U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION V

Report No. 50-312/80-06

Docket No. 50-312 License No. DPR-54 Safeguards Group ____

Licensee: Sacramento Municipal Utility District

P. O. Box 15830

Sacramento, California 95813

Facility Name: Rancho Seco

Inspection at: Herald, California (Rancho Seco Site)

Inspection conducted: February 1-29, 1980

Inspectors: for Harvey L. Canter, Senior Resident Inspector — March 14, 1980 Date Signed

John Elin, Reactor Inspector — March 14, 1980 Date Signed

William Wagner, Reactor Inspector — 3-14-80 Date Signed

Approved by: for Robert Dodds, Chief, Reactor Engineering Support Section — 3-14-80 Date Signed

B. H. Faulkenberry, Chief, Reactor Projects Section #2, Reactor Operations and Nuclear Support Branch — March 14, 1980 Date Signed


Summary:

Inspection between February 1 and 29, 1980 (Report No. 50-312/80-06)

Areas Inspected: Routine inspections of preparation for refueling activities; refueling activities; LER follow-up; facility modifications; and, independent inspection effort. The inspection involved 100 inspection-hours by the NRC Senior Resident Inspector and 30 inspector-hours by three NRC Region V based inspectors.

Results: No items of noncompliance were identified in four areas. Five items of noncompliance were identified in one area (LER followup).


RV Form 219 (2)

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# DETAILS

1. Persons Contacted

*J. Mattimoe, Assistant General Manager and Chief Engineer
*S. Anderson, Associate Nuclear Engineer
D. Blachly, Mechanical Engineer
*N. Brock, Electrical/I&C Maintenance Supervisor
T. Cassidy, Dresser Corporation Technician
*R. Colombo, Technical Assistant
*G. Coward, Maintenance Supervisor
*B. Daniels, Supervising Electrical Engineer
*J. Dunn, Manager of Transmission and Distribution
*W. Ford, Operating Supervisor
M. Fratt, Level II Inspector, Mobile Inspection Services
*H. Hechert, Engineering Technician
*W. Latham, Assistant General Manager, Operations
*R. Miller, Chemistry/Radiological Supervisor
*P. Oubre, Plant Superintendent
*D. Raasch, Manager of Generation Engineering Department
*L. Schwieger, Quality Assurance Director
*J. Sullivan, Quality Assurance Supervisor
*B. Wichert, Mechanical Engineer

The inspectors also interviewed and talked with other licensee employees during the course of the inspection. These included shift supervisors, reactor operators, auxiliary operators, maintenance personnel, security personnel, plant technicians, engineers, quality assurance personnel, and quality control personnel.

*Denotes those attending the Exit Interview on March 5, 1980.

NRC Region V personnel attending the March 5, 1980 Exit Interview included the following:

R. H. Engelken, Director, Region V Office of Inspection and Enforcement

J. L. Crews, Chief, Reactor Operations and Nuclear Support Branch

B. H. Faulkenberry, Chief, Reactor Projects Section 2,
Reactor Operations and Nuclear Support Branch

A. D. Johnson, Reactor Inspector/Enforcement Coordinator

2. Preparation for Refueling Activities

Rancho Seco was shut down for refueling on January 12, 1980. In preparation for witnessing fuel movements the resident inspector examined STP 221, "REFUELING RANCHO SECO FOR CYCLE 4."

This procedure received the group Supervisor's approval on January 22, 1980. The inspector's comments resulting from this examination were given to the licensee and resolved to the inspectors satisfaction.

Fuel handling procedures were available for both incore and ex-core fuel handling activities.

No items of noncompliance or deviations were identified.

3. Refueling Activities

Prior to the licensee's completion of fuel handling within the core, the inspector examined surveillance testing records of technical specification related tests to verify completion.

The inspector witnessed fuel handling operations during several different shifts. All activities witnessed appeared to be in accordance with technical specification (TS 3.8) and approved procedures. Containment integrity appeared to be appropriate.

Housekeeping around the fuel canal and spent fuel pool was adequate.

Licensee staffing was noted to be in accordance with the technical specifications and applicable procedures.

During the fuel movement portion of the refueling and maintenance outage an item occurred which is of concern. This involved several fuel assemblies that had their spacer grids torn, causing small pieces of grid material to fall into the lower vessel internals area.

This problem resulted when fuel assemblies that had experienced fuel burnup induced bowing were removed and/or inserted into the core and physically contacted adjacent assemblies causing the spacer grids to hang up and subsequently tear. Using an underwater TV camera the licensee observed a few pieces in the lower internals area. These have been removed with the use of a vacuum hose. The licensee believes that pieces that were removed are representative of other pieces remaining in the core which they have been unable to see. The licensee estimates that as much as three square inches of spacer grid material, involving seven separate pieces, with the largest piece being 1/2" X 1 3/16", may be in the lower internals area. The licensee contacted B&W with regard to this problem and B&W had told the licensee that this amount of material will not adversely affect operational safety. Region V referred this problem to IE:HQ's and NRC for their review. Both NRC and IE:HQ's informed Region V that, based upon the B&W analyses and discussions with licensee personnel, no items of safety concern were identified.

No items of noncompliance or deviations were identified.

4. Licensee Event Reports (LERs)

The inspectors conducted an initial screening, examination and followup on the following licensee event reports. This inspection activity was conducted in part to determine whether the reporting requirements had been met, the reports were adequate to assess the event, the cause appeared accurate and was supported by report details, corrective were are appropriate to correct the cause, generic aspects of the events had been considered, and the LER forms were complete and accurate.

The following LER's were reviewed:

a. LER 79-23 (dated 1-14-80) Closed

Improper Valve Lineup on the Discharge Cross-Tie Valves between the A and B High Pressure Injection Loops.

b. LER 79-24 (dated 1-25-80) Closed

Improper Valve Lineup on the Nuclear Service Raw Water supply and discharge to Make-Up Pump Coolers (oil and room) while the "B" High Pressure Injection Pump was out of service.

c. LER 80-03 (dated 2-6-80) Open

Improper Electrical Lineup to the Make-Up Tank Isolation Valve, SFV 23508.

d. LER 80-06 (dated 2-18-80) Open

Failure of a stud in a Stainless Steel Valve Manufactured by the Anchor Valve Company.

A number of significant issues were developed during the review of these LERs. The following is a discussion of the significant issues associated with each LER. These items were discussed in the Exit Interview held on site on March 5, 1980. (see paragraph 8)

e. LER 79-23

On December 17, 1979 at 8:45 PM, the High Pressure Injection Pump P-238B ("B" HPI Pump), was declared out of service to facilitate fire protection modifications per Amendment 19. The Make-up Pump P-236 was valved in to replace the "B" HPI Pump. However, under the Shift Supervisor's direction an improper valve lineup was made. Contrary to procedure A-15, Revision 8, both SIM 038 and SIM 039, the discharge Cross-Tie valves between the A and B High Pressure Injections Loops, were closed. At 9:03 PM on December 27, 1979 the problem was discovered and corrected by opening valve SIM-038. The discovery was made by an operator while he was performing a surveillance test.

Procedure A.15, "Make-up, Purification and Letdown System" states in step 7.15.5 to "close HPI Pump P-238A suction header isolation valves SIM-054 and SIM-055 and discharge header isolation valve SIM-039."

This procedural step was not implemented properly in that SIM-038 was also closed. Technical specification 6.8.1 states in part "Written procedures shall be estaolished, implemented and maintained...." The licensees actions in this case appear to be a violation of this technical specification in that the operator did not properly implement procedure A-15. (80-06-02)

For the ten day period involved the licensee apparently operated outside an NRC License Condition imposed by the NRC in its Exemption dated December 15, 1978 which states, in part, in paragraph IV.(1), "Until implementation of the modifications....the facility shall be operated in accordance with the procedures for operator action described in licensee's letter of April 14, 1978, as supplemented by letters dated April 21, and July 7, 1978, except that the maximum time for completion of operator action shall be ten minutes after initiation of the event...."

The referenced letters refer to a condition in that if a small break is postulated to occur in the RCS piping between the RCS pump discharge and the reactor vessel, part of the high pressure injection flow injected into this line would flow out the break. Therefore, for the worst combination of break location and single failure, only a portion of the flow from a single high pressure ECCS pump would contribute to maintaining the coolant inventory in the reactor vessel. This situation had not been previously analyzed and B&W had indicated that the limits specified in 10 CFR 50.46 may be exceeded.

Consequently, to facilitate an action to preclude the above stated problem, and to justify a return to full power operation, the licensee committed to maintain one of the series connected, manually operated Cross-Tie valves normally open. However, for a period of ten days the licensee apparently operated in violation of the referenced Exemption License Condition by operating with both Cross-Tie valves closed. (80-06-01)

Two apparent items of noncompliance at the level of violation were identified.

f. LER 79-24

As noted in the discussion of LER 79-23, the "B" HPI pump was taken out of service at 8:45 PM on December 17, 1979 to facilitate fire protection modifications. When the fire protection changes were reviewed to determine their effect on plant operations, changes where made to the Locked Valve List (SP 214.03) and the Nuclear Service Raw Water procedure A.25. Procedure A.15 was not changed and it was apparently this procedure that was used to remove HPI Pump P-238B from SFAS standby service for the fire protection modifications.

Consequently, the Nuclear Service Raw Water (NSRW) valves in the supply and discharge lines to the make-up pump coolers (oil and room) were not valved in to supply NSRW cooling water when the make-up pump was aligned to operate as the "B" HPI pump, even though an approved procedure was available for use and the operators were instructed by a senior licensed operator in how to line up the system.

The improper valve lineup was discovered by the licensee on January 9, 1980 at about 3:30 am. Upon discovery of the problem, the shift supervisor returned the "B" High Pressure Injection pump to service and placed it in SFAS standby.

Technical specification 6.8.1 states in part "Written procedures shall be established, implemented and maintained...." Procedure A.15 was apparently not maintained in accordance with either the Locked Valve List or the NSRW procedure. This is an apparent item of noncompliance at the level of an infraction. (80-06-04)

Technical Specification 3.3.1 specifies that the reactor shall not remain critical unless, among other things, (1) two out of three high pressure injection pumps shall be operable, and (2) the manual valves in the suction and discharge lines of all operable heat exchangers served by the Nuclear Service Raw Water System are locked in their throttled or open position.

Contrary to this requirement the reactor remained critical and was operated at greater than 10% power for approximately 22 days, except for a 16 hour shutdown period on January 5, 1980, in noncompliance with this limiting condition for operation. This is an apparent item of noncompliance at the level of an infraction (80-06-03).

Two apparent items of noncompliance, both at level of an infraction, were identified.

g. LER 80-03

As stated in the discussion on LER 79-24, the licensee returned the "B" HPI pump to service at 3:30 AM on January 9, 1980. When this occurred the operator apparently did not follow procedure A.15, Revision 8, with respect to the electrical realignment of SFV-23508 (the make-up tank isolation valve).

This problem was not discovered until 9:20 PM on January 10, 1980 when a control room operator noted that an indicator light on his safety features panel for the valve in question, was not lit.

Procedure A.15, Revision 8, step 7.15.12.1 states: "Change make-up tank isolation valve SFV-23508, power supply from 52-2B166 to 52-2A172 and place A key interlock to CLOSE at H8DSFV-23508 in makeup pump room." Technical specification 6.8.1 states in part, "Written procedures shall be established, implemented, and maintained. . ."

Contrary to these requirements, it appears the operator did not properly implement procedure A-15.

As a result of the electrical alignment problem discussed above and the valve alignment problems discussed under LER's 79-23 and 79-24, the licensee has instructed all operations personnel to perform independent (two man) dual verification of ECCS valve and breaker lineups.

This LER will remain open pending inspector verification of the implementation and documentation of the corrective actions listed in the LER.

One apparent item of noncompliance was identified at the level of an infraction. (80-06-05)

h. LER 80-06

The reported stud failure in an Anchor Valve Company supplied valve was examined by the following methods:

1) A documentation review which included a review of the minutes of plant review committee meetings, a failure analysis report of the fractured studs by Anamet Laboratories dated January 11, 1980, and inplace stud hardness test results performed by Mobile Inspection Services;

2) A visual inspection of the fractured stud and several other studs in the borated water system, including the studs in the valve with the failed stud; and,

3) Discussions with SMUD engineers and test personnel.

The Calibration of hardness testing equipment was also observed.

After experiencing failure of a stud from a stainless steel valve, manufactured by Anc or Valve Company, the licensee sent the failed stud to Anamet L boratories, Inc., for failure analysis. In addition, the material, designated as ASTM A193 B6 (416) on the manufacturer's drawings, was tested for compliance with the ASTM A193 specifications.

The licensee indicated that the stud failure was due to intergranular fracture. Intergranular fracture is a failure mode usually due to improper heat treatment and/or an environmentally-induced phenomenon such as stress corrosion cracking.

The tests for compliance with ASTM A193 revealed that the stud material was not in compliance with the current specifications in two areas: percentage of sulfur in the material, and the hardness of the material using the Rockwell C scale.

The higher-than-specified sulfur content is attributed to the material's being Type 416 stainless steel. Type 416 is a resulfurized 12% chromium steel. This material was within ASTM A193 specifications at the time of purchase. However, the current specification ASTM A193 does not include a resulfurized (Type 416) steel.

The average of the hardness tests run on a cross section of the stud resulted in a reading of 37 HRC. This hardness value can be obtained in Type 416 by tempering in the 900-1050$^{\circ}$F range. ASTM A194, which lists Type 416, specifies a minimum tempering temperature of 1100$^{\circ}$F and limits the hardness to 28 HRC maximum.

The licensee concluded that the combination of Type 416 stainless steel and the high hardness values (probably caused by low temperatures during the tempering process) makes the material quite susceptible to the type of failure experienced.

Further analysis of a sampling of studs is underway by Anamet Laboratories to supplement the previous failure analysis report of January 11, 1980. The status, of this program was discussed with the licensee. The results of all the testing programs will be used to determine the need for replacement of susceptable studs. The tests are designed to correlate mechanical properties such as yield strength with hardness values.

By the conclusion of this inspection on February 21, 1980 the results from some of Mobile Inspection Services inplace hardness testing was available. A summary of those results follows:

A 193B-6 (410)

| | | |
|---|---|---|
| 4 valves (28 studs) | - | 22-26 $HR_c$ |
| 1 valve (16 studs) | - | 28-30 $HR_c$ |

A 193B-6 (Unknown)

| | | |
|---|---|---|
| 6 valves (52 studs) | - | 20-26 $HR_c$ |
| 1 valve (16 studs) | - | 26-28 $HR_c$ |

A 193 B-6 (416)

| | | |
|---|---|---|
| 4 valves (24 studs) | - | 20-26 $HR_c$ |
| 2 valves (22 studs) | - | 28-32 $HR_c$ |

Valve BWS-018 (Valve with failed stud)

| | | |
|---|---|---|
| 4 studs of B-7 | - | 22 $HR_c$ |
| 1 stud of B-6 (416) | - | 24 $HR_c$ |

The inspectors examined the Anchor Company supplied valves located in the borated water system in the tank farm area. The valve with the failed stud (BWS-018) was inspected along with similar valves located around the borated water storage tank. Visual examination of valve BWS-018 failed to reveal any possible failure cause due to corrosion. However, it was noted that the studs were identified as B6 or CB7 with B8 nuts. B6 is martensitic steel, B8 is austenitic steel and CB7 is ferritic steel.

A previous fracture of a Type 416 Anchor Company supplied valve occurred approximately one year ago. However, this was not reported because it was considered as an isolated failure in a non-safety related system. Examination by the inspectors of the past maintenance history on the valve with the failed stud (BWS-018) disclosed that two valve bonnet studs had broken in 1974 "under slight torque" while attempting to tighten the bonnet. This would explain the reason for the different types of studs used for this valve.

The licensee has ordered enough replacement studs for all Anchor valves. The studs are stainless steel type 17-4 PH. The intent is to replace all "416" studs in safety-related Anchor valves inside containment prior to startup. By April 2, 1980 a replacement program for "416" studs in Anchor Valves outside containment will be available.

This LER is open pending further review of the stud replacement program.

5. Facility Modifications

The inspector reviewed the electrical modifications made in response to NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations. The installations were compared to descriptions provided by the licensee in letters to the Division of Operating Reactors, U.S. Nuclear Regulatory Commission, on October 18, 1979, November 26, 1979, and January 7, 1980. The installations were inspected for conformance to Reg. Guide 1.75 and IEEE 384. Specific documentation associated with installation and qualification certification of equipment was not reviewed, although the licensee's plans in this area were discussed.

The specific systems reviewed were:

a. Emergency Power for Pressurizer Heaters.

Two class 1E breakers, previously used as BUS-TIE breakers between redundant class 1E switchgear, have been utilized as the safety grade power supply for two 126 KW sets of pressurizer heaters (non-class 1E equipment). These breakers have, as part of their original installation, both indication and control in the control room, and a trip derived from the ESF actuaction signal. These breakers therefore serve as isolation devices per Regulatory Guide 1.75 and IEE 384 and provide the necessary separation of non-class 1E and class 1E circuits.

b. Emergency Power to PORV and PROV Block Valves.

The non-class 1E PORV block valve has been powered from a class 1E motor control center (2SB1). New class 1E cables, meeting the separation requirements of IEEE 384 have been installed from 2SB1 to the valve actuator.

The valve actuator, which was installed as a non-class 1E item is of a type generally qualified for safety related use. Although certificates of conformance from the vendor are not available, the licensee stated he would review the maintenance history of the device to insure that changes have not been made to alter this device from qualified actuators and will confirm qualification to the extent practical.

The PORV is powered from a non-class 1E battery. The D.C. power supply to this battery will receive power from a class 1E motor control center (2SA1). The licensee stated that the battery charger is of the same type as the class 1E chargers but certificates of conformance are not available. The licensee stated that the maintenance history will be reviewed to insure that changes have not been made which alter this device from qualified battery chargers. The licensee stated by letter on November 26, 1979 that "upgrading the power source does not seem likely to improve reliability of the system."

c. Subcooling Meter.

The licensee stated by letter to the Division of Operating Reactors on January 7, 1980, that a non-safety grade system would be installed this outage. The inspector questioned the methods used to provide isolation of this non-safety system from safety grade signal inputs such as pressure (H4SAA & H4SAB). The licensee stated that the isolation was provided by isolation amplifiers within the specific safety grade cabinets.

No items of noncompliance or deviations were identified.

6. Independent Inspection Effort

Discussions were held with operations, security and maintenance personnel in an attempt to better understand problems they may have which are related to nuclear safety. These discussions will continue as a standard practice.

On numerous occasions, during the month of February, the Resident Inspector attended outage status meetings. These meetings are held by the Outage Coordinator to provide all disciplines onsite with a shift by shift update on the plant status and ongoing maintenance work.

In addition to the above, independent inspection effort was performed on the following items:

a. Pressurizer Code Safety Valves

A number of concerns were identified in IE Report 50-312/80-03 on the work performed on the pressurizer code safety valves BRO 9499, BM 9684, and BMO 9685. All items identified as Unresolved Item 80-03-02 in the referenced IE report have been resolved. Resolution was obtained as a result of the inspectors's witnessing of rework on valve BRO 9499, discussions with the Dresser Corporation representative onsite on February 8, 1980, and by reviewing documentation between the licensee and the vendors involved in safety valve maintenance, design and repair.

The items of concern as identified in IE Report 50-312/80-03 and the answers to those items are as follows:

1) Item of Concern: Action was requested to resolve whether a plugged bonnet adversely affects the intended safety function of the pressurizer safety valves.

   Answer: If the bellows were to leak severely, then subsequent back pressure build up may reduce valve lift or cause chatter. The reduced valve lift, however, would not reduce the valve capacity significantly. Moreover, the initial opening set point would not be affected, therefore the valve would perform its intended function.

2) Item of Concern: Action was requested to determine whether the bellows on valve BRO 9499 was sound.

   Answer: A bellows leak test was performed. There was no indication of leakage. The valve was disassembled and the bellows was inspected. The bellows was sound.

3) Item of Concern: Technical justification was requested for not performing the bellows leak test recommended by the Dresser Industries Manual on BRO 9499.

   Answer: The test was run as described in 2) above. The valve manufacturer recommends running the bellows leak test only when a leak is suspected. No leak has been suspected on BRO 9449.

4) Item of Concern: It was requested that the need for installing or not installing manual actuation lift handles be technically resolved and the resultant conclusion implemented consistently.

   Answer: There is no need to install the manual lift handles because the handles will not be used during plant operation. The handles are supplied as a customer convenience. The licensee intends to remove the manual handles. The removed handles will not affect the seismic response of the valves according to the architect-engineer.

5) Item of Concern: The valve repair procedures appeared to require revision to include quality control verification of critical work operations.

Answer: A licensee representative stated that a review of repair and installation procedures would be made with the intent of incorporating information from the October 1978 Dresser Industries Manual for the 31700 Closed Bonnet Maxi-flow Safety Valves.

The procedures will be revised prior to the time the current spare safety valve is rebuilt.

6) Item of Concern: Technical justification was requested for performing one satisfactory lift pressure test when the Dresser Industries Manual recommends three consecutive successful lifts for assurance of accurate lifts.

Answer: It is the owner's responsibility to prepare the detailed test procedure. The content of the procedure is usually based on the valve manufacturer's recommendations. However, in this particular case the licensee decided, based on experience in this area at Rancho Seco, to lift the valve at set pressure only once unless conditions warrant further lifts. This procedure is consistant with ASME code requirements. The licensee has stated they will revise their procedure to require at least three lifts at plateaus on up to the set pressure for the purpose of looking for erratic behavior. Abnormalities will be pursued on a case by case basis.

7) Item of Concern: Technical justification was requested for not having valve body drains piped to a safe area, as recommended by the Dresser Industries Manual, to prevent corrosion.

Answer: The requirement to pipe the body drains to a safe area was directed toward valves having a vertical riser at the discharge of the valve. At Rancho Seco, the valve relieves downward to the PRT so that it is unlikely that fluid will collect in the body.

8) Item of Concern: Verification was requested that the Dresser Industries Manual of October 1978 applied to valve BRO-9499, since BRO-9499 is a Model 31759 valve and the Dresser Industries Manual lists Model 3-31759A-1 and 6-31759A-2 as applicable model numbers.

   Answer: The valve is a 3-31759A-1 valve and the October 1978 manual applies. The manual was entered into the document control system about April 4, 1979. The old manual was in effect for all work performed on the valve prior to this date.

9) Item of Concern: It appeared necessary to provide verification that blowdown adjustments are currently proper for the valves to be re-installed for operation.

   Answer: Discussions with the vendor representative onsite for the rework on BRO 9499 indicated that care was taken to return the blowdown rings to their proper location. Factory tolerances were met, so there is reasonable assurance that correct blowdown will be obtained.

10) Item of Concern: It appeared necessary to provide verification that critical internal dimensions, and clearances were properly established on valves to be re-installed for operation.

   Answer: Vendor communications with the licensee verified dimensions were proper.

11) Item of Concern: The inspector had a question regarding the absence of lead seals on the lockwire attached to adjusting ring locknuts.

   Answer: The licensee will install seals on the lockwire attached to adjusting ring locknuts.

The answers given above resolve item 80-03-02 in paragraph 8a and 10 of the referenced IE inspection report. The procedure revisions committed to by the licensee in items 5) and 6) above will be reviewed by the inspector during a subsequent inspection (80-06-08).

b. Surveillance Testing During Refueling

A licensee representative stated that a technical specification clarification of technical specification requirements for surveillance testing during refueling had been drafted. The position stated in Unresolved Item 80-03-03 is appropriate, as long as no technical specification requirement is violated.

This item is resolved and is considered acceptable. 80-03-03 is closed.

c. Seismic Monitoring Systems

Further investigation into the Rancho Seco seismic monitoring system occurred during the month of February.

Item 80-03-04 remains unresolved with the following additional information provided as an update. On February 15, 1980, a conference call between Region V NRC personnel and SMUD personnel took place. During the conference call a licensee representative stated:

1) SMUD will produce updated maintenance procedures for all three subsystems.

2) SMUD will replace the electrical cable from the triggers to the control room recorder and battery. This should correct the current problem which causes the power supply (battery) to discharge.

3) SMUD will investigate the feasibility of replacing currently installed accelerometers with compatible new ones to increase system reliability. If determined to be feasable this will be done.

4) SMUD will immediately initiate a program to provide week day surveillance of the battery to assure that the battery is maintained with a full charge. This daily surveillance will continue until the new electrical cable described in item 1 above is installed.

5) SMUD will seal the instrument pit from moisture and dirt so that the seismic instruments will be located in a more suitable environment for reliable operation.

6) SMUD will expedite calibration of the present system. The seismic event triggers (4) will be repaired and calibrated.

7) SMUD will upgrade their earthquake procedures to include actions required by operators to confirm spurious trips.

8) SMUD will develop procedures to specify the proper method for processing and evaluating data. Also they will take whatever actions necessary to assure that site personnel are aware of the importance of maintaining the seismic instrumentation and following the associated procedures.

9) SMUD will actively investigate and pursue a program for long term upgrading of the current system or installation of a new system. The requirements contained in 10 CFR 10C, Appendix A and Regulatory Guide 1.12 will be considered in the evaluation of a new system.

This item will remain unresolved pending followup on the above items and verification that FSAR committments regarding the seismic system have been met. (80-03-04)

d. Snubbers

The inspector noted that four of the seven snubbers that failed operational testing during this refueling outage also failed operational testing during the 1978 refueling outage. All four were Bergen-Paterson snubbers. Due to the number of failures found on the Bergen-Patersen units, all 53 units had to be tested this outage.

The following is a summary of failures that have been discovered during this outage:

| Failure Mode | Footnote | ID | Model Number |
|---|---|---|---|
| No lockup either direction | 1,2 | 5SW-20530-7A | HSSA-10-6 |
| No lockup either direction | 3,4 | 1SW-206021-11A | HSSA-3-6 |
| No lockup in retract | 1,2,4 | 4SW-26101-2 | HSSA-10-6 |
| No lockup in retract | 1,2,4 | 1SW-26021-7A | HSSA-3-6 |
| No lockup in retract | 3,4 | 5SW-20529-3A | HSSA-3-6 |
| No lockup in extension | 2 | 4SW-53520-3 | HSSA-3-6 |
| Unacceptable bleedoff rate | 2 | 5SW-32141-1A | HSSA-10-6 |

1 Found to have broken poppet spring(s) upon disassembly.

2 Found to have worn poppets and seats upon disassembly.

3 Found to have worn "O" ring upon disassembly.

4 Failed during 1978 refueling outage.

This item will be followed up as part of the inspector's review of LER 80-7. (80-06-06)

e. Document Control

Vendor supplied maintenance and technical manuals have been noticed by the Senior Resident Inspector to exist in various places throughout the plant in various stages of control. It is not clear to the inspector when various manuals have arrived on site for use, how many copies of each manual exist, their locations, or whether or not the manual being used is the latest and/or appropriate publication.

Various manuals were examined in reviewing the above control, including the CDR manual, the pressurizer code safety valve manual, and five diesel generator manuals.

This item is unresolved pending an examination of licensee compliance with Criterion VI to 10 CFR 50, Appendix B on Document Control. (80-06-07)

No items of noncompliance or deviations were identified.

7. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance or deviations. One new unresolved item was disclosed during this inspection and is discussed in paragraph 6e. (80-06-07)

One additional unresolved item was updated but remains unresolved and is discussed in paragraph 6c. (80-03-04)

Two previously unresolved items were resolved in this report and are discussed in paragraph 6a (80-03-02) and paragraph 6b (80-03-03).

8. Exit Interview

The Director of the Region V Office of Inspection and Enforcement was joined by other members of the Region V staff and the Senior Resident Inspector for an exit interview with the Assistant General Manager and Chief Engineer, members of his staff and the plant superintendent on March 5, 1980. (See paragraph 1)

During this meeting, the Senior Resident Inspector summarized the scope and findings of his February 1980 inspection effort. The Region V Director and his staff members discussed the apparent items of noncompliance identified in paragraphs 4.e., 4.f., and 4.g. of this report. Region V management stated that based upon the number and the type of items of noncompliance identified in this IE inspection report, the Region V staff will recommend escalated enforcement action to NRC Headquarters. The licensee asked for a clarification on the term "escalated" enforcement action. The response was that Region V would probably recommend enforcement action at the level of civil penalty. The licensee acknowledged this information.

Prior to the exit interview discussed above, the Senior Resident Inspector reviewed his monthly inspection activities with plant staff members.