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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of APPLICATION OF TEXAS UTILITIES GENERATING COMPANY, *ET AL.* FOR AN OPERATING LICENSE FOR COMANCHE PEAK STEAM ELECTRIC STATION UNITS #1 AND #2 (CPSES)	§	Docket Nos. 50-445 and 50-446

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CASE'S BRIEF REGARDING CONSIDERATION OF LOCA IN DESIGN CRITERIA FOR PIPE SUPPORTS

Pursuant to the Licensing Board's April 7, 1983, Order during the conference call with all parties, CASE (Citizens Association for Sound Energy), Intervenor herein, hereby files this, its Brief Regarding Consideration of LOCA in the Design Criteria for Pipe Supports.

BACKGROUND

On Thursday, April 7, 1983, a telephone conference call was initiated by the Licensing Board with all parties. During that call, the Board ordered that all parties file briefs "in order to provide a correct interpretation of the application of the Commission's regulations to whether or not LOCA" (loss-of-coolant accident) "conditions must be considered in the design criteria for pipe supports." The Board Chairman stated that in doing that, he would like to have "a logical discussion of the relationship between the different regulatory materials including the design criteria, the standard review plan, the staff guidance, the staff practice, and applicable industry codes."

DISCUSSION

The inclusion of LOCA in the design criteria for pipe supports is one of the underlying issues of the concerns of CASE witnesses Mark Walsh and Jack Doyle. Indeed, it was because Messrs. Walsh and Doyle were instructed to discontinue including LOCA conditions in their STRUDL (Structural Design Language computer program) calculations (in addition to their other concerns) that they resigned their positions at Comanche Peak. (See Messrs. Walsh and Doyle's testimonies.)

Regulations, Codes, etc., require the consideration of LOCA conditions in the design criteria for pipe supports.

10 CFR Part 50, Appendix A - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

There are several portions of Appendix A which are pertinent to the subject of the consideration of LOCA conditions. For instance:

INTRODUCTION

"Pursuant to the provisions of 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

"These General Design Criteria establish minimum requirements for the principal design criteria...

"The development of these General Design Criteria is not yet complete... Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:...

"(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)...

"It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

"There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety..."

--(Emphases added)

"CRITERIA - I. Overall Requirements

"Criterion 1 -- Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the plant."

--(Emphases added.)

"Criterion 2 -- Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena...without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:...(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

--(Emphases added.)

> "Criterion 4 -- Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit." (Emphases added.)

As can be seen from the preceding, these minimum requirements mandate that all components important to safety be designed with consideration given to LOCA. Further, the Introduction to Appendix A clearly states that applicants must consider in the design of the plant whatever is necessary to satisfy the necessary safety requirements. Specifically included are the design requirements to suitably protect against postulated loss-of-coolant accidents (LOCA).

In addition, Criterion 1 requires that if generally recognized codes and standards are used (such as the ASME Code), those codes and standards must be evaluated and supplemented or modified as necessary to assure a quality product that will operate the way it is supposed to. In other words, operability must be assured.

A quality assurance program must be implemented for those components to assure that they will satisfactorily perform their safety functions, and appropriate records of the design shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the plant.

Criterion 2 requires that components important to safety shall be designed such that their operability is assured, and that the design bases for those components shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and the importance of the safety functions to be performed.

Criterion 4 requires that components important to safety shall be designed with proper consideration given to environmental conditions associated with postulated accidents, including loss-of-coolant accidents. This would also include the environmental temperature following a LOCA and the effects from that increased temperature.

The Criteria found in 10 CFR, Part 50, Appendix A, are the primary controlling regulations of the Nuclear Regulatory Commission regarding whether or not LOCA must be included in the design criteria for pipe supports. As demonstrated in the preceding, the regulations clearly state that LOCA conditions must be considered. Further, the regulations clearly state that all structures, systems, and components important to safety must remain operable and be able to function as they are intended to, even under LOCA conditions.

Regulatory Guide 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports," Revision 1, January 1978

Although they do not carry the force of law or of the regulations as set forth in 10 CFR, Part 50, Appendix A, Regulatory Guides published by the NRC offer guidance for compliance with regulations. Such guidance regarding whether or not LOCA conditions must be considered in the design criteria for pipe supports is contained in Regulatory Guide 1.124, Rev. 1, January 1978 (CASE Exhibit 743). In their discussion regarding applicable Regulatory Guides, Applicants state regarding this Regulatory Guide:

"Regulatory Guide 1.124

"Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports"

"Discussion

"All non-NSSS" (Nuclear Steam Supply System = Westinghouse) "supplied Class I linear-type supports comply with Revision 1 (1/78) of this regulatory guide.

"Also refer to Appendix 1A(N)."

--Applicants' FSAR, page 1A(B)-52

(Appendix 1A(N) discusses the CPSES NSSS position on Revision 1 (1/78) of Regulatory Guide 1.124. Since Westinghouse does not, as far as CASE is aware, supply pipe supports at CPSES, this would not be applicable. In any event, Messrs. Walsh and Doyle's concerns deal primarily with pipe supports supplied by NPSI and ITT Grinnel, and for that reason Appendix 1A(N) would not be applicable. It appears to CASE that the only reason for including the discussion in Appendix 1A(N) is the fact that Applicants relied on Westinghouse to prepare FSAR Section 3.9N.3, ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures, as well as Section 3.9N.1, Special Topics for Mechanical Components /1/.)

As discussed in Regulatory Guide 1.124, it was formulated with the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 2, in mind. The Guide states, in part:

"The failure of members designed to support safety-related components could jeopardize the ability of the supported component to perform its safety function.

"This guide delineates acceptable levels of service limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of Class 1 linear-type component supports as defined in Subsection NF of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This guide applies to light-water-cooled reactors..." (A. INTRODUCTION; emphasis added.)

/1/. See CFUR's First Set of Interrogatories to Applicants dated 2/26/81, and Applicants' Answers to CFUR's First Set dated 4/13/81, Questions 1, 2, 3, 5, 7, 11, 12, 15, 18, 19, and especially 17.

"Load-bearing members classified as component supports are essential to the safety of nuclear power plants since they retain components in place during the loadings associated with normal and upset plant conditions under the stress of specified seismic events, thereby permitting system components to function properly. They also prevent excessive component movement during the loadings associated with emergency and faulted plant conditions combined with the specified seismic event, thus helping to mitigate the consequences of system damage. Component supports are deformation sensitive because large deformations in them may significantly change the stress distribution in the support system and its supported components.

"In order to provide uniform requirements for construction, the component supports should, as a minimum, have the same ASME Boiler and Pressure Vessel Code classification as that of the supported components. This guide delineates levels of service limits and loading combinations, in addition to supplementary criteria, for ASME Class 1 linear-type component supports as defined by NF-1213 of Section III.

"...the (ASME) Code does not specify loading combinations...guidance is required to provide a consistent basis for the design of component supports..." (B. DISCUSSION; emphases added.)

"The design of component supports is an integral part of the design of the system and its components. A complete and consistent design is possible only when system/component/component-support interaction is properly considered. When all three are evaluated on an elastic basis, the interaction is usually valid because individual deformations are small. However, if plastic analysis methods are employed in the design process, large deformations that would result in substantially different stress distributions may occur." (B. 4. Large Deformation; emphases added.)

"C. REGULATORY POSITION. ASME Code"..."Class 1 linear-type component supports...should be constructed to the rules of Subsection NF of Section III as supplemented by the following...

"...5. Component supports subjected to the combined loadings of system mechanical loadings associated with (1) either (a) the Code design condition or (b) the normal or upset plant conditions and (2) the vibratory motion of the OBE should be designed within the following limits:...

"a. The stress limits of XVII-2000 of Section II and Regulatory Position 3 of this guide should not be exceeded for component supports designed by the linear elastic analysis method. These stress limits may be increased according to the provisions of NF-3231.1(a) of Section III and Regulatory Position 4 of this guide when effects resulting from constraints of free-end displacements are added to the loading combination.

"...8. Component supports in systems whose normal function is to prevent or mitigate the consequences of events associated with an emergency or faulted plant condition should be designed within the limits described in Regulatory Position 5 or other justifiable limits provided by the Code..." (Emphasis added.)

"D. IMPLEMENTATION...If an applicant wishes to use this regulatory guide in developing submittals for construction permit applications docketed on or before January 10, 1978, the pertinent portions of the application will be evaluated on the basis of this guide." (Emphasis added.)

As stated in the preceding, the proper functioning of component supports is vital to assure the safety of nuclear power plants, not only during normal and upset plant conditions under the stress of specified seismic events, but also during emergency and faulted plant conditions (which would include LOCA) combined with the specified seismic event[2].

Although it is postulated, there is no accurate method of determining exactly which item may experience an emergency or faulted condition. For this reason, and for the reasons stated in the preceding paragraph, each pipe support must be considered in two ways: (1) as though it were supporting the item which was involved in the faulted load of a LOCA; and (2) as though it were supporting the item which was not involved with the faulted load but receives the effects of the LOCA. One of the effects of a LOCA which must be considered is a probable increase in air temperature to 280°F within two minutes[3]. (It should be noted that even this is not conservative[4].)

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2 Regulatory Guide 1.124 defines "Emergency Plant Condition" as "Those operating conditions that have a low probability of occurrence" and "Faulted Plant Condition" as "Those operating conditions associated with postulated events of extremely low probability." Both would include LOCA.

3 See CASE Exhibit 659, page 2, 2nd full paragraph, Mark Walsh 7/28/82 direct testimony as corrected on transcript pages 3127-3128; and CASE Exhibit 659C, page 2, attachment to Walsh testimony, 10/9/81 letter from Gibbs & Hill to TUGCO.

4 See CASE Exhibit 659C, page 1, last paragraph.

Only by considering each pipe support in these two ways can the operability of the supports (and thereby the operability of the items which they are supporting) be assured under all conditions which they may experience. This assurance of operability is required by 10 CFR Part 50, Appendix A, by Regulatory Guide 1.124, and by simple logic. All items important to safety must be designed to assure that they will retain their capability to perform their required safety functions.

Considering a pipe support in the second of the two ways (as though it were supporting the item which was not involved with the faulted load but receives the effects of the LOCA) is discussed under C.8. and C.5.a. of Regulatory Guide 1.124. As stated therein, it is permissible to increase the stress limits according to the provisions of NF-3231.1(a) of Section III and Regulatory Position 4 of the Regulatory Guide only when effects resulting from constraints of free-end displacements are added to the loading combination.

NF-3231.1(a) of Section III of ASME (CASE Exhibit 744, page 37) states:

> "Design, Normal, and Upset Conditions. The stress limits for Design, Normal, and Upset Conditions are identical and are given in Appendix XVII. The allowable stress for the combined mechanical loads and effects which result from constraint of free-end displacements (NF-3213.10), but not thermal or peak stresses, shall be limited to three times the stress limits of XVII-2000." (Emphasis added.)

Since there has been much discussion both in prefiled and cross-examination testimony regarding thermal stresses and thermal expansion stresses, and since the ASME Code is not as specific on this point as it might have been[5], it is necessary at this point to clarify exactly what Messrs. Walsh

[5] Applicants' witness Reedy confirmed this difficulty in the ASME Code during his cross-examination testimony. In response to Mr. Walsh's question "Does ASME have different definitions for different subsections?" Mr. Reedy stated in this regard, however, "I think for your stress analysis terms that you're considering, the definitions are the same." (Tr. 5222/24-5223.)

and Doyle are in fact concerned with in this regard. Their concerns relate not to thermal stress as specifically defined in the ASME Code, but to what is defined in the Code as thermal expansion stress (or constraint of free-end displacement).

The Code is specific in neglecting the effects of thermal stress as specifically defined in the Code, but the Code also recognizes the complexity of thermal conditions and has subdivided the effects of temperature into two categories: thermal stresses; and constraint of free-end displacement (or expansion stress).

A thermal stress as defined in NB 3213.13 (CASE Exhibit 699) is "a self-balancing stress produced by a non-uniform distribution of temperature or by differing thermal coefficients of expansion." An example of a non-uniform distribution of temperature is where a pipe is in contact with the pipe support. At the point of contact, the temperature of the steel is the same as the temperature of the pipe. But a few inches away from the point of contact, the temperature has decreased a considerable amount. These changes of temperature will cause a stress distribution within the pipe support and will occur during the life of the plant and perhaps will not adversely affect the overall structural capacity of the pipe supports. An example of a thermal stress of differing thermal coefficients of expansion would be the case where the weld electrode material has a coefficient of expansion different from the base metal. This also will have the same effect as the non-uniform distribution of temperature.

The ASME Code defines free-end displacement in NF-3213.10 (CASE Exhibit 744, page 32):

"Free End Displacement consists of the relative motions that would occur between an attachment and connected structure or equipment if the two members were separated."

Two examples of this form of motion are differential movements, and expansion and contraction of a member. For differential movement, consider a member attached to a ceiling and attached to a floor. In a seismic event the ceiling may want to move downward 1/2 inch and the floor may want to move up 1/2 inch. In that case, the stress in the member in compression would be comparable to the stress that would occur if the member were to be applied with a load that would give a displacement of 1 inch, provided the material were elastic and had a large enough cross-sectional area to withstand the displacements.

Expansion and contraction of a member would have a similar philosophy. If a member were attached from floor to ceiling, and the air temperature were to increase a certain amount, the member would want to increase in length, and this length could be, for example, one inch. The stress in the member would be comparable to the stress that would occur if the member were to be applied with a load that would give a displacement of 1 inch, provided the material were elastic and had a large enough cross-sectional area to withstand the displacements. This form of stress is designated as an expansion stress in ASME, Section III, Subsection NF.

The definition of an expansion stress as constraint of free-end displacement is exemplified in Article NF-1121(a) (CASE Exhibit 745):

"Rules for Supports.

"(a) The rules of Subsection NF provide requirements for new construction and include consideration of mechanical stresses and effects which result from the constraint of free-end displacements, designated as P_e in NF-3222.3 but not thermal or peak stresses." (Emphasis added.)

NF-3222.3 (CASE Exhibit 744, Pages 34-36) is entitled "Expansion Stress Intensity, P_e." Although this paragraph is under NF-3220 DESIGN OF PLATE AND SHELL TYPE SUPPORTS BY ANALYSIS (rather than under linear supports), what is important here is that this is the definition to be used according to NF-1121(a). NF-3222.3 states:

> "Expansion Stress Intensity, P_e. This stress intensity is the highest value of stress, neglecting local structural discontinuities, produced at any point across the thickness of a section by the loadings which result from restraint of free end displacement and the effect of anchor point motions..." (Emphasis added.)

Further, under NF-3200 DESIGN OF CLASS 1 COMPONENT SUPPORTS, NF-3210 GENERAL REQUIREMENTS, NF-3217 Classification of Stresses (CASE Exhibit 744, page 33), it is stated:

> "Table NF-3217-1 provides assistance in the determination of the category to which a stress should be assigned."

Table NF-3217-1 CLASSIFICATION OF STRESSES FOR SOME TYPICAL CASES (CASE Exhibit 744, page 34) (which is for plate and shell but for which the definitions are the same as for design of linear type supports by analysis regarding stresses) states:

> "Origin of Stress...Expansion[1]
>
> "/1/ Stresses which result from the constraint of free end displacement and the effect of differential support or restraint motions. Considers the effects of discontinuities but not local stress concentrations."

. . . and we are again referred to NF-3213:

> "...For definition of types of stress, see NF-3213." (See page 11 of this pleading.)

In addition, Article NF-3111 (CASE Exhibit 659B, page 29, Attachment to Mark Walsh direct testimony) states that:

> "The loadings as specified in the Design Specifications (NA-3250) that shall be taken into account in designing a component support

include, but are not limited to, the following:...

"(e) Restrained thermal expansion;...

"(g) Environmental loads such as wind and snow loads."

--(Emphasis added.)

This is consistent with Criterion 4 of 10 CFR Part 50, Appendix A. One of the environmental conditions involved with a LOCA is an increase in the ambient temperature. This temperature reaches 280^{o}F in two minutes after the postulated pipe break (see Footnote 3, page 8, of this pleading). The increase in air temperature results in an expansion stress; i.e., stress resulting from the restraint of free-end displacement, or thermal expansion stress.

Appendix F of ASME Section III (CASE Exhibit 746) has been misused by both Applicants and the NRC Staff.

The March 8, 1982, TUSI Memorandum to Pipe Support Engineering "Design Criteria File" (CASE Exhibit 659E, Attachment to Mark Walsh direct testimony) states, in part:

"1. It was agreed that ASME Section III, Subsection NF does not require that thermal expansion be considered in pipe support design." (Emphasis in the original.)

"2. It was stated that information received from Gabe Bove of Westinghouse, who is a member of the NF committee, verifies that thermal expansion in supports is not generally considered in the nuclear industry. Exceptions do exist, such as members spanning between walls, floor to ceiling, critical anchors, etc...

"7. Specification 2323-MS-46A mentions LOCA environmental conditions, but references sub-section NF for support design.

"8. It was agreed that thermal expansion in pipe support designs would only be considered in special cases based on engineering judgement. If any stress conditions are encountered, they will be further analyzed in detail before any design changes are proposed."

Although the March 8 Memo does not specifically mention Appendix F as being the basis for Applicants' conclusions regarding thermal expansion and LOCA, Applicants' witness Chang, in his 8/21/82 deposition (CASE Exhibit 677, pages 46, 47 and others) referenced Appendix F.

In the June 25, 1983, TUSI Memorandum from Applicants' witness Finneran to PSE "Design Criteria File" (CASE Exhibit 659G, Attachment to Mark Walsh direct testimony), which was for the purpose of expanding on the 3/8/82 and finalizing PSE's position regarding LOCA temperature considerations relative to pipe support designs, it is stated, in part:

"Item 8 of the previous letter indicated that thermal expansion in pipe support design needs to be considered in 'special cases'. It is PSE's position that the only 'special cases' that need to be considered are when members span between walls or from floor to ceiling. In all other cases local yielding will relieve these stresses to a point where they are not a problem.

"By copy of this letter G. Krishnan is requested to <u>not</u> run any STRUDL thermal analysis unless specifically requested to do so by the PSE Engineer."

In the NRC Staff's prefiled testimony prepared for the September 1982 hearings (Staff Exhibit 201), Mr. Tapia stated (Answer 8, page 5):

"'Rules for Evaluation of Faulted Conditions,' ASME Section III, Appendix F, excludes thermal stresses resulting from faulted conditions in the design procedures. This exclusion is based on the fact that the thermal stresses are relieved by ductile displacement. <u>The evaluations of plant faulted conditions are intended to demonstrate the structural capability of the system, to ensure operability of the piping</u>. The evaluation allows the material to be stressed above the yield point provided that sufficient ductility exists in the material to allow relaxation of constrained thermal expansion stresses prior to the material reaching failure strain." (Emphasis added.)

Similar reasoning is contained in the NRC Special Inspection Team's investigation report regarding the Walsh/Doyle allegations, Report 50-445/82-26, 50-446/82-14, bottom of page 17 continued on page 18, which states, in part:

> "The Special Inspection Team determined, from interviews with cognizant design engineers and from calculation reviews, that the Applicant had not considered LOCA thermal expansion effects on concrete inserts and bolts in the design of individual pipe supports and associated concrete anchors...This decision was based primarily on the ASME Code Section III, Appendix F, 'Rules for Evaluation of Faulted Conditions,' which does not require that differential thermal expansion stresses resulting from faulted conditions be included in the design procedure. This exclusion is based on the ASME Code rationale that these stresses occur once in the lifetime of the plant, are self-limiting in nature and are relieved by small deformations and displacements." (Emphasis added.)

It is not clear what the Special Inspection Team meant regarding "the ASME Code rationale that these stresses occur once in the lifetime of the plant, are self-limiting in nature and are relieved by small deformations and displacements," or what the basis for that statement is. Further, this is contrary to what is stated in Regulatory Guide 1.124 (see especially page 7 of this pleading: paragraph 1, last sentence; paragraph 4, last sentence; and last paragraph).

Further, Appendix F (CASE Exhibit 746, page 481) clearly states:

"F-1220. LIMITS OF DESIGN PROCEDURES.

> "(a) The Faulted Condition design procedures contained in F-1300 are provided for limiting the consequences of the specified event. They are intended (NA-1130) to assure that violation of the pressure retaining boundary will not occur in components or supports which are in compliance with these procedures. These procedures are not intended to assure the safe operability or reoperability of the system either during or following the postulated event." (Emphasis added.)

Section 3.2.2 of Applicants' FSAR (Applicants' Exhibit 3), beginning on page 3.2-7), SYSTEM QUALITY GROUP CLASSIFICATION, discusses the system quality group classifications and safety class definitions. This section states, in part:

"2. Safety Class Definitions...Supports that have a nuclear safety function shall be the same safety class as the components that they support..."

Table 17A-1 of the FSAR, a LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS, also includes listings by Safety Class.

In addition, Table 3.11B-1 of the FSAR lists EQUIPMENT REQUIRED TO FUNCTION DURING AND AFTER AN ACCIDENT.

The FSAR also addresses requirements for ASME Code Class components (FSAR page 3.9N-69):

"3.9N.3 ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

"The ASME Code Class components are built to accepted ASME Code, Section III requirements. For Code Class 1 components, very stringent requirements are imposed and are met. For Code Class 2 and 3 components, the requirements are less stringent but adequate, in accordance with the lower classification.

"See Section 3.9N.1 for more detailed discussions on ASME Code Class 1 components."

Included in FSAR Section 3.9N MECHANICAL SYSTEMS AND COMPONENTS are the following (page 3.9N-26):

"3.9N1.4.2 Analysis of the Reactor Coolant Loop and Supports

"The reactor coolant loop piping is evaluated in accordance with the criteria of ASME III, NB-3650 <u>and Appendix F</u>." (Emphasis added.)

. . . and (page 3.9N-50):

"3.9N.1.4.7 Stress Criteria for Class 1 Components and Component Supports

"<u>All Class 1 components and supports</u> are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code, Section III. The design analysis or test methods and associated stress or load allowable limits that

> will be used to evaluation of faulted condition are those that are defined in Appendix F of the ASME code with...supplementary option..." (Emphases added.)

The manner in which pipe supports are designed and constructed is also important as they pertain to safety-related active valves. Section 3.9N.3.2 of the FSAR, regarding Pump and Valve Operability Assurance (page 3.9N-72) details the extensive methods used to demonstrate the operability of active pumps and valves, including requirements for operability under the specified plant conditions and defining appropriate acceptance criteria to ensure that the program requirements are satisfied. It also states (page 3.9N-76):

> "Safety-related active valves must perform their mechanical motion in times of an accident. Assurance is supplied that these valves will operate during a seismic event. Qualification tests accompanied by analyses are conducted for all active valves..."

But Section 3.9N.3.4 Component Supports states (page 3.9N-82):

> "Active valves are, in general, supported only by the pipe attached to the valve. Exterior supports on the valve are generally not used." (Emphasis added.)

This means that the safety-related active valves are supported by pipe which is in turn supported by pipe supports whose operability must be assured.

Thus, Applicants (with the blessings of the NRC Staff) are designing and constructing vitally important component supports, whose operability must be assured, in accordance with Appendix F of the ASME Code, which does not (as stated in Appendix F itself) assure the safe operability or reoperability of the system either during or following the postulated event.

While it might be argued that it is permissible to assume that a certain item will be affected by a LOCA, and that therefore it is permissible to use Appendix F in the design and construction of that item, this ignores the second of the two ways in which it is necessary to consider each support in order to assure operability of the support under all conditions which they may experience. (See discussion on pages 8 and 9 of this pleading.) When we consider this same item in the second of the two ways (as though it were supporting the item which was not involved with the faulted load but receives the effects of the LOCA), we get an entirely different picture.

If we have a Class 1 pipe (which we'll call P for convenience) being supported by a Class 1 pipe support (which we'll call PS) which is involved in a LOCA, it may be permissible to use Appendix F in the design and construction of P and PS -- but only insofar as we have considered the first of the two ways necessary to assure operability. In this case, another Class 1 pipe (which we'll call P2) being supported by a Class 1 pipe support (which we'll call PS2) is required to remain operable and must therefore be designed to Level A or B allowables (design, normal and upset conditions), rather than to Level D allowables (faulted conditions) permissible under Appendix F.

But what if it is P2 and PS2 which are involved in the LOCA? In that case, it would be permissible to design P2 and PS2 to Level D allowables; but P and PS would have to be designed to Level A or B allowables.

What Applicants and the NRC Staff are in effect saying is that should there be a LOCA,and P and PS have been designed to Level D allowables, it is permissible to also have designed P2 and PS2 (and P3 and PS3, etc.) to Level D allowables. But this approach does not assure operability or reoperability. In effect, it increases the possibility that, should there be a LOCA, not only P and PS will be involved in it, but P2 and PS2 will then not be able to function due to the increased temperature resulting from the LOCA, which will in turn increase the possibility that the initial LOCA will have a domino effect leading to an additional LOCA from the failure of P2 and PS2 and other Class 1 items which have been designed to Level D allowables.

By using Appendix F, Applicants and the NRC Staff are in effect saying that if there is a LOCA, all the Class 1 items which have been designed to Level D allowables can be allowed to fail. Clearly this is contrary to the intent of all NRC regulations and regulatory guides. And CASE cannot believe that this was the intent of the ASME Code either. Finally, this approach flies in the face of all logic and common sense.

In order to assure operability and reoperability, and to mitigate the consequences of a LOCA, the most conservative approach would be to design all Class 1 items with faulted loads and to normal operating allowables (Level A or B allowables which would include consideration of the constraint of free-end displacement, or thermal expansion stresses). This would recognize the faulted load condition of a sudden increase in load due to a pipe break (similar to the effect of suddenly turning on a garden hose, for example), while at the same time assuring operability under design, normal, and upset conditions.

Realistically, however, in the real world the combination of faulted loads with constraint of free-end displacement (thermal expansion stress) and Level A or B allowables may not be practical. A more reasonable approach might be to make two analyses, giving consideration to the two ways in which it is necessary to consider each item in order to assure operability under all conditions which may be experienced.

The first would be to consider the load combination of constraint of free-end displacement (thermal expansion stress) with faulted loads, using Level D allowables. The second method of analysis would consider constraint of free-end displacement (thermal expansion stress) with normal upset loads and Level B allowables. Both analyses must be done with each item in order to assure operability and reoperability of the system during and following the postulated event. (It should be noted that this does not consider the effects placed on the Richmond Insert, which is not covered by ASME but by American Concrete Institute (ACI); this will be addressed briefly in the following and in more detail during the upcoming hearings.)

Class 2 and Class 3 items must also be designed to assure operability under all conditions which may be experienced. To this point, we have discussed only Class 1 items. However, the same requirements should be met for Class 2 and Class 3 as for Class 1, as stated in the ASME Code:

"NF-3300 DESIGN OF CLASS 2 AND CLASS MC COMPONENT SUPPORTS

"NF-3330 DESIGN OF LINEAR TYPE SUPPORTS BY ANALYSIS

"The design rules and stress limits which must be satisfied for the Design and Operating Conditions are as given in NF-3230"

(See CASE Exhibit 747, especially pages 44 and 37.)

"NF-3400 DESIGN OF CLASS 3 COMPONENT SUPPORTS

"The design of Class 3 component supports shall be in accordance with the requirements of NF-3300 using one of the design procedures indicated in Table NF-3132.1(b)-1 for Class 3 construction. The applicable table of allowable stresses for a given material to be used with a specific design procedure is stipulated in Table NF-2121-1."

(See CASE Exhibits 747, especially page 45; 744, especially page 31; and 748.)

It should also be noted that we have not attempted to address here the use of gang hangers (where more than one pipe is shared by the same support) at Comanche Peak. This is a common practice throughout the containment, and should also be properly considered.

The 11/20/81 Memorandum for All NRR Personnel from Harold R. Denton, Director, Office of Nuclear Reactor Regulation, NRC, Washington, regarding Standard Definitions for Commonly-Used Safety Classification Terms, further supports CASE's positions in this Brief. This Memorandum (which CASE received 4/18/83 during preparation of this Brief) offers guidance regarding the use of certain commonly-used safety classification terms. Mr. Denton states, in part (CASE Exhibit 749, page A-1):

"...I am endorsing and prescribing for use by all NRR personnel the standard definitions set forth in the enclosure to this letter... For the time being...the definitions in the enclosure should be considered 'standard' and should be applied consistently by all NRR personnel in all aspects of our safety review and licensing activities and should be appropriately reflected in our regulatory guidance documents..."

The enclosure to Mr. Denton's memorandum, DEFINITION OF TERMS, states in part (CASE Exhibit 749, enclosure page):

"Important to Safety

". Definition - From 10 CFR 50, Appendix A (General Design Criteria) - see first paragraph of 'Introduction.'

"'Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.'

". Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in important way to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation).

". Includes Safety-Grade (or Safety-Related) as a subset.

"Safety-Related

". Definition - From 10 CFR 100, Appendix A - see sections III.(c), VI.a.(1), and VI.b.(3).

"Those structure, systems, or components designed to remain functional for the SEE (also termed 'safety features') necessary to assure required safety functions, i.e.:

"(1) the integrity of the reactor coolant pressure boundary;

"(2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or

"(3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of this part.

". Subset of 'Important to Safety'

". Regulatory Guide 1.29 provides an LWR-generic, function-oriented listing of 'safety-related' structures, systems, and components needed to provide or perform required safety functions. Additional information (e.g., NSSS type, BOP design A-E, etc.) is needed to generate the complete listing of safety-related SSC's for any specific facility..."

--(Emphases in the original.)

Safety Class Definitions are contained in Applicants' FSAR (Applicants' Exhibit 3), Section 3.2.2, beginning on page 3.2-7. Based on those definitions and on all of the preceding, it is obvious that LOCA (including thermal expansion stresses, or free-end displacement) must be considered in the design of all equipment inside the containment at a minimum (including pipe supports) and must therefore be included as part of the design criteria for Comanche Peak.

Applicants have not considered certain necessary ingredients in their analyses. Although there will be more detailed discussion and cross-examination in this regard in the upcoming hearings, there are a few noteworthy examples of these omissions by Applicants which CASE would call to the attention of the Licensing Board.

In Applicants' supplemental direct testimony in the September 1982 hearings (Applicants' Exhibit 142F, Answer 1), Mr. Finneran states:

> "General Design Criterion 4 requires that structures important to safety be designed to accommodate the environmental effects of LOCA. This is a general requirement which does not specify the methods by which the criterion must be satisfied. Specific implementation of the criterion is further defined in the FSAR and project design specifications which specify ASME Section III, Subsection NF as the basis for the design of pipe supports.
>
> "ASME Section III, Subsection NF recognizes the self-limiting characteristics of thermal stresses and thus does not require their consideration in the design of linear pipe supports. Aside from the Code, we have shown that slippage of anchors will relieve thermal stresses in worst case situations during a LOCA without loss of function of the supports. As such, the requirements of General Design Criterion 4 have been satisfied."

--(Emphases added.)

CASE does not agree that Applicants "have shown that slippage of anchors will relieve thermal stresses in worst case situations during a LOCA without loss of function of the supports" or that "the requirements of General Design Criterion 4 have been satisfied." We will be addressing this in detail in the upcoming hearings, but the following should be noted as well.

It is implied that Mr. Finneran, in the preceding statements, is speaking of Applicants' Exhibit 142D (Attachment to Applicants Exhibit 142, Rebuttal Testimony regarding Walsh/Doyle allegations). Although there are three different types of configurations of supports included in 142D, Messrs. Walsh and Doyle were primarily addressing the use of Richmond inserts (Applicants' Case 2, 9th page, Exhibit 142D) and we will address ourselves to this.

In this example, this reaction is due to constraint of free-end displacement (or thermal expansion stress). However, this reaction does not consider: bending stresses in the bolts; mechanical loads; different configurations of pipe supports; operability of the piping system; or the method used for the allowable load on the bolts. Further, the analysis is based on assumed load displacements of a Richmond insert.

Regarding the bending stresses in the bolts, CASE agrees with that portion of the NRC Special Inspection Team's statements in Investigation Report 82-26/82-14 regarding the Walsh/Doyle allegations which states (Report, bottom of page 21 continued on top of page 22):

> "Calculation by the Special Inspection Team of the stresses resulting from the shear, tension, and bending moments for the five loading cases analyzed, indicates that bending stresses in the bolt for the worst-case conditions are 15 times larger than the stresses resulting from shear. Although bending in the bolt may result in reducing shear on the insert, <u>it imparts an additional bending stress in the bolt which has not been calculated</u>...While there have been questions about whether the bolting is governed by the ASME Code, the NRC staff believes that <u>the total stress (including the bending stress) in the bolts should be evaluated to assure that the value for allowable stress has not been exceeded</u>. The NRC staff requires that <u>this value shall not exceed the ASME Code allowable stress for bolting</u>." (Emphases added.)

Regarding the method used for the allowable load on the bolts, the NRC Special Inspection Team stated in Investigation Report 82/26, 82/14 (page 18, top of page):

> "Although the ASME Code is not directly applicable to the design of the concrete anchorages, the Applicant adopted the ASME Code philosophy in the design of the concrete inserts. This design approach is documented in Sections 3.8.3.3.3 and 3.8.4.3.3 of the FSAR, where it states, '. . . thermal loads are neglected when they are secondary and self-limiting in nature and when the material is ductile.'
>
> "With respect to the design of inserts such as Richmond inserts, the Special Inspection Team found that these components are not governed by the ASME Code nor by any other standard which the NRC has adopted as a regulatory requirement. Thus, the only applicable regulatory standards are the requirements of 10 CFR Part 50, Appendix A - General Design Criteria For Nuclear Power Plants, Criteria 1 and 2, which require that such components be capable of performing their intended design function which is to carry the imposed loads without failure."

<u>But</u> the ASME Code <u>addresses</u> this in ASME NF-1132.5 (CASE Exhibit 705):

> "<u>NF-1132.5 Nonintegral Support Connected to Building Structure</u>. The jurisdictional boundary between a building structure and a non-integral support shall be the surface of the building structure. Such means of mechanically attaching the support to the building structure shall fall within the jurisdiction of Sub-section NF."

Therefore, the bolt which is screwed into the Richmond insert (which is embedded in the concrete) is part of the pipe support and thus is governed by the ASME Code. The Richmond insert and the wall itself are governed by the American Concrete Institute (ACI) Code (specifically, ACI 349-80 "Code Requirements for Nuclear Safety Related Concrete Structures), as the document which is more directly applicable than any other presently available at this time. It is CASE's understanding that a Regulatory Guide is in preparation at this time to implement the requirements of Appendix B to ACI 349.

In his testimony in the June hearings, Applicants' witness Scheppele testified that concrete is a brittle material (Tr. 853):

> "If you can visualize something like concrete, which is a brittle material when it's subjected to tension it would tend to crack..."

This is also confirmed by the book THE TESTING AND INSPECTION OF ENGINEERING MATERIALS (CASE Exhibit 750, page 37). This book is part of the McGraw-Hill Civil Engineering Series. It states:

> "Ductility is that property of a material which enables it to be drawn out to a considerable extent before rupture and at the same time to sustain appreciable load. Mild steel is a ductile material. <u>A non-ductile material is said to be brittle</u>, i.e., it fractures with relatively little or no elongation. Cast iron and <u>concrete are brittle materials</u>..." (Emphases added.)

Therefore, concrete is a <u>non-ductile</u> material. Since the Richmond insert is embedded in the concrete, in shear loading the insert bears against the concrete, and in tension loading the insert is held by the concrete. Thus, what happens to the concrete has a major bearing on what happens to the insert; the concrete is a major factor in the performance of a Richmond insert.

In Applicants' Exhibit 142E, page 1, it states:

> "Prior to failure, four fine cracks emanated from the insert on the top of the block on both specimens."

In Applicants' Exhibit 142E, page 2, it states:

> "Six cracks emanated from the insert on the top of the slab and extended down on four side surfaces to the reinforcement."

Applicants' Exhibit 142D, pages 25, 26, and 27 state:

> "The concrete spalled in a 2" radius..."
> "The concrete spalled in a 3" radius..."
> "The concrete spalled in a 4" radius..."

As indicated in each of the preceding Exhibits, the concrete in all of these tests on the Richmond inserts was reinforced concrete[6].

The definition of spalling is "chipping or fragmenting, especially of stone" (Webster's dictionary).

All of the preceding examples from Applicants' 142D and 142E are further demonstrations that the Richmond insert itself, which is embedded in the concrete, must also be treated as non-ductile. Therefore, the statement by the Applicants in their FSAR (as discussed by the Special Inspection Team - see p. 25 of this pleading) that ". . . thermal loads are neglected when they are secondary and self-limiting in nature and when the material is ductile" (emphasis added) is not applicable to the concrete or to the Richmond inserts which are embedded in the concrete. Since one of the effects of a LOCA is an increase in air temperature (which would induce a thermal load), it should be considered in the design and use of Richmond inserts (along with other thermal loads).

The Special Inspection Team stated (Investigation Report 82/26, 82/14, page 20, second full paragraph):

> "The Applicant has stated that ACI 349-80, 'Code Requirements for Nuclear Safety Related Concrete Structures,' an industry standard not adopted by the NRC as a regulatory requirement, allows a factor of safety of two for concrete inserts. The Special Inspection Team found that the ACI standard specifies load factors and capacity reduction factors and requires consideration of the forces caused by thermal effects under accident conditions. In addition, the ACI standard requires a testing program far broader than that which has been carried out for the Richmond inserts. The Special Inspection Team cannot concur that the ACI standard allows a factor of safety of two to be used in the manner in which it has been used by the Applicant." (Emphases added.)

---

[6] It should be noted that this use of reinforced concrete is contrary to the testing requirements of the American Concrete Institute (ACI). This is logical since there is no way of being certain that the Richmond inserts will only be used in locations next to reinforcing steel.

As pointed out in ASME, Appendix XVII-2271.3 (CASE Exhibit 707):

> "XVII-2271.3 Provision for Expansion. Adequate provision shall be made for expansion and contraction appropriate to the function of the support structure."

By considering this Code requirement, the effects of LOCA could be lessened. One way in which this could be implemented would be by the us· of slotted connections at the point where the bolt screws into the Richmond insert, as suggested by Mr. Walsh in his testimony.

There are other aspects of this matter which will be discussed further during the next hearings.

On page 17 (bottom of page) of the Special Inspection Team's Investigation Report 82/26, 82/14, it is stated:

> "The Special Inspection Team determined, from interviews with cognizant design engineers and from calculation reviews, that the Applicant had not considered LOCA thermal expansion effects on concrete inserts and bolts in the design of individual pipe supports and associated concrete anchors." (Emphasis added.)

From the preceding discussions, it is now obvious that the NRC Staff should require that Applicants consider LOCA thermal expansion effects on concrete inserts, the bolts which screw into the inserts, and the steel used for the pipe support in the design of pipe supports and associated concrete anchors (including Richmond inserts).

In regard to the Applicants' claim that they have considered the slippage at the intersection of a Richmond insert and the tube steel connection in their prefiled supplemental testimony (see page 23 of this Brief), Applicants' Exhibit 142D (page 9) contains an analysis of a 6x6x3/8 inch tube anchored with 1-1/2 inch diameter Richmond inserts with threaded rods. The analysis considered only a temperature differential of 210°F. The results indicate a reaction of 21 kips bearing on the Richmond insert. In their analysis, the Applicants have made several engineering errors.

As demonstrated in the preceding, the Applicants' presumption that the threaded rod is not covered by ASME Subsection NF is incorrect; the threaded rod <u>is</u> covered by ASME. The threaded rods used at Comanche Peak are SA36 rods. Since SA36 rods are not listed in Table XVII-2461.1-1 of ASME, Appendix XVII (CASE Exhibit 752), and since they are similar to A307 (CASE Exhibit 753, from AISC Steel Manual), the allowable shear strength of the bolt is .3YS (yield strength). Also in the AISC Manual of Steel Construction (CASE Exhibit 753), the allowable shear strength of an A36 bolt is $.3F_y$ (yield strength). The bolt's yield strength at 300°F is 31.9 as shown in Code Case N71-10 (CASE Exhibit 751). This is the applicable Code Case, as indicated in CASE Exhibit 754, NRC Regulatory Guide 1.85, Materials Code Case Acceptability ASME Section III Division 1. Regulatory Guide 1.85 states that:

"C. REGULATORY POSITION

"1. The Section III ASME Code Cases...listed below...are acceptable to the NRC Staff for application in the construction of components for light-water-cooled nuclear power plants...within the limitations stated..."

It also indicates that other "Code Cases that are not listed herein are either not endorsed or will require supplementary provisions on an individual basis to attain endorsement status."

For a 1-1/2 inch diameter bolt, the allowable shear on the bolt, under Level B allowables, is $[\pi (1.5)^2/4]$ (.3)(31.9) = 16.9 kips. This is less than the allowable of 25 kips used by the Applicants. Regulatory Guide 1.124 (CASE Exhibit 743, Position 4) permits an increase of 50% on bolted connections when constraint of free-end displacement is considered in the analysis. This would increase the allowable of 16.9 kips to 25 kips. This 25 kips allowable is only applicable when an analysis of thermal expansion is included with the mechanical loads; otherwise, the allowable of 16.9 kips is the allowable.

The value of the Richmond insert the Applicants currently use is based on the Prestressed Concrete Institute (PCI) handbook, as stated in the NRC Staff's Investigation Report 82-26/82-14 (page 20, first full paragraph). This method is commonly called the ultimate strength concept. When using this form of analysis, loads are increased with load factors and the material is reduced with under-capacity factors.

Although CASE does not, in the interest of time and brevity, wish to go into detail at this time regarding how the Applicants have misused the formula from PCI, the loading for the Richmond insert should be pointed out. As shown in CASE Exhibit 755, ANSI/ACI 349-76, the load combinations given in equations 7 and 8 under 9.3--Required Strength include accident temperature effects, as well as the faulted load combination and a 15% increase from the OBE earthquake. Using this load combination does not increase the allowable strength of the Richmond insert. As can be seen from the two equations, the Applicants considered only one item out of eleven possible additional loads.

The Applicants stated that their example was a worst-case condition. The condition they analyzed was for a 6-foot long member. The NRC Special

Inspection Team stated that the worst-case condition which they evaluated was for an 11-foot long member that exists in the containment and is part of the feedwater system gang hanger with an overall span of approximately 30 feet (Investigation Report 82-26/82-14, page 18, last paragraph). Since the expansion load exhibits a linear relationship, the ratio of 11 to 6 will provide the load on the threaded rod and Richmond insert. The load is (11/6) 21 = 38.5 kips. This value far exceeds the allowable used by the Applicants for the bolt and the Richmond insert.

Also, neither of the worst cases analyzed by the Applicants and the NRC Special Inspection Team included the results of two intersecting members, such as can be seen in CASE Exhibit 659B, 13WW (Attachment to Jack Doyle deposition/testimony). Had the Applicants or the NRC Staff analyzed two members intersecting at the Richmond insert, there would be two shear loads of 21 kips each on the threaded rod and Richmond insert. These loads would have a resultant shear force of $\sqrt{21^2 + 21^2} = 30$ kips -- again above the established allowables used by the Applicants.

Also, in the analysis the displacement used by the Applicants for the Richmond insert is <u>estimated</u> since this 1-1/2" diameter bolt has <u>never been tested in shear</u>. This is confirmed by the Special Inspection Team (Investigation Report 82-26/82-14, page 18, last full sentence, continued on top of page 19):

> "...there are no deflection test data for 1 1/2-inch Richmond inserts in shear loading. For the reasons discussed below the Special Inspection Team concludes that additional test data is required for 1 1/2-inch Richmond inserts."

The values for displacement and load can decrease with an increase in concrete strength, for example as shown for the 1-1/4" diameter Hilti

Super Kwik-Bolt allowables (CASE Exhibit 756) going from 4,000 PSI concrete to 6,000 PSI concrete shows a decrease of 435 lbs. in strength. This decrease in strength, although not indicated, is also a decrease in the deflection of the bolt.

And finally, during the deposition of Kenneth Scheppele (Vice-President of Gibbs & Hill), Applicants witness Scheppele stated (CASE Exhibit 757):

> "The structural steel within that building, I would say has considered the thermal effects of a LOCA by several measures, either by means of -- and this is where we talk about major structures, I'm not talking about structures which are relatively short and which would be attached to concrete by means of bolts, things of this sort but I would say as far as major structures are concerned, these are elements which would be normally reviewed from the viewpoint of temperature expansion."

In reviewing the calculations of the wall-to-wall steam generator upper lateral restraint (referred to in Investigation Report 82-26/82-14, page 25, middle paragraph)(CASE Exhibit 758), it is clearly indicated that the calculations in regard to LOCA were done after Mr. Walsh had testified in the July hearings, and that LOCA was not previously considered in those calculations. In reviewing the calculations, the engineer has made two errors which should be noted. First is that he is using the yield strength of the material at room temperature, not at accident temperature; a similar mistake was made on the threaded rod which screws into the Richmond insert. The second mistake was that the engineer is using the main wall reinforcement for shear reinforcement. The ACI Code (318-71, paragraph 12.13) is explicit in stating that stirrups will be U-shaped or multiple U-shaped to carry the shear forces that the concrete cannot withstand. Stirrups are not used within the walls at Comanche Peak.

There are many other errors regarding these matters which will be discussed further during the next hearings.

Although we are still analyzing some of the documents received recently from the Applicants on discovery, there are a couple of specific examples which we will definitely be addressing further in the next hearings which are especially noteworthy.

One example is in further reference to the steam generator upper lateral restraint mentioned on the preceding page of this Brief. From our preliminary review, it appears that the Applicants used an incorrect method in determining the forec applied to walls A and B of the steam generator compartments. Using energy methods, it appears that the force exerted on the walls is three times larger than the value the Applicants obtained in their calculations (which the NRC Special Inspection Team reviewed in preparing their Investigation Report 82-26/82-14).

Another example is in regards to the floor-to-wall moment restraint also discussed in the middle paragraph on page 25 of Investigation Report 82-26/82-14. It appears that the method used by the Applicants is providing erroneous values for reactions and displacement of the attached pipe. One major flaw in their analysis of the stiffness of the base plates is that they did not consider the shear lugs underneath the base plates. The value they used in their analysis for translational stiffness at a Richmond insert was 83 kips per inch. If one were to consider the shear lugs acting with the Richmond inserts, the stiffness becomes approximately 49,000 kips per inch rather than 83 kips per inch as assumed by the Applicants (and reviewed by the NRC Special Inspection Team). The Applicants claim that slippage at the support points relieves the expansion stresses and that if one were to model in the stiffness of the support, the thermal expansion stresses during a LOCA would be relieved and that the supports could still perform their required function. However, it is obvious that with a stiffness of 49,000 kips per inch, those stresses aren't just going to disappear. This moment limiting restraint has displacements in excess of 2/10". These displacements are important since the support is adjacent to the penetration at the containment wall. Large displacements of the support at the point where the pipe is may cause the pipe to shear in the event of a LOCA.

As we have indicated, these are merely preliminary calculations and analyses which we will be addressing further later.

IN CONCLUSION

As demonstrated herein, there is no justification to be found in 10 CFR, NRC Regulatory Guides, other NRC regulations, industry codes, or elsewhere for the position of Applicants and the NRC Staff of not including LOCA in the design criteria for Comanche Peak. In fact, their approach practically assures that some items not involved in a LOCA will not remain operable during a LOCA or be reoperable following a LOCA and may even lead to other LOCA(s) due to failure of other items not involved in the initial LOCA.

It is most disturbing to CASE that not only the Applicants have taken this position, but the NRC Staff has also in its testimony and in its investigation report regarding the Walsh/Doyle allegations. From the discussions contained herein, it is obvious that both Applicants and the NRC Staff are not complying with the NRC's own regulations. CASE should not even have to be writing this Brief. The NRC Special Inspection Team _should_ have uncovered these problems (and the many other problems which will now have to be litigated in the upcoming hearings) during their investigation of the Walsh/Doyle allegations (just as the NRC Region IV inspectors and investigators _should_ have uncovered the many problems recently identified by the Construction Appraisal Team (CAT) in its recently released report). The NRC Special Inspection Team made no effort to contact Messrs. Walsh or Doyle regarding their concerns, but rather contacted _only_ the Applicants and representatives of their contractors, sub-contractors, and suppliers (see page 8 of Investigation Report 82-26/82-14, item 1. Persons Contacted).

Had the NRC Special Inspection Team made an effort to contact Messrs. Walsh or Doyle regarding their concerns, much of the information contained in this Brief would have already been known and acted upon, and there is a good possibility that the upcoming hearings would not even have been necessary. LOCA is only one of several major concerns and problems which will be addressed by CASE further in the upcoming hearings. When all the facts are presented, it will become obvious that the concerns of Messrs. Walsh and Doyle were well-founded, just as their concerns regarding the inclusion of LOCA in the design criteria for pipe supports were well-founded.

In conclusion, _LOCA conditions must be considered in the design_ criteria for pipe supports. Further, _each_ item important to safety should be analyzed _in two ways_: (1) as though it _were_ involved in the faulted load of a LOCA; and (2) as though it _were not_ involved in the faulted load of a LOCA but received the effects of the LOCA, including the constraint of free-end displacement (or thermal expansion stresses) resulting from the increased temperature due to the LOCA. Only by considering each item in these two ways can the operability and reoperability of the systems important to safety be assured under all conditions which they may experience. Only by the inclusion of LOCA conditions in the design criteria for pipe supports and other items important to safety can the public health and safety be protected (assuming the many other problems at Comanche Peak are also addressed and corrected prior to fuel-loading).

Further, and more importantly, these problems must be addressed and corrected not only on paper, but also in actual practice.

Respectfully submitted,

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

APPLICATION OF TEXAS UTILITIES GENERATING COMPANY, ET AL. FOR AN OPERATING LICENSE FOR COMANCHE PEAK STEAM ELECTRIC STATION UNITS #1 AND #2 (CPSES)

Docket Nos. 50-445 and 50-446

DOCKETED USNRC APR 25

CERTIFICATE OF SERVICE

By my signature below, I hereby certify that true and correct copies of CASE'S BRIEF REGARDING CONSIDERATION OF LOCA IN DESIGN CRITERIA FOR PIPE SUPPORTS have been sent to the names listed below this 20th day of April, 1983, by: Express Mail where indicated by * and First Class Mail elsewhere.

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