UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-445
TEXAS UTILITIES GENERATING COMPANY	)	50-446
et al.	)	
	)	
(Comanche Peak Steam Electric	)	
Station, Units 1 and 2)	)	

NRC STAFF'S PROVISIONAL PROPOSED FINDINGS OF FACT

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NRC STAFF'S PROVISIONAL PROPOSED FINDINGS OF FACT

I. BACKGROUND

1. These findings relate to an operating license application submitted by Texas Utilities Generating Company ("TUGCO"), as applicant and agent for the owners Dallas Power and Light Company; Texas Electric Service Company; Texas Power and Light Company; Texas Municipal Power Agency; Brazos Electric Power Cooperative, Inc., and Tex-La Electric Cooperative of Texas, Inc. ("Applicants").[1/] The application is for the operation of two pressurized water nuclear reactors, designated Comanche Peak Steam Electric Station, Units 1 and 2 ("CPSES"), in Somervell County, Texas. Notice of the Commission's receipt of the operating license application for CPSES was published in the Federal Register on May 12, 1978.[2/]

1/ The Board's findings generally refer to Texas Utilities Generating Company, *et al*. ("TUGCO") as "the Applicants", since they have applied for licenses to operate Comanche Peak. However, since TUGCO, *et al*. have construction permits for Comanche Peak, the Board may also refer to TUGCO, *et al*. as "the licensee." These terms may appear interchangeably in these findings.

2/ 43 Fed. Reg. 20583.

Construction Permits CPPR-126 and CPPR-127, issued December 19, 1974, authorized construction of CPSES.

2. Following the receipt of the application, a Notice of Opportunity for Hearing was published in the Federal Register.[3/] Citizens Association for Sound Energy ("CASE"), Citizens for Fair Utility Regulation ("CFUR"), and the Texas Association of Community Organizations for Reform Now/West Texas Legal Services ("ACORN") filed timely petitions to intervene and requests for hearing, in accordance with 10 C.F.R. Section 2.714. The State of Texas filed a timely petition to participate as an interested state, pursuant to 10 C.F.R. Section 2.715(c). An Atomic Safety and Licensing Board ("Board") was established[4/] to rule on these petitions and requests for hearing and to preside over the proceeding in the event that a hearing was necessary.[5/] After a prehearing conference on May 22, 1979, the Board admitted CASE, CFUR and ACORN as intervenors, and the State of Texas as an interested state. "Order Relative to Standing of Petitioners to Intervene," June 27, 1979. Twenty-five (25) contentions were originally admitted by the Board, along with three "Board Questions." "Order Subsequent to the Prehearing Conference of April 30, 1980," June 16, 1980.

3/ 44 Fed. Reg. 6995 (Feb. 5, 1979).

4/ 44 Fed. Reg. 15813 (March 15, 1979).

5/ The Atomic Safety and Licensing Board ("Board") appointed to consider this matter was reconstituted on July 3, 1979, 44 Fed. Reg. 40461 (July 10, 1979); August 1, 1980, 45 Fed. Reg. 53288 (August 11, 1980), and June 19, 1981, 46 Fed. Reg. 32717 (June 24, 1981).

3. The Board struck one contention[6/] and dismissed two contentions as a result of stipulations between the parties[7/] and five contentions as a result of withdrawal of the contention by the sponsoring intervenor.[8/] The Board granted summary disposition of two contentions.[9/] As a result of the withdrawal of Intervenor ACORN, the Board ultimately dismissed its contentions.[10/] Thus, only three contentions remained before the Board for the hearing:

> Financial Qualifications (Contention 25);
>
> Construction Quality Assurance ("QA") and Quality Control ("QC") (Contention 5), and
>
> Emergency Planning (Contention 22)

4. Evidentiary hearings were held on December 2-3, 1981;[11/] June 7-11, 1982; July 26-30, 1982, and September 13-17, 1982. A number of limited

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6/ "Memorandum and Order," LBP-81-22, 14 NRC 150, 154 (1981).

7/ "Order Subsequent to Prehearing Conference of December 1, 1981," December 18, 1981, at 2; "Order Dismissing Contentions 1, 4 and 6," January 25, 1982.

8/ "Order" August 21, 1981; "Memorandum and Order," October 23, 1981, at 6; "Order Dismissing Contentions 1, 4 and 6," January 25, 1982; "Order Following Conference Call," April 2, 1982.

9/ "Order Granting Summary Disposition of Contentions 2 and 7," March 5, 1982.

10/ "Memorandum and Order," July 24, 1981; "Order," January 12, 1982.

11/ Testimony regarding Contention 25, relating to financial qualifications, was presented at the December hearing session. In view of the Commission's rule eliminating financial qualifications review of electric utilities in licensing hearings for nuclear power reactors (47 Fed. Reg. 13750, March 31, 1982), the Board subsequently ordered that no further consideration would be given to Contention 25. "Order Following Conference Call," April 2, 1982, at 3.

appearance statements were received from members of the public during this time.

5. The decisional record of this proceeding consists of: (1) the Commission's Notice of Opportunity for Hearing; (2) the pleadings filed by the parties; (3) the transcripts of the hearing, and (4) the exhibits received into evidence.

II. BOARD QUESTIONS

6. In an operating license proceeding, the Board is authorized only to decide the issues in controversy among the parties (10 C.F.R. § 2.760a and Appendix A to 10 C.F.R. Part 2, Section VIII) and issues which the Board has determined raise serious safety, environmental, or common defense and security matters. The only matters in controversy between the parties remaining in the proceeding are those encompassed by Contentions 5 and 22. The Board did pose "Board Questions" which it requested the Applicants and Staff to address. However, in posing these questions, the Board did not determine that these matters constituted serious safety, environmental or common defense and security matters pursuant to 10 C.F.R. § 2.760a. Rather, the Board sought information to assist it in making such a determination if appropriate. The Board has addressed each of these questions below.

7. In the Board's "Order Subsequent to Prehearing Conference of April 30, 1980," June 16, 1980, the Board requested that the Applicants and the Staff address the following "Board Questions":

> Board Question 1
>
> Describe in detail the planned method of handling any hydrogen gas in the CPSES containment structure.

Board Question 2

Applicant and Staff should describe in detail the operating quality assurance program for CPSES. A description of the provisions for conduct of QA audits should be provided, including a description of how reactor operations and reactor operator training will be audited.

Board Question 3

Describe the status of resolution of Safety Issue TAP A-9 (ATWS) as it relates to CPSES 1 and 2.

8. By Order dated April 2, 1982, the Board requested that in light of information presented in CFUR's motion for voluntary withdrawal, that the NRC Staff, and the Applicants if they so desired, could present pertinent information regarding the deletion of the Boron Injection Tank ("BIT") for Comanche Peak. Specifically, the Board sought a description of the system which is to be deleted, the purpose of the system, its status with reference to the Comanche Peak reactors, the bases for its deletion, and the means by which its functions will be performed if there is not to be a boron injection tank.

A. Board Question 2

9. At the evidentiary hearing on December 2-3, 1981, the Applicants presented the following witnesses as a panel to address Board Question 2: R. A. Jones, Manager of Plant Operations (Jones, Tr. 527); Antonio Vega, Quality Assurance Services Supervisor (TUGCO), Quality Assurance Division (Vega, Tr. 531); David N. Chapman, TUGCO Manager of Quality Assurance (Chapman, Tr. 522); B. R. Clements, Vice President, Nuclear, TUGCO (Clements, Tr. 519), and David E. Deviney, Operations Quality Assurance

Supervisor at Comanche Peak.[12/] These witnesses were cross-examined by the Intervenors (Tr. 537-556) and were also questioned by the Board. Tr. 556-640.

10. The Staff presented as its witness John G. Spraul, a Senior Quality Assurance Engineer (Nuclear) in the (NRC) Office of Nuclear Reactor Regulation, Quality Assurance Branch (Spraul, Tr. 645). The Intervenors cross-examined Mr. Spraul (Tr. 648-656) and the Board also questioned him. Tr. 656-662.

11. The Board has determined that the Applicants and Staff fully addressed this question and that the information provided is sufficient to demonstrate that no "serious safety environmental, or common defense and security matter exists." Thus, the Board need make no finding on this question under 10 C.F.R. § 2.760a.

B. Board Questions 1 and 3

12. With respect to Board Questions 1 and 3, in the Board's April 2, 1982, Order, the Board indicated that the Applicants or the Staff may file information with the Board for its evaluation which answers these questions.

13. On April 19, 1982, the Applicants filed information on Board Question 1 in the form of the affidavit of Fred W. Madden, Jr. Board Exhibit 1B. On May 7, 1982, the NRC Staff filed a response to Applicants' Answer to Board Question 1, which was supported by the affidavits of David Shum and Robert Palla. Board Exhibits 1D and 1E.

12/ Mr. Deviney did not pre-file any written testimony, but was part of the panel testifying on this question (Tr. 506-644).

14. On May 7, 1982, the NRC Staff filed its answer to Board Question 3, which included the affidavits of David W. Pyatt, James W. Clifford, and Marvin W. Hodges. Board Exhibit 3.

15. Testimony regarding Board Questions 1 and 3 was not presented at the evidentiary hearing sessions since the Board determined that the information which the Applicants and the Staff supplied was "sufficient for the purposes for which the Board raised those questions." Tr. 693, 730-731. Thus, the Board determined, on the basis of this information, that no "serious safety, environmental, or common defense and security matter exists."

C. Deletion of the BIT

16. On May 7, 1982, the NRC Staff submitted its responses to the Board's questions regarding the deletion of the BIT, which included the affidavit of Sammy Diab. Board Exhibit 2. Mr. Diab also appeared as a witness at the June hearing session. Tr. 772-783.

17. The Applicants presented two witnesses to testify about the deletion of the BIT, Fred W. Madden, Jr. and Melita P. Osborne. Tr. 734-770.

18. The Board considers that the information provided by the Applicants and the Staff regarding deletion of the BIT to be comprehensive and responsive. Thus, the Board has determined, on the basis of that information, that no "serious safety, environmental or common defense and security matter exists."

III. FINDINGS OF FACT

A. Contention 5

19. Contention 5, as admitted by the Licensing Board in its "Order Subsequent to Prehearing Conference of April 30, 1980," June 16, 1980, states as follows:

> The Applicants' failure to adhere to the quality assurance/quality control provisions required by the construction permits for Comanche Peak, Units 1 and 2, and the requirements of Appendix B of 10 CFR Part 50, and the construction practices employed, specifically in regard to concrete work; mortar blocks; steel; fracture toughness testing; expansion joints; placement of the reactor vessel for Unit 2; welding; inspection and testing; materials used; craft labor qualifications and working conditions (as they may affect QA/QC) and training and organization of QA/QC personnel, have raised substantial questions as to the adequacy of the construction of the facility. As a result, the Commission cannot make the findings required by 10 CFR § 50.47(a) necessary for issuance of an operating license for Comanche Peak.

20. CASE did not initially present any witnesses in support of this contention. The testimony of the Applicants and the Staff originally addressed the concerns raised by CASE in support of this contention. Once the hearing began, CASE raised other concerns, either through documentary evidence or through the testimony of witnesses whom CASE presented at the hearing sessions. The Board will first address the issues originally in this contention and will then address the allegations CASE presented at the hearing, either through documentary evidence it presented or through its witnesses.

21. Applicants' initial testimony on Contention 5 was presented by Ronald G. Tolson, Construction Quality Assurance Supervisor, TUGCO;

David N. Chapman, Manager Quality Assurance, TUGCO (Chapman, Applicants' Exhibit 42 and attachments); Susan L. Spencer, Quality Assurance Auditor, TUGCO (Spencer, Applicants' Exhibit 44 and attachments); Roger F. Reedy, President, R. F. Reedy, Inc. and Partner, Reedy, Herbert, Gibbons & Associates (Reedy, Applicants' Exhibit 46 and attachments); Raymond J. Vurpillat, Power Group Quality Assurance Manager, Brown & Root, Inc. (Vurpillat, Applicants' Exhibit 45 and attachments); and Antonio Vega, Supervisor, Quality Assurance Services, TUGCO (Vega, Applicants' Exhibit 43 and attachments). (See also Tr. 1402-1710; 1738-1904; 1927-2145).

22. The Staff's initial testimony was presented by William A. Crossman, Chief, Reactor Project Section B, Reactor Project Branch I, NRC Region IV, Robert C. Stewart, Reactor Inspector, NRC Region IV, and Robert G. Taylor, Senior NRC Resident Inspector (Construction), Comanche Peak Steam Electric Station, Glen Rose, Texas. NRC Staff Exhibit 13 ("Crossman," "Stewart," "Taylor"). See also Tr. 1711-1738; 2312-2452. In addition to the testimony of these individuals, the Staff also introduced into evidence (Tr. 2336) all of the inspection and investigation reports cited in its initial testimony (NRC Staff Exhibits 13A-178).

23. During the period relevant to Comanche Peak construction and up until March 7, 1982, Mr. Crossman was responsible for the supervision of the project inspectors who inspected nuclear power plants under construction in Region IV, including Comanche Peak. (He is now responsible for the supervision of project inspectors who inspect other nuclear power

plants under construction in Region IV). Crossman (NRC Staff Exhibit 13), at 1-2.[13/] Mr. Stewart was the principal NRC Comanche Peak inspector from June 1974 to January 1978, during which time he had responsibility for coordinating all safety-related inspections of Comanche Peak construction. During that period of time, he performed or participated in numerous inspections of Comanche Peak construction. He also conducted special investigations concerning allegations of improprieties in Comanche Peak construction. Stewart, at 16-17. Mr. Taylor has been the resident reactor inspector since 1978. During that time, he has had responsibility for coordinating all safety-related inspection efforts by the NRC Region at the site. In addition, he maintains a field office, develops and recommends enforcement action, and acts as a liasion with regional, state and local agencies. Taylor, at 17.

24. Messrs. Crossman, Stewart and Taylor also presented supplemental testimony consisting of 1) errata and addenda correcting and updating their testimony pre-filed May 24, 1982 (incorporated as part of NRC Staff Exhibit 13), and 2) the findings contained in the Staff's annual assessment and analyses of the Applicants' performance ("Supplemental Testimony of William A. Crossman, Robert C. Stewart and Robert G. Taylor Regarding Annual Assessments of Applicants' Performance," NRC Staff Exhibit 180).

25. The adequacy of the Applicants' construction and the construction QA/QC program has primarily been determined by a review of the

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13/ This testimony, and most of the other witnesses' testimony, was not bound into the record. All testimony was given an exhibit number. Except for the testimony which was bound into the record with transcript page numbers (See Applicants' Exhibit 141, for example), references are to the page numbers of the testimony, rather than to transcript pages. Although the first reference to the testimony includes the exhibit number of the testimony, unless otherwise noted, subsequent references include only the name of the witness testifying and the page number of the testimony.

Staff's inspection and investigation program at Comanche Peak. An understanding of that program is a prerequisite to understand ; the adequacy of construction at Comanche Peak and the Applicants' QA/QC program during construction. Accordingly, the Board will first discuss that inspection and investigation program.

1. Role of NRC During Construction

26. Each NRC applicant is responsible for assuring that its nuclear power plants are built and operated safely and in conformance with the NRC regulations. An applicant also is required to assure that its suppliers meet the applicable NRC criteria. In this respect an applicant is responsible for functions such as product inspection and nondestructive testing of reactor components, structures, and systems even though it may, on occasion, delegate the actual performance of the activity to another organization. Crossman, at 3.

27. NRC looks to the power plant owners, the utilities themselves, to take the leadership role in assuring the quality of their plants and operations. This requires careful attention the selection of engineering specifications and Quality Assurance (QA) procedures and practices for each task and their implementation by the workers on the job. And, most importantly, there must be adequate resources of qualified personnel at management, operating, and staff levels. The NRC places the highest emphasis on the active involvement of top management in QA programs. The NRC evaluates these programs, an applicant executes them, and the NRC assesses performance. Crossman, at 3-4.

28. To meet the NRC's regulatory requirements, an applicant must develop and implement a pyramid control system which, at the bottom, assures, through detailed inspection and test programs, that all safety significant actions are properly done. These detailed verification programs require up to 100% inspection by an applicant's quality control personnel of a multitude of individual actions. These programs also provide the basis for accept/reject decisions on specific equipment, instrumentation, technician or operator actions, and procedures. Crossman, at 4.

29. Moving up the pyramid, an applicant must have a quality assurance program which includes audits to oversee and test the adequacy of the performance of the detailed quality control tests and inspections. These programs provide feedback to the lower level of this system in the form of specifying changes in training, modification of procedures, upgrading or improving testing methods or equipment, requalification methods, if required, and other programmatic improvements. This feedback assures and enhances the reliability of the program as a whole which, in turn, assures and verifies that all actions which are of safety significance have been, and will be properly carried out. Crossman, at 4-5.

30. At the top of the pyramid, an applicant's management must provide adequate organizational independence and manpower for its quality assurance and quality control programs and provide policy guidance to all elements of an applicant's organization in order to assure quality performance in all safety aspects of the contruction and operation of its nuclear facility. Crossman, at 5.

31. Another basic element of the NRC program is the defense-in-depth concept which requires multiple barriers and redundancy in equipment and operating options. This approach assures that, even if an item of equipment malfunctions or an incident of human error occurs, there will nevertheless be adequate protection of the public. Crossman, at 5.

32. The mission of NRC Region IV is to conduct inspections to assure that an applicant meets license and regulatory requirements as well as commitments in its Safety Analysis Reports. Region IV takes enforcement action where necessary, to obtain corrective action for specific or programmatic deficiencies. These Region IV activities interface with the Office of Nuclear Reactor Regulation (NRR) whose mission is to evaluate the adequacy of an applicant's proposals, specify license conditions, amendments, and technical specifications. Simply stated, NRR evaluates what an applicant proposes, commits to, or is required to do, whereas the NRC regional offices inspect to determine that an applicant does what it is required or committed to do and takes enforcement action, if needed. Crossman, at 5-6.

33. It is not the responsibility of the NRC regional offices to inspect each and every activity during the construction of a facility. The NRC regional offices are charged with providing assurance, through direct inspection, that an applicant's performance meets NRC's regulatory requirements and other commitments. Considering the extensive applicant control programs referred to above, the regional office inspection program may be viewed as the apex of the pyramid which provides overall assurance of adequate quality in the contruction and operation of nuclear facilities. The regional office inspection program is one of selective

auditi~g and not 100% verification of all phases of an applicant's program. This inspection of hardware, observation of testing, review of procedures, and all other inspection activities are not aimed at approval of individual components, actions, or procedures, but rather, at evaluating whether or not an applicant's management control system are working. Crossman, at 6-7.

34. Whenever deficiencies are identified, the NRC requires an applicant to take action to prevent recurrence as well as to correct the specific deficiencies. If the results of a single inspection, or a sequence of inspections, indicate a deterioration in the performance of an applicant's program in several areas, the NRC requires the applicant to examine its program in depth and upgrade the degree of control exercised at the highest level of the control system pyramid to assure that such deterioration is checked and the program as a whole returned to a satisfactory level of quality. Crossman, at 7.

2. NRC Inspection of QA/QC Program

35. The NRC conducts periodic scheduled and unannounced field inspections of an applicant's QA program implementation as well as those of its contractors and suppliers. These inspections start prior to docketing of the application and continue throughout the construction phase, the preoperational test program, and the operating lifetime of the facility. The NRC inspection program is carried out by region based inspectors. The NRC inspection program is not designed to duplicate an applicant's QA program, or to perform a redundant, independent review of every accept/reject determination. Rather, it is a regulatory program

aimed at determining, by spot checking and sampling, whether or not an applicant is in fact providing adequate assurance of quality in the construction and operation of its facility. The NRC audit or sampling program is not a statistical random sample. The specific areas reviewed in detail are selected from those considered to be the most important from a nuclear safety standpoint. By a specific spot checking and sampling review of QC actions the NRC can, therefore, test whether an applicant's QA program is really working. The NRC review of the overall program gives considerable confidence that its spot checking and sampling review provides an accurate assessment of an applicant's performance in meeting regulatory requirements. Crossman, at 7-8.

36. The NRC enforcement program complements the sampling or auditing inspection philosophy. By assuring that upgrading of an applicant's program results from specific noncompliance identified by the inspector, continued reliance can be placed on the validity of NRC's inspections which place heavy emphasis on an evaluation of an applicant's quality assurance program. Crossman, at 8.

37. Inspections conducted during the construction phase include: (1) reviewing an applicant's QA performance, including audits of an applicant's QA records and documentation; (2) witnessing the construction practices and inspection of the facility at various stages of construction; and (3) reviewing the qualifications and training of the construction personnel (where requirements exist) as well as those of the quality assurance and quality control (QA/QC) personnel. Inspection reports, resulting from the above inspections, document inspection findings (items of noncompliance and deviations), as well as unresolved

items. These items are entered in the Region IV tracking system and remain open until appropriately resolved. Crossman, at 9.

38. Inspection of the implementation of an applicant's quality assurance program is a key element in the determination of its adequacy. This inspection activity, still a nonrandom sample, involves checking whether actual work activities are in accordance with procedures, license requirements, technical specifications, plans, and code requirements. NRC inspectors question craftsmen and operators to determine if they understand, and are adhering to, applicable limits and requirements. The NRC inspectors observe operating instruments and recorder charts to determine that operations are being conducted within regulatory requirements. They observe instruments being calibrated. Observations are made as equipment is started up, shutdown, or otherwise changed in the operating mode. These observations and individual discussions with, and questions of, people actually doing the work provide a basis for determining how well an applicant is actually implementing its quality assurance program. Crossman, at 9-10.

39. The NRC's quality assurance requirements are contained in Appendix B to Part 50 of Title 10 of the Code of Federal Regulations, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." These criteria provide a basis upon which the NRC assesses the acceptability of QA programs. The criteria of Appendix B apply to all activities affecting safety-related functions of nuclear power plant structures, systems, and components. Crossman, at 10.

40. The NRC inspection reports assessing compliance with 10 C.F.R. Part 50, Appendix B, contain the terms "item of non-compliance," "devia-

tion." Crossman, at 10. NRC Staff witness Mr. Crossman explained the meaning of these terms. An item of noncompliance refers to an applicant's failure to comply with the various regulatory requirements of the NRC or the applicant's specifications. During the time covered by the Staff's testimony, such terms were categorized into three levels of severity: violations, infractions and deficiencies. When any level of severity was found a Notice of Violation was attached to the inspection report wherein the item was reported and the level of severity was set forth. Crossman, at 10-11.

41. A violation was the most severe item of noncompliance and was issued when the fabrication, construction, testing or operation of a Safety Related Category I system was such that the function or integrity of the system was lost. In contrast, an infraction was a less serious finding that a Safety Related Category I system was impaired, rather than lost. Crossman, at 11.

42. A deficiency was an item of noncompliance in which the threat to the health, safety, or interest of the public was remote; deficiencies included such items of noncompliance as failure to follow procedures, and posting or labeling requirements which are not serious enough to amount to infractions. Crossman, at 11.

43. In addition, at times an applicant would promise that certain controls or procedures would be implemented which were not required by the NRC. Nonetheless, when an applicant did not conform to its commitments to the NRC, even though such commitments were not regulatory requirements, such failure was cited and referred to as a deviation. Crossman, at 11.

44. The above classification system was revised in October 1980 to provide for six severity levels of violation. (See 45 Fed. Reg., No. 196, October 7, 1980). On March 9, 1982 the enforcement policy was revised to reduce the number of severity levels from six to five. (See 47 Fed. Reg., No. 46, March 9, 1982). Crossman, at 11-12.

3. Construction at Comanche Peak

45. TUGCO began construction at the Comanche Peak site under a Limited Work Authorization (LWA) on October 17, 1974. Further construction activities were commenced when construction permits were issued on December 19, 1974. Crossman, at 13.

46. Brown & Root Co., Inc. (B&R) has constructor responsibilities and has American Society of Mechanical Engineers ("ASME") certification. Crossman, at 13.

47. Westinghouse Electric Corporation (W) is Nuclear Steam Supply Systems (NSSS) supplier for the two four-loop pressurized water reactors rated at 1150 MWe each. Crossman, at 14.

48. The architect-engineer (AE) with responsibility for design and engineering of the facility is Gibbs and Hill, Inc. Crossman, at 14.

49. Freeze and Nichols (F&N) and Mason-Johnson Associates (MJA) were responsible for design, inspection, and testing of the Safe Shutdown Impoundment Dam. Specifically, F&N supplied the design and construction specifications and performed oversight of the construction. MJA provided geotechnical and inspection functions. Crossman, at 14.

4. QA/QC Organization at Comanche Peak

50. Applicants have established a QA/QC program during the construction phase at Comanche Peak which assigns QA/QC functions among Texas Utilities Generating Company/Texas Utilities Services, ("TUGCO/TUSI"), Gibbs & Hill, Inc., the Architect-Engineer, Brown & Root, Inc., the Construction Manager/Constructor, and Westinghouse Electric Corporation, the nuclear steam supply system supplier. Chapman (Applicants' Exhibit 42), at 2-5.

51. TUGCO, as the lead applicant, has ultimate responsibility for quality assurance activities at Comanche Peak. TUSI is designated by TUGCO to have the authority to conduct the required support activities for implementation of the QA program at the site. Chapman, at 2-3; Vega (Applicants' Exhibit 43), at 2-3.

52. Interfaces among participating organizations in the QA program have been established for QA functions at Comanche Peak. Chapman, at 9-10.

5. Satisfaction of 10 C.F.R. Part 50, Appendix B Criteria

53. The QA program at Comanche Peak establishes procedures and requirements to address each of the criteria set forth in 10 C.F.R. Part 50, Appendix B. Vega, at 2.

54. The CPSES QA program is described in the CPSES Quality Assurance Plan (Applicants' Exhibit 43A) and the Final Safety Analysis Report ("FSAR"), admitted as Applicants' Exhibit 3 in this proceeding. Vega, at 2.

55. Issuance of construction permits to TUGCO for the construction of CPSES was contingent on development of a satisfactory QA program.

Subsequent to issuance of the permits and during construction, implementation of the program has been monitored by the NRC by review of further procedure development and observation of work activities. Crossman, et al., at 14.

56. During the first 6 months of 1978, TUGCO made a series of organizational changes which were considered to have a positive impact on the CPSES QA program. Chapman, at 3; Tr. 1807-1808; Crossman, at 13. At the same time, Texas Utilities Services, Inc. ("TUSI") implemented comparable changes to exercise greater control over engineering and construction. The active management of the B&R QA/QC staff was assumed by TUGCO except for that work accomplished under the American Society of Mechanical Engineers ("ASME") code certification program. This reduced B&R headquarters QA involvement in on-site activities. Also, at that time, TUGCO assumed management of the on-site soils and concrete testing laboratory, formerly under a subcontract. A revised program description for QA during design and construction of CPSES was submitted to the NRC on September 22, 1978, and accepted by NRR on November 3, 1978. Region IV's audit of the CPSES QA program was completed in December 1978 and the results are contained in NRC Inspection Report 78-23 (NRC Staff Exhibit 14). Crossman, et al., at 13-14.

6. Results of the NRC QA/QC Inspection Program at Comanche Peak

57. The NRC inspection program at Comanche Peak has identified items of noncompliance, deviations from commitments, and weaknesses in the QA/QC program, but in each instance the Applicants have responded properly and taken adequate corrective action. Crossman, at 12.

58. Based on experience gained in the implementation of the QA program, as monitored by the NRC, TUGCO revised certain aspects of their QA program in early 1978. These revisions, in the form of organizational changes (see paragraph 55), have essentially been in effect since mid-1978. Work accomplished prior to that time included only early civil construction activities and was concluded to have been performed satisfactorily. Subsequently, the major construction phases, such as piping, electrical, etc., have been performed utilizing the improved organization. Since 1978, the Applicants have initiated minor program revisions which have strengthened quality control. Crossman, at 15.

59. During December 1978, NRC regional inspectors performed an in-depth QA/QC inspection approximately mid-term through the construction period. No items of noncompliance or deviations were identified as a result of the inspection, which was documented in NRC Inspection Report 78-23 (Staff Exhibit 14). Crossman, at 12.

60. The TUGCO QA program has, to date, been determined to comply with NRC QA criteria established in 10 C.F.R. Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Chapman, at 11; Crossman, at 15. The TUGCO QA program is interpreted to include their prime contractors, subcontractors, and vendors. Crossman, at 15.

61. In most cases, the Applicants' own QA program has identified and corrected significant construction deficiencies within the context of 10 C.F.R. § 50.55(e). Crossman, at 15.

62. Additional inspections of the Applicants' QA/QC program by the NRC are planned. The NRC will ensure that safety matters identified during

the past and subsequent inspections will be adequately resolved prior to authorization for the Applicants to load fuel or operate CPSES. Crossman, at 15-16.

63. Systematic assessments of licensee performance ("SALP") on an annual basis have also determined that the Applicants have adequately developed and implemented a QA/QC program. Crossman, at 12.

64. Starting in 1980, the NRC Staff initiated a "Systematic Assessment of Licensee Performance" or "SALP" program. The SALP program was implemented in accordance with the commitments of the "Action Plan for Implementing Recommendations of the President's Commission and Other Studies of the TMI-2 Accident." The SALP is an integrated NRC Staff effort to collect available observations and data on an annual basis and to evaluate licensee performance utilizing such data and observations. The integrated systematic assessment is intended to be sufficiently diagnostic to provide a rational basis for allocating NRC resources and to provide meaningful guidance to licensee management. Crossman Supplemental Testimony (NRC Staff Exhibit 180), at 1-2.

65. The first "SALP" for Comanche Peak was conducted in 1980, covering the period August 1, 1979 through July 31, 1980. That SALP involved evaluation of the licensee's performance during the apparisal period by a SALP "Review Board" consisting of Region IV management and inspectors as well as the NRC Office of Nuclear Reactor Regulation ("NRR") Project Manager for Comanche Peak. The SALP evaluated the licensee's performance in the following areas:

- Enforcement history
- Status and Summary of Noncompliance Items

- Construction Deficiency Reports
- Escalated Enforcement Action
- Licensee's Responsiveness and Ability to Take Meaningful Corrective Action
- Effectiveness and Attitudes of Licensee Personnel in Complying with NRC Requirements

Crossman Supplemental Testimony, at 2.

66. The SALP evaluation of licensee performance was discussed at a meeting with the TUGCO corporate management. Inspection Report 80-25, November 12, 1980 (NRC Staff Exhibit 181) documents that meeting and contains a "Licensee Performance Evaluation." As indicated in Inspection Report 80-25, that SALP concluded that the licensee's performance was generally acceptable although continued improvement in certain areas would be desirable. On the basis of the evaluation, Region IV did not see a need to make any adjustments in the NRC inspection programs related to Comanche Peak. Crossman Supplemental Testimony, at 2-3.

67. The next SALP for Comanche Peak covered the period July 1, 1980 through June 30, 1981 and is documented in Inspection Report 81-20, April 27, 1982 (NRC Staff Exhibit 15). That report is the product of the revised NRC policy for the conduct of the SALP program. That SALP evaluated the performance of the licensee in the following functional areas:

- Piping Systems and Supports
- Safety-Related Components
- Support Systems
- Electrical Power Supply and Distribution

- Instrumentation and Control Systems
- Licensing Activities

Crossman Supplemental Testimony, at 3.

68. The SALP documented in Inspection Report 81-20 concluded that the licensee demonstrated an overall combination of attributes exhibiting Category I performance (the highest rating) during the appraisal period. This evaluation was based upon the three primary areas where the construction efforts, and therefore, the NRC inspection effort, were directed; i.e., piping, electrical, and instrumentation installations. Crossman, at 12; Crossman Supplemental Testimony, at 4.

69. Starting in 1977 (prior to the implementation of the SALP program in 1980), the NRC Region IV inspectors prepared documents entitled "trend analyses" for each of their assigned licensees, including Comanche Peak. The "trend analyses" were prepared at the request of Mr. William C. Seidle, (then) Chief, Reactor Construction and Engineering Support Branch (Region IV), and Mr. Crossman, Chief, Projects Section. Crossman Supplemental Testimony, at 5.

70. The analyses were prepared to provide Mr. Seidle with information about licensee performance which would assist him in allocating the inspection resources available to him to achieve the greatest effectiveness. The analyses were prepared informally, were not transmitted to or otherwise made known to the Applicants, and were not made part of the official docket files. Crossman Supplemental Testimony, at 6.

71. The specific areas considered in these analyses include the following:

- Number and repetitiveness of Construction Deficiency Reports.
- Enforcement history, e.g., number and repetitiveness of non-compliance items.
- Responsiveness of licensee to enforcement action.
- Number of outstanding unresolved items - timeliness of resolution.
- Corporate management involvement in regulatory matters.
- Effectiveness of QA/QC programs.
- Any other trends indicative of poor performance.

Crossman Supplemental Testimony, at 5.

72. The trend analyses for Comanche Peak cover the years 1976, 1977, 1978 and 1979. (NRC Staff Exhibits 182-195). Since Mr. Stewart was the Comanche Peak Project Inspector during the years 1976 and 1977, he analyzed Comanche Peak performance for 1976 and 1977. Mr. Taylor, who became the Comanche Peak Resident Reactor Inspector in 1978, did the analyses for Comanche Peak for 1978 and 1979. Crossman Supplemental Testimony, at 5.

73. These "trend analyses" were not official NRC inspections or audits. All official activities related to a docket are documented in an inspection report and are made publicly available. The term "audit" does not appear within Staff procedures as a Staff activity where docket work is involved. Although the concept of auditing as used in the nuclear industry and the Staff's concept of inspection are not significantly different, the term "inspection" is consistently used by the Staff. Also, inspection reports must be factual to the maximum extent and devoid of opinion as completely as possible. In contrast, the analyses

performed in response to Mr. Seidle's request consisted of both factual information, such as the number of Construction Deficiency Reports, and the opinions of the inspector on such items as repetitiveness, responsiveness, management involvement and effectiveness. The analyses did not actually contain trends, but rather an assessment of the Applicants' performance for the preceeding year. Those statements that might appear to be trends were, in reality, simply a sharpening of the inspector's perceptions of what usually were situations or conditions that had existed previously but were perhaps not apparent to him earlier. Crossman Supplemental Testimony, at 5-6.

74. The formal assessments of Applicants' performance (the "SALP" reports) as well as perceptions contained in the analyses made before the formal assessments began, show a positive change in the Applicants' performance during this entire period. At the beginning of the analysis period, there was some question as to experience level of QA personnel (Taylor, Tr. 1730) and as to whether B&R was exercising domineering and overpowering control over both TUGCO QA and B&R QA/QC personnel. Crossman, Stewart, Taylor Supplemental Testimony, at 23. However, changes were made by the Applicants, the most important of which was the significant organization and management change initiated January 3, 1978 by which TUGCO assumed active management of the Brown and Root ("B&R") QA/QC staff (except for work accomplished under the ASME code certification program). While some later weaknesses, as discussed herein, were noted, there was progressive improvement in the effectiveness of the QA/QC program as construction advanced. Crossman, Stewart, Taylor Supplemental Testimony, at 23.

7. NRC Staff Inspection and Investigative Findings Regarding Construction Activities Spcified In Contention 5

75. In addition to these overall inspection findings, the Staff presented testimony regarding its inspection and investigative findings with respect to the construction activities specified in Contention 5, viz, concrete work; mortar blocks; steel; fracture toughness testing; expansion joints; placement of the reactor vessel for Unit 2; welding; inspection and testing; materials used; craft labor qualifications and working conditions (as they may affect QA/QC) and training and organization of QA/QC personnel. Crossman, Stewart, Taylor (NRC Staff Exhibit 13), at 18-103, as corrected and up-dated in "Errata and Addenda To NRC Staff's Direct Testimony Pre-Filed on May 24, 1982." Also see NRC Staff Exhibits 13A-178 (inspection and investigative reports relating to these activities). The Board's findings with respect to all of these activities are set forth below.

(a) Concrete

76. The NRC inspections of concrete construction activities involved both the direct work and the Quality Assurance/Quality Control (QA/QC) tests and inspections, including the installation of the reinforcing steel; the mechanical splicing of the reinforcing steel into continuous lengths (usually by the proprietary process called "Cadweld"); testing of aggregates and cement; operation of the concrete batch plant and transportation of the fresh concrete; the testing of the fresh concrete; the formwork before and during placement; the actual placement and consolidation of the concrete; and finally, the curing of the newly placed concrete. Stewart, Taylor (NRC Staff Exhibit 13), at 18.

77. Such inspections also included materials that are embedded in the concrete such as steel structures for subsequent attachments and electrical conduits. From early 1975 to date, there have been in excess of 45 inspections of these activities covering both Units 1 and 2. There was no set frequency for the inspections, but every effort was made to arrange the inspections to occur at the times when major activities were in progress. Stewart, Taylor, at 18-19.

78. Approximately 75% of the routine inspections resulted in findings that the Applicants and their contractors had complied with their commitments to fulfill the requirements of the Safety Analysis Report and Appendix B of 10 C.F.R. Part 50. The other 25% of the routine inspections revealed either items of noncompliance or deviations. As might be expected, five instances occurred in 1975 when the concrete work was just getting underway. During 1976, only two such instances were identified and, in 1977, none. With the advent of the resident inspector at CPSES, the instances of noncompliance or deviation rose to three, which probably reflects the increased time that was devoted to direct observation of the work. In 1979 and 1980, there was one instance of noncompliance or deviation each year. Stewart, Taylor at 19.

79. The NRC negative inspection findings regarding concrete work, as summarized by the testimony of the Region IV inspectors (Stewart and Taylor, at 19-24) and as set forth in the NRC Inspection Reports introduced into evidence (CASE Exhibit 15; NRC Staff Exhibits 11, 20, 24, 28, 31, 35, 41, 44, 48, 49, 53 and 54) reveal certain items of noncompliance or deviations. However, with respect to each of these negative findings, the Applicants made commitments to take corrective actions. Stewart and Taylor, at 24.

80. These commitments were reviewed by Region IV management and by the cognizant inspector to assure that the corrective action as stated would be effective and of a nature that could be verified. Id. The NRC Region IV witnesses summarized these commitments by the Applicants (Stewart and Taylor, at 24-28), which are also set forth in the NRC inspection reports and letters and in the Applicants' letters to NRC introduced into evidence (CASE Exhibit 15; NRC Staff Exhibits 10, 17, 18, 19, 21, 22, 25, 26, 29, 30, 32, 33, 36, 37, 42, 43, 45, 46, 49, 50, 51, 55, and 56 and Applicants' Exhibit 26).

81. The NRC Staff conducted follow-up inspections to determine whether the Applicants in fact implemented the corrective action they represented they would undertake. Stewart and Taylor, at 28-29; NRC inspection reports introduced into evidence (NRC Staff Exhibits 11, 12, 23, 24, 27, 34, 38, 47, 52, 57, and 58, and Applicants' Exhibit 37).

82. The NRC inspectors concluded that in view of the corrective actions taken by the Applicants as a result of these findings, as verified by the NRC Staff, these inspection findings do not raise substantial questions as to the adequacy of the construction of the facility. Stewart and Taylor, at 33.

83. In addition to inspections of construction activities involving concrete, investigations were also undertaken by Region IV of allegations of improprieties involving concrete work. Stewart and Taylor, at 34-40; NRC Staff Exhibits 35, 59, 63, 64 and 72. In many instances, the investigations did not result in any evidence which could substantiate the allegation. (e.g., Stewart and Taylor, at 36). Some of the allegations were substantiated (e.g., Stewart and Taylor, at 34) and certain items of

non-compliance were identified during the course of such investigations (e.g., Stewart, at 34). The Applicants responded to negative findings resulting from these investigations. Stewart and Taylor, at 36, 39; NRC Staff Exhibits 60, 65 and 66. The NRC Staff accepted these responses, subject to verification. Stewart and Taylor, at 36, 40; NRC Staff Exhibits 61, 67, and 68. The NRC Staff conducted followup inspections to determine whether the Applicants, in fact, implemented the corrective actions they represented they would undertake as a result of any adverse investigative findings. Stewart and Taylor, at 41-42; NRC Staff Exhibits 62, 69, 70 and 71.

(b) Mortar Blocks

84. The NRC inspection program for construction does not include an inspection of mortar blocks except as they relate to testing methods for cement and the water used in concrete. The NRC inspectors were not completely familiar with the term "mortar block construction" but assumed that it might be a reference to the use of precast concrete block construction sometimes referred to as "cinder block" construction. Such construction is frequently used in some parts of the country in home construction and is used in nuclear plants, on occasion, to build walls of a somewhat temporary nature such as personnel shields covering entrances into potential radiation areas and as divider walls. Generally, they are not designed or used as structural members and carry little or no safety loads. NRC Staff witness Mr. Taylor, the resident inspector (construction) for Comanche Peak, did some research on this matter as it might concern Comanche Peak. Stewart and Taylor, at 42-43; Taylor, at 43.

85. Items 33 and 34 appearing on page 25 of CASE's answers to the NRC's Fourth Set of Interrogatories refer to mortar blocks in the walls of the control room. Mr. Taylor reviewed the construction drawings for the control room area to determine if there were any precast concrete blocks used and if so, the quality requirements imposed by the designer. He found that for several walls, the engineer had specified block walls as divider walls that in effect make small rooms out of a large room. The large room is adjacent to the main control room and its wall are of reinforced concrete construction to seismic Category I standards. Failure of the block walls within the large room would not jeopardize any of the equipment or personnel within the control room. The walls were not considered by the engineer to have any effect on safety and therefore, no quality assurance requirements were imposed on their construction. Based on examination of the drawings and the rooms involved in the matter, Mr. Taylor stated his belief that the engineer's action complied with NRC requirements. Taylor, at 43-44.

(c) Steel

86. Steel, along with concrete, is the most commonly used material of construction in a nuclear power plant. The reinforcing steel used in concrete construction is considered during the inspection of concrete work. Steel is used in piping systems and the support thereof and is considered during inspection of these elements. Likewise, a vast majority of the mechanical components are made of steel and that is considered during the inspection of the component and the related documentation. In each case, the material quality is just one relatively small phase

of an overall inspection effort for a given category. It is thus nearly impossible to estimate the number of inspections that have involved steel. In reviewing the CASE answers to the NRC Staff's Fourth Set of Interrogatories, the instances cited that were categorized as dealing with steel as a material actually deal with the installation of components made of steel. Three of the instances which relate to the installation of concrete embedments have been discussed in regard to the inspection and investigative findings involving concrete. Another instance, i.e., the difference between the actual method used to install supports for the reactor coolant pumps and the steam generators and the method depicted in the Preliminary Safety Analysis Report ("PSAR"), is a fairly common occurrence since the PSAR usually is conceptual in nature (this is one of the reasons why there is also a Final Safety Analysis Report or "FSAR"). The remaining specific instance cited in support of Contention 5 dealt with a finding in 1975 that B&R had not fulfilled a procedural commitment to independently test reinforcing steel but rather accepted the supplier's test results. The B&R procedural commitment was in excess of that required by the NRC or by the engineering specifications for the projects and was subsequently deleted from the procedure. The item was resolved to the Staff's satisfaction as indicated by NRC Inspection Report 50-445/76-03. (NRC Staff Exhibit 27). Taylor, at 44-45.

87. The NRC inspectors were not aware of any inspection findings that would indicate a failure by the Applicants to adhere to their commitments to the NRC. Taylor, at 45-46.

(d) Fracture Toughness Testing

88. Fracture toughness is a characteristic of steel materials and generally relates to a determination as to the temperature at which the mode of failure of the steel to withstand an impact changes from brittle to ductile and the impact energy required to cause failure. All steel materials used in nuclear plants do not require such testing, only those that have a likelihood of having to withstand impact loadings. This is an engineering evaluation. The Staff inspects for compliance with this requirement as an element of the Staff's evaluation of the steel which in turn is inspected as discussed in regard to "steel." Taylor, at 46.

89. Referring again to the CASE answers to the NRC Staff's Fourth Set of Interrogatories, it appears that one of the original intervenors in this proceeding was of the opinion an equipment supplier should not perform the tests for fracture toughness of the materials in the reactor vessels, steam generators or the pressurizer. As a general rule, these tests are actually performed initially by the steel mill supplying the steel to the component fabricator since this steel characteristic is affected by the composition of the steel, the melting practices involved, and by the heat treatments given the steel during and after processing into its final form, in this case, plate steel. The steel mill is by far the most qualified party to perform the tests. The test is again repeated when the weld procedure for joining the plates is qualified. Generally, this test would be performed on materials that are comparable but not identical to the materials used in fabrication. NRC inspections have not identified any specific deficiencies in the Applicants' QA/QC program for fracture toughness testing. Taylor, at 46-47.

(e) Reactor Vessel for Unit 2

90. Although not an inspection specifically, NRC Inspection Report 50-445/79-03 (NRC Staff Exhibit 49) discussed the initial discovery by the Applicants that the supports and the periphery structure around the Unit 2 reactor vessel were rotated approximately 45° from a usable position. The Unit 2 structure was mirror imaged from Unit 1. Mirror imaging is a common engineering method of developing a second design from an original design provided that everything is completely symmetrical. (If one component is not symmetrical, mirror imaging will not work). In this case, the Unit 1 and 2 reactor vessels are identical units but need to be placed in position 180° in rotation. Following the determination that the physical structure (the supports and the periphery structure around the Unit 2 reactor vessel) could be rebuilt in a relatively short time, the Applicants' engineer devised the method which was coordinated with the design review personnel of the NRC. A comprehensive inspection of the rework effort was documented in NRC Inspection Report 50-446/79-07 (NRC Staff Exhibit 73). There were several short follow-up inspections by this inspector, culminating in NRC Inspection Report 50-446/79-13 (NRC Staff Exhibit 74). The entire reconstruction effort including the engineering aspects took approximately three (3) months, ending with the placement of the vessel on its supports in June 1979. Taylor, at 47-48.

91. It was not shown that any specific failure by design QA or with construction QA/QC had occurred. From a design standpoint, the misdesign had been through the entire design review process including interface review by the nuclear steam supply system (NSSS) vendor. The facility area had been constructed in accordance with the original design even

though that design was incorrect. It appeared that a very basic drafting concept error had been made early in the design effort for Unit 2 and was not detected in subsequent reviews, perhaps because of its very simplicity. Taylor, at 48-49.

92. The matter was reported to the NRC resident inspector, Mr. Taylor, and through him to other components of the NRC, but was not the subject of a 10 C.F.R. § 50.55(e) report by the Applicants. That provision of the regulations requires notification of deficiencies in design and construction, which if uncorrected, could have adversely affected the safety of operations of the nuclear power plant. In this case there could not have been a safety problem with the reactor because, with the vessel rotated from its correct position, the reactor could not have been assembled into a system. Taylor, at 49.

93. Based upon the actions taken by the Applicants in this matter, as reviewed for design by personnel of the Office of Nuclear Reactor Regulation and verified by the personnel of Region IV, there is nothing that would indicate that the as constructed placement of the reactor vessel in Unit 2 is less than satisfactory or unsafe. Taylor, at 50.

(f) Welding

94. The NRC reactor construction inspection program included inspections of all safety-related welding. Such welding included welding of reinforcing steel used in concrete work; the welding of the seams in the containment building liners; the welding of piping system components; and the welding of structural steel into component supports and restraints. The inspections included all aspects of the

welding, from the materials involved to the finished weld. Also included was the qualification testing of welders; the qualification of nondestructive testing personnel and the application and results of nondestructive testing activities. The inspections varied somewhat in depth, mostly depending on the importance of the welding activity. There was no set frequency for the inspections but every effort was and is made to arrange to have the inspections conducted to coincide with the early phase of a new welding activity and to have additional inspections of each activity as the work progresses. During the construction of Comanche Peak, the Staff has conducted some sixty-five (65) inspections dealing with various aspects of the Applicants' program for quality assurance/quality control of welding. Stewart and Taylor, at 50.

95. Approximately 83% of the routine inspections resulted in findings that the Applicants and their contractor had complied with their commitments to fulfill the requirements of the Safety Analysis Report and Appendix B of 10 C.F.R. Part 50. Approximately 17% of the routine inspections revealed either items of noncompliance or deviations. Although B&R as the principal contractor at Comanche Peak is directly responsible for the vast bulk of the welding activities, other contractors have also been involved in welding, both on-site and off-site. Such contractors include Chicago Bridge & Iron Company for the containment liners and for certain off-site fabricated welded steel components and ITT-Grinnell for performing off-site fabrication of pipe assemblies. Stewart and Taylor, at 51.

96. The Brown and Root (B&R) scope of work is such that several different craft groups were involved in welding such as ironworkers, millwrights, and pipefitters. The QA/QC scheme within B&R for control

of welding is essentially the same regardless of the craft type, but the specific application has varied depending on the welding process involved and also with the construction activity at that time. The present system of controls has evolved from a relatively complex method of documentation to a less complex but more effective system. Stewart and Taylor, at 51-52.

97. There are two primary code systems dealing with welding. The American Welding Society (AWS) has a code widely used throughout industry and in the nuclear field for structural welding applications. The American Society of Mechanical Engineers (ASME) also has a code system for welding that is widely applied in the manufacture and erection of pressure retaining systems such as boilers for heating and power generation. The parts of the ASME code most used in the nuclear power industry are Sections III, V, and IX. The differences between the two codes in regard to control of welding processes and the qualification standards for welders are relatively insignificant. There is, however, considerable difference in the two codes with respect to the criteria for acceptance of a given weld, particularly since the ASME Section III Code has several subdivisions, each with its own inspection and acceptance requirements. Thus, a QA/QC program for control of welding is necessarily complex because of such a variables. Stewart and Taylor, at 52.

98. The negative NRC inspection findings regarding welding, as summarized by the NRC inspectors (Stewart and Taylor, at 52-55) and in the NRC inspection reports introduced into evidence (NRC Staff Exhibits 31, 41, 69, 70, 75, 81, 86, 93, 102, 103, 104, 107, 112, 112, 114, 118, and 118C) reveal items of non-compliance and deviations.

99. However, as described in the testimony of NRC inspectors and in the NRC inspection reports and letters and in Applicants' letters to the NRC introduced into evidence, the Applicants made commitments to take corrective actions with regard to these negative findings. Taylor and Stewart, at 55-60; NRC Staff Exhibits 32, 33, 41, 69, 70, 75, 76, 77, 78, 81, 82, 83, 84, 87, 88, 89, 90, 91, 94, 95, 96, 97, 98, 99, 100, 101, 102, 103, 104, 105, 106, 108, 109, 110, 111, 115, 116, and 118c. These commitments were reviewed by Region IV management and by the NRC inspector involved to assure that the corrective action as stated would be effective and of a nature that could be verified. Taylor and Stewart, at 55.

100. Moreover, the NRC inspectors conducted follow-up inspections to determine whether the Applicants in fact implemented the corrective actions they represented they would undertake. Stewart and Taylor, at 60. The follow-up inspections were described briefly by the NRC inspectors (Stewart and Taylor, at 60-63) and are set forth in NRC Staff Exhibits 34, 38, 41, 58, 80, 92, 102, 103, 107, 112, and 117. The NRC inspectors concluded that in view of the corrective actions taken by the Applicants as a result of the NRC Staff findings, which have been verified by the Staff, these findings do not raise substantial questions as to the adequacy of the construction of the facility. Stewart and Taylor, at 63.

101. In addition to routine inspections, the NRC Staff also conducted investigations of allegations of improprieties in construction activities involving welding. Taylor, at 64; NRC Staff Exhibits 119, 120, 121, 122, 123, 124 and 125. Although some allegations investigated were substantiated (e.g., Taylor, at 64) only one negative finding (a Notice of Violation) resulted from these investigations. Taylor, at 67.

Many of these allegations that were substantiated had no technical merit or safety significance. e.g., Taylor, at 64, 75. The Applicants responded to this Notice of Violation by committing to take corrective actions which were subsequently verified by the NRC Staff. Taylor, at 68; NRC Staff Exhibit 114.

(g) Expansion Joints

102. Inspection of expansion joints is included as a part of the normal inspection of concrete placement. There have been no negative inspection findings with regard to expansion joints. According to NRC inspectors, the Applicants have not failed to adhere to the quality assurance/quality control provisions referenced by the construction permits for Comanche Peak or the requirements of Appendix B to 10 C.F.R. Part 50 in regard to expansion joints. Stewart, at 69.

103. As a result of an investigation of an alleged impropriety in construction activities involving an expansion joint (Stewart, at 70; NRC Staff Exhibit 128), the Applicants initiated an inspection and documentation program to assure that the required gap between Category I seismic structures was being maintained in the as-built condition. Stewart, at 70. The NRC Staff conducted a follow-up review of the Applicants' inspection and documentation program which was initiated to assure that all temporarily installed rotofoam blocks had been removed from between Category I seismic structures after concrete placement. Stewart and Taylor, at 71; NRC Staff Exhibit 80. The NRC inspector found that the procedural requirements were being implemented and considered this item closed. Id.

(h) Inspection and Testing

104. The essential element of any quality control program is the performance of inspection or testing, or both, for the purposes of determining whether a function or a product satisfies the requirements established for the function or product. Nearly all of the criteria of Appendix B of 10 C.F.R. Part 50 relate in some way to a function or product inspection to verify performance. Therefore, nearly any NRC inspection relates in some way to inspection and testing. NRC inspections during construction are classified according to general activity areas such as civil or structural (concrete) type construction activities and are carried out to determine conformance with the requirements in the regulations and/or committed to in the Safety Analysis Reports. The same is true for the broad activity areas generally designated as mechanical and electrical. Thus, it can be seen that an observation that a test of fresh concrete was not properly performed could readily be assigned to the area of "inspection and testing" whereas NRC inspectors have assigned that type of finding to "concrete" because that was the activity inspected. Stewart and Taylor, at 71-72.

105. Seven of NRC inspection findings have been assigned this aspect of Contention 5. These findings do not relate to more specific categories or welding, but relate to the mechanical and electrical areas of the NRC Staff inspection program. Stewart and Taylor, at 72.

106. The negative NRC inspection findings in the category of "inspection and testing" were summarized by the NRC inspector (Taylor, at 73-74) and are set forth in NRC inspection reports introduced by the Staff (NRC Staff Exhibits 52, 58, 129, 132, 140, 145, 146, and 148B).

With the exception of Notice of Violation 82-11 (NRC Staff Exhibit 148B), the Applicants made commitments to the corrective actions as a result of these negative inspection findings. (At the time of the Staff's testimony, the time period for Applicants' to respond to that Notice had not expired.) Taylor, at 74.

107. These corrective actions were reviewed by the Region IV management and the cognizant inspector for probable effectiveness and if acceptable, were acknowledged by Region IV by letter. Id. The NRC Staff inspectors summarized these corrective actions (Taylor, at 74-77), which are also set forth in NRC Staff Exhibits 130, 131, 133, 134, 135, 136, 137, 138, 141, 142, 143, 147, and 148.

108. The NRC also conducted follow-up inspections, as described in the Staff testimony (Taylor, at 77-79; Staff Exhibits 41, 52, 58, 70, 107, 139, 148A), to determine whether the Applicants in fact implemented the corrective actions they represented they would undertake. Taylor, at 77. The NRC inspectors concluded that considering the nature of these inspection findings and the Applicants' corrective actions, these inspection findings do not raise substantial questions as to the adequacy of construction.

109. In addition to the routine inspections described above, the NRC Staff conducted an investigation of allegations that might be assignable to this aspect of Contention 5. The investigation was documented in NRC Inspection Report 81-04 (NRC Staff Exhibit 149) and involved allegations that electrical inspectors were directed by their supervision not to follow procedures; that electrical inspectors had not performed required inspections; that certain electrical drawings utilized for inspections were, in some cases, obsolete. Taylor, at 80. None of the allegations was substantiated and no enforcement action was taken. Id., at 80-81.

(i) Materials Used

110. The materials used in any phase of construction which become part of the permanent installation are inspected as an element of the overall specific area inspected. As an example, when an inspection of concrete activities is made, the constituent materials of concrete such as cement, coarse and fine aggregates, and the water will be inspected via review of test records generated in accordance with the specifications. Whenever possible the inspectors will observe this type of testing to establish the credibility of the testing organization. None of the adverse inspection findings in regard to construction at Comanche Peak indicates any instance where materials as such were unsatisfactory. Stewart and Taylor, at 81.

(j) Craft Labor Qualifications and Working Conditions

111. In general, there are no NRC regulatory requirements or policy concerning the qualifications of the craft labor force or the conditions which may be imposed on the labor force during construction. The NRC has, however, established or adopted qualification standards for welders and for persons performing reinforcing steel splicing, usually referred to as Cadwelding. The NRC requires that an applicant commit to constructing pressure retaining components under one of the appropriate parts of the ASME Code for Boilers and Pressure Vessels. The NRC also requires that an applicant commit to use the American Institute for Steel Construction (AISC) Code for most other applications of steel construction where the ASME code is not applicable. Both codes contain direct or referenced requirements for the testing of welders prior to production work. The

ASME code for concrete reactor containment buildings contains a requirement for the qualification testing of Cadweld splicers. NRC Staff inspections covered implementation by Applicants of these requirements concerning the qualification of welders and Cadwelders. Taylor, at 82.

112. There have been no findings that would indicate that welders have not been qualified as required by the applicable code. One instance was identified where Cadweld splicer helpers were not fully tested and qualified. (NRC Staff Exhibit 44). That matter was discussed in regard to construction activities involving concrete. Taylor, at 83.

113. In addition to routine inspections concerning the qualifications of welders and cadwelders, the NRC Staff conducted investigations of allegations that involved the qualifications of the welders and Cadwelders. NRC Inspection Reports 79-15 and 79-22 (NRC Staff Exhibits 120 and 121) discussed, among other charges, an allegation that welders were unqualified. In each instance the investigation revealed that the particular welders had been qualified in accordance with the applicable requirements. As stated above, the inspections and investigations have revealed no instance where a welder performing production welding had not been qualified by the testing methods stipulated by the codes. Taylor, at 83.

(k) <u>Training and Organization of QA/QC Personnel</u>

114. The training aspect of the QA/QC personnel is examined during most of the routine inspections since training is one factor in the determination of the qualifications of a given person to perform an inspection and in the determination of the acceptability of what is inspected. The inspection of the organization is performed much less frequently since it

is generally reasonably static. Emphasis is placed on the organizational aspect of QA/QC in the period prior to issuance of a construction permit and again, at about the mid-point in the construction period. Changes in QA/QC organization, such as the changes which occurred at Comanche Peak during 1978 (as discussed previously), are examined in greater detail by the regional inspectors and by the NRC headquarters personnel who are responsible for assuring that an adequate organization is presented by the applicant in the Safety Analysis Report. The ability and the integrity of the persons in an organization is generally far more important than the exact structure of the organization and generally can only be determined by continued, long-term observation rather than by specific inspections. Taylor and Stewart, at 84-85; Taylor, Tr. 1728-1729.

115. The negative NRC inspection findings regarding training and organization of QA/QC personnel, as summarized by the NRC Staff inspectors (Stewart and Taylor, at 85-87) and as set forth in the NRC Inspection Reports introduced into evidence (NRC Staff Exhibits 54, 69, 70, 86, 107, 139, 150, 156, 157, 158, 163, 164, 168, 169, 173 and 174) reveal certain deficiencies. Taylor and Stewart, at 85-87.

116. However, as described in the testimony of the NRC Staff inspectors, the NRC inspection reports and letters and in the Applicants' letters to the NRC introduced into evidence, the Applicants made commitments to take corrective actions with respect to these negative findings. Stewart and Taylor, at 88-93; NRC Staff Exhibits 56, 88, 89, 90, 91, 101, 110, 111, 139A, 139B, 151, 152, 153, 154, 159, 160, 161, 162, 165, 166, 167, 170, 171, 175 and 176. Taylor and Stewart, at 88-93.

117. The NRC Staff inspectors conducted follow-up inspections to determine that the Applicants had implemented their commitments and that such commitments were effective. Stewart and Taylor, at 93-94. These follow-up inspections were described by the NRC Staff inspectors (Stewart and Taylor, at 94-98) and are set forth in the NRC inspection reports which the Staff introduced into evidence. (NRC Staff Exhibits 38, 54, 58, 103, 112, 155, 172, 177, and 177A). The NRC Staff inspectors concluded that in view of the corrective actions taken by the Applicants, as verified by the NRC Staff, these findings do not raise substantial questions as to the adequacy of the construction of Comanche Peak. Stewart and Taylor, at 98.

118. In addition to the routine inspections described above, the NRC Staff conducted investigations of allegations that either the training of QC personnel was inadequate or that there were improprieties in the organization of the QA/QC personnel. As documented in NRC Inspection Report 81-12 (NRC Staff Exhibit 178), the Staff investigated allegations that 1) QC personnel were not performing their prescribed inspection of fastening devices referred to as "Hilti" bolts in a correct manner; 2) QC personnel were careless or had lost control of an inspection marking media called "Torque Seal," and 3) a repair was made to concrete without the knowledge of QC personnel after a "Hilti" bolt was removed. Taylor, at 98-99. The investigation did not substantiate any of these allegations. Id., at 99.

(1) Hayward-Tyler Pumps

119. The NRC has investigated allegations of problems in the manufacture of pumps by one of the Applicants' vendors, the Hayward-Tyler Pump Co. This investigation has not yet been completed. Taylor, at 99.

120. Hayward-Tyler delivered four pumps used in the safety-related service water system at Comanche Peak. Review of vendor data packages in the possession of the Applicants indicates that Hayward-Tyler apparently only tested for final performance the two pumps associated with Unit 1. These records indicate that nearly all of the manufacturing, assembly and non-destructive testing activity on these pumps was accomplished by AMPCO Metals Division of the AMPCO-Pittsburgh Corporation under the auspices of Babcock & Wilcox Canada Limited, with whom the Applicants had placed the original contract for the pumps. Most of the pressure boundary parts for the Unit 2 pumps appear to have been manufactured under the same circumstances, but the records indicate that Hayward-Tyler may have fabricated some of the non-pressure boundary parts. Hayward-Tyler appears to have assembled as well as tested the Unit 2 pumps. Taylor, at 100.

121. There is no indication in the vendor data records reviewed by NRC Resident Inspector Mr. Taylor that the pumps are defective. However, Hayward-Tyler reported to the NRC pursuant to 10 C.F.R Part 21 that a spare part (pump shaft) may be defective. Taylor, at 100.

122. The two pumps for Unit 1 have been operated intermittently since March 1980 to supply service water in performance of certain tests and to supply cooling water to assist in the cooling of the Unit 1 containment building during times of peak heat to provide a more agreeable working

environment for the construction labor force. One of the pumps for Unit 2 was temporarily substituted for one of the Unit 1 pumps when the Unit 1 pump had to be taken out of service for examination following a maintenance error. To date, the pumps have performed satisfactorily, but a substantial amount of testing for all of the pumps remains to be done prior to issuance of operating licenses for Comanche Peak. Any findings concerning the Comanche Peak pumps resulting from the investigation of Hayward-Tyler will be followed-up by the NRC Staff as necessary. Taylor, at 100-101.

(m) Conclusion

123. The NRC inspection and investigation findings at Comanche Peak to date do not substantiate the allegation in Contention 5 that there are substantial questions as to the adequacy of the construction of the facility. Specifically:

a. The NRC inspections and investigations, as well as assessments of the Applicants' performance on an annual basis, have determined that the Applicants have adequately developed and implemented a QA/QC program meeting NRC requirements. Crossman, Stewart and Taylor, at 101; Taylor, Tr. 1728. Moreover, NRC Staff witness Mr. Taylor, who has been the NRC resident inspector during the period when the bulk of the construction activity took place (including electrical and piping installation and testing) unequivocally expressed his confidence in the quality of construction. Taylor, Tr. 1735.

b. The inspection and investigation findings that were indicative of actual defects in construction were resolved by correction of the defective construction, as necessary, along with correction of the underlying QA/QC program deficiency. All but one of the matters raised in the NRC Inspection Reports cited in support of Contention 5 as negative findings have been resolved and that resolution verified by the NRC Staff. This matter relates to a small amount of concrete of indeterminate quality in the Unit 1 containment which will be finally resolved by the containment structural integrity test. Crossman, Stewart and Taylor, at 101-102.

c. The negative findings in the Staff's inspection reports that were essentially programmatic in nature (i.e., no actual construction deficiency was identified) occurred early in each given phase of construction and were corrected in a timely manner such that there is little likelihood that any significant defective construction has not been detected. Crossman, Stewart and Taylor, at 102; Chapman, Tolson, Vurpillat, Tr. 2105-2106.

d. Although the construction of Unit 1 of the Comanche Peak Station is essentially complete, additional final construction testing remains to be done, along with system tests that are prerequisites to the issuance of operating licenses. These tests are designed to reveal any deficiencies in either the design or construction of the facility that could adversely affect the safe operation of the facility at a future time. Additional inspections, including monitoring of these tests, will be conducted by the NRC Staff to ensure that safety-related construction matters identified will be adequately resolved prior to recommending authorization for the loading of fuel and the operation of Comanche Peak. Crossman, Stewart and Taylor, at 102-103; Taylor, Tr. 1733.

8. Allegations By CASE of Deficiencies in Applicants' QA/QC Program

124. Intervenor CASE raised allegations of specific deficiencies in plant construction or in the QA/QC program. These allegations relate to the rock overbreak which occurred during Comanche Peak excavation; shrinkage cracks in the Unit No. 1 reactor cavity shield wall; expiration of Brown and Root's ASME certificates of authorization; the "Lobbin Report," and the number of NCR's generated by the Comanche Peak QA/QC Program. Although CASE did not present any testimony regarding these matters, the direct testimony of either the Applicants or Staff (or in some case both) addressed these issues. In addition, CASE presented other allegations of deficien-

cies in plant construction or quality assurance/quality control through its witnesses John Junior Gates; Stanley G. Miles; Cordella and Robert Hamilton; Mark Walsh; Jack Doyle;[14/] Charles Atchison and Darlene and Henry Stiner

125. The Board has considered these issues to ascertain the safety significance of the deficiency alleged and whether there are so many confirmed defects in plant construction that, taken as a whole, such defects indicate a breakdown in the QA/QC program.

(a) <u>Rock Overbreak and Cracks in Unit No. 1 Reactor Cavity Shield Walls</u>

126. On June 3, 1982, CASE raised allegations concerning cracks in the Unit No. 1 reactor cavity shield wall and the rock overbreak which occurred during Comanche Peak construction.[15/] At the Licensing Board's request (Tr. 694), the Applicants and the Staff presented oral direct testimony regarding these allegations (Tr. 789-1401). The Applicants presented a panel consisting of Raymond C. Mason; John T. Merritt, Jr.; Kenneth L. Scheppele; Ralph E. McGrane, and Ronald G. Tolson. The

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14/ In accordance with the schedule established by the Board, findings on the "Walsh/Doyle" allegations are not included here, but will be included later. "Reconsideration of December 7, 1982 Order," December 21, 1982, at 2.

15/ On June 3, 1982, CASE filed before the Atomic Safety and Licensing Appeal Board "Intervenor Citizens Association for Sound Energy's Motion to Stay Licensing Board's Order Scheduling Hearings on Contention 5 for June 7." In an "Order" dated June 4, 1982, the Appeal Board denied relief sought by CASE in its entirety. Order, at 1.

Staff's panel of witnesses consisted of William A. Crossman; Robert C. Stewart; Joseph I. Tapia, and Robert G. Taylor. CASE did not present any testimony on these two allegations.

(i) Rock Overbreak

127. The foundations for all CPSES structures are set on the Glen Rose limestone rock formation, a marine formation of Cretaceous genesis. The Glen Rose limestone is not homogeneous, being separated by embedded clay stones, sand, and hard, crystalline layers. Mason, Tr. 803-04, 857-58; Merritt, Tr. 1210.

128. Raymond Mason, Applicants' geotechnical engineering consultant, characterized the Glen Rose limestone as a "weak rock", in which it is difficult to control fracturing and rock overbreak due to blasting. Mason, Tr. 1211.

129. Excavation procedures for Category 1 (safety related) and non-Category 1 structures at CPSES were the same. Mason, Tr. 805.

130. The initial excavation for CPSES removed 60 feet of the Comanche Peak summit to a level plateau, which is defined by Applicants as "plant grade." This overburden was removed during October, November, and December of 1974. Mason, Tr. 805-06; Merritt, Tr. 1239.

131. Approximately two to four feet of soil was removed in this phase of the excavation. The remaining material excavated to plant grade was composed of Glen Rose limestone. The upper portion of the limestone was severely weathered, and was removed by conventional earth-handling equipment. Weathering of the limestone decreased with depth. Unweathered limestone was removed by ripping and blasting. Mason, Tr. 806-07.

132. The excavation for all CPSES structures, including Category 1 structures, commenced from plant grade. The reactor containments required an excavation approximately 40 feet below plant grade. Mason, Tr. 807-08; 813; 922.

133. Mason and Johnson Associates, Applicants' geotechnical engineering consultants, recommended that all structures, both Category 1 and non-Category 1, be constructed on in-situ, or intact, foundation materials. They also recommended that no fills be constructed to support any plant structure. Finally, they recommended that for Category 1 structures, the concrete comprising the structures be placed against intact rock. Mason, Tr. 808-09.

134. Brown and Root, the contractor for Applicants, employed a method known as "line drilling" for excavation of the CPSES reactor containments. Line drilling was employed in order to comply with Mason and Johnson Associates' recommendations. Mason, Tr. 839.

135. Line drilling of the containment excavation involved the drilling of two to four inch diameter holes spaced two feet apart which outline the circular perimeter of the containment. In addition, a series of holes were drilled in the center of the excavation. The blasting holes were loaded with explosives which were detonated with time delay blasting caps. The perimeter explosives were set off so that the detonation would proceed sequentially in a clockwise direction around the perimeter, for a total time delay of 16 milliseconds. The explosives in the center of the excavation were detonated several milliseconds after the perimeter detonation was completed. Mason, Tr. 809-812.

136. The reason for the blasting delay sequence was to create a controlled crack around the perimeter separating the cylinder of rock to be removed from the outer containment foundation, while preventing the formation of cracks past the perimeter which could damage foundation rock. The strength of the Glen Rose limestone is such that a perimeter crack could be propagated by a series of detonations which are closely spaced in time. Simultaneous detonation of the perimeter line explosives would have involved a greater risk of damaging the foundation rock beyond the excavation perimeter. Mason, Tr. 811-12, 948-49.

137. The blasting procedure, blasting sequence, and the size of the explosive charges were developed primarily by the Applicants' contractor, Brown and Root, after separate discussions with the DuPont Powder Company and the Hercules Powder Company. Mason-Johnson contributed some input to the blasting procedures through a discussion with the Applicants' construction manager, and the Brown and Root administrative chief and superintendent of excavation. Mason, Tr. 810, 919-20.

138. Mr. Mason testified that his contribution to the blasting and excavation procedure took into account his previous geotechnical engineering experience with excavations in limestone formations. Mason, Tr. 835.

139. The blasting of the perimeter line for Containment 1 occurred on January 5, 1975. Rock which had been fractured and separated from the center of the excavation was removed by front-end loaders. Mason, Tr. 930, 923, 813; Tolson, Tr. 924.

140. There was little rock that could be excavated as a result of the perimeter and center blast. The remaining rock in Containment 1

excavation was removed after additional blasting. This was achieved by drilling 40 feet deep holes on radii extending from the center of excavation to the perimeter. Additional charges were placed in these holes and detonated. The fractured and separated rock was then removed by front end loaders. Mason, Tr. 813-814.

141. As the rock inside the excavation for Containment 1 was removed, a rock overbreak condition became apparent. The record is unclear or silent on the date that the rock overbreak was first discovered. The upper ten feet of the forty feet deep rock wall was cracked and tilted upward to the center of the excavation. Void spaces were present; the displacement of rock was in the range of several inches. The majority of the overbreak cracks were horizontal, due to the vertical displacement of rock. However, vertical cracks not originally present in the limestone were found, created by blasting gases escaping to the side walls. The overbreak decreased with depth, and was confined to the upper ten feet of the containment excavation. No overbreak occurred in the deeper sections of the containment. The overbreak was inspected and photographed by the Mason-Johnson staff geologist. Mason, Tr. 815-16; 837; CASE Exhibits 17A-Y; 18A-O.

142. The blasting for the excavation of Containment 2 occurred on January 22, 1975. However, the full extent of the rock overbreak in Containment 1 did not become known until January 23, 1975, or one day after the blasting for Containment 2. Tolson, Tr. 924; Mason, Tr. 926-29, 828-30.

143. Mr. Mason testified that the blasting procedure for Containment 2 was changed to prevent the occurrence of rock overbreak at the

Containment 2 excavation. These changes included: (1) reduction of the distance separating the perimeter line drilling holes; (2) selection of a different powder company to supply explosives; (3) selection of a new powder consultant; and (4) reduction of the explosive charges. Mason, Tr. 926, 927, 829.

144. The changes in blasting procedure were inadequate to prevent the occurrence of rock overbreak in Containment 2 which became apparent in early February, 1975. The amount of overbreak in the Containment 2 excavation was approximately equal to that in Containment 1 excavation. Mason, Tr. 829-30, 927-28.

145. Mr. Mason stated his professional opinion that the characteristics of the Glen Rose limestone formation are such that it is not possible to obtain an intact rock foundation using the line drilling and blasting technique. Mr. Mason's professional experience includes geotechnical work for several projects which were constructed on limestone. Mr. Mason testified that on one of these projects (construction of the Fort Randall Dam), rock overbreak in a "niarera" chalk approached the magnitude that was present in the CPSES excavation. Mason, Tr. 830, 945-46, 1210-12.

146. Although overbreak occurs in all excavations involving blasting, the extent of the overbreak in the CPSES excavations was greater than expected. Mason, Tr. 835, 1175, 1179; Merritt, Tr. 1208-10.

147. Although there were several alternatives for solving the rock overbreak problem, the Applicants ultimately adopted the alternative recommended by Mason-Johnson Associates, and concurred in by Gibbs and

fill. In the opinion of Mr. Mason, the solution implemented by Applicants was the best option from a structural viewpoint. Mason, Tr. 817-18.

148. The alternative employed by Applicants to repair the rock overbreak for Containments 1 and 2 involved inspection of the rock near the surface to determine the radial extent of overbreak cracks, careful removal of all displaced rock, and restoration of the original foundation geometry through the use of dental concrete.[16/] Mason, Tr. 817.

149. Applicants determined the extent of the rock overbreak by excavating a circular trench no more than three feet deep, approximately five feet beyond the perimeter of each containment. The surface layers of limestone are most sensitive to rock overbreak, due to the characteristics of crack propagation in limestone containing claystone layers. The field geologist examined the walls of the trench to determine if a crack exposal on the perimeter side of the trench wall continued on the trench wall away from the perimeter. If such a crack were found, trenches were dug at greater distances from the perimeter, until cracking disappeared. At this distance, the overbreak rock was excavated to a depth of ten feet, the vertical limit of the overbreak. Mason, Tr. 817-21, 944-52.

150. Overbreak rock was removed by the technique known as "drilling and broaching." In this technique, holes were drilled which were tangent to one another along the perimeter. An expansion tool was inserted into the holes to fracture the rock. The rock was then removed by hand tools

16/ Dental concrete is a construction term given to concrete that is used to repair a defect in rock. Mason, Tr. 826.

and front end loaders. No blasting of in-situ overbreak rock occurred. Mason, Tr. 922, 817, 821-22.

151. The rock surfaces exposed after removal of the rock overbreak were broom cleaned, air hosed, and watered. The surfaces were then examined for cracking. The cracks were grouted with grout pipes, after placement of the containment base mat. Pressures of the grouting were monitored to prevent the grout from propagating a crack in the limestone foundation, or lifting up the base mat. Mason, Tr. 817, 823-24, 954-55.

152. Dental concrete was placed in the ten feet deep excavation formerly occupied by the rock overbreak. The dental concrete was mixed on-site, and placed on clean rock as the base and exterior form. An interior form was utilized to prevent the dental concrete from spilling into the original containment excavation. Mason, Tr. 825-26.

153. The dental concrete employed to repair the rock overbreak had a strength of approximately 2,500 pounds per square inch (p.s.i.) at an age of 28 days. The strength of the dental concrete was less than the strength of the concrete employed in the CPSES structures, but was greater than the strength of the Glen Rose limestone. No reinforcing steel was used in the placement of dental concrete. Mason, Tr. 826-28.

154. Mr. Mason stated that there was no need to place dental concrete which is stronger than the limestone base material; and that rather than place concrete equal to the strength of the Glen Rose limestone, 2,500 p.s.i. dental concrete was placed as a flexible, economical mix. Reinforcing steel was not used, since the object of placing dental concrete was to reproduce the characteristics of the Glen Rose limestone. Mason, Tr. 827-28, 955-56.

155. The placement of dental concrete, rather than the use of the original limestone rock, does not have an impact on seismic analysis, according to Applicants witness, Mr. Scheppele. Mr. Mason testified that seismic characteristics of the foundation were improved by the substitution of a stronger material (dental concrete), for the weaker, non-homogeneous Glen Rose limestone. Scheppele, Tr. 958; Mason, Tr. 827-28, 957-58.

156. The Applicants' expert witnesses on the rock overbreak unanimously concluded that conditions at the conclusion of the removal and repair of the rock overbreak and rock displacement were equal to or superior to the original design concept. Mason, Tr. 828, 835, 838; Scheppele, Tr. 838; McGrane, Tr. 838-39.

157. Excavation of the CPSES fuel building resulted in rock displacement and cracking in the walls and base of the excavation in the limestone foundation. Mason, Tr. 824, 831-32.

158. Rock which had been horizontally displaced and was intended to give resistance against a wall was removed, using the procedures described for the removal of rock overbreak at the containment excavations. Rock which had not been displaced was repaired by grout injection. Dental concrete was placed in the void formerly filled by displaced rock. Mason, Tr. 833.

159. Removal of the displaced and fractured rock at the fuel building excavation by jackhammers and clay spades resulted in a plane surface with two to four inch undulations. The undulations act as a continuous keyway, so that there is intimate, shear resistant contact between the dental concrete and the undisturbed rock. Mason, Tr. 834.

160. The excavation activities for Category 1 structures at CPSES were not subject to 10 C.F.R. Part 50, Appendix B Quality Assurance ("QA") procedures and controls, since Gibbs and Hill did not specify that these QA procedures were applicable. Tolson, Tr. 841, 1040, 1172; McGrane, Tr. 1041.

161. The repair of the excavations for the containments and fuel building were subject to QA procedures, since the specifications were amended to impose the QA requirements of 10 C.F.R. Part 50, Appendix B. Tolson, Tr. 834-44.

162. The rock overbreak for all CPSES building excavations was verbally reported by Applicants on approximately February 4-5th to Mr. Robert Stewart, who was the NRC principal reactor inspector for CPSES. Tolson, Tr. 845; Stewart, Tr. 1270.

163. Applicants later sent a written interim report to the NRC Region IV Office in February, 1975, and a final report in December, 1975, in accordance with the requirements of 10 C.F.R. Section 50.55(e). Tolson, Tr. 845-46; Stewart, Tr. 1270.

164. Mr. Stewart personally inspected the rock overbreak at the containment, fuel building, safeguards buildings, and auxiliary building excavations. Mr. Tolson described the extent of the overbreak fracturing as "quite visible" and "very obvious". According to Mr. Stewart, the overbreak was so extensive that there was no need to distinguish between building boundaries, since all building excavations were affected. Stewart, Tr. 1269-70; Tolson, Tr. 846-49.

165. Rock fractures present in the overbreak were adequately identified, mapped, and repaired by grouting in accordance with Gibbs and

Hill specifications, and Brown and Root and Mason-Johnson, Associates construction procedures, based on Mr. Stewarts' personal observations, inspections and evaluations. Stewart, Tr. 1272.

166. Repair of the rock overbreak, including the cleaning of rock surfaces, removal of rock, and placement of dental concrete, was personally observed and inspected by Mr. Stewart. Mr. Stewart had discussions with the Mason-Johnson geologist on-site. He also reviewed the relevant Gibbs and Hill specifications, Brown and Root procedures, and Mason-Johnson Associates procedures. Mr. Stewart testified that the repairs of the rock overbreak were made in accordance with these specifications and procedures. Stewart, Tr. 1271.

167. NRC Inspection Report 75-05 (CASE Exhibit 15), documents Mr. Stewart's finding that the Applicants failed to develop and implement appropriate inspection procedures for blasting and excavation of CPSES Containments 1 and 2. Applicants responded to this Inspection Report by committing to developing appropriate changes in the written procedures for blasting and inspection, consistent with current administrative controls for Category 1 structure excavations. Stewart, Tr. 1272-76, 1279-80; CASE Exhibit 15, Applicants' Exhibit 26, NRC Staff Exhibit 10.

168. Applicants developed CPS-QCPS 3S, "Plantsite Rock Blasting Surveillance," in compliance with their committments to develop procedures. Tolson, Tr. 1056-57.

169. Mr. Stewart confirmed that the Applicants' fulfilled their commitments to amend their procedures for blasting and inspection by personally reviewing these documents. His findings are reported in NRC Inspection Reports 75-06 (NRC Staff Exhibit 11), 75-07 (NRC Staff

Exhibit 12) and 75-04 (CASE Exhibit 37). Stewart, Tr. 1276-1237 (NRC Staff Exhibits 11 and 12); CASE Exhibit 37.

170. Mr. Mason testified that one factor in the decision not to classify the excavations for Category 1 structures as safety-related was the original Gibbs and Hill specifications, which required the presence of an engineering geologist at all times. The geologist would verify the quality of the rock upon which concrete would be placed, and photograph all excavations. The specifications were made in accordance with a commitment by Texas Utilities Services, Incorporated ("TUSI") to the Atomic Energy Commission ("AEC") and subsequently to the NRC that an engineering geologist would be present at all times during excavation. Prior to 1975, the TUGCO QA manual included a job description for an engineering geologist. Mason, Tr. 1046-48, 1174-75.

171. The engineering geologist present at the CPSES site excavation was Mr. Herb Crowder, Mason-Johnston Associates. He evaluated and provided his professional judgments on the quality of the excavations, in accordance with the Gibbs and Hill specifications. Mason, Tr. 1047-48.

172. The principal difference between excavation without QA procedures, and with those procedures, is the lack of independent verification surveillance or inspection of the contractors' blasting procedures. With the exception of the lack of surveillance records, there is no difference between the records actually kept, and records that would have been kept if QA procedures were applicable. Tolson, Tr. 1173, Mason, Tr. 1176.

173. In Mr. Mason's professional opinion, the overbreak problem would not have been better identified if a formal QA program was present

from the start of the excavation activities, since there was no distinguishable change in Mason-Johnson Associates' activities following the establishment of a formal QA program for excavation at CPSES. Mason, Tr. 1176-77.

174. The technical evaluations and repair of the rock overbreak would not have been different if the formal QA/QC program were applicable from the start of excavation activities for CPSES. Tolson, Tr. 1173-74, Mason, Tr. 1176.

175. The rock overbreak was not caused by sink holes and solutions channels in the Glen Rose limestone attributable to the presence of an aquifer below the CPSES site. Mason, Tr. 947.

176. The Board finds that the occurrence of the rock overbreak during CPSES excavation does not represent a breakdown in construction QA/QC. The overbreak was carefully removed and repaired in accordance with the recommendations of the Applicants' geotechnical engineering consultants and QA procedures. There is no safety significance to be attributed to the occurrence of the overbreak, since the overbreak was adequately repaired to provide a foundation as good if not better than the originally contemplated foundation (Glen Rose limestone).

(ii) Shrinkage Cracks

177. The foundation for the CPSES containments is comprised of dental concrete and the unfractured rock surfaces of the containment excavations. Scheppele, Tr. 849.

178. A twelve feet thick horizontal base mat (also referred to as the foundation mat) is placed on the floor of the containment excavation. Scheppele, Tr. 850-52.

179. A twelve feet high cylindrical wall, twelve feet in thickness, is placed on the base mat. At the top of this wall, and extending outward approximately seventy-five feet from the center of the containment, is the upper base mat. The upper base mat is placed partially on dental concrete, and partially on unfractured rock. The elevation of the top surface of the upper base mat is 805 feet, 6 inches. Scheppele, Tr. 850-51, 855-56.

180. The upper and lower base mats, and the vertical cylindrical wall are composed of concrete reinforced with steel bars ("rebars"). Scheppele, Tr. 851.

181. The base mat is the supporting element for the containment. Scheppele, Tr. 850.

182. A steel liner, which serves as a pressure boundary for the containment, is laid against the inner surfaces of the base mat and the cylindrical wall. All concrete structures within the containment which are utilized to support the reactor vessel, steam generators, and coolant pumps, are placed on the steel liner plate. Scheppele, Tr. 851, 853-54, 861-62.

183. The reactor shield wall (reactor cavity wall) is a hollow cylindrical concrete structure inside containment. The shield wall is approximately twenty-five to thirty-five feet in height, with a minimum wall thickness of eight feet 6 inches. Scheppele, Tr. 854-56, 860-61, 866.

184. The reactor vessel is suspended within the reactor shield wall cavity. The vessel is supported at the two reactor vessel cold legs and two hot legs. Tapia, Tr. 1296.

185. The primary purpose of the reactor shield wall is to provide radiation shielding. The 8.5 feet minimum thickness of the reactor shield wall was set primarily because of radiation shielding requirements, not because of structural requirements. Scheppele, Tr. 855, 866, 869; Tapia, Tr. 1300, 1334-35.

186. The reactor shield wall concrete is heavily reinforced with reinforcing steel bars. Scheppele, Tr. 866.

187. Concrete is utilized as a compression element and not as a tensile element in reinforced concrete design. A compression element is one where the forces acting on the concrete are pushing towards one another. A tensile element is one where tensile forces acting on the concrete are attempting to pull the concrete apart. Concrete is never used as a tensile element because concrete in tension is a weak, brittle material. Scheppele, Tr. 852-53, 866, 1024.

188. Steel has a high tensile strength. Therefore, steel bars are used in reinforced concrete design to accept the tensile forces imposed on a reinforced concrete element. In order for reinforcing steel to work in a reinforced concrete element, it is necessary for the concrete to crack, in most situations. Scheppele, Tr. 853, 856.

189. Two shrinkage cracks are located in the reactor shield wall for Unit 1. One crack is on the north side of the shield wall (crack "A" on Applicants' Exhibits 22 and 23); the other crack is on the South side of the shield wall (Crack "B" on Applicants' Exhibits 22 and 23).

190. Shrinkage cracks A and B are not located in the containment base mat, as erroneously identified in a newspaper article in the Thursday, June 3, 1982 edition of the Dallas Morning News. Mason, Tr. 1181; Scheppele, Tr. 854; McGrane, Tr. 1184-86.

191. Shrinkage crack A is located in a concrete placement approximately 6.5 feet high, extending from the top surface to the upper base mat, elevation 805.6, to elevation 812. The in-plane dimensions of the crack are approximately 8.5 by 6.5 feet. Scheppele, Tr. 856; McGrane, 997, 1001.

192. Shrinkage crack B is located in a concrete placement approximately thirteen feet high, and is part of a structural element thirty-five feet high. The placement at the location of shrinkage crack B spans a 20 feet cavity for housing the in-core instrumentation tubes for the reactor core. The in-plane dimensions of the crack are approximately 9.5 by 13 feet. Scheppele, Tr. 856, 863, 1190; Tapia, Tr. 1393-94; McGrane, Tr. 999-1001.

193. Applicants' Exhibit 22 shows a plane view of the two shrinkage cracks, as well as the limits of the concrete placement where the cracks are located. Scheppele, Tr. 867, 868, 872-73, 876-77.

194. The reactor support leg labeled Y on Applicants' Exhibit 38 is over the in-core instrumentation cavity, and is displaced approximately five feet in the horizontal plane from the centerline of shrinkage crack B. The reactor support leg labelled Z on Applicants' Exhibit 38 is not over the instrumentation cavity. Tapia, Tr. 1326, 1329.

195. Shrinkage cracks A and B are hairline in appearance, and appear to be jagged, according to Applicants' witness, Mr. McGrane, who personally viewed the shrinkage cracks. McGrane, Tr. 870, 996.

196. In the professional opinions of Staff witnesses Messrs. Taylor and Tapia, shrinkage cracks A and B extend at most approximately 2 inches into the concrete. Mr. Taylor based his opinion on his professional experience that shrinkage cracking is a surface phenomenon which proceeds inward until its first encounter with reinforcing steel within the concrete. Mr. Tapia based his conclusion on his professional experience, as well as a review of the professional literature. Taylor, Tr. 1366, 1368, 1374; Tapia, 1375, 1378.

197. Shrinkage cracks are formed in concrete that is restrained, in order to relieve the concrete of tensile stresses generated by: (1) concrete shrinking, and (2) thermal gradients between the surface and the interior of the concrete, due to heat generated by the chemical reaction of curing concrete. Which force is more important in causing shrinkage cracks is dependent upon a number of factors, including the configuration of the placement, and the mass of concrete in the placement. Tapia, Tr. 1379-1381; Scheppele, Tr. 870-71.

198. The shrinking of concrete is due to the chemical reaction (curing or drying) which proceeds during the placement of concrete. The shrinkage due to concrete curing is approximately 500 millionths of an inch per inch of concrete. Tapia, Tr. 1377-78; Scheppele, 870-71.

199. Concrete shrinkage was probably more important than thermal stress in causing shrinkage cracks A and B, according to Mr. Tapia. Tapia, Tr. 1380.

200. The occurrence of shrinkage cracks A and B is attributable to: (1) the configuration of the concrete placement, and (2) the lack of construction joints where the cracks are located. Both Mr. Tapia and Mr. Scheppele explained that two large areas of the placement are connected by relatively thin concrete sections which were placed monolithically. As the two large masses of concrete contract, tensile forces are created at the surface of the reduced concrete sections connecting the two large masses. Accordingly, one would expect to find shrinkage cracks at the surface of those sections, especially if a construction joint were absent. Tapia, Tr. 1384-87; Scheppele, Tr. 882.

201. The shrinkage cracks were minor deficiencies in construction, and were not the type of significant construction deficiencies that should have been reported to the NRC pursuant to 10 C.F.R. 50.55(e), in the opinion of both Mr. Scheppele and Mr. Tolson. Scheppele, Tr. 884-85, 892; Tolson, Tr. 904-906.

202. The shrinkage cracks act as a natural construction joint. The rough surfaces of the shrinkage cracks due to the aggregate serves to transmit shear forces. Scheppele, Tr. 883; McGrane, Tr. 1182.

203. The concrete placement in the area of both shrinkage cracks is reinforced by large amounts of No. 11 reinforcing steel bars, which are 1-3/8 inches in diameter. The bottom of the placement spans the in-core instrumentation cavity, and is heavily reinforced with horizontal steel bars in order to accept the tension loads on the concrete placement. McGrane, Tr. 897; Scheppele, Tr. 1190; Tolson, Tr. 1204, 1206-07; Tapia, Tr. 1393-94, 1399.

204. There would be little, if any, structural distinction between a continuous placement, and a placement with construction joints. Scheppele, Tr. 883, 1192-94.

205. Use of a construction joint at the site of each of the shrinkage cracks would not have changed the scheme of reinforcing bars employed in the reactor shield wall. Scheppele, Tr. 1192-93.

206. In the opinion of Applicants' witnesses, the shrinkage cracks do not impair the structural integrity of the reactor shield wall, since the stresses in the concrete are very low because the thickness of the wall was not set by structural considerations. Scheppele, Tr. 883-885; McGrane, Tr. 1183.

207. The radiation shielding capability of the shrinkage cracks has not been impaired, since the rough surface of the cracks act as a keyway. Scheppele, Tr. 883, 885-886.

208. NRC Staff witness, Mr. Tapia, also stated that the structural capability of the reactor shield wall to carry the loads it contains will not be impaired. Tapia, Tr. 1300-1301.

209. In the opinion of Mr. Tapia, the shrinkage cracks do not have a safety significance such that they would be deemed reportable under 10 C.F.R. Section 50.55(e). Tapia, Tr. 1311.

210. The shrinkage cracks were cosmetically repaired to create a smooth, even surface for painting, and to prevent spalling of the concrete. Spalling includes the accidental removal of concrete from the surface of a placement, through chipping, stresses in the placement, cracking, or a surface hit. There was no reason to repair the cracks from a structural standpoint. Scheppele, Tr. 886-888; Tapia, Tr. 1313.

211. The exposed surface of the shrinkage cracks was repaired by creating a V-shaped groove (notch) along the center line of the crack, and filling the groove with grouting material. Scheppele, Tr. 886, 891-92; McGrane, Tr. 892; Tapia, Tr. 1313.

212. The repair procedure for the cracks was developed by Gibbs and Hill in close consultation with the contractor, Brown and Root. Scheppele, Tr. 914.

213. In the opinion of Mr. Scheppele and Mr. McGrane, the procedure developed for repairing the shrinkage cracks was adequate. Scheppele, Tr. 915; McGrane, Tr. 914.

214. The shrinkage cracks were discovered in April 1977 by James D. Pace, a Brown and Root QC inspector. The cracks were initially documented in an informal memorandum to construction from QC and subsequently formally documented on a NCR issued during June 1977. CASE Exhibit 23. Repair of the cracks did not commence until February or March 1979. McGrane, Tr. 893; Merrit, Tr. 893-94; Tolson, Tr. 899-903.

215. No special measures were taken by Applicants to prevent moisture from getting into the shrinkage cracks. Placement of concrete over the cracks isolated the cracks such that the possibility of moisture penetrating the cracks is extremely remote. Scheppele, Tr. 898; Merritt, Tr. 1022-1023.

216. Curing water from the concrete placement placed above the placement containing the shrinkage cracks may have seeped into the cracks for several days after the initial placement of concrete. McGrane, Tr. 395-97.

217. There is no possibility that there could have been structurally significant corrosion of the steel reinforcing bars in the placement containing the shrinkage cracks due to curing water, or other water sources. McGrane, Tr. 894-95, 897-98; Merritt, Tr. 1022-1024.

218. Experience with concrete structures submerged in water shows that there have been no failures due to corrosion of steel rebars exposed by shrinkage cracks in those structures. McGrane, Tr. 815; Merritt, Tr. 1023.

219. The north side of the reactor shield wall functions as a concrete wall, transferring the vertical loadings of the reactor vessel to the base mat. There is no structural significance to shrinkage crack A in the north side of the reactor shield wall. Scheppele, Tr. 869; Tapia, Tr. 1298.

220. The south side of the reactor shield wall functions as a beam, transferring the reactor vessel loadings across the in-core instrumentation tunnel, and then down to the base mat. Mr. Scheppele assumed the worst case condition that crack B went through the entire placement. Nonetheless, he concluded that the shrinkage crack does not impair the ability of the beam to accept the load imposed by the reactor vessel at support Y. Scheppele, Tr. 873-874, 885.

221. Mr. Tapia also assumed the worst case condition that crack B went through the entire placement at the location of the cracks. Nonetheless, he concluded that the shrinkage crack B does not impair the ability of the placement reactor wall to carry the load imposed on the wall by reactor vessel support Y. Mr. Tapia's conclusion was based on: (1) the large amount of reinforcing steel in the placement which will

accept the tension load, and (2) the additional reinforcing steel and concrete placements above the placement containing shrinkage crack B, which will also carry some of the loads from the reactor vessel. Mr. Tapia also testified that his conclusions regarding the structural adequacy of the placement acting as a beam over the in-core instrumentation tunnel took into account the relative inability of concrete to accept shear forces. Tapia, Tr. 1372-74, 1381-82, 1393-95, 1398.

222. The Applicants have performed analyses which confirm that the actual stress levels in the placements containing the cracks are considerably below the actual structural capabilities of the placements. McGrane, Tr. 1183.

223. The reactor shield wall at crack A may be considered as a beam approximately twenty feet high to the reactor support leg elevation. The twenty-foot high concrete shield wall has the capability to support four times the load actually placed on the wall by the reactor. The steel in the wall, considered by itself, has the capability to support sixteen times the load actually placed on the wall by the reactor. Mason, Tr. 1186-88; Scheppele, Tr. 1191-92.

224. The reactor shield wall at crack B may also be considered as a twenty-foot high beam. The reinforcing steel in that beam is more than sufficient to transfer the sheer loads imposed by the reactor vessel at reactor support leg Y. Scheppele, Tr. 1191-92; McGrane, Tr. 1190-91.

225. Mr. McGrane stated that there was no connection between the rock overbreak and its repair, and the occurrence of shrinkage cracks A and B. His opinion is based upon the relatively unyielding rock foundation

hundreds of feet deep, upon which is constructed the relatively immovable concrete base mat heavily reinforced with steel bars. McGrane, Tr. 1214-16.

226. None of Applicants' witnesses were aware of any reinforcing bars being omitted from any placement in the reactor shield wall. Intervenor did not present any evidence regarding the omission of rebars. Scheppele, Tr. 1025-26; McGrane, Tr. 1206-07; Tolson, Tr. 1206.

227. The Board finds that the shrinkage cracks have no safety significance. The cracks do not adversely affect the structural or radiation shielding capabilities of the reactor shield wall. The cracks were repaired to prevent spalling and do not raise any question regarding the adequacy of the Applicants' QA/QC procedures.

(b) American Society of Mechanical Engineers Certificates of Authorization

228. During discovery, CASE raised as an issue the expiration of Brown and Root's ASME certificates of authorization. Applicants' witnesses Messrs. Vurpillat and Reedy, who testified as part of Applicant's initial panel of witnesses, testified regarding this issue. "Testimony of Raymond J. Vurpillat Regarding Brown & Root Response to ASME Survey and Resurvey" ("Vurpillat," Applicants' Exhibit 45) and "Testimony of Roger F. Reedy Regarding ASME Survey of Brown and Root at Comanche Peak" ("Reedy," Applicants' Exhibit 46).

229. The American Society of Mechanical Engineers ("ASME") writes and publishes a set of Engineering Safety Standards known as the ASME Boiler and Pressure Vessel Code. Section III of that Code, known as the ASME Nuclear Code, provides rules for the design and construction of pressure vessels, piping systems, pumps, valves, storage tanks, component

supports and core support structures used in nuclear power plants. The Nuclear Code presents a set of minimum requirements developed to assure the integrity and safety of the pressure containing equipment and related systems in the nuclear power plant. Reedy (Applicants' Exhibit 46), at 2-3.

230. The ASME Nuclear Code provides requirements for the construction and installation of nuclear components. NRC regulations (10 C.F.R. § 50.55a) require that for nuclear power plants with construction permits docketed prior to 1982, such as Comanche Peak, the systems and components of those plants be designed, fabricated, installed, tested and inspected in accordance with generally recognized codes and standards, such as the requirements of the ASME Nuclear Code. These regulations do not, however, require Code stamping. Reedy, at 3.

231. All work performed pursuant to the ASME Nuclear Code must be certified by an ASME Certificate Holder as complying with the applicable requirements of that Code. In addition, the certification is authorized by ASME's issuance of Certificates of Authorization and ASME Code Symbol Stamps which permit the stamping of the item with an ASME Code Symbol Stamp following completion of the work being performed. These Certificates and Stamps are issued only upon a satisfactory demonstration to the ASME that an acceptable Quality Assurance Program exists and is being properly implemented for the work performed pursuant to the requested authority. Also, an Authorized Nuclear Inspector (ANI) must concur that the requirements of the Code were met. This is based upon his independent inspection of the Code work performed. Reedy, at 3-4.

232. Before the ASME will issue Certificates of Authorization to any organization and thereby authorize application of the ASME Code Symbol

Stamp to any code item, an ASME Survey Team must review the facilities and programs of the organization to assure that the Quality Assurance Manual and related controlling procedures comply with the applicable requirements of the ASME Nuclear Code. The Survey Team also must audit for verification of proper implementation of the QA Program. Reedy, at 4.

233. The verification of implementation includes review by sampling work in progress and appropriate records, including some records of already completed work, and interviewing personnel. Once this verification has been made, the Certificates of Authorization and applicable Code Symbol Stamp will be issued to the organization upon the recommendation of the Survey Team and with the concurrence of the ASME Subcommittee on Nuclear Accreditation. The Certificates of Authorization allow the organization to apply the ASME Code Symbol Stamp to components. This authorization is valid for a period of three years. Reedy, at 4-5.

234. There are three types of ASME Code Symbol Stamps. The "N" Symbol Stamp is the stamp which signifies that all requirements for design and construction of ASME Nuclear Code components have been met and accepted. The N Symbol Stamp is used by the organization which accepts overall responsibility for the completed component. The "NPT" Symbol Stamp is used for parts of ASME Code components and signifies that the work for that portion of a component has been acceptably accomplished. An NPT Stamp does not include responsibility for the design. The "NA" Symbol Stamp is a stamp which indicates that the equipment and related piping has been installed or assembled in accordance with Code require-

ments. At Comanche Peak, Brown & Root has been issued both the NA and NPT Symbol Stamps. Reedy, at 5.

235. A fundamental controlling principle of the ASME Boiler and Pressure Vessel Code is that a component must receive "third-party authorized inspection" during construction and acceptance testing to assure compliance with all Code requirements. This inspection process is in addition to, not in lieu of, a full Quality Assurance Program of the Certificate holder. Reedy, at 5-6.

236. The third-party inspection for nuclear components is performed by the Authorized Nuclear Inspector ("ANI"), who is employed by the Authorized Inspection Agency ("AIA"), here the insurance company which underwrites insurance for the type of equipment being constructed at Comanche Peak, viz., the Hartford Steam Boiler Inspection and Insurance Company. The signature of an ANI on an ASME Data Report, verifying that the the component has been constructed in accordance with the applicable Code requirements, is a key element in the ASME accreditation system. The insurance company inspectors are independent of both the manufacturer and the operator of the equipment and have principal interest in safety because of the potential financial obligation of the insurance company on the equipment. Reedy, at 6.

237. The National Board of Boiler and Pressure Vessel Inspectors is a professional group comprised of the Chief Inspectors of each of the States which have adopted the ASME Code rules for boilers, pressure vessels or nuclear components. The National Board trains inspectors, prepares and administers examinations and certifies the qualifications of the ANIs. The National Board assists the ASME by participating on the

Survey Teams who review and qualify manufacturers and installers of both nuclear components and other ASME pressure vessel and boiler construction. The National Board reviews the effectiveness of the Authorized Inspector program to assure that Inspectors are performing their duties in accordance with the requirements of the National Board and the ASME Code rules. Reedy, at 6-7.

238. Before construction activities on a nuclear component can be initiated, it is necessary for the organization doing the work to have an agreement with an Authorized Inspection Agency for providing inspection services. Before entering into such an agreement, the AIA must review and accept the Quality Assurance Manual of the organization and determine that the organization is capable of properly implementing the QA Manual. Reedy, at 7.

239. Once these activities have been accomplished, it is acceptable for the organization to begin ASME Code work on the item which is to be certified and Code Stamped with the concurrence of the ANI. That ASME Code work may include Code design and construction activities engaged in prior to receiving a Certificate of Authorization from ASME. Published ASME interpretations of the Nuclear Code established such practice as acceptable and permissible under Code rules. Applicants' Exhibit 46A is a copy of such interpretations. All Code work performed during this time is subject to review during the ASME Survey Team's review and audit of the QA Program. It is an accepted practice for organizations to begin work in this way, in order that the ASME Survey Team can verify implementation of the QA Program on actual Code work when the survey is performed. Reedy, at 8.

240. In general, the purpose of an ASME Quality Assurance Program is to establish an organizational control plan for performing and assuring that the work on a project is accomplished in accordance with applicable design, fabrication and installation requirements. Reedy, at 9.

241. The Quality Assurance Manual describes the essential features of the QA Program by detailing the responsibilities of personnel and controls needed for the work performed. ASME Code Section III, NCA-4134.2(c).[17/] The QA Manual, in conjunction with the necessary supplementary implementing procedures, provides the specific details for control and documentation of the work. ASME Code Section III, NCA-4134.2(a) and NCA-4134.5. Some QA personnel prefer that the controls be outlined in the QA Manual with specific details being described in procedures. Some ASME Survey Teams prefer that the specific controlling details be described in the QA Manual with less reliance on implementing procedures. Reedy, at 9-10.

242. In the ASME Code, it is a requirement that the QA Manual describe the essential controls of the Quality Assurance system. However, the Code permits the QA Manual to be supplemented by procedures. The principal reason that the essential controls are required to be described in the QA Manual is that revisions of the ASME QA Manual require more approvals than revisions to procedures, thus providing additional controls over the QA Program. ASME Code Section III, NCA-4134.2(d). Reedy, at 10.

17/ All references to the ASME Code are to the 1980 Edition through the Winter 1981 Addenda. Reedy, at 9. All Code provisions cited herein are set forth in Applicants' Exhibit 46B.

243. In performing a review of an ASME QA Program, an ASME Survey Team will first examine the subject QA Manual in detail. After the QA Manual has been reviewed, the Survey Team must establish that Code work is capable of being performed in accordance with the provisions of the Manual. Comments on the ASME QA program for Comanche Peak are discussed in the ASME letter of November 23, 1981 (Applicants' Exhibit 45A). Reedy, at 10-11.

244. At Comanche Peak, although Brown & Root has functional management under their own QA Program of activities within the jurisdiction of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 (ASME Code), TUGCO has ultimate responsibility for such work. Chapman, at 3.

245. On October 12-14, 1981, an ASME Survey Team conducted a survey of Brown & Root's ASME QA Program at Comanche Peak for renewal of the NA & NPT certificates. The ASME Survey Team made findings both with respect to the Brown & Root ASME QA Manual and with respect to implementation of that manual. The first six comments of the ASME Survey Team concerned the manual. The remaining comments concerned implementation of the manual. Vurpillat (Applicants' Exhibit 45), at 2.

246. Brown & Root's initial ASME certificate of authorization expired on January 8, 1982. Reedy, at 8.

247. Work performed while a lapse in the Certificate of Authorization is in effect, as occurred at Comanche Peak following expiration of the Brown & Root initial Certificates on January 8, 1982 and prior to their reissuance on March 15, 1982, is treated in the same manner as work performed by a contractor prior to the initial ASME Survey. The acceptance

of the work performed during this period of time is at the discretion of the ANI and his supervisor. Reedy, at 8.

248. Each matter raised by the ASME Survey Team at the October 12-14, 1981 survey of the Brown & Root ASME QA Program at Comanche Peak was addressed and corrective action taken by Brown & Root. Vurpillat, at 2.

249. Brown & Root has taken measures to assure that matters identified by the ASME Survey Team will not recur. Vurpillat, at 2.

250. On January 18-20, 1982, an ASME Survey Team conducted a resurvey of the Brown & Root Quality Assurance Program at Comanche Peak. Reedy, at 2. That resurvey resulted in a recommendation by the ASME Survey Team to the ASME Subcommittee on Nuclear Accreditation that the B&R Certificates of Authorization be renewed. This recommendation was contingent upon the completion of three actions and verification of such completion by the Authorized Nuclear Inspection Supervisor. Reedy, at 28.

251. Brown & Root has taken corrective actions in response to the findings of the ASME Survey Team at the January 18-20, 1982 resurvey and the Authorized Nuclear Inspector has verified completion of these items. Vurpillat, at 17.

252. On February 8, 1982, the Authorized Inspection Agency ("AIA") for Brown & Root at Comanche Peak, Hartford Steam Boiler Inspection and Insurance Company, transmitted verification to ASME that action for those three items had been verified as complete. Applicants' Exhibit 45B; Reedy, at 28-29. These matters, together with their resolution as stated in the AIA letter, evidence satisfaction of the technical requirements of the ASME Code and do not pose a concern that work performed prior to or since their resolution does not meet Code standards. In addition, the

Survey Team leader returned to Comanche Peak on May 6, 1982, and verified that the three items had been satisfactorily resolved. Reedy, at 28-29.

253. The Brown & Root ASME Certificates of Authorization for Comanche Peak were reissued on March 15, 1982. Vurpillat, at 20. Issuance of these certificates indicates that ASME is satisfied that the Brown & Root ASME QA Program has fulfilled the requirements of the Code. Vurpillat, at 21.

254. Each of the actions taken by Brown & Root in response to the ASME Survey Team findings at both the October 12-14, 1981 survey and the January 18-20, 1982 resurvey resolved those findings satisfactorily. Reedy, at 29. Brown & Root has satisfactorily demonstrated, pursuant to the requirements of the ASME Code, that ASME Code work they have performed at the Comanche Peak site, which was the subject of the ASME survey and resurvey, fully satisfies the applicable ASME Code requirements. All matters raised by the ASME Survey Team at both the October 1981 Survey and the January 1982 Resurvey were resolved by Brown & Root. Specifically, Brown & Root has demonstrated to the proper authorities implementation of an appropriate QA program for ASME Code work completed at Comanche Peak up to the present time. Accordingly, no safety issues were raised during the survey process, and the matters which were raised have been resolved satisfactorily. Reedy, at 29.

(c) The Number of NCRs

255. CASE introduced into evidence scores of NCRs, construction records such as Deficiency and Disposition Reports ("DDRs"), Deficiency

Reports ("DRs") and logs of DDRs and DRs (see e.g., CASE Exhibits 305-570, 626 through 628) which it contends show that the construction QA/QC program is not working. CASE did not offer any testimony in support of that assertion. In contrast, Applicants and Staff presented testimony which refutes CASE's assertion.

256. Applicants' witness, Mr. Chapman, described the principal records which document the involvement of QA in Comanche Peak construction activities. Chapman (Applicants' Exhibit 42), at 10-11.

257. The Inspection Report ("IR") is the primary record documenting QA involvement in non-ASME construction activities. It contains the attributes and results of the inspections conducted of the various non-ASME construction activities. The Weld Data Cards ("WDC") and Nondestructive Examination ("NDE") Reports are the principal records which document the involvement of QA in Comanche Peak ASME construction activities. Unsatisfactory conditions that cannot readily be corrected in the field are documented on either a Nonconformance Report ("NCR") or Field Deficiency Report ("FDR"). Id., at 10.

258. The NCR is the form used to document formally a nonconforming condition that requires further action by the project construction and/or engineering organizations. The FDR is the form used to identify and resolve relatively minor matters discovered through routine inspection. In addition, a Corrective Action Request ("CAR") can be issued to document the need for corrective action involving more important or repetitive conditions. Id., at 10-11.

259. NRC Staff witnesses Messrs. Taylor, Stewart and Crossman, specifically addressed the issue of the number of NCRs. Mr. Taylor has

been present on-site as the resident inspector since 1978, a period which has encompassed almost all of the construction work on-site (the bulk of the electrical and piping installation and testing) except the early civil concrete construction phase. Taylor, Tr. 1731. Mr. Taylor unequivocally rejected CASE's assertion that the number of NCRs shows that the QA/AC program is not working. Id., Tr. 1712-1713. He testified that the sheer number of NCRs is evidence that the QA/QC program is working. Id., Tr. 1712. Mr. Taylor noted that a lack of NCRs would demonstrate an ineffective QA program. Id., Tr. 1713.

260. As part of their routine inspections at Comanche Peak, NRC inspectors reviewed logs of NCRs and other current documents to look for significant deficiencies that might be reportable pursuant to 10 C.F.R. § 50.55(e). Stewart, Tr. 1282; Crossman, Tr. 3021. Not every NCR or DDR is reportable pursuant to 10 C.F.R. § 50.55(e). The NRC inspectors, through a random sampling, seek to determine if significant deficiencies are being reported. Stewart, Tr. 1285.

261. NRC witness Mr. Crossman testified that although NRC inspectors do not examine each and every NCR, they do look at a sufficient number to detect trends indicative of problems. Crossman, Tr. 3021. Also, with respect to certain matters reported by the Applicants pursuant to 10 C.F.R. § 55.55(e), the NRC Staff reviews many of the NCRs relating to the particular matter. Id. In such instances, the NRC Staff also examines corrective actions and engineering evaluations to determine whether the Staff agrees with such evaluation. Id., Tr. 3022. Further, the Staff follows through by actually observing the work to verify the implementation of the engineering corrective action. Id. Mr. Crossman also stated

that the NRC Staff may track NCRs in certain areas to determine whether the numbers of NCRs is excessive, which could be indicative of problems in a particular activity. Id., Tr. 3023.

262. The evidence shows that the number of NCRs generated by the Comanche Peak QA/QC program does not demonstrate a failure of the QA/QC program.

(d) The Lobbin Report

263. CASE asserted that a study known as the "Lobbin Report" (Applicants' Exhibit 48) demonstrates that the Applicants' QA/QC program is inadequate. CASE did not, however, present any evidence supporting that assertion.

264. Applicants' witnesses Mr. Clements, TUGCO Vice President, Nuclear; Mr. Vega, Supervisor of TUGCO QA Services; Mr. Chapman, TUGCO Manager of QA; and Mr. Lobbin testified regarding the Lobbin Report. See Tr. 1445-1450, 2147-2200; "Testimony of B. R. Clements Regarding Reviews of Management Control Program and Activities of Texas Utilities Company Quality Assurance Organization" ("Clements," Applicants' Exhibit 118) and "Testimony of Frederick B. Lobbin Regarding Review of Management Control Program and Activities of Texas Utilities Generating Company QA Organization" ("Lobbin," Applicants' Exhibit 119).

265. As Vice President, Nuclear, Mr. Clements is the corporate officer responsible for assuring implementation of a Quality Assurance program for the design, construction and operational phases of Comanche Peak Steam Electric Station. Quality Control is an element of the Quality Assurance program. Clements (Applicants' Exhibit 118), at 1-2.

266. Mr. Clements' testimony describes the TUGCO management's initiative to commission a review of the TUGCO QA program by Mr. Lobbin and describes the response by the TUGCO QA organization to that review. Id., at 2. Mr. Lobbin's testimony describes the purpose and substance of the management review he conducted for TUGCO. Lobbin, at 1.

267. Mr. Lobbin is an independent engineering and management consultant involved primarily in areas related to design and evaluation of management systems for power plant engineering, construction, testing and operation. Vega, Tr. 1446; Clements, at 2; Lobbin, at 1. Mr. Lobbin was selected upon the recommendation of another utility for which he performed a similar evaluation. Clements, at 2, Tr. 2156.

268. The purpose of Mr. Lobbin's review was to provide Mr. Clements with an independent assessment of the effectiveness and implementation of management controls and activities within the TUGCO QA organization, with primary emphasis on the TUGCO QA Audit Group, and to identify for Mr. Clements any areas that might require further management evaluation. Vega, Tr. 1445; 1456, 2160, 2162; Clements, at 2; Lobbin, at 2-3.

269. The scope included a review of the effectiveness and implementation of management controls within the TUGCO QA organization. Mr. Lobbin did not evaluate the effectiveness and implementation of the QA programs for TUGCO's prime contractors, such as Gibbs & Hill, the Architect/Engineer, and Brown & Root, the Constructor. Clements, at 3; Lobbin, at 2, Tr. 2163-2164.

270. The management controls which were the subject of Mr. Lobbin's review included such areas of management concern as utilization of resources, timeliness of the resolution of deficiencies, establishment

of programmatic priorities, and staffing levels and qualifications. Clements, at 3.

271. Mr. Clements was not required by 10 C.F.R. Part 50, Appendix B, or any other regulation or commitment to have the review performed. Clements, at 3; Lobbin, at 2. He commissioned the report on his own initiative. Clements, at 3; Lobbin, at 2. He did not have reason to believe there were any deficiencies in the TUGCO QA organization. Clements, at 3.

272. Mr. Clements arranged for this review to be done because the CPSES project is approaching a unique phase involving construction activities, turnover activities, startup testing activities, and pre-operational testing activities being performed simultaneously. Clements, at 3-4.

273. Even before the review was conducted, the TUGCO QA staff had recognized the need for increased QA audit activity at this phase of the project, as reflected in an increased audit schedule and auditor manpower authorizations finalized prior to the review. Mr. Clements wanted an independent review to confirm this conclusion relative to the need for increased audit activity and auditor manpower resources. Clements, at 4, Tr. 2176. He also wanted Mr. Lobbin to identify any areas in which he felt Mr. Clements should seek further investigation in view of the combination of activities being conducted at this phase of the project. Clements, at 4.

274. Mr. Lobbin was given a general description of the priority documents, such as the Final Safety Analysis Report (FSAR), TUGCO's QA Program, the CPSES QA Plan, and CPSES organization, duties and responsi-

bilities. Mr. Lobbin was given unlimited access to all manuals, files and personnel. Id.

275. The review was not an audit. Clements, at 4; Lobbin, at 2. An audit, by definition, is carried out to verify compliance with specified aspects of the QA program. This was a management review to look at management aspects of the QA organization. Clements, at 4; Lobbin, at 2-3. The conclusions in Mr. Lobbin's review were based primarily on his personal judgment and experience. Lobbin, at 3.

276. Mr. Lobbin primarily reviewed documents and interviewed personnel in the TUGCO QA organization. Because of the large number of documents available, it was Mr. Lobbin's plan to limit the depth of examination in any one area, and to identify in his report any area that appeared to merit further investigation. It would then be up to Mr. Clements to assure the matter was evaluated in depth. Clements, at 4-5.

277. Mr. Lobbin spent two weeks on his review, the weeks of December 14-18, 1981, and January 4-8, 1982. Clements, at 5.

278. The methodology Mr. Lobbin used can best be described as "Directed Self Evaluation." When time is limited, as it was in this review, the consultant's primary objective is to identify those key areas which have apparent weakness. These areas are brought to management's attention who, in turn, direct the evaluated organization to respond to the findings. In the process of developing its response this organization is, in effect, forced to evaluate itself in areas which it might otherwise not have examined. The organization's response is provided to top management who is the final arbiter. Lobbin, at 3.

279. Mr. Lobbin utilized this approach in his review for two reasons. First, he believed that the evaluated organization should be a participant in the evaluation process for the results to have any real impact on that organization. Second, by involving the evaluated organization in the evaluation process the effective man-effort is increased far above what could be provided solely by the management consultant. Lobbin, at 3-4.

280. Some of the statements in Mr. Lobbin's report were overstatements or intentionally strong in order to put TUGCO's QA organization on the defensive. Lobbin, Tr. 2168. Since Mr. Lobbin did not have the facts to back up some of his concerns, he had to use strong language on purpose in order to ensure that he got a response from the people who were reviewing his report. Id., Tr. 2170.

281. The evaluation involved the review of QA department records, including audit reports, procedures and personnel records, and interviews with a number of QA department personnel. The time directly involved in these reviews and interviews was less than 60 hours. Lobbin, at 4.

282. Based on the results of Mr. Lobbin's reviews and discussions he identified specific areas which, in his opinion, could be improved. These findings were presented to Mr. Clements via Mr. Lobbin's report. Lobbin, at 4. Mr. Clements directed the QA department to respond to Mr. Lobbin's findings. TUGCO's response constituted the self-evaluation process. Mr. Lobbin's findings were the direction. Id.

283. Mr. Lobbin did not identify any significant problems during his review. Lobbin, at 5. Nothing identified in the review is representative of a significant breakdown in any portion of the quality

assurance program for design and construction of the Comanche Peak Steam Electric Station. Id.; Lobbin, Tr. 2169.

284. Moreover, his report does not identify any instance where the Applicants' program failed to comply with any regulatory requirement or commitment. Lobbin, a 5; Vega, Tr. 1446.

285. The findings in the report represent Mr. Lobbin's personal opinion or impression on what improvements can be made to enhance the TUGCO QA function. Lobbin, at 5-6, Tr. 2171. Some of his opinions are based on a comparison of activity at another nuclear project; others are based on direct observations; still others are mere statements of his concurrence with actions already planned by the Applicants prior to the review. Lobbin, at 5-6.

286. Mr. Clements reviewed the report prepared by Mr. Lobbin. He agreed that all matters identified by Mr. Lobbin as warranting additional attention should be evaluated. Clements, at 5.

287. In response to the report, Mr. Clements transmitted the report to the TUGCO QA organization and directed that it be reviewed, and a response prepared within thirty days. Clements, at 5; Lobbin, at 4. Mr. Lobbin's assignment ended with his report to Mr. Clements. Lobbin, at 4.

288. Each item in Mr. Lobbin's report was evaluated in depth and the results presented in a response document. (Applicants' Exhibit 49). This response presented the results of the in-depth investigation and the conclusion reached on each item. Where actions to address an item were in progress, the status was described. If additional measures were

deemed necessary, a description of the proposed actions, and an implementation schedule was included. Clements, at 5.

289. Mr. Clements was satisfied with the responses of the TUGCO QA organization. He reviewed the response and in some cases had detailed discussions with QA staff members to satisfy himself that the response was complete. Id., at 6.

290. Mr. Clements has a standing directive requiring quarterly progress reports until all items are closed. The first of these quarterly reports was submitted on May 24, 1982. (Applicants' Exhibit 50). All action items have been completed with two exceptions. TUGCO is still increasing its audit staff and is in the process of implementing its audit schedule. Clements, at 6.

291. Mr. Lobbin has read the response (Applicants' Exhibit 49) and the First Quarterly Report to Mr. Clements (Applicants' Exhibit 50). Lobbin, at 4. Mr. Lobbin stated that the Vice President, Nuclear should be the arbiter for the resolution of differences of opinion between the consultant and the QA department. He believes that to involve either party in more than one round of findings and responses is not productive and tends to result in a loss of benefits realized as part of the directed self evaluation process. Lobbin, at 4-5.

292. Mr. Lobbin's report states that the level of experience within the TUGCO QA organization in commercial nuclear plant design and construction QA experience is low. Lobbin, at 6. Mr. Lobbin meant that there were not to his knowledge people within the organization who have been through upcoming phases such as start-up. Lobbin, Tr. 2165. The Applicants acknowledged this in their response to his Report, but

indicated that they supplemented their staff with consultant expertise as is commonly done throughout the nuclear industry. Mr. Lobbin's statements should not, therefore, be taken as criticism of the overall level of QA staffing and experience for the Comanche Peak project since his evaluation did not include organizations, such as Gibbs & Hill, Brown & Root and other contractors whose personnel also perform QA functions for design and construction activities. Lobbin, at 6, Tr. 2174-2175; Vega, Tr. 1447.

293. Although Mr. Lobbin did not evaluate the QA programs of either Brown and Root or Gibbs & Hill (Lobbin, Tr. 2174), in his report, he stated that these contractors have a low level of commercial nuclear power plant design, construction and QA experience. Applicants' Exhibit 48, at 6, Lobbin, Tr. 2172-2173. In making this statement, Mr. Lobbin did not take into account the great deal of experience which Gibbs and Hill has in foreign work (Lobbin, Tr. 2174), nor did he actually evaluate either Brown & Root's or Gibbs & Hill's QA programs. Id. According to Mr. Lobbin, this statement just represents his general feeling, based on his knowledge of these contractor's experience in the field. Lobbin, Tr. 2174-2175.

294. Mr. Lobbin did not, at any time, verbally or in his report or otherwise, state or think that the QA staff at Comanche Peak is too small and too inexperienced to monitor properly potential safety problems in design and construction. Lobbin, at 6, Tr. 2185.

295. Mr. Lobbin's report states that the number of audits should be increased, especially of site engineering and construction activities. Lobbin, at 7. His statement is based on a comparison of audits performed

at another nuclear project with which he has had some association. As stated in his report, there may be a number of reasons for a difference in the number of audits performed for one project versus another. His primary objective in drawing this comparison was to put a lot of pressure on TUGCO (Lobbin, Tr. 2167-2168) and to ensure that the TUGCO QA Department undertake a self evaluation of its audit program in its response to his findings. Mr. Lobbin stated that certain of his findings regarding audits contain overstatements. Lobbin, Tr. 2168-2169. In its response and subsequent quarterly report the QA Department performed this directed self evaluation for Mr. Clements. Lobbin, at 7.

296. In response to the Lobbin report, TUGCO revised its procedures to require more input from key personnel in establishing an audit schedule. Vega, Tr. 2178. Also, TUGCO did a survey to determine how its number of audits compared with the rest of the nuclear industry and found the number of audits by TUGCO to be consistent. Id., Tr. 2178-2179. Also, TUGCO is now placing more emphasis not only on adverse findings in audits, but on the overall scope of the audit, to ensure that the overall function of the QA program being audited is examined. Chapman, Tr. 2180-2181.

297. Mr. Lobbin's report includes observations and recommendations in the area of surveillance. Appendix B is quite specific regarding the need for both an inspection and an audit program. TUGCO has established such programs. Appendix B does not, however, include specific requirements for surveillance. In fact, there is not even a uniform definition of the term surveillance throughout the nuclear industry. Lobbin, at 7-8.

298. The surveillance function, in Mr. Lobbin's experience, combines aspects of both inspection and audit and is used more as a

management tool to identify potential problem areas. As such, the surveillance function can provide an extra level of assurance beyond that required by regulation alone. Since TUGCO had taken the initiative to establish a surveillance program, Mr. Lobbin devoted some attention to the program to determine how it was organized and implemented. As a result, he identified certain aspects of the program which in his opinion could be improved. He included these as findings in order to direct the self evaluation of the program by TUGCO QA. Lobbin, at ›.

299. As a result of Mr. Lobbin's findings on the surveillance function, TUGCO is utilizing its surveillance personnel in a manner more consistent with their backgrounds. Vega, Tr. 2178, 2183-2184. Also, the surveillance group now reports to Mr. Tolson, the TUGCO Construction QA supervisor. Clements, Tr. 2177.

300. At no time, verbally or in his report or otherwise, did Mr. Lobbin express a concern for product adequacy or the adequacy of any work done at Comache Peak Electric Station. Lobbin, at 8, Tr. 2185.

301. Similarly, at no time, verbally or in his report or otherwise, did Mr. Lobbin recommend that Applicants go back and reverify the adequacy of any product or any work done prior to his evaluation. Id., at 8-9, Tr. 2185.

302. Mr. Lobbin stated his overall assessment of the TUGCO Quality Assurance Department for the Comanche Peak Steam Electric Station. Id., at 9. Specifically, based on his brief review, he found that overall the quality of the organization and activities of the TUGCO QA Department was good. Lobbin, at 8-9.

303. Mr. Clements does not have any areas of concern as to the adequacy of his QA organization after reviewing the report, the response, and the progress report. However, he will continue to monitor the measures to increase the audit staff and continue to review all audit reports. Clements, at 6.

304. Based on the findings of the report and the responses to the report, Mr. Clements has a high level of confidence that the TUGCO QA program is in compliance with 10 C.F.R. Part 50, Appendix B, and other applicable regulatory requirements and commitments. Id. He believes that the program is being implemented so as to assure Comanche Peak will be a safe and reliable source of electricity for the Applicants' customers. Id., at 7.

305. The Board finds on the basis of the record evidence that although the "Lobbin Report" recommends certain improvements in TUGCO's QA/QC program it does not demonstrate that TUGCO's QA/QC program fails to meet the requirements of 10 C.F.R. Part 50, Appendix B or is inadequate.

(e) John Junior Gates

306. As part of its direct case, Intervenor CASE called as a witness John Junior Gates (Tr. 2756), who testified that he was employed as a carpenter at Comanche Peak from November 24, 1976 to March 8, 1979. "Testimony of John Junior Gates witness for Intervenors CASE," CASE Exhibit 651 (Gates), at 5.

307. CASE offered Mr. Gates not as an expert witness, but as a lay person. Tr. 2755. Mr. Gates admitted that he does not possess a certificate of any kind in any field of construction (Gates, at 2-4) and

that he lacks qualification as an engineer or expert in any field of construction, other than carpentry. Id.

308. Mr. Gates' testimony consists of a rambling account of various alleged deficiencies in Comanche Peak, primarily that there is "sloppy work." See, e.g., Gates at 24, 26. Such alleged deficiencies include: concrete work at Comanche Peak is sloppy (Gates, at 24), as evidenced in a photograph of the Unit 1 and Unit 2 containment (Board Exhibit 4); formwork for concrete placements at Comanche Peak should have been built correctly the first time (Gates, at 20); there was no concern for the safety of the construction labor force, (Gates, at 30); drugs were used on-site (Gates, at 32); there were cost overruns (Gates, at 33); concrete was poured in the Safeguard Building #1 Tunnel which contained too much water and did not meet the slump test (Gates, at 36-37); drawings were used in construction which did not show changes (Gates, at 35); during inspections, QA/QC inspectors used documents not showing changes (Gates, at 35); the steel liner plate for Unit 1 is four inches out of plumb (Gates, at 37), and water stops were improperly installed and were not water tight (Gates, at 37).

309. The Board struck as irrelevant to Contention 5, those portions of Mr. Gates' testimony relating to the cost of construction (Tr. 2776-2777); the use of drugs on-site (Tr. 2923), and the safety of the construction labor force (Tr. 2923).

310. The only asserted basis for Mr. Gates' opinion regarding "sloppy workmanship" was visual observation. Gates, Tr. 2811. Mr. Gates admitted that inasmuch as he is not a professional engineer, he could not evaluate the structural significance of any of the defects he alleged.

Gates, Tr. 2767. Moreover, he admitted that there was no relationship between his allegations of sloppy formwork and quality assurance. Gates, Tr. 2808. He merely stated his desire that the work should be "up to par," meaning "first class" work in his judgment. Gates, Tr. 2884.

311. Although Mr. Gates' testimony is filled with references to many alleged defects in construction (See above), he could not provide any specific information, such as the date the incident giving rise to the alleged defect occurred, or even the location of the alleged defect. See, e.g., Gates, Tr. 2833-2835. For example, with respect to his allegation that changes to design documents were not recorded (Gates, at 35), he merely stated that such occurrence related to formwork and concrete work (Gates, Tr. 2835) in "the Safeguard 1 & 2 [buildings]," but he could not provide an elevation, or date or any other details. Id. Similarly, regarding his allegation that "QC inspectors . . . used documents not showing changes to do inspection" (Gates, at 35), he was not at all specific, beyond referring to "Safeguard 1 & Safeguard 2." Gates, Tr. 2837. Regarding his allegation that concrete poured in the Safeguard Building Tunnel did not meet the slump test, Mr. Gates couldn't recall having witnessed the test. Gates, Tr. 2844. It is the Board's judgment that these unsupported allegations do not constitute evidence of confirmed defects.

312. Mr. Gates characterized many of the alleged defects as not being "safety-related" (see e.g., Gates, at 11). However, he did not seem to understand the meaning of the term (Gates, Tr. 2840), despite having used it in his testimony. Gates, at 11. The numerous examples of deficiencies alleged by Mr. Gates stem from his mistaken belief that all

construction in a nuclear plant should be "safety-related." Gates, Tr. 2882.

313. Similarly, despite testifying about matters relating to quality assurance, Mr. Gates admitted that he was not qualified to judge the significance of these alleged defects. See, e.g., Gates, at 25, where Mr. Gates states that he is not qualified to judge the significance of honeycombing in concrete. Moreover, Mr. Gates did not even appear to understand the term quality assurance (Tr. 2819) and requested that the term "be made plain for him." Id. He even confused the term "quality assurance" with what he called "quality insurance." Gates, Tr. 2820, 2869.

314. Mr. Gates admitted that his concern that formwork for concrete should have been built correctly the first time was not "quality related." Gates, Tr. 2795. He acknowledged however, that rebuilding of the formwork showed a concern that work be done correctly. Gates, Tr. 2882. His objection to having to re-build formwork resulted from his questioning of the authority and judgment of the quality control engineer and his objections to having to conform his work to quality control standards. Gates, Tr. 2880.

315. In support of his allegation that there is "sloppy formwork" at Comanche Peak, Mr. Gates relied upon a photograph of the Units 1 and 2 containments appearing in a newspaper advertisement by TUGCO (Board Exhibit #4). According to Mr. Gates, there is too much variation in the vertical lines in the Unit 2 containment (Gates, Tr. 2799-2800) and there are bulges in the formwork, all of which show the concrete is out of plumb. Gates, Tr. 2803.

316. Mr. Gates accepted the representation that this picture was taken from a distance of several hundred feet from the reactor. Tr. 2800.

317. Applicants' witness, Mr. McGrane, a civil engineer and a registered professional engineer in the States of Texas, New Jersey and Ohio, testified that he saw nothing unusual in the photograph of the containments. McGrane, Tr. 2990-2991. Similarly, having viewed the containments first-hand, Mr. McGrane did not note any anomalies. Id., Tr. 2991. The appearance of the surface appeared to be straight Id., Tr. 2997. Mr. McGrane saw no large or unexpected bulges or protrusions or indications which would lead him to believe that the workmanship and quality of the structure was anything less than desired. Id., Tr. 2997. Mr. McGrane concluded that even assuming the existence of the anamolies in the containments described by Mr. Gates, such anamolies would not have any structural significance. McGrane, Tr. 2991-2992. According to Mr. McCrane, the vertical portions containment the walls are 4½ feet thick and have a very large quantity of No. 18 rebars. McGrane, Tr. 2992, 2994. The anomalies perceived by Mr. Gates are such a small proportion of this total mass that they would have no effect on the integrity of the structure. Id., Tr. 2992.

318. Concerning Mr. Gates' allegation that the steel liner plate for Unit 1 is four inches out of plumb, Mr. Gates was merely told that this was the case. Gates, Tr. 2848. He also alleged that this condition was not corrected. Gates, at 37.

319. According to Applicants' witness Mr. Merritt, in the initial installation of the steel liner plate by Chicago Bridge & Iron, in the first twenty feet, it was determined that the liner was out of alignment

by some four inches, which is still within the specification tolerances. Merritt, Tr. 2992. The work at that time was stopped for approximately three weeks, at which time the problem was corrected through realignment and the addition of stiff backs or "stiffeners," to the inside of the liner. These actions brought it back in line, at which time the installation of the liner was continued on through the dome. Merritt, Tr. 2992-2993. The final condition of the liner plate meets all of the requirements, as set out in Gibbs & Hill specification SS-9 as well as the American Concrete Institute, which prescribes the placing tolerances for concrete. Id., Tr. 2988.

320. Mr. Gates also alleged that water stops, which were installed in the "Safeguard 1, Safeguard 2 and Fuel Building", to prevent seepage of water, were not properly installed. Gates, at 37, Tr. 2845.

321. According to Applicants' witness Mr. Merritt, a water stop is a neoprene strip, from six to nine inches wide, (depending on the location of the joint), half of which is embedded in concrete on each side of a construction joint. Merritt, Tr. 2989. Under normal construction practice, nails are put in the water stops (tacking them to the forms), which does not affect the water tight integrity of the joints. Id., Tr. 2988-2989. Such nails are removed when the formwork to which the nails are attached is removed. Id., Tr. 2989. Because neoprene is a rubber-like material, the nail holes in the neoprene would tighten up and close when the nails are removed. Id. The nails were located at a distance away from the actual joint so that even if the nail holes remained, they would not impair the water tightness of the water stop.

Id., Tr. 2989-2990. There is no water seepage in any of the safety-related buildings at Comanche Peak. Merritt, Tr. 2993-2994.

322. The Board finds that the only confirmed construction anomaly pointed out by Mr. Gates (relating to the steel liner plate for Unit 1) was satisfactorily resolved and that that anomaly does not demonstrate a breakdown in the QA program since it was properly detected and corrected.

(f) Stanley G. Miles

323. CASE also called as a witness Stanley G. Miles (Tr. 2756), another ex-construction worker at Comanche Peak, who testified that he was employed at Comanche Peak from March 1977 to May 5, 1982. Testimony and Supplemental Testimony of Stanley G. Miles, CASE Exhibits 655 and 657, (Miles and Miles Supplemental). Miles (CASE Exhibit 655), at 1.

324. Mr. Miles is not an engineer (professional or otherwise) or a QC inspector. Miles, Tr. 2782-2784. CASE offered him not as an expert but as a lay witness. Tr. 2927. Despite having rendered a judgment in his direct testimony about the likelihood of an accident at Comanche Peak (Miles Suplemental, at 8), Mr. Miles admitted that he had no expertise to do so. Miles, Tr. 2784.

325. Mr. Miles' testimony contained numerous allegations relating to Comanche Peak, including the following examples: "shims were put in the polar crane as cosmetic iron working instead of as it should have been done" (Miles, at 11); there was a morale problem and "someone was getting rich off the job" (Miles, at 49); all of the top QC people and lot of engineers came from the South Texas Project (Miles Supplemental, at 1); rigging work posed dangers to workers (Miles Supplemental, at 2), and

several general foremen were fired from the mechanics shop for having welding rods there (Miles Supplemental, at 3).

326. The Board struck as irrelevant to Contention 5 those portions of Mr. Miles' testimony relating to morale problems and people "getting rich off the job," and his allegations concerning engineers at South Texas and rigging operations in the field. Tr. 2984.

327. Mr. Miles, stating that CASE's representative Mrs. Ellis had prepared his supplemental testimony (Tr. 2953-2955), modified and struck portions of his written testimony while on the stand after having studied the testimony. Miles, Tr. 2966-2970; 2974-2976. Although he testified that he felt a major accident at Comanche Peak is likely (Miles Supplemental, at 8), he changed his testimony to state that by "major accident," he meant "a leak into the reservoir which would go down Squaw Creek into the Brazos River into Lake Whitney and so on, a radioactive leak." (Tr. 2975-2976).

328. With respect to his allegations regarding the firing of several general foreman (Miles Supplemental, at 3), Mr. Miles explained that the foremen were fired because they were warehousing welding rods (a QC violation). Miles, Tr. 2971. He admitted that the foremen were supposed to control the rods and because of the QC violation, they were dismissed. Id. According to Mr. Miles, in this instance, the QC program worked. Id.

The Polar Crane

329. With regard to Mr. Miles' allegation pertaining to the polar crane, he testified that he related this allegation to Lanny Sinkin, one

of CASE's representatives, who later contacted the NRC about it. Miles, at 24-25, 42. According to Mr. Miles, the NRC initiated an investigation, based on the same allegation, and contacted him as part of the investigation. Miles, at 42.

330. Based on an NRC inspection conducted during June 1982 as a result of Mr. Miles' allegations, the NRC issued Notice of Violation 82-11, July 7, 1982 (NRC Staff Exhibit 148B). This inspection found a failure by the Applicants to perform inspections of installation activities related to the Unit 1 polar crane. Specifically, the NRC Senior Resident Inspector (construction), determined through interviews with craft labor supervision and the Comanche Peak Project Engineer for Civil/Structural activities that no inspections of the fabrication or installation of certain shims specified by Design Authorization 9872, dated March 30, 1981, and drawing SK-82032, revision 0, were performed. The shims specified by the Design Authorization and the drawing referenced above form a portion of the seismic restraint system for the containment polar crane. Taylor (NRC Staff Exhibit 13), at 74.

331. Applicants' witness Mr. Merritt also testified about Mr. Miles' allegation relating to the insertion of shims in the polar crane girder support bracket assemblies for Comanche Peak Unit 1. "Testimony of John T. Merritt, Jr., Regarding Placement of Shims in Polar Crane Girder Support Bracket Assemblies" ("Merritt," Applicants' Exhibit 127) at 1.

332. These construction activities, which were conducted in the summer of 1981, entailed the insertion of shims between two vertical plates - one vertical plate attached to the polar crane girder and the

other vertical plate attached to the support bracket assemblies. Merritt, at 1-2.

333. Shims are metal plates fabricated for insertion between two surfaces to maintain spacing or to transfer load. Merritt, at 2. The purpose of the polar crane shims is to maintain contact or near contact in the space remaining after installation of the ring girders for the polar crane, between the opposing vertical plates of the horizontal support brackets at the level of the top flange of the girders. The support brackets, with shims, transmit the horizontal loads from the polar crane to the containment wall. "Testimony of Kenneth L. Scheppele Regarding Structural Significance of Shims in Assemblies," ("Scheppele," Applicants' Exhibit 128), at 2.

334. In assembling any complex structure such as the polar crane support girders, it is necessary to make allowance for variation in dimensions and locations within specified tolerances. In the case of the polar crane support bracket assemblies, a 1/4 inch shim gap is shown on the construction drawing referenced below, for the purpose of providing a close fit between support girders and the connections to the containment wall following assembly of the polar crane support girders. Fitting a shim to fill this gap provides a means to account for slight variations in dimensions and locations. Merritt, at 2.

335. The shims in question are called for in the design document titled "R.B. Containment Liner Details, Sh. No. 5 Polar Crane Support Det's," Drawing No. 2323-S1-0515 (February 6, 1976). A copy of that drawing is attached to Mr. Merritt's testimony as Attachment A (Applicants' Exhibit 127A). On that drawing, Section 3-3 (at coordinates

1-E) and Section 4B-4B (at coordinates F-3) depict where the shims are called for by the design. These sections show that the shims are to be placed between the polar crane girders and the support brackets on the containment liner. Also attached to Mr. Merritt's testimony as Attachments B through G are photographs of the polar crane bracket assemblies and the shims in question. Merritt, at 3. (Applicants' Exhibits 127B through 127G).

336. As construction planning evolved, it became necessary to change the design for insertion of the shims. While the original design called for 1/4 inch shims, it was determined in the plant that a variety of shim thicknesses was necessary. Accordingly, a Design Change Authorization was initiated in March of 1981 to add additional shims. A copy of that Design Change Authorization (DCA No. 9872) dated March 30, 1981, is attached to Mr. Merritt's testimony as Attachment H (Applicants' Exhibit 127H). Attached to DCA No. 9872 is a drawing titled "Polar Crane Supports Additional Shim Requirements" (Drawing No. SK-82032), designated as Attachment I to Mr. Merritt's testimony (Applicants' Exhibit 127I). Merritt, at 3-4, Tr. 2276.

337. Applicants have conducted visual inspections of thirty-five shims that were inserted in the eighty-four polar crane girder support bracket assemblies for CPSES Unit 1. Of the shims inspected, they found that six shims had been clipped such that some "fingers" did not extend to the full depth specified in Drawing No. SK-82032. Merritt, at 4, Tr. 2276-2277. Applicants will inspect all eighty-four polar crane girder support bracket assemblies. Merritt, Tr. 2277.

338. The construction personnel who performed these work activities apparently clipped portions of the "fingers" to facilitate shim installation. For example, in some cases there was a slightly tapered gap between the veritical plates into which full length "fingers" could not be inserted. Merritt, at 4.

339. According to Mr. Merritt, there was no indication whatsoever of an attempt to mask the construction activities. The "fingers" that were clipped are outside "fingers" on the shims, and were readily visible to the inspectors. Furthermore, the shims were required by DCA 9872 to be tack welded in place to prevent movement. This has not been done. Mr. Merritt testified that the Applicants were fully aware of these remaining work activities as they were identified on the Master System Punchlist, Item No. 9X, System No. 8104. Merritt, at 5.

340. Applicants have developed a remedial plan to remove all shims in the polar crane girder support bracket assemblies and inspect each for conformance to design specifications. As to shims on which "fingers" were clipped, an evaluation will be performed by the CPSES Engineer and disposition on each shim will be effected, with the possible results being: (1) use as is, (2) replace the shim or (3) make additional modifications to the shim. Merritt, at 6.

341. The craft will be instructed to complete the work by the Construction Area Management Group. As part of completing their work, the craft will call for inspection to verify the installation. When the work is complete, the Completion Group will verify that the work has been done and all documentation requirements (including QA/QC

records) have been satisfied before removing item 9X on System 8104 from the Master System Punchlist. Merritt, at 6.

342. Applicants' witnesses Messrs. Merritt and Tolson testified that the shims as installed could not have gone undetected or uninspected. Merritt, at 5 and "Testimony of Ronald G. Tolson Regarding Quality Assurance/Quality Control at Comanche Peak," ("Tolson" Applicants' Exhibit 122), at 6, 8. According to Mr. Tolson, the Comanche Peak QA program has reliable, redundant measures that would have found the inconsistencies, probably in August or September, 1982 and would have led to correction of the matter by the construction and engineering organizations. Tolson, at 9.

343. According to the testimony of Applicants' witness Mr. Scheppele (Applicants' Exhibit 128), the maximum loads that must be transferred from the polar crane girder to the support brackets have been calculated. Scheppele, at 2. The maximum loads to the brackets occur when the plant is in the operating mode and a safe shutdown earthquake (SSE) occurs. Id., Tr. 2305. Depending on the locations of the vertical plate areas in contact either directly or through the shims, less than ten percent of the vertical plate area would be sufficient to transfer adequately the calculated loads. Id., at 3.

344. The Board finds that the Applicants have adopted appropriate measures for resolving the only confirmed construction deficiency (relating to the Unit 1 polar crane) raised by Mr. Miles and that such deficiency does not demonstrate a breakdown in the QA/QC program.

(g) Charles A. Atchison

345. Charles A. Atchison also testified as a witness for Intervenor CASE. See "Testimony of Charles A. Atchison, Witness for Intervenor CASE," ("Atchison," CASE Exhibit 650); "Supplementary Testimony of Charles A. Atchison Witness For Intervenor CASE (Citizens Association For Sound Energy)," ("Atchison Supplemental # 1," CASE Exhibit 656); "Supplemental Testimony of Charles A. Atchison Witness For Intervenor CASE (Citizens Association For Sound Energy)" ("Atchison Supplemental #2," CASE Exhibit 684).

346. Mr. Atchison was employed at Comanche Peak from February 27, 1979 until his termination on April 12, 1982. Atchison, at 5, 7-8. During his employment he held the positions of documentation specialist (February-September 1979) (Atchison at 9, Tr. 3214); training coordinator for Brown and Root (September 1979 - December 1981) (Id., at 8), and field inspector (December 1981 - April 12, 1982) (Atchison, at 9, Tr. 3214).

347. Although Mr. Atchison initally stated he was testifying as a "qualified layman" (Tr. 3208), the Board pointed out that there is no such thing. Id. It became clear that Mr. Atchison was tendered as a lay witness (Id.), who considered himself qualified as a QC inspector. Tr. 3209. In his last position at Comanche Peak, Mr. Atchison had a total of four (4) months experience in the field inspecting welds. Atchison, Tr. 3214-3215.

348. In both his direct testimony and through cross-examination, Mr. Atchison admitted that he made several misrepresentations in the job applications he submitted for employment at Comanche Peak. First, on page 2 of the application which Mr. Atchison submitted for employement

with Brown and Root on February 21, 1979 (Applicants' Exhibit 132), Mr. Atchison represented that he had obtained a "Liberal arts" degree from Tarrant County Junior College. In fact, Mr. Atchison did not at the time or later, obtain any such degree. Atchison, at 6; Tr. 3271, 3437.18/ Secondly, the resume that Mr. Atchison used for seeking employment at TUGCO in January 1982 (Applicants' Exhibit 133), contains a copy of Brown and Root's "Request for Verification of Education from Tarrant County Junior College" in which Mr. Atchison stated that he had attended Tarrant County Junior College from September 1968 to January 1971 and had obtained an "Associate-Liberal Arts" Degree. During cross-examination, Mr. Atchison admitted that he had altered the information provided by Tarrant County Junior College (Applicants' Exhibit 137). Atchison, Tr. 3340. In particular, the official verified copy of the Brown and Root Verification of Education Form, retained in the files of Tarrant County Junior College (Applicants' Exhibit 137) shows that the college stated in the "Comments/Exceptions" portion of the form, as follows: "Dates attended: 8/70-12/71; no degree obtained." Applicants' Exhibit 137. Mr. Atchison admitted having altered that document by deleting the word "no" and submitted the document, as altered, as part of his job application (Applicants' Exhibit 133). Atchison, Tr. 3340-3341.

349. Moreover, in his direct testimony (CASE Exhibit 650, at 2) and in the "resume" which Mr. Atchison submitted as an attachment to his

18/ Although Mr. Atchison testified that he did not prepare that resume (Atchison, at 6), during cross-examination, he admitted that he actually wrote the resume containing the misrepresentation. Atchison, Tr. 3437.

direct testimony (CASE Exhibit 650A), Mr. Atchison stated that he had earned one hundred and twenty-three (123) credits at Tarrant County Junior College and that he attended that school from 1968 to 1971. During cross-examination, Mr. Atchison admitted that he had only earned twenty-three (23) to twenty-seven (27) credits (Atchison, Tr. 3356) and that during the period September 1968 to January 1971 he only attended that school for three full semesters (Id.)

350. It is thus apparent that Mr. Atchison made misrepresentations not only in seeking employment at Comanche Peak but in testifying in this proceeding. See CASE Exhibit 650, at 2 and CASE 650A. Such misrepresentations are serious and the Board cannot ignore them in assessing Mr. Atchison's credibility.

351. Mr. Atchison's direct testimony consists of a disjointed account of his experiences while employed at Comanche Peak, including the circumstances surrounding his firing from Comanche Peak and later from Waterford[19/] (CASE Exhibit 684). Mr. Atchison's testimony contains numerous allegations of specific deficiencies in construction and in the QA/QC program at Comanche Peak, including the following:

- there is defective welding in pipe whip restraints and pipe moment restraints at Comanche Peak (see, e.g., Atchison at 22-26, 33, 39-43; Atchison Supplemental #1, at 2-7; Tr. 3458-3460);
- a Non-Conformance Report ("NCR") which Mr. Atchison initiated against Mr. Ronald Tolson, site QA manager, was not handled in the usual way because it never went to the site Authorized Nuclear Inspectors ("ANI") (Atchison, at 12-14);

19/ Waterford Steam Electric Station, Unit 3.

- there were two NCRs which Mr. Atchison prepared which never got into the system and disappeared (Atchison, at 22, Tr. 3242-3245, 3247);
- there is warpage in pipe whip restraints supplied by NPSI (Atchison, at 33, Tr. 3458-3460);
- downhill welding was used at Comanche Peak, in violation of AWS (American Welding Society) and ASME (American Society of Mechanical Engineers) criteria (Atchison, at 48);
- TUGCO employees borrowed Mr. Atchison's penetrant testing ("PT") kit (Atchison, at 51-52);
- hazardous lubricant was used at Comanche Peak (Atchison, at 55);
- the ratio of craft workers to inspectors is too high (Atchison, at 57);
- nothing happened about an NCR which Mr. Atchison submitted regarding cold springing of pipe (Atchison, at 63);
- workers at the plant were not told what their rights were, particularly regarding the Occupational Safety and Health Administration ("OSHA") (Atchison, at 66);
- NRC investigations at Comanche Peak are ineffective (Atchison, at 67)
- a shipment of weld rods by air freight to Comanche Peak arrived three days after a shipment by truck (Atchison Supplemental #1, at 11, Tr. 3364-3367);
- a purchase order came through to pay Westinghouse $1,500,000 for a trailer (Atchison Supplemental #1, at 11; Tr. 3367-3369).

352. The Board struck as irrelevant to Contention 5 those portions of Mr. Atchison's initial direct and supplemental testimony relating to occupational safety and health (Tr. 3464); the shipment of weld rods by air freight, and the cost of a trailer (Tr. 3465). With respect to Mr. Atchison's further supplemental testimony (CASE Exhibit 684), regard-

ing his firing by Tompkins-Beckwith at Waterford, the Board struck all portions of that testimony except for lines 1-8. Tr. 4455-4456.

353. The Board will first consider Mr. Atchison's specific allegations and will then consider his allegations regarding his termination. See, e.g. Atchison, at 53.

(i) Defects In Pipe Moment Restraints and Pipe Whip Restraints

354. Applicants' provided testimony of Messrs. Tolson, (Applicants' Exhibit 122); Chapman (Applicants' Exhibit 123); Boren (Applicants' Exhibit 124); Barber (Applicants' Exhibit 125), and McGrane (Applicants' Exhibit 126) regarding the general subject of defects in moment limiting restraints and pipe whip restraints. Also see Tr. 2201-2307.

355. The portion of the Staff's direct testimony (Staff Exhibit 13) concerning welding discusses certain inspection and investigation findings relating to Mr. Atchison's allegations regarding welding on pipe moment restraints and pipe whip restraints. (NRC Staff Exhibit 13, at 55, 59, 60, 63, 66-68). In addition, the supplemental testimony of Staff witnesses Robert G. Taylor and Donald D. Driskill also addressed these allegations. See "Testimony of NRC Staff Members Robert G. Taylor and Donald D. Driskill Regarding NRC Staff Investigation & Inspection Findings on Allegations by Charles Atchison," ("Taylor/Driskill," NRC Staff Exhibit 197, at 1-8) and NRC Staff Exhibits 113, 114, 115, 116, 117, 123, 124, 125, 126, and 199.

356. Pipe moment restraints are steel structures attached between a building structure and a pipe to limit bending motion of the pipe. Taylor/Driskill Supplemental, at 7; Barber, Tr. 2245; McGrane, Tr. 2246-2247.

357. A pipe whip restraint is a high-energy device which is designed to restrain a pipe if it breaks. Taylor/Driskill Supplemental, at 7; Barber, Tr. 2245.

358. According to Applicants' witness Mr. Tolson, the QA/QC Program for Comanche Peak Steam Electric Station (CPSES) found non-conforming conditions with respect to moment limiting restraints. Tolson (Applicants' Exhibit 122), at 1-2. An apparent violation of Non-Destructive Examination (NDE) procedures was identified during a routine audit/source inspection by Brown & Root QA at the facility of Chicago Bridge & Iron Company (CB&I) in Salt Lake City, Utah in July, 1980. Tolson, at 2. The Brown & Root Audit Deficiency Report is included as Applicants' Exhibit 122A.

359. As stated in Applicant's Exhibit 122A, the auditor considered the surface condition of some welds to be unacceptable for proper interpretation of NDE processes such as magnetic particle examination. Id.

360. The NRC received allegations from a Brown & Root employee (Mr. Atchison) regarding these components and investigated the matter. The results of its investigation and related inspection are documented in NRC Inspection Report 80-22 (NRC Staff Exhibit 123) and in NRC Inspection Report 80-20 (Applicants' Exhibit 44A; NRC Staff Exhibit 125). Id.; Taylor, at 66-67. The NRC found that the apparent violation of Nondestructive Examination (NDE) procedures was an infraction, and issued a Notice of Violation (NRC Notice of Violation 80-20, Applicants' Exhibit 122B; NRC Staff Exhibit 124) to that effect. Tolson, at 2. As set forth in greater detail in the September 24, 1980 letter from the NRC to the Applicants (Applicants' Exhibit 122B, NRC Staff Exhibit 124), four of the moment limiting restraints inspected revealed surface

conditions considered by the NRC to render the NDE results to be unacceptable. Tolson, at 2; Barber (Applicants' Exhibit 125), at 4.

361. As documented in Applicants' October 20, 1980 letter to the NRC (Applicants' Exhibit 122C; NRC Staff Exhibit 126) all moment limiting restraints received at the site on the relevant purchase order through the date of NRC Report 80-20 were identified by Applicants as potentially non-conforming. All of these restraints were reinspected for conformance with specified requirements, and rework of relevant welding indications was effected to achieve full compliance with these requirements. In addition, CPSES QA personnel were assigned to CB&I Salt Lake City to perform in-process and release inspections of shipments from CB&I. Tolson, at 4. Taylor, at 67-68. A follow-up inspection was made by the NRC inspector and documented in NRC Inspection Report 80-23 (NRC Staff Exhibit 114). Taylor, at 68.

362. In that follow-up inspection NRC inspector Mr. Taylor observed a portion of the reinspection and rework effort in regard to the components and was satisfied that it was being accomplished in a sound manner. He also interviewed the Brown and Root employee (Mr. Atchison) who was the basis for the allegation and found that he had been assigned to monitor the vendor's onsite activities involving the reinspection and rework effort. Mr. Atchison indicated that he was satisfied with the effort. Mr. Taylor had no further questions regarding the components themselves. NRC Staff Exhibit 114, at 2-3. According to that report, the licensee's commitment in regard to future inspections will be

evaluated during future routine inspections. The item of non-compliance was therefore considered closed. Id.; Tolson at 3.

363. Mr. Taylor addressed the potential safety consequences of the unacceptable weld surfaces (Inspection Report 80-20, NRC Staff Exhibit 125). He testified that the weld surfaces were unacceptable from the standpoint that a meaningful non-destructive examination could not have been done as required by the engineering drawings for the components. The rework observed by Mr. Taylor was almost entirely that of grinding the surface more smooth. Some rewelding had to be done after the initial grinding since in some cases the welds became undersized as a result of the grinding. Taylor, at 68.

364. The NRC investigation (NRC Inspection Report 80-22, NRC Staff Exhibit 123) included a related allegation by Mr. Atchison that discrepancies identified in vendor-manufactured components were being waived by direction of a TUGCO QA auditor in order that equipment could be shipped to Comanche Peak. Taylor, 66-67; Taylor/Driskill, at 4. That investigation identified one instance where a vendor, Chicago Bridge and Iron ("CB&I"), of Salt Lake City, Utah, manufactured components known as pipe moment restraints which were disapproved for shipment to Comanche Peak, due to deficient manufacturing, but were waived and shipped to Comanche Peak. The allegation that a TUGCO QA auditor had instructed that waivers be prepared on non-conforming components was not substantiated. Details on this aspect of the investigation are provided on pp. 6-9 of Inspection Report 80-22 (NRC Staff Exhibit 123). Taylor, 66-67; Taylor/Driskill, at 5.

365. As part of this investigation, Mr. Driskill personally interviewed Mr. Atchison as well as other persons identified as having

knowledge of the matters which were raised in the allegation. Taylor/Driskill, at 3, F; Driskill, Tr. 2478-2479. Mr. Driskill also reviewed Comanche Peak B&R QA procedures relating to both vendor audit and vendor-manufactured components release inspections. Taylor/Driskill, at 5.

366. As part of a later investigation of subsequent allegations made by Mr. Atchison to the NRC (Investigation Report 82-10/82-05, NRC Staff Exhibit 199), the NRC inspector was informed of the steps that had been initiated at the CB&I factory to verify that proper QC was being performed. The NRC inspector acknowledged the actions that had been taken and indicated that whether these steps were adequate to ensure that the restraints arriving at CPSES were capable of performing their intended function would remain unresolved until the NRC had the opportunity to review this program in detail. NRC Staff Exhibit 199, at 11-12.

367. Inspection Report 80-23 (NRC Staff Exhibit 114), also discusses the finding that certain welds in other components supplied by CB&I, known as "pipe whip restraints", were not full penetration, as required. Taylor, at 55; Taylor/Driskill, at 3-4.

368. This item of non-compliance was identified to the Applicants by an official notification on October 16, 1980 (NRC Staff Exhibit 113). The Applicants responded to the finding in Inspection Report 80-23 in their letter dated November 12, 1980 (NRC Staff Exhibit 115). The response indicated that several of the components involved would be reworked to revised engineering standards and that future inspections would be based on engineering drawings rather than shop fabrication drawings. The Applicants' response was acknowledged by Region IV letter dated December 23, 1980 (NRC Staff Exhibit 116). Taylor, at 59-60. NRC

Inspection Report 81-14 (NRC Staff Exhibit 117) documented the follow-up inspection of the finding in NRC Inspection Report 80-23 regarding the lack of full penetration welds. The NRC inspector reported that the components requiring rework had been reworked and that other components were satisfactory from a strength standpoint even though they were not reworked based on engineering calculations. Taylor, at 63.

369. Applicants' witness Mr. Tolson testified, with respect to pipe whip restraints, that the QA/QC program for CPSES also found non-conforming conditions. Tolson, at 3. In March 1982, a shipment of Unit 2 pipe restraints was received at CPSES and released to construction for installation. Following sandblasting and application of a light protective coating, construction personnel expressed concern to a CPSES QC Inspector (Mr. Atchison) that certain welds on the restraints may not conform fully to the accept/reject criteria (AWS D1.1) at the site. The Inspector reinspected four of the restraint assemblies to these criteria, marked areas on the assemblies that he considered to be rejectable indications, and documented the inspection results on NCR No. M-82-0296, Rev. 0 (Applicants' Exhibit 122D). Tolson, at 3-4.

370. Following issuance of the NCR, the fabricator, CB&I, advised the Applicants that the design drawing permitted the fabrication of pipe whip restraint assemblies either to Subsection NF of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 or to the requirements of AWS D1.1, which was utilized by the CPSES QC Inspector. The fabricator advised that it had elected to use Subsection NF of the ASME Code. Reinspection of the assemblies in accordance with Subsection NF was accomplished by the CPSES Lead QC Inspector of the responsible QC group,

and the initial NCR was revised and reissued as NCR No. M-82-0296, Rev. 1 (Applicants' Exhibits 122F through 122L). Tolson, at 4. These restraints were reworked, as necessary, and installed in the plant. Several types of indications involved are illustrated in the seven photographs attached to Mr. Tolson's testimony as Attachments F through L (Applicants' Exhibits 122 F through L). Id.

371. Following consultation with the TUGCO Manager of Quality Assurance, Mr. Tolson directed a complete reinspection of fifty-two other restraint assemblies that had been received prior to or subsequent to the four restraints documented in Applicants' Exhibit 122D. These inspections revealed welding indications similar to the initial inspections but to a lesser degree. Other corrective actions are described in the testimony of Messrs. Chapman and Boren (Applicants' Exhibits 123 and 124, respectively).

372. The length of weld on all fifty-six restraints which were reinspected totals 55,000 linear inches. Tolson, Tr. 2265. The accumulation of welding indications on all fifty-six restraints represents substantially less than one percent of that 55,000 inches of weld. Tolson, Tr. 2264-2265.

373. Restraints received after implementation of such corrective action were determined to be free of relevant indications and the reinspection efforts at the site were therefore concluded. All of these inspections have been documented in accordance with the CPSES QA Program, and have been or will be resolved in accordance with approved procedures. Tolson, at 5.

374. As indicated in the testimony of Messrs. Chapman and Boren, Mr. Chapman, as TUGCO QA Manager, was aware of the apparent non-conforming condition of the pipe whip restraints fabricated by CB&I. Chapman (Applicants' Exhibit 123), at 1-2.

375. A trip to the CB&I facility was made in March 1982. The purpose of the trip was to meet with CB&I plant management during one of TUGCO's inspections and to develop improvements in their inspection system. Mr. Boren, the TUGCO QA Vendor Compliance Supervisor, represented Mr. Chapman in discussions with CB&I management. TUGCO QA believed that such improvements were necessary because TUGCO inspectors were discovering too many non-conformances during source release inspections which should have been discovered by CB&I's own QA program. The results of that meeting are described in Mr. Boren's testimony (Applicants' Exhibit 124). Chapman, at 3.

376. In addition, in May, 1982, Mr. Chapman called the President of CB&I and requested that he direct his personal attention to improving the inspection effectiveness in the Salt Lake City facility. Chapman, at 3. As a result of the conversation, CB&I upper management requested a May 7 meeting at the CPSES job site. At this meeting, CB&I committed to certain corrective measures. These measures are discussed in Mr. Boren's testimony. Id.

377. According to Mr. Chapman, these corrective measures have been effective and the quality of CB&I's product is greatly improved. Id. TUGCO QA inspectors continue to perform release inspections of all CB&I restraints at Salt Lake City prior to shipment. Id., at 4.

378. Testimony by Messrs. Boren and Barber, two of Applicant's witnesses with seven and twenty-five years experience in inspection of welding (and thus greater experience and qualifications in welding inspection than Mr. Atchison) (Boren, at 1, Tr. 2286; Barber, at 1-2, Tr. 2286), revealed that they did not consider all of the weld indications noted by Mr. Atchison in CB&I pipe whip and moment restraints to be rejectable. Specifically, in the spring of 1982, three TUGCO QA Vendor Compliance Inspectors examined several CB&I pipe whip restraints previously inspected by Mr. Atchison. Boren, at 2. This inspection revealed that most of the marked indications were clearly acceptable and should not have been designated as non-conforming. Id., at 3. The inspectors felt that the welding was acceptable and that the inspection had been performed to some unknown, arbitrary criteria. Id. Mr. Boren personally inspected two restraints at CPSES on May 7, 1982. Id., at 4. Of the approximately eighty spot areas marked by the QC inspector as rejectable, Mr. Boren found all but eight of them to be acceptable per the ASME Code. Id., at 4.

379. Similarly, with respect to the alleged discrepancies in the CB&I moment restraints discussed by Mr. Tolson, Mr. Barber testified that his personal examination of a shipment of moment restraints indicated that not all of the welded surfaces were properly prepared for the NDE examination but that with proper surface preparation and use of proper examination techniques, the entire welded surface of the restraints could be examined. Barber, at 3.

380. Mr. Barber was also personally involved in the QA/QC activities which lead to the identification of certain weld indications in pipe whip restraints supplied by CB&I for Unit 2. Id., at 5. On

April 2, 1982, Mr. Barber personally examined two restraints on which Mr. Atchison had marked numerous indications, determining that they were welded and inspected in accordance with the requirements of ASME Code, Section III, Inspection NF. Id., at 5-6. Mr. Barber determined that most of the marked areas did not violate the NF requirements and should not have been designated as non-conforming. Id., at 6. On another routine visit in April, 1982, Mr. Barber was requested to observe the inspection results on approximately nine additional restraints. He found that the majority of the marked indications were not in violation of the NF criteria and were more of a cosmetic type indication. Id., at 7.

381. Applicants' witness Mr. McGrane, Assistant Chief Structural Engineer at Gibbs & Hill, testified about the structural significance of weld indications in pipe whip restraints. McGrane, Applicants' Exhibit 126, at 1. He testified that he examined documentation of potential weld deficiencies in the pipe whip restraint structures discussed in the testimony of Messrs. Tolson, Barber and Boren. Id.

382. On the basis of his examination of pipe whip restraint design calculations he concluded that there is sufficient design conservatism to offset those weld indications that have been identified in the documentation he examined, even assuming that the weld indications reflect actual deficiencies and even assuming that those deficiencies were not repaired. Id., at 2, Tr. 2297-2300.

383. Mr. McGrane presented three bases for his conclusions. First, weld sizes are determined on the basis of the maximum load at any one point along the weld and welds are of uniform size throughout their length, even if not so required by calculated stresses. Second, fillet

welds joining two right angle surfaces are generally oversized. Calculations indicate a minimum reserve capacity of 5 percent in the fillet welds, and for most weldments the reserve capacity is larger. Third, since the minimum yield strength of all weld material exceeds the minimum yield strength of the base material (i.e., the materials being joined together), for full penetration welds the joint strengths exceed the strength of the materials being joined. Therefore, it is Mr. McGrane's professional opinion based on the foregoing that if the indications of potential weld deficiencies (as reported in the above referenced documentation) had gone undetected, the safety function of the affected pipe whip restraints would not have been impaired. Id., at 2-3.

384. The Board finds, on the basis of the record evidence, that there were certain deficiencies in Applicants' QA/QC program with respect to welding of pipe moment restraints and pipe whip restraints. However, even if such deficiencies had gone undetected, they would not have affected the safety function of the affected components. Nor do such deficiencies demonstrate a breakdown in the QA/QC program.

(ii) An NCR Which Mr. Atchison Prepared Against Mr. Tolson Was Not Handled Properly

385. Mr. Atchison testified about an NCR he initiated (NCR No. M-2685, CASE Exhibit 662) against Mr. Tolson, the TUGCO site QA manager for using Mr. Atchison's keys (in his position as the B&R training coordinator, Mr. Atchison was custodian of the certification files) to gain access to files in the QA vault. Atchison, at 12-17. According to Mr. Atchison, Mr. Tolson got the files in response to the request of Mr. Taylor, the

NRC Resident Inspector. Id., at 13-14. Mr. Atchison testified that this NCR was not handled properly because it did not go through the "ANI's" (Authorized Nuclear Inspectors). Id., at 14.

386. In cross-examination, Mr. Atchison specifically stated that ANI review is applicable to all NCR's (Atchison, Tr. 3222), although he admitted that he was incorrect in that statement as for non-ASME matters, the ANI need not sign off on the disposition of NCRs. Atchison, Tr. 3222-3223. He also stated that after issuance of the NCR, procedures were changed. Atchison, at 16.

387. Applicants' witness Mr. Tolson testified that, in response to the request of Mr. Taylor, the NRC Resident Inspector, for some certification files for electrical QC inspectors, Mr. Tolson's staff provided Mr. Taylor with access to the training records. Tolson (Applicants' Exhibit #141), Tr. 4659-4660. Applicants are required to grant the NRC Resident Inspector access to the training records. Id., Tr. 4660. As Mr. Tolson testified and as NCR No. M-2685, CASE Exhibit #662 explicitly states, following issuance of the NCR, the procedures were not amended with respect to obtaining access to the training records. Id.

388. The Brown and Root QA manual requires ANI approval of NCR dispositions on ASME code items. However, as the subject of this NCR was the training files for electrical QC inspectors, a non-ASME activity, Mr. Atchison incorrectly indicated that this matter concerned ASME Code items. Id.

389. The Board finds that the evidence does not support Mr. Atchison's allegation that this NCR was not handled properly.

(iii) Two NCRs Which Mr. Atchison Prepared Never Got Into The System

390. According to Mr. Atchison, there were two instances in which NCR's he submitted never got into the system. It is not very clear from Mr. Atchison's written testimony that there were two such instances. See, e.g. Atchison, at 22. However, through cross-examination and questioning by the Board, it became clear that Mr. Atchison was referring to two such instances (Atchison, Tr. 3244) which related to vendor-supplied items. Atchison, Tr. 3247.

391. Mr. Atchison testified that in January 1982 he left an NCR on the non-ASME NCR coordinator's desk, which was not the normal, routine procedure for submitting NCRs. Atchison, Tr. 3242-3243. He stated that she had left early for the day and he left it on her desk to have a number assigned to it. Atchison, Tr. 3243. According to Mr. Atchison, this NCR never entered into the system nor was a number assigned to it (Id.); it simply disappeared. Id. This NCR related to the areas of vendor-supplied items which Mr. Atchison had been noting throughout the pipe whip restraint area. Atchison, Tr. 3247.

392. With respect to this instance (January 1982), Mr. Atchison admitted during cross-examination that there were written procedures governing the submittal of NCRs, (Atchison, Tr. 3390-3391), which required that a QC inspector submit a draft NCR to his immediate supervisor before submitting the NCR to the NCR coordinator. Atchison, Tr. 3391-3392. He stated that this procedure was not always followed on the site. Id., Tr. 3392. Mr. Atchison stated that in this instance, he did not submit the NCR to his immediate supervisor for review and initialing before the NCR was submitted to the NCR coordinator. Atchison, Tr. 3393. He merely

took the NCR to her for assignment of a number and, as she was not in her office, he left it on her desk. Atchison, Tr. 3393. He went back several times to find it and could not do so. Id. Mr. Atchison explained that his failure to determine what happened to this NCR was due to his workload. Id.

393. This allegation was one of a second series of allegations Mr. Atchison made to the NRC in April 1982, after his termination. Taylor/Driskill (NRC Staff Exhibit 197), at 8-9. NRC investigator Mr. Driskill participated in an investigation of these allegations, whose purpose was to assess the validity of the allegations and their impact on safety-related components and systems. Id. The investigation is documented in NRC Inspection Report 50-445/82-10 and 50-446/82-05 (NRC Staff Exhibit 199). Id. As part of that investigation, Mr. Driskill personally interviewed Mr. Atchison, the Quality Control ("QC") Mechanical Equipment Supervisor and the non-American Society of Mechanical Engineers ("non-ASME") NCR Coordinator. Id., at 9. He also reviewed the non-ASME log. Id.

394. Investigation of the allegation that a NCR written by Mr. Atchison in about January 1982 concerning weld defects in a number of CB&I pipe whip restraints was never documented or resolved, disclosed that the NCR prepared by him was not properly submitted by him; therefore it was never entered into the corrective action system. Interview of Mr. Atchison's immediate supervisor disclosed that the topic of defective welds in some CB&I restraints was discussed with Mr. Atchison; however, a draft NCR was never submitted to the supervisor for approval and formal submittal. Interview of the non-ASME NCR Coordinator disclosed that Mr. Atchison did discuss with her the format of an NCR regarding CB&I

restraints, during approximately January 1982, however, it was never submitted. Details regarding this aspect of the investigation are contained in Inspection Report 82-10/82-05, pp. 3-4 (NRC Staff Exhibit 199). Id., at 10.

395. As Mr. Driskill noted, the instructions for submitting NCRs include three criteria that have to be met in submitting NCRs, none of which were met in this instance. Driskill, Tr. 2545. According to NRC Staff witness Mr. Taylor, these instructions, which are contained in a site-procedure that is distributed throughout the site, are the subject of considerable site training. Taylor, Tr. 2549. Mr. Driskill found no evidence in the investigation to support the allegation that Mr. Atchison had in fact submitted the NCR (Driskill, Tr. 2633), although he did find evidence that Mr. Atchison had prepared the NCR. Driskill, Tr. 2633-2634.

396. The second instance occurred in March 1982. Atchison, at 22. This NCR (CASE Exhibit 660A), dated March 20, 1982 related to pipe whip restraints at location A22, Reactor Building 1, Pressurizer Tank Room, Elevation 882. Atchison, at 22; Tr. 3253. According to Mr. Atchison, while he was doing his inspection and completing the liquid penetrant test, he noticed a number of defects in the vendor-supplied welds that did not meet AWS (American Welding Society) D1.1 criteria pursuant to which he was inspecting. Atchison, at 23. According to Mr. Atchison, these defects were obvious through the paint. Id.

397. Mr. Atchison stated that he prepared the draft NCR, attached drawings to it and turned it in. Id. Mr. Atchison testified that he took the NCR to his supervisor and it was reviewed and the matter was never entered into the office NCR log. Tr. 3249. Mr. Atchison

testified that the supervisor of his immediate supervisor did not find the welds he had noted to be defective. Atchison, at 23. According to Mr. Atchison, after he issued this NCR, he requested clarification from Bill Hartshorn, a QA engineer, concerning the scope of Mr. Atchison's inspections. Atchison, Tr. 3253.

398. As was pointed out to Mr. Atchison during cross-examination, the date on the document he submitted requesting clarification (CASE Exhibit 650W) was March 19, 1982, one day before he prepared the NCR in question. Tr. 3249-3252. In response to Mr. Atchison's request for clarification, his supervisor, Mr. Randall Smith, clarified in writing the scope of the inspections Mr. Atchison was to perform. CASE Exhibit 650W. Mr. Atchison was satisfied with this instruction (Atchison, at 26, Tr. 3263), which he considered to broaden the scope of the surveillance inspection authorized by instruction numbers QI-QP-11.14.3, Revision 3 (Applicants' Exhibit 131). He stated his belief that this instruction authorized him to conduct random surveillances throughout the plant (Atchison, Tr. 3231) and did not constrain him in the issuance of NCRs. Atchison, Tr. 3236.

399. Applicants' witness Mr. Brandt testified regarding the procedure for obtaining NCRs (Applicants' Exhibit 141), Tr. 4657-4659. According to Mr. Brandt, Mr. Atchison did not follow these procedures when he wrote the two draft NCRs which Mr. Atchison alleges were not properly entered into the system. Id., Tr. 4658-4659. Mr. Brandt testified that had Mr. Atchison correctly followed these procedures, the NCRs would have been entered into the system and would have been properly addressed. Id., Tr. 4659. He further stated that in neither case that Mr. Atchison

referenced did he properly obtain an NCR number. Id. In addition, Mr. Atchison failed to apply a hold tag with respect to the matter reflected in CASE Exhibit 660A. Id. Also, Mr. Brandt testified that when Mr. Atchison identified unacceptable indications on the pipe whip restraints which ultimately became NCR M-82-00296 (Applicants' Exhibits 122D and 122E), he neither applied a hold tag nor obtained a number upon noting the non-conforming condition; Id. This was indicated by the fact that when Mr. Brandt first observed the indications which Mr. Atchison had marked on March 18, 1982, no hold tags were affixed to any of the assemblies. Id.

400. Mr. Brandt testified that he directed the issuance of an NCR with regard to the matters raised in the drawings by Mr. Atchison relating to the 822' pressurizer tank room (the second instance to which Mr. Atchison referred). Id., Tr. 4664. When Intervenor CASE introduced Mr. Atchison's draft NCR on the subject (CASE Exhibit 660A) at the licensing hearings in late July 1982, Mr. Brandt directed his staff to have the craft remove the paint from the subject welds, as Mr. Atchison had never had it removed. Id. When the paint was removed, Mr. Brandt and Mr. C.C. Randall went jointly to inspect the welding that Mr. Atchison had questioned. Id. The results of that inspection are documented in NCR-M-82-01236, Revision 1 (Applicants' Exhibit 141D). Id. Mr. Brandt testified that previous action had not been taken because Mr. Atchison had failed to submit an NCR on the matter in accordance with procedures, and consequently Applicants were unaware of its existence. Id. As a result of Mr. Brandt's reinspection, as he had surmised, the linear indications

identified by Mr. Atchison were indeed cracks in the paint. The porosity which Mr. Atchison had identified was not rejectable. Id., Tr. 4665.

401. The Board finds on the basis of the record evidence that Mr. Atchison did not follow the procedures for submitting NCRs when he wrote the two NCRs which Mr. Atchison alleged were not properly entered into the system.

(iv) Warpage In Pipe Whip Restraints Supplied By NPSI

402. In his written direct testimony, Mr. Atchison testified that pipe whip restraints supplied by NPSI are "in as bad if not worse shape than the CB&I pipe whip restraints." Atchison, at 33. According to Mr. Atchison, there are "basic defects in welding and . . . a lot more problems with fit-ups because they were not built to design specification". Id. He alleged that "another part of that same problem is the welding procedure specification No. 10,046. It did not address the welding configuration that's being used." Id. He stated that "I discussed this problem with our engineering and they had problems with it and wanted welding engineering to adopt ASIC [sic] configuration to that procedure. In effect, they're saying we've got something similar but not exactly like it but we'll use it anyway." Id.

403. During questioning of Mr. Atchison by the Board at the hearing, Judge Cole asked Mr. Atchison whether "there are any physical defects at Comanche Peak Nuclear Power Station of any nuclear safety significance that you have personal knowledge of that have not been corrected?" Tr. 3458. In response to the question, Mr. Atchison stated that in NPSI pipe whip restraints, which:

> have not been fully looked at or investigated, the 588 material that is used on those during the welding process has extreme warpage on it. The angle provided for a fit-up on the main steam lines for these was not addressed in Welding Procedure WPS-10047 at that site. The configurations of these, and the warpage of the pre-welded, or the vendor welded items, are as bad and in some cases worse than those supplied on the CB&I pipe whip restraints... Atchison, Tr. 3458-3459.

404. Mr. Atchison indicated his concern that "as a utility payer, an inspector on the jobsite if I'm going to pay for a Cadillac, I want a Cadillac, I don't want a Ford..." Id., Tr. 3459.

405. According to Mr. Atchison, these "items" were the subject of his investigations or inspections, but he was terminated and no NCR was generated on the vendor defects of the welds on the NPSI pipe whip restraints. Atchison, Tr. 3459-3460.

406. Applicants' witness Mr. Brandt testified regarding Mr. Atchison's allegations regarding warpage of NPSI pipe whip restraints. Brandt, et al. (Applicants' Exhibit 141), Tr. 4684. According to Mr. Brandt, Mr. Atchison was referring to crushable bumpers provided by NPSI which attach to pipe whip restraint structures. These bumpers consist of a piece of plate fillet welded to a piece of pipe to provide a crushable structure to absorb impact loads in the event of a pipe whipping. The warpage that Mr. Atchison is referring to is warpage caused by the welding process. However, the warpage is within the acceptable limits established by the Code and has absolutely no structural or safety significance. In performing a weld in this configuration warpage in the plate would be expected. Id.

407. The Board finds that the evidence does not support Mr. Atchison's allegation that there are defects in NPSI pipe whip restraints having structural or safety significance.

(v) Downhill Welding

408. Mr. Atchison alleged that downhill welding was used at Comanche Peak in violation of AWS (American Welding Society) and ASME (American Society of Mechanical Engineers) criteria. Atchison, at 48. Specifically, Mr. Atchison alleged that this involved a CB&I pipe whip restraint on which Mr. Atchison marked indications because of downhill welding. Id., Tr. 3376.

409. Mr. Atchison was not familiar with the process with which the weld was made (Atchison, Tr. 3376), although he assumed it was "shielded arc metal because of the weld spots that occurred in some areas. . ." Id.

410. According to Applicants' witness Mr. Brandt, ASME Section III, Subsection NF is the fabrication code applicable to the welding of the CB&I pipe whip restraints which were addressed by Mr. Atchison on p. 48 of his direct testimony. Brandt, Tr. 4677. ASME Section IX is the appropriate provision specified in ASME Section III, Section NF for the qualification of welding procedures. Id. According to Mr. Brandt, ASME Section IX allows downhill welding. Id.

411. CB&I has a welding procedure qualified in accordance with ASME Section IX, and approved by Brown & Root, for fabrication of pipe whip restraints which permits downhill welding. Id. WPS-E7018-82105 (Applicants' Exhibit 141H) states that vertical welds shall be made uphill

except that the root and cover pass may be made either uphill or downhill. Id., Tr. 4677-4678. The only welding which was inspected by Mr. Atchison was the cover pass. The procedure clearly indicates that the direction of vertical welding may be in either direction for the cover pass. Mr. Brandt therefore concluded that no deficiency existed. Id., Tr. 4678.

412. WPS-10046 is a Brown & Root welding procedure qualified in accordance with AWS D1.1. WPS-10046 does not apply to the welding by CB&I of the pipe whip restraints identified by Mr. Atchison on page 48 of his testimony. WPS-10046 is only applicable for Brown & Root welding performed in accordance with AWS D1.1. Id., Tr. 4678.

413. In Mr. Brandt's opinion, Mr. Atchison's allegation that downhill welding was performed in violation of specified requirements is an additional indication that Mr. Atchison neither took the time to properly investigate the appropriate acceptance criteria for vendor-supplied welds nor properly identified his concerns in accordance with established procedures. Id.

414. The Board finds that the record does not demonstrate the existence of any construction deficiencies as a result of downhill welding.

(vi) Use of PT Kit

415. Mr. Atchison testified that TUGCO employees borrowed his penetrant testing ("PT") kit but that he did not know whether or not such employees had used his PT kit. Atchison, at 51-52.

416. Applicants' witness, Mr. Brandt, testified that to his knowledge, no one other than properly-certified individuals performed NDE

at Comanche Peak. Brandt, Tr. 4679. He testified that he is familiar with the allegation made by Mr. Atchison in his testimony at page 51 (CASE Exhibit 650) regarding the unauthorized use of penetrant testing (PT) kits by Ebasco inspectors. Id. Mr. Atchison testified in the Department of Labor (DOL) hearings[20/] on August 19-21, that he was referring to Mr. David Brown and Mr. C. C. Randall as the two inspectors who borrowed his PT kit. Id. Mr. Randall is employed by Ebasco Services and Mr. Brown is employed by United Engineers and Constructors. Both are under contract to Texas Utilities. These inspectors are not certified to perform PT at Comanche Peak. Id.

417. There is no evidence that indicates Mr. Brown or Mr. Randall has ever performed code-required PT at Comanche Peak. Id, Tr. 4680. A records check found no indication that PT has been performed by either of these inspectors to this date. Id. The only instance in which they would have borrowed a PT kit would have been to obtain prerequisite training for certification. Id.

418. During cross-examination, Mr. Brandt stated that after his written direct testimony was prepared, he checked with Messrs. Brown and Randall and both indicated that they had performed PT (Brandt, Tr. 4624), but it was for information only in one case and in another case, was just to get some training. Id., Tr. 4624-4625.

20/ As discussed infra, the Department of Labor ("DOL") held hearings on Brown and Root's appeal of DOL's initial determination that Brown and Root's termination of Mr. Atchison violated Section 210(a) of the Energy Reorganization Act.

419. The Board finds that the record evidence does not demonstrate the existence of any construction deficiencies as a result of performance of PT.

(vii) Cable Lubricant

420. Mr. Atchison alleged that "hazardous" lubricant was used at Comanche Peak. Atchison, at 55. Specifically, he alleged that "a lubricant used to assist in the pulling of electrical conduit or a cable through electrical conduit was found to be flammable..." Id. According to Mr. Atchison, the laboratory test performed on the conduit, in which it did not burn, did not accurately simulate field conditions. He stated that after the laboratory test was performed, he tested a piece of the cable coated with lubricant and it burned. Mr. Atchison admitted that he neither knew what the laboratory test conditions were, nor had he witnessed the tests and that he is not an expert in the testing of lubricants. Id., at 56, Tr. 3327.

421. Applicants' witness, Mr. Tolson, testified regarding this allegation. Tolson, Tr. 4683. The cable lubricant in question (Yellow 77) tended to support combustion upon dehydration. An NCR was prepared (Applicants' Exhibit 141I). It was also reported to the NRC as a potential significant deficiency under 10 CFR §50.55(e). The lubricant was tested by a cable manufacturer in a dehydrated state and the results were found to be satisfactory. It is currently in use at Comanche Peak. The test at the cable manufacturer was witnessed by the NRC Resident Inspector, Mr. Taylor. This matter has been removed from the potential significant deficiency list. Id.

422. The Board finds on the basis of the record evidence that use of the cable lubricant in question (Yellow 77) is not hazardous and that there are no construction deficiences as a result of the use of that cable lubricant.

(viii) Inspector: Craft Ratio

423. Mr. Atchison alleged that the ratio of craft to inspectors is too high. Atchison, at 57. He testified that there was approximately one inspector for each two hundred craftsmen (Atchison, Tr. 3328), and a total of one hundred fifty-seven inspectors. Id.

424. Mr. Atchison admitted that if these numbers were correct, there would be over thirty-one thousand craft (31,000), when in fact he stated there are about four thousand (4,000) craft. Id., Tr. 3328-3329.

425. Mr. Vurpillat, Power Group QA Manager for Brown & Root, testified regarding the inspector to craft ratio at Comanche Peak. Vurpillat, Tr. 4693-4694.

426. The total number of craft workers in both safety and non-safety areas and QC Inspectors during the years 1979, 1980, 1981 and 1982 are approximately as follows:

	Craft	QC Inspectors
1979	3,000	160
1980	2,900	140
1981	2,600	130
1982	2,750	240

Tr. 4693.

427. All of the inspectors and no more than 75% of the craft personnel were involved in safety-related activities. Vurpillat, Tr. 4693. The ratios of safety-related craft to inspectors were 14:1 in 1979, 16:1 in 1980, 15:1 in 1981 and 9:1 in 1982. Id.

428. In Mr. Vurpillat's opinion, a ratio of 15:1 or 20:1 is one which would generally be considered an appropriate average for safety-related nuclear site construction. However, many considerations could contribute to a higher ratio being more than satisfactory to accomplish the job well. Likewise, a lower ratio may be appropriate in certain circumstances (such as the current heavy schedule of hydrostatic testing at the site). However, the amount of inspection is not a function of the number of inspectors. Inspection points and hold points are set regardless of the size of the inspection force. The number of inspectors merely influences how efficiently the job can be done and how little time is wasted waiting for inspections to be performed. Id., Tr. 4694.

429. The Board finds on the basis of the record evidence Mr. Atchison's allegation regarding the ratio of inspectors to craft lacks basis and that the ratio of craft to inspectors is an appropriate average for safety-related nuclear site construction.

(ix) Cold Springing of Pipe

430. Mr. Atchison alleged that during repairs of compartment No. 3 of the reactor coolant pump, two lines were disengaged and "cold sprung (which means they were forced together initially) and they made a loud audible noise." Atchison, at 63. According to Mr. Atchison, it is a violation of ASME and site procedures for piping to be cold sprung. Id.

He alleged that nothing happened about the NCR on this, that it was never submitted to QA/QC for final "buy-off" and that it eventually went to Mr. Tolson, who "had no problem with it." Id.

431. Applicants' witness, Mr. Brandt, testified that the situation Mr. Atchison described was identified in NCR M-3215SR1 (Applicants' Exhibit 141W). The piping was part of the component cooling water system, not the reactor coolant system described by Mr. Atchison. Brandt, Tr. 4691.

432. The term "cold spring" is used to describe the use of external forces to move pipe into alignment in order to make a connection. The use of external forces induces stresses within the pipe and its adjoining welds. Id., Tr. 4691-4692.

433. This matter was not correctly documented by Mr. Atchison. He originally identified it on the incorrect form (NDER). NDERs were used at that time to document minimum wall violations, arc strikes and base metal defects. All other defects should have been on an NCR. Id., Tr. 4692.

434. The issue of the cold sprung pipe has been resolved. The NCR was dispositioned to separate the flanged connection, cut out two welds, reweld and then rejoin the flanges. The work has been completed and the NCR is closed. Mr. Atchison's statement that the NCR was dispositioned use as is and that it was never submitted to QA/QC for "final buy-off" is incorrect. Id.

435. Applicants' witness, Mr. Tolson, disagreed with Mr. Atchison's statement on page 63 of his testimony that the NCR was submitted to Mr. Tolson and that he had no problem with it. Tolson, Tr. 4692.

Mr. Tolson stated that he is not routinely involved with disposition and subsequent "close-out" of Brown & Root NCRs, such as the NCR written on this matter. Processing of Brown & Root NCRs is described in Brown & Root's Quality Assurance Manual, and does not require his involvement. Id.

436. The Board finds that the record evidence refutes Mr. Atchison's allegation regarding this NCR and that the issue of the cold sprung pipe has been properly resolved.

(x) NRC Investigations at Comanche Peak Are Ineffective

437. Mr. Atchison alleged that although he respects the NRC Resident Inspector, Mr. Taylor, the investigations which have been made by the NRC are not sufficient to assure the plant's safety. Atchison, at 44, 67. According to Mr. Atchison, "lack of manpower, fear by employees who come forward to report defects--fear they will lose their job, Mr. Taylor's familiarity with the plant management top personnel means that the NRC has been ineffective on the jobsite and lacks sufficient manpower to cover the site." Id.

438. Mr. Atchison admitted that he has no experience as an investigator. Atchison, Tr. 3394. Although he questioned the NRC manpower on-site, he also admitted that Mr. Taylor is not the only inspector on-site. Id., Tr. 3396-3397.

439. He explained his statement that Mr. Taylor was overly familiar with "top management" by alleging that there was a willingness to overlook things or to give information to the Applicants so they could correct deficiencies before detrimental reports were written. Id., Tr. 3395.

According to Mr. Atchison, by "top management," he meant "Mr. Ron Tolson and the top administrative staff for TUGCO at Comanche Peak job-site." Id., Tr. 3397.

440. Mr. Atchison acknowledged that these are the people on site ultimately responsible for the proper functioning of the QA/QC program, to whom the NRC findings of any deficiencies should be brought. Id., Tr. 3398. In addition, Mr. Atchison could not cite any specific instances in which the NRC resident inspector was ineffective on the job site because of familiarity with top management. Id., Tr. 3399. Even though Mr. Atchison claimed that the NRC was ineffective, he actually cited an instance in his supplemental testimony regarding an NRC backfit program required of TUGCO which he admitted showed they were effective. Atchison Supplemental #1, at 3, Tr. 3400-3401.

441. The Board finds that the record evidence does not support Mr. Atchison's allegation that NRC investigations at Comanche Peak are ineffective.

(xi) Mr. Atchison's Firing

442. In addition to the specific allegations of defects in Comanche Peak construction discussed above, Mr. Atchison also alleged that he was improperly fired from his position as a QC inspector at Comanche Peak. Mr. Atchison claimed that he was fired for "over-inspecting" and submitting NCRs. See, e.g., Atchison at 53.

443. The Applicants' and Staff presented testimony addressing Mr. Atchison's termination. See Applicants' testimony of Brandt, et al.,

Tr. 4660-4677 and Staff testimony of Taylor/Driskill (NRC Staff Exhibit 197), at 8-11 and NRC Staff Exhibit 199.

444. The question whether Mr. Atchison's rights were adversely affected as a result of his termination such that he is entitled to remedies must be left to other forums (i.e., the U.S. Department of Labor). Specifically, on April 15, 1982, Mr. Atchison filed a complaint with the U.S. Department of Labor. Staff Exhibit 199, at 9; Atchison, at 59-60. On May 14, 1982, the Department of Labor, Fort Worth, Texas, determined that "the weight of the evidence to date indicates that [he] was a protected employee engaging in a protected activity within the ambit of the Energy Reorganization Act and that discrimination, as defined and prohibited by the statute, was a factor in the actions which comprised his complaint." Staff Exhibit 199, at 9; Atchison, at 59-60; CASE Exhibit No. 650B.

445. As the Applicants and Staff further noted, at the time that testimony was presented on Mr. Atchison's firing, a hearing was pending before the Department of Labor regarding Mr. Atchison's firing. Taylor/Driskill (NRC Staff Exhibit 197), at 11; Brandt, Tr. 4677. As a result of Brown & Root's appeal of this determination, a hearing was held on August 19, 20 and 21, 1982 before a Department of Labor Administrative Law Judge (ALJ). Brandt, Tr. 4677. A "Recommended Decision" by the Department of Labor ALJ dated December 3, 1982, upheld the initial determination and concluded that "Brown and Root terminated complainant because he was engaged in protected activities within the Act's [Energy Reorganization Act, Section 210(a)], and that respondent [Brown and Root] violated the Act and regulations in so acting." CASE Exhibit 738, at 26.

446. The Board has pursued the question of Mr. Atchison's firing to the extent that it is relevant here, i.e., whether his firing indicates an intent on the part of Applicants' QA/QC program to prevent the detection of deficiencies in plant construction.

447. Mr. Atchison testified that he believes he was terminated because he and Mr. Brandt had "a previous run-in while I was training coordinator, and there were some animosities still there." Atchison, at 53. In testifying about his firing Mr. Atchison referred to an NCR he initiated stating that "All PT & MT performed an non-ASME items are invalid due to the fact that Non-ASME group has no training or certification program for qualifying inspections. Inspections were performed by ASME trained and certified inspector, not non-ASME qualified personnel." Atchison, at 52; Applicants' Exhibit 135. According to Mr. Atchison, he obtained an NCR number for this NCR off the door of the NCR coordinator's office because she had gone home early. Id., at 52. He testified that he left a note with the NCR stating "# taken not issued yet. Open to pow. wow. on subject. Black or white no grey area's Chuck. [sic]" Id., Tr. 3306, Applicants' Exhibit 135. He stated that he wanted to "pow-wow" because if there was a problem, "then we would have a massive backfit program . . . to determine if other TUGCO employees had performed liquid penetrant stuff on production work." Id., at 52-53. The NCR was voided. Id., at 53. Mr. Atchison testified that he wanted "to have a pow wow with them because ... we had a procedure that said Brown & Root would perform all the NDE's." Texas Utilities didn't have any method of saying, we're going to train our people for PT or MP Examination to handle the non-ASME criteria or anything to this effect." Id.

According to Mr. Atchison if this was determined by upper management to be a problem, "then we would have a massive backfit program..." He denied that this request for a "pow wow" had anything to do with his request for a raise. Id. Mr. Atchison also alleged that the "real reason" he was fired was "because I found some defects that they didn't want discovered which would cause them to go back and do a backfit "program." Atchison, at 53. The defects to which he was referring were in the welding on pipe whip restraints. Id., at 52-53.

448. The NRC Staff testimony of Messrs. Taylor and Driskill regarding NRC inspections and investigations of allegations by Mr. Atchison addressed Mr. Driskill's investigation of allegations Mr. Atchison made to the NRC after he was fired, including the allegation that he was fired for submitting NCRs. Taylor/Driskill/NRC Staff Exhibit 197, at 9.

449. In April and May, 1982, Mr. Driskill participated in the investigation of allegations made in April 1982 to the NRC by Mr. Atchison. The purpose of that investigation was to assess the validity of these allegations and their impact on safety-related components and systems. This investigation is documented in NRC Inspection Report 50-445/82-10 and 50-446/82-05 (NRC Staff Exhibit 199). Taylor/Driskill, at 8. In his capacity as the Comanche Peak Resident Inspector, Mr. Taylor assisted in that investigation, although he did not participate directly in it. Id.

450. The investigation covered the two allegations made by Mr. Atchison to the NRC. The allegation concerning the draft NCR which Mr. Atchison claimed he submitted in January 1982, has been discussed previously. The investigation also covered Mr. Atchison's allegation that in late March 1982 and early April 1982 he submitted several NCRs

which brought him into disfavor with site QA management and resulted in his termination. Id. As part of the investigation of this allegation, Mr. Driskill interviewed Mr. Atchison, the non-ASME QA/QC supervisors, and the non-ASME NCR Coordinator. Id., at 10. This investigation also covered other matters which have previously been discussed relating to CBI-supplied pipe moment restraints and pipe whip restraints. Id.

451. In early 1982, Mr. Atchison, who was working as the ASME training coordinator on the QA department's staff, was transferred at his own request to the field QC group. NRC Staff Exhibit 199, at 6. Subsequently, Mr. Atchison was transferred to the mechanical inspection staff where he was primarily responsible for the inspection of pipe whip restraints. Id. That function was transferred to the non-ASME mechanical group under TUGCO, and Mr. Atchison was transferred to that group where he continued to conduct inspection of restraints. Id.. Mr. Atchison was in that position at the time of his termination. Id.

452. The NRC Staff's investigation of the allegation that Mr. Atchison was terminated for writing NCRs did not substantiate or refute the allegation. Taylor/Driskill, at 10-11; Driskill, Tr. 2585, 2663-2664. A review of NCRs submitted by Mr. Atchison, to which he attributed his termination, disclosed that all were appropriately documented and each was pending review or in final disposition. Taylor/Driskill, at 11. Interviews of TUGCO and Brown and Root (B&R) Managers disclosed that they had been dissatisfied with Mr. Atchison's performance (Id.; Tr. 2582-2583); however these concerns were never documented nor was he counseled regarding his performance. Taylor/Driskill, at 11. Interviews with two former supervisors disclosed that they considered his performance "good" to

"excellent". The B&R Quality Assurance (QA) Manager stated Mr. Atchison had requested transfer from the TUGCO non-ASME Quality Control (QC) staff at the same time that TUGCO management had decided that it was necessary to transfer Mr. Atchison out of their group back to ASME QC staff. Id.

453. Since no position was open in this group, Mr. Atchison was terminated. Details regarding this aspect of the investigation are provided on pp. 4-9 of Inspection Report 82-10/82-05 (NRC Staff Exhibit 199). Taylor/Driskill, at 11.

454. Mr. Driskill testified that his investigation did not uncover any evidence that Mr. Atchison was forbidden to write NCRs. Driskill, Tr. 2577. According to Mr. Driskill, the evidence was simply inconclusive to support or refute Mr. Atchison's allegation about the reasons for his firing. Id., Tr. 2585. Mr. Driskill stated that it was possible that Mr. Atchison may have been terminated for writing NCRs (Id., Tr.2665 2666), but that he could not prove that charge on the evidence he had. Id., Tr. 2666. Mr. Driskill did not believe that Mr. Atchison was fired for overinspecting; nor did he believe that Mr. Atchison's firing was due to poor performance. Id., Tr. 2699. According to Mr. Driskill, there was just a difference of opinion regarding Mr. Atchison's performance. Id. Further, Mr. Driskill did not believe that any of the individuals he interviewed as part of his investigation lied. Id., Tr. 2688. Mr. Driskill also testified that he thinks Mr. Atchison is "relatively credible," if Mr. Atchison has the information. Id.

455. In the Staff's investigation report (NRC Staff Exhibit 199) as well as another investigation report cited in the testimony of Messrs. Taylor and Driskill (NRC Staff Exhibit 123), the Staff reported the findings of its investigations concerning the allegations made by an

individual designated only as "A", and reported on related information provided by persons identified only by letter designation and job title. NRC Staff Exhibits 123 and 199. During cross-examination of Messrs. Taylor and Driskill, the Board ordered the Staff to identify these individuals. Tr. 2479, 2481, 2484. See also Tr. 2488, 2492-93, 2499, 3046.[21/] The Staff, asserting the informer's privilege, stated that it was "not at this time going to disclose the identities of the individuals designated by letter." Tr. 2484. See also Tr. 3047-3049, 3051-3054, 3060. The Board denied the Staff's request that the Licensing Board stay its order so that prompt review by the Appeal Board could be obtained. Tr. 3072-3073. Staff counsel produced expurgated copies of signed witness statements and voluntarily produced the investigator's notes, from which names and other identifying information had been deleted. Tr. 2965, 3041, 3159; see Tr. 2750, 3042.

456. After the Staff declined to identify its informants, the Licensing Board permitted Mr. Tolson, an employee of Applicants, to testify as to his understanding of the identities of the Staff's informants in Staff Exhibits 123 and 199--in most cases with an asserted 100% certainty. Tr. 2506, 2508-13. Mr. Tolson's identifications were later substantially confirmed by Mr. Atchison (Tr. 3442-56), and the Licensing Board admitted

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21/ The Licensing Board Chairman also ordered the Staff to disclose the names of individuals identified by letter in Investigation Report 81-12, admitted into evidence as Staff Exhibit 178, concerning the unrelated allegations made by another individual (Tr. 3558). However, at the hearing sessions held in September 1982, the Chairman effectively rescinded his order compelling the disclosure of those identities, on the grounds that confidentiality appeared to have been requested by and granted to all of those individuals (Tr. 4064, 4068).

into evidence over Staff's objection a copy of Mr. Atchison's unexpurgated signed witness statement. Tr. 3466-68. That witness statement (CASE Exhibit 663) identifies by name many of the persons named by Mr. Atchison to the Staff's investigators. Further identification of the informants in Staff Exhibit 199 was inadvertently made by Staff witnesses during sustained cross-examination and Board questioning. Tr. 2573, 2593, 2698. Applicants' witnesses, Messrs. Brandt, Purdy, Smith and Tolson, testifying in rebuttal to Mr. Atchison's allegations, later identified themselves. Tr. 4673-4674.22/

457. Applicants' witness Mr. Brandt, who is certified Level III mechanical inspector by TUGCO and Ebasco Services (Brandt, Tr. 4663-4664), discussed his involvement in the termination of Mr. Atchison. Brandt, Tr. 4661. Mr. Brandt transferred Mr. Atchison back to Mr. Gordon Purdy who was the Brown & Root Site QA Manager, as his services were no longer needed by Mr. Brandt's group. Because Mr. Atchison was a Brown & Root employee, Mr. Brandt had no authority to terminate him. Mr. Brandt determined that he no longer required Mr. Atchison's services in his group because as he had previously discussed on two occasions with

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22/ The Licensing Board subsequently issued an "Order to Show Cause" against the Staff (August 4, 1982), to which the Staff responded on August 24, 1982 (See "NRC Staff's Response to Order to Show Cause and Motion for Reconsideration"). On September 30, 1982, the Licensing Board issued an Order denying the Staff's request for reconsideration and on October 8, 1982 the Staff timely filed with the Appeal Board exceptions to the Licensing Board's Order. On November 17, 1982, the Staff its brief in support of its exceptions as well as a motion for directed certification. Both the Applicants and CASE filed briefs in response to the Staff's appeal. The matter is currently before the Appeal Board. (See the Appeal Board's December 27, 1982 letter and December 30, 1982 Order.)

Mr. Purdy, he deemed Mr. Atchison's level of competence as a QC welding inspector to be unsatisfactory. Brandt, Tr. 4661. On two occasions Mr. Brandt personally observed Mr. Atchison's work. On both occasions he considered that work to be unsatisfactory and to demonstrate that Mr. Atchison lacked the capability and judgment necessary for performing visual welding inspections. Id. Mr. Brandt described both instances where he observed Mr. Atchison's inspection activities. Brandt, Tr. 4661. Mr. Brandt's second opportunity to observe Mr. Atchison's inspection activities occurred in March 1982, when Mr. Brandt received a draft copy of NCR M-82-00296 (Applicants' Exhibits 122D and 122E). Brandt, Tr. 4662.

458. Applicants' witness Mr. Smith, recalled the matters described by Mr. Brandt and his discussions with Mr. Smith regarding Mr. Atchison's inspection capabilities. Smith, Tr. 4663. Mr. Smith accompanied Mr. Brandt on his reinspection efforts in the first situation (Reactor 1, 822' pressurizer tank room), and participated directly in the reinspection of NCR M-8200296. Id.

459. Following each of those reinspections, Mr. Smith advised Mr. Atchison of his inability to judge the acceptability of porosity. In Mr. Smith's opinion Mr. Atchison was aware that Mr. Brandt considered Mr. Atchison's performance to be unsatisfactory. Id.

460. Mr. Brandt testified that he told Mr. Gordon Purdy in February 1982 when he transferred Mr. Atchison to Mr. Brandt's group that Mr. Brandt doubted Mr. Atchison's performance capabilities. Mr. Brandt based his views upon a discussion he had with Mr. Atchison in late 1981 regarding visual inspection. Thereafter, in March 1982, after the incident on the 822' pressurizer tank room, Mr. Brandt verbally discussed the matter with

Mr. Purdy. After that, on April 8, 1982, Mr. Brandt advised Mr. Purdy that he would not be requiring Mr. Atchison's services much longer. Mr. Brandt also discussed with Mr. Purdy that he had been advised that Mr. Atchison was seeking a transfer back to the ASME group. Brandt, Tr. 4666.

461. The Applicants witnesses testifying about Mr. Atchison's termination (Messrs. Brandt, Tolson, Smith, Purdy and Vurpillat) testified that on April 12, 1982 (the day of Mr. Atchison's termination), they were not aware that Mr. Atchison had made allegations to or threats to contact the NRC regarding what he perceived as construction deficiencies at Comanche Peak. Brandt, et al. Tr. 4666-4667.

462. Mr. Brandt also testified regarding the chronology of events which occurred on the morning of April 12, 1982, the day Mr. Atchison was terminated. Brandt, Tr. 4667-4669.

463. On the morning of April 12, Mr. Brandt first received a request from Mr. Atchison to Mr. Smith requesting permission to seek employment elsewhere at Comanche Peak. Mr. Brandt verbally approved the request. Mr. Brandt then received a request from Mr. Atchison to Mr. Smith seeking permission to transfer back to the ASME Mechanical Equipment Group. Mr. Brandt approved the request, provided that Mr. Purdy concurred with the transfer, and signed the request. Mr. Brandt later received Non-Conformance Report M-82-00361 with a note attached (Applicants' Exhibit 135). He perceived the note as an attempt by Mr. Atchison to leverage a promotion which he had been seeking the previous week. Id., Tr. 4667. As the note indicated that the NCR number had been assigned, yet the NCR had not been issued, Mr. Brandt interpreted Mr. Atchison's statement "open to pow. wow. [sic] on subject . . . black or white no

grey area's [sic]" as an attempt to arbitrate the issuance of the NCR. The implication in his mind was that Mr. Atchison was offering not to process the NCR in return for approval of a promotion request which had been previously submitted to and rejected by Mr. Brandt. The magnitude of the deficiency described by Mr. Atchison would have been tremendous had it been valid It was not valid, and Mr. Brandt voided it. Brandt, Tr. 4667-4668; Applicants' Exhibit 135. The impression this left with Mr. Brandt was that Mr. Atchison had picked a major matter so that his bargaining position would be enhanced. Id., Tr. 4668.

464. Mr. Brandt then telephoned his supervisor, Mr. Tolson, described the situation, and sought direction. Mr. Tolson advised Mr. Brandt to bring Mr. Gordon Purdy to a meeting in Mr. Tolson's office. When the three met, Mr. Tolson reviewed the "pow wow" note and concluded independently that the note was an attempt by Mr. Atchison to use the NCR as a lever. Id.

465. When Mr. Brandt called Mr. Purdy, he came into Mr. Tolson's office promptly. When he arrived, Mr. Brandt handed Mr. Purdy a copy of the "pow wow" note and the attached NCR. That was the first time that Mr. Purdy had seen the note or NCR or known of their contents. He studied the note and NCR and concluded independently that Mr. Atchison was "after something," and that this was an attempt to obtain it through abnormal means. Purdy, Tr. 4668-4669. Mr. Brandt advised Mr. Purdy at that time that his organization no longer required the services of Mr. Atchison and that Mr. Atchison was being returned to Mr. Purdy in his capacity as Brown & Root Site QA Manager. Mr. Brandt then sent a memoran-

dum to Mr. Purdy confirming their verbal discussion (CASE Exhibit 650C). Purdy, Tr. 4668-4669.

466. Mr. Purdy testified that he was not directed by anyone to terminate Mr. Atchison. As Brown & Root Site QA Manager, Mr. Purdy has the sole responsibility and authority to terminate his employees. Mr. Purdy made the decision to terminate Mr. Atchison himself.

467. When Mr. Purdy received the memorandum from Mr. Brandt (CASE Exhibit 650C), he contacted four of his supervisory personnel (Messrs. Sanders, Ragan, Opelski and Leigh) to determine whether any of them had a position for which Mr. Atchison was qualified. Each stated that he did not. Mr. Purdy then contacted Mr. Vurpillat, his supervisor in Houston, to determine whether he had a position for which Mr. Atchison was qualified. Mr. Vurpillat stated that he did not. Mr. Purdy then determined that Mr. Atchison's services were no longer required by Brown & Root and effected his termination. Purdy, Tr. 4669.

468. Mr. Vurpillat recalled the conversation, and the substance of it was as Mr. Purdy has described. Vurpillat, Tr. 4670.

469. Mr. Smith testified that he had no role in the decision to terminate Mr. Atchison. Smith, Tr. 4670. At the time, Mr. Smith thought that Mr. Atchison was a qualified inspector and that he should not have been terminated. Id. Mr. Brandt, not Mr. Smith, is responsbile in Mr. Smith's organization for the verification of technical competence of QC welding inspectors. In view of Mr. Brandt's responsibilities in that regard, and based on observations that Mr. Smith made of Mr. Atchison's past performance, Mr. Smith testified that he now understands Mr. Brandt's decision. Id. Based upon his experience over the last several months

in observing work previously inspected by Mr. Atchison, Mr. Smith now questions Mr. Atchison's qualifications as a QC inspector. Mr. Smith testified that his reaction at the time of Mr. Atchison's termination reflected his limited experience in the supervision of technical personnel and his reluctance to discipline and instruct his inspectors on adherence to proper acceptance criteria. Id., Tr. 4670-4671.

470. Mr. Brandt testified that at the time of Mr. Atchison's termination, Mr. Brandt was not aware of the falsification of documents by Mr. Atchison. Brandt, Tr. 4671. As the Level III responsible for evaluating the education of non-ASME mechanical inspectors, which includes welding inspectors, Mr. Brandt stated that Mr. Atchison's lack of education did not necessarily relate to his competence as a QC welding inspector. Id. However, in Mr. Brandt's opinion, without having obtained the associate's degree which Mr. Atchison claimed to have, Mr. Atchison does not qualify under the requirements of ANSI N45.2.6, and could not have become a Level II visual inspector. Id.

471. Applicants' witness Mr. Purdy, who is also a Level III mechanical inspector (Purdy, Tr. 4672), testified regarding the reasons why Mr. Atchison was terminated. Purdy, Tr. 4674-4675. According to Mr. Purdy, Mr. Atchison was terminated by Mr. Purdy in view of Mr. Brandt's lack of desire to further utilize his services due to Mr. Atchison's incompetence in performing visual welding inspections. Mr. Brandt reached that conclusion, and Mr. Purdy has a high level of confidence in Mr. Brandt's expertise and judgment. Id., Tr. 4674. Additionally, Mr. Atchison was terminated for what was perceived by Messrs. Tolson and Brandt, and Mr. Purdy as an effort on Mr. Atchison's

part to utilize an NCR as a negotiation tool. Finally, he was terminated because none of Mr. Purdy's key supervisors had a position for which Mr. Atchison was qualified. Id., Tr. 4674-4675.

472. Mr. Purdy disagreed that Mr. Atchison was terminated for finding and reporting safety problems at Comanche Peak. Mr. Purdy testified that he was unaware at the time of Mr. Atchison's termination that he was the author of M-82-00296 (Applicants' Exhibits 122D and 122E). On April 12, the only NCR that Mr. Atchison had written, to Mr. Purdy's knowledge, was NCR M-82-00361 (Applicants' Exhibit 135), which had been voided by Mr. Brandt for reasons stated on the NCR. Id., Tr. 4675.

473. Mr. Brandt denied that Mr. Atchison was returned to Brown & Root by him for finding and reporting safety problems at Comanche Peak. Mr. Atchison was not returned to Brown & Root because he issued too many NCRs or identified safety problems. Mr. Brandt considers it part of an inspector's responsibility to identify nonconforming conditions and document them in accordance with procedures. In fact, Mr. Atchison wrote approximately a dozen NCRs during his four months as a QC inspector, not forty NCRs as Mr. Atchison claims in his testimony at page 21 (CASE Exhibit 650). This is not a large number of NCRs for a QC inspector to issue in a four-month period. Brandt, Tr. 4675.

474. Further, Mr. Brandt was unaware at the time of his decision to return Mr. Atchison to Brown & Root that Mr. Atchison had alleged anything to the NRC, and was unaware of any threat by Mr. Atchison to contact the NRC. According to Mr. Brandt, Mr. Atchison was returned to Brown & Root because Mr. Brandt determined that he was not competent to perform his

duties as a QC welding inspector, and because he felt that he was using an NCR as a lever to obtain a pay increase. Id., Tr. 4675-4676.

475. The record shows that Mr. Atchison's firing may have been improper. However, no evidence was presented which would lead to the conclusion that Mr. Atchison's firing demonstrated a pervasive intent on the part of the Applicants' QA/QC program to prevent the detection and reporting of construction deficiencies. In fact, there was no evidence that Mr. Atchison was forbidden to write NCRs. Tr. 2577.

476. Mr. Atchison also raised an issue regarding his subsequent firing from Waterford. (Atchison Supplemental #2). Although the Board admitted into evidence the Department of Labor's initial determination that Mr. Atchison was improperly fired from Waterford (CASE Exhibit 684D), that matter has no significant relationship to the issues under consideration here.

(h) The Hamiltons

477. Cordella and Robert Hamilton, both ex-workers at Comanche Peak, also testified as witnesses for CASE. "Testimony of Cordella Marie Hamilton Witness for Intervenor CASE (Citizens Association For Sound Energy)", CASE Exhibits 652, 652-A, ("C. Hamilton") and "Testimony of Robert L. Hamilton, Witness For Intervenor CASE (Citizens Association For Sound Energy)", CASE Exhibits 653, 653A-F, ("R. Hamilton").

478. Mrs. Hamilton testified that she was employed at Comanche Peak from July 8, 1980 to March 9, 1982, as a laborer, inspector trainee and documentation clerk. C. Hamilton, at 1-2. Mr. Hamilton testified that he was employed at Comanche Peak from November 1976 to March 9, 1982, as

a Cadweld inspector and a QC paint inspector. R. Hamilton, at 1. Mrs. Hamilton stated that she quit her position voluntarily when her husband was fired. C. Hamilton, at 4. CASE offered both Mr. & Mrs. Hamilton as lay witnesses. Tr. 3489.

479. Mrs. Hamilton's main concerns related to an alleged lack of documentation and traceability of documentation. C. Hamilton, at 24. She also expressed a concern about the alleged use of drugs on-site (Id., at 22) and stated her opinion that these matters "may jeopardize the health and safety of the public if the . . . plant is allowed to go into operation." Id., at 21.

480. The Board struck as irrelevant to Contention 5 that portion of Mrs. Hamilton's testimony relating to the alleged use of drugs on-site (the last line of page 22 and the first half of page 23). Tr. 3527.

481. Mrs. Hamilton admitted that she had no expertise to make a judgement that operation of Comanche Peak would jeopardize the health and safety of the public. C. Hamilton, Tr. 3490. Although she expressed a concern about documentation for protective coatings, she was not sure of the purpose of such coatings. Id., Tr. 3491. She also limited her expression of concerns about the QA/QC program to protective coatings only. Id., Tr. 3492. She offered no perspective as to the overall adequacy of the Comanche Peak QA/QC program, stating that she has only heard rumors. Id., Tr. 3492-3493.

482. Mr. Hamilton testified that he was terminated for failing to perform an inspection he had been ordered to do "on the liner plate wall located 105 feet above the nearest floor level on the rotating platform rail of the Unit 2 Containment." R. Hamilton at 7, Tr. 3493. He felt

that performing the inspection would jeopardize his safety. R. Hamilton, at 7-10, Tr. 3496.

383. Mr. Hamilton expressed a number of concerns, including the following: the disposition of NCRs he wrote regarding paint (e.g., R. Hamilton, 22); costs of plant construction (Id., at 59); worker safety (Id., at 62), and that he was never told to be more careful because he was working on a nuclear plant (Id., at 65). His concerns about "quality and safety" (Id., at 12) were confined to the quality of protective coatings (Tr. 3497) and safety on the job-site (R. Hamilton, Tr. 3497).

484. The Board struck as irrelevant to Contention 5 those portions of Mr. Hamilton's testimony relating to the monetary cost of construction. Tr. 3526.

485. Although Mr. Hamilton stated his belief that the public health and safety would be jeopardized by plant operation (R. Hamilton, at 64), he admitted that he had no expertise to make such a judgement. Id., Tr. 3512. He also admitted that he did not know of any problems significant to safety which had not been corrected. Id., Tr. 3520-3521.

486. Regarding the NCRs Mr. Hamilton wrote, he stated that the disposition on most of them was "use as is" (R. Hamilton, at 21), and that he did not agree with that disposition. Id. According to Mr. Hamilton, in 1981 he wrote an NCR on oil and grease in paint and the final disposition "was to strain it". Id., at 21, 29. He questioned how one can strain oil and grease out of paint. Id., at 31.

487. The matter to which Mr. Hamilton referred took place in 1979 (R. Hamilton, Tr. 3498-3507) and was recorded on NCR C-1729 (Applicants' Exhibit 138), which Mr. Hamilton prepared. Id., Tr. 3499. Although that

NCR stated that the disposition of that matter was to strain the paint (Applicants' Exhibit 138), a revision to the NCR prepared by Mr. Hamilton on the very same day (Applicants' Exhibit 139) revised the disposition to call for return of the paint to the vendor, because the straining process did not eliminate the foreign particles in the paint. Id., Tr. 3502-3503.

488. Mr. Hamilton admitted that he received a copy of the revision to the NCR. Id., Tr. 3525. He explained the lack of basis for his allegation by stating that he did not mean to imply that the final disposition of the matter was to strain the paint. Id., Tr. 3508. Although Mr. Hamilton maintained that some of the paint containers returned to the vendor were half-empty (indicating that some of the defective paint was used between the time NCRs C-1729 and C-1729, Rev. 1 were prepared), he admitted that some of the paint could have been used for straining. Id., Tr. 3504. Having alleged that this was a matter of "safety significance" (Id., Tr. 3506), Mr. Hamilton nevertheless admitted that he was not an expert on paint and coatings. Id., Tr. 3510.

489. Although Mr. Hamilton implied that he was never told to be more careful because he was working on a nuclear plant (R. Hamilton, at 65), he admitted during cross-examination that during the training he received when he began work at Comanche Peak, he was told that nuclear power plant construction was a matter related to the health and safety of the public. See Applicants Exhibit 140; R. Hamilton, Tr. 3514.

490. The Board finds that the concerns raised by the Hamiltons, which are either not relevant to Contention 5 or lack any foundation in

fact, do not establish the existence of any deficiencies in either QA/QC or in construction at Comanche Peak.

(i) The Stiners

491. Darlene and Henry Stiner also testified as witnesses for Intervenor CASE. See "Testimony of Darlene K. Stiner Witness for Intervener CASE (Citizens Association For Sound Energy)", CASE Exhibit 667 ("D. Stiner), Tr. 4125-4198, and "Testimony of Henry A. Stiner Witness for Intervenor CASE (Citizen's Association for Sound Energy)," CASE Exhibit 666, ("H. Stiner), Tr. 4203-4267.

492. Darlene Stiner testified that she started work at Comanche Peak in August 1977 and is currently employed by Brown and Root as a "Level D QC Inspector on the non-ASME side of the house." D. Stiner, Tr. 4127.

493. Henry Stiner was employed at Comanche Peak by Brown and Root as a welder from November 1979 to December 1980 and from June 1981 to July 1981. H. Stiner (CASE Exhibit 666A), Tr. 4255.

494. In their testimony, the Stiners made a series of allegations of deficiencies on Comanche Peak construction, including the following: numerous pipe supports were fabricated using weave welds which are prohibited by procedure (Tr. 4147; 4210); there was a lack of control at the site of "torque seal" and improper torquing of "Hilti-bolts" (Tr. 4147; 4222, 4225); there is lack of control of weld rods and other weld filler material (Tr. 4147); an NCR which Mrs. Stiner wrote on the polar crane was unfairly voided (Tr. 4177), and plug welding is a common practice at Comanche Peak (Tr. 4154; 4254).

495. In addition to these allegations of specific deficiencies in Comanche Peak construction, the Stiners also generally questioned the safety of the Comanche Peak plant, including its effect on the health and safety of the public (see, e.g., D. Stiner, Tr. 4126). They also criticized the NRC Staff investigation of their allegations (Investigation Report 50-445/81-12; 50-446/81-12, NRC Staff Exhibit 178). See e.g., Tr. 4152, 4204).

496. The Board struck as irrelevant to Contention 5 the portions of the Stiner's testimony concerning the alleged waste of construction material (Tr. 4119); Mr. Stiner's first termination from his job at Comanche Peak (Tr. 4352) and Mr. Stiner's fears that he and his wife could be killed for testifying about Comanche Peak (Tr. 4353). The Board also struck CASE's Motion for a Protective Order (CASE Exhibit 667W1-4) which was included as an exhibit to Mrs. Stiner's testimony (Tr. 4121).

497. The Board has a concern about Mr. Stiner's credibility. Although the Board did not permit the Applicants to impeach Mr. Stiner's credibility through cross-examination concerning the criminal convictions to which he referred in his direct testimony (H. Stiner, Tr. 4249-4250),[23/] other admissible evidence casting doubt on Mr. Stiner's credibility was presented. First, Mr. Stiner admitted during cross-examination that in his first application for work at Comanche Peak (Applicants' Exhibit 145), he did not reveal the existence of his criminal record. H. Stiner, Tr. 4484. Secondly, in his application for re-employment by Brown and

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23/ The basis for the Board's ruling is set forth at Tr. 4471-4474.

Root (Applicants' Exhibit 146), Mr. Stiner admitted that he did not fully disclose the extent of his criminal record. Id., Tr. 4488.

498. The Applicants presented rebuttal testimony to the Stiner's testimony. See "Rebuttal Testimony of C. Thomas Brandt, Ronald G. Tolson, Gordon R. Purdy, Raymond J. Vurpillat and Randall D. Smith Regarding Quality Assurance/Quality Control" (Applicants' Exhibit 141, Tr. 4656-4694).

499. The Staff's testimony presenting its inspection and investigation findings on construction at Comanche Peak (NRC Staff Exhibit 13) addressed Investigation Report 81-12 (NRC Staff Exhibit 178), documenting the Staff's investigation of the allegations made to the NRC Staff by the Stiners. Taylor (NRC Staff Exhibit 13), at 98-99. See also Driskill, Tr. 2463-2467.

500. The Board discusses below each of the substantive allegations raised by the Stiners.

(i) Weave Welding

501. According to Mrs. Stiner, although weave welding (or "weave beading") is not allowed at all at Comanche Peak (D. Stiner, Tr. 4100, 4147), she personally witnessed weave welding. Id., Tr. 4149. Mrs. Stiner described a weave bead as a bead made using transverse oscillation. Id., Tr. 4086. In contrast, a stringer bead, according to Mrs. Stiner, is a bead made using no transverse oscillation. Id., Tr. 4087. Mrs. Stiner alleged that the stops and starts used in "weave beading" result in a weak weld (Id., Tr. 4088-4089); that porosity would be likely in a weave bead (Id., Tr. 4089), and that if weave welds with undetected defects were under stress, the weld could break open and ruin the support con-

taining the welds. Id., Tr. 4092. Mrs. Stiner included as an attachment to her testimony one NCR (NCR M-82-00584) she wrote regarding weave welding. (CASE Exhibit 667Q; Applicants' Exhibit 141J). According to Mrs. Stiner, the disposition on this NCR was "use as is." D. Stiner, Tr. 4154. Mr. Stiner made similar allegations regarding weave welding. (See, e.g. Tr. 4211-4219).

502. Applicants' witness, Mr. Brandt, testified that he is familiar with the issue raised by the Stiners in their testimony relating to the use of weave welding on component supports and the subsequent repair of those welds. Brandt, Tr. 4684. He reviewed several NCRs issued and dispositioned on this subject matter. Id., Tr. 4684-4685. Copies of four of those NCRs are included as Applicants' Exhibits 141J through 141M. Id., Tr. 4685.

503. As defined by ASME Section IX, a weave weld is a weld made with significant transverse oscillation in the welding process. Brandt, Tr. 4412, 4685. Weave welding is permitted by the AWS and ASME, Section IX Codes (Brandt, Tr. 4412) and by Brown & Root welding procedures for component supports. Id., Tr. 4685. Specifically, weave welding up to four times the core diameter of the weld rod being utilized is permitted by Brown & Root welding procedures. Brandt, Tr. 4413, 4635-4636, 4641-4642, 4685).

504. If the term weave welding is interpreted to include those welds in excess of four core wire diameters, then the Brown & Root welding procedures do not permit weave welding. Brandt, Tr. 4412, 4420. It is physically impossible to make a weld that is four-core wire diameters wide without oscillating the arc. Id., Tr. 4428-4429. Generally, the

welds on pipe hangars at Comanche Peak are two to three times the electrode diameter. Id., Tr. 4637. Examples of welding procedures which permit the use of transverse oscillation are Applicants' Exhibits 141N through 141V. Id., Tr. 4528. Bead width determines whether a bead is a weave or stringer bead. Id., Tr. 4589.

505. The technical justification for restricting bead width for weave welding is two-fold. Id., Tr. 4685. First, it is desirable to qualify welding procedures which incorporate as wide a range of material thicknesses as possible. This permits joining of material thicknesses which are great enough to require impact testing as well as material thicknesses which do not require impact testing. Secondly, bead width is limited to control effective heat input which is a factor only for those materials which require impact testing. By utilizing a welding procedure which permits joining a wide variety of material thicknesses, bead width for those materials which do not require impact testing can be effectively limited even though there is no requirement to do so. Id., Tr. 4413, 4685-4686.

506. Weave beading does not necessarily use more filler material than stringer beading. Brandt, Tr. 4410. A weld, whether it is a fillet weld, a partial penetration weld, or a full penetration weld, is three dimensional. Id., Tr. 4411. Consequently, there is a definite volume to be filled. Id. Whether the weld is filled using a weave bead technique or a stringer bead technique, the volume of weld metal deposited is essentially the same. Id.

507. Weave beading is indirectly addressed by the ASME Code as a supplementary essential variable in that Section IX of the ASME Code

implies that it is a factor which must be considered when impact testing is required on the base materials being joined. Id., Tr. 4411-4412.

508. There is no structural significance to starts and stops in welding, if properly done. Id., Tr. 4414. As long as the start and stop is properly tied in and proper fusion exists between the two, there should not be any structural significance to such starts and stops. Id. Depending on the length of the weld, there could be starts and stops with a stringer bead. Id., Tr. 4431. Starts and stops are required to be ground prior to incorporation in the next pass, and the slag must be cleaned off the weld. Id., Tr. 4432.

509. Weave beading does not generally take less time than stringer beading. Id., Tr. 4416-4419. It is easier to do stringer beading than to do weave beading. Id., Tr. 4419.

510. The basic concern with a weave bead is with heat input. Brandt, Tr. 4638-4639. It would be a factor both in carbon steel materials and materials which require impact testing and austenitic stainless steels. Id. Tr. 4639. Overheating of such sections of carbon steel tends to cause embrittlement and loss of ductility. Id. Overheating of austenitic stainless steel causes sensitization or carbon precipitation (i.e., the formation of chrome carbides). Id., Tr. 4639-4640. If the heat input is not controlled, the chromium combines with the carbon to drastically change the grain structure. Id. This process, which occurs at certain temperatures in austenitic steels, can cause corrosion cracking when the weld is placed under stress. Id., Tr. 4640.

511. Welding instructors at Comanche Peak have taught welders that they are not to perform weave beading. Purdy, Tr. 4643. The confusion

about whether or not Brown & Root procedures permit weave beading may have arisen because of the question as to the definition of weave beading. Id. Welders receive instruction as part of their training, which includes face to face communication between welding technicians (and other people directly assigned to the welding engineering group) and the welders in which the performance of the welder is evaluated. Id. Tr. 4644. Additionally, welders are given a "welding grammar guide" when they pick up their electrodes for a given day, which lists the parameters for the welding process to use, including the maximum bead width. Brandt, Tr. 4645.

512. There is a verification program for welding. Brandt, Tr. 4646. Prior to turn over to the operations group, the inspection records for individual supports will be examined on a case-by-case basis for evidence on the inspection record that there was a final visual inspection and other required inspections. Id. In the case of ASME supports (Class 1, 2 or 3), Brown and Root will perform that function. Tolson, Tr. 4647. With respect to Class 5 supports in safety-related buidlings, Mr. Brandt's organization (mechanical/civil QA/QC) and the person in charge of the QA vault (who reports to Mr. Tolson, the site QA supervisor) will jointly perform that function. Id., Tr. 4647. The inspection for the Class 5 supports will be done on a room completion basis, tied in with the room completion schedule. Id. The Brown & Root verification program is ongoing, as of September 14, 1982. Id., Tr. 4648.

513. With regard to the NCR Mrs. Stiner mentioned in her testimony (NCR M-82-00584, CASE Exhibit 667Q; Applicant's Exhibit 141J) the disposition of the NCR is in accordance with appropriate procedures and

criteria. Brandt, Tr. 4686. As stated above, when materials do not require impact testing, excessive bead width generally is not detrimental to the weld. There is no reason here to question the engineering evaluation which led to the disposition of this NCR. Id.

514. The Board finds on the basis of the record evidence that weave welding is not necessarily prohibited at Comanche Peak and that the Stiner's allegations do not demonstrate the existence of any construction deficiencies at Comanche Peak as a result of weave welding.

(ii) Torquing of Hilti-Bolts; Use of Torque Seal

515. According to Mrs. Stiner, there is a lack of control of a product called "Torque Seal" which is used on "Hilti-bolts" once they have been torqued and inspected. D. Stiner, Tr. 4155; 4083-4086. Specifically, she alleged that she had personal knowledge of instances in which the craft possessed torque seal. Id. She testified that the reason "Torque Seal" is controlled is because if craft personnel have access to it, there would be no way to prove that a Hilti-bolt was sealed by QC. Id., Tr. 4156.

516. According to Mrs. Stiner, an inspector should personally witness the torquing of the bolts by the craftsperson. Id., Tr. 4157. She stated that the inspector should check the calibration of the wrench and assure that it is torqued to meet the specification of "CEI 20." Id. After the inspector witnesses that, the torque seal should be applied. Mrs. Stiner believes that torque seal, if properly controlled, would prevent the craft from tampering with Hilti bolts. Id., Tr. 4049.

517. She alleged that her supervisor instructed her that if the documentation regarding Hilti-bolt torquing did not contain a QC inspector's signature, she was to disregard that and assume that since torque seal was a controlled substance that QC had applied it. Id., Tr. 4085. Mrs. Stiner referred to an NCR she wrote (NCR M-81-01570, CASE Exhibit 667R) regarding lack of documentation to verify that torque seal had been applied by QC. Id., Tr. 4160. Mr. Stiner expressed the same concerns. H. Stiner, Tr. 4222-4226; 4254; 4300.

518. Hilti-bolts are concrete expansion on anchors which are used for attaching structural steel members and supports of all types to concrete structures. "Hilti" is a trade name. Brandt, Tr. 4686.

519. Hilti-bolts are installed by drilling a hole into the concrete wall, driving the bolt into the wall and then torquing the bolt in order to set the wedges on the embedded end of the bolt in concrete. Id. The process of torquing of Hilti-bolts is inspected by QC. Id., Tr. 4686-4687. Inspection procedures require that a minimum of one bolt per base plate shall be inspected by a QC inspector to verify that the correct torque has been applied. The inspection personnel are directed to verify the proper torquing of 100% of all bolts in safety-related structures. Id., Tr. 4687; 4533. Specifically, they are instructed to inspect torquing of all concrete expansion anchor bolts in safety-related structures. Id., Tr. 4533.

520. A QC inspector verifies that a bolt has been properly torqued as follows: First, a QC inspector must verify that the torque wrench has been set at the proper torque setting. Then he must assure that the required torque value has been reached as evidenced by a clicking sound

made by the torque wrench. Id., Tr. 4687; 4537. Last, he must verify that the setting on the torque wrench after torquing remains as it was set prior to torquing. Id., Tr. 4687.

521. Torque seal is an orange fluid which dries to a wax-like consistency which is applied to the interface between the nut and the bolt to serve as a preliminary indicator that a satisfactory torque value has been attained and to reduce the necessity for retorquing. Id., Tr. 4687; 4533-4535; 4649. If the nut has been turned in an effort to remove it, the interface would be gone (Id., Tr. 4649) and the torque seal would be broken. Id., Tr. 4650. It would be evident that this had occurred. Id., Tr. 4649-4650.

522. At Comanche Peak, QC buys torque seal and it is maintained by Mr. Harry Williams, one of the supervisors who works for Mr. Brandt. Brandt, Tr. 4535-4536; 4543. As part of Mr. Brandt's normal supervisory function, he checks on Mr. Williams' performance of his duties. Id., Tr. 4636.

523. Torque seal is assigned directly to the QC inspectors. Id., Tr. 4536. There are no written procedures at Comanche Peak relating to the control of torque seal. Brandt, Tr. 4543.

524. Mr. Brandt testified that he is sure that there have been cases where different tubes of torque seal were in the possession of individuals other than QC inspectors. Id., Tr. 4536. However, he did not consider the fact that craft may have access to torque seal at Comanche Peak to be a problem. Id., Tr. 4544. Normally, the QC inspector performing the verification of the torque operation applies the torque seal. In those cases where physical access to the Hilti-bolt is

limited to one person, the inspector may hand the torque sealant to the craftsman for application provided that the inspector verifies that the sealant is applied. Id., Tr. 4687-4688, 4540, 4542.

525. Mr. Brandt testified that he was familiar with the matter raised in NCR M-81-01570 (CASE Exhibit 667R). Id., Tr. 4688. The substance of that NCR was that no documentation existed to indicate the torquing had been verified by QC, yet torque seal had been applied to the Hilti-bolts. The disposition of the NCR indicates that QC should verify the torquing on the Hilti-bolts. Id.

526. Hilti-bolt verification is documented on an inspection report that ties a unique identifier to an individual support. Tolson, Tr. 4600. There is an individual inspection report for each component support. Id., Tr. 4652. Mr. Tolson testified that there would be no safety significance if torque sealant were applied by unauthorized personnel. Id., Tr. 4688. The final verification of proper torquing on Hilti-bolts is through a record review prior to turnover to TUGCO Operations. Id. Review of these inspection reports will verify that sufficient inspection records exist to substantiate that QC performed all required torque verifications. Id., Tr. 4600; Brandt, Tr. 4544-4545.

527. The Board finds that the record evidence demonstrates that there are adequate procedures at Comanche Peak for assuring the proper torquing of Hilti-bolts. The record evidence does not demonstrate the existence of any construction deficiencies as a result of the torquing of Hilti-bolts.

(iii) Weld Rod Control

528. Both Stiners alleged that there is a lack of weld rod control at Comanche Peak. D. Stiner, Tr. 4162-4168, H. Stiner, Tr. 4230. Mrs. Stiner described the circumstances surrounding the issuance of an NCR she wrote regarding lack of weld rod control (NCR M-81-01299, CASE Exhibit 667S). Id., Tr. 4163. In this instance, Mrs. Stiner discovered that the can of a portable rod oven had been unplugged for a greater period of time than allowed. Id.. According to Mrs. Stiner, the safety significance of this is that the welding rods were not in the proper heat range and could have collected moisture and resulted in "contaminated welds." Id., Tr. 4164. She stated that an incident in which she found abandoned weld rods showed that the procedures governing weld rod control were not being followed. Id. Tr. 4165-4166. Mrs. Stiner testified that as a welder she frequently observed weld rods unattended and in places where they should not have been, although she could not provide specific instances other than the one incident she described. Id., Tr. 4095-4099. According to Mrs. Stiner, without proper control of weld rods, there is no way to verify the integrity of the welded supports. Id., Tr. 4099.

529. There are procedures, which are the same for non-ASME and ASME welders, for controlling the use of weld rods. Brandt, Tr. 4688; Purdy, Tr. 4689. The use of weld rods is controlled for non-ASME welding to assure that safety-related weld rods and the proper type of weld rods are used in the correct applications. Brandt, Tr. 4689. The use of weld rods is controlled for ASME welding in accordance with ASME Section III, subsections NB, NC and ND, which provide that heat traceability shall be maintained for any particular weld. Purdy, Tr. 4689. This means that

the heat or lot number must be shown for all weld rods which are incorporated into a given weld. Id.

530. Mr. Brandt described the procedure for controlling weld rods for non-ASME welding. Specifically, a weld rod is issued by one of two methods, either on a Weld Filler Material Log (WFML) or by the use of weld rod issuance slips. Brandt, Tr. 4689.

531. According to Mr. Purdy, weld rod control for ASME is similar to that described by Mr. Brandt. Purdy, Tr. 4689. The WFML system is the only system employed by the ASME program. Id.

532. There have been instances where weld rod control procedures have not been followed. Brandt, Tr. 4690. Such instances were documented NCRs. An example of such an NCR is CASE Exhibit 667S. Id.

533. Mr. Brandt described the technical justification for requiring low hydrogen electrodes (rods) to be issued in heated containers. Id. Electrodes are issued in heated containers to minimize the possibility of moisture accumulation in the electrode coating. Id.

534. He also addressed the significance of use of weld rods that are not maintained in a heated condition. Id. The most visible indication would be that the weld might contain excessive amounts of porosity. This would be due to moisture contained in the electrode coating being introduced into the weld as steam and upon escape leaving a gas pocket (porosity). Such porosity would be detected upon visual inspection, and appropriate acceptance criteria would be applied in accordance with applicable inspection instructions. Id.

535. The Board finds that the record evidence does not demonstrate the existence of any unresolved confirmed construction deficiencies as a result of weld rod control.

(iv) The Polar Crane NCR

536. According to Mrs. Stiner, an NCR which she wrote on the polar crane (NCR M-81-00144, CASE Exhibit 667U) was unfairly voided. D. Stiner, Tr. 4177-4178. According to Mrs. Stiner, the "bus box" had been burned extensively. Id., Tr. 4071, 4177. She described the incident leading to the issuance by her of this NCR. Id., Tr. 4177-4178. She stated that the NCR was voided because the polar crane bus-box is "non-Q." Id., Tr. 4178. She disagreed with the disposition of this NCR. Id.

537. Mrs. Stiner's direct testimony implied that as a result of this incident, there was actually a hole in the polar crane rail. Id. Tr. 4179. According to Mrs. Stiner, "running the polar crane on a track with a hole in it could have great safety implications." Id.

538. During cross-examination, Mrs. Stiner first repeated that there was a hole in the polar crane track. Id., Tr. 4072. Then she stated that there was a corroded area on the rail itself and there was a hole in the bus box itself. Id., Tr. 4073-4074. She admitted that she wasn't sure how the polar crane runs on the rail, but that it was just her opinion that the pitting and the hole in the bus bar was significant. Id., Tr. 4075.

539. The Board finds on the basis of the record evidence that the incident regarding the polar crane does not demonstrate the existence of a construction deficiency having safety significance.

(v) Plug Welding

540. Mrs. Stiner testified that plug welding is a common practice at Comanche Peak (D. Stiner, Tr. 4154) and that she "was instructed on numerous occasions to plug weld holes on Q hangars without a QC inspector being present." Id. According to Mrs. Stiner, the safety significance of plug welding is that if "the plug tempered with the base metal, it could create a weak spot. Consequently the weld itself could break." Id. Mr. Stiner made similar allegations and implied that plug welding is "illegal" at Comanche Peak. H. Stiner, Tr. 4219-4222.

541. Applicants' witness Mr. Brandt testified that he was familiar with the issues raised by the Stiners regarding plug welding. Brandt, Tr. 4690-4691. According to Mr. Brandt, the Stiners were referring to the fillet welding of holes which are drilled in incorrect locations in structural shapes. Id., Tr. 4691. That type of fillet welding is permitted at Comanche Peak to repair holes which were drilled in the wrong location on structural shapes. Id.

542. The procedures require that a final visual inspection is to be performed by the QC inspector. Any plug weld in safety related structures which was not inspected would violate procedures. Id., Tr. 4629.

543. The Board finds on the basis of the record evidence that the Stiner's allegations regarding plug welding do not demonstrate the existence of any confirmed construction deficiencies.

(vi) General Allegations

544. Although Mrs. Stiner generally alleged that there are QC problems at Comanche Peak which may have an affect on the health and safety of the public (D. Stiner, Tr. 4126), she admitted that in making this statement, she was just speculating. Id., Tr. 4039. She also stated that she did not have any background in any field such as health physics, which would qualify her to render a professional opinion concerning the health effects of Comanche Peak operation. Id., Tr. 4042.

545. Both of the Stiners were dissatisfied with the findings in the NRC Staff Investigation Report (81-12, NRC Staff Exhibit 178) concerning allegations made to the NRC before they testified. (See, e.g., D. Stiner, Tr. 4033, 4152, 4204). Mr. Stiner in particular alleged that the facts in Report 81-12 do not reflect all the allegations made to the investigators (H. Stiner, Tr. 4204) and that the facts do not appear to be supported by the investigators' sanitized notes (CASE Exhibit 666C-6-666C-39). Id. He also alleged that the investigators did not contact all the individuals he identified who could corroborate his allegations. See e.g., H. Stiner, Tr. 4223.

546. NRC Staff Investigation Report 81-12 (NRC Staff Exhibit 178) discusses all of the allegations made by Mr. Stiner in the signed witness statement he provided to the NRC investigator, specifically: weave welding; plug welds; control of torque seal, and torquing of Hilti-bolts. Details regarding the investigation of each of these allegations are contained on pp. 3-9 of the report. As noted in the report, numerous interviews of Comanche Peak construction personnel, QC inspectors and craft supervisors were conducted, in addition to inspections of areas

identified which allegedly exemplified the allegations. NRC Staff Exhibit 178, at 2.

547. CASE's representative Mrs. Ellis had the opportunity to cross-examine the primary NRC investigator, Mr. Driskill. Tr. 2464-2467. Mr. Driskill stated that he was given names of certain individuals who could support the allegations. Driskill, Tr. 2466. Mr. Driskill testified that he interviewed each of these individuals (Id., Tr. 2467) and that none of them could provide him with any information which supported the allegations. Id., Tr. 2466-2467. He did state that there may have been occasions in which an individual he interviewed did indicate that problems similar to those allegations existed (Id., Tr. 2467) but that to that individual's knowledge, all such problems had been corrected or weren't actually problems to the extent alleged. Id.

548. Concerning Mrs. Stiner's concern that not all of the Stiner's allegations were actually investigated, during cross-examination, Mrs. Stiner was asked whether she felt that the NRC is obligated to address every concern that might be raised by anyone, whether or not the concern is insignificant. Tr. 4030. She stated that every effort should be made to verify the safety significance of a concern and if there is none, there should be "another section" to which the concern could be reported. Id. She acknowledged that there may be situations in which an individual, such as herself, might raise an issue with the NRC which appears to be a problem and yet which, upon investigation, turns out not to be a problem. Id., Tr. 4030-4031.

549. Although Mr. Stiner contended that Investigation Report 81-12 does not contain all the allegations he made to the NRC Staff investiga-

tors, in his signed witness statement, introduced by CASE (CASE Exhibit 666C1-C5) he certified that the statement of his allegations was "true and correct." CASE Exhibit 666C-5. Moreover, in that certification, he also stated that "I have read over, understand, made corrections to and initialled errors in the foregoing . . . statement." Id. When questioned about this certification, Mr. Stiner first claimed that when he read it, he wasn't "looking for typographical errors . . . and deletions" (H. Stiner, Tr. 4320) and that he just "thought it was a statement in general to get the investigation going." Id. Although Mr. Stiner contends that the statement doesn't contain all of the allegations made to the investigator (H. Stiner, Tr. 4325), he nevertheless acknowledged that "this statement is true and correct." Id. The Board finds Mr. Stiner's contention, when viewed against the statement he certified as being true and correct, to be incredulous.

550. Although the Board has expressed concerns regarding the adequacy of NRC Staff investigations, that issue is not before the Board for adjudication, but rather, the adequacy of the QA/QC program at Comanche Peak. Further, in view of the doubts cast on Mr. Stiner's credibility, his allegations taking issue with NRC Investigation Report 81-12 and disputing the statements therein are not to be given great weight.

B. Contention 22, (Emergency Planning)

551. Contention 22, as admitted in the Board's Order of June 30, 1982, states as follows:

Applicants have failed to comply with 10 C.F.R. Part 50, Appendix E, regarding emergency planning, for the following reasons:

a. The FSAR does not identify state or regional authorities responsible for emergency planning or who have special qualifications for dealing with emergencies.

b. No agreements have been reached with local and state officials and agencies for the early warning and evacuation of the public, including the identification of the principal officials by titles and agencies.

c. There is no description of the arrangements for services of physicians and other medical personnel qualified to handle radiation emergencies and arrangements for the transportation of injured or contaminated individuals beyond the site boundary.

d. There are no adequate plans for testing by periodic drills of emergency plans and provisions for participation in the drills by persons whose assistance may be needed, other than employees of the Applicant.

e. There is no provision for medical facilities in the immediate vicinity of the site, which includes Glen Rose.

f. There is no provision for emergency planning for Glen Rose or the Dallas/Ft. Worth metroplex.

552. CASE did not present any direct evidence on this contention.

553. Applicants presented the testimony of witnesses Richard A. Jones, Bobby T. Lancaster and Roger E. Linneman, M.D. (See "Testimony of Richard A Jones, Bobby T. Lancaster and Roger E. Linneman, M.D., Concerning Contention 22 on Emergency Planning," Applicants' Exhibit 143), and Alton B. Armstrong, Jr., Clarence L. Born, Larry J. Skiles, and Arthur C. Tate (See "Testimony of Alton B. Armstrong, Jr., Clarence L. Born, Larry J. Skiles, and Arthur C. Tate Concerning Contention 22 on Emergency Planning," Applicants' Exhibit 144).

554. The NRC Staff presented the testimony of NRC Staff member David M. Rohrer ("NRC Staff Testimony of David M. Rohrer Regarding Emergency Planning (Contention 22)," NRC Staff Exhibit 202) and Federal Emergency Management Agency (FEMA) Staff members Albert Lookabaugh and John Benton ("FEMA Staff Testimony of Albert Lookabaugh and John Benton Regarding Emergency Planning (Contention 22)," NRC Staff Exhibit 203).

1. <u>Contention 22(a)</u>

555. The Hood and Somervell County Emergency Organizations are the key emergency planning organizations at the local level. Section 1.3.1 of the Applicants' Emergency Plan, dated May 21, 1982, Applicants' Exhibit 143D. These organizations are the Hood and Somervell Counties' sheriff departments, fire departments, hospitals, and ambulance services responsible for planning and implementing protective measures for citizens in the respective counties. (Rohrer, Tr. 5686; Jones, Tr. 5442-43; Skiles, Tr. 5515).

556. Written agreements have been reached between the Applicants and the Hood and Somervell Counties' Emergency Organizations to provide support in the event of an emergency at CPSES. Appendix H, Section 15.0 of the Applicants' Emergency Plan. Mr. Jones described these agreements in his written testimony. Jones, Tr. 5443-44.

557. Both Somervell and Hood Counties have an "Emergency Operations Plan," a "Fixed Nuclear Facility Response Plan," and a "Manual of Emergency Procedures." Applicants' Exhibit 144E. These documents describe the county emergency organization, assign specific responsibility and tasks to individual agencies, and prescribe a response based on a

particular emergency action level. Official, signed copies of these plans are on file in the Division of Emergency Management. Skiles, Tr. 5514; Armstrong, Tr. 5514-15.

558. The County Judges for Hood and Somervell Counties are the individuals who are responsible for directing the operations of their respective county emergency organizations, as stated in Section 1.3.1 of the Applicants' Emergency Plan. Rohrer, Tr. 5686; Skiles, Tr. 5515; Jones, Tr. 5475-76.

559. The Bureau of Radiation Control of the Texas Department of Health is the lead agency in the State of Texas for responses to radiological emergencies, as stated in Section 1.3.2 of the Applicants' Emergency Plan. The individual responsible for directing the Bureau of Radiation Control is the Bureau's Chief, Mr. David Lacker. Rohrer, Tr. 5686; Tate, Tr. 5516; Jones, Tr. 5444.

560. The Bureau of Radiation Control will send an emergency response team to the ten-mile plume exposure pathway emergency planning zone ("EPZ"). This team will provide environmental and radiological monitoring, assessment of off-site hazards, and recommendations for protective actions to State and county officials. Appendix F, Annex L to the State Plan. Tate, Tr. 5517, 5533.

561. The Division of Emergency Management of the State of Texas is responsible for the planning, direction, and control of the overall emergency response by State agencies and departments. The Division of Emergency Management is part of the Texas Department of Public Safety. The current Director is Mr. James Adams. Rohrer, Tr. 5686; Jones, Tr. 5444, 5476; Armstrong, Tr. 5515-5516.

562. The Texas Department of Public Safety ("DPS") will assist in the coordination of local law enforcement agencies, traffic control assistance, roadblock establishment, assistance in protection of life and property, and assistance in alerting and warning persons in the affected area. The DPS communications network will serve as the primary communications link between the Applicants, the State, and Hood and Somervell Counties. Annex R, State of Texas Emergency Management Plan. Jones, Tr. 5444-45; Armstrong, Tr. 5517, 5518.

563. A letter of Agreement between the DPS and the Applicants is contained in Appendix H, Section 15.0 of the Applicants' Emergency Plan. Jones, Tr. 5444-45.

564. The Applicants also have a written agreement with the Bureau of Radiation Control to provide support in the event of radiological emergencies at CPSES in accordance with its Radiological Emergency Response Plan and the State Fixed Nuclear Facility Response Plan. Appendix H, Section 15.0 of the Applicants' Emergency Plan. Jones, Tr. 5445.

565. The State's principal radiological emergency planning documents are (1) the State of Texas Emergency Management Plan (Applicants' Exhibit 144F); (2) the Texas Department of Health Annex L to the basic state plan (Applicants' Exhibit 144F); (3) Appendix 7 (Radiological Emergency Response) to Annex L (Applicants' Exhibit 144G), and (4) Tab 1 (Fixed Nuclear Facilities) to Appendix 7. Armstrong, Tr. 5516.

566. The Staff's expert witness on emergency preparedness, Mr. Rohrer, concluded that the Applicants' Emergency Plan adequately identifies the State and local (County) government organizations and

individuals with the responsibility and authority for emergency response planning. Rohrer, Tr. 5686.

567. The assignment of emergency response and regulatory functions among federal agencies, the State of Texas, and the Hood and Somervell County governments is set forth in Tab 1, "Fixed Nuclear Facility Accidents", of Appendix 7, "Radiological Emergency Response", to Annex L, "Texas Department of Health", of the Texas Emergency Management Plan ("State Plan"). Lookabaugh and Benton, Tr. 5747.

568. The role of federal agencies in the event of an emergency at CPSES is described in Mr. Jones' written testimony. Jones, Tr. 5445-47.

569. The expert witnesses on emergency preparedness for FEMA, Messrs. Benton and Lookabaugh, stated that the State Plan and the Hood County and Somervell County Emergency Operations Plans adequately identify the appropriate State and county officials who are responsible for emergency planning. Lookabaugh and Benton, Tr. 5747. This finding will be confirmed in FEMA's Preliminary Findings regarding off-site emergency planning for CPSES. Id.

2. Contention 22(b)

570. Section 3.0 of the Applicants' Emergency Plan sets forth procedures for emergency notification of the public, including the delineation of the notification criteria for each emergency action level, the time constraints on initial and close-out information messages, the methodology for notifying emergency response personnel, and the details on call-back verification of telephone and radio communications. A description of the offsite protective actions is contained in Section 8.2 of the Applicants' Emergency Plan. Jones, Tr. 5448-49.

571. Should an incident occur at CPSES, the County Judges for Hood and Somervell Counties will be notified by the Texas Department of Health, the Texas Department of Public Safety, or the Applicants. Annex A, Paragraph V.C., "Fixed Nuclear Facility Incident", and Attachment D to the Hood and Somervell Counties Emergency Operations Plans. Lookabaugh and Benton, Tr. 5748-49.

572. Applicants will normally notify the Department of Public Safety District Office in Waco, and the NRC Incident Response Center in Bethesda. The DPS District Office will in turn notify the Sheriff's Office of the appropriate Emergency Action Levels ("EALs"), as stated in Attachment D to the Hood County and Somervell County Emergency Operations Plans. The DPS District Office will have dedicated telephone lines to the two Sheriff's offices in Hood and Somervell Counties. These lines cannot be interrupted and act as open channels, which eliminates the problem of busy signals. Lookabaugh and Benton, Tr. 5749; Jones, Tr. 5452; Skiles, Tr. 5577-78.

573. The DPS District Office will notify the Sheriff's Office of either Hood or Somervell Counties by Department of Public Safety ("DPS") radio, or the Texas Law Enforcement Teletype System ("TLETS"). Commercial telephone will be used, if necessary, according to Attachment D to the Hood County and Somervell County Emergency Operations Plans. The operation of the DPS communications system during an emergency is discussed in greater detail by Applicants' expert witness, Mr. Armstrong, in his written testimony. Lookabaugh and Benton, Tr. 5749; Armstrong, Tr. 5518-20.

574. The County Judges for Hood and Somervell Counties will decide whether the public in the county for which they are responsible should be warned. The Texas Department of Health may recommend to the County Judges whether to warn the public. Lookabaugh and Benton, Tr. 5748-49; Lancaster, Tr. 5481-82.

575. The public will be warned by 40 outdoor pole-mounted sirens located in threatened areas within the 10 mile Emergency Planning Zone ("EPZ"), and by mobile public address ("PA") systems mounted in vehicles. The designated Warning Officer in the Sheriff's Office will activate the sirens and dispatch the warning vehicles. A Warning Officer will be present 24 hours a day at the Sheriff's Offices in the two counties. Lookabaugh and Benton, Tr. 5749; Jones, Tr. 5449, 5451-52; Skiles, Tr. 5587.

576. The siren system is designed to provide coverage of the entire 10 mile EPZ, including urban, rural, and recreational areas. The system will be activated and operated by the local officials of Hood and Somervell Counties. All 40 sirens can be activated simultaneously from either Hood or Somervell Counties' emergency operations facilities. Each County can also activate those sirens located in their respective counties. Lookabaugh and Benton, Tr. 5749-50; Jones, Tr. 5449-50, 5477; Lancaster, Tr. 5477-78, 5479.

577. Once members of the public have been warned, they must be advised as to what protective actions they should take. Protective action information will be provided to the public by the Emergency Broadcast System ("EBS"). Lookabaugh and Benton, Tr. 5748-49.

578. Generally, the public may be advised either to seek shelter where they are (in place sheltering), or to evacuate (relocate). Notifi-

cation messages broadcast over the EBS will convey information concerning the type and nature of the emergency condition, the affected area, and the protective actions which should be taken by the affected public. Examples of such protective action instructions are to take shelter, close all doors and windows, and listen to the radio or television for instructions. Additional messages may include, when necessary, evacuation instructions specifying the location of shelter areas, what to take, and what equipment and facilities will be available at the shelter area. The EBS would also be used to advise the public of changes in recommended protective actions and to issue an all clear announcement. Jones, Tr. 5450; Lookabaugh and Benton, Tr. 5748-5749.

579. The County Judge activates the EBS by contacting WBAP radio/TV in Fort Worth, Texas, and directing them to initiate the EBS procedures. Lookabaugh and Benton, Tr. 5750; Jones, Tr. 5450.

580. In addition to EBS, the County Judge may simultaneously contact the National Weather Service's Weather Radio System to request that information be broadcast to the public on the system. Lookabaugh and Benton, Tr. 5750.

581. In the event that EBS cannot be utilized, the Counties' mobile PA vehicles will be dispatched to provide information to the public, as provided in Attachment F to the Hood and Somervell Counties' Emergency Operations Plans. Lookabaugh and Benton, Tr. 5750.

582. Each County Judge will decide whether to evacuate the county for which he is responsible. The County Judges will be advised by the Texas Department of Health of what protective action is appropriate.

Attachment G to the Hood and Somervell Counties' Emergency Operations Plans. Lookabaugh and Benton, Tr. 5748-50; Jones, Tr. 5450-51.

583. The County Judge's evacuation order is implemented by the County Sheriff. The Sheriff will receive advice and guidance from the Texas Department of Health and/or the Applicants regarding which areas within the 10 mile EPZ should be evacuated. The Sheriff will establish roadblocks, notify residents of designated evacuation areas, and assist in the relocation of evacuees. Attachment G to the Hood and Somervell Counties' Emergency Operations Plans. Lookabaugh and Benton, Tr. 5750-51; Jones, Tr. 5451-52.

584. FEMA will withhold final approval on warning systems until the systems are installed, tested, and evaluated in accordance with FEMA rules and regulations. Lookabaugh and Benton, Tr. 5750.

585. The Applicants have obtained written agreements with the various state and local agencies responsible for warning and evacuation of the public in the event of an emergency at CPSES. Copies of the Letters of Agreement are in Appendix H, Section 15.0 of the Applicants' Emergency Plan. Applicant's expert witness, Mr. Jones, briefly describes those agreements in his written testimony. Jones, Tr. 5451-52.

586. Mr. Lookabaugh and Mr. Benton concluded that the State Plan, and the Hood and Somervell Counties' Emergency Operations Plans contain adequate provisions for the notification and evacuation of the public located within the 10 mile EPZ. Lookabaugh and Benton, Tr. 5751. This conclusion will be confirmed in FEMA's Preliminary Findings regarding off-site emergency planning for CPSES. Id.

3. Contention 22(c)

587. Hood General Hospital in Granbury, Texas, is designated in Section 1.3.1.4 of the Applicants' Emergency Plan as the local facility for the receipt and treatment of injured personnel from CPSES who are contaminated with radioactive material or who have received an overexposure to radiation which requires medical evaluation. Treatment includes gross decontamination, life saving activities, and patient stabilization. Hood General Hospital is 16 road miles from CPSES. Rohrer, Tr. 5688, 5690-91; Linnemann, Tr. 5457.

588. Applicants have contracted with Radiation Management Corporation ("RMC") to provide expertise, facilities, and equipment to Hood General Hospital, to assure a comprehensive emergency medical assistance program at Hood General. Section 10.1 of Applicants' Emergency Plan. RMC will conduct a semi-annual review of plant and hospital procedures; provide annual training for plant, ambulance, and hospital personnel involved in the radiation emergency medical program; prepare a radiation accident scenario, coordinate a medical emergency drill, and prepare evaluation reports of the drills and exercises. RMC will also conduct an annual seminar on management of radiation accidents for physicians and other medical personnel. Rohrer, Tr. 5688; Linnemann, Tr. 5459.

589. Northwestern Memorial Hospital in Chicago, Illinois, is affiliated with RMC. It is designated in Section 1.3.1.4 of the Applicants' Emergency Plan as the back-up medical facility for treating radiologically-contaminated individuals. The medical capabilities of the Northwestern Memorial Hospital include a fully-equipped radiosurgery

suite, reverse isolation units, facilities for white cell transfusion, bone marrow transfusion, and chromosome analysis. A more complete description of the capabilities of Northwestern Memorial Hospital is set forth in Section 10.1 of Revision 4 to the Applicants' Emergency Plan. Rohrer, Tr. 5690-5691; Linnemann, Tr. 5457-58; Jones, Tr. 5435-38.

590. RMC will provide 24-hour, seven days a week availability of expert consultants, and the services of a Radiation Emergency Medical ("REM") Team. The REM Team consists of a licensed physician experienced in radiation medicine, a certified health physicist, and technicians with portable instruments who will be flown in from out-of-state to respond to Applicants' request for expert assistance. If requested by Applicants, RMC will also provide the services of its bioassay laboratory and whole body counting facility. Rohrer, Tr. 5675-76, 5688; Linneman, Tr. 5459.

591. Letters of agreement between Applicants, and both Hood General Hospital and RMC are contained in Section 15, Appendix H of the Applicants' Emergency Plan. The letter of Agreement between RMC and Northwestern Hospital was attached to Mr. Linnemann's written testimony as Attachment C, Applicants' Exhibit 143C. Linnemann, Tr. 5434, 5457-58; Rohrer, Tr. 5667-68, 5689.

592. The State of Texas Emergency Management Plan designates Hood General Hospital as a facility for treatment of radiologically-contaminated persons. In addition, the State Plan designates two other hospitals for treatment of radiological-contaminated persons which are not listed in the Applicants' Emergency Plan: Stephenville Hospital in Stephenville; and Johnson County Memorial Hospital in Cleburne. Tab 1, Appendix 7,

Annex L of the Texas Emergency Management Plan. Lookabaugh and Benton, Tr. 5752, 5759.

593. The Hood and Somervell Counties' Emergency Operations Plans do not identify any facilities for treatment of radiologically-contaminated persons. Lookabaugh and Benton, Tr. 5752.

594. Arrangements have been made between the State, and the Johnson County and Stephenville hospitals for the receipt and treatment of radiologically contaminated individuals. Born, Tr. 5520.

595. Letters of agreement between the County governments, and the Hood and Stephenville hospitals which are identified in the State Plan as hospitals for the receipt and treatment of radiologically contaminated individuals are not contained or referenced in that Plan. However, since Hood General Hospital is a county-owned hospital, FEMA witnesses Messrs. Benton and Lookabaugh concluded in their oral testimony that no arrangements or letters of agreement need be referenced by the State or Hood County in their emergency plans. A Letter of Agreement has been reached between Johnson County Memorial Hospital and the City of Cleburne for provision of medical services (Applicants' Exhibit 144H). Messrs. Lookabaugh and Benton stated that they are satisfied with that Letter of Agreement for the purpose of satisfying Criterion L-3 of NUREG-0654/FEMA-REP-1. Lookabaugh and Benton, Tr. 5708-09, 5710-12, 5752; Born, Tr. 5520, 5662; Skiles, Tr. 5663.

596. Mr. Rohrer testified that the Applicants' Emergency Plan adequately identifies the medical personnel, services, and primary and back-up facilities for the treatment of radiologically-contaminated persons. Rohrer, Tr. 5689, 5691.

597. The State Plan and the Counties' Emergency Operations Plans do not sufficiently describe the capability of Hood General, Stephenville, and Johnson County Memorial hospitals to receive, evaluate, and treat radiologically-contaminated or injured individuals, contrary to Criterion L-1. Lookabaugh and Benton, Tr. 5714, 5752.

598. The State and county plans were submitted to FEMA for review on June 17, 1982. Shortly thereafter, FEMA was given a verbal commitment by the State and County officials responsible for writing the plans that letters of agreement with any non-governmental organization having an assigned responsibility within the plans would be either incorporated or referenced in the plans. Conversations with the Texas Bureau of Radiation Control indicate that additional information regarding the medical capabilities and resources of Hood General, Stephenville, and Johnson County Memorial hospitals will be incorporated in the plans. In his oral testimony, Mr. Lookabaugh reaffirmed that the State will be providing letters of agreement and additional information on the medical capabilities and resources of the three hospitals. Lookabaugh and Benton, Tr. 5711-13, 5716-17, 5753.

599. Messrs. Lookabaugh and Benton concluded that the State, Hood County, and Somervell County emergency plans presently do not satisfy Criteria L.1 and L.3 of NUREG-0654/FEMA-REP-1, and therefore do not adequately address the FEMA guidance criteria for provision of medical facilities and personnel to treat radiologically-contaminated individuals. However, on the basis of the verbal commitments they received from officials for the State and two counties, they believe that this inadequacy

will be rectified. Lookabaugh and Benton, Tr. 5716-17, 5752-53. FEMA's findings will be confirmed in FEMA's Preliminary Findings for CPSES. Id.

600. A plant emergency vehicle is available at CPSES to transport injured personnel, including radiologically-contaminated personnel, to offsite medical facilities, including Hood General Hospital for radiologically-contaminated individuals. Applicants' Emergency Plan, Section 10.2. Rohrer, Tr. 5689; Jones, Tr. 5460.

601. Radiologically-injured individuals who need to be treated at Northwestern Memorial Hospital will be transported to Chicago by private surface and air transportation services. Appropriate protective measures (such as isolated transportation, if necessary, and attendance by technicians trained in transporting and handling radiologically contaminated patients) will be instituted by RMC and the Applicants. Linnemann, Tr. 5458-59.

602. The Glen Rose/Somervell County Volunteer Fire Department Ambulance Service, and the Hood General Hospital Ambulance Service have agreed to provide back-up ambulance service to the Applicants' on-site plant emergency vehicle. Applicants' Emergency Plan, Section 10.2. Rohrer, Tr. 5689; Jones, Tr. 5460.

603. It is unlikely that the unavailability of back-up ambulance services would affect the capability of Applicants to transport on-site personnel in need of medical attention, since the on-site emergency vehicle is dedicated to providing that service. Rohrer, Tr. 5676-78.

604. RMC will train and exercise ambulance service personnel in the transportation and handling of radiologically injured patients. A

training session for those persons was conducted by RMC at CPSES in August 1982. Linnemann, Tr. 5461.

605. Letters of agreement between the Applicants and both the Hood General Hospital Ambulance Service, and the Glen Rose/Somervell County Volunteer Fire, Rescue, and Ambulance Service are contained in Appendix H of the Applicants' Emergency Plan. Rohrer, Tr. 5689; Jones, Tr. 5460.

606. Mr. Rohrer concluded that the Applicants' Emergency plan adequately describes the provisions for transportation of injured persons, including those who are radiologically-contaminated. Rohrer, Tr. 5690.

607. Messrs. Lookabaugh and Benton concluded that the State Plan, and the Counties' Emergency Operations Plans presently do not satisfy Criterion L-4, since the plans do not describe the provisions for transportation of radiologically-contaminated individuals to any of the three hospitals identified in the State Plan which will accept radiologically-contaminated individuals for treatment. In addition, the State and Counties' Plans presently do not contain or reference letters of agreement between private ambulance services, and the State and County governments confirming the ambulance services' willingness to handle and transport radiologically contaminated or injured individuals. However, on the basis of verbal commitments Messrs. Lookabaugh and Benton received from officials of the State and two counties, they believe that this inadequacy will be rectified. Lookabaugh and Benton, Tr. 5717-18, 5653. FEMA's finding will be confirmed in FEMA's Preliminary Findings for CPSES. Id.

4. Contention 22(d)

608. Section 12.0 of the Applicants' Emergency Plan sets forth the Applicants' provisions for conducting emergency preparedness exercises and drills. Lancaster, Tr. 5453.

609. The Applicants, Hood and Somervell Counties, the State (including the Bureau of Radiation Control), and Federal agencies will participate in annual exercises. The annual exercises will be observed by representatives from participating State agencies, and Federal agencies. Section 12.0 of the Applicants' Emergency Plan; Attachment 4, Tab 1, "Fixed Nuclear Facility Accidents", Appendix 7, Annex L to the State Plan. Lookabaugh and Benton, Tr. 5754-55; Lancaster, Tr. 5453; Born, Tr. 5521-22.

610. The scope of these annual exercises will be in accordance with requirements identified by FEMA. Exercise scenarios will be designed to test major components of the State's, Counties', and Applicants' Emergency plans, and will be scheduled to demonstrate 24-hour operating capabilities starting at any time of day or night in any type of weather. Scenarios for annual exercises will be developed by the Bureau of Radiation Control in cooperation with the Applicants and the Division of Emergency Management. Other participants will be included in scenario development covering the involvement of their agencies. Lookabaugh and Benton, Tr. 5754-55; Lancaster, Tr. 5453.

611. Following each annual exercise a critique will be conducted, observer comments will be evaluated, necessary changes to appropriate plan elements will be incorporated, and plan updates will be issued. Lookabaugh and Benton, Tr. 5755.

612. Monthly communications drills designed to test the ability of the State system to: (1) receive a simulated message from the Applicants, (2) relay that message from the Department of Public Safety District Office through Department of Public Safety Headquarters Communications, the Division of Emergency Management and the Department of Health's Disaster Response Program, to the Bureau of Radiation Control, and (3) transmit messages to the Bureau in an understandable form. Similar drills may be conducted wherein messages are sent from the Applicants to the Hood and Somervell Counties' governments, either direct or via relay through the Department of Public Safety District Office(s). Lookabaugh and Benton, Tr. 5755-56; Lancaster, Tr. 5453-54.

613. Message verification during Hood and Somervell Counties' drills will be in accordance with the respective county's procedures. Message verification will not be tested in State drills since the initial notification to the State will be via dedicated telephone line from the Applicants to the Department of Public Safety District Office; or will be verified by the Department of Public Safety in accordance with Standing Operating Procedures of the Department. Lookabaugh and Benton, Tr. 5756.

614. Semi-annual Health Physics drills will be conducted in which each four-man shift of the accident assessment team receives, evaluates, and develops recommendations for protective responses to simulated elevated airborne and liquid releases and direct radiation measurements in the environment. Typical drills will include use of the mini-computer in the mobile laboratory, appropriate models, and the computer graphics terminal and printer which will be available in the Applicants' Emergency

Operations Facility. Lookabaugh and Benton, Tr. 5756; Lancaster, Tr. 5454; Born, Tr. 5521-22.

615. Offsite ambulance and medical support services will participate in either periodic drills or the annual exercises, as well as training programs in transporting and handling radiologically injured persons. Linnemann, Tr. 5461; Lancaster, Tr. 5454.

616. Communications between the Applicants, the State and local emergency operations center, and the Radiological Monitoring teams will be tested annually. Monthly communications drills involving Radiological Monitoring teams is not deemed to be necessary, since the communications equipment involved is in daily use by the Department of Public Safety members of those teams. Message content will be familiar to Bureau of Radiation Control team members, and Department of Public Safety team members will be present at all times during actual response to give technical assistance in equipment use. Lookabaugh and Benton, Tr. 5757; Lancaster, Tr. 5454; Born, Tr. 5521-22.

617. Station personnel who are assigned to the radiological monitoring teams will participate in annual drills which involve responses to all aspects of environmental monitoring, both on site and offsite. Lancaster, Tr. 5454.

618. Communications with Federal emergency response organizations will be tested quarterly. Separate monthly exercises or drills are not necessary since the primary communications mode between the Division of Emergency Management and FEMA's Region VI offices in Denton, Texas is by telephone. National Warning System ("NAWAS"), the secondary communications system is tested every day, and the Civil Defense National Radio

System, the tertiary system, is tested on a weekly basis. Lookabaugh and Benton, Tr. 5757; Lancaster, Tr. 5454.

619. Monthly drills involving Radiological Monitoring teams are not necessary because the skills and procedures involved are identical to those used in routine sampling at other locations, and because the Radiological Monitoring team's communications and record keeping systems are in daily use. Lookabaugh and Benton, Tr. 5757-58.

620. Personnel from the Hood and Somervell Counties Sheriff's Departments will participate in periodic drills, annual exercises and site specific emergency response training sessions provided by Applicants personnel at mutually agreed times and locations. The training sessions will include instruction regarding procedures for communication, and familiarization with the Applicants' Emergency Plan and CPSES emergency facilities. Copies of the Letters of Agreements with the sheriffs are included in Appendix H, Section 15.0 of the Applicants' Emergency Plan. Lancaster, Tr. 5455.

621. The Glen Rose-Somervell County Volunteer Fire Department and Ambulance Service has agreed to participate in periodic drills, annual exercises, and site specific training sessions. A copy of that agreement is found at Appendix H, Section 15.0 of the Applicants' Emergency Plan. These training sessions will be conducted by Applicants' personnel and Radiation Management Corp., and will include procedures for notification, basic radiation protection, site access, and emergency response functions. A similar letter of agreement with the Granbury Volunteer Fire Department is being renegotiated and will be added to the Plan when it becomes available. Similarly, Hood General Hospital has agreed that its personnel

will participate in site specific exercises, drills, and training provided by TUGCO and RMC. A copy of that agreement is provided in Appendix H, Section 15.0 of the Applicants' Emergency Plan at Appendix H to Section 15.0. Lancaster, Tr. 5455-56.

622. The Texas Department of Public Safety has agreed to participate in periodic communication drills and exercises involving CPSES. Lancaster, Tr. 5456.

623. Squaw Creek Park, Inc. (SCPI), which operates a 470 acre recreation area adjacent to Squaw Creek Reservoir, has agreed to participate in annual exercises, drills, and site specific training sessions with regard to emergency evacuation of the park and reservoir. SCPI is responsible for controlling access to the park and reservoir and is responsible for accountability and evacuation of the park and reservoir in the event such action is necessary because of an emergency at CPSES. A copy of that agreement is found at Appendix H, Section 15.0 of the Applicants' Emergency Plan. Lancaster, Tr. 5447-48, 5456-57.

624. Messrs. Lookabaugh and Benton concluded that there are adequate provisions in those plans for periodic drills and exercises. Lookabaugh and Benton, Tr. 5758. This finding will be confirmed in FEMA's Preliminary Findings regarding the off-site emergency plans for CPSES. Id.

5. Contention 22(e)

625. Section 1.3.1.4 of the Applicants' Emergency Plan states that injured individuals whose medical treatment is not complicated by radio-

logical considerations may be sent to Hood General Hospital in Granbury, or to Marks English Hospital in Glen Rose. Rohrer, Tr. 5691-92; Linnemann, Tr. 5460, 5461.

626. The Counties of Hood and Somervell have committed to listing supporting medical facilities in Attachment Q to their respective Emergency Operations Plans. This Attachment should identify any medical facilities that are located in Glen Rose. Lookabaugh and Benton, Tr. 5759.

6. Contention 22(f)

627. The City of Glen Rose ("City") is located in Somervell County, and is within the ten-mile EPZ for CPSES. The Somervell County-City of Glen Rose Emergency Operations Plan ("County Plan") is a joint emergency plan containing the emergency planning provisions, including emergency notification and evacuation, for Somervell County and Glen Rose. The County Plan assigns specific responsibilities and tasks to members of the Glen Rose government, and City departments and agencies. Tab 1 to Annex F of the County-City Plan depicts these assignments. Section V of Annex F describes these responsibilities. Lookabaugh and Benton, Tr. 5759-60; Skiles, Tr. 5522, 5569; Born, Tr. 5573-74.

628. In the event that the mayor of Glen Rose is unable to perform his role as emergency director, Part IX of the County Plan establishes a line of succession for Glen Rose running from the mayor to the mayor pro tem to the remaining members of the council. Skiles, Tr. 5523.

629. Annex F to the County Plan establishes the specific response plan for the City of Glen Rose and for Somervell County in the event of an accident at a fixed nuclear facility. The responsibility for emer-

gency operations and the assignment of specific responsibilities for the City of Glen Rose are set forth at Section V, Annex F of the County Plan. The County Plan also contains specific provisions for notifying persons living, working or traveling within the 10-mile EPZ for CPSES (including Glen Rose), and procedures for further contact and possible evacuation. Lookabaugh and Benton, Tr. 5759-60; Skiles, Tr. 5523.

630. In the event of an off-site release warning at CPSES, the County Plan provides for notification of the appropriate City officials and the Marks English Hospital; a determination of the need for evacuation based upon information received from the state and other sources; and procedures for evacuation of persons from affected areas. Skiles, Tr. 5522-23.

631. The Administrator of Marks English Hospital is responsible for coordinating medical care and treatment for injured individuals. Part VI(9), and Annex F, Section V(J) of the County Plan. Specific procedures for coordination of medical treatment are set forth in an Appendix to Section IV, "Manual of Emergency Procedures for Incidents Involving the Comanche Peak Steam Electric Station." These procedures include specific instructions to the Hospital Administrator to be followed in the event of an Unusual Event, an Alert, a Site Area Emergency, and a General Emergency at CPSES. Skiles, Tr. 5523-5524.

632. Messrs. Lookabaugh and Benton concluded that the Somervell County Emergency Operations Plan is adequate with regard to emergency notification and evacuation. Lookabaugh and Benton, Tr. 5760. This finding will be confirmed in FEMA's Preliminary Findings regarding off-site emergency planning for CPSES. Id.

633. Portions of the Dallas/Fort Worth metroplex are within the 50 mile Ingestion Exposure Pathway EPZ. The emergency plan for residents located within the 50 mile EPZ is in the Texas Emergency Management Plan. The primary concern is preventing public ingestion of radioactive contamination from water supplies, and agricultural products produced within the 50 mile EPZ. Tab 1, Appendix 7, Annex L of the State Plan. Lookabaugh and Benton, Tr. 5760; Born, Tr. 5524, 5526, 5534-35, 5574.

634. Protective actions for prevention of public consumption of radioactively-contaminated water supplies and agricultural materials are summarized by Applicant's expert witness, Mr. Born in his written testimony. Tab 1, Appendix 7, Annex L to the State Plan. Born, Tr. 5524-26, 5534-35, 5575.

635. Emergency actions to prevent ingestion of radioactively-contaminated agricultural products will be ordered by the State, after consultation with the United States Department of Agriculture. Coordination of these actions will be effected by the County Judges, the County Agricultural Agents, and County Emergency Boards. Lookabaugh and Benton, Tr. 5760; Testimony of Born, Tr. 5524-25.

636. Mr. Born, the Applicant's expert witness representing the Bureau of Radiation Control, stated that if emergency actions are required beyond the 50-mile Ingestion Exposure Pathway EPZ, as identified by the State's radiological monitoring teams, the State would implement those measures. Both Mr. Born and Mr. Armstrong, Applicants' witnesses representing the Division of Emergency Management Plan, indicated that the State Plan has explicit procedures and provisions for warning individuals

and implementing emergency actions beyond the 50 mile EPZ. Born, Tr. 5546-47, 5569-70, 5573-74; Armstrong, Tr. 5547-48.

637. Messrs. Lookabaugh and Benton concluded that the emergency planning provisions contained in the State Plan for the 50 mile EPZ are adequate. Lookabaugh and Benton, Tr. 5760. FEMA's findings on this subject will be confirmed in FEMA's Preliminary Findings regarding off-site emergency planning for CPSES. Id.

Respectfully submitted,

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Counsel for NRC Staff

Geary S. Mizuno

Geary S. Mizuno
Counsel for NRC Staff

Dated this 24th day of February, 1983
at Bethesda, Maryland

APPENDIX A

NRC STAFF EXHIBITS (CONTENTION 5), COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

The following NRC Staff Exhibits were marked for identification and/or received into evidence:

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
1	Safety Evaluation Report related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797), July, 1981.	290	291
2	Supplement No. 1, Safety Evaluation Report related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797, Supplement 1), October, 1981	290	291
3	Final Environmental Statement related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0775), September 1981	290	291
4	NRC Staff Testimony of Jim C. Petersen Regarding Financial Qualifications (Contention 25)	297	301
5	NRC Staff Testimony of John G. Spraul Regarding Operating Quality Assurance (Board Question No. 2)	646	648
6	Professional Qualifications of William A. Crossman	1262	1263
7	Professional Qualifications of Robert C. Stewart	1264	1264
8	Professional Qualifications of Joseph I. Tapia	1266	1266
9	Professional Qualifications of Robert G. Taylor	1267	1267
10	Letter from NRC to TUGCO re NRC Inspection Report 75-05 4/22/75	1275	Order (January 31, 1983)

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
11	NRC Inspection Report 75-06 4/30/75	1277	Order (January 31, 1983)
12	NRC Inspection Report 75-07 6/11/75	1277	Order (January 31, 1983)
13	NRC Staff's Testimony prefiled on May 24, 1982	2315	2336
13A	Federal Register Notice, "Proposed General Statement of Policy and Procedure for Enforcement Actions," 45 Fed. Reg. 66754 (October 7, 1980)	2315	2336
13B	Federal Register Notice, "General Statement of Policy and Procedure for Enforcement Actions," 46 Fed. Reg. 9987 (March 9, 1982)	2315	2336
14	NRC Inspection Report 78-23 1/5/79	2315	2336
15	NRC Inspection Report 81-20 4/27/82	2315	2336
16	Letter from NRC to TUGCO re Findings in Inspection Report 75-05 6/4/75	2315	2336
17	Letter from Applicants to NRC re Findings in Inspection Report 75-06 5/29/75	2315	2336
18	Letter from Applicants to NRC re Findings in Inspection Report 75-06 6/2/75	2315	2336
19	Letter to Applicants from NRC re Findings in NRC Inspection Report 75-06 6/6/75	2315	2336
20	NRC Inspection Report 75-10 8/7/75	2315	2336
21	Letter from Applicants to NRC re Findings in NRC Inspection Report 75-10 9/5/75	2315	2336
22	Letter to Applicants from NRC re Findings in NRC Inspection Report 75-10 9/10/75	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
23	NRC Inspection Report 75-12 10/01/75	2315	2336
24	NRC Inspection Report 75-13 12/6/75	2315	2336
25	Letter from Applicants to NRC re Findings in NRC Inspection Report 75-13 1/15/76	2315	2336
26	Letter to Applicants from NRC re Findings in Inspection Report 76-13 2/12/76	2315	2336
27	NRC Inspection Report 76-03 4/1/76	2315	2336
28	NRC Inspection Report 76-01 1/29/76	2315	2336
29	Letter from TUGCO to NRC re Findings in NRC Inspection Report 76-01 2/27/76	2315	2336
30	Letter from NRC to TUGCO re Findings in NRC Inspection Report 76-01 3/10/76	2315	2336
31	NRC Inspection Report 76-07 8/5/76	2315	2336
32	Letter from TUGCO to NRC re Findings in NRC Inspection Report 76-07 9/2/76	2315	2336
33	Letter from NRC to TUGCO re Findings in NRC Inspection Report 76-07 9/17/76	2315	2336
34	NRC Inspection Report 76-10 10/14/76	2315	2336
35	NRC Inspection Report 78-07 5/10/78	2315	2336
36	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-07 6/5/78	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
37	Letter from NRC to TUGCO re Findings in Inspection Report 78-07 6/20/78	2315	2336
38	NRC Inspection Report 78-20 12/18/78	2315	2336
39	DELETE		
40	DELETE		
41	NRC Inspection Report 78-13 9/19/78	2315	2336
42	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-13 TXX 2895 10/18/78	2315	2336
43	Letter from NRC to TUGCO re Inspection Report 78-13 10/24/78	2315	2336
44	NRC Inspection Report 78-16 11/17/78	2315	2336
45	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-16 12/8/78	2315	2336
46	Letter from NRC to TUGCO re Findings in NRC Inspection Report 78-16 12/21/78	2315	2336
47	NRC Inspection Report 78-22 1/10/79	2315	2336
48	Letter from NRC to TUGCO re NRC Inspection Report 79-03 2/20/79	2315	2336
49	NRC Inspection Report 79-03 3/14/79	2315	2336
50	Letter from TUGCO to NRC re Findings in NRC Inspection Report 79-03 3/6/79	2315	2336
51	Letter from NRC to TUGCO re Findings in NRC Inspection Report 79-03 3/13/79	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
52	NRC Inspection Report 79-06 5/10/79	2315	2336
53	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-08 4/2/80	2315	2336
54	NRC Inspection Report 80-08 4/18/80	2315	2336
55	Letter from TUGCO to NRC re Findings in NRC Inspection Report 80-08 4/21/80	2315	2336
56	Letter from NRC to TUGCO re NRC Inspection Report 80-08 4/30/80	2315	2336
56A	Letter from TUGCO to NRC re Findings in NRC Inspection Report 80-08 TXX 3142 5/14/80	2315	2336
56B	Letter from NRC to TUGCO re NRC Inspection Report 80-08 5/22/80	2315	2336
57	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-11 4/9/80	2315	2336
58	NRC Inspection Report 80-11 6/17/80	2315	2336
59	NRC Inspection Report 77-02 3/23/77	2315	2336
60	Letter from TUGCO to NRC re Findings in NRC Inspection Report 77-02 4/25/77	2315	2336
61	Letter from NRC to TUGCO re Findings in NRC Inspection Report 77-02 5/11/77	2315	2336
62	NRC Inspection Report 77-06 5/27/79	2315	2336
63	NRC Inspection Report 79-09 6/8/79	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
64	NRC Inspection Report 79-11 5/14/79	2315	2336
65	Letter from TUGCO to NRC re Findings in NRC Inspection Report 79-11 6/12/79	2315	2336
66	Letter from TUGCO to NRC re Findings in NRC Inspection Report 79-11 9/17/79	2315	2336
67	Letter from NRC to TUGCO re NRC Inspection Report 79-11 7/5/79	2315	2336
68	Letter from NRC to TUGCO re NRC Inspection Report 79-11 10/10/79	2315	2336
69	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 79-18 9/24/79	2315	2336
70	NRC Inspection Report 79-18 11/9/79	2315	2336
71	NRC Inspection Report 79-24/79-23 11/27/79	2315	2336
72	NRC Inspection Report 79-20 10/5/79	2315	2336
73	NRC Inspection Report 79-07 3/23/79	2315	2336
74	NRC Inspection Report 79-13 6/18/79	2315	2336
75	NRC Inspection Report 77-10 10/19/77	2315	2336
76	Letter from TUGCO to NRC re Findings in NRC Inspection Report 77-10 11/17/77	2315	2336
77	Letter from TUGCO to NRC re Findings in NRC Inspection Report 77-10 1/3/78	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
78	Letter from NRC to TUGCO re NRC Inspection Report 77-10 12/14/77	2315	2336
79	DELETE		
80	NRC Inspection Report 78-01 2/1/78	2315	2336
81	NRC Inspection Report 78-05 4/6/78	2315	2336
82	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-05 5/04/78	2315	2336
83	Letter from NRC to TUGCO re NRC Inspection Report 78-05 5/12/78	2315	2336
84	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-05 7/18/78	2315	2336
85	Letter from NRC to TUGCO re NRC Inspection Report 78-05 7/24/78	2315	2336
86	NRC Inspection Report 78-12 9/7/78	2315	2336
87	Letter from NRC notifying Applicants of NRC Inspection Reports 78-13/78-12 8/30/78	2315	2336
88	Letter from TUGCO to NRC re Findings in NRC Inspection Reports 78-12/78-13 9/21/78	2315	2336
89	Letter from NRC to TUGCO re NRC Inspection Reports 78-1 and 78-13 10/10/78	2315	2336
90	Letter from TUGCO to NRC re Findings in NRC Inspection Reports 78-12/78-13 11/6/78	2315	2336
91	Letter from NRC to TUGCO re NRC Inspection Reports 78-12 and 78-13 11/17/78	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
92	NRC Inspection Report 79-01 2/22/79	2315	2336
93	NRC Inspection Report 78-18 10/20/78	2315	2336
94	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-18 11/16/78	2315	2336
95	Letter from NRC to TUGCO re NRC Inspection Report 78-18 12/8/78	2315	2336
96	Letter from TUGCO to NRC re Findings in NRC Inspection Report 78-18 and NRC requests for information 12/20/78	2315	2336
97	Letter from NRC to TUGCO re NRC Inspection Report 78-18 1/4/79	2315	2336
98	Letter from TUGCO to NRC re modification earlier committments concerning NRC Inspection Report 78-18 10/1/79	2315	2336
99	Letter from NRC to TUGCO re modification of earlier committments concerning NRC Inspection Report 78-18 11/2/79	2315	2336
100	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-18 11/29/29	2315	2336
101	Letter from NRC to TUGCO re NRC Inspection Report 79-18 1/16/80	2315	2336
102	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-01 1/23/80	2315	2336
103	NRC Inspection Report 80-01 2/15/80	2315	2336
104	NRC Inspection Report 79-31/79-29 1/29/80	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
105	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-31/79-29 2/20/80	2315	2336
106	Letter from NRC to TUGCO re NRC Inspection Report 79-31/79-29 3/11/80	2315	2336
107	NRC Inspection Report 80-13 5/21/80	2315	2336
108	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-01 2/19/80	2315	2336
109	Letter from NRC to TUGCO re NRC Inspection Report 80-01 2/27/80	2315	2336
110	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-13 6/6/80	2315	2336
111	Letter from NRC to TUGCO re NRC Inspection Report 80-13 6/19/80	2315	2336
112	NRC Inspection Report 80-18 9/19/80	2315	2336
113	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-23 10/16/80	2315	2336
114	NRC Inspection Report 80-23 11/19/80	2315	2336
115	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-23 11/12/80	2315	2336
116	Letter from NRC to TUGCO re NRC Inspection Report 80-23 12/23/80	2315	2336
117	NRC Inspection Report 81-14 7/30/81	2315	2336
118	NRC Inspection Report 80-17 7/31/80	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
118A	DELETE		
118B	DELETE		
118C	NRC Inspection Report 82-03/82-02 6/14/82	2315	2336
119	NRC Inspection Report 79-12 8/22/79	2315	2336
120	NRC Inspection Report 79-15 7/2/79	2315	2336
121	NRC Inspection Report 79-22/79-21 11/27/79	2315	2336
122	NRC Inspection Report 80-02 3/20/80	2315	2336
123	NRC Inspection Report 80-22 1/21/81	2315	2336
124	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-20 9/24/80	2315	2336
125	NRC Inspection Report 80-20 10/21/80	2315	2336
126	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-20 10/20/80	2315	2336
127	Letter from NRC to TUGCO re NRC Inspection Report 80-20 11/5/80	2315	2336
128	NRC Inspection Report 77-13 12/20/77	2315	2336
129	NRC Inspection Report 78-11 6/30/78	2315	2336
130	Letter from TUGCO to NRC re findings in NRC Inspection Report 78-11 7/28/78	2315	2336
131	Letter from NRC to TUGCO re NRC Inspection Report 78-11 8/4/78	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
132	NRC Inspection Report 79-04 2/27/79	2315	2336
133	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-04 3/20/79	2315	2336
134	Letter from TUGCO to NRC supplementing 3/20/1974 letter re NRC Inspection Report 79-04 4/12/79	2315	2336
135	Letter from NRC to TUGCO re NRC Inspection Report 79-04 4/2/79	2315	2336
136	Letter from NRC to TUGCO re NRC Inspection Report 79-04 4/26/79	2315	2336
137	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-06 6/5/79	2315	2336
138	Letter from NRC to TUGCO re NRC Inspection Report 79-06 6/20/79	2315	2336
139	NRC Inspection Report 79-19 9/7/79	2315	2336
139A	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-19 TXX-3049 9/27/79	2315	2336
139B	Letter from NRC to TUGCO re NRC Inspection Report 79-19 10/10/79	2315	2336
140	NRC Inspection Report 79-28/79-27 1/11/80	2315	2336
141	Letter from TUGCO to NRC re findings in NRC Inspection Reports 79-28/79-27 2/8/80	2315	2336
142	Letter from NRC to TUGCO re NRC Inspection Reports 79-28/79-27 2/20/80	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
143	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-11 5/5/80	2315	2336
144	DELETE		
145	Letter from NRC to TUGCO re NRC Inspection Report 81-15 10/26/81	2315	2336
146	NRC Inspection Report 81-15 11/6/81	2315	2336
147	Letter from TUGCO to NRC re findings in NRC Inspection Report 81-15 TXX-3439 11/19/81	2315	2336
148	Letter from NRC to TUGCO re NRC Inspection Report 81-15 1/19/82	2315	2336
148A	NRC Inspection Report 82-09/82-04 7/6/82	2315	2336
148B	Letter from NRC to TUGCO re NRC Inspection Report 82-11 7/7/82	2315	2336
149	NRC Inspection Report 81-04 5/5/81	2315	2336
150	NRC Inspection Report 76-08 8/20/76	2315	2336
151	Letter from TUGCO to NRC re findings in NRC Inspection Report 76-08 TXX-1913 9/21/76	2315	2336
152	Letter from NRC to TUGCO re: additional questions pursuant to NRC Inspection Report 76-08 10/8/76	2315	2336
153	Letter from TUGCO to NRC re: additional answers to questions pursuant to NRC Inspection Report 76-08 TXX-1998 10/26/76	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
154	Letter from NRC to TUGCO re NRC Inspection Report 76-08 11/23/76	2315	2336
155	NRC Inspection Report 76-11 11/24/76	2315	2336
156	Letter from NRC to TUGCO notifying Applicants of NRC Inpsection Report 79-27/79-26 11/15/79	2315	2336
157	Letter from NRC to TUGCO notifying Applicants of NRC Inpsection Report 79-27/79-26 11/21/79	2315	2336
158	NRC Inspection Report 79-27/79-26 1/8/80	2315	2336
159	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-27/79-26 TXX-3080 12/18/79	2315	2336
160	Letter from TUGCO to NRC re findings in NRC Inspection Report 79-27/79-26 TXX-3081 12/21/79	2315	2336
161	Letter from NRC to TUGCO re NRC Inspection Report 79-27/79-26 1/2/80	2315	2336
162	Letter from NRC to TUGCO re NRC Inspection Report 79-27/79-26 1/11/80	2315	2336
163	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-03 2/7/80	2315	2336
164	NRC Inspection Report 80-03 3/20/80	2315	2336
165	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-03 TXX-3105 3/5/80	2315	2336
166	Lette. from NRC to TUGCO re NRC Inspection Report 80-03 3/11/80	2315	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
167	Letter from TUGCO to NRC re findings on NRC Inspection Report 80-08 5/14/80	2315	2336
168	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 80-15 6/23/80	2315	2336
169	NRC Inspection Report 80-15 7/23/80	2315	2336
170	Letter from TUGCO to NRC re findings in NRC Inspection Report 80-15 TXX-3177 8/18/80	2315	2336
171	Letter from NRC to TUGCO re NRC Inspection Report 80-15 9/12/80	2315	2336
172	NRC Inspection Report 80-27 1/15/81	2315	2336
173	Letter from NRC to TUGCO notifying Applicants of NRC Inspection Report 81-02 2/25/81	2315	2336
174	NRC Inspection Report 81-02 3/20/81	2315	2336
175	Letter from TUGCO to NRC re findings in NRC Inspection Report 81-02 TXX-3289 3/19/81	2315	2336
176	Letter from NRC to TUGCO re NRC Inspection Report 81-02 4/2/81	2315	2336
177	NRC Inspection Report 81-05 4/17/81	2315	2336
177A	NRC Inspection Report 79-25/79-24	2315	2336
178	NRC Inspection Report 81-12 4/16/82	2315	2336
179	Errata and Addenda to Staff's Prefiled Written Direct Testimony of May 24, 1982	2316	2324

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
180	Supplemental Direct Testimony of Crossman/Stewart/Taylor	2320	2336
181	Inspection Report 80-25	2320	2336
182	Memo from W. Seidle to W. Crossman re: Trend Analysis for 1976 1/4/77	2320	2336
183	Memo from W. Crossman to W. Hubacek, R.G. Taylor, R.C. Stewart and C.R. Oberg re 1976 Trend Analysis 1/14/77	2320	2336
184	Comanche Peak Trend Analysis - 1976	2320	2336
185	Memo from W. Crossman to W.G. Hubacek, R.G. Taylor, R.C. Stewart and C.R. Oberg re 1977 Trend Analysis 1/6/78	2320	2336
186	Memo from W. Seidle to K.V. Seyfrit re use of licensee performance evaluation information 10/6/78	2320	2336
187	Comanche Peak Trend Analysis for 1977	2320	2336
188	Memo from W. Seidle, re Trend Analysis for 1978 2/1/79	2320	2336
189	Memo from W. Crossman to W.G. Hubacek, R.C. Stewart, R.G. Taylor and C.R. Oberg re Trend Analysis for 1978 2/2/79	2320	2336
190	Memo from R.C. Stewart to W. Crossman, re Trend Analysis - 1978 10/19/79	2320	2336
191	Trend Analysis - 1978	2320	2336
192	Memo from W. Seidle to W. Crossman, re Trend Analysis - 1979 1/4/80	2320	2336
193	Memo from W. Crossman to W.G. Hubacek, R.C. Stewart, R.G. Taylor, C.R. Oberg, H.S. Phillips, re Trend Analysis, 1979 1/17/80	2320	2336

EXHIBIT NUMBER		IDENTIFIED	ADMITTED
194	Memo from C.R. Oberg to W. Crossman, re Trend Analysis, 1979 3/3/80	2320	2336
195	Trend Analysis, 1979	2320	2336
196	NRC Inspection Report 80-04 2/29/80	2320	2336
197	Direct Testimony of Taylor/Driskill	2461	2461
198	Professional Qualifications of Driskill	2456	2456
199	NRC Inspection Report 82-10/82-05 7/7/82	2461	2461
200	Deposition of Charles Atchison 6/21/82	3403	3433
201	NRC Staff Testimony of Joseph I. Tapia and W. Paul Chen in Rebuttal to the Testimony of Mark Anthony Walsh Concerning the Design of Pipe Supports	5327	
201A	Professional Qualifications Statement of W. Paul Chen	5330	5332
201B	NRC Inspection Report 82-05 5/27/80	5332	5332
201C	NRC Inspection and Enforcement Bulletin 79-14 7/2/79	5560	5561
202	NRC Staff Testimony of David M. Rohrer Regarding Emergency Planning (Contention 22)	5666	5679
202A	Professional Qualifications Statement of David M. Rohrer	5666	5679
203	FEMA Staff Testimony of Albert Lookabaugh and John Benton Regarding Emergency Planning	5698	5743
203A	Professional Qualifications Statement of John Benton	5698	5743
203B	Professional Qualifications Statement of Albert Lookabaugh	5698	5743