August 5, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

APPLICANTS' PROPOSED FINDINGS OF FACT IN THE FORM OF A PARTIAL INITIAL DECISION

In accordance with 10 C.F.R. §2.754, Texas Utilities Generating Company, et al. ("Applicants") hereby submit proposed findings of fact in the form of a partial initial decision on pipe support design questions. Applicants intend to respond to the findings of the other parties in accordance with 10 C.F.R. §2.754(a)(3).

Respectfully submitted,

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[Applicants' Proposed Findings of Fact and Conclusions of Law in the Form of a]

PARTIAL INITIAL DECISION

(Concerning Pipe Support Design Questions)

I. BACKGROUND

A. General

This Partial Initial Decision concerns allegations regarding the design of pipe supports at the Comanche Peak Steam Electric Station, Units 1 and 2. This is the second decision we have issued in this proceeding. The first, issued as a proposed partial initial decision on July 29, 1983, concerned portions of Contentions 5 (QA/QC) and 22 (Emergency Planning), Board Questions 1-3 and the deletion of the boron injection tank. The instant decision addresses allegations raised during this proceeding regarding various aspects of the design of pipe supports at Comanche Peak. While these allegations were not clearly within the scope of any admitted contention, this Board (as previously constituted[1]) permitted their litigation. We

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1 Since these allegations were first raised, two of the three positions on this Board have changed hands.

conclude in this decision that there is reasonable assurance that pipe supports designed for and constructed at Comanche Peak will perform their intended functions. The allegations raised regarding those designs were shown through testimony of highly qualified and experienced experts of both Applicants and the NRC Staff to be without merit, to involve impacts so insignificant that based on sound engineering judgment they need not be evaluated, or to have already been accounted for in Applicants' pipe support design program. Accordingly, we find that these designs satisfy all applicable codes and standards regarding the design of component supports for nuclear power reactors.

B. Witnesses and Testimony

1. CASE

CASE presented two witnesses to testify regarding the design of pipe supports at Comanche Peak. Mr. Mark A. Walsh, who initially appeared to present a limited appearance statement, was given leave by the Board to submit testimony regarding his allegations. Mr. Walsh presented his testimony and was cross-examined at those hearings (CASE Exhibit 659). Mr. Walsh also submitted supplementary testimony at the September 1982 hearing session (CASE Exhibit 668). CASE also presented testimony of Mr. Jack Doyle. Upon agreement of the parties, the Board permitted Mr. Doyle's deposition to be submitted as testimony, including the cross-examination conducted by Applicants during the deposition (CASE Exhibits 669 and 669A; Tr. 3592). Mr. Doyle also supplemented his deposition with testimony at the September hearing (CASE Exhibit 683). Both witnesses presented surrebuttal

testimony to the NRC Staff's rebuttal testimony at the May 1983 hearings (CASE Exhibits 761-763, 805 (Mr. Doyle) and 766 (Mr. Walsh)).

Mr. Walsh tendered himself as an expert in structural engineering (Tr. 3084). Mr. Walsh is a degreed engineer although not a registered professional engineer (CASE Exhibit 659A; Tr. 3091). He had approximately two and one-half years experience working on pipe support designs (Tr. 3092-93; CASE Exhibit 659A). Mr. Walsh had been employed at Comanche Peak in the STRUDL Group (see Section II.B.3, below), a service group assigned the task of inputting key parameters of pipe support designs into the STRUDL computer program, checking that the data printed on the output correctly reflected the support model, and returning the computer output to the designer for his evaluation (Applicants' Exhibit 142 at 9-10; Tr. 3095). Mr. Doyle also had been employed at Comanche Peak in the STRUDL Group (Tr. 3887). His only formal education beyond the high school level was one semester at a junior college (Tr. 3867-68).

2. Applicants

In response to the allegations made by Messrs. Walsh and Doyle, Applicants presented at the September, 1982 hearing a panel of five witnesses with expertise in pipe support design and related fields. These witnesses submitted written testimony regarding the pipe support design allegations and were cross-examined (Applicants' Exhibits 142 and 142F). In addition, oral direct testimony was presented by three of these witnesses and another individual as a panel in the May 1983 hearing.

Applicants presented Mr. Kenneth L. Scheppele as an expert in structural engineering (Applicants' Exhibit 142 at 1). Mr. Scheppele is Senior Vice President of the architect/engineer for Comanche Peak, Gibbs & Hill, Inc., and is a registered professional engineer. His qualifications in the field of structural engineering are extensive. (Applicants' Exhibit 19; Tr. 3086.)

Applicants also presented Mr. Roger F. Reedy as an expert in the development, interpretation and application of the ASME Code with regard to general requirements, materials, fabrication, examinations, design and analysis. Mr. Reedy has extensive experience in his field of expertise. He is a registered structural engineer in Illinois and a registered professional engineer (civil) in five states. He has been involved in the design of components for nuclear power plants since 1956. He has served as the responsible registered professional engineer for the design of nuclear reactor vessels, containment vessels, piping and supports. He has been chairman of the ASME Section III Code Committee since early 1977. He assisted in the development of Section III prior to its publication in 1963 and has been a member of the ASME Code Committee since 1969. He personally compiled the Code rules and Subsections NC, ND and NE for inclusion in the 1974 Code Edition, and provided guidance to the task group developing the rules for Subsection NF prior to its adoption into Section III. Mr. Reedy was a founding member of the ASME Pressure Vessel and Piping Division and Chairman of the Professional Division in 1979. In 1982, Mr. Reedy was awarded the honor of ASME Life Fellow because of his ASME Code work and design developments for multi-layered vessels. (Applicants' Exhibits 142 at 2-4; 41.)

Dr. Peter S. Y. Chang was presented by Applicants as an expert in pipe support engineering and STRUDL analysis. Dr. Chang has a Ph.D. in civil engineering and a registered professional engineer. Dr. Chang is the Chief Engineer, Pipe Support Engineering for Comanche Peak. He has eleven years of practical experience in the design and analysis of power plant structures, the last nine years being on nuclear plants. He is experienced in the application of the ASME Code, Section III to containment vessel, pipe stress and pipe support analysis and design. Dr. Chang is experienced in the development of computer programs for modelling static, thermal, seismic and other transient loadings for nuclear power plants. His experience with the application of the STRUDL Code has included advanced lectures and seminars on STRUDL, in addition to graduate course work in topics related to STRUDL analysis. Dr. Chang served as a supervising engineer responsible for structural analysis and design for static, thermal, seismic and other loads for all safety-related buildings at another nuclear project. Since coming to Comanche Peak in 1981, he has been responsible for small bore ASME pipe stress analysis and ASME NF pipe support design. (Applicants' Exhibit 142 at 4-5; 142A.)

Mr. John C. Finneran, Jr. presented testimony for Applicants as an expert in structural engineering. Mr. Finneran has Bachelor's and Master's degrees in Civil Engineering and is a member of the American Society of Civil Engineers. He is a registered professional engineer. Mr. Finneran is the Pipe Support Engineering Supervisor for Comanche Peak. He has several

years experience in structural engineering in design and analysis of substation and transmission structures for power plants, and he has been a supervisor of structural engineering groups at Comanche Peak for three years. (Applicants' Exhibits 142 at 7; 142B.)

Also, Mr. Gary Krishnan was presented by Applicants as an expert in pipe stress analysis. Mr. Krishnan is the Site Stress Analysis Group Supervisor for Comanche Peak. Mr. Krishnan has Bachelor's and Master's degrees in mechanical engineering. His Master's degree is in the area of stress analysis. He has eight years experience in pipe stress analysis at nuclear facilities. He has been a Senior Engineer for Gibbs & Hill for three years, performing pipe stress analyses of safety class piping. (Applicants' Exhibits 142 at 8-9; 142C.)

Finally, Applicants presented Mr. Michael A. Vivirito as an expert in structural engineering (on a panel with Messrs. Reedy, Finneran and Chang) during the May 1983 hearings to testify in response to NRC Staff testimony and the surrebuttal testimony of CASE's witnesses. Mr. Vivirito is the Vice President - Power Engineering of Gibbs & Hill. Mr. Vivirito is a registered professional engineer and has thirty-five years experience in structural engineering, including 17 years experience in the design and construction of nuclear power reactor facilities. He is a member of the American Society of Civil Engineers and has served on numerous professional committees. (Applicants' Exhibit 154.)

3. NRC Staff

The NRC initially presented in the September 1982 hearings two witnesses to address the pipe support design allegations. Mr. Joseph I. Tapia and Dr. W. Paul Chen, submitted prefiled testimony on this matter (identified as NRC Exhibit 201), but because they had not had an opportunity to complete their review of Mr. Doyle's allegations, the Board suspended the taking of evidence on that question until such time as the Staff was prepared to proceed (Tr. 5407, 5410). Upon completion of its review of the pipe support design allegations, the Staff issued an inspection report (I&E Report 82-26/82-14, cover letter dated February 15, 1983). That report was received into evidence at the May 1983 hearings (NRC Exhibit 207). The Staff also submitted the testimony of Mr. Tapia and Dr. Chen regarding pipe support design, and supplemental testimony of Messrs. Tapia, Spottswood Burwell, Robert G. Taylor and Drs. Chen and Jai Raj N. Rajan on the same topic, as well as with respect to the NRC Construction Appraisal Inspection Team ("CAT") report for Comanche Peak (NRC Staff Testimony and Supplemental Testimony, following Tr. 6402). In addition, the Staff presented the testimony of Mr. A. B. Beach, as a member of the CAT, regarding the pipe support findings of the CAT (following Tr. 6283).[2]

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[2] The Construction Appraisal Team is an NRC-commissioned team of inspectors who are charged with conducting reviews of the adequacy of construction at facilities nearing completion. This team presented testimony at the June 1983 hearing regarding its findings, and our decision on the CAT Report will be issued at a later time. We address in this decision only those aspects of the CAT Report (NRC Exhibit 206) that concern pipe supports.

Mr. Tapia is the Reactor Inspector in the Engineering Section of the Division of Resident, Reactor Projects and Engineering Programs, NRC Region IV. He had held this position since 1976. Mr. Tapia has Bachelor's and Master's Degrees in Civil Engineering. Mr. Tapia is a member of the American Society of Civil Engineers; the International Society of Soil, Mechanics and Foundation Engineering; and the American Concrete Institute, serving as a member of that Institute's Committee on Quality Assurance Systems for Concrete. (NRC Exhibit 8.)

Dr. Chen is the Manager of the Stress Analysis Unit of the Systems Engineering Department of the Energy Technology Engineering Center, a U.S. Department of Energy Laboratory. Dr. Chen has Bachelor's and Master's Degrees in Civil Engineering and Applied Mechanics, and a Ph.D. in theoretical and applied mechanics. Dr. Chen is responsible for the technical review of portions of the FSAR, including the pipe support stress analysis performed by Applicants. Dr. Chen has extensive experience in areas relating to material properties and stress analysis. He is responsible for performance of ASME compliance analysis of piping and components for ETEC. (Chen Statement of Qualifications, Attached to NRC Staff Testimony following Tr. 6402.)

Mr. Burwell is the NRC Operating License Project Manager for Comanche Peak. He is responsible for managing the participating in the safety and environmental reviews, analyses and evaluations associated with licensing actions at Comanche Peak. Mr. Burwell has Bachelor's and Master's Degrees in Mechanical Engineering, and is a registered professional engineer. Mr. Burwell has

extensive experience in the design and construction of components for nuclear power reactors. He has worked at the NRC since 1969. (Burwell Statement of Qualifications, attached to NRC Staff Supplemental Testimony, following Tr. 6402)

Dr. Rajan is the mechanical engineer responsible for reviewing and evaluating safety analysis reports with regard to the dynamic analysis and testing of safety-related systems and components, and the criteria for protection against the dynamic effects associated with postulated failures of fluid systems for nuclear facilities. Dr. Rajan has Bachelor's Degrees in physics, mathematics and chemistry and civil engineering; a Master's Degree in applied mechanics and a Ph.D. in Fluid Mechanics. He has extensive experience in the design, analysis, testing and evaluation of fluid piping systems and power fluid systems of nuclear reactors. He has authored papers in various professional journals, and is a part-time professor in the fields of mechanics, materials, fluid mechanics and applied mechanics. (Rajan Statement of Qualifications, attachment to NRC Supplemental Testimony, following Tr. 6402.)

Mr. Taylor is the Resident Reactor Inspector at Comanche Peak, a position he has held since 1978. He is responsible for conducting and coordinating all safety-related inspection efforts by the NRC Region at the site. Mr. Taylor is a registered professional engineer, specializing in quality control engineering. Mr. Taylor has thirty years of experience in the quality engineering field, including fifteen years of quality assurance and reactor inspection in the nuclear power reactor

field. Mr. Taylor joined the NRC is 1976 and served as the reactor inspector at two other power reactors prior to being assigned to Comanche Peak. (NRC Exhibit 9.)

C. Observations

The issues raised regarding pipe support design involve highly technical questions of structural mechanics and interpretation of applicable codes, regulations and guidance. In addition, we have found that the analysis of pipe support designs requires exercise of sound engineering judgment in many areas where precise analytical techniques are not otherwise established. With respect to each of these considerations, Applicants' and the NRC Staff's witnesses exhibited substantial training and expertise. Their testimony was founded on years of experience in fields related to all aspects of these allegations. Accordingly, we consider them to be experts in their fields, and have placed substantial weight on their judgment.

CASE's witnesses on the other hand had relatively limited experience and/or little formal training in these areas. In addition, they exhibited little capacity for exercising judgment as to the significance of their assertions in terms of the adequacy of pipe support design of the supports. We consider such a capacity to be crucial to the presentation of a meaningful case. While many of CASE's assertions were relevant to pipe support design, they were not material, i.e., important, to the adequacy of those designs. Nonetheless, bearing in mind our responsibility to protect the public health and safety, we have examined these allegations as if they were all significant.

II. PIPE SUPPORT DESIGN ALLEGATIONS

A. Thermal Stress

The principal question raised regarding the design of pipe supports at Comanche Peak was whether stresses within linear-type pipe supports resulting from the restraint of thermal expansion of the support (induced by elevated temperatures in a loss-of-coolant accident ("LOCA") environment) must be evaluated in the design of individual supports. Each of the parties presented extensive testimony[3] and briefs[4] regarding the applicable requirements of various codes and regulations, and provisions of the Standard Review Plan and NRC Regulatory Guides. In view of the importance of resolving this issue at as early a date as possible, we separated this issue from the remaining pipe support design questions and issued a Memorandum and Order addressing the provisions of NRC Regulations and guidance, as well as the ASME Code regarding the consideration of thermal stresses.[5] We do not

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3 See discussion supra in Section I. B.

4 Each party submitted three briefs regarding this matter. An initial brief was filed by each party on April 20, 1983. Subsequently, CASE, the Staff and Applicants submitted reply briefs (dated May 3, 4 and 5, 1983, respectively) to respond to matters raised in the initial briefs and to answer a particular question we felt was left unresolved in our reading of those briefs. Finally, we again requested the parties to file briefs regarding the relationship of certain provisions in the ASME Code. CASE's brief on this question was filed on May 9, 1983, and Applicants' and Staff's briefs on May 11, 1983.

5 Memorandum and Order (Thermal Stress in Pipe Supports), July 6, 1983.

here reiterate the extensive arguments and our conclusions on the issue. Rather, we incorporate our Memorandum and Order by reference.

For the reasons stated in our Memorandum and Order, we find that Applicants and the NRC Staff were correct in their position that thermal stress created in linear pipe supports by the restraint of the thermal expansion of those supports under LOCA conditions need not be evaluated in the design of those supports. The assertions of CASE's witnesses on this matter are, therefore, incorrect. Accordingly, we find that Applicants' approach to the consideration of thermal stresses in the design of pipe supports at Comanche Peak is valid from both technical and regulatory viewpoints. We conclude that there is no basis for concern regarding the adequacy of the designs for those supports.

B. Organizational Interfaces

CASE's witnesses alleged that the organizational interfaces between the pipe support design groups for Comanche Peak were inadequate and led to the existence of design inconsistencies (Tr. 3706, 3852, 3864, 3925 and 3973). The scope of this allegation concerns both the interfaces between the pipe support design organizations and the interfaces between those organizations and the analysis service group in which the CASE witnesses were employed. The purported bases for this allegation are the provisions of 10 C.F.R. Part 50, Appendix B, and ANSI Standards N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants," and N45.2.11, "Quality Assurance Requirements for Design of Nuclear Power Plants." (CASE Exhibit

687 and Applicants' Exhibit 148.) Specifically, CASE's witnesses argued that there were inadequate measures for the identification and control of design interfaces and for coordination among participating design organizations (Tr. 6986-93).

We find, for the reasons discussed below, that the Applicants have implemented detailed procedures establishing design interfaces which satisfy all applicable requirements. We further find that the witnesses for the intervenor were simply unaware of the procedures established for design interfaces because their job positions provided them no exposure to or responsibility for the implementation of those procedures.

The NRC has endorsed the use of ANSI N45.2 and N45.2.11 in Regulatory Guides 1.28 and 1.64, respectively. The N45.2 standard is a general requirements document which provides a basis for complying with the provisions of 10 C.F.R. Part 50, Appendix B. The N45.2.11 standard is applicable in part to the specific design control measures described in 10 C.F.R. Part 50, Appendix B, Criterion III, and thus may be referred to for more detailed guidance regarding that provision.

In applicable part, 10 C.F.R. Part 50, Appendix B, Criterion III provides, as follows, with respect to design interfaces:

> Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release distribution and revision of documents involving design interfaces.

ANSI N45.2.11 establishes criteria for the control of both external (relationships between design groups from different companies) and internal (relationships between design groups or organizations within a company) design interfaces. As discussed below, the evidence of record clearly demonstrates that Applicants have satisfied all requirements concerning both external and internal design interface control.

1. Pipe Support Design Organizations

The responsibility for the design of pipe supports at Comanche Peak is assigned to three organizations: ITT-Grinnell, NPS Industries ("NPSI") and the Pipe Support Engineering ("PSE") Group of Comanche Peak Project Engineering. The evolution of the assignment of pipe support design responsibility to various organizations occurred over several years as Applicants determined that additional capability was necessary for the design and fabrication of pipe supports. This evolution of the pipe support design effort resulted in the assignment to each of the above organizations the separate and distinct responsibility for the design of pipe supports in particular areas. Specifically, ITT-Grinnell is responsible for the design of pipe supports in buildings associated with Unit 1 and common areas. The pipe supports in the containment building itself are the responsibility of NPSI (although a few containment supports were assigned to ITT-Grinnell). In Unit 2, the design responsibility is assigned to NPSI with the exception of certain non-safety related supports of main steamlines on the turbine deck. The PSE Group is responsible for all small-bore (2-inch and under) piping

supports and certain isolated cases of large-bore supports and fire protection piping. (Applicants' Exhibit 142 at 9; NRC Exhibit 207 at 12; Tr. 5277-78.)

2. External Design Control Interfaces

The provisions of 10 C.F.R. Part 50, Appendix B and ANSI Standard N45.2.11 concerning external design interfaces are intended to assure that each design organization has a clear scope of responsibility and that there are documented paths of communication among the organizations when the responsibility for an activity shifts from one organization to another. At Comanche Peak each of the three pipe support design organizations have their own specific scope of responsibility, described above. The Staff testified, and we so find, that there is no need for cross-communication between these design organizations. Accordingly, Applicants need not establish procedures for communications between the design organizations themselves. (NRC Exhibit 207 at 12-13.)

With respect to communications between the pipe support design organizations and other organizations, we observe that each support design organization provides its design information to Applicants' Architect/Engineer ("A/E"), Gibbs & Hill, Inc., or nuclear steam supply system ("NSSS") supplier, Westinghouse Electric Corporation, for utilization in pipe stress analyses. <u>See</u> discussion below in Section II.C 1. This process involves communication between the pipe support design organizations and the A/E or NSSS-supplier to assure that complete packages are available for the pipe stress design review. In this regard, the

NRC Staff examined the pipe support design specification manual and procedures[6] applicable to communications between each design organization and Gibbs & Hill or Westinghouse. The Staff presented uncontroverted testimony that these documents define and establish the appropriate responsibilities and lines of communication. (NRC Exhibit 207 at 13.)

Accordingly, we find that the applicable standards regarding external design control interfaces are satisfied with respect to the pipe support design organizations.

3. Internal Design Control Interfaces

A portion of CASE's allegations regarding design interfaces concerns the interfaces between the service group in which CASE's witnesses were employed and the design organizations. CASE alleged that there were not adequate guidelines for controlling design of pipe supports between the service group and the design organizations. (Tr. 5028-32, 5034, 6984-92.) We find below that such guidelines and procedures were not required to be implemented at Comanche Peak.

T ͐ group in which CASE's witnesses were employed performs analyses of particular support frames using the Structural Design Language ("STRUDL") computer program The STRUDL Group is a subgroup within the Site Stress Analysis Group ("SSAG"). The entire SSAG is a service organization with no responsibility for

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6 Each pipe support design organization establishes design guidelines in accordance with overall pipe support design specifications. The Staff testified, and we so find, that this specification also satisfies the requirements for design input information as described by N45 2-11, paragraph 3.2. (NRC Exhibit 207 at 12.)

the design of pipe supports.[7] The STRUDL Group's function is to develop a mathematical model of pipe supports based on information provided by the pipe support design organization, to conduct an analysis using the STRUDL computer program employing the data provided, and to return the results of that computer analysis to the designer. (Applicants' Exhibit 142 at 9-10.) The STRUDL Group performs only a service function and is not organized or called upon to evaluate the results of its computer analyses. As such, that Group does not perform a design function and, therefore, its activities do not come within the scope of ANSI N45.2.11. Thus, written procedures regarding interfaces between the SSAG or STRUDL Group and the pipe support design organizations are not required. (Applicants' Exhibit 142 at 9-10; Staff Exhibit 207 at 10-11; Tr. 4834, 5035, 5051.)

Nevertheless, Applicants established detailed procedures to document the responsibilities of each engineering organization, including the SSAG, and the lines of communication between the different organizations (NRC Exhibit 207 at 11; see also CASE Exhibits 825 and 826). We find, therefore, that Applicants have gone beyond the minimum requirements applicable to establishment of design interfaces. In any event, we observe that CASE's witnesses were unable to point out any example of how the absence of the guideline which they contended should exist with respect to the STRUDL Group would affect the safety of the plant (Tr.

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7 There is one exception to this rule, and that is with respect to particular pipe runs (2.5 inches to 4 inches) on which the SSAG performs analyses for Gibbs & Hill. The STRUDL Group is not involved in this program. (NRC Exhibit 207 at 10-11.)

6990-91). Accordingly, we find no basis for concern regarding the adequacy of Applicants' pipe support design efforts due to design organization interfaces.

C. Piping and Support Design Process

A substantial portion of the allegations raised by CASE concerns the design of individual pipe supports (See CASE Exhibit 659B). We address later the particular allegations regarding individual supports or types of pipe support designs. We address below, however, the piping and support design process at Comanche Peak and the implications of the process for the concerns raised by CASE's witnesses. In view of the comprehensive nature of that process, we conclude that even were the designs for a few supports discussed by CASE's witnesses found to be inadequate, this does not present a concern for the design of the piping and pipe support systems at Comanche Peak. This is so because the designs raised by those witnesses were taken from the initial stages of a carefully designed and comprehensive iterative design process and thus do not (nor were they intended to) reflect the quality of the final pipe support designs at Comanche Peak. An understanding of this iterative process is important to an understanding of our disposition of the allegations by CASE regarding support designs. Accordingly, we describe that process below.

Further, CASE alleged that certain aspects of the design process violated both 10 C.F.R. § 50.55(e) and 10 C.F.R. Part 50, Appendix B. Specifically, it alleged that each of the instances in which an initial pipe support design was determined to be

inadequate by subsequent review during the normal course of the design process, or whenever a final stress analysis was not performed at the time certain allegedly inadequate supports were designed, a notice to the NRC of a reportable deficiency should have been made pursuant to 10 C.F.R. §50.55(e). In addition, there is a general allegation regarding the iterative design process that designs later found to be unacceptable should be reported as non-conformances, pursuant to 10 C.F.R. Part 50, Appendix B. (Tr. 6674-76, 6706-7.) We find below that none of these allegations has merit.

1. Iterative Nature of Design Process

The process for the design of piping and supports is iterative in nature. In fact, it is unrealistic to expect to design piping and supports to satisfy all applicable requirements the first time through the process. Such an iterative design approach is employed throughout the nuclear industry, and is utilized in the design of other nuclear components as well. Briefly, the design of an individual support begins with an initial design based on the known initial piping stress analysis. When it is impractical to construct the support as originally designed, a new support scheme is required and an update of the original piping analysis will be performed. This process continues until the final as-built analysis confirms the adequacy of both the piping and supports. (Applicants' Exhibit 142 at 33-34; Tr. 4969 5184, 7155-57.)

The iterative design process was described by Applicants and is summarized in NRC Exhibit 207 at 14-16. As described therein, the process focuses upon a piping "stress problem" which consists of a designated length of pipe for which a pipe support is an accessory that cannot be designed separately from the length of pipe. The steps in this iterative design process are, as follows:

1. A conceptual design for a length of pipe is prepared using the piping plan and elevation and/or isometric drawings for the plant

2. An initial pipe stress analysis on the conceptual piping design is performed to calculate the forces and types of loads on proposed supports on the conceptual piping design.

3. The description of the acceptable piping layout (including the proposed support locations with accompanying directions of restraint and magnitude of forces) are sent to one of the three support design groups.

4. During installation of the supports, field engineers are available to authorize changes to support designs as necessary to produce a usable design.

5. Once piping and some of the accompanying supports are installed, a QA inspection of the as-built dimensions of the piping and installed pipe supports is performed. The drawings utilized at this step are then stamped "as-built verified" and transmitted as a package to the appropriate piping stress analysis organization (Gibbs & Hill or Westinghouse) for a preliminary stress analysis.

6. The pipe stress analysis organization conducts its preliminary stress analysis, adjusting the piping stress problem for any new factors which impact on the pipe or support stresses. The stress problem is rerun to determine new stresses in the pipe and new loads on the pipe supports.

7. The stress package is then returned to the appropriate design group, which reviews the new piping loads to determine whether the particular hanger is still appropriate. Supports which are found to be satisfactory are stamped "vendor certified" and if

found to be unsatisfactory are modified and a new as-built design package is sent to the pipe stress analysis organization.

8. Upon completion of installation of all supports, a stress problem package (incorporating changes to the supports since the problem was last run) is prepared and provided to the pipe stress analysis organization for reanalysis. A pipe stress problem will be rerun if the new as-built configuration impacts the pipe stresses.

9. This package is once again returned to the appropriate design group to determine whether any supports need be modified as a result of the new stress problem and if so, will be modified and returned once again to the pipe stress analysis organization until all pipe stresses are acceptable and all pipe supports are vendor certified to the loads developed in the last run of the stress problem.

(Applicants' Exhibits 142 at 33-35 150 and 151; NRC Exhibit 207 at 14-16; Tr. 5286-91, 7152-54.)

The above described as-built program is established in accordance with the requirements of NRC I&E Bulletin 79-14 (NRC Exhibit 201C; Applicants' Exhibit 142 at 34-35).

Further, Applicants have at least two processes in place to check the validity of the final vendor certification process. The first is a design control group within the pipe support engineering organization on site which is responsible for randomly sampling final vendor certified drawings to assure satisfaction of applicable requirements. Second, Applicants audit the vendor certification process and final designs from both a programmatic and technical viewpoint. (Tr. 7143, 7173-75.) Accordingly, we find that adequate controls are in place to assure the effectiveness of the iterative design process.

2. Incorporation of Design Changes

Another aspect of the pipe support design process misunderstood by CASE's witnesses, and which misunderstanding underlies CASE's repeated attempts to criticize pipe support drawings from the early stages of the design process, is the manner in which drawings are updated and labelled. Accordingly, we address below the process of updating drawings and reanalyzing supports.

One aspect of this process is the need to accommodate design changes caused by construction interferences. Pipe support field engineers are responsible for working directly with the craft to resolve interference problems. Modifications to supports are documented on Component Modification Cards ("CMC") which are forwarded to the proper design organization for review and approval. The majority of these changes is minor in nature, involving small dimensional changes or similar matters. In any event, the field engineers have demonstrated a high level of competence in making these changes. In all but a small fraction of the cases the field initiated modifications have been determined to be acceptable If the responsible design organization does determine that the intended modification is not appropriate, the design will be subsequently modified in accordance with that organization's direction. Each design organization has a group on site to review the CMC's as they are generated. (Tr. 4958, 4965-66, and 5184.)

To incorporate the CMCs into the design process, they are first drafted onto as-built drawings which then undergo field verification before being ent to the pipe stress analysis

organization as as-built verified drawings. Applicants' audit group has audited the as-built verification process and determined that adequate controls have been established and implemented to assure the accuracy of the as-built verified drawings. The pipe stress analysis organization analyzes the revised (as-built verified) drawing in accordance with the procedures described above. That analysis is then forwarded to the responsible support design organization for certification, and if so certified, it is designated as vendor certified. (Tr. 7141-54.) We believe this process is a reasonable means for addressing a complicated and lengthy process of design and construction of pipe supports.

To illustrate the pipe support design process and the associated steps in the development of pipe support design drawings, Applicants submitted a sample package of pipe support design drawings (Applicants' Exhibit 147). The original design drawing for this particular support was received from Applicants' pipe support designer (Applicants' Exhibit 147 at 1; Tr. 5399). A second iteration of this document simply involved the relabelling of the approval block. (Applicants' Exhibit 147 at 2; Tr. 5400). The next step in the process involved the preparation of comments on the drawing by the field survey team (Applicants' Exhibit 143 at 3; Tr. 5194). The label pasted on top of the original vendor label is used to supply construction-related information. (Tr. 5200). This drawing, with the surveyor's comments, was sent to the appropriate analysis organization for final as-built piping verification (Tr. 5195).

Simultaneously, the same drawing was sent to the pipe support design review group (Applicants' Exhibit 147 at 4; Tr. 5196). In this example, because the surveyor identified an angularity in the structure not on the original drawing, a CMC was generated (Applicants' Exhibit 147 at 6; Tr. 5196). Finally, the organization performing the as-built analysis reviewd the completed analysis with the final as-built piping stress loads (Applicants' Exhibit 147 at 7; Tr. 5196-97). This analysis was verified by the design review group and certified by the engineer of the vendor organization, as indicated by the vendor certified "stamp" in the center of the drawing (Tr. 5197). None of the exhibits submitted by CASE witnesses Walsh and Doyle were vendor certified drawings (Applicants' Exhibit 142F at 6; NRC Exhibit 207 at 16; Tr. 5198), contrary to what they believed (e.g., Tr. 3719).

3. Professional Qualifications

A question raised by the Board concerning the iterative design process was whether the system was adequate to detect errors that might be introduced into the process at the early stages of design by relatively young and inexperienced engineers. This concern arose from a comment by the NRC Resident Reactor Inspector that of the 19 broad concerns raised by CASE's witnesses, six of those concerns could be described as arising because of a "somewhat inexperienced engineering staff" (Tr. 6403). Both Applicants and the NRC Staff testified, however, that the programmatic controls in place in the vendor certification program provide a high level of assurance that any

deficiencies occurring in the earlier phases in the pipe support design process would be identified (Tr. 6404, 7164-70). This assurance arises because at the later phases of the design review process, those persons who perform the reviews are, and were planned to be, highly experienced and capable engineers (Tr. 6404-05). Specifically, while the engineers who prepared the initial design for pipe supports and CMCs are qualified for just those purposes, the final vendor certification process is performed by a registered professional engineer (for all Class 1 piping systems) or by degreed engineers with several years of experience (Class 2 and 3 piping systems) (Tr. 7167-69). We conclude that the qualifications of personnel assigned tasks in the pipe support design process are in accord with applicable requirements and are commensurate with their levels of responsibility within that process.

4. Timing of Vendor Certification Process

A peripheral question also raised by the Board, was whether the time left to perform the final vendor certification was adequate in view of the number of engineers involved in that review and the large number of supports to be certified (Tr. 7170). Applicants' witnesses testified that no time constraints that could affect the final review process are imposed on that process, and that the schedule allowed adequate time for completion of the certifications prior to the plant completion

date (Tr. 7172).[8] There is no evidence of record that leads us to conclude otherwise.

5. Reportability

With respect to the allegation that Applicants should report, pursuant to 10 C.F R. §50.55(e), any deficiency in pipe support design even where such design was still subject to further analysis as part of the design process (Tr 6674-75), we find no basis for reaching such a conclusion. Rather, as a matter of law, an item is not reportable pursuant to 10 C.F.R. §50.55(e) until it has been proven to be significantly deficient. Further, NRC Regulations do not impose a time constraint with respect to the conduct of analyses (See Tr. 6675-76). This is not to say that any deficiency in design escapes the reaches of 10 C.F.R. §50.55(e) simply because the design is subject to further review. The question is one of degree, viz., the significance of the deficiency.

We questioned the Staff witnesses as to whether any of the pipe support design deficiencies identified during its review would be considered significant. None of the Staff witnesses knew of any deficiencies that would come within that category

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8 Applicants also submitted, at our request, an affidavit of their witness regarding the schedule and work remaining to be completed in the vendor certification process. See "Affidavit of John C. Finneran Regarding Board Inquiries Concerning Status of Pipe Support Design Verification and Unstable Supports," June 3, 1983. Because the scheduled plant completion date was subsequently changed to December, 1983, we are not concerned with the availability of adequate time to complete the process. We note nonetheless, that even had that schedule not been revised, we would have been satisfied with the evidence of record regarding the schedule for completion of the vendor certification process.

The witnesses specifically addressed the question of whether the unstable pipe supports[9] identified by Applicants or during the Staff's review would present significant deficiencies that warranted reporting under 10 C.F.R. §50.55(e). None of the Staff witnesses or CASE witnesses could identify any instance where an unstable support would warrant reporting as a significant deficiency. (Tr. 6680-6689; 6695-6705; 6722-28.) We agree that there is no reason to consider the alleged design deficiencies, including the unstable supports, to fall within the scope of the reporting requirements of 10 C.F.R. §50.55(e).

6. Documentation of "Nonconformances"

With respect to the allegation that Nonconformance Reports ("NCRs") should have been written against pipe support designs which were found to be inadequate, the NRC Staff testified, and the Board agrees, that 10 C.F.R. Part 50, Appendix B does not address inadequate designs but rather addresses the conformance of installed hardware and the inspection thereof to the design. With respect to 10 C.F.R. Part 50, Appendix B, Criterion III, concerning design control, that provision establishes review procedures, and does not involve reporting of nonconformances. (Tr. 6707-10.) Accordingly, we find there is no requirement for the identification of inadequate pipe support designs as nonconforming conditions. The iterative design process for pipe

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9 The question of the safety implications of unstable supports is discussed, _infra_, in Section II.F.

supports (including the internal checks in that process) discussed herein supra, Section II.C.1, assures that inadequate designs or unstable supports are identified and corrected.

D. Design Analyses of Richmond Inserts and Hilti Bolts

The allegations concerning the design of pipe support concrete anchors (Richmond Inserts and Hilti Bolts) concern two aspects of those designs. First, CASE's witnesses alleged that the loads induced on the concrete anchors due to thermal expansion of pipe support material under LOCA conditions were excluded from the design of the anchors. Second, they alleged that the method of analysis regarding sheer and moment stresses at the anchors is erroneous. (CASE Exhibits 659 at 1-2, 4; 659H at 3; 669B Attachments 9 and 10; Tr. 3752-57; 3769-72.) To address these concerns, Applicants performed a worst case analysis of these conditions to demonstrate generically that the approach employed in the design of the anchors was valid. The NRC Staff also conducted independent analysis, and requested of Applicants additional testing of the particular anchor utilized within the containment building. Upon considering the allegations in light of the evidence presented by Applicants and the NRC Staff, we find that the allegations are meritless and that the design of the pipe support anchors at Comanche Peak is adequate.

Before addressing the particular allegations regarding the analysis of inserts, we note that the intervenor attempted to apply the provisions of ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants," to the materials

used in the inserts at Comanche Peak. Specifically, criterion nine on page D-11 of that standard concerns the identification and control of materials, parts and components. (CASE Exhibit 687; Tr. 6415-16.) However, undisputed testimony of the NRC Staff demonstrated that this criterion applies to the control of materials, parts and components for materials in which material traceability is important. Because the insert at issue here is an off-the-shelf item for which specifications are provided by the manufacturer and for which it is not necessary to assure material traceability for any of the parts of the insert, that criterion is inapplicable. (Tr. 6415-18.) Accordingly, we find that ANSI N45.2-1971 Criterion 9, is inapplicable to the design of inserts at Comanche Peak.

1. Consideration of Thermal Expansion of Pipe Support Tube Steel

As noted previously, one of the principal allegations of CASE's witnesses is that Applicants had not considered the stresses imposed on concrete inserts and bolts (used to attach pipe supports to concrete structures) which arise from the thermal expansion of the pipe support members under LOCA conditions. As discussed below, Applicants performed a generic study that demonstrates that the self-limiting characteristics of the thermal stresses imposed by the expansion of the supports under LOCA conditions also would relieve the stress imposed upon the anchors. This conclusion was based on the same principles

set forth in the ASME Code concerning consideration of differential thermal expansion stresses which we addressed at some length in our Memorandum and Order dated July 6, 1983.

The NRC Staff testified that while the ASME Code does not apply directly to the design of concrete anchorages, 10 C.F.R. Part 50, Appendix A, Criteria 1 and 2 require that components be capable of performing their intended design function. To this end, the Staff requested that Applicants demonstrate through testing that the inserts are capable of performing their intended design function under imposed loads without failure. (NRC Exhibit 207 at 18.) We are convinced that the analyses and testing performed by Applicants demonstrate the ability of the inserts to perform their intended design function, including the ability to withstand the stresses induced by the expansion of the pipe support members under LOCA conditions.

Applicants' generic analysis of the effects of LOCA-induced thermal expansion of supports on the anchors is based on the use of test data results of load versus displacement characteristics of various types of anchors (Hilti Bolts, Richmond Inserts and Nelson Studs). Applicants' analysis utilized accepted test data and coefficients of thermal expansion for comparison with the ultimate displacement of the various anchors under appropriate loading conditions to arrive at a factor of safety of greater than 2.0. Applicants considered this factor of safety to be acceptable noting that it is used in the ASME Code for displacements of Nelson Studs which attach the steel liner plate to the reinforced concrete containment. (Applicants' Exhibits

142 at 22-24, 142D, Attachment 2.) The analyses performed in Applicants' Exhibit 142D utilize examples of worst case conditions (Applicants' Exhibit 142 at 22; Tr. 5250). We believe that Applicants' analyses alone would be sufficient to demonstrate the validity of their position regarding the ability of concrete anchorages to withstand the stresses under consideration here. However, there is additional evidence in the form of actual tests performed by Applicants under NRC supervision which confirms of adequacy of the inserts.

2. Testing of Concrete Inserts

Although not disagreeing with Applicants' generic analysis described above, the NRC Staff determined[10] that it would be more conservative to perform actual sheer tests on the 1 1/2 inch diameter Richmond inserts rather than relying (as did the analyses) upon extrapolation of test data from 1 inch and 1 1/4 inch inserts to the 1 1/2 inch (Applicants' Exhibit 142D, Attachment C and Reference Fc; NRC Exhibit 207 at 18-19). Specifically, the Staff concluded that Applicants could utilize a safety factor of two for the inserts if its conclusions were based on test data for the size of inserts used in the containment and tests were performed in a manner which modelled actual configuration in the plant (NRC Exhibit 207 at 20; Tr. 6435-36).

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10 In fact, the Staff did not find any violation of NRC requirements by Applicants prior to the performance of the above-described tests. (Tr. 6418-20.) The Staff cited 10 C.F.R. Part 50, Appendix A, Criteria 1 and 2 as applicalbe to this question, but did not claim that additional tests were <u>required</u> by those provisions. (NRC Exhibit 207 at 18.)

In response to the NRC's request in this regard, Applicants performed tests to verify that the allowable loads at Comanche Peak for the 1 1/2 inch Richmond inserts (which are the inserts used in the containment at Comanche Peak) were appropriate, and to determine the actual factors of safety being used in these connections. The tests were conducted utilizing a methodology representative of the manner in which these connections are employed at Comanche Peak. (Tr. 7084-86, 6514-19, 6441-45; CASE Exhibit 834.)

a. Factors of safety

The factors of safety calculated in the insert tests were determined by dividing the experimentally derived ultimate strength of the insert by the manufacturer's allowable strength. CASE's witnesses argued that, in addition, a load factor or capacity reduction factor should have been employed in calculating the safety factor. However, the Staff testified, and we so find, that the load factors and capacity reduction factors are substitutes for a safety factor and are not to be employed in addition to safety factor calculations (Tr. 6430-31, 6434-36).

Further, we note that those tests were performed for the purpose of determining the factors of safety which exist for one-time static loads under conditions postulated to occur in the event of a LOCA. CASE argued that these tests were inadequate because they did not demonstrate the adequacy of the insert for normal operating conditions (Tr. 6537). However, these tests were not necessary to demonstrate the adequacy of the inserts to perform their function during normal operating cyclical loads.

That consideration is accounted for by application of the allowable stress values for the inserts. Indeed the test results here are irrelevant to the question of whether the allowables for normal and upset conditions were satisfied. (Tr. 6540, 6547-48 and 7085.)

In any event, Applicants' witness testified that the test of the assembly (consisting of the bolt and the insert imbedded in concrete) demonstrated that there is no pronounced yield in the assembly. Thus, the yield point for the assembly may be considered to be near the ultimate yield. Given these facts, the stress levels observed during the static tests with the resultant factors of safety (yield greater than three times the allowable) actually confirm the allowable stress values for normal operation in that, as described in the ASME Code, when the total stress level is at or below twice yield, "shakedown" in cyclic loadings is achieved. Maintaining stress levels within the "shakedown" range will assure that under repeated loads there will be no incremental additive distortions. (Applicants Exhibit 142 at 16-19; Tr. 6556-58; 7085-86.) In sum, the test data derived during the static tests performed to demonstrate ultimate yield under continuously applied loads does not compromise, and in fact confirms, the use of established allowables for design, normal and upset conditions.

b. Allowable stresses

Another allegation regarding the adequacy of the insert tests was that the design load allowable utilized in calculating the factors of safety based on the maximum applied load was

incorrect (Tr. 6451). The specific concern was that the design load allowable for the inserts tested should be 25 Kips rather than the 17.67 value employed in the factor of safety calculations (See CASE Exhibit 774 at page 17; Tr. 6451. See also CASE Exhibit 724 at page 2 and CASE Exhibit 834 at page 8, specimen numbers 7, 8 and 9; Tr. 6453). As we find below, however, this concern, was premised on a misunderstanding of the test results and the applicable design allowables for the insert assemblies tested

As clearly explained by the NRC Staff's witness, CASE was confusing the design allowable applicable for the bolt in the insert assembly with the design allowable for the insert itself. The design allowable for the bolt was utilized in calculating the factor of safety because the tests for that assembly demonstrated that the bolt was the controlling item, i.e., the bolt had a lower capacity for shear stress than did the insert. In short, where an insert assembly is tested, the controlling allowable load for the assembly depends on the relative strengths of the bolts and the inserts into which the bolts are placed. Different combinations of assemblies and bolts will result in either the bolt or the insert being the controlling component of the assembly. In the type of assembly tested here, and that which is used at Comanche Peak, it is the bolt which is controlling, i.e., will fail first, for the loads imposed by shear forces. (Tr.

6453-56.)[11] Thus, we find Applicants correctly selected the design allowables for the insert assemblies to calculate appropriate factors of safety.

c. Combined shear and tension testing

CASE alleged that there had been no testing of combined shear and tension of the insert assemblies as was stated should be performed in NRC Exhibit 207, at page 20 (Tr. 6502). The Staff testified, however, that this recommendation was premised on the assumption that a factor of safety of two would be used. Because a factor of safety of three was shown to exist by the insert shear tests, such combined shear and tension testing was not necessary. (Tr. 6503, 6508.) No evidence was presented which challenges the conclusions of the Staff on this matter. Accordingly, we find that combined shear and tension testing need not be performed.

d. Bending stresses

Another aspect of the insert tests questioned by CASE was premised on a statement in NRC Exhibit 207 that Applicants had not calculated the bending stresses in the inserts which result from the imposition of the shear force on the bolt offset from the concrete surface by the use of a 1 inch washer between the

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11 The unresolved items in NRC Exhibit 207 concerning the performance of Richmond inserts were formerly closed by I&E Report 83-12/83-07, dated May 13, 1983 (NRC Exhibit 208). The only assertion regarding the accuracy of NRC Exhibit 208 was that the use of a 17.67 Kip design allowable rather than a 25 Kip allowable as utilized in Applicants' Exhibit 142D, is not consistent with the practice at Comanche Peak. (Tr. 6486-87.) The testimony was clear, however, that the Applicants' employed the appropriate design allowable for the type of insert assembly being analyzed. (Tr. 6488-89.)

concrete and the support steel (NRC Exhibit 207 at 21). CASE contended that the analysis performed by the NRC Staff showed that these bending stresses were significant and should have been calculated. Specifically, CASE referred to calculations performed by the NRC Staff to compare Applicants' calculations of bending stresses with those derived by use of the STARDYNE computer model. (CASE Exhibit 775 at 5-7, following Tr. 6513; Tr. 6511-12, 6532-33.)

The NRC Staff testified that the calculations made in CASE Exhibit 775 demonstrated that Applicants methodology was overly conservative (Tr 6535). Further, the entire question of analyzing bending stresses in the bolts was resolved by the test performed utilizing the 1 inch washer which created the subject bending stresses. In addition, the NRC witnesses testified that the bolt primarily would be in shear because of the relative thicknesses of the washer and the bolt, and that the calculations derived from the STARDYNE computer were overly conservative and unrealistic in this aspect of its calculations (Tr. 6533-34, 6552; CASE Exhibit 834 at 8-9) and in any event would be unnecessary given the engineering judgment that the stresses would primarily be in shear (Tr. 6566). In response to this testimony the CASE witness stated that actually its concern was with the consideration of bending stresses under normal operating conditions and that these tests were performed to analyze the static loading which may occur under LOCA conditions (Tr. 6542-43). However, the applicability of these tests to confirmation of the design allowables for normal operating conditions was

discussed above in Section D.2.a. Further, CASE presented no evidence that leads the Board to conclude that its concern is valid. Accordingly, we find there is no valid area of concern regarding bending stresses.

e. Modelling of moments

The next concern raised by CASE concerned the modelling of moments within the pipe support tube steel by coupling the moment into the bolt to arrive at a tension in the bolt. The specific point of disagreement was over the assumption that the compressive area for the coupling or transfer of the moment from the tube steel to the bolt extended to the end of the washer plate between the tube steel and the concrete. CASE argued that the washer is not, as assumed by Staff and Applicants, subject to compressive forces over its entire surface area. (Tr. 6889-93.)

The Staff testified that it reviewed this concern and that the calculational techniques utilized by the design organization for computing this effect is correct. The Staff concluded that the washer is sufficiently thick to permit the consideration of the effective width of the washer to be its entire surface area. (Tr. 6889-92.) To clarify this question we requested that drawings be prepared by the Staff's and CASE's witnesses (NRC Exhibits 209 and 210; Tr. 6894-6901). Through the discussion of these drawings, we ascertained that the point of disagreement is with the amount of tension produced in the bolt as a result of the different assumptions regarding the moment arm applicable to the modelling of this configuration (Tr. 6903).

We find helpful the testimony of Applicants' expect witness that in reality CASE was applying a detailed analytical technique applicable to the aerospace industry where the factors of safety are only 1.1 rather than the factors of 3 or 4 applicable to nuclear plants (Tr. 6910). Further, the witness testified that the assumptions used by the Staff are valid because the configuration being considered would (if the loadings were sufficient) deform to clearly provide for a moment arm as designated by the Staff (Tr. 6911-12). This however, would be only a worst case situation and the safety factor applied to the design of this component indicates that that point would not be reached (Tr. 6927). Applicants' witness testified, and we so find, that this is an instance where the use of bounding assumptions and engineering judgment is proper and that more detailed analytical techniques need not be applied in view of the high factors of safety incorporated in the designs (Tr. 6913-15). Such a situation is consistent with the engineering approach utilized in the ASME Code where only if detailed calculations are necessary to analyze a particular process for design will the Code specifically identify such calculational techniques (Tr. 6915-16).

In view of all the above evidence, we find that the analytical technique employed by Applicants and confirmed by the NRC Staff is valid and raises no concern regarding the adequacy of the design of the supports.

f. Shear keys

An additional allegation made regarding the modelling of the Richmond inserts was that those tests should have considered the use of shear keys on the base plates of supports or moment restraints which utilized Richmond inserts (Tr. 6578, 6580). It was clearly demonstrated, however, that even assuming that the base plates utilized shear keys, there would be no significant impact on the analysis, and that in fact the assumption that shear keys do not exist is conservative for the purpose of maximizing the loads on the bolts (Tr. 6584-86). Accordingly, we find that not including shear keys in these analyses is a conservative assumption and we find no basis for questioning the analytical technique of Applicants or the NRC Staff in this regard.

g. Worst case analysis

Finally, CASE alleged that the worst case scenario analyzed by Applicants (Applicants Exhibit 142D) and relied upon by the Staff (NRC Exhibit 207 at 18) was not the worst case because the support reviewed for worst case stresses actually had a longer overall span than the member length used in the analysis. (Tr. 6587.) However, it was demonstrated that the longest overall member of the large support analyzed as the worst case scenario by the NRC was 11 feet, as assumed by the NRC. Further, Applicants and the NRC Staff testified, and the Board agrees, that it would be inaccurate to assume the large support should be

analyzed as a single member in determining the thermal expansion effects on the attachments to the wall. Thus, the use of the 11 foot member as the worst case was appropriate. (Tr. 6586-92.)

We find that the evidence presented regarding the ability of the concrete anchors to withstand stresses induced by the expansion of the supports under LOCA conditions overwhelmingly demonstrates the adequacy of those anchors. Accordingly, we conclude there is reasonable assurance that the anchors will be capable of performing their intended design function.

E. Design Considerations for Wall-to-Wall, Floor-to-Ceiling and Floor-to-Wall Supports

CASE's witnesses testified that various effects on wall-to-wall, floor-to-ceiling and floor-to-wall supports were not considered at Comanche Peak (CASE Exhibit 659 at 3, CASE Exhibit 668; Tr. 3120-22 and 3141-43). In particular, CASE's witnesses testified that the LOCA-induced differential temperature effects on service water floor-to-ceiling pipe supports were not considered (Tr. 3141-43). Further, a concern was expressed regarding the differential seismic displacement of those supports and concrete creep displacement effects.[12] The NRC Staff and Applicants presented uncontroverted testimony that each of the effects which CASE's witnesses alleged were not being considered had in fact been identified by Applicants as requiring analysis

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12 CASE Exhibit 669 at 62-63, 118-121, 145-151, 214-215, 307-309; CASE Exhibit 669A at 4-7; CASE Exhibit 669B, Attachments 7C-7D, 14D-14E, 14I-14K and 18.

and each of the supports specifically referenced by those witnesses either had been or would have been identified during the normal design process as warranting further analysis.

1. Differential Thermal Expansion Effects

The special case of differential thermal expansion effects in this classification of supports had already been identified by Applicants as an area in which special analyses may be necessary despite the general conclusion that thermal stresses need not considered in the design of linear pipe supports (Applicants' Exhibit 142 at 11-12; CASE Exhibit 659E). The NRC Staff confirmed that instructions had been given to pipe support design organizations that such effects may need to be considered in the design of these categories of supports. In addition, the Staff evaluated Applicants' assessment of the differential thermal expansion effects on three particular supports and found that Applicants satisfied their commitments in the FSAR in this regard (NRC Exhibit 207 at 25). In fact, two of these three supports examined by the NRC were the subject of detailed testimony in this proceeding. Specifically, the moment restraint and the steam generator upper lateral restraint identified at page 25 of NRC Exhibit 207 were examined in detail. These restraints are discussed below in Section II.T.

Even though Applicants had determined that an evaluation of the thermal expansion effects of this category of restraints may be necessary, they presented testimony which demonstrated that even in these cases there is no concern for the integrity of the restraint as a result of constrained thermal expansion in the

event of a LOCA. Specifically, Applicants' witnesses testified that for supports which span between structures (such as floor-to-ceiling or wall-to-wall supports), there can be displacements which relieve the thermal stress. These displacements would occur either in the structures to which the supports are attached (Tr. 5239) or by local yielding of the structural member (Tr. 5253-56) The latter yielding will result in minute plastic deformation of the beam, thereby relieving the thermal stress (Tr. 5256). Accordingly, we do not consider the possibility that these effects may occur to be significant from the standpoint of maintaining the functional capability of the support.

2. Differential Seismic Displacements

This allegation concerned both LOCA and differential seismic displacement of certain Service Water pipe supports which extend from floor-to-ceiling. Applicants and the NRC staff presented testimony that these supports had previously been identified by Applicants and that guidelines at Comanche Peak for each of the design organizations required consideration of these effects.

The specific Service Water supports identified by CASE's witnesses were identified by Applicants in late 1981 in the normal design process as possibly being inconsistent with seismic guidelines utilized at Comanche Peak (Applicants' Exhibit 142 at 25; Tr. 3142). However, Applicants' analysis of these supports showed that it was not necessary to modify the supports to account for seismic displacement considerations. Nonetheless, Applicants modified the supports, an obviously conservative approach. (Staff Exhibit 208, Appendix at 7.)

In addition, the NRC Staff verified that Applicants' design group (PSE) guidelines require that large frames which span across corridors or from floor-to-ceiling must be designed with a slip joint at one end. Although the other design organizations (ITT-Grinnell and NPSI) did not have similar requirements, they either have designed only a limited number of supports which fall within this category and which will be reanalyzed (ITT-Grinnell), or have not designed any such supports (NPSI). Nevertheless, Applicants issued a memorandum to those design organizations which directs them to use the same seismic guidelines as those contained in the PSE guidelines for this class of supports. (NRC Exhibit 207 at 25-26; NRC Staff Testimony at 11, following Tr. 6402.) We thus find that there is no basis for concern regarding differential seismic displacements as to this category of supports.

As for the concerns regarding LOCA-induced thermal stesses in these supports, CASE's witness admitted under cross-examination that in view of their location outside containment that thermal stresses (if any) would not be LOCA-induced. He was unable to identify any particular evidence that thermal effects would be important for these supports. In fact, he admitted thermal effects would not be a significant consideration and that seismic effects would be the controlling concern. (Tr. 7053-55.) Accordingly, we conclude there is no basis for any concern regarding differential thermal effects on these supports.

3. Concrete Creep Displacement

CASE's witness alleged that the stresses which could be induced as a result of the creep of concrete slabs to which wall-to-wall and floor-to-ceiling supports were attached were not being considered, and identified particular supports for which he contended such effects should be considered (CASE Exhibit 669B, Attachment 7B). The NRC Staff determined that the effects alleged to occur here would be insignificant in that the magnitude of the effect is greatest during the first twelve months following concrete placement and that the supports in question had not been constructed until more than two years after the concrete to which they were attached was placed (NRC Exhibit 207 at 26). This conclusion was premised on a portion of an American Concrete Institute committee report attached to the testimony of CASE's witness (CASE Exhibit 669B, Attachment 7B, Figure 9.2).

For the above reasons, we conclude that the allegations regarding LOCA thermal expansion loads and concrete creep displacement on wall-to-wall, floor-to-ceiling and floor-to-wall supports are without technical merit. As for the seismic displacement effects, those were identified in the course of Applicants' normal design review program and have been addressed. Thus, we find that this allegation to be unfounded.

F. Pipe Support Stability

CASE's witnesses expressed a concern that certain pipe supports, the designs for which they observed in their positions in the STRUDL Group, were unstable. Specifically, they alleged

that certain types of supports could be characterized as three-bar linkages which would be unstable if the supported piping was able to rotate within the box frame or U-bolt attaching the pipe to the support. Further, other instances of instability could arise even where such gaps did not exist initially but were created by movement or deformation of the U-bolt or by insufficient friction of the box frame on the supported piping. (CASE Exhibits 669 at 95-104, 669B, Attachments 4 and 13. See also CASE Exhibit 659H at 1; Tr. 3103-05, 3109.)

The question of stability of pipe supports is another instance of concerns arising because CASE's witnesses had only a limited knowledge by virtue of their limited roles in the entire design process for pipe supports and were unaware of measures beyond their scope of responsibility to identify and correct unstable supports. As was demonstrated by Applicants' and the NRC Staff's witnesses, the potential instability of pipe supports at Comanche Peak was routinely identified in the normal design review process. Only a very small fraction of the total initial design of supports at Comanche Peak were found to be potentially unstable. (E.g., Tr. 5204, 6720.) Thus, regardless of the fact that CASE's witnesses made these allegations, we are convinced that any unstable support at Comanche Peak would have been identified during the design review process and its design revised, if necessary Nonetheless, we examine below the general question of pipe support stability and specific instances of alleged instability raised by CASE's witnesses.

The first principle to be established is that the stability or instability of a particular support segregated from the piping system is not significant. The important consideration is whether the entire piping system and associated supports are stable when considered as a single system. (NRC Exhibit 207 at 27; Tr. 6695-99; Applicants' Exhibit 142 at 28.) At least one of CASE's witnesses (Mr. Doyle) appeared to recognize this principle (CASE Exhibit 669 at 210). Thus, one aspect of our review of this question is whether any supports alleged by CASE to be potentially unstable when segregated from the piping system (or to be unstable even as p rt of the piping system) create any concern for the overall stability of the piping system.

Applicants testified that unstable pipe supports at Comanche Peak have been identified during the normal design review process and will be modified as necessary to assure stability (Applicants' Exhibit 142 at 27). The NRC Staff confirmed this, and noted that Applicants also have committed to assess the stability of all non-rigid box-frame supports and to modify those supports if necessary, to prevent rotation of the box-frame about to the pipe to assure stability. Applicants' proposed modifications provide a positive means for preventing rotation of the box frame about the axis of the supported pipe, thus assuring stability and functional adequacy of the piping system. (NRC Exhibit 207 at 28.)

In addition, to confirm its conclusion that potential unstable pipe supports do not raise a concern regarding the adequacy of Applicants' support and piping designs, the NRC Staff

identified approximately 30 individual supports alleged by CASE to be unstable. The piping systems on which these supports were constructed were reviewed to confirm that these supports were not grouped together so as to create an instability problem with the piping system. (Tr. 6716-19.) The Staff further premised its conclusion that the pipe support design process would routinely identify each unstable support on the fact that its review of the Gibbs & Hill review process noted instances of potential instabilities being identified for corrective action and the Staff's own review of 100 vendor certified supports, wherein no instances of potential instability problems were identified. (Tr. 6721.)

In view of the above evidence, we reiterate our finging that the Applicants' pipe support design process should identify and correct, if necessary, any instance of unstable pipe supports designs at Comanche Peak.

G. Utilization of U-bolts in Pipe Support Design

CASE's witnesses raised several allegations regarding the use of U-bolts at Comanche Peak in pipe support design. First, they argued that the constraint on the piping system induced by U-bolts oriented with their principal axis in the direction of the design load was not considered or that the corresponding lateral load on the U-bolt was not considered (CASE Exhibit 669 at 87-88) Second, it was alleged that U-bolt deformations were not included in calculations regarding support deflections (CASE Exhibit 669 at 195-197). Finally, it was argued that U-bolts were cinched down on piping, thereby inducing pre-loading

stresses and stresses arising from the constraint of differential thermal expansion which were not considered in either the U-bolt analysis or piping stress analysis. In this regard it was also contended that the supports may become unstable upon yielding and permanent deformation of the U-bolt. (CASE Exhibit 669 at 318.) Approximately thirty separate supports were identified by CASE's witness as improperly utilizing U-bolts in their design (CASE Exhibits 669 at 195-213; 669B at 13).

1. One-way U-bolt Supports

With regard to the first concern, the NRC Staff determined that Applicants had already identified the same matter as part of their as-built design verification program. Consequently, Applicants' A/E issued a memorandum requiring reevaluation of piping stress analyses if the piping thermal movement in the unrestrained (lateral) direction is greater than 1/16 inch. This reanalysis was undertaken because the original assumption was that the piping would be unrestrained in that direction. In addition to the reanalysis, Applicants have undertaken to modify or replace all U-bolt supports which are identified as being subjected to greater than 1/16 inch lateral deflection. With respect to deflections less than 1/16 inch or for displacements resulting from seismic induced movements, it was determined that those displacements resulted in negligible loads and stresses. (NRC Exhibit 207 at 29-31.) This approach was not challenged or refuted by CASE Accordingly, we concur with the NRC Staff that

the allegations regarding the use of U-bolts in one-way supports is adequately addressed and raises no concerns regarding the adequacy of these designs.

2. U-bolt Deformations

As to the second area of the allegations, regarding consideration of U-bolt deformation in design calculations, the NRC Staff testified, without contradiction, that the movements which could create deflection of U-bolts were insignificant. The basis for this conclusion is that the limitations imposed by Applicants on U-bolt deflections limit the movement of the U-bolt to small (elastic) movement not giving rise to deformation. (NRC Exhibit 207 at 31-32.) Accordingly, this question presents no concern for the adequacy of the support designs.

With respect to the allegation that the pre-loading stresses induced in the U-bolts were not considered, it was demonstrated that Applicants have implemented a procedure that governs the installation of U-bolts to limit the tightness of those bolts. The pre-loading stresses resulting from this procedure are within acceptable limits and are consistent with usual industry practice. (NRC Exhibit 207 at 32; Tr. 6742-43, 6746.) In any event, if the U-bolts had been too greatly tightened, stripping of the bolts would have resulted which would be obvious to inspectors (Tr. 6743-44). In this regard, Applicants will conduct field verifications prior to pre-operational testing to assure proper installation of U-bolts (NRC Exhibit 207 at 32; Tr. 6744).

3. Differential Thermal Expansion

The question of stresses induced in piping and the U-bolts as a result of differential thermal expansion was addressed in some detail by Applicants and the NRC Staff. Their testimony demonstrated that such effects are insignificant in that the differential thermal expansion between the pipe and the U-bolt is negligible as a result of the thermal contact between the pipe and the U-bolt. It was determined that the relative difference in expansion between the U-bolt and the pipe would be approximately 1/128 inch, which produces negligible differential thermal expansion effects. Years of practical experience in this area have shown that such effects indeed are insignificant, and we find that consideration of them is unwarranted. (NRC Exhibit 207 at 32-33; Applicants' Exhibit 142F at 5.)

As to the allegation that differential thermal expansion effects would occur in box frame supports with zero clearances, it was determined that the procedures at Comanche Peak were to allow for a gap between the pipe and the box frame if the calculated diametrical thermal growth of the pipe exceeds 1/32 inch at design temperatures. Further, it was determined that these configurations were used only on low temperature lines where temperature differentials would be small and the resulting stresses in the frame and pipe would be insignificant. (NRC Exhibit 207 at 33; Applicants' Exhibit 142F at 7.)

Accordingly, we conclude that the stresses from differential thermal expansion induced in piping, U-bolts or box frames are negligible and need not be considered.

H. Seismic Acceleration

CASE's witnesses expressed a concern regarding the effect of loads on pipe supports resulting from the seismic acceleration of the supports themselves. In support of these allegations, one of CASE's witnesses assumed a natural frequency for the supports based on the peak of the response spectra curve for the buildings. (CASE Exhibits 659H at 1; 669B, Attachment 12A; Tr. 3100.) As discussed below, the testimony presented by Applicants and the NRC Staff demonstrated that these concerns were unfounded or involved negligible stresses. During the course of the hearing, CASE's witnesses raised an additional concern not previously presented in their testimony regarding the effect of oversized bolt holes in support base plates on the seismic acceleration of supports (Tr. 6615-19).[13] Applicants presented testimony regarding this assertion which we believe resolves the allegation. The NRC Staff intends to conduct further analyses of this matter, but in view of Applicants' testimony we do not believe it to be necessary for our decision.

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13 We note that CASE raised several new issues during the course of the hearings, and in doing so surprised Applicants and the NRC Staff. While at an unfair disadvantage, the Applicants and Staff presented evidence to respond to the new matters. While the proper course is for parties to identify issues prior to trial and for presiding Boards to prohibit the raising of new issues at trial (absent a showing of good cause), we allowed this novel practice to proceed to assure that no important safety issues were involved. We are satisfied that the record establishes that no issue raised in this matter has safety implications.

1. Effect of Seismic Acceleration Loads

Applicants and the NRC Staff testified that the Applicants already evaluate the seismic accelerations of supports in the small bore piping area. However, for large bore piping the seismic acceleration loads of the supports themselves had been generically determined to be relatively low in comparison with the design loads imposed on piping and therefore were not included in the pipe support design process. (NRC Exhibit 207 at 34; Applicants' Exhibit 142 at 30.) In addition, Applicants testified that they already consider the seismic acceleration of large frames in the unbraced direction in original support designs (Applicants' Exhibit 142 at 30).

To confirm the validity of Applicants' assumptions regarding seismic accelerations of supports on large bore piping, a random sample of least conservative supports was selected for detailed analysis (Applicants' Exhibit 142 at 31-31). These supports included unbraced cantilevers, large frames braced in one direction, and large structures with relatively small pipe loads. In this reanalysis, the results of which were confirmed by the NRC Staff, it was demonstrated that the loads and stresses on the supports from the seismic acceleration of supports were in all cases within allowable limits of ASME Code Section III and were in the majority of cases, negligible. Further, reanalysis of other supports was also performed which demonstrated the seismic acceleration loads also to be negligible. The majority of the supports analyzed was found to have fundamental frequencies in excess of 33 Hz., and thus may be treated as rigid bodies. In no

case did the inclusion of the seismic acceleration of the support create an overstressed condition and in fact, as noted above, in the majori'y of cases the loads were negligible. Thus, both Applicants and the NRC Staff posited, and we so find, that the decision not to consider the effects of the seismic acceleration of supports was a valid exercise of engineering judgment. (Applicants' Exhibit 142 at 32; NRC Exhibit 207 at 35.)

2. Design Criteria Applicable to Support Rigidity

With respect to the assumption by CASE's witnesses that the support response frequency corresponded to the peak acceleration of the floor response spectra (Tr. 3100), it was demonstrated by Applicants and the NRC Staff, and we so find, that such an assumption was erroneous. Indeed, use of the assumption produces unrealistic results in that the actual worst case natural support frequency is at least double (rather than equivalent to) the building frequency and thus more rigid. The NRC Staff confirmed that the design criteria used by the pipe support design organizations ensure a rigid design. (Applicants' Exhibit 142 at 31; NRC Exhibit 207 at 35-36.)

3. Effect of Support Loads on Pipe Stresses.

CASE's witness alleged that the weight of certain types of supports would produce additional loadings on the piping which were not adequately considered in the stress analysis (CASE Exhibit 669B, Attachment 12A). The evidence produced in response to this allegation demonstrated that inclusion of this

contribution to the piping stress load produced only a negligible increase in pipe stresses, and that this allegation raises no safety concern.

Applicants' piping stress organizations (Gibbs & Hill and Westinghouse) add the contribution of the support loads to the piping loads in a piping stress analysis on a case-by-case basis if the contribution is considered significant to the piping stress analysis. Examples of these calculations were reviewed by the NRC Staff and found to be appropriate. In addition, the Staff evaluated the torsional loading on the piping from the supports. Based on their sample calculations they determined that the increases in pipe stresses from this effect also were negligible. (NRC Exhibit 207 at 36-37.) CASE did not demonstrate why this approach is unsatisfactory.

Accordingly, we find that the concerns raised by CASE's witnesses regarding the effect of the weight of supports on the piping is appropriately considered by Applicants.

4. Effect of Anchorage Slippage on Support Seismic Response

During the cross-examination of NRC Staff witnesses, a new concern was raised by CASE regarding seismic acceleration of supports. This concern was that the slippage of the support during a seismic event due to bolt holes being larger than the bolts (albeit within construction tolerances) alters the flexibility of the support so that it cannot be considered rigid (Tr. 6614-15). Applicants presented testimony on this matter, and we find that Applicants' testimony disposes of the issue.

Applicants' witness testified that neglecting the existence of oversize bolt holes is conservative. The piping system (pipe and supports) responds to earthquake excitation as a mass-elastic spring system. The seismic energy imparted to the system is by the elastic spring. However, as the system goes through a seismic vibration excursion, the effective spring energy, and thus system response, is reduced when it travels through the oversize bolt hole gap, and, therefore, the seismic response of the system is reduced by the presence of oversized bolt holes. (Tr. 7079.) No evidence was presented to contradict this testimony.

In view of Applicants' analyses, we further find that whatever effect on support stiffnesses may result from the postulated small slippages in bolt holes will have no significant impact on the piping system. Accordingly, we find this allegation to be without merit.

I. Welded Stanchions on Pipes

CASE's witnesses alleged that the effects due to welded stanchions on certain piping runs were not included in the piping stress analyses. These effects include moment restraints introduced into the piping system and local stresses in the pipe wall. (CASE Exhibit 659 at 4; CASE Exhibits 668 at 1, 668A, Item 2, and 669B, Attachments 11LL-NN and 12N-12P; Tr. 3694-98.) Applicants and Staff testified that these concerns had previously been identified and were being addressed at Comanche Peak. Their testimony demonstrated that the stresses are being adequately considered and give rise to no safety concern.

Applicants testified that the question of stresses induced by welded attachments had already been addressed at Comanche Peak. These stresses are not considered until the final as-built piping and support verification program because it is not known at the time the piping is originally designed whether welded attachments will be necessary. CASE's witnesses apparently were unaware of this analysis because they had no responsibility for addressing this issue in their positions at Comanche Peak. (Applicants' Exhibit 142 at 25-26.)

To confirm Applicants' analysis of these effects, the Staff reviewed this aspect of the as-built verification program. This review showed that in accordance with directions issued in April, 1982, that program requires consideration of restraint characteristics of as-built supports and stresses due to welded attachments. In addition, the Staff selected a typical piping run for review to determine the adequacy of Applicants' verification program in this area. Upon reviewing this run the Staff concluded that the concern of inducing moment restraints into piping systems by welded attachments was being appropriately addressed. (NRC Exhibit 207 at 38-39; Tr. 6658-61.)

With regard to local stresses, the Staff found that Applicants' use of computer programs designed to analyze these stresses is an acceptable method of analysis. The Staff also concluded that these stresses are properly combined with other stresses at the support locations. (NRC Exhibit 207 at 39.)

Accordingly, we find that the above concerns of CASE's witnesses had already been identified by Applicants as matters

for review and are being appropriately considered in the piping analyses. We find no basis for concern regarding the adequacy of Applicants' consideration of these stresses.

J. Deflections and Local Stresses

CASE's witness alleged that Applicants had not considered the effects of attached brackets to pipe support structures which give rise to deflections and local stresses (CASE Exhibits 669 at 169-72, 669B, Attachment 11A). In response to this allegation, the NRC Staff evaluated particular supports identified by CASE's witnesses as examples of these concerns. As was demonstrated by testimony of the Staff in this proceeding, in all but two instances these concerns are either incorrect or had already been identified and addressed by Applicants and the Staff. In one of those instances the matter was subsequently resolved (by testing) and in the other instance the matter will be resolved by redesign (NRC Exhibit 207 at 42). In addition, there are two matters remaining for resolution which were raised by the NRC Staff (not by CASE) in the course of their review of these allegations. These two items will be the subject of NRC confirmation. [14]

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14 Applicants are responding to the Staff's questions by providing selected piping and support calculations using actual support stiffnesses to demonstrate that the supports and piping are within the bounds of Applicants' generic stiffness study, discussed below (See NRC Exhibit 207 at 40-43). The Board finds that these questions are properly left to be addressed by Applicants and the Staff. Louisiana Power & Light Company (Waterford Steam Electric Station, Unit 3), ALAB-732, ___ NRC ___ (June 30, 1983), slip op. at 44-45, 51-52.

In response to CASE's concerns regarding excessive deflections in pipe supports, Applicants described their use of a generic stiffness value for installed supports in analyzing the seismic response of the piping system. As discussed below, Applicants have performed a detailed analysis of their generic stiffness assumptions and have conducted a sensitivity study to determine the impact on variations in generic or actual support stiffness. We have no basis to conclude that these efforts are not adequate to demonstrate the validity of Applicants' engineering approach to support stiffnesses. Accordingly, we find that this concern is resolved.

Applicants' witness described the analyses which have been performed for Comanche Peak to demonstrate the appropriateness of the selection of generic stiffness values for supports.[15] The analyses were performed for all piping systems utilizing a generic stiffness based on the type and size of piping for a system.[16] As part of the analyses, a sensitivity study was

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15 Applicants' witness also described the experience at other power plants (non-nuclear) which have withstood seismic accelerations five times greater than that postulated at Comanche Peak without failure of the piping systems. Such events have involved movements of piping systems several inches without affecting the operability of active components. Thus, Applicants' witness suggests that the concerns here (with movements on the order of 1/16 of an inch) are insignificant. (Tr. 7061-65.) We need not rely on this information for our decision, but mention it to illustrate the relatively small displacements we are considering here.

16 The Staff noted that this approach is a common industry practice and is acceptable, provided the generic stiffnesses adequately represent the stiffness of the installed supports (NRC Exhibit 207 at 40).

performed to determine the effect of those variations in the generic support stiffnesses on the dynamic seismic response of the piping systems. Through this study it was demonstrated that actual variations in support stiffnesses have little effect on the dynamic response of the piping system. The analysis also demonstrated that Applicants' use of generic stiffness values for supports adequately encompasses potential variations in frequencies of individual supports. In addition, Applicants performed a confirmatory analysis of an actual piping system on which a support had been identified with a stiffness which was a small fraction of the generic stiffness value. That analysis of the piping system confirmed that the system design was adaquate and that no modifications to the system were necessary. (CASE Exhibit 823; Tr. 7066-78.)

With respect to CASE's allegations regarding individual supports, we find those assertions to be unfounded. CASE's witness first alleged that the displacement and local stresses in one member of a particular support would exceed allowable stresses under certain loading combinations (CASE Exhibit 669, Attachments 11FF-II). However, Applicants' analysis of this support had already identified an over-stress condition that was scheduled to be rectified as part of its normal design iteration process. Applicants also committed to provide a status report of this support. (NRC Exhibit 207 at 41-42). We find no basis for concern regarding the adequacy of Applicants review of this support.

The next support identified by CASE's witness (CASE Exhibit 669B, Attachments 4G-H, 11B) was claimed to be subjected to loads that will exceed the ultimate material stress. The NRC Staff determined that the calculations utilized by CASE's witnesses were in error, leading to unrealistic stress values. Accordingly, we find that this allegation is unfounded. (NRC Exhibit 207 at 42.)

Another support, alleged to exceed the 1/16 inch deflection guideline (CASE Exhibit 669B, Attachment 13DD-13GG), was shown to have been modified during the normal course of design review to assure that the maximum deflection guideline and ASME Code requirements were satisfied (NRC Exhibit 207 at 42). Accordingly, we find that this concern also is unfounded.

Finally, the NRC Staff noted (as a matter not raised by CASE's witnesses) that there did not appear to be explicit guidelines regarding the consideration of local stresses resulting from bracket loads or for considering deflection contributions from localized effects (NRC Exhibit 207 at 42). Applicants noted that although there were no such guidelines, it was routine practice for each of the pipe support design groups to consider these effects in the review of their designs. The NRC Staff confirmed this position as a result of discussion with engineers performing the design work and the review of a large sample of vendor certified supports. (NRC Exhibit 207 at 42; Tr. 7030-31.) We find that this matter is resolved.

K. Consideration of Friction Loads

CASE's witness expressed a concern regarding the different coefficients of friction utilized by the different pipe support design organizations. In addition, CASE's witness was concerned that the design organizations utilized different loading combinations in computing frictional loads. (CASE Exhibit 659H at 3.) As demonstrated by Applicants and the NRC Staff, these differences are attributable to different engineering judgments regarding appropriate methods of analysis for determining friction loads which result from piping displacements. Those witnesses testified that these assumptions are correct, and that the differences merely result in different degrees of conservatism in these calculations Further, it was established that the piping system is insensitive to variations in frictional loads and therefore minor differences in friction parameters will not have a significant effect on piping stress analyses. (Applicants' Exhibit 142 at 28; NRC Exhibit 207 at 43-44.)

Another aspect of this allegation (first raised during CASE's cross-examination of the NRC Staff's and Applicants' witnesses) was that two of the design organizations incorrectly limited their consideration of friction loads to piping movements greater than 1/16 of an inch. In response to this assertion, the NRC witnesses testified that this limitation is properly founded on the engineering judgment of the design organizations and that consideration of movements of less than 1/16 of an inch is impractical in view of construction tolerances. (Tr. 6751-54.) There is no Code requirement specifically addressing this matter

(Tr. 6768). Further, Applicants' witness testified that the limitations on frictional movement are all based on proper engineering judgment and that a movement of 1/16 of an inch in the support in its weak direction will not give rise to any significant effects in the supports. Accordingly, these deflections do not give rise to any significant effects on the base plate of the support in the direction in which this criteria is applied, as alleged by CASE's witness. (Tr. 6759-63, 6766.) In any event, the Staff believes that there is no impact on the piping stress analyses regardless of whether the 1/16 inch limitation is used (Tr. 6755-57). All of the NRC witnesses were in agreement with Applicants' witness that this assumption was based on proper engineering judgment and that the contribution to the loads on the support would never be significant (Tr. 6768-69).

In order to clarify CASE's concern in this area, we requested that CASE's witness provide a statement as to the significance of his concern in this regard. This statement was introduced into the record (following Tr. 6824). CASE's witness summarized his position as a disagreement in engineering judgment. He admitted that there was no possibility of failure of the support as a result of these deflection stresses (Tr. 6826) and that the basic concern was that his interpretation of the engineering judgment employed by Applicants would lead to a "improper analysis", but in actuality posed no critical concern for the support (Tr. 6827-28).

We find that the decision as to the use of particular coefficients of friction in calculating friction loads is within the realm of matters properly reserved for engineering judgment and that the Applicants' pipe support design organizations have properly applied their engineering judgment in this regard. We attribute the disagreement between CASE's witness on the one hand and the numerous experts presented by Applicants and the NRC Staff on the other to their relative levels of engineering expertise. We find no reason for concern regarding the adequacy of any of the pipe support design calculations on the basis of the use of different coefficients of friction by the different design organizations.

L. Kick Loads

CASE's witness alleged that Applicants were not considering "kick loads" in the design of piping (CASE Exhibit 669B, Attachment 11RR). The NRC Staff reviewed Applicants' procedures for consideration of kick load forces (a component of force imposed by a support not directly aligned (when installed) with its design alignment) in pipe stress analyses. It was determined that the Applicants considered such forces if the support is misaligned by five degrees or more. This approach is considered an appropriate manner of addressing the kick load force component. (NRC Exhibit 207 at 45.) Accordingly, we find this allegation to be erroneous and Applicants method of considering kick loads to be acceptable.

M. Modelling of Wide Flange Members

CASE's witness expressed a concern regarding the modelling of wide flange members by ITT-Grinnell using large torsional rigidity values. This concern was that the assumption of a large torsional rigidity value incorrectly assumes that the wide flange shapes in pipe supports are torsionally stiff. (CASE Exhibits 669 at 180-81; 669B, Attachment 11TT.) As demonstrated by Applicants and the NRC Staff, his concern is invalid, and in fact, CASE's witness apparently was not aware of the purpose of this modelling technique. In any event, this procedure is no longer in use at Comanche Peak, so the issue is moot.

CASE's witness misinterpreted the intention of the criterion by which high torsional rigidity values were employed to model wide flange members. The purpose of using such values is to maximize the torsional moment in the support so as to evaluate conservatively the torsional stress in the wide flange members (Applicants' Exhibit 142F at 8-9). The NRC Staff confirmed that this was the purpose of the procedure and reviewed both the procedure and the implementation of that procedure. The Staff concluded that the use of these values, with the subsequent calculation of torsional shear stresses using the correct torsional constant values for the materials, results in a conservative calculation. (NRC Exhibit 207 at 45-46; Tr. 6628-29, 6634, 6636-37.)

In the course of the examination of the NRC Staff panel, CASE raised for the first time an additional question as to whether the actual deflection of the support was recalculated

using the actual torsional constants following the use of the high torsional rigidity values (Tr. 6642-44). Applicants' witnesses stated that the procedure which utilized the high torsional rigidity value dictated that the value be utilized in modelling only deflection in the unrestrained direction of the support. Deflections in the unrestrained direction are not of concern to the support design and deflection criteria are not required in the unrestrained direction. (Tr. 7103-07.) Thus, whether this deflection is calculated is not significant to the concern raised by CASE's witness. (Tr. 7135-36.) In any event, the procedure of concern was discontinued and the final support verification process will not utilize the procedure (Tr. 7136). Accordingly, we find the concern is clearly invalid and, in any event, that the concern is no longer relevant to the design of supports at Comanche Peak.

N. Effect of Cold Forming on Ductility of Tube Steel

CASE raised a question during the cross-examination of Applicants' witnesses regarding the effect of cold-forming on the ductility of tube steel (Tr. 5078). The NRC addressed this concern by performing a literature review and identifying the results of tests conducted to quantify the effects of cold-forming (NRC Staff Exhibit 207 at 46). The results of the Staff's review demonstrate that CASE's concern is unfounded.

Ductility of a steel is desirable in the design of pipe supports in that the plastic deformation mechanism is relied upon in the design process to accommodate detrimental effects from secondary stresses. The cold-forming of structural steels will

increase the yield and ultimate strengths of a material, but by different amounts, thereby reducing the gap between the yield and ultimate strengths. The premise of CASE's allegation is that this reduction creates a decrease in the elongation capability or ductility of the cold-formed steel. (NRC Exhibit 207 at 46.)

In order to address this matter, the Staff examined several technical papers regarding the effects of cold-forming on structural steel. The results of their review demonstrated that the material in question (A500-Grade B Tube Steel) remains sufficiently ductile for the purposes in which it is used. Thus, the Staff concluded, and we so find, that CASE's allegations are unfounded. (NRC Staff Exhibit 207 at 46-47.)

During cross-examination of Applicants' and the Staff's witnesses CASE raised yet another new question regarding the ductility of cold-formed steel. Specifically, CASE contended that while allowables were published in the ASME Code for tube steel, welding on the steel reduced the yield of the material, and CASE's witness was not aware of whether Applicants reduced the allowables for steel to account for this welding phenomenon. CASE's witness was incorrect, however, in that the ASME Code addresses only welded materials. Thus, the allowables for cold-formed steel in the Code account for welding of cold-formed steel. (Tr. 6789-92.) CASE's witness then attempted to utilize an ASME Code Case to support his assertion that the allowables at Comanche Peak were not consistent with the Code (Tr. 6792-96). However, it was shown that the Code Case was not in effect at the time the components for Comanche Peak in question here were

ordered and, therefore, is inapplicable (Tr. 6801, 6806-08). In any event, had the change in allowables in the Code Cases affected the safety of any Code component, there would have been a ruling by the ASME to that effect, which was not done here (Tr. 6810). Accordingly, we find that this aspect of CASE's concern is also without merit.

O. Operating Condition Loads

CASE's witness alleged that computational errors had been made in the analyses of the loads on a particular support. Specifically, he claimed that because the emergency operating condition load on the support was calculated to be less than the normal and upset operating condition loads, there must have been an error in the piping analysis (CASE Exhibits 669 at 130, 669B, Attachment 8T-8U). As discussed below, the NRC evaluated the Applicants' piping analysis for that support and found the analysis to have been correctly performed. Accordingly, we find no basis for this allegation.

The NRC Staff testified, and CASE did not refute, that the design, normal and upset conditions include the operating basis earthquake ("OBE") loads and the emergency condition includes the safe shutdown earthquake ("SSE") loads. Since the ground acceleration for the SSE is twice the OBE, one would expect the SSE loads to be higher than the OBE loads. However, the ratio of resulting loads is actually less than two, due primarily to the increased damping allowed when considering the SSE. Since the shape and magnitude of the floor response

spectra varies significantly with the assumed damping, the accepted method of response spectra analysis does occasionally result in theoretical loads for OBE greater than SSE. However, designing (as Applicants have done) for both load combinations at the allowable stress levels appropriate for each combination conservatively envelopes both levels of earthquake loading. (NRC Exhibit 207 at 48; Tr. 6780-89.) Accordingly, we find that this support was properly analyzed and that CASE's witness simply was unaware of the method of seismic analysis that determines the appropriate loadings on the support.

Finally, CASE raised during cross-examination a somewhat obscure question as to the need for increasing allowable stresses for the emergency and faulted conditions when the loads calculated on the supports actually decreased in those conditions (Tr. 6786-87). CASE's witness apparently could not grasp the logic of the seismic analyses applicable to this support. As demonstrated by the NRC Staff, the calculation of seismic loads is properly based on the appropriate response spectra at various elevations within the plant. Therefore, the worst case scenario has been evaluated, although it is likely that the spectra within the building will differ depending on the natural frequencies of the system. (Tr. 6784-87.) Accordingly, we find this aspect of CASE's concern also to be meritless.

P. Welding on Pipe Supports

CASE's witness made various allegations regarding the welding practices on pipe supports at Comanche Peak (CASE Exhibit 669 at 111-18). This witness did not purport to be an expert in

welding, and we find that he was not (CASE Exhibit 669 at 316). As discussed below, he was unfamiliar with the welding codes applicable to the welding with which he was concerned, and his allegations were simply unfounded and without merit.

The first allegation concerns "welded stepped connections", (perpendicular joints between pipes or tubes of different sizes) which CASE's witness claimed were prohibited by the AWS Code (CASE Exhibit 669 at 111-18). The NRC Staff examined the support referenced by CASE's witness in connection with this allegation and determined that the requirements of the AWS Code alleged to apply were not applicable. These connections instead are governed by the ASME Code. The Staff confirmed that Applicants' pipe stress analyses employed the applicable Code requirements and verified that those requirements were satisfied. Further, the criteria referred to by CASE's witness are applicable only to pre-qualified welding and thus even had the welding been governed by AWS standard, that criteria would not apply because site welding procedures for safety-related supports are qualified. (NRC Exhibit 207 at 49; Applicants' Exhibit 142F at 7.)

On a related question, although not raised by CASE's witness, the NRC Staff examined the adequacy of design requirements regarding perpendicular tube-to-tube welded connections. The Staff evaluation of this question found that such joints are generally acceptable as long as the member width ratio is greater than 0.4. It was determined that the member width ratios at Comanche Peak are generally 0.67, although one member with a ratio of 0.5 was identified. We find on that basis

that support member width ratios at Comanche Peak are adequate. In addition, because the analysis referred to by the Staff was concerned with ultimate strength values (with failure modes beyond the elastic limit), and the tube steel joints utilized in the designs at Comanche Peak are all designed not to exceed yield loads (i.e., will remain in the elastic range), they would not approach the ultimate strength values tested in the evaluation reviewed by the Staff. (NRC Exhibit 207 at 49-50; NRC Staff Testimony at 12, following Tr. 6402.)

The next concern raised by CASE's witness was that undersized fillet welds existed on three supports provided with his testimony (CASE Exhibit 669B, Attachment 6A). The NRC Staff reviewed these welds and determined that each satisfied the minimum fillet of weld requirements of the ASME Code (NRC Exhibit 207 at 50). In addition, Applicants testified that all welds on supports are examined in the final support review (through which the support drawings Mr. Doyle submitted had not passed) and no support will pass that review if there is a violation of minimum weld criteria (Applicants Exhibit 142F at 8). Finally, during the Staff review of 100 vendor certified supports (NRC Exhibit 207 at 54) all welds were evaluated for adequate size and no discrepancies were identified (NRC Exhibit 207 at 51). Accordingly, we find no basis for concern regarding this aspect of the allegation.

Finally, CASE's witness alleged that fillet welds at skewed joints (skewed welds) violated the AWS Code (CASE Exhibit 669B, Attachment 6). However, these welds are governed by the ASME

Code and Applicants' welding procedures were shown to properly address skewed welds (NRC Exhibit 207 at 51; Applicants' Exhibit 142F at 8). Interestingly, even the weld criteria submitted by CASE's witness (CASE Exhibit 669B, Attachments 6B-F) would have allowed such welding (NRC Exhibit 207 at 51). Accordingly, we find this allegation to be unfounded.

In sum, we find no merit in the allegations regarding welding practices on pipe supports at Comanche Peak.

Q. Section Property Values

CASE's witness alleged that two different sets of member property values for tube steel sections were used in the design of pipe supports for Comanche Peak and that as a result the reactions and deflections calculated for these supports could be off by as much as 25%. In addition, he believed that the difference in these member properties would require Westinghouse to perform a reevaluation of its piping systems. (CASE Exhibit 659 at 5; Tr. 3701-02.) As demonstrated below, the difference in member property values is relatively small and results in only minor variations in stress levels calculated for supports. Thus, there is no basis for concern regarding the adequacy of the design of pipe supports at Comanche Peak as a result of this difference in member property values.

The property differences to which CASE's witness referred arose because the PSE design organization used different manual to obtain these rules over time. Prior to 1981, PSE used the seventh edition of the American Institute of Steel Construction ("AISC") properties. From January 1981 to January 1982, it used

the values set forth in the Welded Steel Tube Institute 1974 Manual of Cold Form Welded Structural Steel Tubing. Since January 1982, PSE has been using the eighth edition of the AISC properties. The Welded Steel Tube Institute since has also amended its 1974 Manual to agree with the eighth edition of the AISC properties. (Applicants' Exhibit 142 at 29; NRC Exhibit 207 at 52.)

Applicants recognized these differences in member properties, and therefore, reanalyzed all large bore and Class 1 small bore pipe support designs using the latest member property values. This reanalysis was initiated in January, 1982, well prior to the raising of this allegation. For small bore Class 2 and 3 pipe supports, the difference in member properties was considered to be sufficiently small to be insignificant and not to warrant reanalysis of those supports. The NRC Staff confirmed the validity of this judgment by determining that the maximum difference in stresses in supports calculated using the different member property values would not exceed 8%. (Applicants' Exhibit 142 at 29; NRC Exhibit 207 at 52.) Accordingly, we find that Applicants' treatment of the different member property values is appropriate and raises no concern as to the adequacy of the pipe support designs.

During cross-examination of the NRC Staff, CASE's alleged that it was also concerned that the other pipe support design organizations (NPSI and ITT-Grinnell) utilized the seventh edition of the AISC Manual (Tr. 6859-62). It was shown that this concern is meaningless in that use of the seventh edition values

is conservative and could not adversely affect the pipe support design. In any event, as was previously demonstrated, the variations in these member properties are insignificant and their impact on the design of supports would be negligible. (Tr. 6867-70.)

Finally, with respect to the allegation that Westinghouse must reanalyze its piping and support analyses, Westinghouse has instructed the design groups at Comanche Peak that the support stiffnesses (which would be affected by these variations in member properties) may increase by 100% and not have any appreciable affect on the piping and support analysis. In that the difference in the member property values in question here is small, there is no valid concern for the pipe support design as a result of these changes. (Tr. 5016-17.) Further, the safety factors incorporated into the piping designs are on the order of magnitude of three Thus, revisions to these manuals are very small in comparison to the total factors of safety included in the designs. (Tr. 5018-20.)

Accordingly, we find that the Applicants have properly addressed the question of variations in member properties and that in any event such variations had they not been addressed could create no concern as to the adequacy of pipe support designs at Comanche Peak.

R. Welding Over Girth Welds

CASE's witness alleged that welding codes required that attachments to piping not be welded over existing pipe (girth) welds (CASE Exhibit 669B, Attachment 2; CASE Exhibit 669 at 82-

85, 314-16.), and submitted examples of supports which he believed involved welding of attachments over girth welds (CASE Exhibit 669B, Attachment 2). As demonstrated below, CASE's witness incorrectly interpreted the applicable welding Codes, and in any event, the supports submitted as examples of this concern were not even safety-related supports.

The welding under consideration here is governed by the ASME Code. Neither ASME Code Sections III nor XI proscribe the welding of lugs, pads or material attachments over existing welds. In fact, it is often necessary to weld support skirts over Category A welds in Class I nuclear vessels and other components. In addition, reinforcing pads around nozzles must often be placed over portions of either Category A or B welds. As for the possibility of welds not being available for inspection, Section XI does not consider there to be any problem in performing any in-service inspection ("ISI") around local areas even where they are covered and inaccessible to ISI. Thus, there is no Code basis for this allegation. Applicants nonetheless have established measures to identify those locations to evaluate the need for examination or evaluation. (Applicants' Exhibit 142F at 3-4.) We conclude that Applicants have properly considered the implications of the welding practices under consideration here.

With respect to the particular supports submitted by CASE's witness, it was determined that these supports were on Class V piping which is not safety-related and thus is not included in the ASME Section XI inspection program. In addition, these

supports did not involve welding over of girth welds as alleged but instead employed removable clamps that would not preclude the inspection of the welds had it been necessary. Accordingly, the concern with respect to individual supports is invalid. (NRC Exhibit 207 at 53.)

S. Damage to Support During Hydrostatic Testing

CASE's witnesses alleged that they observed a support "fail" during hydrostatic testing of Unit 1. They contended that they observed paint cracking from the support during that test. (CASE Exhibit 669 at 72-73; CASE Exhibit 669B, Attachment 11XX.) Applicants and the Staff disproved the allegation by demonstrating that the damage to this support was unrelated to the hydrostatic testing or the adequacy of its design.

Damage to this support was identified long before hydrostatic testing was conducted. Identification of damage to the support was made during of a field walk down prior to hydrostatic testing and was determined to have been caused by installation and adjustment of the system. Further, the support carries no significant loads during hydrostatic testing or dead loads during operation, and could not have failed during that testing. Thus, the paint chipping observed most likely occurred as a result of normal flow-induced vibration in the system during the testing. Applicants have since replaced the original tube steel member which had been damaged with a new piece of structural tubing. (Applicants' Exhibit 142F at 6; Tr. 4791-93, 5010-11, 5202-03; NRC Exhibit 207 at 54.) Accordingly, we find

that the damage to this support did not occur as a result of hydrostatic testing and the allegation in this regard is incorrect.

T. Upper Lateral Steam Generator Support and Moment Restraint

As noted above, to confirm the adequacy of Applicants' assessment of LOCA thermal expansion effects in supports which span between two structures, the NRC Staff evaluated two restraints installed at Comanche Peak. The first was a floor-to-wall moment restraint (CASE Exhibit 669B, Attachments 9Q-S). The second was a wall-to-wall steam generator upper lateral restraint (Applicants' Exhibit 157). As a result of its analyses, the Staff concluded that the Applicants consideration of LOCA thermal expansion effects satisfy FSAR commitments. CASE's witness challenged these conclusions and argued that thermal expansion effects were not adequately evaluated for those two restraints. We requested oral argument specifically on these two restraints during the May hearing session. We find that these restraints are adequately designed to perform their intended function even under stresses imposed as a result of LOCA-induced thermal expansion of the restraints.

1. Upper Lateral Steam Generator Supports

CASE's witness made several allegations regarding the adequacy of Applicants' evaluation of the effects of thermal expansion on this support (CASE Exhibit 750). Specifically, the witness contended that Applicants had miscalculated the stiffnesses of the walls to which the restraint is attached, had

failed to consider various loads on the restraint resulting from the interaction of the restraint with the wall, or to consider loads resulting from the interaction of the restraint and the steam generator (CASE Exhibits 761 at 3; 762 at 2; 805; Tr. 6031-34). CASE's witnesses contended these factors were significant, and submitted his own calculations to support his conclusion (CASE Exhibit 761C) As shown below, the differences between the different calculations are not significant, which leads us to find that Applicants' calculations satisfactorily address the matter. Further, it was demonstrated that effects alleged not to have been considered were, in fact, either considered or were insignificant. Accordingly, we find that this restraint will perform its intended function under LOCA-induced thermal expansion loads.

CASE's witness alleged that Applicants should have employed a different analytical method to calculate the stiffness of the walls to which the restraint is attached and that Applicants had failed to consider certain walls and platforms attendent to the steam generator walls which were claimed to affect the calculated wall stiffness (CASE Exhibits 761 at 3; 805 at 2). In response to this allegation, Applicants' expert witness testified that the differences between the stresses and deflections calculated by Applicants and CASE were insignificant and were well within the degree of accuracy one would expect in such calculations. Accordingly, he concluded, and we so find, that the calculations actually confirm the appropriateness of Applicants' calculations. (Tr. 6041-44, 6050.)

In addition, Applicants' witness testified, without contradiction, that because it was not possible to calculate the deflection of the wall to a high degree of accuracy, proper engineering practice requires consideration of the effects of the lower bound value of wall deflection (zero) and the upper bound value of wall deflection (the free growth of the steal beam, approximately 0.20 inches). He testified that it is obvious that the wall can resist the lower bound value (if there is no strain, i.e., zero deflection, there is no stress). The resultant effects on the support from this lower bound assumption (zero wall deflection) are adequately accounted for by a number of self-relieving mechanisms and conservatisms included in the support design. Thus, there is no concern of the ability of the support or the walls to perform their intended functions under these conditions. As for the upper bound case, the resulting deflection is much less (by a factor of 30-40) than the maximum deflection the wall could withstand. In addition, the stresses in the reinforced concrete wall are allowed to exceed normal allowables for this type of load. Accordingly, such an upper bound deflection also would be acceptable. In actuality then, CASE's attempts to "exactly" calculate the predicted deflection (between the lower and upper bound) is meaningless. The NRC Staff agreed with Applicants' basic approach to this analysis. (Tr. 6046-51, 6059-63.)

We have no reason to disagree with the conclusions reached by Applicants and accordingly find that the restraint will perform its intended function, as designed.

CASE also contended that because the restraint is attached to a point on the reactor cavity wall which is below the elevation of the reactor missile shield on the other side of the wall, concrete could fall onto the reactor if the wall failed (Tr. 6027-30). Applicants' witness testified that the mechanisms for relieving stress in the concrete involved, if anything, only minor cracking and posed no concern for concrete failing on the reactor (Tr. 6080-84). We consider this allegation to be adequately resolved.

The final concern regarding this restraint was that the steam generator itself would impose lateral stresses on the restraint due to its thermal growth during normal operation and under LOCA conditions (Tr. 60311-34). Applicants testified that snubbers were used in one direction and gaps were provided in the other direction to account for movements in the steam generator (Applicants' Exhibit 157, notes 3 and 4; Tr. 6036-39). CASE's witness seemed to agree that this would satisfactorily resolve his concern, but claimed that the snubbers did not exist at Comanche Peak (Tr. 6040). We indicated at that time that we considered this topic to be adequately addressed, if there was evidence that the snubbers did exist at the plant. Applicants subsequently confirmed their existence and the Judge McCollum personally observed them during a site tour. Accordingly, we find no basis for this concern and consider it to be resolved.

In sum, we find that each of the assertions made by CASE regarding the consideration of LOCA-induced thermal expansion stresses in the upper lateral steam generator restraint have been

shown to be incorrect or to concern matters not significant to the ultimate finding that the restraint is capable of withstanding those stresses. There is substantial evidence demonstrating the ability of these restraints to perform their intended function.

2. Moment Restraint

CASE alleged that thermal stresses induced in the floor-to-wall moment restraint evaluated by the NRC Staff were not adequately considered (CASE Exhibit 761 at 2). This concern related to two aspects of the design of the moment restraint. The first was that stresses were not considered at each of the joints of this restraint. The second was that the shear keys welded to the base plates of this support were not considered.

The question of whether thermal stresses in each member of the support frame (and thus at each joint) need be considered was shown by Applicants and the NRC Staff to fall within the scope of our decision regarding consideration of thermal stresses under the ASME Code. Accordingly, we determined that this allegation need not be further pursued. (Tr. 6269.)

The second question concerned the effect of shear keys in the base plates of this support on the analysis of thermal expansion of the suport. We have already addressed the question of shear keys (<u>see</u> Section II.D.2.f , <u>supra</u>) and found that exclusion of the shear keys from the analysis was conservative. Accordingly, we conclude that this allegation also is without

merit and provides no reason to question the ability of this support to withstand stresses resulting from its thermal expansion under LOCA conditions.

U. CAT Report Regarding Pipe Supports

The NRC Construction Appraisal Team ("CAT") conducted an inspection of Comanche Peak in January, 1983. Part of that inspection involved a review of aspects of pipe support construction and design interfaces with construction. The NRC Staff presented testimony of the team leader for the Comanche Peak CAT to describe this aspect of the CAT Report.[17] As stated in that testimony, the objective of the CAT inspection was to review the facility hardware being constructed at Comanche Peak, including pipe supports, to assess the construction quality control program. In this regard, the CAT inspection for Comanche Peak reviewed installed and QC-accepted safety-related piping and pipe supports for conformance to engineering design, and reviewed quality-related records regarding pipe supports to determine whether the completed work was reflected in quality-related records. The CAT reviewed the Applicants' program for design change controls, but did not review the adequacy of design of the supports. (NRC Staff Testimony of A.B. Beach, following Tr. 6283, at 2-3.) The NRC witnesses regarding pipe support design reviewed the CAT Report and found no reason to change their previous conclusions and findings regarding the adequacy of pipe support designs (NRC Staff Supplemental testimony, following Tr.

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17 I&E Report 83-18/83-12, April 11, 1983 (NRC Exhibit 206, following Tr. 6283).

6402 at 8). In short, these two Staff groups reviewed separate aspects of Applicants' pipe support program. Because the issues raised here concern the adequacy of design of pipe supports, we will not address in this decision the aspects of pipe support construction considered in the CAT Report.

III. OTHER MATTERS CONSIDERED

We have addressed in this decision each of the issues raised by CASE (either as part of its initial allegations or during the course of the hearing) regarding the design of pipe supports at Comanche Peak which we perceive could have affected our determination as to the adequacy of those designs. To the extent CASE may have raised other questions, we have considered those also, and found they are either without merit or are insignificant and could not affect our determination here.

IV. CONCLUSION

We conclude that there is reasonable assurance that the design of pipe supports at Comanche Peak will have adequately considered appropriate loads and stresses, in accordance with applicable Codes and NRC Regulations. We conclude that those supports are designed to perform their intended functions under all conditions to which they might be subjected. We note that we have reached these conclusions on the basis of extensive evidence on all aspects of the pipe support design process and on the allegations regarding the adequacy of the resultant designs.

Our examination has gone into extraordinary detail regarding the pipe support design program at Comanche Peak, even to the extent of considering the modelling techniques, design assumptions and calculations for many individual supports and types of supports. We have reviewed the iterative design process for piping and support design in detail. Most of the allegations were rooted in a disagreement by CASE's witnesses with the engineering judgment of Applicants' and the Staff's experts. Given this posture of the issues we conclude that we should find for Applicants on these matters, if for no other reason than the relative expertise of the witnesses. However, we have not based these findings and conclusions solely on an assessment of witness expertise. We have pursued all questions in depth to assure that a proper foundation exists for their technical resolution, including those matters the disposition of which rests on engineering judgment. We conclude that such a foundation has been shown in all instances. Accordingly, we conclude that the design of pipe supports at Comanche Peak satisfies all applicable Codes and NRC Regulations. To the extent aspects of those designs are not governed by particular Codes or Regulations, we conclude that Applicants have employed sound engineering judgment in the development of those designs.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "[Applicants' Proposed Findings of Fact and Conclusion of Law in the Form of a] Partial Initial Decision (Concerning Pipe Support Design Questions)," in the above-captioned matter were served upon the following persons by mail, first class postage prepaid, or by overnight express mail (*) this 5th day of August 1983.

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