ATTACHMENT 2

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TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION REPORT ON THE FIRST
10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN:
TEXAS UTILITIES (TU) ELECTRIC COMPANY,
COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1,
DOCKET NUMBER 50-445

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Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION

EG&G Idaho

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ABSTRACT

This report presents the results of the evaluation of the *Comanche Peak Steam Electric Station (CPSES), Unit 1, First 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0, submitted October 15, 1990, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements that the Licensee has determined to be impractical. The *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during the Nuclear Regulatory Commission (NRC) review before granting an operating license. The requests for relief are evaluated in Section 3 of this report.

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Review of ISI for ASME Code Class 1, 2, and 3 Components

SUMMARY

The Licensee, Texas Utilities (TU) Electric Company, has prepared the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0, to meet the requirements of the 1986 Edition of the ASME Code Section XI. The first 10-year interval began August 13, 1990 and ends August 12, 2000.

The information in the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted October 15, 1990, was reviewed. Included in the review were the requests for relief from the ASME Code Section XI requirements that the Licensee has determined to be impractical. As a result of this review, a request for additional information (RAI) was prepared describing the information and/or clarification required from the Licensee in order to complete the review. The Licensee provided the requested information in the submittal dated September 13, 1991.

Based on the review of the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan*, Revision 0, the Licensee's response to the Nuclear Regulatory Commission's RAI, and the recommendations for granting relief from the ISI examinations that cannot be performed to the extent required by Section XI of the ASME Code, it is concluded that the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan*, Revision 0, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

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TECHNICAL EVALUATION REPORT ON THE FIRST 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN: TEXAS UTILITIES (TU) ELECTRIC COMPANY, COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1, DOCKET NUMBER 50-445

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1. INTRODUCTION

Throughout the service life of a water-cooled nuclear power facility, 10 CFR 50.55a(g)(4) (Reference 1) requires that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, Class 2, and Class 3 meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Reference 2), to the extent practical within the limitations of design, geometry, and materials of construction of the components. This section of the regulations also requires that inservice examinations of components and system pressure tests conducted during the initial 120-month inspection interval comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed therein. The components (including supports) may meet requirements set forth in subsequent editions and addenda of this Code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The Licensee, TU Electric Company, has prepared the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0 (Reference 3), to meet the requirements of the 1986 Edition of the ASME Code Section XI. The first 10-year interval began August 13, 1990 and ends August 12, 2000.

As required by 10 CFR 50.55a(g)(5), if the licensee determines that certain Code examination requirements are impractical and requests relief from them, the licensee shall submit information and justifications to the Nuclear Regulatory Commission (NRC) to support that determination.

Pursuant to 10 CFR 50.55a(g)(6), the NRC will evaluate the licensee's determination that Code requirements are impractical to implement. The NRC may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Alternatively, pursuant to 10 CFR 50.55a(a)(3), the NRC will evaluate the Licensee's determination that either (i) the proposed alternatives provide an acceptable level of quality and safety or (ii) Code compliance would result in hardship or unusual difficulty without a compensating increase in safety. Proposed alternatives may be used when authorized by the NRC.

The information in the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval ISI Program Plan*, Revision 0, submitted October 15, 1990, was reviewed, including the requests for relief from the ASME Code Section XI requirements that the Licensee has determined to be impractical. The review of the ISI Program Plan was performed using the Standard Review Plans of NUREG-0800 (Reference 4), Section 5.2.4, "Reactor Coolant Boundary Inservice Inspections and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components."

In a letter dated August 12, 1991 (Reference 5), the NRC requested additional information that was required in order to complete the review of the ISI Program Plan. The requested information was provided by the Licensee in the "Response to Request for Information Related to the Inservice Inspection Program Plan" dated September 13, 1991 (Reference 6).

As a result of telephone conversations with the Licensee on October 4, 1991, and January 23, 1992, the Licensee submitted further information in a letter dated January 24, 1992 (Reference 7), regarding the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval ISI Program Plan*.

The *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval ISI Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of

Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during the NRC's previous reviews.

The requests for relief are evaluated in Section 3 of this report. Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1986 Edition. Specific inservice test (IST) programs for pumps and valves are being evaluated in other reports.

2. EVALUATION OF INSERVICE INSPECTION PROGRAM PLAN

This evaluation consists of a review of the applicable program documents to determine whether or not they are in compliance with the Code requirements and any previous license conditions pertinent to ISI activities. This section describes the submittals reviewed and the results of the review.

2.1 Documents Evaluated

Review has been completed on the following submittals from the Licensee:

(a) *Comanche Peak Steam Electric Station (CPSES), Unit 1, First 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted October 15, 1990 (Reference 3).

(b) Letter, dated August 21, 1991 (Reference 8), containing interim change requests for the ISI Program Plan, a clarification regarding the weld marking system, and correction of several errors in the plan.

(c) Letter, dated September 13, 1991 (Reference 6), response to the NRC request for additional information dated August 12, 1991.

(d) Letter, dated October 16, 1991 (Reference 9), containing Request for Relief B-1.

(e) Letter, dated October 30, 1991 (Reference 10), containing additional information regarding Request for Relief B-1.

(f) Letter, dated January 24, 1992 (Reference 7), containing CPSES *Augmented ISI Plan*, CPSES/FSAR commitment to comply with NRC Regulatory Guide 1.150, Westinghouse Technical Bulletin NSD-TB-75-1 regarding the reactor coolant pump thermal barrier, and ISI boundary diagrams for the component cooling water system.

2.2 Compliance with Code Requirements

2.2.1 Compliance with Applicable Code Editions

The Inservice Inspection Program Plan shall be based on the Code editions defined in 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b). Based on the starting date of August 13, 1990, the Code applicable to the first ISI interval is the 1986 Edition. As stated in Section 1 of this report, the Licensee has prepared the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year ISI Program* to meet the requirements of 1986 Edition of the Code.

2.2.2 Acceptability of the Examination Sample

Inservice volumetric, surface, and visual examinations shall be performed on ASME Code Class 1, 2, and 3 components and their supports using sampling schedules described in Section XI of the ASME Code and 10 CFR 50.55a(b). Sample size and weld selection have been implemented in accordance with the Code and 10 CFR 50.55a(b) and appear to be correct.

2.2.3 Exemption Criteria

The criteria used to exempt components from examination shall be consistent with Paragraphs IWB-1220, IWC-1220, IWC-1230, IWD-1220, and 10 CFR 50.55a(b). The exemption criteria have been applied by the Licensee in accordance with the Code, as discussed in the ISI Program Plan, and appear to be correct.

2.2.4 Augmented Examination Commitments

In addition to the requirements as specified in Section XI of the ASME Code, the Licensee has committed to perform augmented examinations according to the following documents:

(a) NRC Regulatory Guide 1.14, *Reactor Coolant Pump Motor Flywheel Integrity*, Revision 1 (Reference 11).

(b) NUREG-0797, *Safety Evaluation Report Related to the Operation Of Comanche Peak Steam Electric Station, Units 1 and 2*, Supplemental Safety Evaluation Report 12, regarding safety injection pump shrouds (Reference 12).

(c) NRC Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors*, (Reference 13) and CPSES-9006199.

(d) MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment", (Reference 14) and FSAR 6.C.8.

(e) NRC Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*, (Reference 15).

2.3 Conclusions

Based on the review of the documents listed above, it is concluded that the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval ISI Program Plan*, Revision 0, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

3. EVALUATION OF RELIEF REQUESTS

The requests for relief from the ASME Code requirements that the Licensee has determined to be impractical for the first 10-year inspection interval are evaluated in the following sections.

3.1 Class 1 Components (No relief requests)

3.2 Class 2 Components

3.2.1 Pressure Vessels (No relief requests)

3.2.2 Piping

3.2.2.1 Request for Relief No. C-1, Examination Category C-C, Item C3.20, Integrally Welded Attachments to Containment Spray Piping

Code Requirement: Section XI, Table IWC-2500-1, Examination Category C-C, Item C3.20, requires a surface examination as defined by Figure IWC-2500-5.

Licensee's Code Relief Request: Relief is requested from performing 100% of the Code-required surface examination on the eight welded lugs of support number CT-1-024-003-S22R on containment spray line number 16-CT-024-301R-2.

Licensee's Basis for requesting relief: The Licensee states that the eight welded lugs were examined with liquid penetrant to the maximum extent practical without removing an adjacent welded clamp. The clamp prohibits examination of the 1/2 inch examination zone on one side of each of the lugs.

Licensee's Proposed Alternative Examination: None. The Code-required surface examination was performed to the maximum extent practical.

Evaluation: Table IWC-2500-1, Examination Category C-C, Item C3.20 requires a surface examination per Figure IWC-2500-5. The Licensee states that a best effort surface examination was performed on the welded lugs, but that a 1/2 inch wide portion of each lug was obstructed by an adjacent welded clamp. As shown in the drawing attached to the Licensee's relief request, the clamp obstructs a portion of the required examination area, making the surface examination impractical to perform to the extent required by the Code. In order to perform the Code-required examination, the lugs would have to be redesigned and replaced. Imposition of the requirement on the Licensee would cause a burden that would not be compensated by an increase in safety above that provided by the limited examination.

Conclusion: Based on the above, it is concluded that the Code required surface examination is impractical to perform to the full extent required by the Code at Comanche Peak, Unit 1, and that public health and safety will not be endangered by allowing the limited examination in lieu of the Code requirement. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted.

3.2.3 Pumps (No relief requests)

3.2.4 Valves (No relief requests)

3.2.5 General (No relief requests)

3.3 Class 3 Components (No relief requests)

3.4 Pressure Tests

3.4.1 Class 1 System Pressure Tests

3.4.1.1 Request for Relief No. B-1, Examination Category B-P, Item 15.50, Reactor Coolant Piping System Leakage Test

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-P, Item B15.50, requires a VT-2 visual examination during the System Leakage Test. As required by IWB-5221, the System Leakage Test is performed at or above the nominal operating pressure associated with 100% rated reactor pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required VT-2 visual examination during the System Leakage Test for reactor coolant piping between the reactor pressure vessel (RPV) and concrete barrier wall, and within the concrete wall penetration.

Licensee's Basis for requesting relief: The Licensee states that the reactor vessel at Comanche Peak, Unit 1, is surrounded by a concrete barrier wall. The length of the sleeve in the wall where the hotleg (reactor vessel outlet) of the main coolant loop penetrates, measures approximately 74 inches, and approximately 132 inches where the coldleg (reactor vessel inlet) penetrates. The area where the nozzles penetrate the wall is inaccessible for direct visual examination due to the limited separation between the coolant pipe and the penetration sleeve. The separation between the insulation and the sleeve is 3 inches on the hotleg, and 2 inches on the coldleg.

The reactor nozzle areas are accessible from the reactor cavity through access ports over each of the eight nozzles. Entry into the nozzle areas renders approximately 30 inches of main coolant piping accessible for examination.

Since the System Leakage Tests are performed at elevated temperatures, access to the nozzle penetrations represents a personnel hazard due to the extreme heat within the confined space. Additionally, the nozzle penetrations are anticipated to represent significant radiation exposure areas with an estimated dose accumulation of approximately 12 man-REM.

The efforts associated with the removal of the nozzle access covers, the personnel hazards, and the radiation dose accumulations associated with performing the direct visual examination of this limited area do not result in a corresponding increase in quality and safety.

Licensee's Proposed Alternative Examination: None. The area in which the main coolant piping exits the concrete barrier wall will be examined for steam or other signs of leakage. "The source for any steam present shall be examined for evidence of leakage, including any residue or discoloration." Additionally, existing plant monitoring systems provide further assurance that any significant leakage will be detected.

Evaluation: Relief is requested from the Code-required VT-2 visual examination during the System Leakage Test for the reactor coolant piping between the reactor vessel and the outside of the concrete barrier that surrounds the vessel.

Paragraph IWA-5241 addresses visual examinations for noninsulated components as follows:

> (a) The visual examination VT-2 shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage.
>
> (b) For components whose external surfaces are inaccessible for direct visual examination VT-2, only the examination of surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage shall be required.

The portion of piping that penetrates the concrete wall is inaccessible due to limited clearance (2-3 inches) between the piping and the concrete. As stated in paragraph IWA-524 for components that are not accessible for direct visual examination, only examination of surrounding areas for evidence of leakage is required. Since direct visual examination is not possible, relief is not required for the reactor coolant piping located within the concrete wall penetration, provided that the surrounding areas are examined for leakage as required by the Code.

For the portion of piping between the RPV and the concrete wall, the Code required visual examination is impractical to perform due to the high radiation levels and the extreme heat generated during the System Leakage Test. In order to perform the examination to the extent required by the Code, the reactor coolant system would require extensive design modifications. Imposition of this Code requirement on the Licensee would cause a burden that would not be compensated by an increase in safety above that provided by the limited examination.

Conclusion: It is concluded that the VT-2 visual examination is impractical to perform at Comanche Peak, Unit 1, to the extent required by the Code. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted for the portions of piping between the reactor vessel and the concrete barrier wall. For the portions of piping within the sleeve penetrating the concrete barrier wall, it is concluded that relief is not required, provided that the surrounding areas are examined for evidence of leakage.

3.4.2 Class 2 System Pressure Tests (No relief requests)

3.4.3 Class 3 System Pressure Tests

3.4.3.1 Request for Relief No. D-1, Paragraph IWD-5223, Hydrostatic Pressure Testing of Reactor Coolant Pump (RCP) Thermal Barrier Heat Exchangers

Code Requirement: Section XI, paragraph IWD-5223(a), requires a system hydrostatic test pressure of at least 1.25 times the system pressure (P_{sv}) for systems with design temperature above 200°F.

Licensee's Code Relief Request: Relief is requested from the Code-required hydrostatic test pressures of paragraph IWD-5223(a) for reactor coolant pump (RCP) thermal barrier heat exchangers.

Licensee's Basis for requesting relief: The Licensee states that the P_{sv} for the portion of the component cooling water system containing the RCP thermal barrier heat exchangers is 2485 psi with a design temperature of 650°F. This yields a test pressure of 3106.25 psi. To attempt to pressurize the tube side of the RCP thermal barrier heat exchangers would potentially damage these components. The manufacturer (Westinghouse) has issued a technical bulletin advising that the maximum allowable field hydrostatic pressure is 225 psi for the component cooling water side of these RCP thermal barrier heat exchangers. These heat exchangers are designed for high differential pressures in the direction opposite from that imposed by a component cooling water hydrostatic test (i.e. from the reactor coolant system). To meet the conditions that could exist in the event of a heat exchanger leak inside of the RCP, the external connections and adjacent piping are designed for 2500 psi internal pressure.

Licensee's Proposed Alternative Examination: The portion of the component cooling water system that constitutes the tube side of the RCP thermal barrier heat exchangers shall be hydrostatically tested along with the portions other than those sections designed for 2500 psi. The design pressure of these portions is 150 psi and the test pressure shall be the required 1.25 times P_{sv}.

Evaluation: Paragraph IWD-5223(a) requires test pressures of 1.25 P_{sv} for systems with design temperatures above 200°F. The P_{sv} for the portion containing the RCP thermal barrier heat exchangers is 2485 psi with a design temperature of 650°F, yielding a test pressure of 3106.25 psi. Westinghouse Technical Bulletin NSD-TB-75-1 (included in Reference 7) advises that the maximum internal hydrostatic test pressure for the thermal barrier heat exchanger is 225 psi. The Licensee's proposed alternative is to perform the hydrostatic pressure tests of the subject heat exchangers at 1.25 times P_{sv} (which has not been specified).

Pressurizing the tube side of the RCP thermal barrier heat exchanger above the 225 psi maximum could potentially damage these components. Therefore, the Code requirement is impractical. In order to perform the examination to the extent required by the Code, the RCP would require extensive design modifications. Imposition of this Code requirement on the Licensee would cause a burden that would not be compensated by an increase in safety above that provided by the Licensee's proposed alternative.

Conclusion: It is concluded that hydrostatic pressure test is impractical to perform at Comanche Peak, Unit 1, at the test pressure required by the Code. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted.

3.4.4 General (No relief requests)

3.5 General (No relief requests)

4. CONCLUSION

Pursuant to 10 CFR 50.55a(g)(6), it has been determined that certain inservice examinations cannot be performed to the extent required by Section XI of the ASME Code. For the Relief Requests C-1 and D-1, the Licensee has demonstrated that specific Section XI requirements are impractical. For Request for Relief B-1, it is concluded that relief may be granted in part, and that relief is not required for the remainder of the request.

This technical evaluation has not identified any practical method by which the Licensee can meet all the specific inservice inspection requirements of Section XI of the ASME Code for the existing Comanche Peak Steam Electric Station, Unit 1, facility. Compliance with all the exact Section XI required inspections would necessitate redesign of a significant number of plant systems, sufficient replacement components to be obtained, installation of the new components, and a baseline examination of these components. Even after the redesign efforts, complete compliance with the Section XI examination requirements probably could not be achieved. Therefore, it is concluded that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55a(g)(6), relief is allowed from the requirements that are impractical to implement. Relief may be granted only if the relief will not endanger life, property, or the common defense and security and is otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

The Licensee should continue to monitor the development of new or improved examination techniques. As improvements in these areas are achieved, the Licensee should incorporate these techniques into the ISI program plan examination requirements.

Based on the review of the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan*, Revision 0, the Licensee's response to the NRC's Request for Additional Information, and the recommendations for granting relief from the ISI examination requirements that have been determined to be impractical, it is concluded that the *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection*

Program Plan, Revision 0, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

5. REFERENCES

1. Code of Federal Regulations, Title 10, Part 50.

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1:
   1985 Edition

3. *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan*, Revision 0, dated October 15, 1990.

4. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, Section 5.2.4, "Reactor Coolant Boundary Inservice Inspection and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components," July 1981.

5. Letter, dated August 12, 1991, T. A. Bergman (NRC) to W. J. Cahill Jr. (TU Electric Company), containing request for additional information on the First 10-Year Interval ISI Program Plan.

6. Letter, dated September 13, 1991, W. H. Cahill (TU Electric Company) to Document Control Desk (NRC), containing response to NRC request for additional information.

7. Letter, dated January 24, 1992, W. H. Cahill (TU Electric Company) to Document Control Desk (NRC), containing additional information regarding the First 10-Year Interval ISI Program Plan.

8. Submittal, dated August 21, 1991, TU Electric to Document Control Desk (NRC), containing interim change requests for the ISI Program Plan, a clarification regarding the weld marking system, and correction of several errors in the plan.

9. Letter, dated October 16, 1991, W. J. Cahill (TU Electric Company) to Document Control Desk (NRC), containing Relief Request B-1.

10. Letter, dated October 30, 1991, W. J. Cahill (TU Electric Company) to Document Control Desk (NRC), containing additional information regarding Relief Request B-1.

11. NRC Regulatory Guide 1.14, *Reactor Coolant Pump Motor Flywheel Integrity*, Revision 1, August 1975.

12. NUREG-0797, *Safety Evaluation Report Related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2*, Supplemental Safety Evaluation Report 12, July 31, 1981.

13. NRC Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors*, July 26, 1988.

14. Generic Letter 87-11, dated June 19, 1987, containing Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 2, June 1987.

15. NRC Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*, Revision 1, February 1983.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report presents the results of the evaluation of the *Comanche Peak Steam Electric Station (CPSES), Unit 1, First 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0, submitted October 15, 1990, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements that the Licensee has determined to be impractical. The *Comanche Peak Steam Electric Station, Unit 1, First 10-Year Interval Inservice Inspection Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during the Nuclear Regulatory Commission (NRC) review before granting an operating license. The requests for relief are evaluated in Section 3 of this report.

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