
Date Signed

Inspection Summary

Inspection on May 13 through June 23, 1993 (Report Nos. 50-206/93-11, 50-361/93-11, 50-362/93-11)

Areas Inspected: Routine, announced resident inspection of Units 1, 2 and 3 Operations Program including the following areas: operational safety verification, radiological protection, security, evaluation of plant trips and events, bi-monthly surveillance activities, monthly maintenance activities, refueling activities, independent inspection, licensee event report review, NRC Bulletin followup, plant modification and refueling activities, and followup of previously identified items. Inspection procedures 35702, 37700, 40500, 41500, 60705, 60710, 61726, 62700, 62703, 64704, 71500, 71707, 71710, 82301, 92700, and 93703 were covered.

Safety Issues Management System (SIMS) Items: None

Results:

General Conclusions and Specific Findings:

Strengths:

The inspector considered that the licensee's efforts to perform design basis testing of motor operated valves by using strain gauges, pressure gauges, and flow measurements a strength (Paragraph 6.b).

The inspector observed that the licensee performed an outage risk assessment (Paragraph 8.c) and had implemented a significant amount of Nuclear Oversight Division review of refueling activities (Paragraph 8.d). The inspector considered these efforts to be strengths by minimizing the potential for shutdown events.

Weaknesses:

The inspectors noted a number of examples in which both contractor and licensee personnel did not adhere to procedural requirements as follows:

- Personnel did not properly secure components near safety-related equipment in a number of areas throughout the plant (Paragraph 3.e).
- Personnel missed a step in a construction work order which resulted in an inadvertent diesel generator start in Unit 2 (Paragraph 4.a).
- Personnel did not perform an evaluation of a scaffolding modification that added dead-weight load to a safety-related pipe support (Paragraph 7.b).
- Personnel did not properly control fire extinguishers in two areas of Unit 2 (Paragraph 3.f).
- Personnel did not close a Technical Specification fire door after completing activities in a Unit 3 charging pump room (Paragraph 3.g).
- Personnel did not observe the improper operation of a Unit 3 control room recorder (Paragraph 3.h).
- Personnel parked a tractor-trailer within the seismic exclusion zone of Unit 3 for the fire tanker truck restraints (Paragraph 3.d).

One result of these procedural adherence discrepancies was inadequate configuration control of components during pre-outage and outage activities that potentially jeopardized safety-related equipment (Paragraph 3).

The inspector noted several examples of uncontrolled tools that were half-in and half-out of contaminated areas that had the potential for

unwanted spread of contamination in Units 1 and 2 (Paragraph 3.c).

The inspector observed a number of components leaking borated water on the Unit 1 spent fuel pool cooling heat exchanger on several occasions. These leaks apparently had not been identified by the licensee until the inspector identified the discrepancy. In addition, contamination was found in a clean area of Unit 1 after several individuals had their shoes contaminated (Paragraph 3.a). The inspector considered that the licensee should ensure that adequate diligence is focused on controlling leaks and contamination control of Unit 1.

The inspector considered that the licensee did not appear to have evaluated the implications for Unit 3 when a broken pedestal on a Unit 2 weir gate was first identified. It was not apparent that it was formally evaluated until approximately 4 days later when licensee supervision was made aware of the condition (Paragraph 3.i).

The inspector noted two examples in which procedures provided conflicting guidance. One example involved confusing guidance for requirements to perform shiftly surveillances by the refueling supervisors (Paragraph 7.a). The second example involved conflicting scaffolding seismic restraint criteria (Paragraph 7.b).

The inspector noted that the licensee's June 10, 1993, response to NRC Bulletin 93-02 was not clear. While the licensee had previously identified and evaluated fibrous material in the normal air coolers in the containment building and concluded that there was not a potential for the fibrous material to cause strainer plugging, this evaluation was not reflected in the licensee's response. Thus, the inspector considered that this indicated a need for clearer communications in the future (Paragraph 8.e).

Significant Safety Matters: None

Summary of Violations:

One violation with multiple examples was identified concerning failure to follow procedures. The examples involved personnel not properly securing components near safety-related equipment in a number of areas throughout the plant (Paragraph 3.e) and the failure of personnel to perform an evaluation of a scaffolding modification that added a dead-weight load to a safety-related pipe support (Paragraph 7.b).

Four non-cited violations of failures to adhere to procedures were identified involving: uncontrolled fire extinguishers (Paragraph 3.f), a fire door left open (Paragraph 3.g), failure to recognize a control room recorder for RCS reference average temperature was not working properly (Paragraph 3.h), and steps of a procedure being skipped resulting in an inadvertent diesel generator start (Paragraph 4.a).

Open Items Summary:

During this report period, 13 new followup items were opened and six were closed.

DETAILS

1. Persons Contacted

Southern California Edison Company

H. Ray, Senior Vice President, Power Systems
*R. Krieger, Vice President and Site Manager
R. Rosenblum, Vice President, Nuclear Engineering & Technical Support
J. Reilly, Manager, Nuclear Engineering & Construction
*B. Katz, Manager, Nuclear Oversight
*K. Slagle, Deputy Station Manager
*R. Waldo, Operations Manager
*L. Cash, Maintenance Manager
*D. Breig, Manager, Station Technical
M. Short, Manager, Site Technical Services
M. Wharton, Manager, Nuclear Design Engineering
P. Knapp, Manager, Health Physics
*W. Zintl, Manager, Emergency Preparedness
*D. Herbst, Manager, Quality Assurance
C. Chiu, Manager, Quality Engineering
*G. Moore, Plant Superintendent, Unit 1
V. Fisher, Plant Superintendent, Units 2/3
*G. Gibson, Supervisor, Onsite Nuclear Licensing
J. Reeder, Manager, Nuclear Training
H. Newton, Manager, Site Support Services
*D. Irvine, Supervisor, Technical Support
*J. Fee, Assistant Manager, Health Physics
*A. Tally, Supervisor, Health Physics
*K. O'Connor, Manager, Construction
*M. Tolson, Engineer, Emergency Preparedness
*D. Axline, Engineer, Onsite Nuclear Licensing
*R. Giroux, Engineer, Onsite Nuclear Licensing

San Diego Gas and Electric Company

*R. Erickson, Site Representative

City of Riverside

*C. Harris, Site Representative

*Denotes those attending the exit meeting on June 30, 1993.

The inspectors also contacted other licensee employees during the course of the inspection, including operations shift superintendents, control room supervisors, control room operators, QA and QC engineers, compliance engineers, maintenance craftsmen, and health physics engineers and technicians.

2. Plant Status

Unit 1

The Unit was permanently shutdown on November 30, 1992. The core was off-loaded to the spent fuel pool (SFP) on March 6, 1993. Primary and secondary systems were in a "SAFSTOR" condition. Final containment closure was achieved on June 17, 1993.

Unit 2

The Unit began the inspection report period at 100% power. On June 3, 1993, Unit power was reduced to 80% in order to begin performance of a pre-refueling outage, circulating water system heat treatment and to perform main steam safety valve testing. The Unit entered Mode 3 on June 4, 1993, and Mode 6 was reached on June 10, 1993, for the Cycle VII refueling outage.

Unit 3

The Unit began the inspection report period at 100% power. On May 28, 1993, power was reduced to 80% to allow for circulation water system heat treatment. On May 30, 1993, the Unit resumed full power operation and continued at 100% power for the remainder of the inspection period.

3. Operational Safety Verification (71707, 64704)

The inspectors performed several plant tours and verified the operability of selected emergency systems, reviewed the tag out log and verified proper return to service of affected components. Particular attention was given to housekeeping, examination for potential fire hazards, fluid leaks, excessive vibration, and verification that maintenance requests had been initiated for equipment in need of maintenance. The inspectors also observed selected activities by licensee radiological protection and security personnel to confirm proper implementation of and conformance with facility policies and procedures in these areas. A number of discrepancies were noted as discussed below.

a. Spent Fuel Pool Heat Exchanger Leaks In Unit 1

On May 27, 1993, the inspector observed a number of leaking components associated with the Unit 1 Spent Fuel Pool (SFP) cooling heat exchanger. In particular, leaks were noted from valves SFP-323, SFP-304, and SFP-305; the pipe cap downstream of valve SFP-355; and a flange to the heat exchanger. The inspector noted that the leaks were not contained and that deficiency tags were not attached to the equipment. The inspector discussed this with the shift superintendent who indicated that these leaks could have resulted from a heat treatment that was conducted several weeks earlier.

The inspector noted several days later that the licensee had

corrected the deficiencies. However, on June 22, the inspector toured the Unit again and noted that there were active leaks on valves SFP-304 and CCW-387 that were not contained and did not have deficiency tags attached. SFP-304 was one of the leaking valves identified on May 27. Discussions with the licensee indicated that the packing on valve SFP-304 was adjusted and the drip bag was removed prior to the inspector's tour on June 22. Health Physics (HP) management indicated that the drip bags were removed after packing adjustments were made and the leaks were believed to have been fixed. However, it appeared that no one had reevaluated the situation until the leaks were observed by the inspector.

On June 22, 1993, an NRC individual's shoe became contaminated while walking in a "clean" area near the Unit 1 component cooling water (CCW) and SFP equipment. As a result of this contamination, the licensee performed a survey and found that the CCW platform was contaminated with detectable levels of radioactivity (approximately 100 cpm for a large area massilin). A review of the licensee's logs indicated that one other individual's shoe had become contaminated on the sole by walking in "clean" areas of Unit 1. In addition, for a few weeks prior to June 22, several other individuals had caused portal monitor alarms, but surveys indicated that the contamination levels on their shoes were less than 100 cpm above background. The inspector learned that the licensee had a floor drain on the CCW platform back up with contaminated water several days earlier and that the licensee considered that this was possibly the source of contamination for the individuals.

The inspector was concerned that the licensee was not focusing adequate attention during tours, since the active leaks were not noted by Operations or other organizations until identified by the NRC. The inspector was also concerned that the licensee had not adequately controlled loose surface radioactive contamination in clean areas of Unit 1. The inspector will continue to monitor Unit 1 as part of the routine inspection effort as inspector followup item (IFI 50-206/93-11-13) .

b. Turbine Gantry Crane Side Boom Parked Over Components In Unit 2

On June 3, 1993, the inspector noted that the turbine gantry crane side boom was parked directly over Unit 2 feedwater block valve 2HV-4051. This condition was discussed with the Maintenance Manager, who performed an evaluation of the concern. He considered that there were sufficient and redundant brake controls such that there was an extremely low probability that the brakes would fail, causing the gantry crane side boom to impact any plant equipment. However, the Maintenance Manager indicated that the crane operators would review the procedure and determine if any enhancements should be made. The inspector considered that the licensee's response was adequate.

c. Control Of Tools In Contaminated Areas Of Units 1 and 2

On May 27, 1993, the inspector noted a number of tools that were lying across the contaminated barrier tape in the Unit 2 Train 'A' safety injection pump room, Room 5. These tools were apparently being used in conjunction with the modification to cross-connect the containment spray pumps with the SFP cooling system. However, the tools were not in use at the time. The inspector was concerned that these tools could be contaminated and could be easily mistaken for clean tools since they were half-in/half-out of the contaminated area. This concern was discussed with the licensee who promptly cleaned up the area.

On June 22, 1993, the inspector observed a number of potentially contaminated items in Unit 1. For example, two items noted were as follows:

- An unsecured hose was found half-in/half-out of a contaminated area near the refueling water filter pump.
- A rubber glove used with protective clothing (PCs) was found in a clean area under a safety injection pump.

These items were discussed with the licensee, who promptly cleaned up the area. The inspector will continue to monitor the licensee's efforts to control tools in contaminated areas as part of the routine inspection effort.

d. Control Of Equipment In A Seismic Exclusion Zone In Unit 3

On June 23, 1993, the inspector observed that a contractor tractor-trailer was parked very close to the seismic exclusion zone (SEZ) near Unit 3. The SEZ is used to store fire tanker trucks. The inspector observed that the tractor-trailer was parked more than 15 feet from the tanker seismic restraints, as required. However, the tractor-trailer appeared to be parked within the "two times the height" distance requirement specified on a nearby sign.

Discussions with the licensee indicated that they had performed three inspections of the area that week. However, it appeared that the tractor-trailer had been parked in the area after these tours had been performed. It also appeared that the signs in the SEZ were not understood clearly by the personnel responsible for choosing the parking site for the tractor-trailer with respect to the "two times the height" requirement.

As a result of the inspector's concern, the licensee indicated that they would have an hourly fire watch tour the area for several weeks to determine if there is a generic problem with equipment being parked within the SEZ, or whether this was an isolated occurrence. The licensee also indicated that they would review the signs used in

the area to determine if the requirements could be better defined to ensure that there would be no confusion. The inspector will review the licensee's evaluation of the control of equipment in and around the SEZ as unresolved item (URI 50-362/93-11-01).

e. Control Of Components Near Safety-Related Equipment In Units 1, 2, And 3

During the inspection report period, the inspector noted a number of components that were not properly controlled that could have become hazards to safety-related (SR) equipment during a seismic event. In particular, the following were noted:

- On May 17, 1993, a tool cart was left unsecured and unattended near an Emergency Safety Features cabinet behind the Unit 3 control room panels.
- On May 27, 1993, a ladder was propped up against a pipe while unsecured and unattended within a few feet of component cooling water pump P-025 transfer switch S21804ED005 in Unit 2 Room 23.
- On May 27, 1993, a gang box was left unsecured and unattended within 10 feet of component cooling water piping and transmitters in Unit 2 Room 23.
- On June 5, 1993, a ladder was standing unsecured and unattended next to Unit 2 auxiliary feedwater pump P141.
- On June 9, 1993, an unsecured and unattended ladder was stored in a scaffolding storage area in Unit 2 component cooling water pump Room 12.
- On June 9, 1993, scaffolding was left unattended and unsecured within one foot of component cooling water pump P-025 transfer switch S21804ED005 in Unit 2 Room 23.
- On June 16, 1993, a ladder was standing unsecured and unattended near reactor coolant loop 1A hot leg injection drain valve 2HV9437 on the 17 foot elevation of the Unit 2 containment.
- On June 16, 1993, an equipment box was secured to a safety-related electrical conduit, CABJ13, on the 45 foot elevation of Unit 2 containment (near column 5).
- On June 24, 1993, a ladder was standing unsecured and unattended behind 480V safety-related switchgear in the Unit 1 4160 VAC switchgear room. Discussions with the licensee indicated that the ladder may have been in the room for a few weeks. As corrective actions, the Unit 1 Superintendent indicated that during shift turnover meetings this incident

would be discussed and the expectations for performing thorough plant walkdowns would be re-emphasized.

Procedure S0123-XVIII-23, "Implementation Of Site Housekeeping And Cleanliness Controls," Paragraph 6.4.1.5, specified that "Excess materials or equipment (such as hoses, portable gear, tools, ladders, temporary power cables) not in use shall be removed or properly secured, in order to prevent it from becoming a hazard to safety-related equipment or personnel during a seismic event." In addition, procedure S0123-I-1.20, "Seismic Controls During Maintenance, Testing, and Inspection," Paragraph 6.1.1, required that "Precautions shall be taken to ensure that no tool, material, or any other item capable of damaging a safety-related piece of equipment be allowed to impact any safety-related equipment." It also stated in Paragraph 6.5.1 that "All storage containers, gang boxes, welding machines and/or power panels shall be anchored or secured as far away from safety-related equipment as practical when in safety-related areas except when being moved." In addition, in Paragraph 6.2.3, it required that "step ladders, when left unattended, shall be laid down on the floor and secured to a permanent civil structure or, if left standing, tied off in 2 places at or near the top to a permanent civil structure." These examples of failures to follow procedures for control of components near safety-related equipment were identified as violation (VIO 50-361/93-11-02).

The licensee responded to many of these concerns by performing a detailed evaluation. The results of the evaluation are discussed in Paragraph 7.b of this report.

f. Uncontrolled Fire Extinguisher In Unit 2 Turbine Lube Oil Tank Room And Unit 2 Containment

During a routine inspection on April 22, 1993, the NRC inspector observed an uncontrolled, unsecured fire extinguisher in the Unit 2 turbine lube oil tank room (fire area 2-TB-7-149), with a service tag that was out of date. The tag was dated both March 5 and 12, 1992. The inspector determined that the fire extinguisher had been stored in the B-62 Tool Room. The primary use of fire extinguishers checked out of that Tool Room was to accompany a flame permit for work involving a possible ignition source (e.g., welding, brazing, or grinding). Maintenance procedure S0123-I-1.41, "Flame Permits," Paragraph 6.6.1.2.5 required that personnel ensure that "the State Fire Marshall Service tag date has not exceeded one year." In addition, Paragraph 6.6.9 of that procedure stated, "At the completion of fire watch posting, return the fire extinguisher(s) to the Tool Room located at B-62." It appears that either the individual signing out the fire extinguisher failed to check the State Fire Marshall Service Tag or that the fire extinguisher had been unsecured since the service tag had expired. In either case, the fire extinguisher should have been returned to Tool Room B-62

after use, as required by procedure S0123-I-1.41.

In addition, during a routine inspection of the Unit 2 containment on June 22, 1993, the inspector observed welding activities performed under maintenance order (MO) 91101357000. The maintenance was being performed on Quality Class 1 equipment (high-point vent-valve safety injection combined header to reactor coolant loop 1A). The MO required that a fire watch be posted during welding activities and possess a fire extinguisher in accordance with the Flame Permit procedure. The inspector noted that the fire extinguisher had a State Fire Marshall Service Tag that was dated 1991, indicating the last time the fire extinguisher was inspected. The inspector questioned the fire watch about the out of date State Fire Marshall Service Tag. The fire watch said he had obtained the extinguisher from a group of extinguishers located near the exterior equipment hatch. The fire watch then went to obtain a fire extinguisher with a current State Fire Marshall Service Tag. The inspector determined that the extinguishers located near the equipment hatch were quarantined extinguishers removed from containment fire extinguisher cabinets that were awaiting removal from containment in order to be serviced.

The inspector considered both of these observations as instances of failure to follow procedures. However, no citation is being issued since the instances were of low safety significance, the licensee took prompt action to remove the uncontrolled extinguishers, and the criteria of Section V.G of the Enforcement Policy were satisfied, non-cited violation (NCV 50-361/93-11-03).

g. Technical Specification Fire Door Left Inadvertently Open In Unit 3

On June 15, 1993, the inspector noted that the Technical Specification (TS) fire door for Charging Pump 3P190 had been left open with no one in the room or nearby area, and no fire impairment identified. Technical Specification 3/4.7.9 required that all fire rated doors separating redundant equipment which could affect the ability to achieve and maintain safe shutdown in the event of a fire be operable and closed. However, this violation is not being cited since there was no safety significance, the door was promptly closed, it was an isolated occurrence, the fire doors for the other two charging pumps were closed, and the criteria of Section V.G of the Enforcement Policy were satisfied, non-cited violation (NCV 50-362/93-11-04).

h. Unit 3 Control Room Observations

On June 15, 1993, the inspector observed that Unit 3 control room reactor coolant system reference average temperature recorder 3TR-021B was not printing. After discussions of this issue with the assistant control operator (ACO), the inspector learned that the recorder had not been printing for several hours. The inspector

verified that the recorder was originally installed to be used for automatic rod control and therefore was not normally used other than for trending purposes and a historical record.

The inspector noted that procedure S023-0-44, "Professional Operator Development and Evaluation Program," Paragraph 6.1.3.7, required in part that "operators continually monitor plant conditions and aggressively resolve problems displayed by abnormal indications." Additionally, the inspector also noted that procedure S0123-0-3, "Control Operator's Responsibilities and Duties," step 6.2.8 required that "the Control Operator shall make periodic inspections of the Control Room instrumentation and controls. The frequency is dependent upon plant status and the conditions that prevail on the individual systems or components. In cases where very frequent observations are necessary, delegation to an ACO may be advisable." The inspector reviewed control operator logs for June 15, 1993, and noted that there were a significant amount of activities prior to the above observation which may have contributed to this concern.

The inspector considered this as an instance of failure to follow procedures. However, this violation is not being cited since appropriate actions were taken as a result of the inspector's observation, there was no safety significance, the recorder was not necessary for operating the plant, and the requirements of Section V.G of the Enforcement Policy were satisfied, non-cited violation (NCV 50-362/93-11-05).

i. Unit 2 Intake Structure Inlet Gate Pedestal Failure

On June 11, 1993, the inspector conducted a walkdown of the Unit 2 intake structure with the system engineer. During the tour the system engineer informed the inspector that the south stainless steel pedestal was missing from the bottom of safety-related weir gate number 3. This weir gate has two stainless steel pedestals and serves as gate stops to maintain sea water available to the salt water cooling system even on closure of the weir gate. This was apparently caused by the failure of the circular weld connecting the pedestal to a mounting plate which was secured to the bottom of the gate. The pedestal was discovered missing on June 8, 1993.

The inspector reviewed non-conformance report (NCR) 93060015, initiated to document this condition, and noted that the NCR did not include an operability assessment to justify the continued operation of the Unit 3 gate number 3 (which was of the same design). The inspector questioned cognizant personnel in the Station Technical (STEC) organization to determine if the operability of the Unit 3 gate had been addressed and was told that a formal evaluation had not been conducted, but there were no operability concerns regarding the Unit 3 gate.

Subsequently, the inspector noted that the licensee had performed in

March 1993 Design Calculation, C-255-3.14, Revision 2, "Intake Structure Design-Recirculation Gates," which provided justification that the number 3 recirculation gates operators on both Units would successfully withstand seismic II/I criteria loads. Additionally, to preclude the gate drive motors from inadvertently closing the gate, the licensee removed the power leads to the motors. During the next Unit 3 scheduled heat treat, tentatively scheduled for mid-July 1993 (when there would be a need to close gate 3), the licensee will conduct a test to determine the condition of the pedestals by driving the gate to its fully closed position.

The inspector reviewed procedure S0123-XV-5, TCN 3-17, "Non-conforming Material, Parts, or Components," and noted that there appeared to be no guidance on addressing the operability of similar components on the same or the other Unit when equipment deficiencies are identified. Additionally, in a Quality Assurance (QA) surveillance, SOS-359-92, QA investigated the history of non-conformance occurrences and related corrective actions for the Unit 2 and 3 RCP baffle bolt failures. It was noted by QA that, "NCR dispositions did not appear to consider applicability of the baffle bolt deficiencies to other components."

As a result of this review, the inspector had two concerns. First, upon initial questioning of cognizant STEC personnel it did not appear to the inspector that the operability of the Unit 3 number 3 gate had been considered. However, in subsequent discussions with the STEC group supervisor, the supervisor indicated that he had considered the operability of the Unit 3 gate and considered its failure non-credible, when made aware of the condition three days after it was discovered. Secondly, the inspector considered that the lack of formal guidance in procedure S0123-XV-5 may have contributed to delaying the assessment of the Unit 3 gate's operability until brought to the attention of STEC division management.

During the exit meeting of June 30, 1993, licensee management committed to review applicable procedures to implement improvements where warranted. The licensee also stated that they do have procedures, such as those used to conduct division investigations, which direct station personnel to consider operability of similar components in the same or other Unit. However, the inspector noted that division investigations usually take at least 30 days to be completed. The inspector will review this issue further as part of routine inspection activities.

Within this area inspected, two examples of one violation for failure to follow procedures, and three non-cited violations were identified.

4. Evaluation of Plant Trips and Events (93702)

a. Inadvertent Diesel Generator Start

On June 7, 1993, the Unit 2 diesel generator (DG) 2G002 was inadvertently started as a result of maintenance activities. In particular, the licensee was moving switches and indicators from the permanent control board to a temporary panel in accordance with construction work order (CWO) 92091615, "DCP 2-6605 Remove and Relocate 2HS1667A1 and 2HS1667B1 To Temporary Panel Position While DF2 Completes CR063 Modification in accordance with SO123-XXVI-6.14 (Circuit and Calibration Tests) on Scheme 2AZL160CAL." The CWO had been initiated to implement portions of Design Change Package 2-6605.08, "Control Panel 2/3 - CR063 Human Factor Modification."

The DG start occurred when a technician installed a jumper to test the safety injection actuation signal (SIAS) override switches, 2HS1667A1 and 2HS1667B1, without verifying that the DG was in maintenance lockout in accordance with the construction work plan (two steps prior to installing the jumper). The inspector noted that all steps in the construction work plan, except the step to verify that the DG was in maintenance lockout, required the technician to sign and date that the step was performed.

The Nuclear Construction (NC) Organization performed a division investigation as a result of this event, Report 93-001, and considered the root cause to be "inattention to detail and weak human factor document planning." To prevent reoccurrence, all NC technicians were briefed on this event. Retraining for formality and attention to detail was given to all technicians within NC. This training was developed from NUREG-1192, "Wrong Unit or Wrong Train Events," and the licensee's Maintenance Self-Checking Program and Circuit and Calibration Testing procedures. In addition, the Unit 2 CWOs were enhanced to provide better guidance. The NC organization indicated that the Unit 3 CWOs will be modified in the same manner. Additionally, administrative procedure SO123-XXIX-2.14, "Construction Work Orders," will be modified to enhance attention to detail by adding guidance such that when function test objectives, conditions and instructions are listed, a signature and date line are required. The inspector reviewed the report and considered that the licensee's assessment and corrective actions were adequate. The inspector considered this failure to follow the CWO is a violation. However, this violation is not being cited since there was low safety significance, there was no damage to the DG or other components, and the requirements of Section V.G of the Enforcement Policy were satisfied. Non-cited violation (NCV 50-361/93-11-06).

b. Damage To Reactor Coolant Pump Seal Heat Exchangers

On June 10, 1993, the licensee performed a hydrotest of the CCW

system piping. As a result of that hydrotest, the reactor coolant pump (RCP) seal heat exchangers were overpressurized to 3150 psig and damaged. The actual heat exchanger design pressure was approximately 90 psia. The licensee completed a preliminary investigation which indicated that the test boundary selected for the hydrotest incorrectly included the heat exchangers. The licensee determined, through discussions with Combustion Engineering (CE), that the heat exchanger design specification was not properly worded and the internal design pressure could be misinterpreted to be 2500 psig. In addition, the drawings and specifications were ambiguous on the American Society of Mechanical Engineers (ASME) status of the heat exchanger. For example, the piping and instrument diagram (P&ID) indicated that the heat exchangers were "non-ASME" while other documentation indicated that they were ASME. For corrective action, the licensee was in the process of replacing the heat exchangers, was planning to revise the documentation discrepancies, review future hydrotests for similar problems, and review previous hydrotest to determine whether past tests had subjected components to pressures higher than its design pressure. The inspector considered the licensee's actions adequate.

c. Potential Enforcement Discretion For Inadequate Boron Concentration

The licensee contacted the NRC on June 9, 1993, indicating the potential need for enforcement discretion due to unequal boron concentration in two portions of the Unit 2 reactor coolant system (RCS). At the time, Unit 2 was in Mode 5 (shutdown for a refueling outage) with the reactor coolant pumps secured, fuel assemblies in the reactor vessel, and the licensee was making preparations for entry into Mode 6. The licensee considered a need for enforcement discretion to be based on the failure to comply with Technical Specification 3.9.1 for maintaining a uniform boron concentration in the filled portions of the RCS when the plant entered Mode 6. The licensee considered the additional dose required to restore the plant to conditions in which the reactor coolant pumps (RCPs) could be started to obtain adequate mixing was not warranted. The licensee later determined that enforcement discretion was not needed because additional RCS sampling had shown that the boron concentration in the RCS was in fact uniform.

The licensee initially believed that there was inadequate mixing of the RCS when the RCPs were secured prior to completion of boration to the Mode 6 value. This apparently occurred due to a lower boration rate subsequent to a design change which resulted in the licensee reaching the point in the procedure for turning off the reactor coolant pumps before boration to Mode 6 conditions was achieved. The procedure lacked appropriate guidance for securing the RCPs during boration activities.

The NRC considered that it was appropriate that the licensee brought this issue to the NRC's attention early. However, the NRC also

considered that the licensee could have evaluated the details in more depth prior to bringing it to NRC management attention. For example, it was not evident that the licensee evaluated the condition of a slug of lower concentration borated water in the intermediate RCS loop and its effect on the reactor during start of an RCP upon completion of refueling activities. The licensee responded to this concern by indicating that they believed that they had assessed the condition thoroughly and that they would fully comply with the TS when they returned to Mode 6 later in the outage.

The licensee indicated that it would perform a division investigation to determine the root cause for the event. The inspector will review the licensee's investigation including their assessment of procedural adequacy upon completion, inspector followup item (IFI 50-361/93-11-07).

d. Unit 3 Steam Generator E088 Leakage

On May 11, 1993, Unit 3 operators noted increased activity levels on condenser air ejector radiation monitors 3RT78970 and 3RT7818. Abnormal Operating Instruction S023-13-14, "Reactor Coolant Leak" was subsequently entered. Chemistry confirmed that there was a primary-to-secondary leakage for steam generator (SG) E088, based on the analysis of air ejector activity with leakage calculated at 2.35 gallons per day (gpd). On May 15, 1993, the leakage increased over a short period of time to 7 gpd. The SG leakage gradually increased to approximately 14 gpd by the end of the inspection period.

The licensee indicated that operators had been trained and sensitized to the importance of prompt determination of leakage and leakage increases. Additionally, in accordance with procedure S023-13-14, operators have guidance such that any increases in leakage that would equate to greater than 60 gpd within any one hour would require plant shutdown. The inspector will continue to monitor licensee activities in this area under routine inspection activities.

e. Plant Protection System/Plant Monitoring System Unexpected Interface

On May 18, 1993, the licensee identified that the nonsafety-related plant monitoring system (PMS) shared common wiring with the safety-related plant protection system (PPS). This discovery resulted from spurious actuations of low SG water level alarms and channel trips generated during semi-annual surveillance testing on April 28 and May 6. The trips were detected by the control room annunciators and PPS panel indications.

The licensee determined that these actuations were generated as a result of the method used by technicians to install instrument card-edge connectors to circuit cards in the PMS cabinets. The result of this improper action was a short in the PMS circuit cards which

resulted in a PPS bi-stable actuation of the low SG water level pre-trip and trip. Subsequent evaluations by the licensee identified that the low SG pressure trip was susceptible to the same type of failure. Additionally, a similar problem, with a different effect, could result in the prevention of the containment high pressure channel from detecting a high containment pressure condition. However, the licensee indicated that this particular containment high pressure trip was not credited in their Final Safety Analysis Report (FSAR) Chapter 15 analysis.

Immediately upon discovery of this condition, the licensee removed the common wiring shared by the PMS and PPS to preclude future inadvertent actuations. The purpose of the wiring connecting the PMS to the PPS had been to provide setpoint indication to the plant computer of the various reactor trip setpoints monitored by the PPS. The licensee noted that the setpoints did not provide a safety-related function and were not normally used by operators. In addition, the need to evaluate long term corrective actions was identified.

The licensee stated that they had received NRC Information Notice, IN 84-86, "Isolation Between Signals Of The Protection System and Non-Safety-Related Equipment," and CE Infobulletin 84-14, "Plant Protection System/Plant computer Interface." These notices identified the potential for creating fault paths through inadvertent interconnection of common components, but they did not identify this condition. The inspector acknowledged the licensee's comment that the CE Infobulletin may have mislead readers because statements were made which indicated that as long as original design principles were retained, the CE-supplied PPS/PC (plant computer) systems reviewed were appropriately isolated. The licensee also indicated that as a result, a reevaluation of IN 84-86 was initiated.

At the end of the inspection period the licensee had not completed their operability assessment to determine reportability of the conditions as found on April 28 and May 6, 1993, and was in the process of reevaluating IN 84-86. The inspector will review these evaluations as inspector follow-up item (IFI 50-361/93-11-08).

f. Acoustic Monitor Mounting Brackets Missing In Units 2 and 3

On May 28, 1993, the licensee declared the four acoustic monitors in each of Units 2 and 3 inoperable due to not having power supplies/converters seismically secured (i.e., missing brackets). The condition was discovered while a design engineer from the Nuclear Engineering Design Organization (NEDO) was performing a walkdown of the affected components with the missing brackets in order to resolve an long outstanding site problem report (SPR) that was initiated in 1992. The engineer realized that an NCR should have been written instead of an SPR, and the acoustic monitors were

declared inoperable shortly thereafter.

The brackets were installed on June 5, 1993, and the acoustic monitors were declared operable. For followup actions, the licensee was going to perform an assessment of the event as discussed in Paragraph 6.b.

Within this area inspected one non-cited violation was identified.

5. Bi-Monthly Surveillance Activities (61726)

During this report period, the inspectors observed or conducted inspection of the following surveillance activities:

Observation of Routine Surveillance Activities (Unit 2)

S023-I-2.5, "Valve Surveillance Testing Of Main Steam Safety Valves."

S023-I-2.56, "Spent Fuel Handling Machine Surveillance."

S023-I-2.17, "Surveillance-Refueling Machine Load Test And Automatic Load Cutoff And Auxiliary Hoist Automatic Load Cutoff."

No violations or deviations were identified.

6. Monthly Maintenance Activities (62703)

During this report period, the inspectors observed or conducted inspection of the following maintenance activities:

a. Observation of Routine Maintenance Activities (Unit 1)

M093031481000, "Inspect Air Tank To Update California Pressure Vessel Permit #35205-77."

M093040046000, "Unit 1 South Diesel Generator Lube Oil Filter Has A Minor Oil Leak At Cover Gasket."

b. Observation of Routine Maintenance Activities (Unit 2)

M093051816000, "Support Maintenance Order For Investigation Of Seismic Brackets For PAMI (Post-accident Monitoring Instrumentation) Installation."

M091101357000, "Welding On Safety Injection Combined Header Valve To Reactor Coolant Loop 1A."

M090101512000, "Combined MOVATS (Motor-operated Valve Analysis and Testing) and Environmental Qualification Preventive Maintenance On Containment Isolation Valve 2HV6371."

MO93031656000, "Provide Temporary Power From Disconnect 2D018 To 2CR050."

The inspector noted some positive observations and some discrepancies with several of the maintenance activities reviewed as follows:

- The inspector observed portions of MO90101512000 conducted on May 28, 1993. This MO was performed in preparation for design basis testing of 2HV6371 in accordance with the licensee's Generic Letter 89-10 program. The inspector observed that, in preparation for this testing, two strain gauges had been attached (one to the valve yoke and one to the valve stem) and calibrated. The inspector considered that the incorporation of strain gauges directly into the measurement of valve thrust, instead of using spring packs only, was a strength. The inspector also observed that pressure gauges had been installed to measure system delta pressure in close proximity to the inlet and outlet of the valve. The inspector was informed that system flow would be measured using control room indications. These preparations were made to support the design basis testing of the valve. The inspector considered this would provide the licensee with an adequate means to measure these parameters and would provide the licensee with the ability to extrapolate design basis delta pressure and flow values. The inspector also considered the placement of these pressure gauges and the measurement of flow as a strength. The inspector had no concerns based on those portions of this MO that were observed.
- The inspector observed work on June 3 to install brackets on the Unit 2 acoustic monitors in accordance with MO93051816000. In general, the work was performed satisfactorily. However, it appeared that configuration control over the brackets had not been maintained. In addition, the inspector noted an apparent weakness in that the licensee had issued an SPR over a year earlier on this issue, but it was not elevated to an NCR. If it had, more timely resolution would have occurred. In addition, a similar condition was identified with equipment in Unit 1 several years ago.

The licensee indicated that the Projects division would perform an evaluation of what happened to the brackets. In addition, STEC initiated an Division Investigation to determine why the SPR was written instead of an NCR. Also, NEDO was in the process of reviewing whether this condition was outside the design basis. The inspector will review these evaluations as unresolved item (URI 50-361/93-11-09).

- On June 15, the licensee attempted to close the containment equipment hatch during preparations for core alterations. The

inspector observed that the licensee could not close the hatch because the breaker in containment for Box 2D018 would not close to energize power to the hatch hoists. Previous to this, the licensee had issued MO 93031656000 for Bechtel Maintenance Support Group (MSG) to provide temporary power to the containment jib crane instead of using welding receptacle power. The licensee did this to ensure that redundant power would be available to the jib crane in order to move the containment equipment hatch ramp out of the way should it become necessary. However, the way the temporary cable was installed prevented the box cover from being closed and also prevented breaker manipulation. (Unit 3 has manual override capability for the equipment hoists. However, this capability has not been installed in Unit 2 as of this report period.)

The inspector was concerned that the licensee might not have been able to close the hatch easily if a loss of shutdown cooling (SDC) had occurred. In fact, it took them several hours to determine the problem and shut the breaker. At the time, the reactor had been shut down for over a week and there would have been a many hours before boiling in the core would have occurred if SDC had been lost. However, at the time that the temporary modification was installed, June 5, 1993, there would have been as little as 82 minutes until core boiling. The licensee indicated that they would have taken expedited actions to restore power to the equipment hatch hoists should it have been necessary.

The inspector noted that for the period of concern, both SDC trains were operable. However, on June 5 through June 7, 1993, the main and unit auxiliary transformers were cleared, and on June 9 through 17, 1993, the reserve auxiliary transformer was cleared such that only one off-site source of power was available for almost the entire period. In addition, from June 10 to June 15, 1993, emergency diesel generator (EDG) 2G003 was inoperable for maintenance.

It appeared that the safety significance of this condition was low since at least two events were necessary to pose a potential for any event of safety significance (i.e., boiling in the core). However, the inspector was concerned that this temporary modification appeared to be not well-controlled nor well-planned before it was installed. In addition, the licensee was performing an investigation to determine what caused this problem. This item will remain open pending the outcome of the licensee's evaluation as unresolved item (URI 50-361/93-11-10).

c. Observation of Routine Maintenance Activities (Unit 3)

M093040485000, "Metal Shavings Found In CCW Valve 3HV6371 Due To Lack Of Proper Actuator Lubrication."

M093051381000, "Troubleshoot Valve Failure Mechanism - Valve Failed To Stroke From The Control Room."

No violations or deviations were identified.

7. Plant Modification and Refueling Activities (37700, 37828, 60705, 60710)

The inspector reviewed the licensee administrative requirements for control of refueling operations and plant conditions during refueling. The inspector also reviewed refueling activities to determine if they were being controlled and conducted as required by TS and approved procedures.

The inspector discussed requirements and responsibilities with various refueling personnel and concluded that the licensee's administrative requirements and controls over refueling activities were adequate. For example, the licensee had established clear lines of supervision, QA/QC observation of key activities were being performed, foreign material exclusion (FME) and cleanliness controls were established, and abnormal operating instructions were available if problems should be encountered. The licensee performed the requisite surveillances of the spent fuel handling and refueling machines. The inspector interviewed key licensee and contractor personnel who were aware of their responsibilities, understood administrative requirements, and felt capable of responding to events. The inspector observed site management involvement in refueling preparations. In addition, the inspector noted that the licensee had established a refueling supervisor position along with approximately five assistants to support work activities in containment. In general, the inspector considered that the refueling activities were proceeding well.

The inspector observed the activities of the refueling supervisor and assistants and considered that their assistance was aiding in the satisfactory completion of work. The inspector also noted that in many cases activities were completed satisfactory and adequately controlled. However, some difficulties were observed, such as follows:

a. Responsibilities of the Refueling Supervisor

The inspector observed that during evolutions to remove the core internals and in-core instrumentation, the refueling supervisors were very busy dealing with problems that developed during the evolutions. The inspector noted that one consequence of this work load was that the refueling supervisors were not performing the shiftly inspection of containment specified in Attachment 8 to procedure S023-I-3.1, "Minor Refueling Procedures." This review had

not been performed on June, 9, 10, or 15, 1993. In addition, it had only been performed during one shift per day on June 8, 11, 12, and 13, 1993.

The inspector discussed this observation with the refueling manager who indicated that management expected this inspection to be completed each shift, even though it was not specifically required by procedure. In fact, the inspector noted that there was conflicting guidance in the procedure about performing these inspections. Step 1.2 indicated that Attachment 8 was an optional form that may be utilized at the discretion of the refueling shift supervisor. However, Attachment 8 indicated that the refueling shift supervisor "shall" inspect the areas specified.

The procedure required that the refueling supervisor inspect a number of items on the 63 foot elevation of containment such as the cavity seal and spent fuel handling building air supplies, FME, fire extinguisher status, scaffolding, loose items, cleanliness, and other items. The refueling manager indicated that these inspection items were redundant to requirements in other procedures. In effect, Attachment 8 just provided an additional check to ensure that no problems existed (essentially serving as another set of eyes). In fact, the inspector noted that the requirements to ensure that the cavity seal and gate seal pressurization systems were adequately pressurized were performed by ACOs as specified in Attachment 1 to procedure S023-0-9, "Routine Rounds And Inspections."

Licensee management indicated that it was their expectation that this shiftly surveillance would be performed once the plant was in a condition that warranted implementation of the checklist. Licensee management also indicated that they would reinforce this expectation with the refueling personnel. The inspector checked the logbook several days later and noted that the refueling supervisors were performing the shiftly surveillances. The licensee indicated that they would review procedure S023-I-3.1 to determine if any enhancements were warranted. The inspector considered the licensee's actions appropriate.

b. Control Of Scaffolding Near Safety-Related Equipment In Unit 2

The inspector noted that there were several anomalies with scaffolding associated construction and maintenance activities in Unit 2. In particular, the following problems were noted:

- A U-clamp attaching a ladder to scaffolding above Train B CCW valve 2HV6228A had only one wing-nut and bolt securing it in place. The nut and bolt may not have been sufficient to ensure the ladder was adequately secured. This scaffolding was installed by maintenance personnel, and had the requisite housekeeping tag and the Occupational Safety and Health Act

(OSHA) green tag. However, the inspector was concerned that the U-Clamp could have broken loose, with the potential for injury to personnel. However, the inspector was not able to determine if the nut was missing before or after the scaffolding was tagged as being acceptable.

- Scaffolding erected in an alcove of the 'B' Train safety injection equipment room over valve S21206MU018, "LPSI (low pressure safety injection) to SFP Cross Connect" to support a modification did not appear to be erected per procedure. The inspector noted that Construction Specification CS-C20, "Scaffolding Erection Criteria" specified that scaffolding required horizontal tie-off for lateral load resistance and/or prevention of over-turning if the over-turning may affect the safety-related equipment. The procedure specified that "There should be at least one tie-off per 5 foot length of scaffolding in each horizontal direction. The tie-offs should be equally spaced. The horizontal tie-offs will be at or near the working level and attached to walls, columns or main beams. The horizontal ties should be spaced maximum of 6 feet apart vertically." However, the inspector noted that the scaffolding did not have any horizontal tie-off and that the first tie-off in the vertical direction was approximately 15 feet overhead. The inspector discussed this with the responsible engineer who indicated that the scaffold did not appear to be constructed in accordance with the procedure.

The licensee responded to the inspector's concern by indicating that it appeared that some scaffolding requirements related to CS-C20 were not easily understood. In fact, the licensee considered that there was no potential for the scaffolding to overturn in this case since it was "entrapped" by the surrounding walls of the alcove. The inspector was concerned that the guidance provided in CS-C20 could create confusion as to what the specific requirements were for erecting scaffolding over safety-related equipment. As a result of this concern, the licensee indicated that Nuclear Construction intended to further clarify the wording in CS-C20 during the upcoming biannual procedure review. The inspector considered the licensee's response adequate.

- On May 28, 1993, the inspector observed work being performed in the Unit 2 Train 'B' safety injection pump room. In particular, work on design change package (DCP) 2-6863, "Containment Spray Cross-Tie," was being performed in the room when an interference was encountered with installed scaffolding. As a result, Bechtel craft modified the scaffolding by replacing the interfering vertical member with two additional members. The new members were supported from pipe supports S2-SI-004-H-0017 and 0019. However, no engineering evaluation of the modification was performed to

determine if the load addition was acceptable.

Procedure S0123-XXVI-14.120, "Construction/Testing Aids," Paragraph 6.2.1.2, requires that "A new Scaffolding Safety Evaluation shall be generated for revisions to existing scaffolding unless the modifications are minor in nature and do not extend the scaffolding over any additional plant equipment." This failure to follow procedure was identified as a violation (50-361/93-11-11).

As a result of the concerns with these scaffolding discrepancies and housekeeping discrepancies discussed in Paragraph 3.e of this report, the licensee corrected the discrepancies and informed the inspector that the majority of the discrepancies involved Bechtel contract workers who were not implementing housekeeping and seismic restraining procedures properly. A number of additional corrective actions were taken by the licensee, including briefings with all Bechtel laborers and carpenters to address the requirements for housekeeping and seismic restraint of components near SR equipment. In addition, the licensee indicated that a meeting was held with senior Bechtel management to emphasize the importance of adherence to procedures, and the need for supervisory involvement to ensure compliance with procedures. Additional management oversight was focused on outage activities and increased monitoring was to be performed by the Nuclear Oversight Division (NOD). A seismic evaluation was performed of the safety-related containment spray system supports, calculation number P-450-1.44, which determined that there was no excessive load to the pipe support. A walkdown of other Bechtel constructed scaffolds in safety-related areas did not reveal any further discrepancies. The licensee also indicated that two independent Bechtel managers were brought in to review Bechtel work practices, man-loading, and training.

Within this area inspected, one violation was identified.

8. Independent Inspection (60710, 40500, 92703)

a. Fuel Assembly Leakage Inspections - Unit 2 (60710)

On June 21, 1993, the nuclear fuels analysis (NFA) group within the NEDO documented an analysis for the location of failed fuel rods in the Unit 2 Cycle VI core. The analysis concluded that there was only one failed fuel rod in the batch G assemblies. Additionally, since only one of the 108 Batch G fuel assemblies will be returned to the reactor core for Cycle VII (the estimated probability that the assembly being returned to the core contained the failed pin was 2 percent), fuel pin inspections of other batches being returned to the core for leaks would not be performed. The inspector considered the licensee's assessment adequate.

b. Commitment Tracking And Followup (40500)

The inspector reviewed the licensee's efforts to implement commitments from past inspection issues in 1991 and 1992. The inspector reviewed inspection report issues, reviewed the licensee's regulatory commitment tracking system (RCTS), and compared these commitments to the licensee's actions for approximately 30 items. The inspector noted that the appropriate items were tracked by RCTS. In addition, the licensee had followed through on all of those commitments with the exception of one. In particular, Inspection Report 50-206/92-03 concerning an enforcement conference for inoperability of the Unit 1 4160 VAC electrical switchgear room halon system documented the licensee's commitment to perform an assessment of the maintenance planner workload and staffing in early 1992. However, upon review, the inspector could not find that this had been performed. This item will remain open pending the licensee's assessment if action on this commitment had been performed as inspector followup item (50-206/93-11-12).

c. Shutdown Risk Evaluation (40500)

The inspector reviewed the April 1993 Nuclear Safety Group (NSG) report. This report contained a summary of the activities of the NSG during that period. The inspector noted that the licensee performed a risk assessment of the Unit 2 Cycle VII refueling outage. The NSG calculated the probabilities of inventory boiling and fuel damage during various stages of the outage. This assessment concluded that the risk would be acceptably low and several recommendations for plant equipment configurations to keep the risk low were provided to Outage Management for incorporation in the outage plan. The inspector had observed this effort during past outages and considered that it remained as a licensee strength.

d. Nuclear Oversight Division Outage Activities (40500)

The inspector reviewed the site quality assurance outage activities that included performance-based observations of various activities. The inspector noted that the QA inspectors had several good observations and issues. This effort was considered to be a strength.

e. Followup To NRC Bulletin 93-02 (92703)

The inspector reviewed the licensee's June 10, 1993, response to Bulletin 93-02, "Debris Plugging Of Emergency Core Cooling Suction Strainers." The response indicated that the licensee had performed a review and concluded that: "...Units 2 and 3 do not have fibrous air filters or other temporary sources of fibrous material installed or stored in containment during Modes 1 through 4 that would cause plugging of the emergency core cooling system (ECCS) suction strainers during a loss of coolant accident (LOCA)." However, the

inspector noted that there were fibrous air filters installed in the containment normal air coolers as part of the original plant design. As a result, the inspector discussed this condition with licensee personnel who indicated that they were aware that there was fibrous material in containment, but confirmed that through an engineering evaluation and design review committee that the fibrous material would not be impacted during a LOCA and that no significant quantities of material would reach the ECCS suction strainers.

The inspector was concerned that the licensee's June 10, 1993, response did not give a clear assessment of the conditions in containment or an indication of how the licensee concluded that there was no fibrous material installed that would cause plugging of the ECCS suction strainers. The inspector discussed this issue with personnel from the Office of Nuclear Reactor Regulation (NRR) who had a similar interpretation of the Bulletin response. The NRR personnel did not see a need for the licensee to revise their response to the Bulletin. However, they agreed that a clearer description of the condition, along with some basis for their conclusions would have been appropriate so that the NRC could better understand the conditions in the plant. This was discussed with the licensee, who indicated that they understood the conditions when preparing the response, but they agreed the response could have been clearer.

No violations or deviations were identified.

10. Follow-Up of Previously Identified Items (92701)

- a. (Closed) Follow-up Item (50-206/92-03-03), "Failure to Correctly Re-configure Battery Charger Output Cables Following Routine Surveillance."

This item referred to a licensee-identified problem in which maintenance personnel failed to properly re-configure the output cables of a safety-related battery charger following a routine surveillance test in December 1992. The improper configuration was reviewed by both a quality control (QC) inspector as well as a maintenance supervisor (MS). However, both individuals failed to identify the error.

The inspector's initial evaluation of this non-conformance issue identified no continuing safety concern in view of the licensee's self-identification and immediate corrective actions. However, the licensee's long term corrective action were to be evaluated during later inspections.

Corrective actions initiated by the licensee included performing a "Charger A-Output Cable Ampacity Analysis", calculation No. A-92-E-003. This analysis concluded that, although the 'A' charger was degraded with only one set of cables in place, the single set of

cables had sufficient capacity to provide for worst-case current loading. The inspector reviewed this calculation and found it to be acceptable. Other actions taken by the licensee included reviewing the incident with all plant electricians, reviewing the implementation of the plant's "Maintenance Stop/Self Checking Program", reviewing QC procedures, and considering disciplinary action. The inspector reviewed the licensee additional corrective actions and found them to be acceptable. No additional operability concerns were identified. This item is closed.

b. (Closed) Follow-up Item (50-206/92-34-01), "Station Service Transformer (SST) Voltage Evaluation".

This item referred to a licensee-identified deficiency related to the incorrect selection of a transformer tap setting on the Unit 1, No. 3, 4160 V/480 V station service transformer (SST) (there are four SSTs at Unit 1). The licensee determined that the transformer had operated an entire fuel cycle using the incorrect transformer tap setting. Specifically, the 'A' tap setting (4368 V/480 V) was used. This setting was normally used during operation in Modes 5 or 6. This setting was selected for Modes 5 and 6 operation because 480 V system loading during these modes tended to be lighter than during higher modes of operation.

The licensee evaluated the problem and found that the reason they failed to select the 'B' tap setting prior startup was attributed to inadequate management of the transformer tap setting program. The licensee identified a general weakness with implementation of the program due to "fragmented responsibility and accountability." Informal authorization was given to allow the tap selection to remain at the 'A' position, contrary to established procedure. With the 'A' tap setting incorrectly selected during operation in Modes 1 through 4, system voltage potentially could have been driven to below required levels as a consequence of heavier equipment loading.

The inspector's original evaluation of this nonconformance issue identified no continuing safety concern for Unit 1 in light of the fact that the unit is being decommissioned. However, there was a concern regarding equipment operability during the preceding operating cycle. The licensee conducted an operability evaluation covering the period that the incorrect tap setting had been selected. The licensee's evaluation determined that, for both normal and accident scenarios, voltage would have been below nominal levels. However, the licensee concluded that voltage levels would have been sufficient to maintain all required equipment in an operable status. The licensee's evaluation and conclusions were reviewed by the inspector. No further concerns were noted and this item is closed.

11. Exit Meeting

On June 30, 1993, an exit meeting was conducted with the licensee representatives identified in Paragraph 1. The inspectors summarized the inspection scope and findings as described in the Results section of this report.

The licensee acknowledged the inspection findings and noted that appropriate corrective actions would be implemented where warranted. The licensee did not identify as proprietary any of the information provided to or reviewed by the inspectors during this inspection.

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