APPENDIX

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-445/83-24
50-446/83-15

Docket: 50-445
50-446

Category: A2

Licensee: Texas Utilities Generating Company (TUGCO)
2001 Bryan Tower
Dallas, Texas, 75201

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

Inspection At: Comanche Peak, Units 1 and 2, Glen Rose, Texas

Inspection Conducted: March through July 1983

Inspectors: for D M Hunnicutt
R. G. Taylor, Senior Resident Inspector
Construction (SRIC)

8/19/83
Date

Approved: D M Hunnicutt
D. M. Hunnicutt, Chief
Reactor Project Section A

8/19/83
Date

Inspection Summary

Inspection Conducted March through July 1983 (Report 50-445/83-24 and 83-446/83-15)

Areas Inspected: Special inspections, announced and unannounced, related to allegations made to various NRC persons including the Atomic Safety and Licensing Board in their procedings regarding the operating license for Comanche Peak Station. The inspections involved 449 inspector-hours by one NRC inspector.

Results: The inspection confirmed the need to issue four violations initially identified by the Construction Appraisal Team (CAT) (NRC Inspection Report 50-445/83-18; 50-446/83-12). These involved the areas of HVAC, Equipment Installation, Document Control, and Storage of Equipment.

Details

1. Persons Contacted

Principal Licensee Employees

*R. G. Tolson, Site QA Supervisor
*C. T. Brandt, Non-ASME QC Supervisor
*J. R. Merritt, Engineering, Construction and Startup Manager
*J. B. George, Project General Manger
*D. N. Chapman, QA Manager
*B. R. Clements, Vice-President, Nuclear

Brown & Root (B&R)

*G. R. Purdy, Project QA Manager
*D. Frankum, Construction Project Manager

The SRIC also interviewed many other licensee, B&R, and subcontractor personnel during the course of the inspection.

*Denotes those persons who attended one or more management interviews with the SRIC.

2. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (50-445/82-22-02), "Analysis of Weld Discrepancies." This unresolved item concerned a substantial number of identified defects in a large whip restraint essentially surrounding the mainsteam and feed water lines located several feet outside of the ASME code boundry point. The device was engineered by the licensee's A/E and manufactured by NPS Industries. Due to the overall size of the structure, it has been nicknamed "George Washington Bridge" by the site labor and quality forces. The licensee had reported the finding of the defects as a potential 50.55(e) item to the SRIC on September 30, 1982, which was subsequently stated not reportable in a letter dated December 27, 1982. An NRC inspector followed up on the matter during a visit to the offices of the A/E, as documented in NRC Inspection Report 50-445/83-12. This review pertained to all of the defects involved with the exception of two cracked welds that had not been analyzed at the time of the inspection. The engineer has recently analyzed these two defects and has determined that had they not been detected, the structure could have fulfilled it's function. The SRIC has reviewed the location of the cracks and their length in relation to the size of the welds and the functional application of the structure. Since the structure has no continuous service application and is essentially subject to a one-time loading, the cracks would not have the potential for further propagation. Further, the cracks are at points in the structure that would receive relatively low stresses in the one-time impact based on their small size in relation to the members being welded. It appears that the cracks formed due to the stresses developed during the tightening of high strength bolting in the immediate vicinity of the welds during the site assembly of the structure. Taken in conjunction with the earlier documented review of the engineers calculations and the SRIC's review of these cracks, the SRIC has concluded that the engineer's overall analysis was adequate and that deficiency(s) were not reportable under 50.55(e). Both the licensee's initial report (CP-82-12) and the above identified unresolved item are considered closed.

It should be noted for the record that this closure only applies to the reportability aspects under 50.55(e) and not to the correction of the defects. The defects, including the cracks, have been documented on a nonconformance report. The final disposition and closure of the NCR will be evaluated during future routine inspections.

3. Review of Licensee Self-Evaluation (Using INPO Criteria)

The SRIC has reviewed a report of the licensee's self- evaluation performed during October 1982 which was based on criteria that has been developed for the purpose by INPO. The evaluation was performed in behalf of the licensee by personnel in the employment of Sargent & Lundy, an architect-engineer firm with substantial nuclear power involvement. A copy of the report was furnished to the NRC, and subsequently, to the Atomic Safety and Licensing Board in the matter of Comanche Peak Station operating license by letter dated May 2, 1983. The purpose of the review by the SRIC was to determine if any of the 47 findings in the report were of a type and of sufficient significance to have been reported to the NRC as required by 10 CFR 50.55(e). The SRIC reviewed each of the 47 findings and the supporting documentation in the report pertaining to each finding. This review revealed that none of the 47 items were based upon identified deficiencies in structures, systems, or components nor were there any significant deficiencies in design, engineering, or testing that would constitute conditions reportable under 10 CFR 50.55(e).

4. Car Wash In Containment

During the limited appearance statement portion of the Atomic Safety and Licensing Board hearing on May 16, 1983, a person stated at transcript page 6152 that he understood that the containment looked something like a car wash. The person stated that it was his understanding that the situation developed at about the same time that there was a meeting at the D/FW Airport between the NRC and any interested parties to discuss NRC decentralization. That meeting took place on April 5, 1983. For the purposes of evaluating this allegation, the SRIC expanded the period of interest to include the 3 weeks prior to the meeting. During this entire period, the Unit 1 reactor system was undergoing what is referred to as "Hot Functional Testing". This particular test is an accurate simulation of the operation of the reactor system and its appurtenances but without a reactor core being in place. The heat and pressure in the system is generated by the reactor coolant pumps in conjunction with the chemical and volume control system charging pumps. The test could readily be construed to be a pressure test but in fact is an operational test at pressure. This particular test extended overall for about 90 days beginning late in February and continuing until late May. The SRIC monitored the test but was by no means continously in the containment. The SRIC interviewed personnel in the licensee's startup test group, QC inspectors who had reason to be in the building and others to obtain a picture of the events that occurred in the Unit 1 Containment Building during the period of interest. The SRIO also reviewed the licensee's control room logs for any indication of operational problems indicative of a major leak in any of the fluid filled systems under test. The picture obtained was that there were several small leaks, generally at the gaskets between valve bodies and their bonnets. In addition, there was a considerable amount of condensation dripping from the reactor coolant pump motor cooling coils. This was caused by the cold water in the coils condensing the humidity from the atmosphere within the building and was not indicative of a leak in the reactor coolant system. The SRIO found from the control room logs that on March 29, a steam leak occurred during one phase of the test when a drain valve was partially open. Perhaps this valve should have remained closed. The room in which the valve was located was apparently filled with steam vapor which would have condensed out on the cooler walls as water. On March 30, the reactor vessel head vent valves were partially opened, which in turn would give some amount of steam blowoff into the reactor refueling cavity area and would rise up into the building until cooled and condensed out as water. None of these events are typical of any major leak indicative of piping or piping component (such as a valve) failure. The type of small events described above are, within the experience of the SRIC, typical of what would be expected during such a test and is one of the reasons for performing the test.

5. Design of the HVAC System Supports

By letters, both dated March 11, 1983, Citizens Association for Sound Energy (CASE) notified the NRC's Offices of Inspection and Enforcement and the Executive Legal Director of a concern that the HVAC system for Comanche Peak had not been properly supported, nor had it been properly considered in regard to seismic load conditions or its treatment as potential missiles. CASE specifically states that from their review of the FSAR, it appears that the licensee has not analyzed the HVAC supports for a seismic load condition. Specific reference is made to Sheet 21 of Table 17A. In addition, the personal observations of Messrs. Walsh and Doyle are relied upon to point out that there are no lateral supports on the HVAC systems within the containment. CASE also states that all HVAC components and supports inside containment should be treated as missiles under Criterion 4 of the General Design Criteria for Nuclear Power Plants, 10 CFR 50, Appendix A.

Sheet 21 of Table 17A of the FSAR lists the containment ventilation systems as being Seismic Category II. Apparently, it has been assumed by CASE that this category excludes seismic loading in the design. This assumption is incorrect since the FSAR, Section 3.2.1.2 defines Seismic Category II as being those portions of systems or components whose continued function is not required but whose failure could reduce the functioning of any Seismic Category I system or component required to satisfy the requirements of C.1.A through C.1.Q of Regulatory Guide 1.29 to an unacceptable safety level or could result in incapacitating injury to occupants of the control room. These systems are designated Non-Nuclear Safety (NNS) Seismic Category II and are designed and constructed so that a safe shutdown earthquake (SSE) will not cause such a failure.

CASE also states that if the HVAC systems within the containment failed during a SSE, this would allow the temperature within the containment to rise quickly to unacceptable levels which could over time cause components and monitoring equipment to fail and which could also mean that it might be impossible for workers to enter the containment due to the heat. Containment heat removal is required by Criterion 38 of the General Design Criteria for Nuclear Power Plants. The system to remove heat from the reactor containment at Comanche Peak does not rely on the HVAC system but rather is composed of two separate containment spray recirculation trains each with 100 percent capacity. Each train contains two separate pumps, one heat exhanger, and seven spray headers, and each system is fed from its individual electrical Class IE bus. The containment heat removal system is designed to ensure that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the system from accomplishing its planned safety function. CASE's concern with being able to enter the containment following certain design basis accidents is unfounded in that it is not a requirement.

In order to assess the adequacy of the design of HVAC supports, an inspection was conducted at the home office of "Corporate Consulting & Development Company, LTD.," the support design consultant. It was determined that all permanent HVAC supports are analyzed for seismic loading. Two methods are utilized: Zero Peak Accleration (ZPA), or 1.5 Times the Peak Acceleration When the Fundamental Frequency Falls Below 20 Hertz. Of the latter method of design, only about 6 out of 4000 supports have been designed that way. A typical HVAC duct run is supported axially at every third support This may explain why Messrs. Walsh and Doyle may have felt that there were no lateral supports on the HVAC systems. The NRC inspector reviewed the design of a typical HVAC duct run at elevation 852'-6" in the Auxiliary Building. Supports were designed utilizing two computer programs entitled FEASA-2D and FEASA-3D. The acronym stands for frame eigenvalue and stress analysis. The -2D version is used on the transverse supports and the -3D version is used on the axial supports. The inclusion of equivalent weights from both up and downstream transverse supports and accesories such as volume dampers and vane turns in the design of the axial supports was verified. This inspection verified the adequacy of the siesmic design techniques being utilized for the design of HVAC supports at Comanche Peak.

The concerns expressed by CASE have been found to be without merit.

Persons contacted during the course of the inspection at Corporate Consulting

& Development Company, LTD. were:

J. Roland Yow, President & Chief Executive Officer
Gary Hughes, Vice-President for Operations
David Lindley, Principal Engineer
Stephen Lehrman, Seismic Department Manager
Daryl Hughes, Project Engineer

6. Heating, Ventilation, and Air Conditioning System (HVAC)

During the CAT inspection (NRC Inspection Report 50-45/83-18; 50-446/83-12), the CAT inspectors noted that a significant portion of the welds on the ducting support structures were deficient in relation to the applicable welding code requirements. The dominate deficient condition noted was that the welds were significantly undersized. Based upon this information the SRIC toured various areas of the facility with special emphasis on the ducting in the Unit 2 Containment Building since that was one of the more recent areas of installation by the HVAC contractor. In accordance with the design drawings, the bulk of the welds should have been fillet welds with ½inch leg size. The SRIC noted by visual comparison to the ½inch thick base metal that very few of the welds were of proper size. The CAT inspectors also found cases where the bolting and gaskets between ducting sections were loose and/or missing. The CAT inspectors also found that some support members were not within the dimensional tolerances on the design drawings. It was noted that the contractor's inspection records did not reveal these various facts, indicating ineffectual QC by the contractor. Further, a review of the licensee's audit program indicated that the licensee was unaware of these several problems in the fabrication, installation, and inspection of the HVAC systems. Based upon the CAT inspectors' findings and his own observations, the SRIC recommended that a notice of violation be issued to the licensee pertaining collectively to these matters (Notice of Violation issued on May 31, 1983. Reference 50-445/83-18 and 50-446/83-12, item 4).

7. Installation of Major Items of Equipment

The CAT inspectors noted during their inspections of certain major items of equipment that there were several variables in how the equipment was fastened to the building equipment pads. In some instances. tanks for example, CAT inspectors found that there were two nuts (double nuts) on the embedded bolts securing the equipment, other bolts had one nut, (single nut) and some had a combination of both single nuts and double nuts on one piece of equipment. The CAT personnel also noted that certain heat exchangers had slotted holes in one of the mounting bases to allow for thermal expansion during operation. The holddown nuts appeared to be installed too tightly and may have prevented freedom of movement. The SRIC obtained the design and installation drawings for two of the referenced heat exchangers identified in the CAT report. Both were found to be horizontal Utube heat exchangers whose function is nonsafety, but whose pressure boundary in the tubes is safety-related since the process fluid could be radioactive. The SRIC found that the construction drawings for the mounting pedestals had a flat steel plate on one pedestal that would be suitable for the type of mounting detail on these heat exchangers. The SRIC then reviewed the installation travelers for each heat exchanger and found that these documents did not note or address the slotted details, the plate, or the fact the bolts should be left loose. The SRIC would note that the vendor manual which provides the details does not provide information on how loose or tight the nuts should be nor how these nuts are to be locked at that looseness or some torque value. The SRIC with the assistance of site QC and craft labor had one of six nuts loosened on heat exchanger TCX-CSAHLD-01. On all six of the studs involved, each had only one nut (single nut). The one nut that was loosened had been very tight, as evidenced by the amount of force required to break the nut loose. On another heat exchanger of comparable design, it was found that each stud was double nuted and when the top nut was loosened, the second nut was approximately one flat (about 1/6 of a turn) from being fully tight. This degree of looseness should allow sufficient freedom of movement. During the document review, the SRIC found that the engineer had specified that all rotating and vibrating equipment should be double nutted and that other equipment could be secured with only one nut. No document could be located that established the identity of vibrating equipment nor were there any apparent provisions made to lock nuts where they must be deliberately left loose. This was considered overall to be a violation of Criterion V of Appendix B to 10 CFR 50 (Notice of Violation was issued on May 31, 1983. Reference: Notice of Violation 50-445/83-18 and 50-446/83-12, item 1).

8. Maintenance of Equipment In Outdoor Storage Areas

The CAT found that a considerable amount of equipment such as pipe support struts, clamps, and like items, normally stored outdoors, was not being properly maintained in accordance with procedure MCP-10, "Storage and Storage Maintenance of Mechanical and Electrical Equipment", as evidenced by rusting bolts and adjustment screws on struts. In addition, the strut bearings were dirty from dust and the bearing load pins, in some instances, were rusted. By a tour of the storage areas, the SRIC confirmed the CAT inspectors findings. The SRIC would also note that the INPO Self-Evaluation Report at page 111 describes essentially the same finding. This situation was determined to be a violation of Criterion XIII of Appendix B to 10 CFR 50 (Notice of Violation issued on May 31, 1983. Reference: Notice of Violation 50-445/83-18 and 50-446/83-12, item 2). The SRIC would note for the record that there is little evidence that any items which indicated substantial deterioration from such storage conditions have in fact been installed in the nuclear power block. It would appear that the various items involved have been cleaned and restored prior to installation such that they can perform the required function.

9. Obsolete and/or Illegible Drawings In The Field

The CAT inspectors found a group of drawings in one particular area adjacent to the control room that were found to be out of date by up to several issues and further, that some drawings in other areas were incomplete in the title and revision blocks. The SRIC discussed the finding with supervisory personnel of the licensee's central document control center who indicated that they had located the drawings identified by the CAT inspectors along with many more that were obsolete in other areas. It was stated that distribution system for engineering drawings had become faulted by the simple volume and by the need for so many points of distribution and audit verification thereof. Since problems are obviously still present, it was determined that the licensee had violated Criterion VI of Appendix B to 10 CFR 50 (Notice of Violation was issued on May 31, 1983. Reference: Notice of Violation 50-445/83-18 and 50-446/83-12, item 3) and that substantial steps would be required to correct the problems.

10. Allegations Relative To Improperly Supported Items In The Control Room

The president of CASE in a letter dated March 11, 1983, addressed to Mr. Richard C. DeYoung, Director of the NRC Office of Inspection and Enforcement, indicated that CASE had received information from an unidentified source to the effect that:

a. There is field run conduit above the control room supported only by wire.

b. There is drywall (or sheet rock) that is supported by wire.

c. There may be lights that are supported by wire.

The SRIC has examined the suspended ceiling and the area above the suspended ceiling in the control room area and has examined the pertinent engineering drawings depicting both in relation to these allegations with the following findings:

a. There is a considerable amount of both safety-related and nonsafety related conduit in the area above the suspended ceiling. The safety-related conduit is supported by Seismic Category I supports typical of those used in other areas of the facility. The nonsafety-related conduits are generally supported by simpler and less substantial supports that are typical of those that the SRIC has observed in large open factories and are not designed to seismic standards. In each case examined, the non-seismic support was structurally paralleled with a small stainless steel cable that would assume the full weight of the conduit were the normal support to fail in a seismic event.

b. The drywall materials were found to be part of the suspended ceiling above the central part of the control room and to form a part of the sloping wall area below the control room observation room. These drywall materials have been securely fastened to a metal frame work (metal batten) which in turn is supported by conventional and non-seismic straps and wires to the concrete primary building. The frame work is also attached to a system of stainless steel cables which in turn also attach to the primary structure such that if normal supports fail during a seismic event, the weight of the framing and drywall will be assumed by the cabling thus preventing the materials from falling.

c. The lighting fixtures in the control room are supported from an intermediate substructure of "unistrut" by light-weight conduit. The substructure is likewise supported by the same type of conduit from the primary structure ceiling. The conduit used appears to be the typical of that supporting the light fixtures in most offices with suspended ceilings. Paralled with each conduit are two small stainless steel cables which would assume the load if the conduit or its attachment were to fail. In the case of the actual light fixtures, the cable is attached to the light fixture at the edge of the reflector assembly.

The SRIC would note for the record that above described design features appear to fully satisfy the intent of the licensee's commitment to comply with NRC Regulatory Guide 1.29, "Seismic Design Classification." The licensee has used terminology in the classification system that is at variance with that of the regulatory guide but is explained and defined in Section 3.2 of the FSAR. In essence, the licensee has defined all safety-related items that must remain fully functional during and after a seismic event as Seismic Category I. Items not having a safety function but whose failure could damage components which have a safety function or cause injury to the occupants of the control room during an event are referred to as Seismic Category II. In the case of the items involved in this allegation, all are Seismic Category II since their falling could cause injury to the control operators. The cabling system described can be expected to prevent such a fall even though the normal supports could possibly fail. The stainless steel cable used in this design feature, which at a short distance away looks much like bright galvanized common steel wire, is of relatively high strength. As an example, the test strength of an 1/8-inch cable is in excess of 1760 pounds. With four cables attached to a light fixture, two at each end, the total support capability of the cables is over 7000 pounds. It is apparent that the designers have elected to use conventional suspended ceiling and light fixture support techniques in order to use conventional and available materials and then provide a high strength backup support system in a seismic event.

No violations or deviations were identified during this special inspection effort.

11. Placement and Curing of Concrete During Freezing Weather

During the limited public appearance portion of the Atomic Safety and Licensing Board (Board) hearing conducted on May 15, 1983, there were two references to the placing of concrete in freezing weather at the Comanche Peak Station which in turn lead to a question from the Board to the NRC staff as to whether there were any NRC personnel present with knowledge of the matter. The two references are at 6106 and 6134 of the hearing transcript while the Board question is at 6109. Also at 6109, an unidentified voice responded to the Board that the matter had been reported in IE inspection reports. Research of the NRC inspection reports revealed that there had been such a discussion in NRC Inspection Report 50-445/77-01 which was categorized as an unresolved item pending the licensee's review and action on their finding of the problem. The unresolved item was further discussed in NRC Inspection Report 50-445/77-04 with the closure of the item by an improvement in the QA procedures.

The SRIC has reviewed the matter, particularily with a view toward determining whether the practices involved actually caused damage to the concrete involved. The primary focus of NRC Inspection Report 50-445/77-01 (Details II, paragragh 5) was directed toward two licensee "Site Surveillance Reports" which had been prepared approximately 2 weeks earlier than the inspection period covered by the inspection report. The first of the licensee's reports (C-134-77) was directed specifically to findings by a licensee inspector that the surface temperature of Concrete Placement 101-2808-001 some 6 hours after the placement was completed were well below freezing in some locations. The other licensee report (C-135-77) was directed toward records and was not considered in this review. The SRIC obtained the necessary records to review the matter and found that placement 101-2808-001 had taken place on December 30, 1976, being completed at approximately 6:00 p.m. Later, the same evening at approximately midnight, the licensee inspector found that some surface areas were chilled to as low as 20°F. The records reflect, however, that there was disagreement between the B&R inspection personnel assigned to monitoring the curing of the placement and the licensee's inspector as to what the surface temperatures actually were. The B&R personnel contended that the licensee inspector was actally measuring the air temperature rather than the temperature of the concrete. No resolution of that disagreement was reflected in the records. The SRIC interviewed the licensee inspector of record during the course of this review to gain a clearer understanding of the events which took place. The licensee inspector stated during the interview that he was confident that his measurements were accurate and also stated that there was no physical evidence that the concrete was frozen even though the surface temperatures were well below freezing. The records also reflect that in order to resolve the issue, swiss hammer tests were run on the suspect areas after the concrete had fully cured. These tests indicated that the suspect areas had attained strengths comparable to known properly cured areas, indicating that the concrete had not been damaged even though the possibility exists that it had been frozen for a period of time. The records reflect that good concrete curing temperatures, i.e., above 40°F were established and maintained shortly after the licensee's inspector's observation.

For the record, the SRIC would note that Placement 101-2801-001 took place in the Unit 1 Reactor Building. The placement became the open area floor at the lowest full floor in the building. This floor area, while supporting some equipment, serves primarily as a walk area. As such, it is fully topped with an architural concrete making the structural concrete no longer accessable.

NRC Inspection Report 50-445/77-01 also discussed comparable events to that documented on Surveillance Report C-135-77. One of these events was documented by Surveillance Report C-068-76 on January 7, 1976, and on B&R deficiency/disposition reports (now titled nonconformance reports). These documents indicate that on January 7, 1976, the surface temperature of Placement 105-2773-001, the foundation basemat for the Unit 1 Safeguards Building, were found frozen as evidenced by frozen wet burlap over certain areas that were not covered by insulating blankets. The records also reveal that the reported finding took place almost 7 days after the placement of the concrete. Although the placement should not have been allowed to freeze in the time frame involved in accordance with the project specification, the placement was accepted "use-as-is" on the premise that the curing temperatures during the 7 days were conducive to a good cure and that after 7 days there would be little free water in the concrete to freeze even though the burlap was froze. This conclusion is considered valid by the SRIC based on his review of publications of the American Concrete Institute and the Bureau of Reclamation. Further, in responding to a separate finding that the field cure test cylinders made for the placement tested lower than allowed by the project specifications, swiss hammer tests were performed. The swiss hammer tests indicated the concrete placement had full specified strength. Relative to the low reported strengths of the field cure cylinders, the SRIC would note that in his experience field cure cylinders will frequently test low under cold weather conditions. The reason is that the cylinders' small mass generates little heat of hydration, thus making them either more vulnerable to freezing and/or curing much slower than normal due to their depressed temperature.

The final events covered by NRC Inspection Report 50-445/77-01 included DDR-C-460 which in turn discussed low temperatures during the curing period of three separate placements that were made during the late December time period of 1976. In each case, the records reflect that the placements were accepted "use-as-is" since the least amount of cure time was 9 days, again with good conditions until the cold weather occurred.

The NRC inspector involved in NRC Inspection Report 50-445/77-04 which closed the unresolved issue has stated that he had visually inspected each of the placements discussed in NRC Inspection Report 50-445/77-01 for evidence of damaged concrete and found none. NRC Inspection Report 50-445/77-04 did not reflect those inspections since the NRC inspector was aware that the concern was for prevention of repetition rather than any specific concern about the quality of the placements involved.

The SRIC would note for the record that there are no regulatory or industry prohibitions on placing concrete in cold weather conditions. The American Concrete Institute and the Bureau of Reclamation both indicate that if the fresh concrete is above 40^{O}F at the time of placement, the chemical process of hydration will generate sufficient heat to prevent the concrete from freezing provided that precautions are taken to prevent heat loss. In mass concrete applications, the greatest danger to the concrete is on the exposed surface areas, particularily at corners and other edges of the placement. It would be exceedingly rare for the mass of the concrete to freeze and sustain damage. These publications also indicate that even if frozen, the concrete will normally cure to full design strengths if temperatures conducive to the hydration process are restored.

12. Allegations Relative To The As-Built Verification and Design Verification Activities.

During April 1983, NRC personnel received allegations to the effect that the QA group performing as-built verifications were not measuring support member dimensions and therefore, the "Vendor Certified Drawings" of the supports would not be accurate. A second allegation from the same person indicated that the QA group charged with responsibility for verifying that design changes have been incorporated into the plant and that the inspection records for the installations accurately reflected that incorporation was being required with the use of a computer generated status document to make the verification of records. The allegation was that the computer listing was faulty and therefore, the verification effort was equally faulted.

The SRIC has examined each of these allegations as to the factualness of the allegation and as to whether the allegation has or will have an effect on the safety of the facility when operating. In regard to the first allegation, the SRIC found that the allegation was and is factual. The allegation, however, does not appear to have any significant impact on safety in that the as-built inspection was not developed to assure that the "Vendor Certified Drawing" was an accurate representation of the support in all aspects. The as-built program was established to assure only that the support location on the supported pipe and the direction of support is accurate for the purposes of performing the final pipe stress analysis. The responsibility for assuring that the support members and other characteristics of the individual support reflect the design drawing requirements reside in other QA groups associated with the fabrication and installation efforts. To also perform these functions in the as-built verification inspection would be a redundant inspection that would not contribute significantly to the safety function of any given support.

Regarding the second allegation, the SRIC found that it too was factual but only at the specific time the allegation was made. When making the allegation, the alleger provided the NRC personnel with a reference to a QC inspection report which he said would fully display his concern. This report, identified as IR DCV-00421, was found to contain notation that the verification was based on a computer tabulation and that the report was being completed at the direction of the inspector's supervisor. The original report was dated April 4, 1983. The permanent file copy was found to have been marked "voided" by the originating inspector as of May 20, 1983, with a notation that the report had been superceded by IR DCV-00423. This latter inspection report was examined by the SRIC and found to document essentially the same inspection effort by the same inspector but without any notation of having been based upon a computer tabulation and without notation of apparent protest of directions given by supervision. The SRIC interviewed the QC inspector who prepared and signed all of the reports noted above in order to ascertain what had and is transpiring in the QC design verification program effort. The inspector stated that the attempt to use the computer based data in the performance of the assigned task was in error from the beginning because of errors by persons generating the computer data. The interviewee stated that only the one verification effort had been done using the computer based data and that all prior and subsequent verifications have been done by the assigned inspectors directly and personally examining the existent quality records in compliance with applicable QC procedures for the task. He stated that the only procedural deviation was the one instance stated in the allegation. Discussions between the group supervisor at the time the allegation was received and the SRIC indicated that he had attempted to use the computer tabulation to expedite the task on a trial basis by management direction and that he had caused the original inspection report to be filed as it was to give management a picture of the faults in the computerized data. It thus appears that the design verification effort has been performed in accordance with procedures except for the one-time pertubation that was subsequent correctly reaccomplished in accordance with approved procedures.

No violation to NRC requirements were revealed during this special inspection effort.

13. Improperly Certified Liquid Penetrant Examination Materials

The CASE informed the Atomic Safety and Licensing Board by a letter dated May 18, 1983, of a potential problem with the liquid penetrant materials in use at the Comanche Peak Station. The letter stated that CASE had been made aware of the potential problem during a phone conversation with Charles A. Atchison, who in turn learned of the "problem" from a Dallas area representative of the Magna-Flux Corporation, the orginal manufacturer of the material. The letter states that the problem surfaced only 7 to 10 days earlier. Based on the date of the letter, it would seem that the problem arose between approximately May 8 to May 11, 1983.

The situation bears close resemblance to the situation outlined beginning with NRC Inspection Report 50-445/82-18;50-446/82-09 based upon an inspection conducted during the period of September 7-10, 1982. The NRC inspector noted that some certified test result documents had been altered by "pen and ink" changes not immediately explainable. The matter was considered unresolved at that time. During a second inspection of the matter, conducted during November 1982 and documented in NRC Inspection Report 50-446/82-11, the inspector found that previous corrective actions were not adequate and further that the "pen and ink" changes sometimes didn't match the type of material being certified. A Notice of Violation was issued as part of the inspection report on the matter. The licensee responded to the Notice of Violation by a letter dated December 21, 1982, wherein he stated that a supplier had altered the certificates but that the original manufacturer had been able to furnish valid certificates and further, that all future purchases would be direct from the manufacturer rather from a "middle-man" supplier. The licensee also stated that specific receiving inspection procedures had been implemented to prevent repetition. NRC Inspection Report 50-445/83-10;50-446/83-05 documented verification that the licensee's actions were acceptable and the matter was closed.

It appears that the situation outlined in the CASE letter parallels the NRC findings in all details except for the dates which probably arose as a result of misunderstood or incomplete communications between the

Magna-Flux representative and Mr. Atchison and/or with CASE.

CASE also posed two questions on the matter as follows:

a. Has an NCR been written on this problem?

   Answer: The above discussed inspection reports document a total of five NCR's that were issued.

b. Has either TUGCO or Texas Utilities or B&R notified the NRC of this problem?

   Answer: The roles of reportability were effectively reversed in that the NRC identified the problem and notified the licensee.

A need for further NRC action on this matter has not been identified and the matter is considered closed.

14. Penetration Seals

This special inspection was undertaken to ascertain the validity and significance of allegations received initially by an NRC Headquarters Duty Officer on or about March 22, 1983, which were confirmed and added to during a telephone interview with the alleger on March 23, 1983, by the SRIC and a NRC inspector assigned to NRC Region I. The allegations, as understood by the SRIC, were:

a. The overlap seal for flexible boots should be 3 inches whereas 2 inches is being used by BISCO.

b. There maybe a problem with the strength of the fabric used in the flexible boots since the material supplier and BISCO are involved in a lawsuit.

c. The aggregate used in a radiation seal may separate giving rise to improper personnel protection.

Since BISCO was and is on the Comanche Peak site installing seals, Region IV was selected for the purpose of this special inspection although the company has involvement at several other nuclear power sites throughout the United States. The SRIC obtained from the BISCO site manager all of the production and quality procedures applicable to the work at CPSES as well as some that are not. The alleger specifically mentioned that the NRC should review Procedures QC-507, SP-504, SP-505, SP-505-1, and SP-505-2 in regard to the flexible boot overlap problem. Each of the above procedures was in the books offered to the SRIC for review. A brief discussion follows as to the contents of these procedures:

a. QCP-507: This procedure covers the final inspection of installed flexible boots. The amount of overlap is not mentioned in the procedure, although the procedure does require that the seam be examined for evidence of poor sealing such as "fish-mouthing" which is taken to mean that the exposed edge of the overlap is puckered and not adhering to the base fabric.

b. SP-504: This procedure provides instructions and a calculation sheet to initially cut the fabric into a shape that would subsequently allow the formation of a truncated cone. The formula on the calculation sheet requires that 1-inch be added at each edge of the fan shaped fabric which is evidently to provide the overlap. The base formula prior to adding the 1-inch provides a dimension just equal to the circumference of the pipe and/or sleeve to which the boot will be attached. Thus, the 1-inch at each edge will provide for 2-inches of overlap, assuming that the pipe and sleeve are concentric. If pipe and sleeve are not concentric, the resulting cone will be skewed and the seam overlap will be something other than 2-inches.

c. SP-505: This is a generic procedure for the installation of flexible boots. It was noted that the procedure requires that the adhesive for the overlap seam be spread over a 3-inch depth from the fabric edge prior to fitting up the fabric where it is to be installed. Although not so stated, it appears that the 3-inch width of adhesive is to provide sufficient area of adhesive in the event the above mentioned cone skewing occurs.

d. SP-505-1 and SP-505-2: These are additions to SP-505 having application when the boots are used as a simple pressure seal only and for when the boot is used as part of a fire protection seal, respectively.

The SRIC interviewed the BISCO site manager as to whether the procedures had ever required a 3-inch overlap. The site manager indicated that 3-inch seam had been used up to sometime in 1979 and that his homeoffice engineering had then changed the seal seam detail. The SRIC reviewed the results of a pressure differential test performed by BISCO in September 1979 which indicated that the fabric boot would withstand a differential pressure of 44 psig without sustaining damage. The project specification (2323-MS-38F) requires that the pressure seal maintain its integrity only up to 2 psig. While the BISCO test data does not specifically state what the overlap seam width was on the test boot, it would strongly appear that the strength margin is so high that even a reduction of 1/3 in the area of the overlap would have the effect of changing the safety factor from 22:1 to approximately 14:1. It is the SRIC's conclusion that while the allegation relative to the reduction in seam from 3 to 2 inches is correct, the reduction would have no significant effect on the performance of the boot in service at CPSES and that, therefore, the allegation has no technical merit.

Regarding the matter of the possibility of some undefined problem with the boot fabric, the BISCO site manager stated that his company has been engaged in a law suit with the supplier of the fabric but only in regard to the performance of the fabric in one application which is understood to involve the tearing of the fabric after being punctured. It is understood that the puncturing has occurred when a gel type radiation seal hardens under radiation. Since the specific design involved is not scheduled for use at CPSES, the allegation has no technical merit.

Regarding the matter of possible separation of the radiation seal aggregate material from the carrier material, the SRIC can only conclude that the allegation is potentially correct but without apparent merit. The BISCO test reports indicate that the seals involved met the engineers specification. The separation of the aggregate (powdered lead) from the carrier (a silicone material) would appear to be process sensitive in that if they are not well mixed, pockets of lead might form with resulting pockets of silicone without sufficient lead. Since the specification and the BISCO procedures require careful control and monitoring of the mixing process, the SRIC can only conclude that these measures are effective in production operations as they were in preparation of the test samples.

15. Electrical Cable Splicing

The SRIC became aware that the Comanche Peak project electrical engineer had authorized the splicing of safety-related and auxiliary electrical cables within several control panels during the inspection period. Since the licensee has committed in FSAR Section 8.1 to comply with IEEE 420, "Trial-Use Guide for Class 1E Control Switchboards for Nuclear Power Generating Stations," which forbids splicing of wiring in such panels, the SRIC judged that the licensee was deviating from these commitments. The licensee engineer indicated that he interpreted the IEEE standard to prohibit such splicing only between the cabinet terminal boards and the cabinet devices and did not prohibit such splicing in the field run cables attaching to the terminal boards. The engineer stated that action had been initiated with the NRC Office of Nuclear Reactor Regulation to clarify the issue in the FSAR. The SRIC confirmed that such action had been initiated by a telephone conversation with the NRR Licensing Program Manager for Comanche Peak. Pending action by NRR, this matter will be considered as an unresolved matter.

16. 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 such item, disclosed during the inspection, is discussed in paragraph 15 above. This item is identified as "Splicing of Electrical Cables in Cabinets." (8324-01)

17. Management Interviews

The SRIC met with one or more of the persons identified in paragraph 1 of this report at frequent intervals during the inspection period to discuss the licensee's position and proposed actions on a significant number of issues which occurred during the period.

NRC FORM 766
(11-81)
IE MC 0535

U.S. NUCLEAR REGULATORY COMMISSION

INSPECTOR'S REPORT
Office of Inspection and Enforcement

PRINCIPAL INSPECTOR (Name, last, first, and middle initial): TAYLOR, R. G

REVIEWER: HUNNICUTT, DM

INSPECTORS: R. G TAYLOR

LICENSEE/VENDOR	TRANSACTION TYPE	DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits)	REPORT NO.	SEQ.	NEXT INSPEC. DATE MO	YR
Texas Utilities Generating Company (TUGCO)	☒ I – INSERT	05000445	8324	A		
	☐ M – MODIFY	05000446	8315	B		
	☐ D – DELETE			C		
	☐ R – REPLACE			D		
	1	2 – 14	15 – 18			

PERIOD OF INVESTIGATION/INSPECTION FROM MO.	DAY	YR.	TO MO.	DAY	YR
03	01	83	07	31	83
20		25	26		31

INSPECTION PERFORMED BY
☐ 1 – REGIONAL OFFICE STAFF
☒ 2 – RESIDENT INSPECTOR
☐ 3 – PERFORMANCE APPRAISAL TEAM
☐ OTHER
32

ORGANIZATION CODE OF REGION/HQ CONDUCTING ACTIVITY (See IEMC 0530 "Manpower Reporting—Weekly Manpower Reporting" for code.)

REGION	DIVISION	BRANCH
4	C	A
33	34	35

REGIONAL ACTION (Check one box only)
☐ 1 – NRC FORM 591
☒ 2 – REGIONAL OFFICE LETTER
36

TYPE OF ACTIVITY CONDUCTED (Check one box only)
☒ 02 – SAFETY
☐ 03 – INCIDENT
☐ 04 – ENFORCEMENT
☐ 05 – MGMT. AUDIT
☐ 06 – MGMT. VISIT
☐ 07 – SPECIAL
☐ 08 – VENDOR
☐ 09 – MAT. ACCT.
☐ 10 – PLANT SEC.
☐ 11 – INVENT. VER.
☐ 12 – SHIPMENT/EXPORT
☐ 13 – IMPORT
☐ 14 – INQUIRY
☐ 15 – INVESTIGATION
37-38

INSPECTION/INVESTIGATION FINDINGS (Check one box only)

A	B	C	D	
				1 – CLEAR
X	X			2 – VIOLATION
				3 – DEVIATION
				4 – VIOLATION & DEVIATION

39

TOTAL NUMBER OF VIOLATIONS AND DEVIATIONS

A	B	C	D
04	04		

40-41

ENFORCEMENT CONFERENCE HELD

A	B	C	D	
				1 – YES

42

REPORT CONTAIN 2.790 INFORMATION

A	B	C	D	
				1 – YES

43

LETTER OR REPORT TRANSMITTAL DATE

NRC FORM 591 OR REG. LETTER ISSUED MO.	DAY	YR.	REPORT SENT TO HQ. FOR ACTION MO.	DAY	YR.
05	24	83			
44		49	50		55

RG

MODULE INFORMATION

REC ORD TYPE	MODULE NUMBER INSP. NUMBER	PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL	SEQ *	PRIORITY	DIRECT INSPECTION EFFORT IN STAFF HOURS EXPENDED THIS INSPECTION	PERCENTAGE COMPLETED TO DATE	STATUS	MODULE REQ. FOLLOWUP PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL
1		2	99	706	E	(A)		249						
						(B)		200						
						C								
						D								
						A								
						B								
						C								
						D								
						A								
						B								
						C								
						D								
						A								
						B								
						C								
						D								

MODULE INFORMATION

REC ORD TYPE	MODULE NUMBER INSP. NUMBER	PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL	SEQ *	PRIORITY	DIRECT INSPECTION EFFORT IN STAFF HOURS EXPENDED THIS INSPECTION	PERCENTAGE COMPLETED TO DATE	STATUS	MODULE REQ. FOLLOWUP PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								

* CIRCLE SEQUENCE IF VIOLATION OR DEVIATION

C FORM 766 A
81)
MC 0535

INSPECTOR'S REPORT
(Continuation)
Office of Inspection and Enforcement

DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits)	REPORT NO.	SEQ.
05000445	8324	A
05000446	8315	B
		C
		D

MODULE NUMBER 299706B 2.D

VIOLATION SEVERITY OR DEVIATION: 1 2 3 4 [✓ 5] 6
27

SITE RELATED: 5 A C; 5 B D
28

OLATION OR DEVIATION (Enter up to 2400 characters for each item. If the text exceeds this number, it will be necessary to paraphrase. Limit lines to 50 characters each.)

Major items of equipment were installed without specific documentation that could establish the identity of vibrating equipment nor were there provisions made to lock nuts (single or double) where the nuts must be left loose to assure allowance for expansion ~~durin~~ during changes in operating conditions. This is a violation of criterion V, 10 CFR 50, Appendix B.

RC FORM 766 A
1-81)
MC 0535

INSPECTOR'S REPORT
(Continuation)
Office of Inspection and Enforcement

DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits)	REPORT NO.	SEQ.
05000445	8324	A
05000446	8315	B
		C
		D

MODULE NUMBER 244706B 2.D

VIOLATION SEVERITY OR DEVIATION 1 ☐ 2 ☐ 3 ☐ 4 ☐ 5 ☒ 6 ☐ 27

SITE 5 A C 5 B D 28

VIOLATION OR DEVIATION (*Enter up to 2400 characters for each item. If the text exceeds this number, it will be necessary to paraphrase. Limit lines to 50 characters each.*)

Major items of equipment were installed without specific documentation that could establish the identity of vibrating equipment nor were there provisions made to lock nuts (single or double) where the nuts must be left loose to assure allowance for expansion ~~durw~~ during changes in operating conditions. This is a violation of criterion V, 10 CFR 50, Appendix B.

U.S. NUCLEAR REGULATORY COMMISSION

C FORM 766 A
C 0535

INSPECTOR'S REPORT
(Continuation)
Office of Inspection and Enforcement

DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits)	REPORT NO.	SEQ.
05000445	83 24	A
05000446	83 15	B
		C
		D

MODULE NUMBER 299706B

2 D

VIOLATION SEVERITY OR DEVIATION

1	2	3	4	5	6	D
			X			☐

27

SITE RELATED
A S C ☐
B S D ☐

28

OLATION OR DEVIATION (Enter up to 2400 characters for each item. If the text exceeds this number, it will be necessary to paraphrase. Limit lines to 50 characters each.)

A group of drawings in one particular area adjacent to the control toom were found to be out of date by up to several issues. This failure to maintain current drawings is a violation of Criterion VI, 10 CFR 50, Appendix B.

NRC FORM 766
(11-81)
IE MC 0535

U.S. NUCLEAR REGULATORY COMMISSION

INSPECTOR'S REPORT

Office of Inspection and Enforcement

PRINCIPAL INSPECTOR *(Name, last, first, and middle initial)*

REVIEWER

INSPECTORS R. G. TAYLOR

LICENSEE/VENDOR	TRANSACTION TYPE	DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits)	REPORT NO.	REPORT SEQ.	NEXT INSPEC. DATE MO.	NEXT INSPEC. DATE YR.
TUGCO	I – INSERT	05000445	8324	A		
	M – MODIFY	05000446	8315	B		
	D – DELETE			C		
	R – REPLACE			D		
	1	2 … 14	15 … 18			

PERIOD OF INVESTIGATION/INSPECTION FROM MO.	DAY	YR.	TO MO.	DAY	YR.	INSPECTION PERFORMED BY	OTHER	ORGANIZATION CODE OF REGION/HQ CONDUCTING ACTIVITY (See IEMC 0530 "Manpower Reporting—Weekly Manpower Reporting" for code.) REGION	DIVISION	BRANCH
						1 – REGIONAL OFFICE STAFF 2 – RESIDENT INSPECTOR 3 – PERFORMANCE APPRAISAL TEAM				
20		25	26		31	32		33	34	35

REGIONAL ACTION (Check one box only)
- ☐ 1 – NRC FORM 591
- ☐ 2 – REGIONAL OFFICE LETTER

36

TYPE OF ACTIVITY CONDUCTED (Check one box only)
- ☐ 02 – SAFETY
- ☐ 03 – INCIDENT
- ☐ 04 – ENFORCEMENT
- ☐ 05 – MGMT. AUDIT
- ☐ 06 – MGMT. VISIT
- ☐ 07 – SPECIAL
- ☐ 08 – VENDOR
- ☐ 09 – MAT. ACCT.
- ☐ 10 – PLANT SEC.
- ☐ 11 – INVENT. VER.
- ☐ 12 – SHIPMENT/EXPORT
- ☐ 13 – IMPORT
- ☐ 14 – INQUIRY
- ☐ 15 – INVESTIGATION

37-38

INSPECTION/INVESTIGATION FINDINGS (Check one box only) A B C D
- 1 – CLEAR
- 2 – VIOLATION
- 3 – DEVIATION
- 4 – VIOLATION & DEVIATION

39

TOTAL NUMBER OF VIOLATIONS AND DEVIATIONS: A | B | C | D — 40-41

ENFORCEMENT CONFERENCE HELD: A | B | C | D — 1 – YES — 42

REPORT CONTAINS 2.790 INFORMATION: A | B | C | D — 1 – YES — 43

LETTER OR REPORT TRANSMITTAL DATE
- NRC FORM 591 OR REG. LETTER ISSUED: MO. DAY YR. — 44 … 49
- REPORT SENT TO HQ. FOR ACTION: MO. DAY YR. — 50 … 55

R

MODULE INFORMATION

REC ORD TYPE	NUMBER	MODULE NUMBER INSP. PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL	SEQ. *	PRIORITY	DIRECT INSPECTION EFFORT IN STAFF HOURS EXPENDED THIS INSPECTION	PERCENTAGE COMPLETED TO DATE	STATUS	MODULE REQ. FOLLOWUP PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								

MODULE INFORMATION

REC ORD TYPE	NUMBER	MODULE NUMBER INSP. PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL	SEQ. *	PRIORITY	DIRECT INSPECTION EFFORT IN STAFF HOURS EXPENDED THIS INSPECTION	PERCENTAGE COMPLETED TO DATE	STATUS	MODULE REQ. FOLLOWUP PHASE	MANUAL CHAPTER	PROCEDURE NUMBER	LEVEL
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								
B						A								
						B								
						C								
						D								

* CIRCLE SEQUENCE IF VIOLATION OR DEVIATION

RC FORM 766 A
(1-81)
MC 0535

INSPECTOR'S REPORT
(Continuation)
Office of Inspection and Enforcement

DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits)	REPORT NO	SEQ.
05000445	8324	A
05000446	8315	B
		C
		D

MODULE NUMBER 299766B   20

VIOLATION SEVERITY OR DEVIATION

1	2	3	4	5	6
			X		

27

SITE RELATED
S A C
S B D

28

VIOLATION OR DEVIATION (Enter up to 2400 characters for each item. If the text exceeds this number, it will be necessary to paraphrase. Limit lines to 50 characters each.)

A significant portion of the welds on the ducting support structures were deficient in relation to the applicable welding code requirements. The dominate deficient condition noted was that the welds were significantly undersized.

2

U.S. NUCLEAR REGULATORY COMMISSION