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#10CFR50, Appendix K

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January 25, 2005

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
SUBMITTAL OF TXU POWER'S APPLICATION OF NON-LOCA TRANSIENT ANALYSIS METHODOLOGIES TO A FEED RING STEAM GENERATOR DESIGN – TOPICAL REPORT #ERX-04-005, REVISION 0

Gentlemen:

As an enclosure to this letter, TXU Generation Company LP (TXU Power) submits Revision 0 of the CPSES Topical Report ERX-04-005; "Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design." This topical report contains those applications of current NRC approved methodologies that will be necessary to support anticipated licensee submittals related to the replacement of the CPSES Unit 1 steam generators.

This supplement is not intended to replace the methodologies currently used for non-LOCA accident analyses on CPSES Unit 1 and 2, but will provide those necessary additions to reflect the physical differences between the current preheater steam generator design and the feed ring design of the replacement steam generators.

This topical report supplement contains no proprietary information.

A001

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2. Should you have any questions, please contact Bob Kidwell at (254) 897-5310.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
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RJK
Enclosure

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Enclosure to TXX-05022

CPSES TOPICAL REPORT
ERX-04-005, Revision 0

APPLICATION OF TXU POWER'S NON-LOCA TRANSIENT ANALYSIS METHODOLOGIES TO A FEED RING STEAM GENERATOR DESIGN

Dated January, 2005

ERX-04-005

TXU

Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design

January, 2005

REACTOR ENGINEERING

TXU POWER
COMANCHE PEAK STEAM ELECTRIC STATION

Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design

ERX - 04 - 005
REVISION 0

January 2005

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TABLE OF CONTENTS

COVER SHEET 1

ACKNOWLEDGEMENTS 2

DISCLAIMER 3

TABLE OF CONTENTS 4

LIST OF TABLES 5

LIST OF FIGURES 6

CHAPTER 1 - INTRODUCTION 7

1.1 Purpose 7

1.2 Background and History 8

1.3 Report Layout 10

CHAPTER 2 - ORIGINAL AND REPLACEMENT STEAM GENERATOR DESIGN FEATURES 11

2.1 Original Steam Generator Design Overview 11

2.2 Replacement Steam Generator Design Overview 12

2.3 Comparisons of Original and Replacement Steam Generators 13

CHAPTER 3 - SUMMARY OF EFFECTS OF THE REPLACEMENT STEAM GENERATOR ON PREVIOUSLY APPROVED METHODOLOGIES 19

3.1 CPSES Accident Analysis Methodology Overview 19

3.2 Initial Conditions and System Performance Characteristics 21

3.3 Evaluation of Effects on Specific Analysis Methods 21

CHAPTER 4 - Δ76 SPECIFIC STEAM GENERATOR ACCIDENT ANALYSIS METHODS 34

4.1 Δ76 Steam Generator Model Description and Qualification 34

4.2 Δ76 Steam Generator Water Level Model 36

4.3 Feedwater System Pipe Break Analysis Methodology for a Feed ring Steam Generator Design 37

CHAPTER 5 - SUMMARY AND CONCLUSIONS 53

CHAPTER 6 – REFERENCES 54

LIST OF TABLES

Table 1 – Comparison of D-4 and Δ76 Design Features 15
Table 2 – Feedwater Line Break Analysis – Sequence of Events 45

LIST OF FIGURES

Figure 1 Westinghouse D-4 Steam Generator Outline 16

Figure 2 Westinghouse Δ76 Steam Generator Outline 17

Figure 3 Δ76 Steam Generator Feed Ring 18

Figure 4 Feedline Break Analysis – Break Flow Rate 46

Figure 5 Feedline Break Analysis – Break Flow Quality 46

Figure 6 Feedline Break Analysis – Affected and Intact Steam Generator Pressures 47

Figure 7 Feedline Break Analysis – RCS Hot and Cold Leg Temperatures and Saturation Temperatures 47

Figure 8 Feedline Break Analysis – Affected Steam Generator Secondary-Side Inventory 48

Figure 9 Feedline Break Analysis – Pressurizer Pressure 48

Figure 10 Feedline Break Analysis - Δ76 and D-4 Comparisons – Break Flow Rate 49

Figure 11 Feedline Break Analysis - Δ76 and D-4 Comparisons – Break Flow Quality 49

Figure 12 Feedline Break Analysis - Δ76 and D-4 Comparisons – Affected and Intact SG Pressures 50

Figure 13 Feedline Break Analysis - Δ76 and D-4 Comparisons – Cold and Hot Leg Temperatures in the Affected Loop 51

Figure 14 Feedline Break Analysis - Δ76 and D-4 Comparisons – Cold and Hot Leg Temperatures in the Intact Loop 51

Figure 15 Feedline Break Analysis - Δ76 and D-4 Comparisons – Affected SG Secondary-Side Mass 52

Figure 16 Feedline Break Analysis - Δ76 and D-4 Comparisons – Pressurizer Pressure 52

CHAPTER 1 - INTRODUCTION

1.1 Purpose

Analyses of postulated transients and accidents are presented in Chapter 15 of the updated Final Safety Analysis Report (FSAR) of the Comanche Peak Steam Electric Station (CPSES). The methodologies used to perform these analyses were developed by CPSES accident analysis engineers and approved by the Nuclear Regulatory Commission for this use. These methodologies were based on the plant configuration at the time the methodologies were developed, which included steam generator designs with integral feedwater preheaters.

In the CPSES preheat steam generator designs (designated as the Westinghouse D-4 for Unit 1 and D-5 for Unit 2), approximately 90% of the main feedwater is injected directly into the cold leg side of the steam generator tube bundle. Baffles direct this main feedwater across the cold leg tube bundle five times before it exits the preheater region and is allowed to mix with the recirculating fluid and continue to flow through the tube bundle. The remainder of the main feedwater flow is injected above the tube bundle through the auxiliary feedwater nozzle. In CPSES Unit 1, the original preheat steam generators are to be replaced with new steam generators which do not contain integral preheaters. The primary difference in the replacement steam generators, designated as the Westinghouse Δ76 design, is the use of feed rings to distribute the main feedwater above the tube bundle in the upper downcomer regions of the steam generators where it mixes with the entirety of the recirculating fluid before entering the tube bundle region.

The purpose of this report is to describe the effects of the Δ76 feed ring steam generators on the non-LOCA transient and accident analysis methodologies currently used for both CPSES Unit 1 and Unit 2. Significant changes to the applications of those methodologies, made necessary by the replacement of the original D-4 preheat steam generator design in Unit 1, are also described. This report is a supplement to the current methodologies which will continue to be used to support both CPSES units.

1.2 Background and History

The CPSES non-LOCA transient and accident analyses, presented in Chapter 15 of the FSAR, are performed using the following methodologies:

- RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology, " June 1994.
- RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation," July 1992.
- RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," September 1993.
- RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," October 1993.
- RXE-91-002-A, "Reactivity Anomaly Events Methodology," October 1993.
- TXX-88306, "Steam Generator Tube Rupture Analysis," March 15, 1988.
- RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.
- RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994.

These methodologies were developed by CPSES accident analysis engineers and approved by the NRC in the early 1990s. The scope of the methodology report RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," originally excluded the analysis methods for the steamline break accidents, the reactivity and power distribution anomaly transients and accidents,

and the steam generator tube rupture accident. Through the Request for Additional Information (RAI) process, however, the scope of the report came to encompass all non-LOCA transient and accident methodologies.

At the time of this writing, these methodology topical reports have been used to support the operation of CPSES Units 1 and 2 for over 10 years, including fourteen reload core configurations. The CPSES accident analysis engineers have developed extensive experience in the use of the tools, models and methodologies. As the design inputs representing the plants' operating and performance characteristics have changed, the models have been updated in accordance with the approved analytical methodologies. These methodologies have been evaluated and determined to remain applicable to the modified plant operating and performance characteristics.

The Δ76 replacement steam generator design, however, is significantly different from the preheater steam generator designs used in the development of the CPSES accident analysis methodologies. The potential for these design differences to affect the CPSES accident analysis methodologies must be considered. The assessment of the continued applicability of these methodologies and any modifications to the methodologies to accommodate the replacement steam generator design is the subject of this report.

This report is a supplement to the previously approved methodology topical reports. The previously approved methodology topical reports will continue to be applied to both CPSES units. The extension of the previously approved methodologies, as described in this report, will only be applied to feed ring steam generator designs.

1.3 Report Layout

The design features of the Δ76 replacement steam generator are described in Chapter 2 of this report. Comparisons with the original D-4 steam generator design are also presented. In Chapter 3, the effects of the Δ76 replacement steam generator on the approved analysis method for each of the transients and accidents presented in Chapter 15 of the CPSES FSAR are addressed in qualitative terms. Those accident-specific methodologies that require revision are addressed in Chapter 4. A summary and conclusions are presented in Chapter 5.

CHAPTER 2 - ORIGINAL AND REPLACEMENT STEAM GENERATOR DESIGN FEATURES

2.1 Original Steam Generator Design Overview

The original steam generator design used in CPSES Unit 1 is a Westinghouse D-4 design (see Figure 1). This steam generator design includes an integral preheater, where approximately 90% of the total main feedwater flow is injected directly into the cold leg side of the tube bundle. This preheater area is physically separated from the bulk of the recirculating fluid within the steam generator. Baffles direct the main feedwater across the tube bundle five times before it exits the preheater region and is allowed to mix with the recirculating fluid and continue to flow through the tube bundle. The remainder of the main feedwater flow is injected through the auxiliary feedwater nozzle where it mixes with the recirculating fluid and flows down to the tube bundle entrance. The use of the auxiliary feedwater nozzle for the main feedwater flow necessitates a connection between the Auxiliary Feedwater System and the Main Feedwater System. Thus, a significant portion of the auxiliary feedwater line is filled with relatively hot fluid from the Main Feedwater System that must be purged before the colder auxiliary feedwater fluid can enter the steam generator.

The D-4 steam generator design incorporates integral flow restrictors in the main feedwater nozzle and the steam nozzle. There are 4578 U-tubes, with a total heat transfer area of 48,300 ft^2. The tubes are fabricated from Alloy 600 Inconel. The outer diameter is 0.75 in., and they are arranged in a square lattice with a pitch of 1.0625 in. The volume of the shell side of the steam generator is approximately 5954 ft^3.

The steam generator water level instrumentation has a nominal span of 233 in. The lower tap is located in the annular downcomer region near the top of the U-tubes. The upper tap is located above the mid-deck plate (above the outlet of the primary separators). The nominal water level is 66.5% span.

2.2 Replacement Steam Generator Design Overview

The replacement steam generator design to be used in CPSES Unit 1 is a Westinghouse Δ76 design (see Figure 2). This steam generator design includes a feed ring (see Figure 3) through which 100% of the main feedwater is distributed throughout the recirculating fluid. Thirty-six spray nozzles, each comprised of 156 holes, one quarter of an inch in diameter, distribute the main feedwater into the upper downcomer region of the steam generator. These spray nozzles also act as strainers for any loose parts contained in the main feedwater flow. The Auxiliary Feedwater System and the Main Feedwater System are completely separated; only relatively cold auxiliary feedwater is contained in the auxiliary feedwater piping.

The Δ76 steam generator design incorporates integral flow restrictors in the steam nozzle only. There are 5532 U-tubes, with a total heat transfer area of 76,000 ft^2. The tubes are fabricated from Alloy 690 Inconel. The outer diameter of the U-tube is 0.75 in., and they are arranged in a triangular lattice with a pitch of 1.03 in. The volume of the shell side of the steam generator is approximately 5330 ft^3.

The steam generator water level instrumentation has a nominal span of 251 in. The lower tap is located above the annular downcomer region well below the top of the U-tubes. The upper tap is located above the mid-deck plate (above the outlet of the primary separators). The nominal water level is 67% span.

The primary and secondary steam separators, as well as the steam nozzles with their integral flow restrictors, are similar to the original D-4 steam generator design.

2.3 Comparisons of Original and Replacement Steam Generators

Comparisons of several design and operating characteristics important to the transient and accident analyses are presented in Table 2-1. A few items of interest, using the D-4 as the base case:

- The 35% larger tube-side volume results in an 11% increase in the overall Reactor Coolant System (RCS) volume.

- The shell-side mass is approximately 6% larger.

- The higher circulation ratio is indicative of lower fluid qualities in the boiling regions, which lead to less shrink in the indicated steam generator water level following a reactor trip.

- The 60% larger heat transfer area allows either the RCS average temperature to be reduced, if the steam pressure is held constant, or the steam pressure to be increased if the RCS average temperature is held constant.

Based on a comparison of the design features of the D-4 and Δ76 steam generator designs, it is possible to make a few general observations concerning the effects of the replacement steam generator design on the expected CPSES response to postulated transients or accidents. For example, even with a larger heat transfer area, the RCS will respond more slowly to transients induced through changes in the main feedwater flow rate or temperature because the main feedwater first mixes with the entire recirculating fluid in the upper downcomer regions before coming into contact with the tube bundle.

The increase in the RCS volume results in an increase in the fluid mass and heat capacity. For cooldown events induced by the secondary plant, there is a greater amount of stored energy that must be removed. If the magnitude of the secondary side transient is constant, the resulting RCS cooldown will be slower, even though the same endpoint will likely be reached. For heatup events induced by the secondary plant, the total heat

capacity of the RCS is greater, resulting in a slower heatup rate; however, the increased mass also results in greater volumetric expansion as the coolant heats up, assuming the endpoints are the same. The slower RCS heatup rate provides additional time for the Reactor Protection System and overpressure protection systems to operate, such that any transient-induced overshoots are minimized.

Finally, because of the separation between the Main Feedwater System and the Auxiliary Feedwater System, the time following auxiliary feedwater actuation required to purge the relatively hot main feedwater from auxiliary feedwater pipes is essentially eliminated. This difference tends to minimize the duration of the RCS heatup following a loss of the main feedwater flow.

Table 1 – Comparison of D-4 and Δ76 Design Features

Parameter	Current Steam Generator (D-4)	Replacement Steam Generator (Δ76)
Number of Tubes	4578	5532
Tube Outer Diameter, in	0.75	0.75
Tube Wall Thickness, in	0.043	0.043
Pitch, in	1.0625, square	1.03, triangular
Tube Material	Inconel 600	Alloy 690 Inconel
Total Secondary Side Heat Transfer Area, ft^2	48,300	76,000
Secondary Side Volume, ft^3	5954	5329
Primary Side Volume, ft^3	967	1303
Nominal Circulation Ratio at full power	2.44	4.10
Narrow Range Instrument Span, in	233	251
Nominal Water Level at power (% narrow range span)	66.5	67.0
Nominal Secondary Side Mass at 100% power, 0% steam generator tube plugging, lbm (approximate)	105,000	112,000

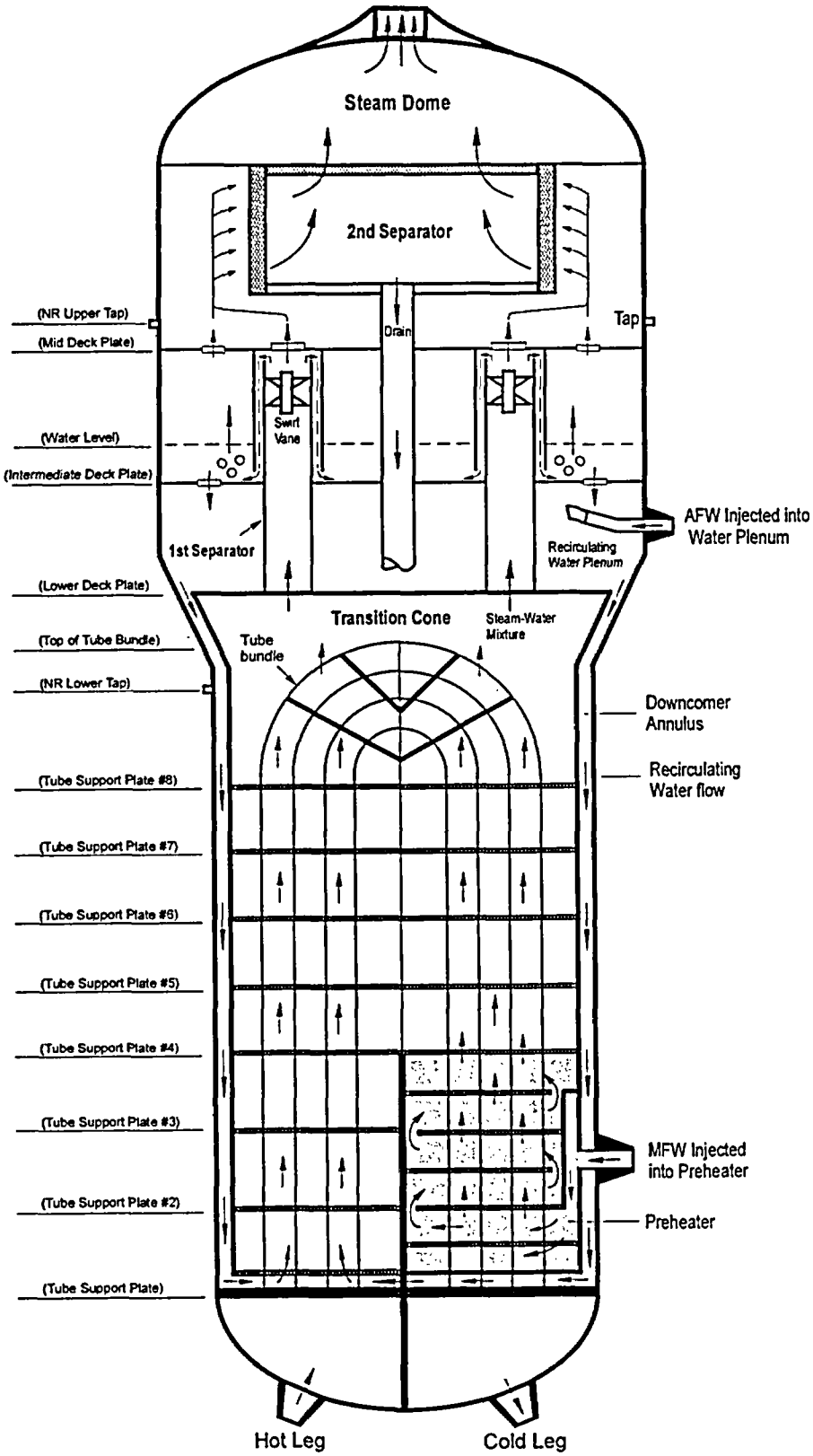

Figure 1 Westinghouse D-4 Steam Generator Outline

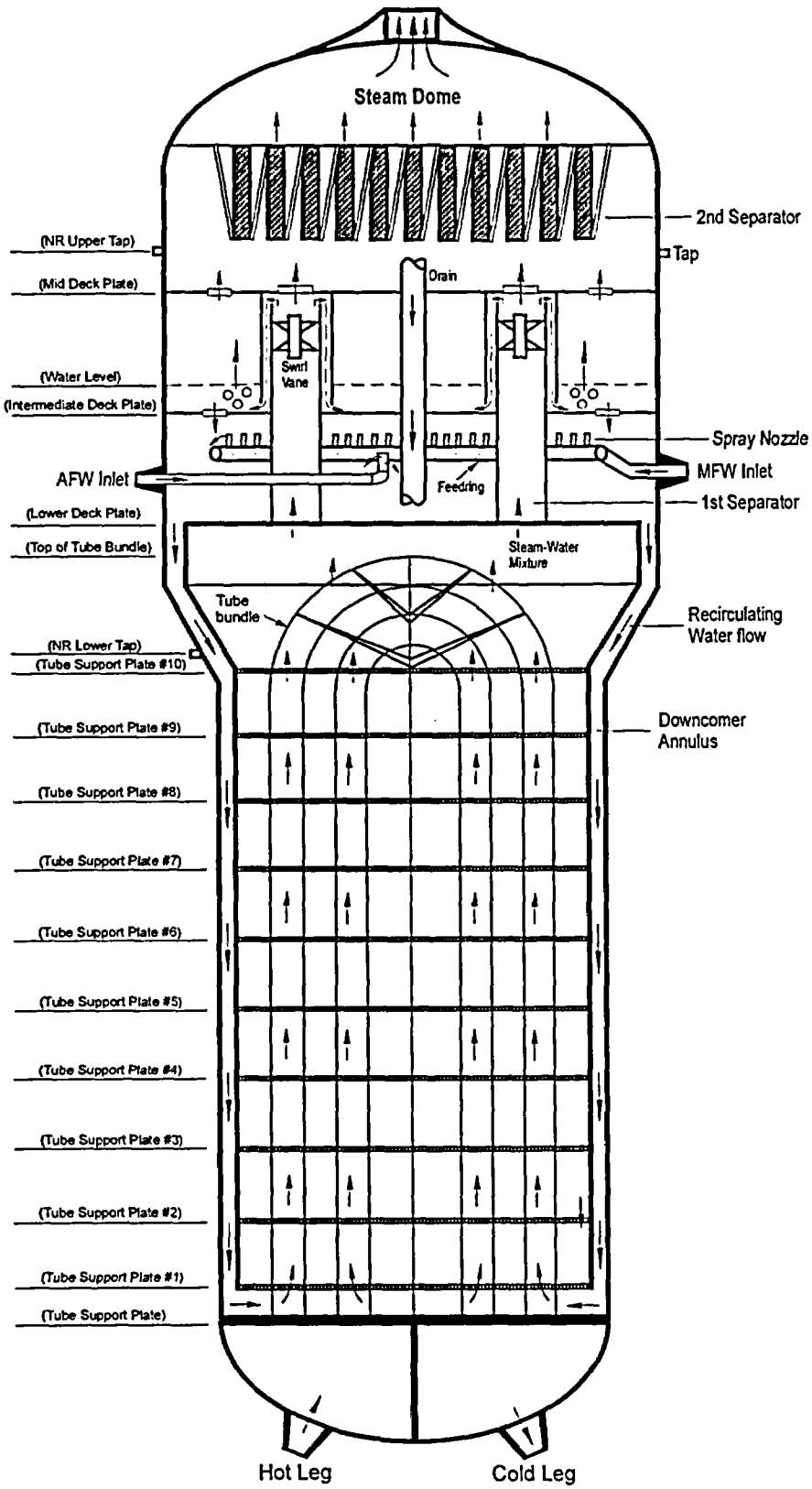

Figure 2 Westinghouse Δ76 Steam Generator Outline

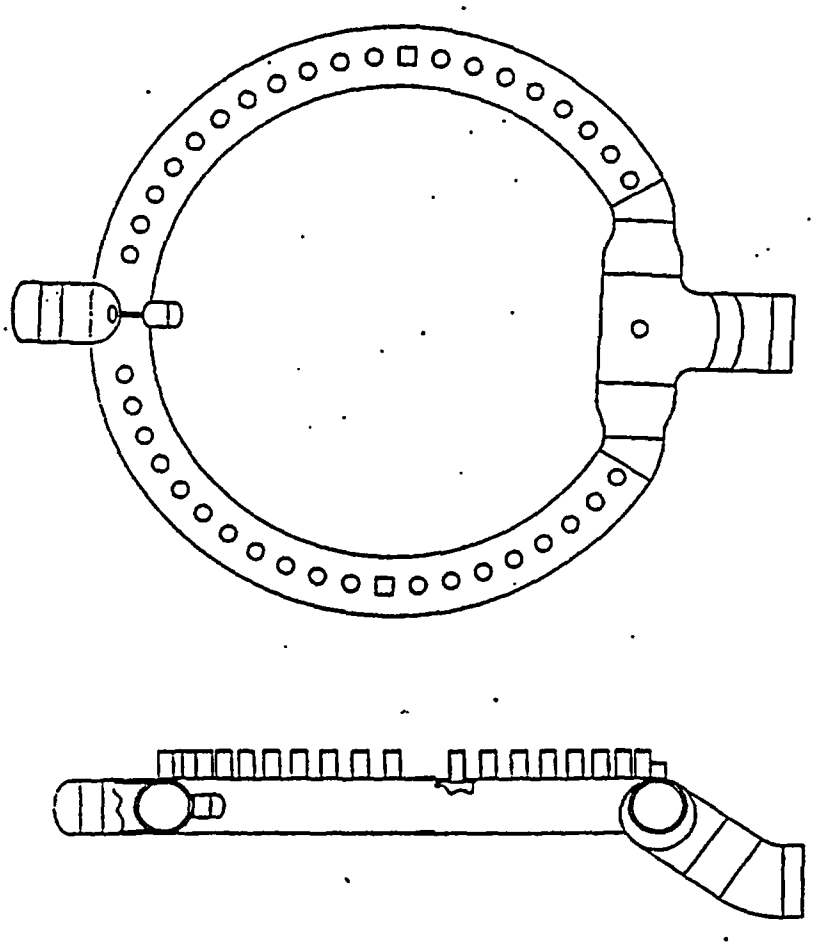

Figure 3 Δ76 Steam Generator Feed Ring

CHAPTER 3 - SUMMARY OF EFFECTS OF THE REPLACEMENT STEAM GENERATOR ON PREVIOUSLY APPROVED METHODOLOGIES

Assessments of the effects of the Δ76 steam generator design on the CPSES accident analysis methodology are provided in this chapter. An overview of the CPSES accident analysis methodology, which includes the use of a qualified system model, is also provided. The assessments of the effects of the Δ76 steam generator design on that methodology follow. The approach for presenting these assessments is:

- Describe the effect of the Δ76 steam generator design on the plant transient or accident.
- Determine whether any differences require a modification to the methodology (e.g., Do all parameters remain with approved limits of the models, correlations, etc.? Are there any new phenomena of importance? Are the same protection and mitigation functions used?)

3.1 CPSES Accident Analysis Methodology Overview

The general accident analysis methodology and philosophy is described in the report RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications" (Reference 4). The methodology is generally discussed in the main body of the topical report; more detailed assumptions for specific analyses are presented in the response to RAI Question 25.

The methodology description focuses on the philosophy for selecting initial conditions, core and reactivity-related parameters, the equipment assumed to be available for protection and mitigation of transients and accidents, as well as their performance capabilities, and the computer codes used in these analyses. The manner in which compliance with the relevant General Design Criteria is demonstrated is discussed as are general guidelines used in the development of the computer code models.

The basis for the selection of initial conditions is unchanged; however, the plant design imposes additional considerations in the selection of some initial conditions. These considerations are discussed elsewhere in this chapter. Otherwise, the methodology used for the non-LOCA transient and accident analyses remains unchanged.

The general guidelines used in the development of the computer code models and described in the report RXE-91-001-A (Reference 4) were used in the development of Δ76 steam generator models for use in the RETRAN-02 system analyses. In those analyses where a simple steam generator model was justified as adequate, a similar Δ76 steam generator model, using the dimensions of the Δ76 steam generator design will continue to be used. In those analyses requiring a more detailed steam generator model, a more detailed model was developed that is compliant with the general guidelines summarized in the report RXE-91-001-A. A description of the more detailed Δ76 steam generator model is presented in Chapter 4.

In the following sections, the effect of the Δ76 steam generator design on each of the non-LOCA transients and accidents is summarized. This effect is then evaluated to justify the adequacy of the existing CPSES accident analysis methodology for the analysis of that transient or accident.

3.2 Initial Conditions and System Performance Characteristics

The initial conditions for each of the transients and accidents will be affected as a direct result of the Δ76 steam generator design features. The larger heat transfer area permits operation with a higher steam pressure for the same Reactor Coolant System average temperature (T_{avg}). Conversely, the value of T_{avg} may be lowered if operation at the same steam pressure is desired. This second approach is expected to be used at CPSES Unit 1. Due to design considerations in the secondary plant, the full power value of T_{avg} will be reduced as necessary to maintain a steam pressure no higher than the maximum value used in the design of the secondary plant. For example, with the new steam generator, this value of T_{avg} is predicted to be approximately 4°F lower and is expected to increase toward the RCS maximum design value as the steam generator tubes become fouled or plugged.

Obviously, these parameters are inputs to the analyses. Regardless of the actual values used, the methods of analyses are unaffected by the choice of these nominal conditions. The methodologies for the selection of initial conditions, as described in Reference 4, remain unchanged: for a given parameter, the importance or sensitivity of the results to that parameter are assessed, the nominal value is determined, uncertainty allowances are specified, and, if appropriate, sensitivity studies are used to assess the conservative directions for the applications of the uncertainty allowances. For sensitive parameters, the nominal value ± the uncertainty allowance is specified as an initial condition or performance characteristic.

3.3 Evaluation of Effects on Specific Analysis Methods

A more detailed discussion of the effects of the replacement steam generator design on the transient and accident analysis methodologies is presented in the following sections, arranged in accordance with the CPSES FSAR Section number. While many of these observations are intuitive, analyses and sensitivity studies of the transients and accidents most affected by the Δ76 steam generator design have been performed to validate these expectations. The results are qualitatively discussed, as is appropriate.

FSAR Section 15.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

This transient is analyzed using the methodology described in topical report RXE-91-001-A (Reference 4). The magnitude of the feedwater temperature decrease is determined by the balance-of-plant response to the bypassing of a feedwater heater, which is unaffected by the Δ76 design. The incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

This transient is analyzed using the methodology described in Reference 4. The magnitude of the feedwater flow increase is a function of the feedwater pump capability and feedwater system configuration. The magnitude of the delivered main feedwater flow may change slightly due to anticipated changes in the feedwater piping layout; however, this value is an input to the analysis. The incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.1.3 Excessive Increase in Secondary Steam Flow

This transient is analyzed using the methodology described in Reference 4. The magnitude of the assumed step change in the secondary steam flow (+10%) is unaffected by the change to the steam generator design. The incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

This transient is analyzed using the methodology described in topical report RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events" (Reference 7). The size of the steam generator relief and safety valves, and thus the magnitude of the break, is unaffected by the Δ76 design. The incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.1.5 Steam System Piping Failure

This accident is analyzed using the methodology described in topical report RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Event" (Reference 7). The Δ76 steam generator design includes an integral flow restrictor in the steam outlet nozzle that has the same relief rate and characteristics as the integral flow restrictor in the D-4 steam generator design. The maximum steam relief rate is unaffected by the incorporation of the Δ76 steam generator design.

As described in Reference 7, the steam generator model used in the evaluation of the core response to a steamline break is a single-node representation. There are no differences between the D-4 and Δ76 steam generator designs that affect the steam generator blowdown characteristics; although, the duration of the blowdown will be slightly longer for the Δ76 steam generator design due to its larger secondary-side inventory. The incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.2.2 Loss of External Load

This transient analysis is bounded by the analysis of the Turbine Trip transient presented in FSAR Section 15.2.3.

FSAR Section 15.2.3 Turbine Trip

This transient is analyzed using the methodology described in Reference 4. As previously noted, the increased RCS mass results in a slightly slower heatup rate and a slightly greater rate of volumetric expansion for a given change in RCS temperature. The net effect is that the rate of pressurizer pressure increase is slightly slower, which

provides additional time for the pressurizer safety valves to respond and mitigate the RCS pressurization effects caused by the volumetric expansion of the RCS fluid. However, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.2.4 Inadvertent Closure of the Main Steam Isolation Valves

This transient is bounded by the analysis of the Turbine Trip transient presented in Section 15.2.3.

FSAR Section 15.2.5 Loss of Condenser Vacuum and Other Events resulting in a Turbine Trip

This transient is bounded by the analysis of the Turbine Trip transient presented in Section 15.2.3.

FSAR Section 15.2.6 Loss of Non-emergency AC Power to the Station Auxiliaries

This transient is analyzed using the methodology described in Reference 4. This transient and the Loss of Normal Feedwater Flow transient are modeled in similar manners; the difference is that at the time of reactor trip and turbine trip, offsite power to the station auxiliaries (including the reactor coolant pumps) is assumed to be lost. As previously noted, the increased RCS mass results in a slightly slower heatup, and a slightly greater rate of volumetric expansion for a given change in RCS temperature. The slower heatup rate provides additional time for the pressurizer safety valves to response and mitigate the RCS pressurization effects caused by the volumetric expansion of the RCS fluid. Nevertheless, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES

analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

There is one difference of importance in the development and selection of input parameters that should be noted. As described in Reference 4, a detailed steam generator model was developed by CPSES accident analysis engineers and used to determine a conservative mass equivalent of the low-low steam generator water level trip setpoint. For the Δ76 steam generator model, sufficient design data is available from the steam generator vendor to conservatively determine this mass equivalency without the need for the development of an additional detailed steam generator model. This difference is not considered to be a methodology difference, as equivalent information is developed and used in the same manner.

FSAR Section 15.2.7 Loss of Normal Feedwater Flow

This transient is analyzed using the methodology described in Reference 4. This transient and the Loss of AC Power to the Station Auxiliaries are modeled in similar manners; the difference is that offsite power is assumed to remain available throughout the transient. All discussions presented for that transient are applicable to the Loss of Normal Feedwater Flow transient analysis. As previously concluded, although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.2.8 Feedwater System Pipe Break

This transient is analyzed using the methodology described in Reference 4. In the preheat steam generator design, a feedwater line break will result in the drainage of subcooled or saturated liquid from the preheater box near the bottom of the secondary side of the steam generator. The RCS fluid will heat up throughout this period due to the lack of main feedwater flow to all of the steam generators. There is no significant depressurization of the secondary side until the mass in the affected steam generator is nearly depleted.

Following a main feedwater line break in the Δ76 feed ring design, saturated liquid will initially be expelled through the feed ring spray nozzles, through the feed ring assembly, and out the main feedwater nozzle. However, within a few seconds, the feed ring spray nozzles will be exposed directly to steam, and the ensuing blowdown will resemble a large steamline break. During this blowdown period, the RCS fluid will cool down, just as is the case for a steamline break. For larger feedwater line break sizes, the main steamlines will quickly depressurize, resulting in steamline isolation at a relatively early time in the event.

The course of this accident is significantly different than the feedwater line break accident described in Reference 4. Even though the same general accident analysis methodology will be used to analyze this accident, the provision of additional details of the application of the CPSES analysis methodology is considered appropriate. This additional information is provided in Chapter 4.

FSAR Section 15.3.1 Partial Loss of Forced Reactor Coolant Flow

This transient is analyzed using the methodology described in Reference 4. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used

within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.3.2 Complete Loss of Forced Reactor Coolant Flow

This transient is analyzed using the methodology described in Reference 4. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)
FSAR Section 15.3.4 Reactor Coolant Pump Shaft Break

These two events are bounded by a single analysis in which the most severe aspects of each event are combined into a single scenario. This accident is analyzed using the methodology described in Reference 4. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.4.1 Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition

This transient is analyzed using the methodology described in topical report RXE-91-002-A, "Reactivity Anomaly Events Methodology" (Reference 5). As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.4.2 Uncontrolled RCCA Bank Withdrawal at Power

This transient is analyzed using the methodology described in Reference 5. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.4.3 Rod Cluster Control Assembly Misalignment

A. One or more dropped RCCAs within the same group

This transient is analyzed using the methodology described in Reference 5. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The

numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

B. One Dropped RCCA bank

This transient is analyzed using the methodology described in topical report Reference 5. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

C. Statically Misaligned RCCA

This transient is analyzed using the methodology described in Reference 5. This analysis is performed at steady-state conditions and has no sensitivity to the steam generator model used.

D. Withdrawal of a Single RCCA

This transient is analyzed using the methodology described in Reference 5. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.4.4 Startup of an Inactive RCP at an Incorrect Temperature

This transient is not analyzed because it is precluded by the CPSES Technical Specifications.

FSAR Section 15.4.6 CVCS Malfunctions that result in a Decrease in Boron Concentration in the Reactor Coolant (Boron Dilution)

Boron dilution events are considered in all six operating modes. For Modes 1 and 2, the methodology described in Reference 5 is used. For Modes 3, 4, and 5, the methodology described in topical report RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5" (Reference 8) is used. Inadvertent boron dilution events are precluded by administrative control of dilution sources in Mode 6.

The increased RCS mass results in a slightly slower rate of boron dilution; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The CPSES RETRAN-02 model is not required for this analysis. No changes to the method of analysis is required for this transient.

FSAR Section 15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (Misloaded Assembly)

This transient is analyzed using the methodology described in Reference 5. This analysis is performed at steady-state conditions and has no sensitivity to the steam generator model used.

FSAR Section 15.4.8 Spectrum of RCCA Ejection Accidents

This transient is analyzed using the methodology described in Reference 5. The duration of this analysis is so short that the results are insensitive to the steam generator model used. For accuracy, the steam generator model will be updated to properly reflect the Δ76 design; however, no changes in the method of analysis are required.

FSAR Section 15.5.1 Inadvertent Operation of the ECCS System During Power Operation

This transient is analyzed using the methodology described in Reference 4. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.5.2 CVCS Malfunction that Increases Reactor Coolant System Inventory

This event is bounded by the analysis presented in Section 15.5.1.

FSAR Section 15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

This transient is analyzed using the methodology described in Reference 4. As previously noted, the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications. Although the steam generator

portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

FSAR Section 15.6.2 Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment

This analysis is performed to assess the radiological consequences of the titled break. Simplified assumptions are used to conservatively determine the mass releases. These assumptions are independent of the steam generator design; therefore, no changes in the method of analysis are required.

FSAR Section 15.6.3 Steam Generator Tube Failure

This transient is analyzed using the methodology described in topical report TXX-88306, "Steam Generator Tube Rupture Analysis" (Reference 6). Both the decreased secondary side volume and the increased RCS mass results in a slightly different transient response; however, the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The affected steam generator does not completely fill with liquid; therefore, the limiting single failure scenario with respect to radiological dose consequences is unchanged. The numerous models that constitute the RETRAN-02 model of CPSES continue to be used within their previously approved ranges and applications.. Although the steam generator portion of the CPSES RETRAN-02 system model must be updated to properly reflect the Δ76 design, no changes in the method of analysis are required.

In summary, the existing methodologies used for the non-LOCA transient and accident analyses presented in Chapter 15 of the Comanche Peak Final Safety Analysis Report remain applicable to the Δ76 steam generator design. However, due to the significant differences in the course of the accident, additional discussions of the application of the methodology to the feedline break accident are presented in Chapter 4.

CHAPTER 4 - Δ76 SPECIFIC STEAM GENERATOR ACCIDENT ANALYSIS METHODS

4.1 Δ76 Steam Generator Model Description and Qualification

The specifics of the steam generator model to be used in each of the transient and accidents analyses is described in each of the cited methodology topical reports, although, the use of alternate models is permitted if justified through sensitivity studies. In several of those transients and accident analyses, where only the gross steam generator performance is of importance, a single-node steam generator was justified as sufficient.

In addition, the use of three-node D-4 steam generator representation was justified for use in those transients and accidents where the steam generator performance was a more significant factor. The three-node D-4 steam model consists of a preheater region, a separated boiling region, and a steam dome. This model was incorporated into the CPSES system model and qualified as an integral part of the RETRAN-02 system model through comparisons to plant transients. For those transients where a reactor trip and/or auxiliary feedwater flow would be initiated on the low-low steam generator water level trip function, a more detailed steam generator model was used in a "stand-alone" mode to develop the mass-equivalent to the low-low steam generator water level trip setpoint.

A single-node RETRAN-02 representation of the Δ76 steam generator design can be easily adapted using the specific geometry of the Δ76 steam generator design. This model is not significantly different from the D-4 steam generator model; the Δ76 model volumes, junctions and heat conductors are simply representative of the Δ76 dimensions. No additional qualification of this model is considered necessary.

Because the Δ76 steam generator does not have an integral preheater, the three-node representation used for the D-4 model can not be directly transformed into a Δ76 representation. Even though a multi-node representation of the Δ76 steam generator could have been developed by CPSES accident analysis engineers based on design drawings and modeling experience, the issue of model qualification would remain; there is no CPSES-specific plant data available for qualification of the Δ76 feed ring steam generator model.

To facilitate the qualification of an appropriate steam generator model for the Δ76 feed ring steam generator design, the multi-node model of a feed ring design steam generator developed and qualified by Westinghouse Electric Company and approved by the NRC in WCAP 14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors Non-LOCA Safety Analyses," (Reference 9) was chosen. The Δ76 steam generator model used in the CPSES Unit 1 RETRAN-02 model was developed by Westinghouse engineers for use at CPSES using the approved methods described in Reference 9. This model, presented graphically in Figure 3.6-2 of Reference 9, includes an explicit model of the internal recirculation loop. The Δ76 steam generator model is used by CPSES accident analysis engineers in the same manner as described in Reference 9.

CPSES accident analysis engineers integrated this Δ76 model into the CPSES RETRAN-02 system model qualified as described in Reference 4. This model was validated through steady-state comparisons to detailed design data provided by the vendor. In addition to the comparisons with the detailed vendor data, the integrated Δ76 system model was used to analyze selected non-LOCA accidents analyses; the results were compared to the results obtained using the currently approved models with the D-4 steam generator design. These comparisons demonstrate that the incorporation of the Δ76 model did not affect other parts of the qualified model.

In summary, the modification of the CPSES single-node steam generator model to reflect the dimensions of the Δ76 steam generator design is a minor change that does not require additional qualifications for its intended use. In addition, a multi-node model of the

CPSES Δ76 steam generator design was developed based upon the NRC-approved methods presented and qualified in WCAP-14882-P-A. These methods are also consistent with the CPSES general modeling guidelines. CPSES accident analysis engineers incorporated this Δ76 model into the previously approved RETRAN-02 system model of CPSES. The adequacy of the installation was verified through comparisons to detailed design data provided by the steam generator vendor and supplemented by comparisons to CPSES engineering calculations performed using the NRC-approved methods and models with the D-4 steam generator design.

4.2 Δ76 Steam Generator Water Level Model

In both the D-4 and Δ76 steam generator designs, the steam generator water level indication is based on the differential pressure between the upper and lower narrow range instrument taps. Due to the coarseness of the D-4 steam generator model used in the non-LOCA transient analyses, an accurate representation of this differential pressure can not be produced. Therefore, in the NRC-approved CPSES transient analysis methods, a more detailed steam generator model was developed and used to produce a transient-specific secondary-side fluid mass that is equivalent to the differential pressure at the setpoint of interest (e.g., 0% narrow range span).

The same difficulties with the representation of the water level indication exist in the multi-node model of the Δ76 steam generator design. The lower narrow range instrument tap is located near the bottom of a non-separated (homogeneous) volume in the internal recirculation loop. With the homogeneous volume, the mass is assumed to be distributed uniformly throughout the volume; thus, for simulation of the low-low water level trip setpoint (at 0% span), an unrealistic amount of mass would be predicted to be within the narrow range span, effectively precluding the prediction of 0% span until the internal circulation was disrupted and the steam generator dried out.

Using the same philosophy presented in Reference 4, a more detailed steam generator model is to be used to develop an appropriate mass-equivalent approximation of a particular steam generator water level setpoint. This approximation is somewhat

transient-specific. For the Loss of Normal Feedwater and Loss of Nonemergency AC Power to the Station Auxiliaries transient analyses, detailed vendor design information is available to determine a conservative secondary-side mass equivalent to the low-low steam generator water trip setpoint of 0% narrow range span used in the accident analyses.

For the analysis of the Δ76 Feedline Break accident analysis, the calculated differential pressure between the upper and lower narrow range taps is used to derive the steam generator water level indication. This approach results in an equivalent mass corresponding to the 0% narrow range span that is significantly smaller that what would be expected based on sensitivity studies and engineering judgement and is, therefore, conservative. This simplified approach is used due to the significant amount of margin to the relevant event acceptance criteria.

4.3 Feedwater System Pipe Break Analysis Methodology for a Feed ring Steam Generator Design

The design-basis feedwater system line break accident (FLB) is defined to be a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain the shell-side fluid inventory. Depending upon the size and location of the postulated rupture and the assumed initial operating conditions, the event can cause either a cooldown or a heatup of the reactor coolant systems. If the break is postulated to occur in the feedwater piping between the steam generator and the first check valve upstream, then not only will feedwater flow be terminated to the faulted steam generator, but fluid from the faulted steam generator will be discharged through the break. A break upstream of the check valve would affect the RCS as a Loss of Normal Feedwater transient.

A FLB reduces the capability of the secondary system to remove heat generated by the core and the reactor coolant pumps. The feedwater flow to the steam generators is reduced or terminated, resulting in a decrease in the shell-side fluid inventory. Moreover, fluid from the faulted steam generator can be totally discharged through the broken pipe,

thereby eliminating the capability of the steam generator to remove heat from the RCS. A broken feedwater line can also reduce the addition of main feedwater to the intact steam generators.

The feedwater line break is one of the events which defines the required minimum capacity of the Auxiliary Feedwater System for removing core residual heat following reactor trip. If sufficient heat removal capability is not provided, the core residual heat following the depressurization of the affected steam generator could eventually increase the RCS coolant temperature to the extent that the resulting fuel damage would compromise the maintenance of a coolable geometry of the core and result in potential radioactive releases. A feedwater line break could also cause a pressure increase in both the RCS and the main steam system challenging the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary until the pressure-relieving devices of the two systems act to limit the magnitude of the increase.

Postulated Sequence of Events

The FLB event for a feed ring steam generator design is significantly different from the FLB event for a preheat steam generator design. In the preheat steam generator design, the break flow from the faulted steam generator will consist of subcooled fluid, eventually turning to steam as the steam generator inventory is almost depleted. The inventory is essentially drained through the feedwater nozzle located at an elevation near the bottom of the steam generator secondary side.

In the feed ring design, the fluid blowdown pathway is through the feed ring spray nozzles located at an elevation slightly below the normal operating water level. With this configuration, the water level will quickly decrease below the feed ring spray nozzles. After that point, the remainder of the affected steam generator blowdown will closely resemble a steamline break accident. The amount of energy removed from the RCS as the affected steam generator's inventory is depleted will be significantly greater for the feed ring design relative to the preheat design.

Following the break initiation and uncovery of the feed ring spray nozzles, the affected steam generator depressurizes through the break. In addition, the intact steam generators depressurize through the main steam header and then through the break. A low compensated steam pressure signal will initiate main steamline isolation and a Safety Injection Actuation Signal. This "S" signal initiates signals which initiate reactor trip, turbine trip, main feedwater isolation, and actuation of the two, 50%-capacity motor-driven auxiliary feedwater pumps. One of these pumps is typically assumed to be the single failure, and the second pump is assumed to preferentially deliver its capacity to the depressurized affected steam generator. Thus, no auxiliary feedwater is available for cooling the RCS until the turbine-driven auxiliary feedwater pump is initiated on low-low steam generator water level signal in more than one steam generator. During this time period, cooling is provided through steam release through the main steam system valves on the intact steam generators. Because the reactor is tripped relatively early in the accident and significant energy will be removed from the RCS via the depressurized steam generator, several hundred seconds may lapse before the low-low steam generator water level is exceeded in the intact steam generators.

At that time, a significant portion of the turbine-driven auxiliary feedwater pump's capacity is preferentially delivered to the depressurized affected steam generator. Operator action to isolate the Auxiliary Feedwater System from the affected steam generator is credited at 30 minutes into the accident. At this time, the capacity of the Auxiliary Feedwater System is sufficient to remove the core decay heat.

Based on sensitivity studies performed for a spectrum of postulated break sizes, it was observed that the low compensated steamline pressure setpoint may not be exceeded for smaller break sizes. However, these break sizes are not sufficient to significantly reduce the amount of main feedwater delivered to the intact steam generators. These break sizes were determined to be non-limiting relative to the relevant event acceptance criteria.

Δ76 Feedline Break Analysis Methodology

The FLB accident is classified as an ANS Condition IV accident. The relevant event acceptance criteria for this accident are:

1. The pressure in the reactor coolant and main steam systems should be maintained below 110% of their design values.

2. Any fuel damage which is calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. This criterion is satisfied by imposing the requirement that boiling is not allowed to occur in the hot leg, i.e., the temperature in the hot leg remains below the saturation temperature.

Cases are evaluated both with and without offsite power available. These cases are evaluated to demonstrate that the heat removal rate from the RCS, aided by the Auxiliary Feedwater System, is sufficient to satisfy the ANS Condition IV event acceptance criteria. The Auxiliary Feedwater System is comprised of two 50%-capacity motor-driven auxiliary feedwater pumps, each delivering flow to two steam generators. A single, 100%-capacity turbine-driven auxiliary feedwater pump delivers flow to all four steam generators. The auxiliary feedwater piping is independent of the Main Feedwater System piping.

The FLB accident evaluated in the following demonstration analysis is a double-ended rupture of the feedline between the affected steam generator and the main feedwater check valve. The break flow from the affected steam generator is limited to 1.118 ft^2 by the flow areas of the spray nozzles located on top of the feed ring. (A spectrum of smaller breaks was considered; for these breaks, significant main feedwater could continue to be delivered to the intact steam generators, as well as, most likely, to the affected steam generator. Under this scenario, the transient resembles a loss of feedwater accident until the feed ring is uncovered in the affected steam generator and a small steamline break thereafter. These events are analyzed elsewhere.)

Following the FLB, the reactor and turbine are tripped when the low compensated steam line pressure setpoint is reached. For the case where offsite power is assumed to be lost, the reactor coolant pumps are assumed to trip concurrently with the reactor and turbine trips. This analysis approach is conservative for demonstrating that the natural circulation heat removal capability of the RCS is sufficient to satisfy the ANS Condition IV event acceptance criteria.

The plant initial conditions are selected to maximize the initial stored energy in the fuel and to maximize the RCS heatup following the depressurization of the affected steam generator. The initial core power, RCS temperatures, and pressurizer pressure assumed to be at their maximum values, including uncertainties. In addition, because a delay in the delivery of the turbine-driven auxiliary feedwater results in a greater RCS heatup, the initial steam generator water level in the affected steam generator is assumed to be at its maximum normal operating level, including uncertainties.

The reactivity parameters used in the FLB analyses are:

1. The maximum moderator density coefficient is assumed. As shown through sensitivity studies, this assumption is conservative for the cooldown period when all steam generators are depressurizing.

2. The maximum Doppler temperature coefficient is assumed in order to retard the decay of power following the reactor trip.

3. The FLB event is not sensitive to the values of the effective delayed neutron fraction and the prompt neutron lifetime.

4. No credit is taken for boration following ECCS injection. The maximum ECCS temperature is assumed. Both of these assumptions have only a minor effect on the transient response.

Other analysis assumptions, such as control system operation and safety analysis limits for trip setpoints, are selected as described in Reference 4.

Due to the course of the Δ76 FLB accident, wherein the two motor-driven auxiliary feedwater pumps are started several hundred seconds before the turbine-driven auxiliary feedwater pump receives a start signal, the limiting single failure for this event is the failure of the auxiliary feedwater pump feeding two intact steam generators. As previously noted, the second motor-driven auxiliary feedwater pump is assumed to feed the affected steam generator without removing any heat.

Following the receipt of a low-low steam generator water level signal from two or more steam generators, starting the turbine-driven auxiliary feedwater pump, an additional delay is included for the pump to start and get up to rated speed. Because the auxiliary feedwater lines are independent of the main feedwater lines, and protected against back fill from the steam generators by check valves, allowances for purging the auxiliary feedwater lines of hotter fluid are not required. Approximately half of the flow from the turbine-driven auxiliary feedwater pump, when it is started, is assumed to feed the affected steam generator without removing any heat. Following the operator action to isolate the Auxiliary Feedwater System from the break, the remaining capacity of the turbine-driven auxiliary feedwater pump plus the capacity of the non-faulted motor-driven auxiliary feedwater pump are assumed to be delivered to the intact steam generators.

Once the break is isolated by the operator and a sufficient, a continuous supply of auxiliary feedwater is restored and the RCS is continuously cooling down, the event is considered terminated from an analysis perspective.

Analysis Results

The FLB cases were analyses with and without the loss of offsite power. The limiting case was determined based on the margin to the event acceptance criterion of no hot leg boiling. For the purpose of the demonstration analyses, only the limiting case, wherein offsite power is assumed to be available, is presented.

The sequence of events during the FLB accident is presented in Table 2. Figures 4 through 9 illustrate the transient responses of key parameters in the primary and secondary systems.

Figures 4 and 5 show the break flow rate and quality from the faulted steam generator. After the FLB starts at 10 seconds, the near-saturated liquid flows through the break until the steam generator water level reaches the top of the break-junction area (the spray nozzles on top of the feed ring). After that, the fluid, containing a steam-water mixture as shown in Figure 5, starts to flow through the break with the quality continuing to increase to 100 percent at the point where the water level drops below the break junction.

When steam starts to be released from the affected steam generator, the pressure in the affected SG begins to decrease at a higher rate (see Figure 6). The low compensated steamline pressure setpoint is exceeded in about 20 seconds. Upon reaching this setpoint, a safety injection actuation signal starts the safety injection pumps and provides a reactor trip signal and a turbine trip signal to close the turbine stop valves. The main steam isolation valves then close, isolating the main steamlines.

Figure 7 shows that the RCS experiences further cooldown after the reactor trip as the steam blowdown through the break continues until the affected steam generator inventory (Figure 8) is almost depleted. After that, a gradual RCS heatup will follow due to the residual heat. As shown in Figure 7, the RCS temperature remains relatively stable after approximately 105 seconds when the core decay heat is absorbed by the heat sinks of the intact steam generators.

As the RCS pressure increases, the pressurizer safety valves function as designed to protect the integrity of the RCS pressure boundary. Figure 9 shows that the pressurizer safety valves open upon the pressure reaching the opening setpoint. From the plots (Figures 9 and 6), it is evident that the pressurizer pressure and main steam pressure remain well below 110% of their design pressures of 2485 and 1185 psig, respectively. Figure 7 illustrates the magnitude of subcooling in the hot leg which shows a subcooling margin greater than 40°F for the cases studied. The reactor operators are credited with isolating the break from the auxiliary feedwater supply lines 30 minutes into the event. At this point, the Auxiliary Feedwater System provides flow to maintain a continuous cooldown of the RCS.

Comparisons between the Δ76 and D-4 FLB Analyses

Plots of system parameters for the comparison with the previous FLB analyses for D-4 steam generator using the CPSES methodology are included in Figures 10 – 16. One key difference stems from the maximum break size. The D-4 steam generator design has integral flow restrictors in the feedwater nozzles, effectively limiting the maximum break size to 0.22 ft^2. In the Δ76 design, the maximum break size is 1.118 ft^2. It is noted that the break fluid flowing out of the D-4 steam generators is at an elevation near the bottom of the SG secondary side while the break fluid flows out of the Δ76 SG through the feed ring at a much higher elevation. This difference affects the amount of energy removed during the blowdown phase of the accident. Primarily because of the large amount of energy removed through the steam blowdown of the Δ76 steam generator design, it may be observed that the Δ76 FLB event is less limiting in challenging the RCS and main steamline pressure boundary as well as RCS subcooling margin.

Table 2 – Feedwater Line Break Analysis – Sequence of Events

Action	Time (Sec)
	100%
Start Problem/Steady State Initial Conditions	0.0
Feedline break occurs, feedwater to intact SG lost	10.0
Low steamline pressure setpoint reached	20
Reactor Trip	21
Main steam isolation valves start to close	22
First main steam safety valve opens(1)	30
Start safety injection (27 sec after low steamline pressure)	47
Lo-Lo SG water level setpoint reached in affected SG	56
Pressurizer safety valves open	432
Lo-Lo SG water level setpoint reached in intact SGs	935
Three intact steam generators receive flow from AFW system	1020
Operator action to isolate AFW system from affected SG; AFW injection flow exceeds that required to remove core decay heat	1810
Problem end	1900

Note (1): Time is the average time for three intact loops.

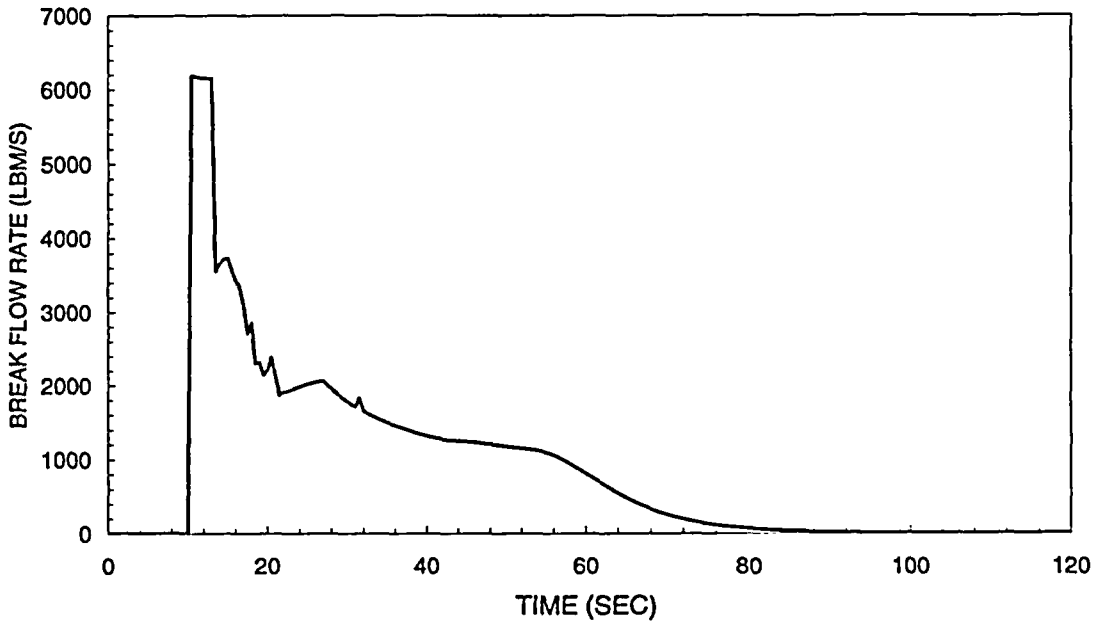

Figure 4 Feedline Break Analysis – Break Flow Rate

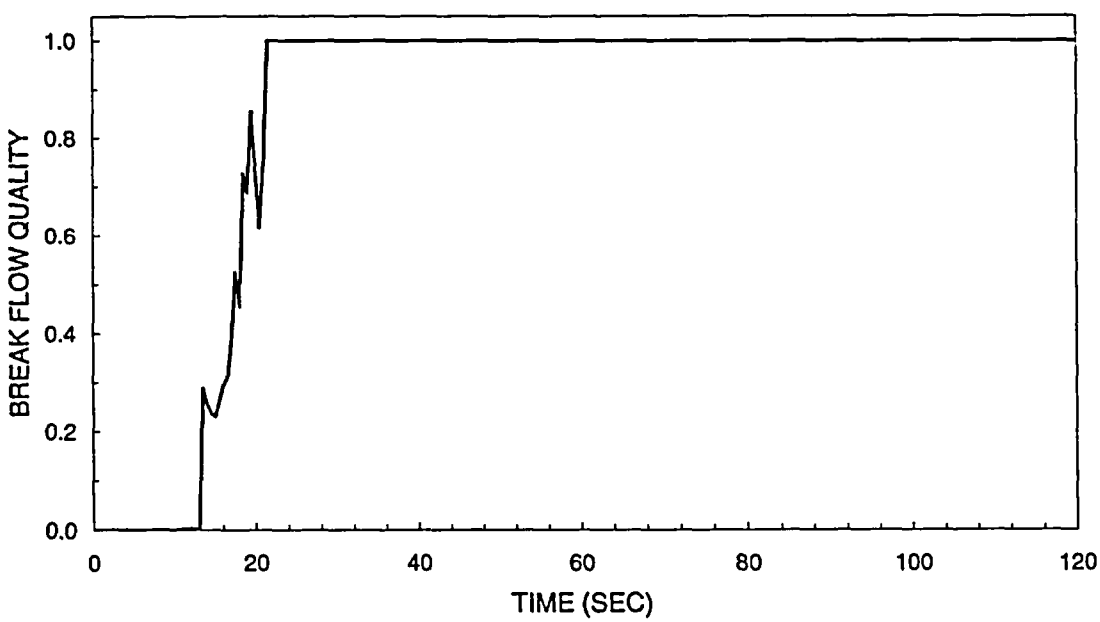

Figure 5 Feedline Break Analysis – Break Flow Quality

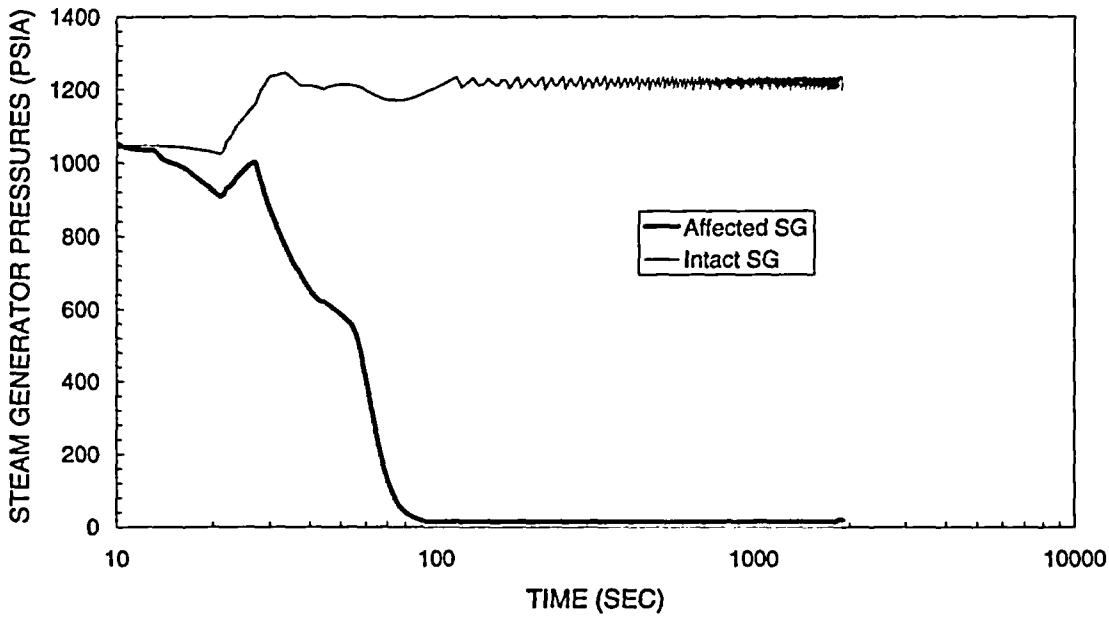

Figure 6 Feedline Break Analysis – Affected and Intact Steam Generator Pressures

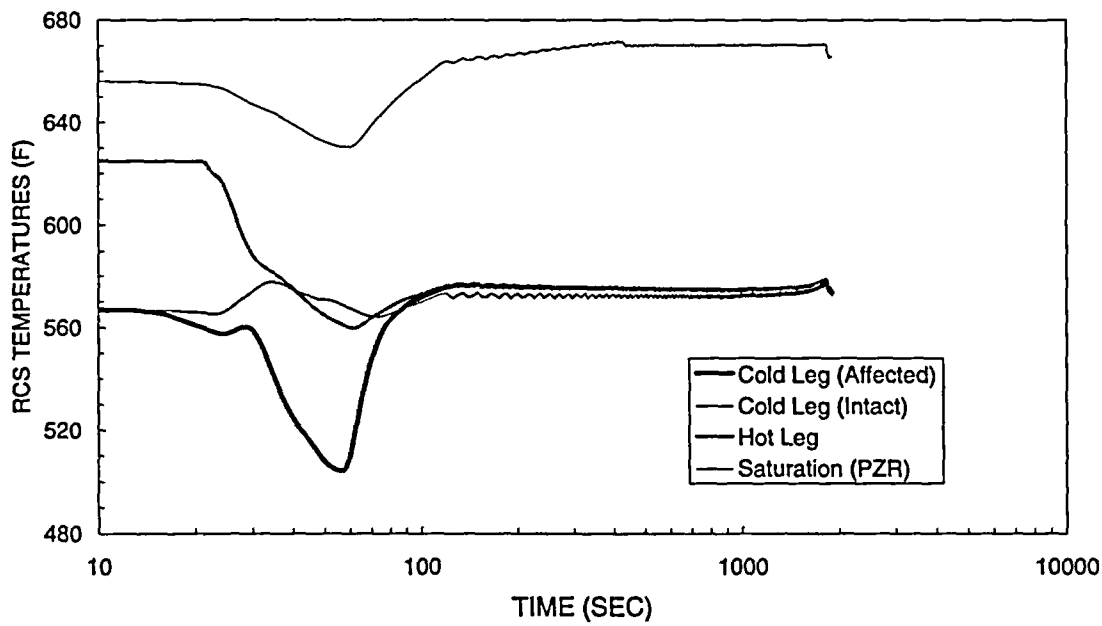

Figure 7 Feedline Break Analysis – RCS Hot and Cold Leg Temperatures and Saturation Temperatures

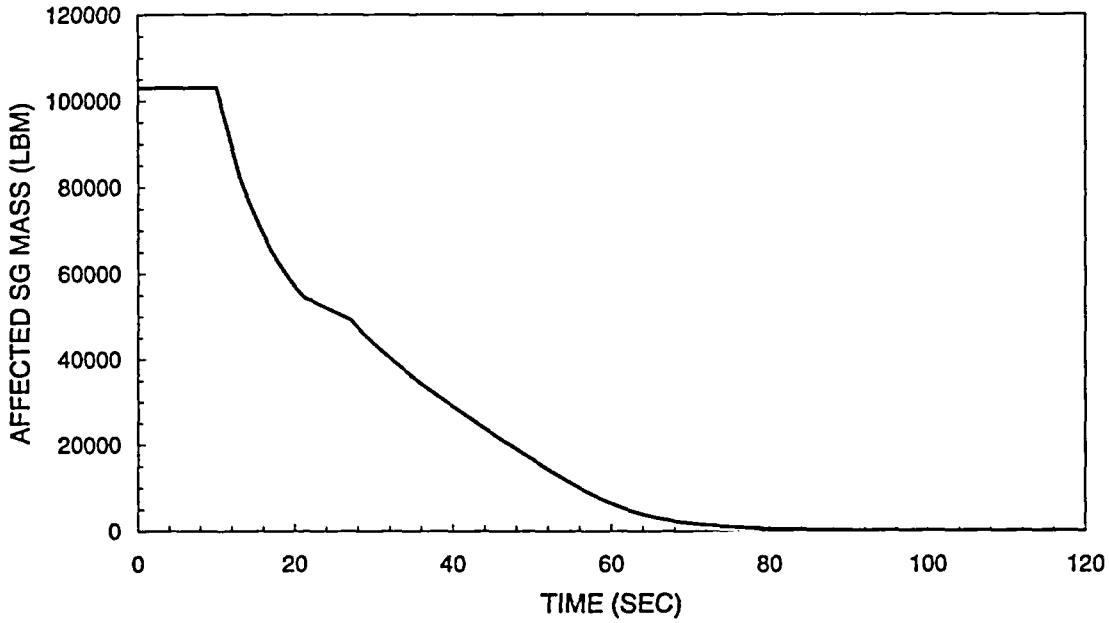

Figure 8 Feedline Break Analysis – Affected Steam Generator Secondary-Side Inventory

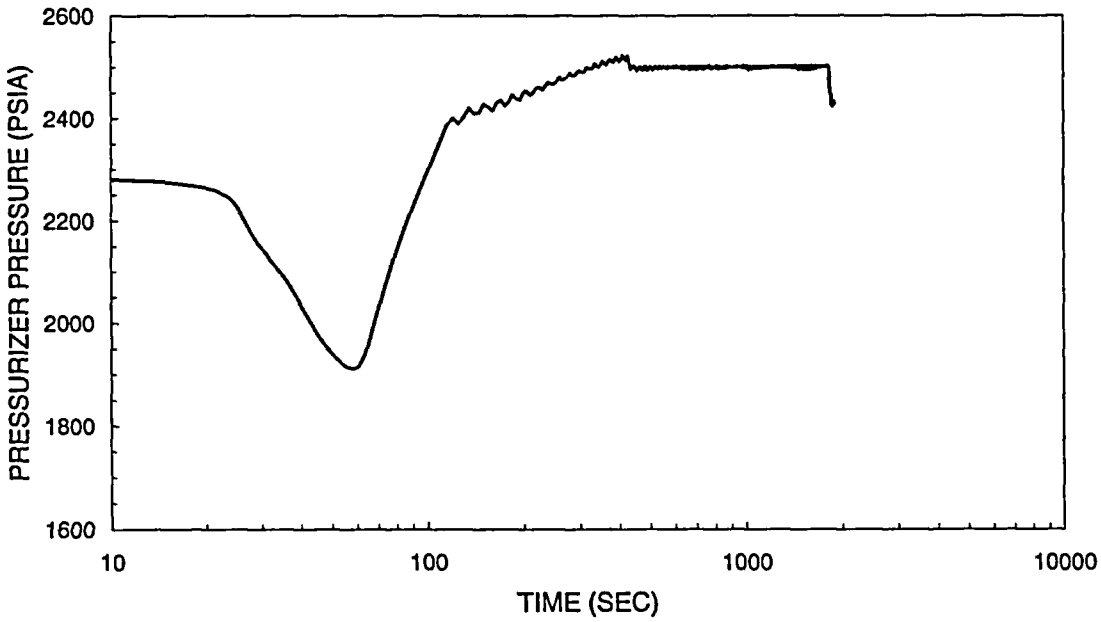

Figure 9 Feedline Break Analysis – Pressurizer Pressure

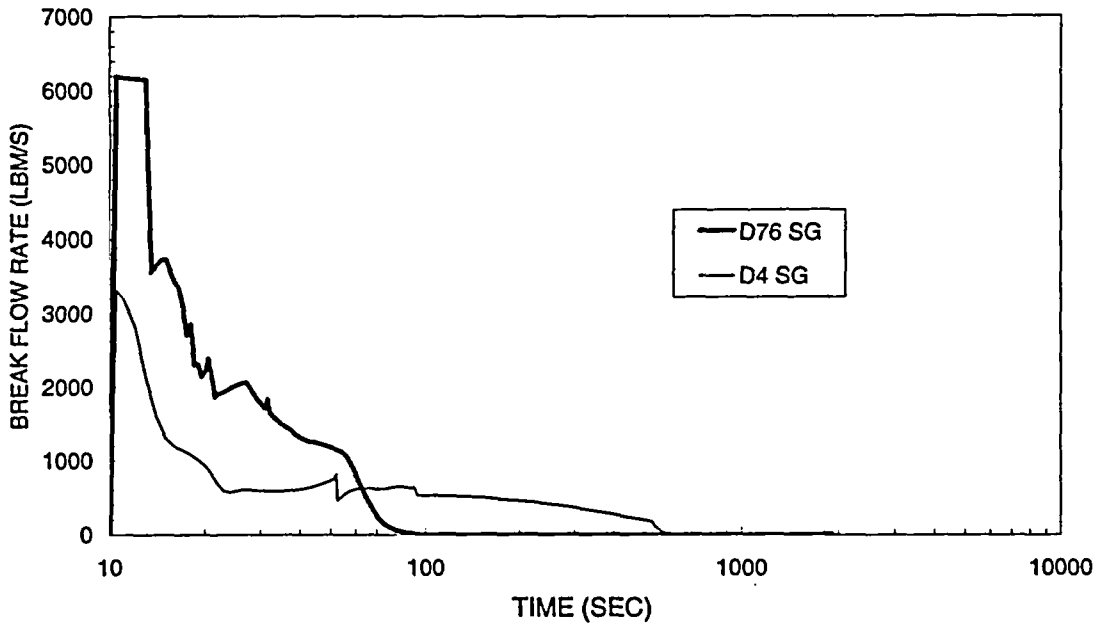

Figure 10 Feedline Break Analysis - Δ76 and D-4 Comparisons – Break Flow Rate

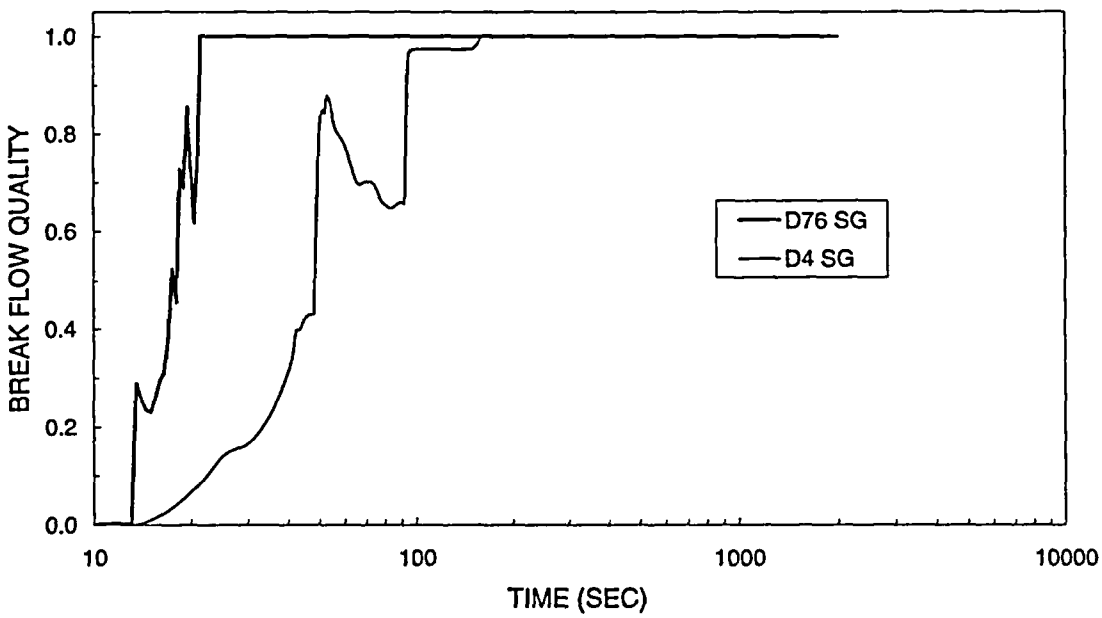

Figure 11 Feedline Break Analysis - Δ76 and D-4 Comparisons – Break Flow Quality

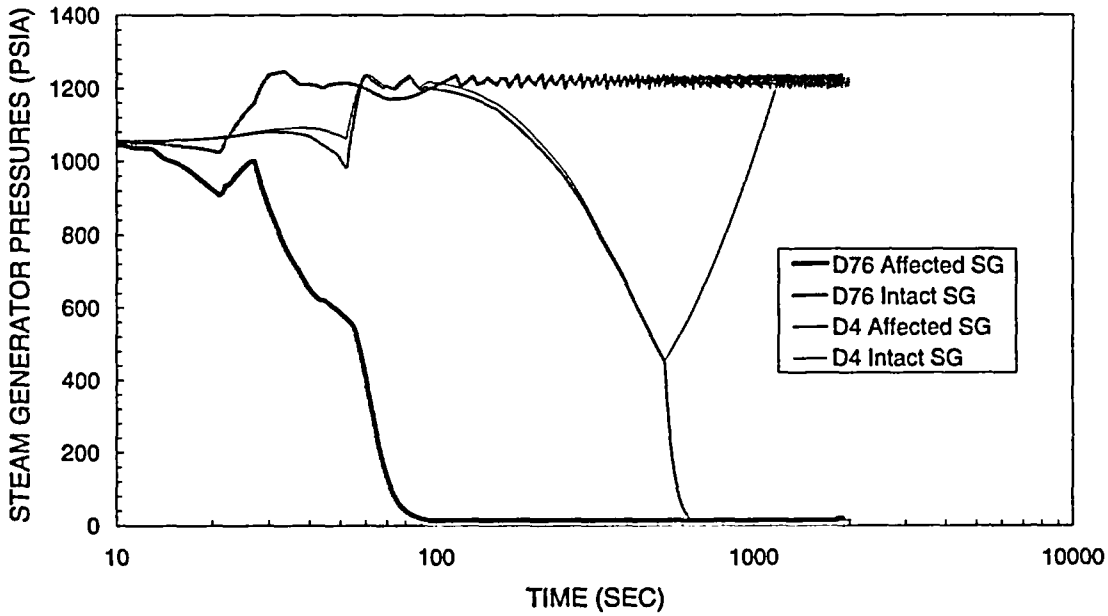

Figure 12 Feedline Break Analysis - Δ76 and D-4 Comparisons – Affected and Intact SG Pressures

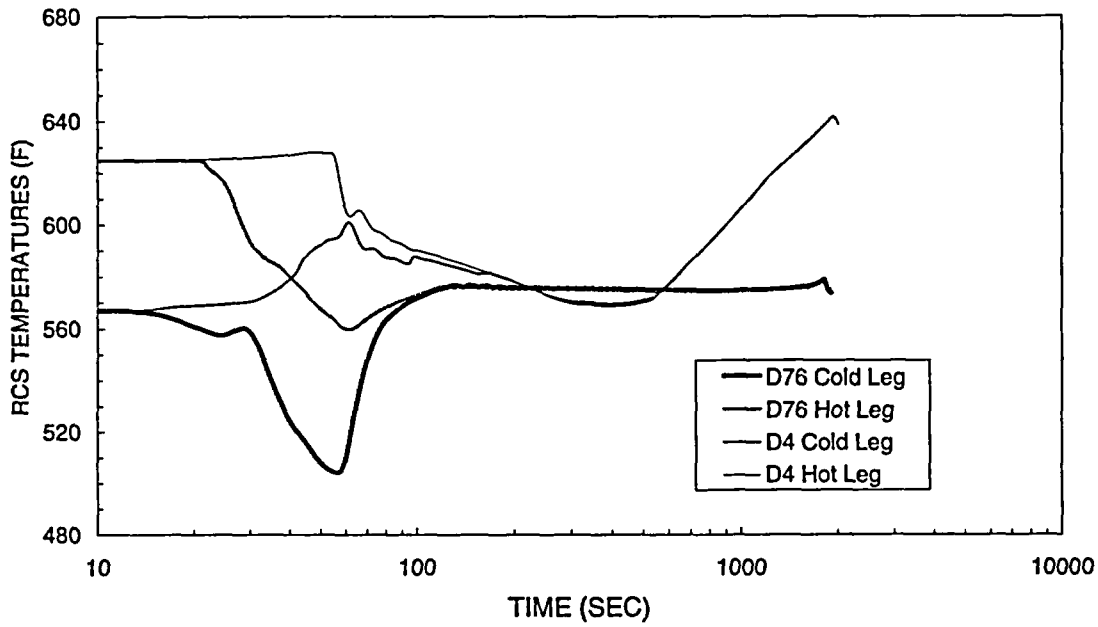

Figure 13 Feedline Break Analysis - Δ76 and D-4 Comparisons – Cold and Hot Leg Temperatures in the Affected Loop

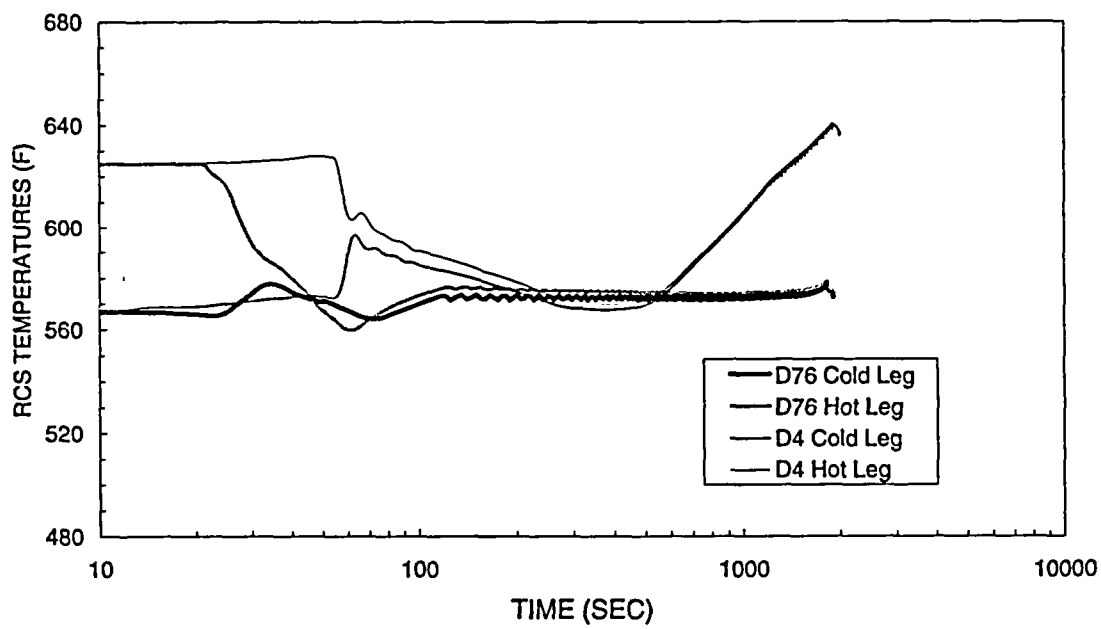

Figure 14 Feedline Break Analysis - Δ76 and D-4 Comparisons – Cold and Hot Leg Temperatures in the Intact Loop

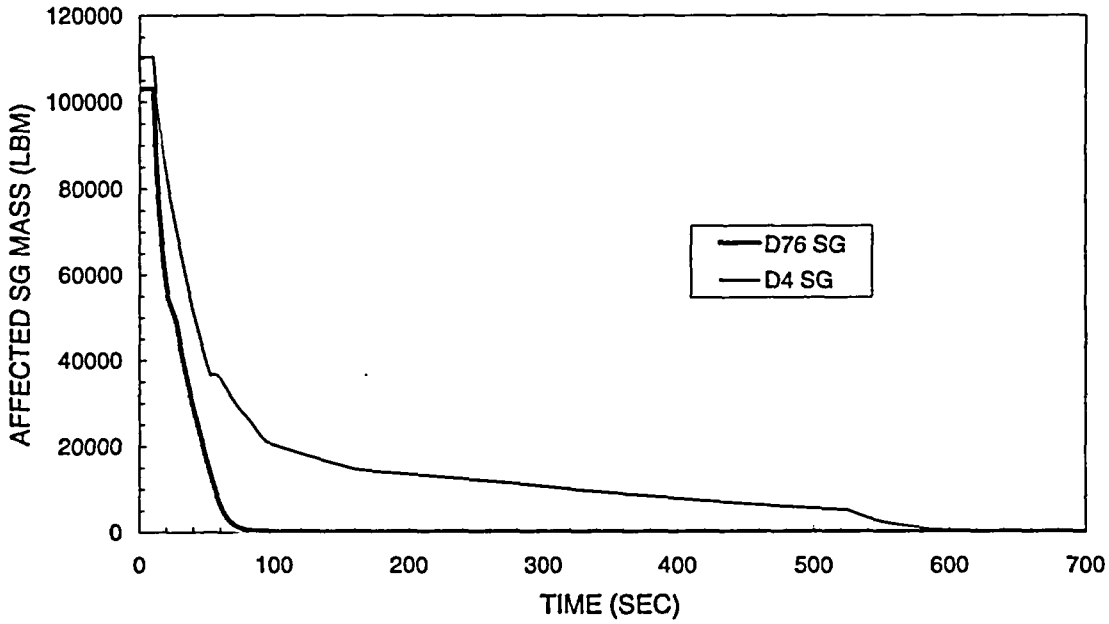

Figure 15 Feedline Break Analysis - Δ76 and D-4 Comparisons – Affected SG Secondary-Side Mass

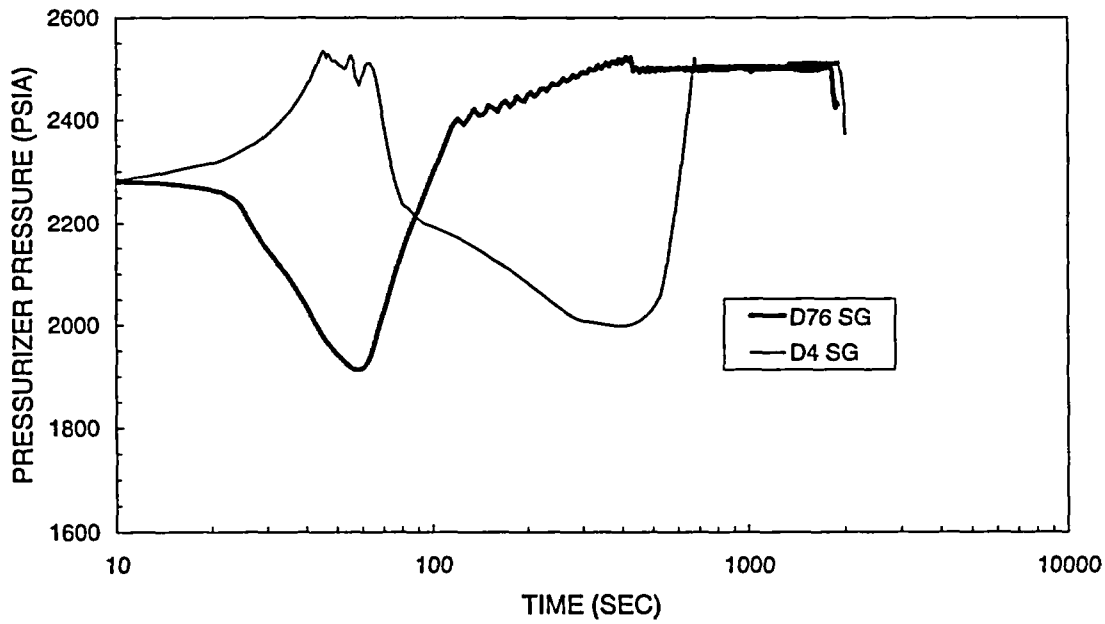

Figure 16 Feedline Break Analysis - Δ76 and D-4 Comparisons – Pressurizer Pressure

CHAPTER 5 - SUMMARY AND CONCLUSIONS

The methodologies used to evaluate the non-LOCA transients and accidents presented in Chapter 15 of the updated FSAR of the CPSES were developed by CPSES accident analysis engineers and approved by the Nuclear Regulatory Commission for this use. These methodologies were developed for use with the current D-4 and D-5 steam generator designs (with integral feedwater preheaters). In accordance with Technical Specification 5.6.5, these methodologies have been applied to over 14 reload core configurations to determine core operating limits such that all applicable limits of the safety analysis are met. CPSES accident analysis engineers have developed extensive experience applying these methods, using both better-estimate and licensing models, as appropriate, to support the design and operations of CPSES.

The purpose of this report is to demonstrate the applicability of the current transient and accident analysis methodologies when used with Δ76 feed ring steam generators rather than the original D-4 preheat steam generator design used in CPSES Unit 1. The effect of a feed ring steam generator design on each of the NRC-approved methodologies for the non-LOCA transients and accidents has been discussed. The methodology application to the analysis of the feedwater system piping break accident is affected to the point that a demonstration analysis of the application of the methodology to the Δ76 steam generator design was considered appropriate. A new analysis, primarily describing the different sequences of events, has been presented; a modified approach to determining the steam generator water level indication was also used. The updated application of the methodology to the Δ76 steam generator design is still considered to be an application of the currently approved methodologies. The underlying philosophies remain unchanged; the methods and modeling techniques are simply applied to a different steam generator design.

CHAPTER 6 – REFERENCES

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3. RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," September 1993.
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5. RXE-91-002-A, "Reactivity Anomaly Events Methodology," October 1993.
6. TXX-88306, "Steam Generator Tube Rupture Analysis," March 15, 1988.
7. RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.
8. RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994.
9. WCAP 14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors Non-LOCA Safety Analyses."