August 29, 1995

APPLICANT: Westinghouse Electric Corporation

FACILITY: AP600

SUBJECT: SUMMARY OF MEETING TO DISCUSS LEAK-BEFORE-BREAK (LBB) FOR THE AP600 DESIGN

On February 14 and 15, 1995, representatives of the Nuclear Regulatory Commission and Westinghouse Electric Corporation met to discuss issues concerning LBB for the AP600 design. Attachment 1 is a list of attendees. Attachment 2 is a copy of the presentation made by Westinghouse.

Westinghouse discussed the leak detection capability of the AP600, LBB methodology that it intended to employ, and methods and results of its pipe rupture evaluation for the AP600. Westinghouse indicated that the design of the sump, containment atmosphere radiation detection capability, ability to measure inventory balance, and capability to monitor humidity made a leak detection capability of 0.5 gpm in the AP600 design feasible. The staff indicated that Westinghouse had presented good arguments supporting the ability of the AP600 to detect 0.5 gpm leakage, and that it may be able to accept that the sump capability, but it was still concerned with how Westinghouse intended to apply this capability to its LBB evaluations. The staff indicated that it was not concerned with large bore piping, but it was concerned with the application of the LBB methodology to small bore piping.

Westinghouse then discussed its responses to specific issues that have been raised in previous requests for additional information and meetings.

Original signed by
Thomas J. Kenyon, Project Manager
Standardization Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 52-003

Attachments:
As stated

cc w/attachments:
See next page

DISTRIBUTION:
See next page

DOCUMENT NAME: A: LBB.LTE

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:DRPM:PDST		SC:DRPM:PDST	E						
NAME	TKenyon:sg		RArchitzel							
DATE	08/28/95		08/29/95							

060080

DF03

cc: Mr. Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Analysis
Nuclear and Advanced Technology Division
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230

Mr. B. A. McIntyre
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
Box 355
Pittsburgh, PA 15230

Mr. John C. Butler
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
Box 355
Pittsburgh, PA 15230

Mr. M. D. Beaumont
Nuclear and Advanced Technology Division
Westinghouse Electric Corporation
One Montrose Metro
11921 Rockville Pike
Suite 350
Rockville, MD 20852

Mr. Sterling Franks
U.S. Department of Energy
NE-42
Washington, DC 20585

Mr. S. M. Modro
EG&G Idaho Inc.
Post Office Box 1625
Idaho Falls, ID 83415

Mr. Charles Thompson, Nuclear Engineer
AP600 Certification
U.S. Department of Energy
NE-451
Washington, DC 20585

Mr. Frank A. Ross
U.S. Department of Energy, NE-42
Office of LWR Safety and Technology
19901 Germantown Road
Germantown, MD 20874

Mr. Ronald Simard, Director
Advanced Reactor Program
Nuclear Energy Institute
1776 Eye Street, N.W.
Suite 300
Washington, DC 20006-3706

STS, Inc.
Ms. Lynn Connor
Suite 610
3 Metro Center
Bethesda, MD 20814

Mr. James E. Quinn, Projects Manager
LMR and SBWR Programs
GE Nuclear Energy
175 Curtner Avenue, M/C 165
San Jose, CA 95125

Mr. John E. Leatherman, Manager
SBWR Design Certification
GE Nuclear Energy, M/C 781
San Jose, CA 95125

Barton Z. Cowan, Esq.
Eckert Seamans Cherin & Mellott
600 Grant Street 42nd Floor
Pittsburgh, PA 15219

Mr. Ed Rodwell, Manager
PWR Design Certification
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

LEAK-BEFORE-BREAK MEETING
ATTENDANCE SHEET
FEBRUARY 14, 1995

NRC	ORGANIZATION
T. KENYON	NRR/PDST
S. HOU	NRR/ECGB
D. TERAO	NRR/ECGB
J. TSAO	NRR/EMCB
W. PAUL CHEN	ETEC
H. L. BRAMMER	NRR/ECGB
P. R. MANDAVA	WESTINGHOUSE
D. C. BHOWMICK	WESTINGHOUSE
D. A. LINDGREN	WESTINGHOUSE
E. R. JOHNSON	WESTINGHOUSE
S. A. SWAMY	WESTINGHOUSE
J. A. SCHOTT	BECHTEL POWER CORP.
G. PICCINI	ALWR/EPRI
A. H. ARASTU	BECHTEL POWER CORP.
P. WARDMAN	NUCLEAR ELEC/WESTINGHOUSE
G. BAGCHI	NRR/DE
R. YOUNG	NRR/SPLB
B. LEFAVE	NRR/SPLB

FEBRUARY 15, 1995

NRC	ORGANIZATION
J. H. WILSON	NRR/PDST
S. SWAMY	WESTINGHOUSE
D. BHOWMICK	WESTINGHOUSE
P. R. MANDAVA	WESTINGHOUSE
D. A. LINGREN	WESTINGHOUSE
E. R. JOHNSON	WESTINGHOUSE
H. L. BRAMMER	NRR/ECGB
S. HOU	NRR/ECGB
W. PAUL CHEN	ETEC
G. BAGCHI	NRR/ECGB
D. TERAO	NRR/ECGB
R. YOUNG	NRR/SPLB
R. ARCHITZEL	NRR/PDST
J. TSAO	NRR/EMCB

Attachment 1

PRESENTATION TO THE

NUCLEAR REGULATORY COMMISSION

AP600 Leak-before-break and Pipe Rupture Issues

NRC Offices, Rockville, MD

February 14 and 15, 1995

Attachment 2

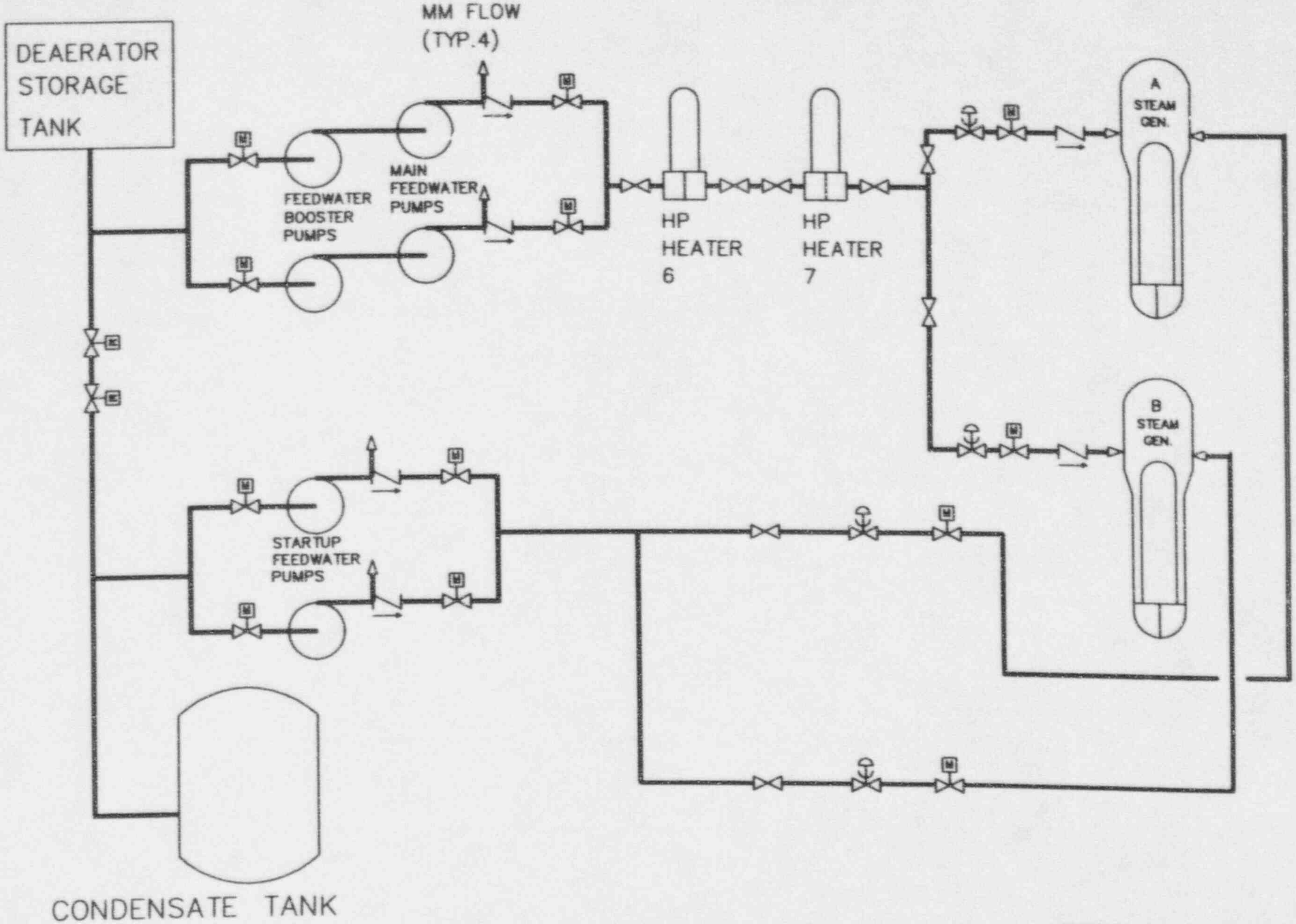
AP600

Agenda
AP600 Leak-before-break and Pipe Rupture Issues

Topic	Presenter
Leak Detection for LBB	Schott
Water Hammer Loads for Feedwater Pipe	Nrastu
LBB Methods	Bhowmick
LBB evaluation for Applicability and Results	Bhowmick
Pipe Rupture Methods	Johnson
Building Design for Pipe Break	Lindgren
Pipe Rupture Hazard Evaluation	Lindgren

Topic	Presenter	Item	DSER
Leak Detection for LBB	Schott	617	3.6.3.6-3
		908	5.2.5.3-1
		909	5.2.5.3-2
		913	5.2.5.3-6
Water Hammer Loads for Feedwater Pipe	Nrastu	614	3.6.3.5-5
LBB Methods	Bhowmick	608	3.6.3.4-1
		609	3.6.3.4-2
		611	3.6.3.5-2
		612	3.6.3.5-3
		613	3.6.3.5-4
		615	3.6.3.6-1
		617	3.6.3.6-3
		618	3.6.3.6-4
		619	3.6.3.6-5
LBB evaluation for Applicability and Results	Bhowmick	614	3.6.3.5-5
		616	3.6.3.6-2
		620	3.6.3.6-6
Pipe Rupture Methods	Johnson	123	3.6.1
		124	3.6.1
		132	3.6.1
		593	3.6.2.1-1
		598	3.6.2.3-4
		607	3.6.2.3-8
Building Design for Pipe Break	Lindgren	128	3.6.1
		592	3.6.2-1
		594	3.6.2.2-1
		599	3.6.2.3-5
		985	6.2.1.2-1
		987	6.2.1.2-3
Pipe Rupture Hazard Evaluation	Lindgren	125	3.6.1
		126	2.6.1
		127	3.6.1
		131	3,6,1
		595	3.6.2.3-1
		596	3.6.2.3-2
		597	3.6.2.3-3

FEEDWATER SYSTEM



[redacted]

Meeting with Westinghouse on Leak Detection Issues
Concerns Expressed By the Staff

- Seismically qualified leak detection monitors should have Class 1E source of power.

- Burden of proof on Westinghouse to make case for using a leakage limit of 0.5 gpm versus 1.0 gpm.

- Westinghouse's draft revision of SSAR Section 5.2.5 should include a discussion the new detection technology (e.g., use of N-13 and F-18 monitors) and any other additions identified in their presentation; draft will be submitted for staff review and comment prior to issuance of an updated SSAR. (Refer to AP 600 Open Item M5.2.5-7 in the Westinghouse database tracking system).

- Westinghouse should provide more details for calibration of leakage detection monitors.

- Effect of using fewer snubbers on piping with a resulting increase in piping seismic loading, and reducing LBB acceptance criteria to accommodate the increased loading; this appears to circumvent the safety issues.

- Lowering LBB acceptance criteria which allows the qualification of smaller diameter piping.

- The staff would like specific examples of locations where pipe seismic loading is a problem in the AP600 design.

- Westinghouse should consider alternatives such as using jet shields or re-routing pipes prior to proposing a reduction in the LBB leakage (0.5 gpm versus 1.0 gpm).

- Westinghouse should provide additional information on differentiation between reactor coolant leakage and feedwater/main steamline leakage inside containment.

- Any change in policy regarding LBB acceptance criteria must be approved by the Commission (use of 0.5 gpm rather than 1.0 gpm).

- Staff recommends that Westinghouse consider safety issues with regard to LBB and state the reasons for requesting that LBB acceptance criteria be lowered, rather than focusing on the capability to detect leakage; staff further suggests that Westinghouse identify piping where 1.0 gpm leakage rate (compared to 0.5 gpm) is not acceptable for LBB; staff emphasized that LBB acceptance criteria impacts several plant design parameters (i.e., pipe sizing, loading, etc.).

AP600 LEAK-BEFORE-BREAK STATUS

Number of LBB analysis packages	16
Primary Loop	
Main steam -B	
Main steam -A	
Main feedwater-B	
Main feedwater-A	
ADS Stage 1, 2, 3/Safety	
Normal RHR	
PRHR Return	
PRHR Supply/ADS 4	
ADS 4 East	
Direct vessel inj-B	
Direct vessel inj-A	
CMT-B	
CMT-A	
Spray	
Number with incompleted pipe layout	7
Main feedwater-B	
Main feedwater-A	
ADS Stage 1, 2, 3/Safety	
Direct vessel inj-B	
Direct vessel inj-A	
Spray	
PRHR Return	
PRHR Supply/ADS 4	
Number with completed preliminary pipe stress	7
Primary Loop	
Main steam -B	
Main feedwater-A	
Surgeline	
Normal RHR	
PRHR Return	
Direct vessel inj-B	
Number with completed LBB	5
Primary Loop	
Surgeline	
PRHR Return	
Main steam -B	
Main feedwater-A	

AP600 LEAK-BEFORE-BREAK STATUS

Westinghouse uses 0.5 gpm leak detection capability for primary and secondary system piping inside containment. We satisfy the NRC's break exclusion zone guidelines outside containment for main steam and feedwater piping.

Westinghouse reviewed unanticipated water hammer events for the main feedwater lines. These events are bounded by the anticipated water hammer due to isolation valve closure.

Westinghouse uses sample approach and preliminary pipe stress analysis. We do not use bounding LBB analysis.

All lines, except the primary loop, are being evaluated with non-conservative response spectra based on 7% damping for interior concrete building. This will be addressed in the intermediate design stage.

Westinghouse uses proprietary method for calculating leakage rates. NRC has accepted on previous applications.

Westinghouse is evaluating the OSU water hammer event.

A portion of the preliminary pipe stress analysis and LBB evaluation is based on superseded P&IDs. This will be addressed in the intermediate design stage.

PRESENTATION

TO

THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

NRC LICENSING OFFICES (ROCKVILLE, MD)

FEBRUARY 14, 1995

LEAK DETECTION

- INTRODUCTION
- DESIGN BASES
- LEAK DETECTION DESCRIPTION
- LEAK DETECTION METHODS

LEAK DETECTION

INTRODUCTION

- Purpose
  - Describe the leak detection components that are used to detect unidentifed leakages inside containment
  - Show that the leak detection components are designed to quantify and identify leakages inside contaiment of 0.5 gpm within one hour
  - Show that the leak detection components can differentiate between reactor coolant leakage and feedwater/main steam line leakage inside containment

LEAK DETECTION

DESIGN BASES

- Designed in accordance with GDC 30
    - Provides detection of reactor coolant leakage
    - Assists in identifying the location of the source of reactor coolant leakage
- Conforms to the recommendations of RG 1.45
    - Leakage sources are identifiable to the extent practical
    - Unidentified leakage is collected by the containment sump and measured with an accuracy of 0.5 gpm or better
    - Three diverse detection methods are employed
    - Intersystem leakage is monitored
    - Unidentified leakage component sensitivities and response times are adequate to detect a leakage rate of 0.5 gpm within one hour
    - The leak detection components are operational following seismic events that do no require plant shutdown; the containment atmosphere radiation monitor and containment sump level monitor will function following an SSE

LEAK DETECTION

DESIGN BASES (CONT'D.)

- Conforms to the recommendations of RG 1.45 (cont'd.)

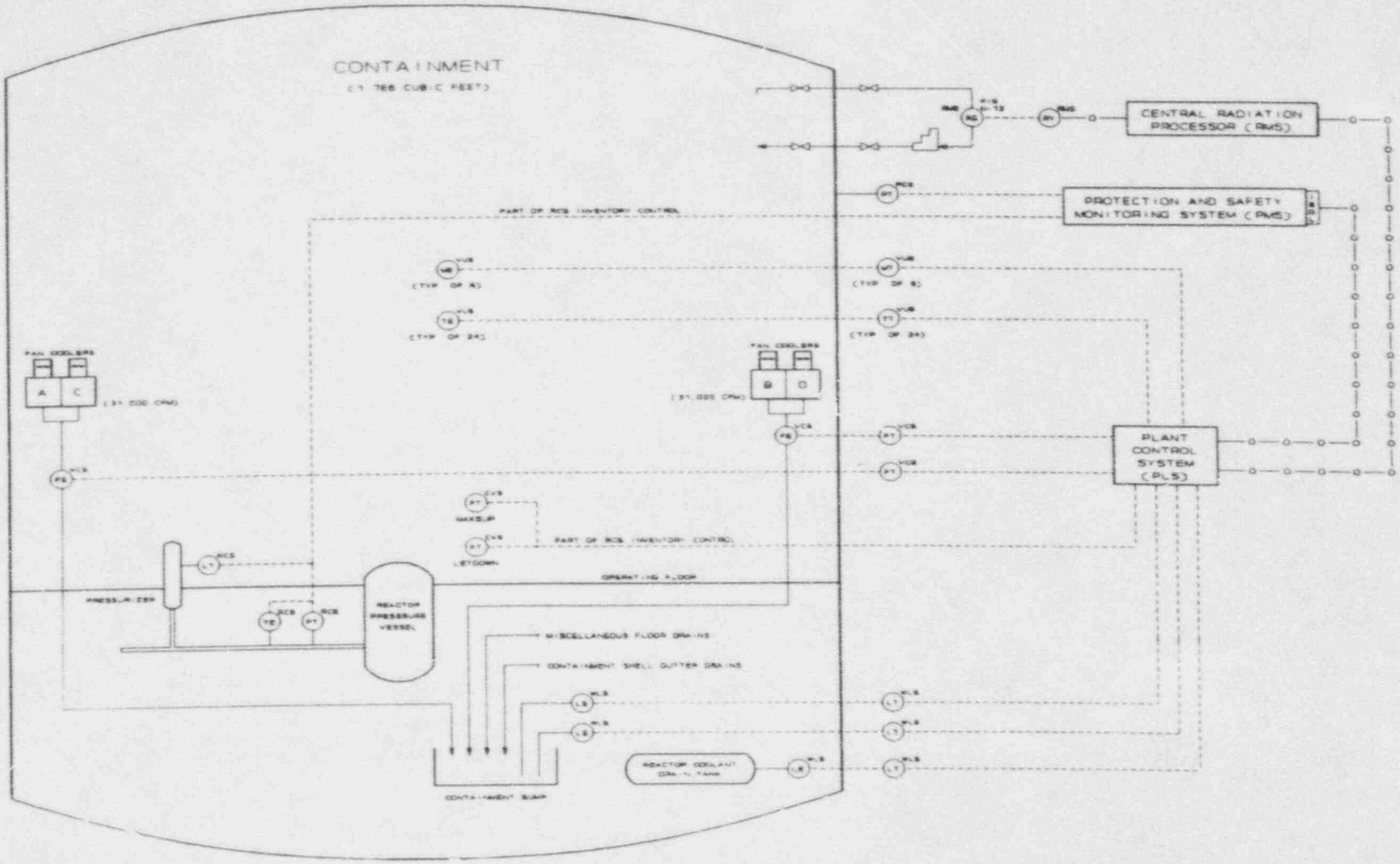
  - Indications and alarms are provided in the MCR; indications are displayed in equivalent leakage rates (gpm)
  - The leak detection system can be calibrated and tested for operability during plant operation
  - SSAR Chapter 16 defines the limiting conditions and operability requirements for reactor coolant pressure boundary leak detection

- AP600 Plant Design (Leak Before Break)

  - Quantify secondary leakage of better than 0.5 gpm inside containment
  - SSAR Chapter 16 defines the limiting conditions and operability requirements for feedwater and main steam line leak detection inside containment

LEAK DETECTION

LEAK DETECTION DESCRIPTION



LEAK DETECTION

LEAK DETECTION DESCRIPTION (CONT'D.)

- Comprised of process sensors from the following systems:
  - Passive Containment Cooling System (PCS)
    - Containment pressure
  - Reactor coolant system inventory control
    - Reactor Coolant System (RCS)
      - RCS pressure
      - RCS temperature
      - Pressurizer level
    - Chemical and Volume Control System (CVS)
      - Makeup flow
      - Letdown flow
  - Radiation Monitoring System (RMS)
    - Containment atmosphere radiation
  - Containment Recirculation Cooling System (VCS)
    - Containment air cooler condensate flow

LEAK DETECTION

LEAK DETECTION DESCRIPTION (CONT'D.)

- Containment Leak Rate Test System (VUS)
    - Containment atmosphere humidity
    - Containment atmosphere temperature

- Liquid Radwaste System (WLS)
    - Containment sump level
    - Reactor coolant drain level

- Plant Control System (PLS) converts process sensor inputs; the leakage rate in gpm is available at the MCR workstation

- PLS interfaces with the following plant computer systems:

    - Protection and Safety Monitoring System (PMS)
        - Containment pressure
        - Reactor coolant system pressure and temperature
        - Pressurizer level

    - RMS Central Radiation Processor (CRP)
        - Containment atmosphere radiation

LEAK DETECTION

LEAK DETECTION METHODS

Leak quantification methods

- Containment sump level
    - WLS level sensors
    - Seismic Category I (non-class 1E)
    - Measures continuous changes in sump level
    - Sensitivity: 0.5 gpm (DB); ≤ 0.4 gpm (vendor)
    - Response time: ≤ 1 hour (DB); ≤ 10 minutes (vendor)

- Containment atmosphere radiation
    - RMS radiation monitor
    - Seismic Category I (non-class 1E)
    - Focuses on core produced radioisotope combination of N-13 and F-18
    - Sensitivity applicable above 20% reactor power
    - Sensitivity: 0.5 gpm (DB); ≤ 0.4 gpm (vendor)
    - Response time: ≤ 1 hour (DB); ≤ 1 hour (vendor)

LEAK DETECTION

LEAK DETECTION METHODS (CONT'D.)

Leak quantification methods (cont'd.)

- Reactor coolant system inventory balance
  - PLS calculation
  - Correlation of RCS pressure and temperature, pressurizer level, and CVS makeup and letdown flow rates
  - Sensitivity: 0.5 gpm (DB); ≤ 0.2 gpm (actual)
  - Response time: ≤ 1 hour (DB); real time (actual)

Other leak quantification methods

- Containment air cooler condensate flow
  - VCS flow sensors
  - Sensitivity: 0.2 gpm (vendor)
  - Response time: ≤ 1 minute (vendor)

LEAK DETECTION

LEAK DETECTION METHODS (CONT'D.)

Other leak quantification methods (cont'd.)

- Containment atmosphere humidity
    - VUS dewpoint monitors

PRESENTATION

TO

UNITED STATES

NUCLEAR REGULATORY COMMISSION

NRC LICENSING OFFICES

FEBRUARY 14, 1995

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

TOPICS OF DISCUSSION

- Types of Water Hammer
- System Description
- Water Hammer Mechanisms
- Past Experience on Feedwater Systems
- Review of FWS for Water Hammer Susceptibility
- Conclusions

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

TYPES OF WATER HAMMER

Anticipated Water Hammer

- Imposed by System Requirements
- System Design Must Accommodate the Water Hammer Loads

Unanticipated Water Hammer

- Not Imposed by System Requirements
- In General, It Should be Prevented or Mitigated

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

SYSTEM DESCRIPTION

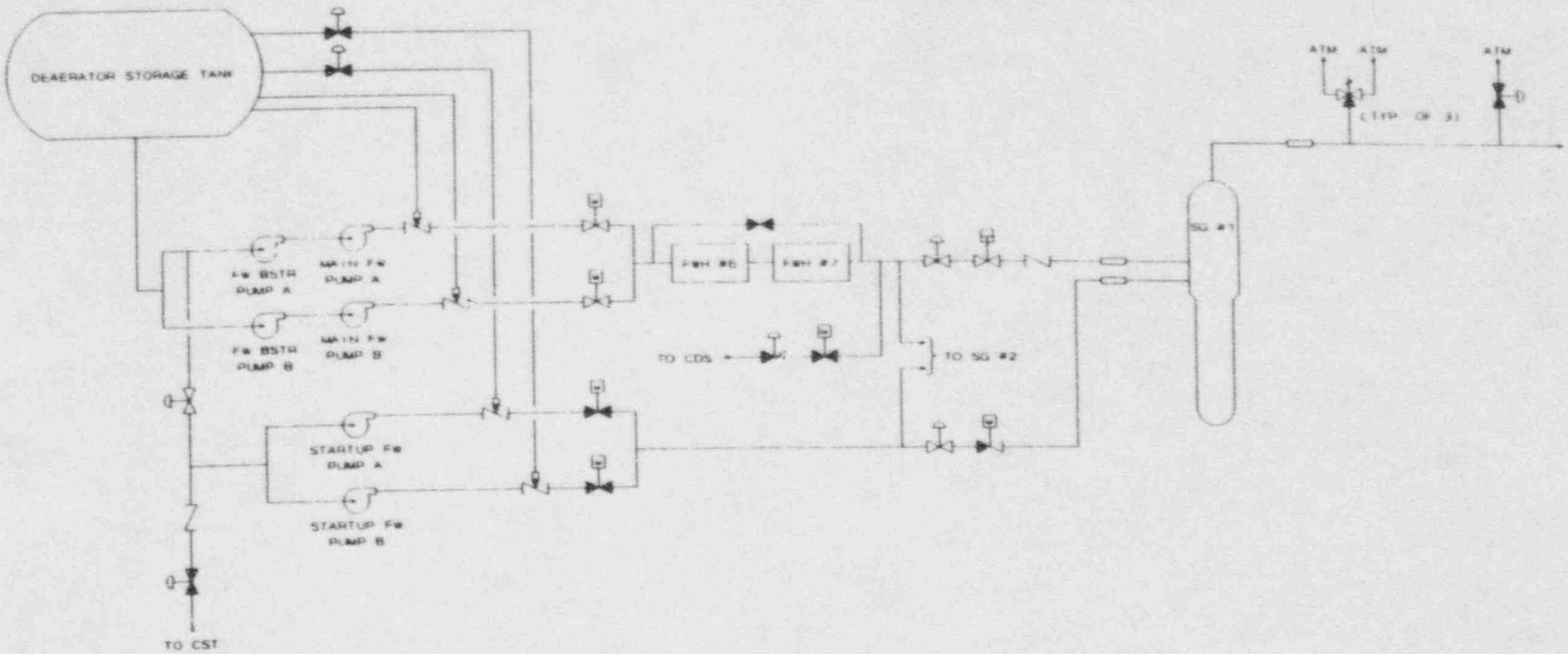
- Condensate System (CDS)
- Main Feedwater System (FWS)
- Startup Feedwater System
- Role of Deaerator in Water Hammer Evaluation

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

DEAERATOR STORAGE TANK

- Large Tank Between Condenser and Main Feed Pumps
- Prevents Pressure Wave Propogation across it
- Transients in CDS Area will not affect LBB Piping

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY



SYSTEM CONFIGURATION

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

EPRI WATER HAMMER RESEARCH PROJECT 2856-03

- Multi-year Project by Stone & Webster (SWEC), Bechtel, Northeast Utilities, and Experts in Water Hammer from Several Universities

- Major Tasks Performed by Bechtel and SWEC

    - Compile & Classify Plant Water Hammer Experience
    - Perform Root Cause Analysis for Water Hammer
    - Compile Experimental Data on Water Hammer
    - Review Analytical Models and Computer Codes
    - Develop Guidelines to Prevent, Diagnose, and Assess Water Hammer for Utility Engineers
    - Review of Plant Systems and Procedures
    - Develop Training Modules

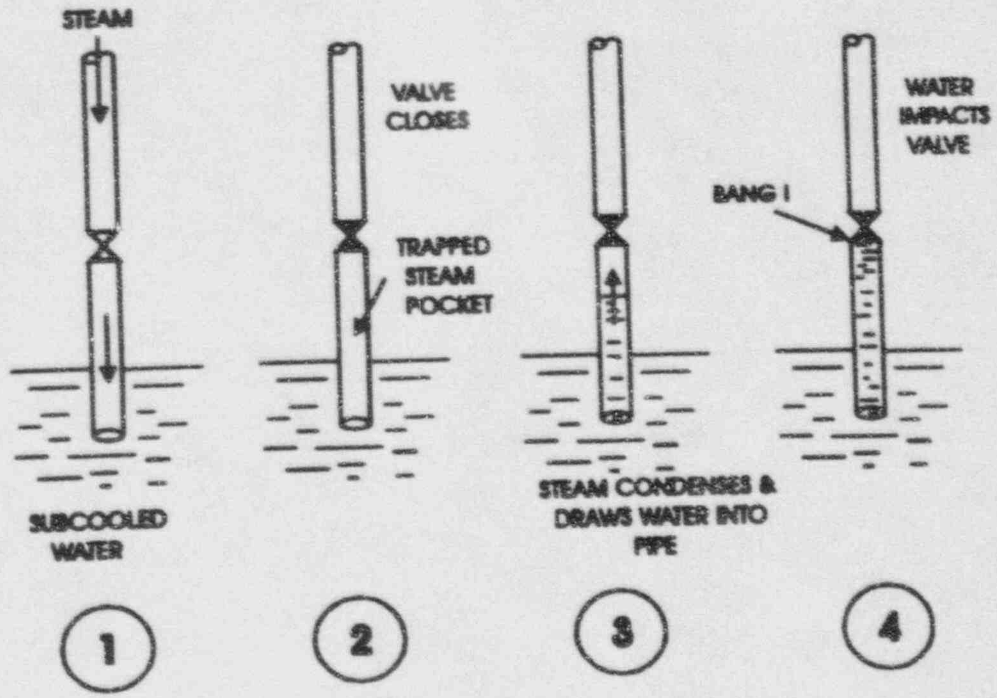
LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER MECHANISMS FROM EPRI STUDY

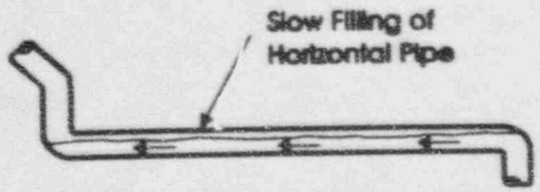
- Mechanism 1: Water Cannon
- Mechanism 2: Steam/Water Counterflow
- Mechanism 3: Pressurized Water Entering a Steam Filled Pipe
- Mechanism 4: Hot Water Entering a Low Pressure Line
- Mechanism 5: Steam Propelled Water Slug
- Mechanism 6: Rapid Valve Actuation
- Mechanism 7: Water Column Separation & rejoining Line

LBB EVALUATION OF THE FEEDWATER LINES - WATERHAMMER SUSCEPTIBILITY

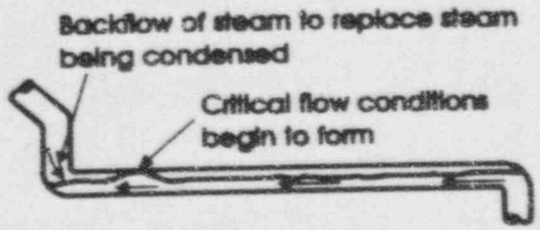


MECHANISM 1: WATER CANNON

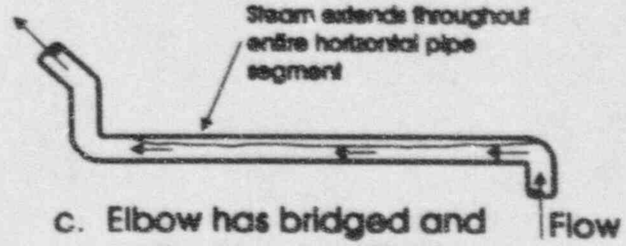
LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY



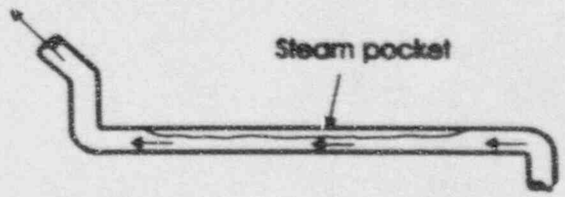
a. Pipe 90% full when cold water flow is stopped



b. Wave instabilities being formed prior to bridging of the vertical pipe



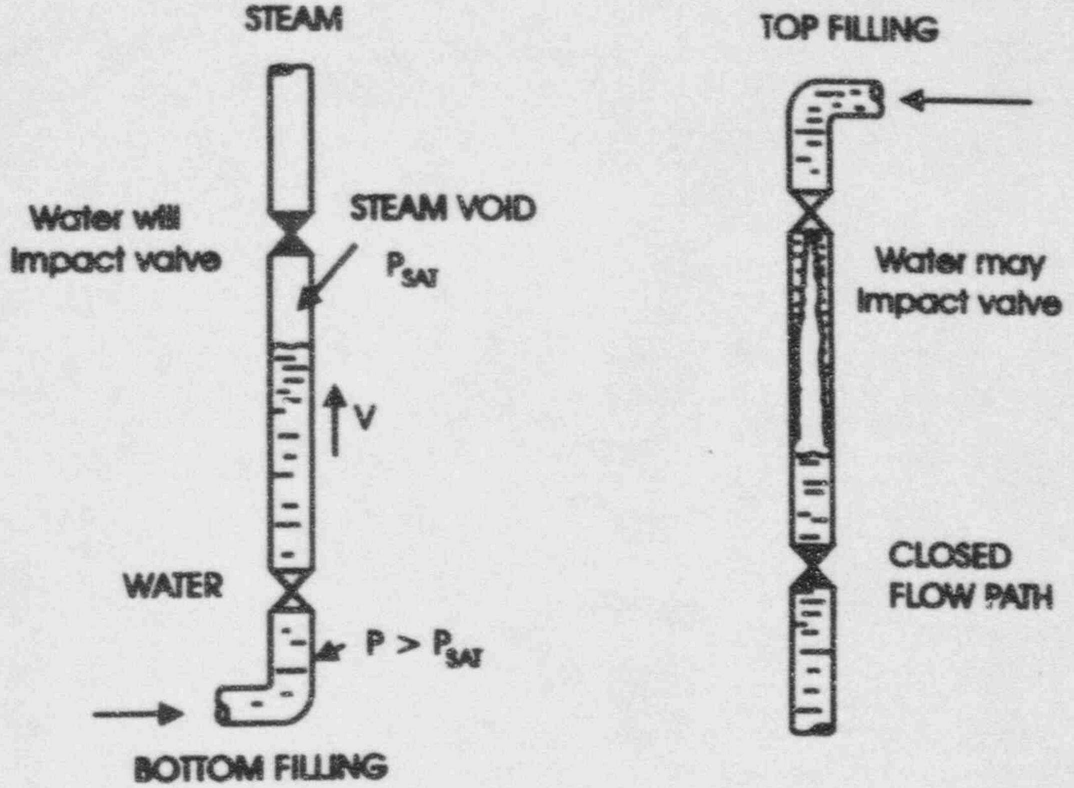
c. Elbow has bridged and vertical leg has filled



d. Probable steam pocket location just prior to water hammer

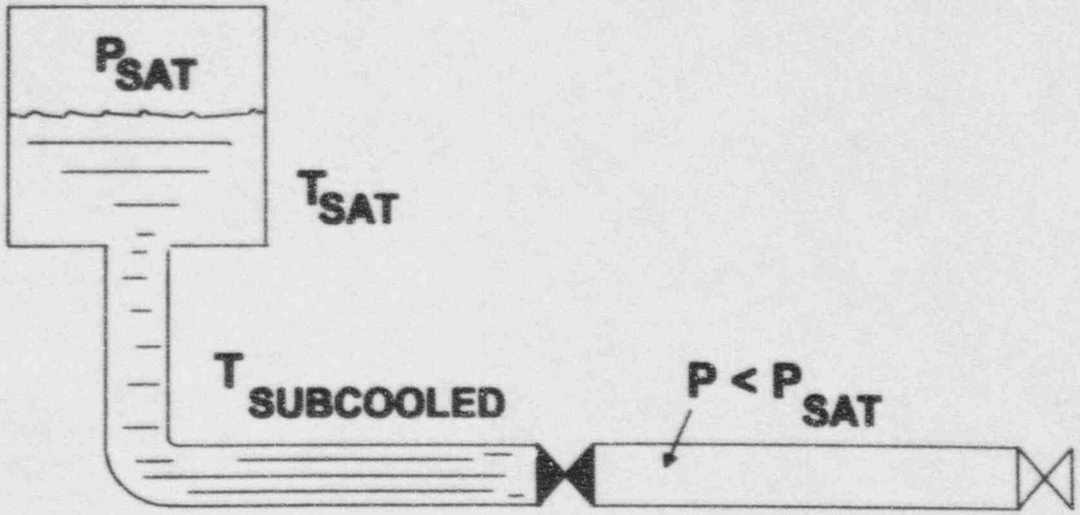
MECHANISM 2: STEAM/WATER COUNTERFLOW

LBB EVALUATION OF THE FEEDWATER LINES - WATERHAMMER SUSCEPTIBILITY



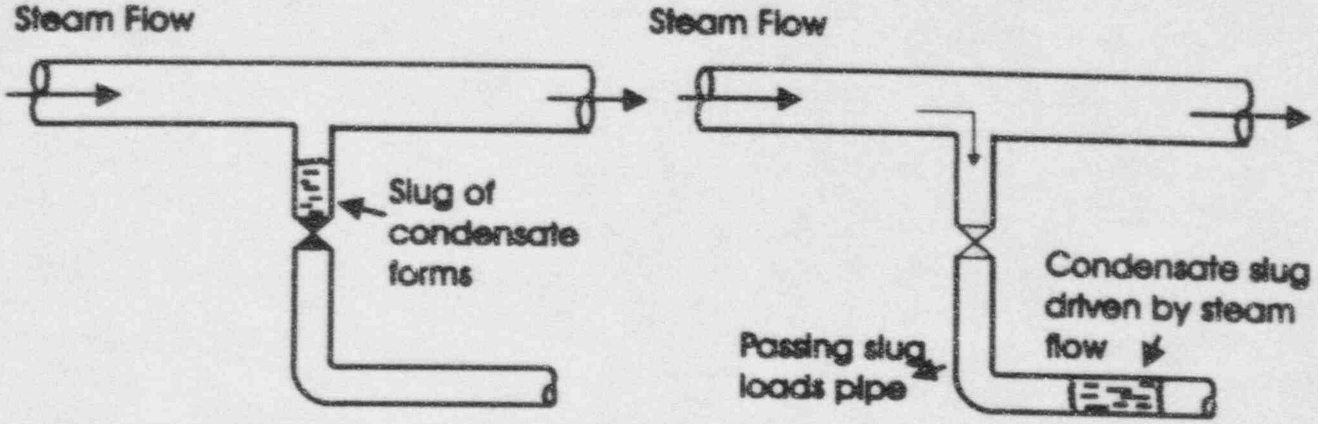
MECHANISM 3: PRESSURIZED WATER ENTERIING A STEAM FILLED PIPE

LBB EVALUATION OF THE FEEDWATER LINES - WATERHAMMER SUSCEPTIBILITY



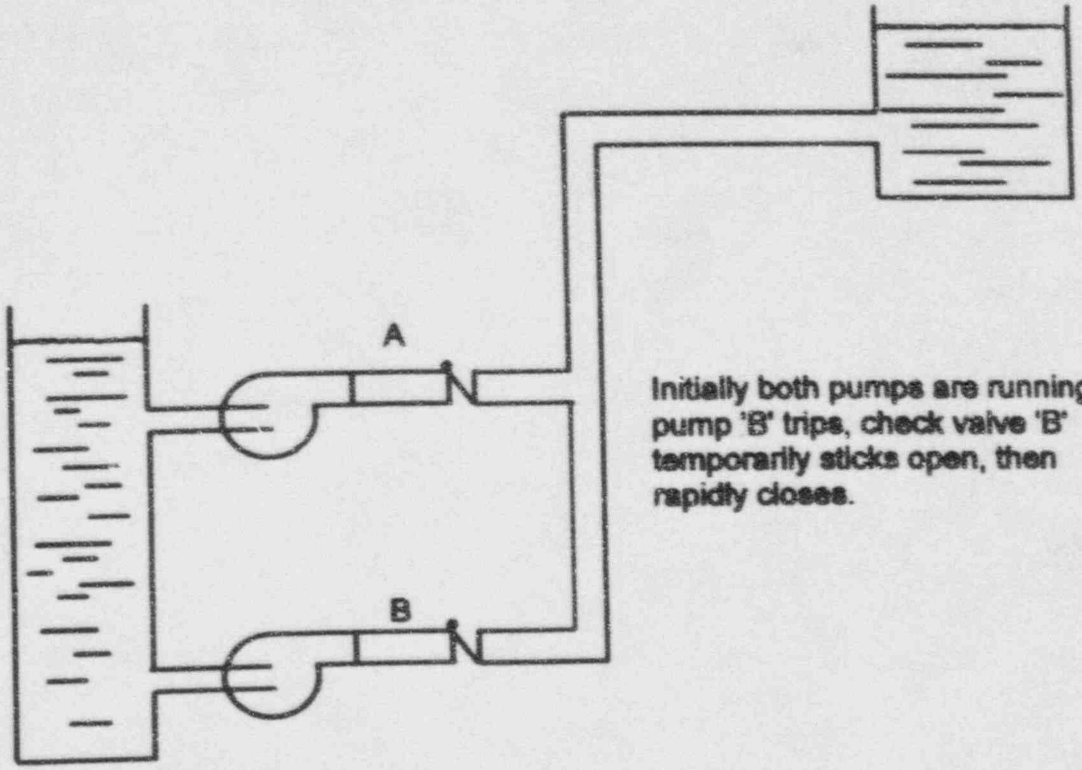
MECHANISM 4: HOT WATER ENTERING A LOW PRESSURE LINE

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY



MECHANISM 5: STEAM PROPELLED WATER SLUG

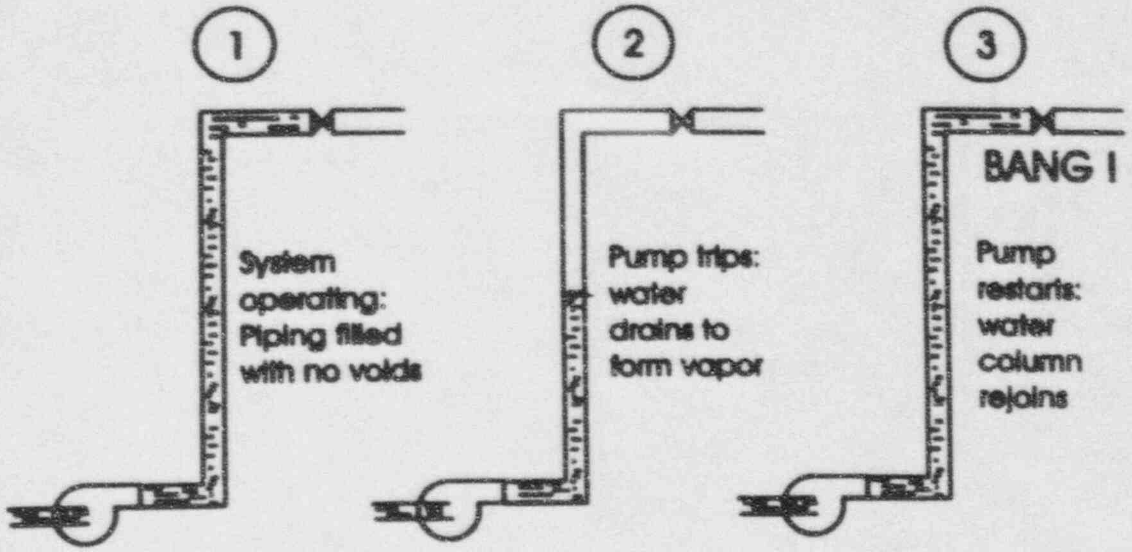
LBB EVALUATION OF THE FEEDWATER LINES - WATERHAMMER SUSCEPTIBILITY



Initially both pumps are running, pump 'B' trips, check valve 'B' temporarily sticks open, then rapidly closes.

MECHANISM 6: RAPID VALVE ACTUATION

LBB EVALUATION OF THE FEEDWATER LINES - WATERHAMMER SUSCEPTIBILITY



MECHANISM 7: WATER COLUMN SEPARATION AND REJOINING

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

PAST EXPERIENCE

Events that have Occurred in FWS

- Steam Generator Water Hammer
- Control Valve Instability Water Hammer

Other One of a Kind Events

- San Onofre Event (Mechanism 2)
- Trojan Event (Mechanism 2)
- Rancho Seco Event (Mechanism 3)

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION

Steam Generator Water Hammer

- Startup Feedwater Direct to Steam Generators
- Use of Spray-Tubes
- Heated Startup Feedwater
- Short Horizontal FW Piping at SG Nozzle
- Sloped FW Piping near the SG Nozzle to Prevent Vapor Accumulation

Control Valve Instability

- Variable Speed Design Main Feed Pumps Perform Part of the Flow Control Function
- Refined 3 Element Valve Control Scheme
- Globe Valve with a Stacked Disk Trim for Finer Control Over the Entire Stroke of Valve

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION

San Onofre Event

- No Source of Cold Water to Cause Void Collapse - Startup Feedwater Direct to Steam Generators

Trojan Event and Rancho Seco Events

- No Thermal Sleeve - Welded Connection Between FW Pipe and Feedring
- No Low Energy Line Interfacing with the High Energy FW Line that Transmits Pressure Waves to LBB Piping

⇒ *Conclusion: No Water Hammer Effects in LBB Piping by the Types of Events that have Occurred in Past*

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

MODES OF OPERATION RELEVANT TO WATER HAMMER

- System Startup
- Normal System Operation
- Normal System Shutdown
- Trip of Both Main Feedpumps - Check Valve Slam
- Loss of Offsite Power (LOOP) without Check Valve Leakage
- Loss of Offsite Power (LOOP) with Check Valve Leakage
- Feedwater Line Depressurization due to a Secondary Side Depressurization

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION (CONT'D)

System Startup

- Main and Startup FWS Started in a Controlled Manner
- At Power Level 2.4% Flow Switched Slowly from Startup to Main Feed Pumps Over 5 Minutes
- Heated Startup Feedwater Direct to Steam Generator

⇒ *No Significant Water Hammer during System Startup*

Normal System Operation

- No High/Low Energy System Interfaces from Where Vapor Bubble Collapse affects LBB Piping
- Stable Flow Control - No Control Valve Instability

⇒ *No Significant Water Hammer during Normal System Operation*

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION (CONT'D)

Normal System Shutdown

- Water Hammer in Deaerator type Designs - Fossil Plants
- Main and Startup Feedpump Suction Lines Connected
- Problem Prevented in AP600 Design by Starting the Startup Pumps Before Main Feed Flow is Terminated
- If Water Hammer Does Occur, the Closed Isolation Valve Will Prevent Wave Propogation to the LBB Piping

⇒ ***No Significant Water Hammer Effects in LBB Piping during Normal System Operation***

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION (CONT'D)

Trip of Both Feed Pumps

- Long Coast Down Time > 90 Seconds
- Recirculation Check Valves Close Slowly When Nearly Closed
- No Reported Water Hammer Problems due to Check Valve Slam

⇒ ***No Significant Water Hammer during Normal System Operation***

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION (CONT'D)

LOOP as Designed

- Rapid Power Reduction
- Main Feed Pumps Continue to Operate

⇒ *No Significant Water Hammer during LOOP as Designed*

LOOP with Turbine Trip But No Check Valve Leakage

- Check Valves Stop Any Steam Flow Ingress into FW Line
- No Significant Water Hammer due to Closing of the Slow Closing Type of Check Valves
- Pressure in FW Line Higher than Saturation - No Voiding

⇒ *No Significant Water Hammer during this Transient*

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION (CONT'D)

LOOP with Turbine Trip if Check Valves Leak

- Turbine Trip Occurs
- Main Feed Pumps Trip
- Large Check Valve Leakage Assumed
- Isolation Valve Closure Upon Low Steam Line Pressure
- Some Steam Ingress Into FW Line
- Void Collapse by Hot Water From Startup Pumps Via S/G

⇒ *No Significant Water Hammer Water Hammer Expected for this Very Low Probability Event*

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

WATER HAMMER SUSCEPTIBILITY EVALUATION (CONT'D)

Steam Generator Depressurization

- Stuck Open Safety Valve or PORV
- Isolation Valve Closes
- Voiding in FW Line
- Filling of the Void by SU pump Flow Via S/G
- SU Flow < 1000 gpm

⇒ *Water Hammer Loads for this Very Low Probability Event Bounded by Isolation Valve Closure Designed Event (See Anticipated Water Hammer)*

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

ANTICIPATED WATER HAMMER

FW Isolation Valve (Gate Type) Closure

- System Designed for Isolation Valve Closure Loads
- Minimum Isolation Closure Time is 3 Seconds
- "Effective Closure Time" for this Gate Valve is Much Shorter than 3 seconds

LBB EVALUATION OF THE FEEDWATER LINES - WATER HAMMER SUSCEPTIBILITY

AP600

CONCLUSIONS

- FWS System Reviewed for Applicability of the Seven Severe Water Hammer Mechanims
- AP600 System Design Precludes Events that have Occurred in the Past in FWS
- No Scenario Identified that Results in a Significant Unanticipated Water Hammer Event
- System Designed for Isolation Valve Closure Water Hammer whose Loads Bound those from All Other Transients for the LBB Piping

MEETING WITH

THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

ON

"AP600 LBB ANALYSIS"

NRC OFFICES (ROCKVILLE, MD)

FEBRUARY 14-15, 1995

AP600 LBB ANALYSIS

ITEM # 608: **DSER Section: 3.6.3.4-1**

QUESTION: Westinghouse should perform and submit for staff review bounding LBB analyses for candidate piping systems including evaluations for susceptibility to degradation mechanisms for the projected 60-year AP600 design life.

RESPONSE: Westinghouse intends to perform the leak-before-break analysis of designated piping systems to which mechanistic pipe break criteria will apply using the preliminary piping analysis loads. The results of each line will be added in the SSAR. This approach was discussed during the piping audit held in 1994 at Pittsburgh. We would like to have more clarification for this item.

AP600 LBB ANALYSIS

ITEM # 609: **DSER Section: 3.6.3.4-2**

QUESTION: Westinghouse should add COL Action Item 3.6.3.4-1 to the SSAR.

RESPONSE: A table is attached for the COL Action and this will be added in the SSAR.

CONTENT OF AS-BUILT RECONCILIATION REPORT FOR THE COL ACTION

ITEM	USED FOR LBB ANALYSIS AS-DESIGNED CONDITION	AS-BUILT CONDITION	AS-BUILT RECONCILIATION COMMENTS
Material			Same
Material Strength			Same or Higher
Nominal Outer Diameter			Same
Minimum Wall Thickness at Weld Undercut			Same or Higher
Normal Stress at Critical Location			Same or Higher
Maximum Faulted Stress at Critical Location			Same or Lower
Welding Process	GTAW		Same
Leak Detection Capability	0.50 GPM		Same
Cast Steel	None		Same

AP600 LBB ANALYSIS

ITEM # 611: **DSER Section: 3.6.3.5-2**

QUESTION: Westinghouse should provide additional discussion concerning the differences in analysis, fabrication, and inspection between Class 1 and 2 systems.

RESPONSE: LBB Analysis and Criteria for Class 2 and 3 systems are same as that of Class 1 systems.

A Fatigue Crack Growth Analysis will be performed on Class 2 and 3 system on which LBB is to be demonstrated. This along with the preservice inspection and Section XI required in service inspection will provide for the integrity of each system.

Class 1, 2 and 3 systems are all subject to regular inservice inspection requirements from ASME Section XI. For Class 1 piping, all terminal ends and dissimilar metal welds must be volumetrically inspected, along with other locations to total 25 percent of the welds. For Class 2 piping, the requirement is to volumetrically inspect all terminal ends and other locations to total 7.5 percent of the welds. For Class 3 systems, the entire system receives periodic visual examinations. These requirements were developed by Section XI consistent with the different safety classes of these systems.

AP600 LBB ANALYSIS

ITEM # 614: **DSER Section: 3.6.3.5-5**

QUESTION: Westinghouse should provide in the SSAR, more detailed discussions with sufficient information to support the inclusion that the MS and FW piping systems do not fall within the limitations delineated in Section 5.1 of Volume 3 of NUREG-1061.

RESPONSE: Detailed discussion will be provided in the SSAR to support the inclusion of the main steam and feedwater systems in LBB Analysis.

A discussion is also provided below:

Stress Corrosion Cracking

A. Mainsteam Lines

Stress corrosion cracking (SCC) is the embrittlement of a material in the form of delayed fracture under static load caused by the combined action of environmental and tensile stresses (residual or applied). Essentially the three factors which contribute to SCC are a susceptible material, a corrosive environment and stress. To mitigate SCC one of the three factors must be eliminated.

The mainsteam piping is constructed from ferritic steel. SCC in ferritic steels commonly results from a caustic environment. A possible source of caustic in the mainsteam piping would be carried over from the steam generator. However, the current secondary side water treatment utilizes All Volatile Treatment (AVT). AVT effectively precludes caustic in the steam generator bulk liquid environment.

Prior to implementing AVT, the phosphate water treatment caused an imbalance of caustic resulting in SCC of steam generator tubing. Under these conditions there was no instance of caustic SCC on the ferritic steam lines indicating no significant caustic carryover.

In summary, the operating secondary side chemistry precludes SCC on the ferritic (SA333 Grade 6) mainsteam lines.

B. Feedwater Lines

Feedwater pipe material which is an alloy steel (SA335 P11) is significantly more resistant to stress corrosion cracking than the carbon steel.

Erosion-Corrosion

A. Main Steam Lines

Erosion-corrosion induced wall thinning is not expected in the mainsteam line. Extensive work has been done for investigating erosion-corrosion in carbon steel pipes, particularly single phase systems. The mainsteam line has very low susceptibility to erosion due to the relatively high operating temperature (532°F) where maximum susceptibility is expected at a lower temperature than 532°F. Susceptibility is also less due to the low erosion potential of the high quality steam in the mainsteam line.

In summary, wall thinning resulting from erosion-corrosion is not a significant affect in the mainsteam line.

B. Feedwater Lines

The erosion-corrosion impact on the AP600 feedwater line is reduced by using the Alloy Steel (SA335 P11). The alloy steel was modeled utilizing EPRI's "Checmate" program to determine erosion/corrosion rates based on AP600 chemistry controls. The calculated wear rates provided significant margin for the proposed feedwater line for the 60 year plant life.

Susceptibility of Failure from Brittle Fracture

A. Mainsteam Lines

There is a potential for low temperature brittle fracture for carbon steel in the mainsteam line. However, at the high normal operating temperature (520°F) for the main steam line this is not a matter for concern.

B. Feedwater Lines

For the feedwater line, the operating temperature is also high (435°F) and therefore brittle fracture is not a concern.

Low Cycle and High Cycle Fatigue

A. Main Steam Lines

Based on operating plant specific analysis, it has been demonstrated that no significant fatigue crack growth will occur over the design life of the plant.

High-cycle fatigue loads in the system result primarily from pump vibrations. The steam generator is designed so that flow-induced vibrations in the tubes are avoided. The loads from pump vibrations are minimized by criteria for pump shaft vibrations during hot functional testing and operation. During operation, an alarm signals when the pump vibration is greater than the limits.

With these precautions taken, the likelihood of leakage in the main steam lines due to fatigue is very small.

- Feedwater Lines

In addition to the above discussion for the Feedwater lines by reducing the thermal stratification and striping loads the fatigue crack growth will not be a concern.

WATER HAMMER

Feedwater Lines : This is discussed separately.

AP600 LBB ANALYSIS

AP600

ITEM # 615: **DSER Section: 3.6.3.6-1**

QUESTION: For all LBB candidate piping systems, Westinghouse should use the worst condition of all potential sites within the scope of the AP600 applications.

RESPONSE: The piping analyses used for input to the LBB evaluation uses parameters that includes the range of sites included in the site interface criteria.

AP600 LBB ANALYSIS

ITEM # 617: **DSER Section: 3.6.3.6-3**

QUESTION: Westinghouse should use a 1.0 gpm leakage rate and a margin of 2 on leakage flaw size in the bounding LBB analyses to be presented for staff review.

RESPONSE: Margin of 2.0 on Leakage Flaw Size

NUREG 1061, Volume 3, Section 5.12 stated that:

- "There are various ways in which conservatism can be incorporated and the large margins are not necessary in each step of the process provided that the overall objective is met. The specific margins recommended could be modified provided that equivalent conservatisms are included elsewhere in the LBB approach. It is the task group's opinion that the NRC staff should have the flexibility to use engineering judgements on a case-by-case basis."
- In general, Westinghouse will use 2.0 margin on flaw size. We may request the NRC to review this on a case-by-case basis.

AP600 LBB ANALYSIS

ITEM # 618: **DSER Section: 3.6.3.6-4**

QUESTION: Westinghouse should benchmark its leak rate evaluation methodology against methods currently accepted by the staff, (such as using the PICEP computer code).

RESPONSE: The NRC has approved the Westinghouse LBB analysis methodology and criteria for various applications. Therefore we believe our methodology is currently accepted by the staff. Westinghouse has successfully demonstrated leak-before-break for the primary coolant loops and auxiliary lines in over 40 plants and obtained NRC approval. Westinghouse leak rate calculations method uses two programs. They are:

- Mechanistic Pipe Break Program (MPBK). This program calculates the crack opening area.

- Fauske-Henry-Griffith Program (FHG). This program calculates the leak from the crack opening area obtained by MPBK (two-phase fluid conditions).

At the request of the NRC, benchmark evaluations were performed to validate Westinghouse methodology. These leak rate and stability calculation methods were accepted by the NRC.

Background Information

The Westinghouse computer code to calculate leak rates was developed in the latter part of the 1970's. The Code was verified by comparison with experimental data and the results were presented in WCAP-9558, Rev. 2 (Reference 1).

In 1986 the NRC staff requested that the code be benchmarked against newer data. Specifically, benchmark calculations were performed in WCAP-11256, Supplement 1 (Reference 2) and compared with the leakage observed in the recirculation pipe at Duane Arnold Nuclear Power Plant. Also the predication using the PICEP code (Reference 3) were compared in that WCAP(Reference 2).

AP600 LBB ANALYSIS

AP600

- A comparison between the experimental data performed by Battelle Columbus Laboratories (Reference 4) and Westinghouse code was shown in Reference 2. The Westinghouse code shows a good correlation.
- Recognizing the need for detailed and accurate leak rate measurement data, Westinghouse, Framatome, Electricite de France (EDF) and Commissariat a l'Energie Atomique (CEA) performed research on the subject. The actual leak rates were conducted by CEA at Aquitane II (Reference 5). A comparison of the experimental results and the analytical predictions was shown In Table 2.4-2 of Reference 2 . The Westinghouse code shows good agreement between experimental data and analytical predictions.
- In April 1987 additional clarifications were also provided to the NRC with reference to the leak rate comparison shown between the Westinghouse code and the Reference 4 data.

AP600 LBB ANALYSIS

From January 30 to February 2, 1989, the NRC and its consultant conducted a detailed technical review of the analysis documented in WCAP-12067. During the Technical Review, which was performed at the Westinghouse offices in Pittsburgh and at the South Texas Site, clarification of leak rate calculations in the evaluation of leak-before-break was provided and documented in Reference 6. Detailed discussion comparing Westinghouse code and PICEP code results were presented in that Reference.

A four-point bend test was conducted on a 4-inch schedule 80 pipe. The crack opening area comparison was shown in that WCAP(Reference 6). The predictions using the Westinghouse code shows good correlation. The PICEP predictions are more conservative especially in higher load levels. From the comparison, the Westinghouse code predicts a more accurate leak rate while still being conservative.

After reviewing the information, the NRC approved the WCAPs (References 2 and 6).

AP600 LBB ANALYSIS

The rationale for the leak rate calculation for LBB is first to obtain a best estimate of the leak rate results. With the estimated leak rate results, a safety factor of ten (10) on leak rate is then applied in order to assure conservative results.

We therefore believe Westinghouse code is the appropriate one to use for the AP600 LBB applications.

REFERENCES:

1. Palusamy, S. S. and Hartmann, A. J., "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," WCAP-9558, Rev. 2, Class 2, May 1981 (Westinghouse Proprietary Class 2).

2. Swamy, S. A., Witt, F. J. and Bamford, W. H., "Technical Bases for Eliminating Pressurizer Surge Line Ruptures as the Structural Design Basis for South Texas Project Units 1 and 2," WCAP-11256, Supplement 1, November 1986 (Westinghouse Proprietary Class 2).

3. Norris, D. M. and Chexal, B., "PICEP: Pipe Crack Evaluation Program (Revision 1)," EPRI NP-3596-SR, Revision 1, Special Report, December 1987.

4. Collier, R. P. , J. S. K. Liu, M. E. Mayfield, F. B. Stulen, "Study of Critical Two-Phase Flow Through Simulated Cracks", Battelle Columbus Laboratories, Interim Report BCL-EPRI-80-1, November 1980.

REFERENCES (cont'd):

5. Chouard, P., Richard, P., "Taux de Fuite dans les Fissures Traversantes: Recherche d'un Modele Analytique a Partir des Resultats Experimentaux"' CEA Laboratoire de Mecanique Appliquee, Note Technique DRE/STRE/LMA85/671.

6. Cranford, E. L., et. al., "Additional Information in Support of the Evaluations for Thermal Stratification of the Pressurizer Surge Lines of the South Texas Projects Units 1 and 2 Nuclear Power Plants," WCAP-12067, Rev. 1, Supplement 1, February 1989 (Westinghouse Proprietary Class 2).

AP600 LBB ANALYSIS

ITEM # 619: **DSER Section: 3.6.3.6-5**

QUESTION: Westinghouse should clarify the provisions (in paragraph 4 of Section 3.6.3.3 of the SSAR) that where applied normal operating stress is low in comparison with faulted stress at critical locations, stability is established by analyzing part-through-wall flaws.

RESPONSE: Westinghouse will use the LBB methodology and criteria for all the lines applicable in the AP600 Design. Stability analysis will be performed using through-wall flaws.

AP600

AP600 LBB Analysis

- Margins (NUREG-1061 VOLUME 3, SRP-3.6.3)
  - Margins of 10 on leak rate
  - Margin of 2 on flaw size
  - Margin of $\sqrt{2}$ on loads
  - Margin of 1.0 on loads is permitted if absolute load summation is used

- Material Strengths

  A. Stainless Steel

  Use ASME code minimum values for leak rate and limit load(stability) calculations.

  Note: AP600 LBB Scope Piping Systems do not include any cast material. The piping systems are made of forged stainless steel such as SA 312 TP 316LN and the welding process uses GTAW (Strength same as base metal). Forged stainless steel and GTAW do not degrade due to thermal aging. Limit load methods will be used for the stability calculations. The same approach was utilized for all the operating plants LBB analysis.

B. For carbon steel (MS Lines) and Alloy Steel (F Lines)

For leak rate and J-integral calculations use material strengths from test results.

AP600 LBB ANALYSIS

MATERIALS TEST PROGRAM

MATERIALS SA335/P11 AND SA333/GR.6 HAVE BEEN SELECTED FOR THE FEEDWATER AND STEAM LINES, RESPECTIVELY FOR AP600. TESTS ESTABLISHED TO DETERMINE THE MECHANICAL PROPERTIES OF THE PIPING MATERIAL AND ASSOCIATED WELDS REQUIRED TO QUALIFY THE FEEDWATER AND THE MAIN STEAM LINE FOR THE LEAK-BEFORE-BREAK (LBB) ANALYSIS. THE MECHANICAL TESTING REQUIRED INCLUDED TENSILE, CHARPY V-NOTCH AND J_{IC} FRACTURE TOUGHNESS.

THE TEST MATRIX IS AS FOLLOWS FOR THE PIPING MATERIAL AND WELDS:

AP600 LBB ANALYSIS

TEST TYPE	TEMPERATURE	NUMBER OF TESTS FOR EACH HEAT
Stress-Strain Curves (σ-ε)	Normal Operating	2
Stress-Strain Curves (σ-ε)	Room Temperature (70°F)	1
J-R Curves	Normal Operating	2
J-R Curves	Room Temperature (70°F)	2
Charpy Tests	Normal Operating	2 each x 2 orientations
Charpy Tests	Room Temperature (70°F)	2 each x 2 orientations

THERMAL STRATIFICATION

- Pressurizer Surge Line

It is well known that the pressurizer surge lines are subjected to thermaL stratification. The effects of stratification are particularly significant during certain modes of heatup and cooldown operation. These effects are evaluated for the AP600 pressurizer surge line. The loads resulting from the stratification effects are used in the LBB evaluation for the AP600 surge line.

AP600 LBB ANALYSIS

MAIN FEEDWATER PIPING

STARTUP FEEDWATER HAS BEEN SEPARATED FROM THE MAIN FEEDWATER LINE TO MINIMIZE LOADING PROBLEMS ASSOCIATED WITH UNFAVORABLE THERMAL STRATIFICATION CONDITIONS

THERMAL STRATIFICATION

WITH CURRENT CONFIGURATION STRATIFICATION WITH A SIGNIFICANT ΔT POTENTIAL CAN ONLY OCCUR DURING A PLANT STARTUP:

COLD FLUID AT 70°F IS IN THE 16 INCH MAIN FEEDWATER PIPING FROM THE DEAERATOR STORAGE TANK IN THE TURBINE BUILDING TO THE STEAM GENERATOR WITH MAIN FEEDWATER CIRCUIT ISOLATED.

AP600 LBB ANALYSIS

HOT FLUID AT 220°F IS DRAWN FROM THE DEAERATOR TANK AS THE MAIN FEEDWATER CONTROL VALVES ARE OPENED

HOT FLUID WILL FORM A TRAVELING WAVE OF STRATIFICATION MOVING TOWARD THE STEAM GENERATOR

TESTING WOULD BE REQUIRED TO DETERMINE THE SLOPE, LENGTH AND SPEED OF THE TRAVELING WAVE

THE CONSERVATIVE STRATIFICATION CASES USED FOR THE PIPE STRESS ANALYSIS ARE AS FOLLOWS:

AP600 LBB ANALYSIS

LENGTH - FROM ANCHOR AT TURBINE BUILDING TO BOTTOM OF VERTICAL RISER INSIDE CONTAINMENT NEAR THE STEAM GENERATOR. SEVERAL CASES ARE CONSIDERED TO REPRESENT STRATIFICATION IN DIFFERENT HORIZONTAL PORTIONS OF THE PIPE

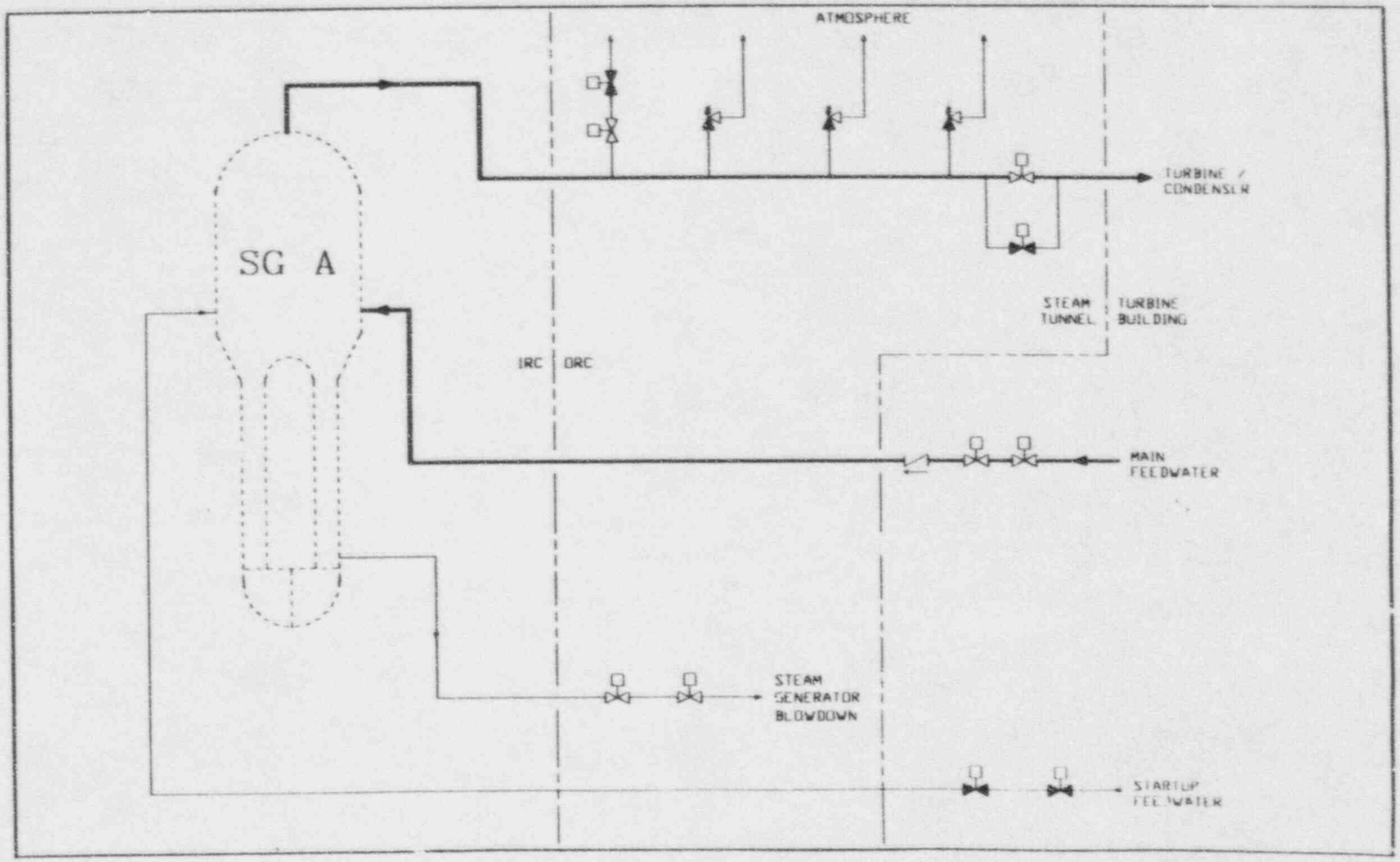
PROFILE - STEP CHANGE IN TEMPERATURE AT PIPE CENTERLINE

T_{TOP}	260°F
T_{BOTTOM}	40°F

RESULTS:

LBB ANALYSIS INCLUDED THE STRATIFICATION LOADS AND THE MARGINS ARE SATISFIED.

Steam Generator System
System Configuration



AP600 LBB ANALYSIS

AP600

THERMAL STRIPING

SEPARATION OF THE STARTUP FEED FROM THE MAIN FEED ELIMINATES THE ΔT AND FLOW FLUCTUATIONS FROM BEING A CONCERN FOR STRIPING IN THE MAIN FEED WATER LINE

SEPARATION OF NOZZLES ELIMINATES THE ΔT ASSOCIATED WITH HOT STANDBY SINCE MAIN FEEDWATER IS ISOLATED, AND SIGNIFICANTLY REDUCES AND COLD STARTUP ΔT POTENTIAL (FLOW FLUCTUATIONS HANDLED BY STARTUP FEED AND THIS PHASE IN STARTUP IS NOT A HOLD POINT)

FLOW FLUCTUATIONS AT STARTUP AND LOW POWER WILL OCCUR PRIMARILY IN THE STARTUP FEED LINE THAT WILL EXPERIENCE FULL MIXING AT FLOW RATES OF CONCERN.

AP600 LBB ANALYSIS

MAIN STEAM LINE FOR AP600 LBB

Material	:	SA333 Grade 6
Outer Diameter	:	32.00 Inch
Thickness	:	1.23 Inch (min.)
Thickness	:	1.44 Inch (nom.)
Pressure	:	815 psia
Temperature	:	520°F

STRESSES:

Normal= 6.53 KSI

Faulted= 17.14 KSI

MAIN STEAM LINE FOR AP600 LBB (Continued)

LBB RESULTS AT CRITICAL LOCATION

Leak Rate Results:

Margin = 10

Stability Results:

For a Flaw Size 2 x Leakage Flaw Size

Margin=2.00

MARGIN ON LOAD = 1.0; USING ABSOLUTE SUMMATION METHOD

FEEDWATER LINE FOR AP600 LBB

Material	:	SA335 P11 (Alloy)
Outer Diameter	:	16.00 Inch
Thickness	:	0.76 Inch (min.)
Thickness	:	0.845 Inch (nom.)
Pressure	:	890 psia
Temperature	:	435°F

AP600 LBB ANALYSIS

FEEDWATER LINE FOR AP600 LBB (Continued)

PARAMETRIC ANALYSIS RESULTS

CASE A:

NORMAL	:	6.47 KSI
FAULTED	:	20.00 KSI

CASE B:

NORMAL	:	13.00 KSI
FAULTED	:	26.00 KSI

0.50 GPM LEAK DETECTION CAPABILITY

AP600 LBB ANALYSIS

FEEDWATER LINE FOR AP600 LBB (Continued)

CASE A

LEAK RATE RESULTS

MARGIN = 10

STABILITY RESULTS (J-integral method)

FOR A FLAW SIZE 2 X LEAKAGE SIZE FLAW

MARGIN = 2

CASE B

LEAK RATE RESULTS

MARGIN = 10

STABILITY RESULTS(J-integral method)

FOR A FLAW SIZE 2 X LEAKAGE SIZE FLAW

MARGIN = 2

MARGIN ON LOAD = 1.0; USING ABSOLUTE SUMMATION METHOD

Table 3.6-1

High-Energy, and Moderate-Energy Fluid Systems Considered for Protection of Essential Systems[(a)]

System	High-energy	Moderate-energy
Reactor coolant (RCS)	•	
Steam generator (SGS)[(b)]	•	
Passive core cooling (PCS)	•	
Passive containment cooling (PXS)		•[(c)]
Main control room habitability (VES)	•	
Chemical and volume control (CVS)	•	
Primary sampling (PSS)	•	
Compressed and instrument air (CAS)		•
Normal residual heat removal (RNS)		•
Component cooling water (CCS)		•
Spent fuel pit cooling (SFS)		•
Demineralized water (DWS)		•
Liquid radwaste (WLS)		•
Radioactive drain (WRS)		•
Central chilled water (VWS)		•
Fire protection system (FPS)		•
Steam generator blowdown (BDS)(d)	•	
Main and startup feedwater (FWS)(d)	•	
Main steam (MSS)(d)	•	

a. Systems included on this list are high-energy, or moderate energy fluid systems located in the containment or the auxiliary building. Systems that operate at or close to atmospheric pressure such as ventilation and gravity drains are not included. See Subsection 3.6.1.2 for additional information.

b. Main and startup feedwater, main steam, and steam generator blowdown lines located in the containment and auxiliary building are part of the steam generator system.

c. The essential portion of the system is at atmospheric pressure.

d. The portion of these systems in the turbine building adjacent to the auxiliary building are evaluated for the effect of a double ended break on the main control room.

ITEM # 123 Meeting open item: M3.6.1-2

QUESTION: Identify all systems excluded from pipe break analysis based on the leak-before-break methodology.

RESPONSE: Table 3.6-1 identifies the high energy systems in the containment and auxiliary buildings. The systems or components inside of these building are the ones that are required to be protected from pipe failures. In addition to the systems in the containment and auxiliary building the main steam, main and startup feedwater, and steam generator blowdown line in turbine building adjacent to auxiliary building are evaluated for the effect of a double ended break on the main control room. These will be added to the SSAR Subsection 3.6.1.2 and Table 3.6-1.

SSAR Revision:

Essential systems are evaluated to demonstrate conformance with the design bases and to determine their susceptibility to the failure effects. Table 3.6-1 identifies systems which contain high and moderate-energy lines. The systems listed include all high- and moderate-energy systems inside containment plus the high- and moderate-energy systems in the auxiliary building near containment penetrations (including access hatches), the main control room, the Class 1E dc and UPS system or the portions of the passive containment cooling system located in the auxiliary building. The table does not list systems that operate at or close to atmospheric pressure including air handling and gravity drains. High energy system piping in the turbine building adjacent to the auxiliary building is evaluated for potential effects on the main control room. These systems are included on Table 3.6-1.

ITEM # 132 Meeting open item: M3.6.1-11

QUESTION: Clarify the piping classifications that are required by RG 1.26. These classifications should extend beyond the outboard restraint unless the restraint is at an isolation valve.

RESPONSE: The equipment classification system for the AP600 is described in Subsection 3.2.2. Subsection 3.6.2.2 of the AP600 DSER found that extension of the break exclusion zone to a nearby anchor to be acceptable.

"In the June 27, 1994 response to Q210.40, Westinghouse indicated that for the MS&FW piping, the auxiliary building anchors are as close as practical to the isolation valves with short sections of piping separating the valves and anchors to permit space for the branch connection for the isolation valve bypass (in the case of the MS line) and inservice inspection of welds. Westinghouse also indicated that these short sections of piping are to be designed to meet the break exclusion zone stress limits and, contrary to the requirements of Section 3.6.2 of the SRP, breaks are not postulated in these sections of piping.

"Relative to the extended AP600 break exclusion zones, during AP600 piping design audits the staff found that for the MS&FW systems the outboard isolation valves were located some 15.2 meters (50 feet) from the containment penetrations and verified that the sections of pipe between the outboard isolation valves and the auxiliary building anchors were short (approximately 1.52 meters (5 feet) in length). The staff has concluded that, since the piping in this short distance is designed to the acceptable criteria for break exclusion zone piping which is in Section 3.6.2.1.1.4 of the SSAR, extending the break exclusion zone for this short distance does not constitute a safety-related deviation from Section 3.6.2 of the SRP, and is acceptable, pending resolution of Open Item 3.6.2.2-1, which is discussed below."

ITEM # 594 DSER Open Item 3.6.2.2-1

QUESTION: Westinghouse should design: 1) the east wall of the east MSIV compartment between the MCR and the compartment, and 2) the floor slab of the east MSIV compartment between the compartment and the safety related electrical equipment room, to accommodate the worst case MS&FW line break. This design should be included in the SSAR.

RESPONSE: The east MSIV area will be evaluated for a main steamline break to consider effect on control room wall and floor of compartment

Scpoe: 1 square foot break for pressure and double ended break for jet impingement and pipe whip.

DISTRIBUTION w/attachment:
Docket File
PUBLIC
PDST R/F
RArchitzel
TKenyon
WHuffman
DJackson

DISTRIBUTION w/o attachment:
WRussell/FMiraglia, O-12 G18
AThadani, O-12 G18
RZimmerman, O-12 G18
DCrutchfield
BGrimes
TQuay
ACRS (11)
EJordan, T-4 D18
JMoore, O-15 B18
WDean, EDO
SHou, O-7 H15
DTerao, O-7 H15
JTsao, O-7 D4
HBrammer, O-7 H15
GBagchi, O-7 H15
RYoung, O-8 D1
BLeFave, O-8 D1
JHWilson