U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

NRC Inspection Report: 50-445/89-89
50-446/89-89

Permits: CPPR-126
CPPR-127

Dockets: 50-445
50-446

Construction Permit
Expiration Dates:
Unit 1: August 1, 1991
Unit 2: August 1, 1992

Applicant: TU Electric
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Facility Name: Comanche Peak Steam Electric Station (CPSES), Units 1 & 2

Inspection At: Comanche Peak Site, Glen Rose, Texas

Inspection Conducted: December 6, 1989, through January 2, 1990

Inspector: M F Runyan 1/8/90
M. F. Runyan, Resident Inspector, Date
Civil Structural (paragraphs 2, 3, 4, 5, and 6)

Consultants: W. Richins, Parameter (paragraphs 4 and 6)
P. Stanish, Parameter (paragraphs 6 and 7)

Reviewed by: H H Livermore 1-8-90
H. H. Livermore, Lead Senior Inspector Date

Inspection Summary:

Inspection Conducted: December 6, 1989, through January 2, 1990 (Report 50-445/89-89; 50-446/89-89)

Areas Inspected: Unannounced, resident safety inspection of applicant's actions on previous inspection findings; follow-up on violations/deviations; action on 10 CFR Part 50.55(e) deficiencies identified by the applicant; assessment of allegations; evaluation of room/area and system completion walkdowns; Nuclear Steam Supply System (NSSS) interface; and general plant areas (tours).

Results: No strengths, weaknesses, violations, or deviations were identified.

Within this report period, the NRC staff completed the assessment of the applicant's room, area, and system turnover programs. These programs resulted in the location and correction of most of the safety-related discrepancies which remained after the completion of construction. Though many less-significant problems were not identified, the number of these minor discrepancies was not large enough to merit a reinspection effort. The NRC inspector determined that the applicant's room/area/system turnover process was adequate (paragraph 6).

DETAILS

1. Persons Contacted

*J. W. Beck, Vice President, Nuclear Engineering, TU Electric
*O. Bhatty, Issue Interface Coordinator, TU Electric
*M. R. Blevins, Manager of Nuclear Operations Support, TU Electric
*H. D. Bruner, Senior Vice President, TU Electric
*A. R. Buhl, IAG
*W. J. Cahill, Executive Vice President, Nuclear, TU Electric
*H. M. Carmichael, Senior Quality Assurance (QA) Program Manager, CECO
*J. T. Conly, APE-Licensing, Stone and Webster Engineering Corporation (SWEC)
*D. E. Deviney, Deputy Director, QA, TU Electric
*F. Dunham, QA Issue Interface, TU Electric
*J. C. Finneran, Jr., Manager, Civil Engineering, TU Electric
*J. L. French, Independent Advisory Group
*B. P. Garde, Attorney, CASE
*W. G. Guldemond, Manager of Site Licensing, TU Electric
*T. L. Heatherly, Licensing Compliance Engineer, TU Electric
*J. C. Hicks, Licensing Compliance Manager, TU Electric
*C. B. Hogg, Chief Engineer, TU Electric
*J. L. LaMarca, Manager of Electrical and I&C Engineering, TU Electric
*F. W. Madden, Mechanical Engineering Manager, TU Electric
*J. W. Muffett, Manager of Project Engineering, TU Electric
*S. S. Palmer, Project Manager, TU Electric
*P. Raysircar, Deputy Director/Senior Engineer Manager, CECO
*J. D. Redding, Executive Assistant, TU Electric
*M. J. Riggs, Plant Evaluation Manager, Operations, TU Electric
*A. B. Scott, Vice President, Nuclear Operations, TU Electric
*J. C. Smith, Plant Operations Staff, TU Electric
*R. L. Spence, TU/QA Senior Advisor, TU Electric
*P. B. Stevens, Manager of Operations Support Engineering, TU Electric
*C. L. Terry, Manager of Projects, TU Electric
*R. D. Walker, Manager of Nuclear Licensing, TU Electric
*R. G. Withrow, EA Manager, TU Electric
*D. R. Woodlan, Docket Licensing Manager, TU Electric

The NRC inspectors also interviewed other applicant employees during this inspection period.

*Denotes personnel present at the January 2, 1990, exit meeting.

2. Applicant's Action on Previous Inspection Findings (92701)

(Closed) Open Item (445/8948-O-01; 446/8948-O-01): This item identified the need for the applicant to correct the design validation calculation for the service water intake structure. This calculation had assumed a nonconservative groundwater level (780 feet) as compared to nearby piezometer readings (783 feet). The calculation 16345/6-CS(B)-058, "Service Water Intake Structure-Exterior Wall Design," was revised by the issuance of Calculation Change Notice No. 2 to reflect the structural impact of the higher groundwater level. New calculations provided in the calculation change notice revealed that the service water intake structure was still adequate as designed and constructed. In addition, Design Basis Document (DBD)-CS-084, "Other Seismic Category I Concrete Structures," was revised to show the higher assumed groundwater level for the service water intake structure.

The NRC inspector reviewed Calculation Change Notice No. 2 to Calculation 16345/6-CS(B)-058 and the revisions to DBD-CS-084. Based on this documentation, the inspector concluded that the applicant had sufficiently addressed this concern. This open item is closed.

3. Follow-up on Violations/Deviations (92702)

(Closed) Violation (445/8938-V-05): This violation involved the applicant's method of accepting nonconforming welds. Welds 1A, 1B, 2A, and 2B located at the joint between the fuel transfer tube expansion bellows and the penetration sleeve in Unit 1 were rejected based on adverse indications revealed by radiography. However, civil/structural engineering contractor personnel dispositioned the welds (per Nonconformance Report [NCR] 89-04023) "use-as-is" based on an analysis of the weld defects, the containment structural integrity test, and a stress analysis which neglected the portions of the welds containing defects. The applicant initiated a proposed change to the FSAR to allow the use of partial penetration welds in this application. At the time the violation was cited, the proposed FSAR change was undergoing internal review by the applicant prior to submittal to the NRC. The applicant recognized that the FSAR change request was being made on an "at risk" basis and that the welds would have to be repaired or reworked if the change was rejected.

After receiving the violation, the applicant withdrew the FSAR change request and revised NCR 89-04023 (to Revision 5) to require rewelding and radiography of the subject welds. The joints were rewelded and accepted by radiographic examination.

The applicant reviewed other proposed FSAR changes to determine whether there were additional cases in which an engineering analysis was used as an alternative to code requirements. Two such cases were identified: (1) Licensing Document Change Request F3.8-57/89-587 regarding examination of full penetration attachment welds to the containment liner and (2) Licensing Document Change Request F3.8-60/89-764 regarding acceptance criteria for nonpressure parts or pads attached by welding to liners. The applicant stated that the NRC had previously been made aware of these issues and that both documents will be submitted to the NRC prior to Unit 1 fuel load.

The applicant considered its current program for design control to be adequate in that NRC approval is required for any nonconservative changes implemented on an "at-risk" basis. Individuals who worked on the fuel transfer tube weld issue were made aware of the violation.

The inspector reviewed NCR 89-04023, Revision 5, and the radiographic examination records for the replacement welds and determined that the applicant had adequately addressed this issue. This violation is closed.

4. <u>Action on 10 CFR Part 50.55(e) Deficiencies Identified by the Applicant (92700)</u>

   a. (Closed) Construction Deficiency (SDAR CP-85-53): "Seismic Category I Platforms and Structures." By letter TXX-4669 dated January 20, 1986, the applicant informed the NRC that a deficiency regarding the fabrication of approximately thirty structural steel platforms/structures in Units 1 and 2 was a potentially reportable item. This deficiency was initially identified during review of Corrective Action Request (CAR)-052 for a platform in the Unit 1 reactor building (beneath the reactor vessel) which was not fabricated in accordance with design documents. The platform had previously been accepted by Quality Control (QC). The applicant performed a document search which indicated that similar conditions did not exist for Unit 2. Note that this issue was initially titled "Seismic Category II Platforms" but was changed to the current title after the applicant determined that the initial concern addressed a Category I platform. The applicant stated in letter TXX-88125 dated February 4, 1988, that this construction deficiency was reportable.

   The deficiencies regarding both Category I and II structural steel platforms have been addressed by a reinspection program under Field Verification Method (FVM)-090. All platforms were reinspected and identified discrepancies were addressed on NCRs. In addition, construction and inspection procedures for structural steel platforms have been substantially improved.

The inspectors have reviewed the FVM-090 program and inspected five structural steel platforms as documented in NRC Inspection Reports 50-445/89-05, 50-446/89-05; 50-445/89-11, 50-446/89-11; 50-445/89-19, 50-446/89-19; and 50-445/89-26, 50-446/89-26.

Based on the previous inspection and review of FVM-090, review of modified construction and inspection procedures (for example, Specification 2323-SS-16B, "Structural Steel/Misc. Steel"), and a review of the applicant's document package for SDAR CP-85-53, the inspector concluded that this issue has been adequately addressed. This construction deficiency is closed.

b. (Closed) Construction Deficiency (SDAR CP-87-64): "Design Basis Tornado [DBT] Analysis for Safety Related Equipment." By letter TXX-6732 dated September 22, 1987, the applicant informed the NRC that a deficiency regarding the pressure relieving capacity of the tornado venting devices in Units 1 and 2 was a potentially reportable item. Specifically, the existing DBT analysis did not include sufficient documentation to show conclusively that safety-related equipment could withstand the differential pressure transients expected during the DBT. Subsequently, by letter TXX-88086 dated January 20, 1988, the applicant determined the issue to be reportable.

The applicant performed evaluations of the effects of tornado venting depressurization for all safety-related systems and those nonsafety systems necessary to meet the requirements of NRC Regulatory Guide 1.117, "Tornado Design Classification." The potential failure of any component which could be affected by a DBT was examined to determine the effect on the ability of the system to perform its safety function. These evaluations are documented in Stone and Webster Report 17530-1805103-B4, "Tornado Venting Depressurization Effects" dated October 31, 1988.

The applicant committed in TXX-88086 to the procurement of safety-related components in accordance with DBD-ME-009, "Design Basis Tornado Analysis." The applicant subsequently issued CAR 89-013, "Design Basis Tornado Analysis," to address deficiencies in the procurement process where the requirements of DBD-ME-009 were not being effectively implemented. The corrective action specified by CAR 89-013 directed engineering to perform a DBT effects evaluation for component material substitutions and to verify that no components were part of the commercial grade critical characteristics evaluation (not directly related to this SDAR). The DBT effects review resulted in no component design changes.

The inspector reviewed the SDAR CP-87-64 package, the Stone and Webster report on DBT effects, DBD-ME-009, and CAR 89-013. The inspector concluded that this issue has been adequately addressed by the applicant. This construction deficiency is closed.

c. (Closed) Construction Deficiency (SDAR CP-87-66): "Structural Bolting in Tension." By letter TXX-88173 dated February 4, 1988, the applicant informed the NRC that a deficiency involving structural bolting in tension was a reportable item. This issue was previously reviewed by the NRC (NRC Inspection Report 50-445/89-65; 50-446/89-65, paragraph 4.d) and left open at that time pending completion of field repairs. During this inspection period, the NRC inspector reviewed (as a sample) the construction documents for five design change authorizations (DCAs) and determined that, based on the applicant's tracking system, all related DCAs are now complete. This construction deficiency is closed for Unit 1.

d. (Closed) Construction Deficiency (SDAR CP-87-115): "Seismic Analysis of Service Water Intake Structure." By letter TXX-6984 dated November 18, 1987, the applicant informed the NRC of a deficiency regarding the seismic analysis of the service water intake structure. This issue was previously reviewed by the NRC inspector (NRC Inspection Report 50-445/89-65; 50-446/89-65) and left open at that time pending completion of the applicant's activities associated with DCA 71229, SDAR CP-86-36, and SDAR CP-86-72. The inspector reviewed the work orders associated with DCA 71229 during this inspection period and verified that the work is now complete. SDARs CP-86-36 and CP-86-72 were closed in NRC Inspection Report 50-445/89-12; 50-446/89-12. This construction deficiency is closed.

5. <u>Assessment of Allegations (99014)</u>

a. (Closed) Allegation (OSP-89-A-0080): An allegation was received by the NRC from an individual who had worked at Comanche Peak in 1978 and 1979. The alleger's concerns included assertions that (a) valves had been installed in incorrect positions in areas around containment, and that the valves would fit both ways but would not seal properly; (b) a concrete slab in Unit 1 was not thick enough as evidenced by a bolt protruding through the other side; and (c) some records had been destroyed by a disgruntled employee and that 90 percent of the records still on hand (in 1978-1979) were forgeries.

The NRC referred this allegation to the applicant for review. The review was performed by the applicant's SAFETEAM organization as Concern 12661. In letter TXX-89783 dated November 10, 1989, the applicant stated that the investigation was complete and that no information surfaced which would support the allegation.

The NRC inspector reviewed documentation of the investigation in the SAFETEAM files for which a brief summary follows. SAFETEAM discovered that very few valves were actually fitup and welded during 1978-79 and that valve issuance from the warehouse was tightly controlled. The applicant's Corrective Action Program (CAP) and Post-Construction Hardware Validation Program (PCHVP) fully checked each valve for compliance to design requirements. SAFETEAM determined that concrete slab thickness was reviewed within Issue-Specific Action Plan (ISAP) VII.c, "Concrete Placement," by CPRT and that no deviations were issued for concrete slab thickness. SAFETEAM discovered that considerable record changing occurred in 1978-79 including discontinuing and reclassifying many records. Some of this activity may have involved destroying old records which may have created a false perception for the alleger. Otherwise, the bulk of the documentation has been rereviewed and corrected as necessary within the CAP. The NRC inspector determined that SAFETEAM had adequately addressed all issues resulting from this allegation and had issued a justifiable conclusion. Consequently, this allegation is not substantiated and is closed.

b. (Closed) Allegation (OSP-89-A-0068): An allegation was received by the NRC staff that delineated 14 concerns. Each concern was numbered sequentially in the order it was presented. Eight of these concerns are addressed in NRC Inspection Reports 50-445/89-76; 50-446/89-76 and 50-445/89-85; 50-446/89-85. The remaining six concerns are addressed below:

Concern No. 1

The alleger stated that the word "witness" means observing work in progress whereas the licensee provided an explanation that this word actually means "verified," which implies the item can be inspected after the work is completed.

Review

The definitions for the words "witness" and "verify" are provided in Procedure AQP-11.1, "General Fabrication and Installation Inspection." This procedure clearly states that witnessing involves physically observing the entire operation being performed whereas verifying requires an examination of the hardware after the operation is complete. The NRC inspector discussed this issue with the QC inspection supervisor, who stated that, to his knowledge, no questions had ever arisen concerning the interpretation of these terms. He also stated that TU Electric had not provided an explanation that "witness" and "verify" are interchangeable terms. The NRC inspector also talked to a QC inspector with 14 years experience on site. The QC inspector provided the correct definition of these terms and stated that he had not heard of any position being promulgated to the contrary.

Conclusion

The inspector could find no evidence to support the alleger's concern. Consequently, the alleger's concern is not substantiated.

Concern No. 2

The alleger stated that it was a widespread practice to simply replace any bolts that were broken during torquing instead of documenting the problem on an NCR. In 1987, the alleger was told to replace two bolts which had broken in Unit 1 Safeguard building, Elevation 778, instead of documenting the problem on an NCR.

Review

In an interview with the NRC inspector, the QC inspection supervisor stated that he was aware of at least one occasion where broken bolts were replaced without documentation on an NCR. According to the QC inspection supervisor, this practice was not widespread and that after the NRC presented a violation for this practice, no further problems have developed. The NRC inspector concluded that the item recalled by the supervisor was Violation 445/8718-V-08. This violation cited the fact that a high-strength capscrew had broken during installation before the design torque was achieved, and that this event was not documented on an NCR. Along with an NCR for the subject capscrew, the applicant issued Deficiency Report (DR) C-87-3885 which stated that the failure to document this particular problem on an NCR was the result of an inadequate definition of the items "flawed" and "failed" materials in plant procedures. The procedures were upgraded and training was conducted to the new procedures. This violation was closed in NRC Inspection Report 50-445/88-56; 50-446/88-52. Based on the NRC closure and the QC inspection supervisor's statement, it appears that the applicant has effectively corrected the problem of plant workers replacing broken bolts without documentation.

The applicant addressed the hardware issues stemming from the failure of a 1-inch Hilti bolt in SDAR CP-87-125. Corrective actions taken by the applicant included procedure changes to prevent torquing bolts to excessive levels. NRC review of SDAR CP-87-125 is documented in NRC Inspection Reports 50-445/88-82; 50-446/88-78 and 50-445/89-05; 50-446/89-05. In these reports, the NRC inspector concluded that defective bolts should be self-identifying in that any bolt which yields prior to reaching its design torque would be rejected either because it would break or because it would fail the QC inspection for applied torque. Since the torque to which the bolt is installed exceeds the torque expected during service, this self-identifying feature of defective bolts diminishes the chances for encountering bolt problems during plant operation. Nevertheless, the applicant's current policy of analyzing each bolt that fails is a prudent action.

Conclusion

The alleger's concern is probably accurate though somewhat exaggerated in scope. The applicant has taken satisfactory action to ensure that NCRs are written for defective bolts. The hardware issues related to any past failures to document defective bolts do not appear to warrant additional investigation.

Concern No. 3

In 1988, the alleger identified to a supervisor pipe supports which were anchored too close to unused Hilti bolt holes, the same problem that had been recently identified in an NCR. The alleger's termination of employment was potentially prompted by the identification of this issue.

Review

The alleger's technical concern was fully addressed by the applicant within the PCHVP as FVM CPE-SWEC-FVM-CS-075, "Concrete Attachments." The applicant generated detailed sketches of 50 concrete surfaces and analyzed the impact of any instance where the spacing between two anchors (including unpatched drill holes) violated current specifications. NRC inspection of this issue is documented in NRC Inspection Reports 50-445/88-76, 50-446/88-72; 50-445/88-82, 50-446/88-78; and 50-445/89-05, 50-446/89-05. The NRC concluded that the applicant had appropriately addressed the issue and that concrete attachments for Unit 1 were adequate. In addition, the applicant initiated a program to patch all unused Hilti bolt holes. The applicant concluded that a patched hole was as strong as the parent concrete.

Regarding the termination of employment, the alleger was advised to contact the U.S. Department of Labor. The alleger expressed an intention to follow this course of action.

Conclusion

The alleger's technical concern is not substantiated. The alleger indicated his labor concern would be referred to the Department of Labor.

Concern No. 4

The alleger's work as a QC inspector was potentially compromised as the result of excessive quotas imposed for pipe support inspections in November and December 1988.

Review

In a discussion with the NRC inspector, the Brown and Root (B&R) site QA manager stated that rough calculations were sometimes made and announced to QC inspectors regarding the number of daily inspections needed to meet a stated goal. These "target" numbers may have been interpreted as quotas by the alleger. However, the B&R site QA manager stated that no individual inspector was ever held accountable to any quantitative standard. Rather, announcements of daily inspection goals were made to evoke a team response.

The inspector interviewed the QC inspection supervisor and two QC inspectors regarding this issue. Their responses corroborated the statements made by the B&R site QA manager. Apparently for them, although there was some knowledge of a management "push" to complete the inspections, time pressure never compromised their inspection efforts.

Conclusion

The inspector did not find any evidence to support the alleger's concern. Additionally, the alleger stated that the quality of this work was only potentially affected. Based on these facts, the alleger's concern is not substantiated.

Concern No. 6

The alleger did not have confidence in the CPSES SAFETEAM's ability to evaluate and resolve safety issues when identified.

Review

The NRC conducted a multi-faceted review of the SAFETEAM organization in February and April 1988 (NRC Inspection Report 50-445/88-23; 50-446/88-20). This inspection encompassed the adequacy and implementation of SAFETEAM's program to orient new employees, conduct interviews, protect confidentiality, classify and trend concerns, investigate concerns, provide feedback, and verify corrective action. The NRC inspection team concluded that SAFETEAM provided an effective means for site personnel to express concerns. No findings were identified supporting the alleger's claim that SAFETEAM lacked the ability to evaluate and resolve safety issues. Additionally, as a point of clarification, SAFETEAM is not chartered to evaluate and resolve technical matters involving reactor safety. SAFETEAM refers these issues to the appropriate engineering organization for review.

Conclusion

The NRC inspection team conducted its review near the end of the alleger's tenure at CPSES and involved an extensive effort. The alleger's concern was presented without supporting details and appeared to be based on second-hand information. Accordingly, the alleger's concern is not substantiated.

Concern No. 9

The alleger stated that, during 1984 and 1985, conduit supports for lighting at Location 6A-CA, Elevation 849 feet, were not constructed according to requirements, but were accepted by QC.

Review

The location referenced by the alleger is 19 feet above the control room floor. In 1985, control room lighting was provided at this elevation as part of a suspended ceiling network. Later, this suspended ceiling was determined to be seismically inadequate and was removed. A new seismic ceiling was installed at an elevation of approximately 842 feet along with a new lighting system. Therefore, the lighting conduit supports referenced by the alleger have no remaining significance and no longer exist. Further, normal plant lighting has been reclassified as nonsafety related.

Conclusion

The alleger's concern is of no consequence to the present status of the plant.

6. Evaluation of Room/Area and System Completion Walkdowns (46053, 48053, 50100, 55100, 51053)

In NRC Inspection Report 50-445/89-76; 50-446/89-76, the NRC inspector expressed the opinion that based on independent NRC walkdowns, the number of nonconforming items missed by the applicant's room/area and system completion walkdown teams was high and that the applicant should perform a generic review of the effectiveness of these programs. During this inspection period, the applicant completed this recommended review and presented the results to the NRC.

The applicant performed an independent walkdown of Room 227 in the Unit 1 Auxiliary building. A total of 29 potentially nonconforming items were identified. Of these, five items were determined to be nonconforming items missed in the original walkdown or which resulted from construction activity following the walkdown. These five items were: (1) a loose hold down clamp for copper tubing, (2) loose shims and support for an instrument tube, (3) loose support clamp for a lighting conduit, (4) a broken sight glass, and (5) an abandoned angle iron. Every one of the nonconforming conditions were evaluated as nonsafety related and were dispositioned "rework." The applicant presented documentation showing that over 3000 attributes were evaluated in Room 227 and expressed the opinion that the number of discrepancies (5) did not represent an error rate which reflected negatively on the effectiveness of the program.

The applicant also evaluated project activities which assure the preservation of the condition of equipment following room/area turnover. These activities include access control, housekeeping walkdowns, and system walkdowns by system engineers. As a result of this evaluation, the applicant's QA department decided to commence enhanced overviews of ongoing work to assure that the condition of the plant is preserved. This QA overview focuses on specific locations where construction work activity has taken place to detect any damage in the vicinity of the completed work.

The applicant stated that the room/area walkdown program is complete and was effective. The inspector reviewed all documentation provided in support of the applicant's evaluation. In addition, four NRC inspectors performed a follow-up plant walkdown and identified the following potential nonconforming items:

a. In Room 51, a face plate was missing on gauge 1-P1-4805.

b. In Room 51, tubing on the north side of Pump CPI-CTAPCS-04 was not hooked up to two installed brackets.

c. In Room 51, a cover was missing on 1-TE-4814-4.

d. In Room 54, gauge for RHR Pump 1-01 Mini Flow 1-F1S-610 was reading high off scale.

e. In Room 54, a spring (H-CC-1-SB-046B-006-3) was bottomed out.

f. In Room 62, the snubber closest to the door, 2 feet off the floor, had an extension piece that did not appear to be straight.

g. In Room 62, rust or stain was found on a condulet (4 feet off floor) associated with Conduit C14K30353.

h. In Room 74, a channel for instrument tubing was bent and not properly attached to a platform on the west and north edge of the walkway.

i. In Room 74, a snubber (MS-1-SB-055-007-3) lock wire was broken.

j. In Room 77S, bent instrument tubing was found on the west wall at Elevation 820 feet.

k. In Room 77S, a sizable gouge in a concrete wall was found in the SE corner at Elevation 830 feet.

l. In Room 88, Valve 1-DD-392 had a valve stem leak.

m. In Room 88, a sizable gouge in a concrete wall was found in the NW corner at Elevation 851 feet.

n. In Steam Generator 1 compartment, the whole body of a snubber (FW-1-019-702-C42K) was found to rotate relative to the end connections.

The applicant provided information regarding each of the NRC field concerns as follows:

a. A work request was issued to replace the face plate.

b. A work order had already been issued on this item but the work is not complete.

c. The applicant did not provide information regarding this discrepancy within the time frame of this inspection, but this item is considered to have little significance.

d. The applicant did not provide information regarding this discrepancy within the time frame of this inspection, but this item is considered to have little significance.

e. The spring rod is at a 5° angle. This caused the load indicator to be canted and appear bottomed out. Under load, the load indicator will level out and the spring will operate as designed.

f. With the use of a survey, the applicant determined that the snubber was aligned properly.

g. This is galvanized piping and the stain is of no concern.

h. The applicant did not provide information regarding this discrepancy within the time frame of this inspection, but this item is considered to have little significance.

i. This lock wire had been damaged and repaired previously. Apparently, construction in the area caused the wire to break for a second time. A work request was issued to replace the lock wire and a platform is being built for the remaining construction.

j. The tube bend is by design as a "controlled bend."

k. A nonstructural (cosmetic) repair will be made to the concrete. Rebar was not exposed.

l. The valve leak had been identified previously (as evidenced by the presence of a bucket to collect the leakage) but a work order had not yet been issued until prompted by the NRC inspection.

m. A nonstructural (cosmetic) repair will be made to the concrete. Rebar was not exposed.

n. The specified torque of 150 foot-pounds on the adapter nut had been released apparently due to continuing manipulation of the snubber body. The snubber, however, was still operational and could only turn 1/4 turn in either direction.

The NRC inspector reviewed the applicant's explanation of the above concerns and concluded that, though some items were missed in the applicant's walkdown, none of them represented safety-significant deficiencies.

In summary, the previous NRC field walkdowns (documented in NRC Inspection Reports 50-445/89-65; 50-446/89-65 and 50-445/89-76; 50-446/89-76), the applicant's reinspection walkdown (documented above), and the most recent NRC walkdown (documented above) all support the same conclusion. The applicant's room/area and system turnover process appeared to be successful in detecting and correcting safety-significant discrepancies. On the other hand, many less-significant problems were not identified. The NRC staff has determined that the number of these minor discrepancies is not large enough to merit a reinspection effort and that the overall room/area/system turnover process as revised and executed was adequate.

7. <u>NSSS Interface (49065)</u>

In this inspection period, the NRC inspector reviewed the interface between Stone & Webster (SWEC) and Westinghouse Electric Corporation (WEC) for the piping analyzed as part of Stress Problem 1-025.

In December, 1988, SWEC letter SWW-0368 transmitted the As-Built Verification Package (ABVP) to WEC for their review and analysis of the Class 1 piping associated with problem 1-025. This package included all associated drawings, wall and floor sleeve penetration verifications, system configuration requirement form, and the exception list. The associated drawings transmitted were:

a. Isometric RC-1-RB-026, Revision CP-1.

b. Support drawings: RC-1-135-001-C41S, Revision 3
RC-1-135-004-C51K, Revision CP-1
RC-1-135-007-C41S, Revision 5
RC-1-135-008-C41K, Revision CP-1
RC-1-135-009-C51R, Revision 5
RC-1-135-010-C41K, Revision CP-2

c. Valve or equipment drawings: 1101J22, Revision 3
1100J48, Revision 8

Also transmitted to WEC by SWEC letter SWW-0396 were the stiffness values calculated for the supports and restraints installed on this Class 1 piping.

The NRC inspector reviewed the isometric referenced above and confirmed that the support drawings included all supports used on the subject piping. Additionally, the NRC inspector reviewed the equipment drawings which detail the pressurizer and its nozzle for the connection of the piping evaluated in this stress problem. The NRC inspector reviewed the support details, along with the Brown & Root hanger location (BRHL) drawing, to ensure that the isometric accurately depicted the type, function, and location of each support. The results of the review of the transmittal package, plus the information forwarded in SWW-0396, indicated that the necessary information had been forwarded to WEC to enable them to update the Class 1 analysis for this piping.

In response to the data, on January 20, 1989, WEC forwarded the final as-built analysis of stress problem 1-025 to the applicant providing the loads and displacement for the supports on this line. This data was subsequently revised and one of the supports was deleted, RC-1-135-004-C51K. This was done apparently because the load on the support in question, due to the original analysis, exceeded the capacity of the snubber that was already installed. When this analysis was rerun without the snubber in question, WEC was able to qualify this piping; therefore, it was deleted from the analysis. This support was removed from the system in accordance with DCA 29532, Revision 3.

The NRC inspector reviewed the final support sketches and confirmed that the final loads and movements predicted by the updated final as-built analysis performed by WEC had been incorporated as part of SWEC's final reconciliation for the supports on this line. Therefore, the NRC inspector concurs that the piping analysis performed by WEC and the analysis for the supports as performed by SWEC reflect the actual as-installed condition of these components. The NRC inspector also reviewed additional inspection data concerning sleeve clearances that SWEC reviewed to ensure that no physical interferences would occur due to anticipated movements during postulated events. Such movement could cause unacceptable interaction between the piping or its insulation and other commodities. This information appeared to be complete and included sufficient detail to enable a thorough analysis.

On August 1, 1989, SWEC forwarded the supplemental "As-Built Verification Packages" for various analyses performed by WEC. One of the packages included in this transmittal was the package for analysis 1-025. This transmittal detailed all changes to the information previously provided to installed components; and, since no revisions were required to the supports on this system (with the exception to the deletion of the one support discussed above), no additional stiffness information was required. This transmittal was made in order to enable WEC to generate their As-Built NIS-2 Design Report for 1-025.

The WEC Design Report was transmitted to the site with WPT-11924 on August 25, 1989. The NRC inspector has reviewed this report and concurs that it is complete and addresses all required conditions. Note that the NRC review documented in this report for this piping (part of the pressurizer surge line) does not include review of the effects of thermal stratification. This issue will be addressed in a separate report.

8. Plant Tours (42051C)

The NRC inspectors made frequent tours of Unit 1, Unit 2, and common areas of the facility to observe items such as housekeeping, equipment protection, and in-process work activities. No violations or deviations were identified and no items of significance were observed.

9. Exit Meeting (30703)

An exit meeting was conducted January 2, 1990, with the applicant's representatives identified in paragraph 1 of this report. No written material was provided to the applicant by the inspectors during this reporting period. The applicant did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. During this meeting, the NRC inspectors summarized the scope and findings of the inspection.

50-445/89-89; 50-446/89-89

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