WESTINGHOUSE ELECTRIC CORPORATION

PRESENTATION

TO

UNITED STATES

NUCLEAR REGULATORY

COMMISSION

WESTINGHOUSE ENERGY CENTER

MAY 13, 1993

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Agenda
AP600 Piping Presentation to NRC
May 13, 1993

8:30	Introduction	McIntyre
8:45	Piping analysis methodology • Analysis and design criteria for safety-related piping systems • Analysis and design criteria for Class D systems • Leak-before-break approach for design certification	Lindgren
9:15	Analysis Status • Scope of piping design included in analysis • Level of detail in the analysis prepared • Leak-before-break analysis status	E. Johnson
9:45	Analysis work plan beyond Design Certification	E. Johnson
10:15	Conceptual pipe layout of selected systems	T. Johnson
11:30	Piping analysis ITAAC and design acceptance criteria • pipe stress analyses • high energy line break analysis • leak-before-break analysis	E. Johnson
12:00	Lunch	
1:00	Potential piping open issues • Advance Reactor Corporation (ARC) piping analysis issues • RAI Responses • SECY-93-087 Leak-Before-Break Position	E. Johnson/ Bagchi
2:30	Discussion	All
3:15	Action Items / Comments	All

AP600 Piping Analysis Methodology

- Analysis and design criteria for safety-related piping systems
- Analysis and design criteria for Class D systems
- Leak-before-break approach for design certification

Safety-Related Piping Systems

- Safety-related piping is AP600 Equipment Class A, B, and C - ASME Class 1, 2, and 3
- The AP600 Equipment Classification is defined in SSAR Section 3.2.
- Safety-related piping is analyzed to ASME Code, Section III requirements.
- Loading conditions, stress limits, and analytical methods for the structural evaluation of the safety-related piping and supports for design conditions, normal conditions, anticipated transients, and postulated accident conditions are discussed in Subsection 3.9.3.
- The design and evaluation requirements for the reactor coolant system are outlined in Subsection 5.4.3.
- The reactor coolant system design transients for normal operation, anticipated transients and postulated accident conditions are discussed in Subsection 3.9.1.

The reactor coolant loop analysis methods and results are presented in Appendix 3C of the SSAR

The integrated reactor coolant loop and supports system model is the basic system model used to compute loadings on components, component supports, and piping.

- The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, and the stiffnesses of auxiliary line piping affecting the system.
- This model is used to determine the static and dynamic loads on the primary loop piping and the component supports and the interfacing loads on the connecting components.

The AP600 reactor coolant loop (RCL) model consists of three-dimensional finite elements such as pipes, beams, elbows, masses, and springs of given dimensions, sizes, and physical properties that mathematically simulate the structural response of the physical system.

- The system model contains the reactor pressure vessel, two steam generators, four reactor coolant pumps, the containment interior building structure, the reactor coolant loop piping, and the primary equipment supports.
- The structural model is subjected to internal pressure, thermal expansion, weight and seismic loadings with boundary conditions imposed on the model.
- The finite element displacement method is used for the analysis.
- The stiffness matrix for each element is assembled into a system of simultaneous linear equations for the entire structure.
- The set of equations are solved by the wave-front technique variation of the Gaussian elimination method.

The reactor coolant loop piping is qualified in according to the requirements of the ASME Code, Section III, Subsection NB.

- The 1989 Edition with 1989 Addenda is the baseline Code.

- The loadings for ASME Code, Section III, Class 1 components are defined in Subsection 3.9.3.

- The following loadings are considered in the reactor coolant loop piping analysis:

- Design pressure
- Weight
- Thermal expansion - normal operating conditions
- Thermal expansion - other transient conditions
- Safe shutdown earthquake

- The reactor coolant loop piping is analyzed for the dynamic effects of a safe shutdown earthquake.

- The damping values used for the reactor coolant loop piping response spectra analysis are those in ASME Code Case N-411.

- When the reactor coolant loop is analyzed using the time history analysis method, 4% damping is used.

The requirements for seismic evaluation are outlined in SSAR Section 3.7.

Seismic design of the AP600 seismic Category I systems and components is based on the safe shutdown earthquake (SSE).

- The safe shutdown earthquake is defined as the maximum potential vibratory ground motion at the generic plant site.

The operating basis earthquake is not used as a design requirement for the AP600.

- Low-level seismic effects are included in the design of certain equipment potentially sensitive to a number of such events based on a percentage of the responses calculated for the safe shutdown earthquake.

Seismic analyses of the nuclear island are performed in conformance with the guidelines provided in ASCE Standard 4-86 and the criteria within SRP 3.7.2.

The methods used for seismic analysis of subsystems include, modal response spectrum analysis, time- history analysis, equivalent static analysis, and "design by rule." The method used is selected by the designer as appropriate for a specific item.

- The equivalent static load method involves equivalent horizontal and vertical static forces applied at the center of gravity of various masses.

- Design by rule is used for small diameter piping systems.

ASME Code Class 1 piping equal to or less than one-inch nominal pipe size and ASME Class 2 and 3 piping with nominal pipe sizes less than or equal to two inches is analyzed using one of the following methods.

- The response spectra method used to evaluate larger pipe systems.
- Equivalent static analysis based on appropriate load factors applied to the response spectra acceleration values; or,
- Seismic qualification by experience is based on the guidelines provided by EPRI Report NP-6628.

Analysis and Design Criteria for Class D Systems

Class D is nonsafety-related with some additional requirements on procurement, inspection or monitoring.

- For Class D structures, systems, and components containing radioactivity, it is demonstrated by conservative analysis that the potential for failure due to a design basis event will not result in exceeding the normal offsite doses per 10 CFR 20. This criterion is in conformance with the definition of Class D in Regulatory Guide 1.26.

- A structure, system or component is classified as Class D when it directly acts to prevent unnecessary actuation of the passive safety systems. Structures, systems and components which are required to support those which directly act to prevent the actuation of passive safety systems are also Class D.

Standard industrial quality assurance standards are applied to Class D structures, systems, and components to provide appropriate integrity and function.

- These standards are consistent with the guidelines for NRC Quality Group D.

The industry standards used for Class D structures, systems and components are widely used high quality standards.

- ASME Code, Section VIII for pressure vessels
- ANSI B 31.1, Code for Power Piping for piping.

The buildings containing Class D structures, systems, and components, as well as the anchorage of the structures and components to the building, are designed to the seismic requirements of the Uniform Building Code.

- The Class D systems and components are normally not designed for either the Uniform Building Code or seismic Category I.

- When Class D structures, systems, and components are located near a Class A, B, or C structure, system, or component, the requirements for seismic Category II may apply.

Piping Failure Protection

Pipe failure protection is provided according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 4.

- In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated piping failure.

- The criteria used to evaluate pipe failure protection are generally consistent with NRC guidelines including those in the Standard Review Plan Sections 3.6.1, 3.6.2 and NUREG-1061 and applicable Branch Technical Positions.

- Dynamic effects of postulated breaks are evaluated except for those analyzable sections of high-energy piping systems that are eliminated by the use of mechanistic pipe break methods.

Piping Failure Protection Requirements

- Subsection 3.6.1 provides the design bases and criteria for the analysis required to demonstrate that essential systems are protected.
  - The high- and moderate-energy systems representing the potential source of dynamic effects are listed.
  - The criteria for separation and the effects of adverse consequences are defined.

- Subsection 3.6.2 defines the criteria for postulated break location and configuration.
  - High-energy pipes are evaluated for the effects of circumferential and longitudinal pipe breaks and through-wall cracks.
  - Moderate-energy pipes are evaluated for the effects of through-wall cracks.

Evaluation of the dynamic effects of postulated breaks in selected piping systems is not required for AP600 based on application of mechanistic pipe break (leak-before-break) considerations.

- Subsection 3.6.3 describes the application of leak-before-break criteria to permit the elimination of pipe rupture dynamic effects considerations.

- Design guidelines aid in the design of piping systems that satisfy the requirements for mechanistic pipe break.

- The systems considered for application of leak-before break criteria include the reactor coolant loop, reactor coolant loop branch lines, main steam and feedwater lines out to anchors adjacent to the isolation valves, and other primary and secondary system piping inside containment equal to or greater than the four-inch nominal pipe size.

- Candidate systems for application of leak-before-break criteria are identified in Appendix 3E. This appendix was provided with the response to RAI Question 210.6.

- Responses to RAIs 252.5 and 252.6 provide clarification for the application of leak-before-break criteria.

Mechanistic pipe break evaluations demonstrate that for piping lines meeting the criteria, sudden catastrophic failure of the pipe is not credible.

- It is demonstrated that piping that satisfies the criteria leaks at a detectable rate from postulated flaws prior to growth of the flaw to a size that would fail because applied loads resulting from normal conditions, anticipated transients, and a postulated safe shutdown earthquake.

- The use of mechanistic pipe break criteria represents a higher level of confidence of the integrity of piping systems based on additional criteria compared to the existing high level of integrity provided by the requirements of the ASME Code.

Piping systems to which mechanistic pipe break are applied are high integrity systems with well understood loading combinations and conditions.

- The piping systems to which it is applied satisfy the requirements of the ASME Code, Section III for Class 1 and 2 systems. ASME Code requirements also apply to the pre-service and in-service inspection which confirm continued integrity.

- The mechanistic pipe break approach is applicable to high-energy piping provided plant design, operating experience, tests, or analyses have indicated low probability of failure from effects of intergranular stress corrosion cracking, water hammer, steam hammer, fatigue (thermal or mechanical), or erosion.

PIPING ANALYSIS STATUS
DESIGN CERTIFICATION

1.0 Piping Analysis Activities

2.0 Piping Analysis Details

3.0 Leak-Before-Break

1.0 Piping Analysis Activities

Preliminary pipe stress and leak-before-break analysis for sample systems:

Primary loop piping

- 2 soil cases - results provided in SSAR Appendix 3B
- enveloping cases - analysis in progress, 12/93 schedule

Pressurizer Safety and Relief Valve Piping

- Scope - 14", 8", 4" - Automatic Depressurization System (ADS) - Stage 1, 2 and 3
- 6" Pressurizer Safety
- Schedule - 12/93

Main steam Piping

- Scope - 32" - inside containment, and main steam isolation valve compartment
- Schedule - 12/93

Feedwater Piping

- Scope - 16" - inside containment and feedwater isolation valve compartment
- Schedule - 12/93

2.0 Piping Analysis Details

Preliminary Pipe Stress Analysis Methodology

- provided in SSAR and responses to RAI's
- followed to extent practical based on status of design data

Preliminary Pipe Stress Analysis Approach

- Goal - define pipe layout, pipe support steel arrangement and module support steel arrangement for the plant model.
- Loadings - thermal expansion, SSE and other limiting loadings
- Structural Models - approximations are used based on status of design data
  - pipe support stiffness
  - module steel stiffness
  - branch line coupling
  - equipment and valve properties
  - seismic response spectra

- Structural Analysis - static, SSE response spectra, SSE time history

- Results

  - ASME pipe stress evaluation
  - equipment and valve interface loads
  - pipe loads for leak-before-break
  - leak-before-break screening criteria

- Analysis Iterations

  - adjustments to pipe layout and support arrangement to achieve acceptable design

3.0 Leak-Before-Break

Objective - provide sample analyses that cover a range of materials, fluid conditions and pipe sizes

Methodology

- provided in SSAR and responses to RAI's

Primary Loop Piping

- sample analysis described in SSAR Appendix 3B

Auxiliary Piping

- pressurizer safety and relief, mainsteam and feedwater
- materials
  - SA 376 - 316 LN
  - SA 333 - Gr 6
  - Alloy Steel
- fluid conditions
  - subcooled liquid (primary and secondary system)
  - saturated steam (primary system)
  - dry steam (secondary system)
- pipe sizes
  - 4, 6, 8, 14, 16 and 32 inches
- schedule 12/93

WORK PLAN BEYOND DESIGN CERTIFICATION

1.0 SCOPE
- FOAKE (THROUGH 9/96)
- POST-FOAKE

2.0 WORKPLAN

3.0 LEAK-BEFORE-BREAK

Pipe Stress Analysis Scope

1.0 Safety Related Piping

1.1 All ASME Class 1 piping

1.2 All ASME Class 2/3 piping inside containment (includes B31.1 extension)

1.3 ASME Class 2/3 piping outside containment that are extensions of major B31.1 piping (> 150°F)

where: Major B31.1 piping includes

1) all large bore
2) small bore extensions to containment isolation valves

1.4 High energy ASME Class 2/3 piping outside containment

2.0 Non-Safety (B31.1) Piping (> 150°F) SSE analysis if Category II

2.1 All large bore piping inside containment

2.2 All large bore piping outside containment

2.3 Small bore piping inside containment in compartments that have safety-related components

2.4 Small bore piping outside containment in compartments that have containment isolation valves

Pipe Support Steel and Module Steel Stress Analysis Scope

- Large Bore Pipe Support Steel and Corresponding Module Steel
  - high energy piping: safety related and non-safety related
  - other piping is post-FOAKE
- Small Bore Pipe Support Steel and Corresponding Module Steel
  - This is post-FOAKE

SYSTEMS THAT HAVE ASME PIPING, WITH LEAK-BEFORE-BREAK (LBB) PIPING

PXS	PASSIVE CORE COOLING
RCS	REACTOR COOLANT
RNS	NORMAL RESIDUAL HEAT REMOVAL
SGS	STEAM GENERATOR

SYSTEMS THAT HAVE ASME PIPING, WITHOUT LBB PIPING

CAS	COMPRESSED AND INSTRUMENT AIR
CCS	COMPONENT COOLING WATER
CVS	CHEMICAL AND VOLUME CONTROL
DWS	DEMINERALIZED WATER TRANSFER AND STORAGE
FPS	FIRE PROTECTION
PCS	PASSIVE CONTAINMENT COOLING
PSS	PRIMARY SAMPLING
PXS	PASSIVE CORE COOLING
RCS	REACTOR COOLANT
RNS	NORMAL RESIDUAL HEAT REMOVAL
SGS	STEAM GENERATOR
SFS	SPENT FUEL PIT COOLING
VES	MAIN CONTROL ROOM EMERGENCY HABITABILITY
VFS	CONTAINMENT AIR FILTRATION
VUS	CONTAINMENT LEAK RATE TEST
VWS	CENTRAL CHILLED WATER
WLS	LIQUID RADWASTE

SYSTEMS THAT HAVE AP600 CLASS D PIPING

BDS	STEAM GENERATOR BLOWDOWN
CCS	COMPONENT COOLING WATER
CPS	CONDENSATE POLISHING
CVS	CHEMICAL AND VOLUME CONTROL
DOS	STANDBY DIESEL AND AUX BOILER FUEL OIL
DWS	DEMINERALIZED WATER TRANSFER AND STORAGE
FWS	MAIN AND STARTUP FEEDWATER
PXS	PASSIVE CORE COOLING
RCS	REACTOR COOLANT
SFS	SPENT FUEL PIT COOLING
SWS	SERVICE WATER
WGS	GASEOUS RADWASTE
WLS	LIQUID RADWASTE
WRS	RADIOACTIVE WASTE DRAIN
WSS	SOLID RADWASTE
WWS	WASTE WATER
ZOS	ONSITE STANDBY POWER

SYSTEMS THAT CONTAIN NO ASME OR AP600 CLASS D PIPING = CLASS E

CDS	CONDENSATE
CFS	TURBINE ISLAND CHEMICAL FEED
CLS	CIRCULATING AND SERVICE WATER CHEMICAL INJECTION
CMS	CONDENSER AIR REMOVAL
CWS	CIRCULATING WATER
DTS	DEMINERALIZED WATER TREATMENT
MSS	MAIN STEAM
PGS	PLANT GAS
PWS	POTABLE WATER
SSS	SECONDARY SAMPLING
TCS	TURBINE BUILDING CLOSED COOLING WATER
VYS	HOT WATER HEATING
WWS	WASTE WATER

FOAKE WORK PLAN

1. Conceptual piping layout

   - separation for safe shutdown
   - access for plant maintenance/inspection
   - modularized construction

2. Design Manuals

   - pipe supports
   - piping and support standardization plan

3. Preliminary piping layout, pipe support configuration, module support configuration

   - governing thermal modes of operation
   - thermal stratification
   - SSE for Category I and II
   - governing valve opening/closing loads
   - LBB screening criteria

4. Hazards review by plant area

   - Seismic Category II
   - Pipe rupture mitigation

5. ASME Design specifications

6. Intermediate piping layout, pipe support design, module support design
   - all modes of operation
   - representative equipment vendor data
   - leak-before-break analysis
   - partial design of pipe support/module steel for large bore high energy piping

7. ASME Design Reports

   - some are post-FOAKE

LBB Piping Systems

System	Analysis Package	
RRCS	*RCS-PLA-01	14, 8, 6, 4" 1st, 2nd and 3rd Stage ADS, Pressurizer Safety
	RCS-PLA-02	6", 4" Pressurizer Spray
	RCS-PLA-03	12" 4th Stage ADS
	RCS-PLA-04	18" Pressurizer Surgeline
	RCS-PLA-05	31", 22" Primary Loop

*Sample LBB evaluations will be completed in 12/93

NSGS	*SGS-PLA-01	-16" Feedwater
	SGS-PLA-02	-16" Feedwater
	*SGS-PLA-03	-32" Mainsteam
	SGS-PLA-04	-32" Mainsteam
NRNS	RNS-PLA-01	12", 10" Normal RHR Suction

*Sample LBB evaluations will be completed in 12/93

:NPXS	PXS-PLA-01	8", 6" Direct Vessel Injection, Normal RHR Return
	PXS-PLA-02	8", 6" Direct Vessel Injection, Normal RHR Return
	PXS-PLA-03	12", 10" Passive RHR Suction, 4th Stage ADS
	PXS-PLA-04	10" Passive RHR Return
	PXS-PLA-05	8" CMT Equalization
	PXS-PLA-06	8" CMT Equalization

AP600 Conceptual Pipe Layout

Reactor Coolant System
- Primary coolant loop layout

Automatic Depressurization System
- Pressurizer ADS valve and piping module
- Pipe routing to spargers

Passive RHR System
- Supply and return valves and piping

Steam Generator System
- Feedwater pipe routing
- Main steam line routing

PIPING ITAAC

1.0 Pipe Stress Analysis

2.0 High Energy Line Break Analysis

3.0 Leak-Before-Break Analysis

AP600 PIPING ANALYSIS ITAAC

Scope:

Safety-Related-Piping

- Seismic Category I
- ASME III Code Class 1, 2, and 3
- AP600 Class A, B, and C

Design Process

ASME Design Specification is prepared including:

- 60 year design life
- piping analysis methods
- functional capability requirements

ASME Design Report is prepared including:

- interface loads/accelerations for line mounted components
- interface loads with other components and pipe support
- interface deflections with adjacent systems, structures and components
- review of as-built configuration

Pipe Rupture Protection Report is prepared including:

- use of leak-before-break to eliminate dynamic effects of high energy pipe break
- demonstration that safe shutdown can be achieved for each postulated pipe rupture
- review of as-built configuration

Acceptance Criteria

Existence of

- ASME Design Specification (ITAAC number 1, Table 4.3-1)
- ASME N Stamp Symbol (ITAAC number 2, Table 4.3-1)
- Pipe Rupture Protection Report (ITAAC number 5, Table 4.2-1)

POTENTIAL PIPING OPEN ISSUES

1.0 Advanced Reactor Corporation (ARC) Issues

2.0 Other Issues

3.0 SECY-93-087 Leak-Before-Break

1.0 ARC Issues

	Appendix G
1.1 Support/Module Deflection Limit	4.3.4
1.2 Decoupling for Branch Pipe and Run Pipe	4.3.4, 4.8.2 4.10.9
1.3 Level D Pipe Stress Limit - Seismic Anchor Motions/Inertia	4.9.1

1.1 Support/Module Deflection Limit

A support deflection limit may be used to simplify the design process for piping that is supported by module steel or piping that is supported by the building structures.

The deflection limit may be applied to either the total deflection of the module steel and pipe support steel, or to the deflection of the support steel only.

The various types of models that may be used for module steel analysis and piping analysis are described below

NO MODULE STEEL-RIGID PIPE SUPPORT STEEL

SSE deflection of pipe support steel is limited to 1/8 inches per ARC report Appendix G section 4.3.4. Support modeling is:

fabricated supports	rigid stiffness
snubbers, limit stops	use vendor stiffness

DECOUPLED MODELS-RIGID PIPE SUPPORT STEEL-RIGID MODULE STEEL

SSE deflection of pipe support steel and module steel is limited to 1/8 inches per ARC report Appendix G section 4.3.4.. Support modeling is:

fabricated supports	rigid stiffness
snubbers, limit stops	use vendor stiffness

DECOUPLED MODELS-RIGID PIPE SUPPORT STEEL-FLEXIBLE MODULE STEEL

Module steel is represented by:

amplified response spectra calculated from the module steel analysis model

or simplified dynamic model is included in piping analysis model

SSE deflection of pipe support steel is limited to 1/8 inches per ARC report Appendix G section 4.3.4. Support modeling is:

fabricated supports	rigid stiffness
snubbers, limit stops	use vendor stiffness

COUPLED MODELS-RIGID PIPE SUPPORT STEEL-FLEXIBLE MODULE STEEL

Dynamic models of module steel is included in the dynamic model of the piping

SSE deflection of pipe support steel is limited to 1/8 inches per ARC report Appendix G section 4.3.4. Support modeling is:

fabricated supports	rigid stiffness
snubbers, limit stops	use vendor stiffness

1.2 Decoupling for Branch and Run Piping

The decoupling criteria described in Appendix G of the ARC report may be applied to decouple equipment from the analysis of the run piping.

BRANCH PIPING ANALYSIS MODELS

The run piping and branch piping can be decoupled based on ARC report Appendix G-sections 4.3.4, and 4.8.2.

- -moment of inertia ratio is equal to or larger than 25.
- -header displacement analysis if larger than 1/8 inches

Decoupling is also permitted when the run pipe to branch pipe diameter ratio is greater than or equal to 3.0.(SSAR 3.7.3.8)

EQUIPMENT DISPLACEMENTS

The equipment and run piping can be decoupled when the ratio of equipment minimum dimension to branch pipe diameter is greater than or equal to 3.0.(SSAR 3.7.3.8.2.1)

The methodology is the same as the branch piping guidelines in ARC report Appendix G-sections 4.3.4, and 4.8.2.

An exception is the primary loop piping model which includes a dynamic model of the steam generator and reactor pressure vessel.

1.3 Level D Pipe Stress Limits-Inertia and Seismic Anchor Motions

The guidelines in ARC report Appendix G, section 4.9.1 may used to evaluate ASME Class 1, 2 or 3 pipe stress to more realistic limits than the 1992 Code.

weight stress	0.25*Sy
SSE SAM stress	5.0*Sy
Equation 9 stress	4.0*Sy

The following positions provided in response to RAI 210.7 will be revised to reflect the ARC report:

Class 2/3	0.25*P*D/t + 0.75*i*MA/Z + i*(MC+M2)/Z < 4.0*Sh.
Class 1	0.5*C2*D*(M1+M2)/I < 6.0*Sm
Class 1/2/3	Equation 9 < 3.0*Sm or 2.0*Sy

2.0 Other Issues

	RAI
2.1 Pipe Stress Limits for Emergency/ Faulted Thermal	210.7
2.2 Composite Model Damping - Structural Frames	210.14
2.3 Break Exclusion Zone - MS/FW/BD	252.6, 410.76
2.4 Subcompartment Pressurization - reactor vessel annulus	410.86
2.5 Independent Support Motion Response Spectra	210.11
2.6 Load combination - SSE and Pipe Rupture	210.7
2.7 NCIG-14 Methodology for Small Bore - Classes A, B, and C (≤ 2")	210.8

2.1 Pipe Stress Limits for Level C and D Thermal Expansion

For systems that must maintain fluid flow path

$$i^*M/Z \leq 3.0\ S_c$$

Based on single anchor movement allowable in ASME Code

2.2 Composite Modal Damping - Structural Frames

Piping systems with dynamic coupled models for piping, equipment, and module steel

Use composite modal damping:

N-411 for piping/equipment
4% for module steel

Piping systems with dynamic coupled models of piping and equipment

Use N-411 damping

2.3 Break Exclusion Zones - MS/FW/BD

Mainsteam and Feedwater Piping

Extends from LBB boundary at the flued head to pipe weld inside steel containment, out to six way anchor at auxiliary building wall

Use 1.0 ft^2 postulated break flow area for subcompartment pressurization per Standard Review Plan 3.6.1

Steam Generator Blowdown Piping

Extends from anchor at steel containment out to six way anchor at auxiliary building wall

Additional details will be provided in response to RAI 410.76 in 8/93

2.4 Subcompartment Pressurization

Reactor Vessel Annulus is designed for 5 gpm leakage crack based on application of LBB to primary loop and direct vessel injection piping

Utility Requirements Document, Volume III, Chapter 1, section 4.5.5.1.3

2.5 Independent Support Motion

Inertia responses of support groups	SRSS Combination
Damping Values	Reg. Guide 1.60

2.6 Load Combination - SSE and pipe rupture

Based on URD position

SRSS of pipe rupture and SSE for:

- primary loop piping
- Class 1 and CS components
- Class 1 component supports

2.7 NCIG-14 (EPRI NP-6628) - Small Bore Piping

Based on earthquake experience
Industry response to NRC questions expected mid-1993

3.0 LEAK-BEFORE-BREAK

SECY-93-087 RECOMMENDS BOUNDING ANALYSIS FOR ALL LINES

AP600 POSITION IS SSAR METHODOLOGY AND PRELIMINARY ANALYSIS FOR A WIDE RANGE OF SYSTEMS

Westinghouse Electric Corporation
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ET-NRC-93-3889
NSRA-APSL-93-0178
Docket No.: STN-52-003

May 13, 1993

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: MR. R. W. BORCHARDT

SUBJECT: INFORMATION PRESENTED AT THE MAY 13, 1993 MEETING TO DISCUSS THE AP600 PIPING ANALYSIS

Dear Mr. Borchardt:

Please find enclosed with this letter information presented by Westinghouse at a May 13, 1993, Westinghouse/NRC meeting on the AP600 Piping Analysis. The purpose of the meeting was to review the piping analysis methodology, analysis status and analysis planning for upcoming analysis. Conceptual piping layouts, piping system ITAACs, and potential piping analysis related open items. Enclosure 1 provides a list of meeting attendees. Enclosure 2 identifies the information provided during the meeting and provided as further enclosures to this letter.

Please contact Donald A. Lindgren on (412) 374-4856 if you have any questions concerning this transmittal.

N. J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

/nja

Enclosures
Attachments

cc: T. Kenyon, NRC (w/o Enclosures/Attachments)
R. Hasselberg, NRC
B. A. McIntyre, Westinghouse (w/o Enclosures/Attachments)

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