Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4873
DCP/NRC0647
Docket No.: STN-52-003

October 30, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: RESPONSES TO NRC LEAK-BEFORE-BREAK OPEN ITEMS

Dear Mr. Quay:

Attached are responses to open items related to LBB open items, DSER Section 3.6.3. These open items were discussed in Attachment 2 of an NRC letter dated August 20, 1996 and in phone calls on August 26 and 27. The response to RAI 210.288 which was part of the transmittal is included.

This submittal will permit completion of the staff review from the NRC staff and preparation of input for the Final Safety Evaluation Report.

Westinghouse needs additional information to complete the response for two of the items related to feedwater line Leak-Before-Break (OITS #614 and #620). Westinghouse needs information from the NRC evaluation of waterhammer loads.

Please contact Donald A. Lindgren on (412) 374-4856 if you have additional questions.

for BAM

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachment

cc: D. T. Jackson - NRC
N. J. Liparulo - Westinghouse (w/o attachments)

E004 1/1

060097

2983A

Enclosed response to NRC questions and comments
Letter NSD-NRC-96-4873

From NRC Letter dated August 20, 1996, Attachment 2

DSER OI 3.6.3.4-1 (OITS #608)
DSER OI 3.6.3.4-2 (OITS #609)
DSER OI 3.6.3.5-2 (OITS #611)
DSER OI 3.6.3.5-5 (OITS #614)
DSER OI 3.6.3.6-1 (OITS #615)
DSER OI 3.6.3.6-3 (OITS #617)
DSER OI 3.6.3.6-6 (OITS #620)
RAI 210.228 (OITS #3518)

The following provides responses to open items related to leak-before-break (LBB) issues for the AP600. An NRC letter dated August 20, 1996 and a phone call on August 27, 1996 provided the latest update of the NRC staff questions. The questions and comments are addressed below. Draft markups of SSAR subsection 3.6.3 and Appendix 3B are also provided to respond to the comments.

OITS #608 DSER 3.6.3.4-1

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.

NRC Comment

Clarify how PSI and (augmented) ISI requirements will provide for integrity of ASME Code Class 2 and 3 piping systems.

Response

A paragraph will be added as shown in attached markup to subsection 3B.5 to clarify that the differences in inspection requirements does not require differences on LBB criteria or methods.

NRC Comment

Clarify: ...the highest stressed point (critical location) has to be less than the bounding analysis curve.

Response

The 5th bullet in subsection 3.6.3.2 will be revised as shown in attached markup to clarify the comparison

NRC Comment

Revise parts of this SSAR section which are inconsistent with bounding analysis approach.

Response

Subsections 3.6.3.2 and 3.6.3.3 will be revised as shown in attached markup to clarify that bounding analysis is the method used for the AP600.

NRC Comment

Provide a description of LBB acceptance criteria and demonstrate how bounding analysis approach satisfies these criteria.

Response

Subsections 3.6.3.2 and 3.6.3.3 will be revised as shown in attached markup to clarify how the bounding analysis method follows NRC guidance.

NRC Comment

Discuss the failure mechanism for non-stainless steel material.

Response

The failure mechanism for ferritic steel piping uses the J-integral method. This information will be included in 3.6.3.3 as shown in attached markup.

NRC Comment

Explain the relationship between the bounding analyses and the requirements of SRP 3.6.3 and NUREG-1061, Vol. 3.

Response

This information was discussed during phone call and additional information will be provided in the SSAR as shown in attached markup .

NRC Comment

Explain if water chemistry requirements will minimize stress corrosion in auxiliary stainless steel piping. If so, provide discussion in 3B.2.2.

Response

This information was discussed during phone call. Information resistance to corrosion is included in the SSAR and no SSAR changes are proposed.

NRC Comment

Explain in 3B.2.3 how pressurizer safety valve discharge loads are considered in analysis.

Response

The first note in 3B.3.3 will be revised to specifically address safety valve opening.

NRC Comment

Explain in 3B.2.4 how fatigue effects due to thermal and other cyclic loads are evaluated in ASME Code, Class 2 and 3 piping

Response

The Class 2 and 3 piping systems comply with the stress range reduction factors of the ASME Code, Section III. This information will be added to subsection 3B.2.4 as shown in attached markup.

NRC Comment

Explain in 3B.2.5 how dynamic strain aging (DSA) effects were evaluated.

Response

The material used in piping systems fabricated of ferritic steel is not susceptible to dynamic strain aging effects. This information will be added to the subsection 3B.2.5 as shown in attached markup.

NRC Comment

Clarify the statement in the third paragraph in auxiliary stainless steel section of 3B.2.6 that unisolable sections of identified candidate LBB piping systems are susceptible to adverse stresses as described in Bulletin 88-08.

Response

There is no unisolable section susceptible to adverse stresses described in Bulletin 88-08. Subsection 3B.2.6 will be revised to state this as shown in attached markup.

NRC Comment

Explain in 3B.2 how susceptibility of failures due to creep fatigue and indirect causes and cleavage type failures were evaluated.

Response

Subsection 3B.2.7 will be added as shown in attached markup to discuss these mechanisms.

NRC Comment

Explain how the bounding analyses are consistent with the methodology in GDC-4, SRP 3.6.3 and NUREG-1061, Vol. 3. Include the following items.

Clarify the load margin in the third bullet of 3.B.3.

Response

Margin on load is 1. This will be included in subsection 3.B.3 as shown in attached markup

NRC Comment

Clarify in 3B.3.1 the inclusion of 304L.

Response

Type 304L is used for the Accumulator line. This will be included in subsection 3.B.3.1.1 as shown in attached markup.

NRC Comment

Clarify in 3B3.1.3 how the lower magnitude of bending stress is selected.

Response

The lower magnitude of bending stress selected is a very small number that is lower than the expected minimum bending stress. This will be included in subsection 3.B.3.1.3 as shown in attached markup.

NRC Comment

Clarify in 3B3.2.4 how the higher magnitude of bending stress is selected.

Response

The higher magnitude of bending stress is selected such that the maximum stress is close to the flow stress. This will be included in subsection 3.B.3.1.4 as shown in attached markup.

NRC Comment

Define Y and Z axes in 3B.3.3.1 through 3B.3.3.3.

Response

The Y and Z axes are lateral axes This will be included in subsection 3.B.3.3.1 as shown in attached markup.

NRC Comment

Revise 3B.4 to include ASME Code Class 3 piping systems (3B.6 indicates there are Class 3 candidate LBB systems).

Response

Subsection 3B.4 will be revised to focus on differences due to material not Code Class as shown in attached markup. Class 3 will be included

NRC Comment

Revise 3B.4 to discuss differences in design analyses for ASME Code Class 1, 2, and 3 systems, instead of differences in LBB analyses.

Response

See previous question.

NRC Comment

Discuss the significance of the differences in inspecti criteria for ASME Code Class 1, 2, and 3 piping systems identified in 3B.5 for LBB ap ications.

Response

LBB evaluation are based on the ability to detect a potential leaking crack. There is no difference in LBB evaluation due to inspection differences. Subsection 3B.5 will be revised as

shown in attached markup to state this. Reference to additional requirements for Class 3 ECCS lines will be added to subsection 3B.6.

NRC Comment

Discuss the significance of the differences in fabrication requirements for ASME Code Class 1, 2, and 3 piping systems identified in 3B.6 for LBB applications.

Response

There is no difference in LBB evaluation due to fabrication differences. Subsection 3B.6 will be revised as shown in attached markup to state this. Additional information on examination requirements will be added.

NRC Comment

Delete 3B.7 on monitoring unanticipated dynamic loads in the Main Feedwater Line.

Response

Westinghouse waiting for additional information on the method the NRC used for the evaluation of waterhammer loads and will provide additional information to support LBB for main feedwater line

NRC Comment

Delete 3B.8 for augmented ISI for Main Feedwater nozzles on steam generators.

Response

See response for previous comment

NRC Comment

Add line size and material to Table 3B-1

Response

Table 3B-1 will be revised as shown in attached markup to include this information.

NRC Comment

Westinghouse should perform sensitivity analyses to assess differences in calculated leak rates for 4-inch ADS line.

Response

The 4-inch ADS line is no longer a candidate LBB line. Another 4-inch line will be selected for audit.

**DSER 3.6.3.4-1 is Resolved** pending formal SSAR revision.

OITS #609 DSER 3.6.3.4-2

Westinghouse should add COL Action item 3.6.3.4-1 to the SSAR.

Response

The COL item in subsection 3.6.4.2 will be revised as shown in attached markup to clarify the comparison to be performed by the COL

**This item is Resolved** pending formal SSAR revision.

OITS #611 DSER 3.6.3.5-2

Westinghouse should provide additional discussion concerning the differences in analysis, fabrication, and inspection between Class 1 and 2 systems. (See Q252.5)

Response

Subsection 3B.5 and 3B.6 will be revised as shown in attached markup to state that there is no difference for ASME Code Class 1, 2, and 3 piping systems in LBB evaluation due to inspection or fabrication differences.

**This item is Resolved** pending formal SSAR revision.

OITS #614 DSER 3.6.3.5-5

Westinghouse should provide in the SSAR, more detailed discussions with sufficient information to support the conclusion that the MS and FW piping systems do not fall within the limitations delineated in Section 5.1 of Volume 3 of NUREG-1061. The 8/20/96 NRC letter requested that the main feedwater line be deleted from list of LBB lines.

Response

Use of LBB on the main steam line has been accepted. The remaining unresolved issue for the feedwater line is size of waterhammer loads. Westinghouse has calculated waterhamer loads for the system that are consistent with the use of leak-before-break to the main feedwater line. Westinghouse needs additional information on the method that the NRC staff used to determine waterhammer loads for their assessment. See open item 620 (DSER item 3.6.3.6-20)

**This item is Action N** pending response to Westinghouse request for information.

OITS #615 DSER 3.6.3.6-1

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

Response

Subsection 3.6.3.4 will be revised to address the acceptability of candidate LBB piping systems based on the enveloped soil conditions.

**This item is Resolved** pending formal SSAR revision.

OITS #617 DSER 3.6.3.6-3

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. Include information from GW-N1-001 in the SSAR.

Response

Questions related to leakage rate and margin on flaw size have been previously resolved. Westinghouse will provide a copy of GW-N1-001 for review when LBB calculations are audited.

**This item is Closed**

OITS #620 DSER 3.6.3.6-6

Westinghouse should address whether the water hammer type loads from condensation events need to be considered in the LBB analyses; if not, Westinghouse should justify why these loads can be excluded. NRC letter dated 8/28/96 - Resolve differences between Westinghouse and NRC estimates for water hammer loads for the main feedwater pipe.

Response

The response to the issue of water hammer in systems containing reactor coolant was provided by NSD-NRC-96-4743. Westinghouse is waiting for additional information from the NRC staff on the methods and loads used by the staff to address the issue of feedwater hammer loads.

**This item is Action N** pending response to Westinghouse request for information.

OITS #3518 RAI 210.228

NUREG/CR-6443 indicates that the effects of: 1) restraint of pressure induced bending, and 2) residual stress can result in gross overestimates of leak rates in small diameter (4-inch) piping. Westinghouse should be prepared to discuss and quantify these effects.

Response

Westinghouse has reviewed NUREG/CR-6443, "Deterministic and Probabilistic Evaluation for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluation," and considered application to small diameter (4-inch) piping in the AP600. There are significant issues with the application of the NUREG/CR-6443 information to small diameter piping at the conditions found in the AP600. These issues are outlined below.

1. The studies in the NUREG with respect to leak rate calculations and crack opening displacement calculations were performed for pipes with thicknesses and radius to thickness ratios (R/t) typical of BWR pipe. Pipe in PWRs is thicker. The pipes in the study have $R_m/t$ ratios of 10 and 6. The $R_m/t$ ratio for the small diameter (4-inch) AP600 pipe is approximately 4.6. The issues identified in the NUREG appear to be more significant for thinner pipe. The extrapolation of the information in the NUREG to the thicker AP600 pipes represents an uncertainty in the application of the information.

2. Although pipe sizes and thicknesses typical of BWRs was used in the study, the pressure used was the much higher pressure typical of PWRs. Thus the study was not prototypical of BWR conditions. The use of a pressure higher than typical operating pressure for the study introduces additional uncertainties in the application of the information.

3. The leak rate flaw sizes are calculated using a roughness associated with intergranular stress corrosion cracking (IGSCC). IGSCC is associated with some BWR pipe degradation and is not found in PWR piping. Piping with susceptibility to IGSCC is specifically excluded from application of LBB by GDC 4. The use of IGSCC crack morphology in evaluating effects on leak rate for PWR piping is not appropriate.

4. The report indicates a significant increase in critical flaw size for the restrained case compared to the unrestrained case. This is a benefit in LBB evaluations. The report does

not satisfactorily address how this increase in critical flaw size may be factored in to the LBB evaluations and the effect on flaw size margins.

5. The influence that residual stress has on leak rate and crack opening displacement appears to be strongly dependent on pipe wall thickness. The effect drops with increasing thickness. The extrapolation of residual stress effects discussed in the report to the thicker PWR pipe may not be appropriate. The residual stress influence is also apparently an issue only with low normal stress.

NUREG/CR-6443 is not directly applicable to the evaluation of leak-before-break for the 4-inch pipes in the AP600 because of the differences in size and conditions between BWRs and the AP600. The adverse effect that restraint could have on the LBB evaluation for AP600 appears to be small and the overall effect could even be positive. The substantial margins on leak rate and critical crack size used in the LBB evaluation are more than sufficient to address any uncertainty that the NUREG may suggest.

In summary, the results and information provided in NUREG/CR-6443 do not support exclusion of 4-inch pipe in the AP600 from application of LBB criteria.

**This item is Action N** pending NRC Review of this response.

In addition to requirements on the design, fabrication, and inspection of the piping systems, the application of mechanistic pipe break requires a qualified leak detection capability. Leak detection systems inside containment meet the guidelines of Regulatory Guide 1.45. See subsection 5.2.5 for a discussion of the leak detection system for the reactor coolant system and connected piping.

### 3.6.3.2 Design Criteria for Leak-before-Break

The methods and criteria to evaluate leak-before-break in the AP600 are consistent with the guidance in NUREG-1061 (Reference 11) and Draft Standard Review Plan 3.6.3 (Reference 12). The application of the mechanistic pipe break in AP600 requires that the following design requirements are met.

- Pre-service inspection of welds is required.

- For ASME Code Class 1, Class 2, and 3 systems for which leak-before break is demonstrated, the ASME Section XI preservice and inservice inspection will provide for the integrity of each system. Appendix 3B describes augmented inspection requirements for the feedwater line. ~~These systems are identified in Appendix 3E.~~

- Inservice inspection and testing of snubbers (if used) are performed to provide for a low snubber failure rate.

- For the maximum stress due to steady-state vibration refer to subsection 3.9.2.

- The leak-before-break bounding analysis curves are developed for each applicable piping system. The bounding analysis methods are described in Appendix 3B. These curves give the design guidance to satisfy the stress limits and leak-before-break acceptance criteria. The highest stressed point (critical location) determined from the piping stress analysis is compared to ~~has to be less than~~ the bounding analysis curve and has to fall on or under the curve. The points on or under the bounding analysis curve satisfy the requirements for leak-before-break.

  The analyzed normal stress and maximum stress are not required to construct the bounding analysis curve. The analyzed stresses are calculated by the equation;

$$\sigma = \frac{F_x}{A} + \frac{M_b}{Z}$$

where:

$\sigma$ is the stress
$F_x$ is the axial force
$M_b$ is the applied bending moment
A is the piping cross-sectional area
Z is the piping section modulus.

The normal stress is calculated by the algebraic summation of load combination method and the maximum stress is calculated by the absolute summation of load combination method.

- The corrosion-resistant piping materials, including base metal and welds, have an appropriate toughness. The piping materials containing primary coolant are wrought stainless steel. The welds in stainless steel pipe are made using the gas tungsten arc (GTAW) process. These materials are very resistant to crack extension. The tensile properties for the leak-before-break evaluation are those found in the Section III Appendices of the ASME Code. During the design stage, the material properties used are based on the ASME Code minimum values. During the as-built reconciliation stage, certified material test report values are reviewed to verify that ASME Code requirements are satisfied.

- For those lines fabricated using non-stainless ferritic materials, the materials used and the associated welds have adequate toughness to demonstrate that leak-before-break criteria are satisfied. The welds are made using the gas tungsten arc (GTAW) process. The tensile properties for the leak-before-break evaluation are obtained from actual material tests. During the design stage, the material properties are based on test results. During the as-built reconciliation stage, certified material test report values are reviewed to verify that the toughness and strength requirements of the ASME Code, Section III are satisfied.

- Potential degradation by erosion, erosion/corrosion and erosion cavitation is examined to provide low probability of pipe failure.

- Wall thicknesses in elbows and other fittings are evaluated to confirm that ASME Code, Section III piping requirements are met as a minimum.

- The as-built condition of the piping and support system is evaluated based on the guidelines in EPRI NP-5630 (Reference 10) and reconciled to the analysis of the leak-before-break criteria based on the design information. The locations and characteristics of the supports, including any gaps between the supports and piping, or other configurations that result in a nonlinear response are included in the as-built evaluation.

- Adjacent structures and components are designed for the safe shutdown earthquake event to provide low probability of indirect pipe failure.

- The piping supports are anchored to reinforced concrete structures, to concrete-filled steel plate structures, or to steel structures anchored to these types of structures. Piping is not supported by masonry block walls.

### 3.6.3.3 Analysis Methods and Criteria

The methods used to develop the bounding analysis curves ~~methods~~ are described in Appendix 3B. Development of the bounding analysis curves provides an evaluation method that is consistent with NRC requirements and guidance.

Analyzable sections run from one terminal end or anchor to another terminal end or anchor. A terminal end is typically a connection to a larger pipe or a component. For the structural analysis, a normally closed valve between pressurized and unpressurized portions of a line is not considered a terminal end. Figure 3.6-3 is a schematic of a portion of a piping system that illustrates the meaning of analyzable segments. In the figure the analyzable portion of the pipe runs from point A to point D.

The leak-before-break evaluation is based on a fracture mechanics stability analysis comparing the selected leakage crack to the critical crack size. The following discussion outlines the analysis method.

~~The analytical steps described in the following~~ The development of leak-before-break bounding analysis curves assume that circumferentially oriented postulated cracks are limiting. ~~At the critical locations, s~~Stability is established by analyzing through-wall flaws.

#### ~~Postulated~~ Leakage Flaw

Through-wall flaws in candidate leak-before break piping systems ~~at selected locations~~ are postulated. The size of the postulated flaws are large enough so that the leakage is detectable with adequate margin, using 10 times the minimum installed leak detection capability when the pipes are subjected to normal operational loads combining by algebraic sum method. That is, the size of the leakage flaw postulated would be expected to have a leak rate 10 times the size of the rated leak rate detection capability.

As noted in subsection 5.2.5, the rated capability of the leak detection systems for the primary coolant inside containment is 0.5 gpm in one hour. The methods used to detect leakage are described in subsection 5.2.5.3. The methods used for primary coolant are the containment sump level, inventory balance, and containment atmosphere radiation. The method used to detect leakage from the main steam and main feedwater line inside containment is the containment sump level. Containment air cooler condensate flow, and containment atmosphere pressure, temperature, and humidity also provide an indication of possible leakage.

**Stability and Critical Flaw Sizes**

The local and global failure mechanisms are evaluated, as appropriate, to provide margin on flaw size and load. The local mode of failure addresses crack tip behavior: blunting, initiation, extension, and instability. The local failure mechanism is evaluated for ferritic ~~nonstainless~~ steel piping systems using the J-integral method. The global mode of failure addresses the behavior of the net section: initial yielding, strain hardening, and plastic hinge formation. The global failure mechanism (limit load method) is evaluated for stainless steel piping with no cast material and GTAW welding. From these evaluations a critical crack size is determined. That is, a crack larger than the critical crack size would have unstable growth characteristics.

**Acceptance Standards**

The results of the preceding evaluations are compared to show that the critical flaw size, which is shown to be stable when the maximum ~~deadweight, themal expansion, pressure, inertial safe shutdown earthquake, and seismic anchor motion~~ loads are combined based on individual absolute values, is at least twice the size (to satisfy margin of 2 on flaw size) of the leakage flaw size. The critical flaw sizes are also shown to be stable when maximum loads are combined by absolute sum. The maximum loads are described in Appendix 3B subsection 3B.3.3. ~~include anticipated static and dynamic loads that are classified as ASME service levels A or B.~~

The torsional moments are not combined with the bending moments since the torsional moment does not have a significant effect on postulated circumferential cracks.

**Bounding Analyses**

~~A series of e~~Evaluations are provided for each different combination of material type, pipe size, pressure, and temperature. These evaluations are used to develop a set of curves of maximum faulted stress versus the corresponding normal stress that satisfy the criteria for leak-before-break. ~~to~~ These curves are used in the design of the piping systems and will be used by the Combined License applicant to verify that the as-built piping satisfies the requirements for leak-before-break.

### 3.6.3.4 Documentation of Leak-before-Break Evaluations

The leak-before-break evaluation is used to support the elimination of dynamic effects of pipe breaks from the loading conditions for the piping analysis. An evaluation of leak-before-break using the as-built configuration of the piping system and supports is required as part of the Design Report of the as-built configuration required to meet ASME Code requirements. Appendix 3B contains a discussion of the bounding analysis methods for the leak-before-break evaluation.

The analysis methods, criteria, and loads used for evaluation of stress in piping systems are outlined in subsections 3.7.3 and 3.9.3. The seismic input bounds the soil design profiles outlined in subsection 3.7.1.4 and Appendicies 2A and 2B. The evaluation also bound soil profiles qualified using site specific evaluations as outlined in subsection 2.5.4.5.5

### 3.6.4 Combined License Information

#### 3.6.4.1 Pipe Break Hazard Analysis

Combined License applicants referencing the AP600 certified design will address as built reconciliation of the pipe break hazards analysis.

#### 3.6.4.2 Leak-before-Break Evaluation

Combined License applicants referencing the AP600 certified design will address: 1) verification that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping excluded from consideration of the dynamic effects of pipe break are bounded by the leak-before-break bounding analysis; ~~and~~ 2) a review of the Certified Material Test Reports or Certifications from the Material Manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied; and 3) complete the leak-before-break evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B.

### 3.6.5 References

1. NUREG/CR-2913, "Two-Phase Jet Loads," January 1983.

2. WCAP-8077, "Ice Condenser Containment Pressure Transient Analysis Methods," March 1977.

3. ASME/ANSI-B31.1, Code for Power Piping, 1989 Addenda to 1989 Edition.

4. ANSI/ANS-58.2-1988, "Design Bases for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture."

5. Moody, F. J., Fluid Reaction and Impingement Loads, paper presented at the ASCE Specialty Conference, Chicago, December 1973.

6. "MULITFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708 (Proprietary) and WCAP-8709 (Nonproprietary), February 1976.

7. WCAP-8252, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," Revision 1, May 1977.

8. EPRI Report, "Piping and Fitting Dynamic Reliability Program, Volume I," (Draft), November 1989.

9. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill Book Company, New York, 1964.

10. EPRI NP-5630, "Guidelines for Piping System Reconciliation" (NCIG-05, Revision 1), May 1988.

11. NUREG-1061, Volume 3, Report of the U. S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks, November 1984.

12. Standard Review Plan 3.6.3, "Leak Before Break Evaluation Procedures," Federal Register, Volume 52, Number 167, Friday, August 28, 1987; Notice (Public Comment Solicited), pp. 32626-32633.

## APPENDIX 3B

## LEAK-BEFORE-BREAK EVALUATION OF THE AP600 PIPING

General Design Criterion 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of conditions associated with normal operation, anticipated transients, and postulated accident conditions. However, the dynamic effects and flooding associated with pipe rupture may be excluded when analysis demonstrates that the probability of fluid system pipe rupture is extremely low. Dynamic effects are not considered for those segments of piping that are shown mechanistically, with a large margin, not to be susceptible to a pipe rupture.

The dynamic effects associated with pipe rupture include effects such as pipe break reaction loads, jets and jet impingement, subcompartment pressurization loads, and transient pipe rupture depressurization loads on other components.

The use of mechanistic pipe break to eliminate evaluation of dynamic effects of pipe rupture includes material selection, inspection, leak detection, and analysis. Subsection 3.6.3 outlines considerations relative to material selection, inspections, and leak detection. Subsection 5.2.5 describes the leak detection system inside containment. This appendix describes the analysis methods used to support the application of mechanistic pipe break to high-energy piping in the AP600.

The analysis and criteria to eliminate dynamic effects of pipe breaks are encompassed in a methodology called leak-before-break (LBB). This methodology has been validated by theoretical investigations and test demonstrations sponsored by the industry and the NRC.

The primary regulatory documents for leak-before-break analyses are General Design Criterion No. 4 (GDC-4), Draft Standard Review Plan 3.6.3 (SRP 3.6.3) (Reference 1), and NUREG-1061, Volume 3 (Reference 2). Although SRP 3.6.3 has been issued only as a draft, its provisions are followed as guidelines to leak-before-break analyses.

Leak-before-break methodology has been applied to the reactor coolant loop and high-energy auxiliary line piping in operating nuclear power plants. The leak-before-break analysis used to support the piping design of the AP600 is an application of the same methodology used in leak-before-beak evaluations previously accepted by the NRC.

In the AP600, leak-before-break evaluations are performed for the reactor coolant loop, the surge line, selected other branch lines containing reactor coolant down to and including 4-inch diameter nominal pipe size, portions of the main steam line, and portions of the main feedwater line. Those lines not qualified to the leak-before-break criteria are evaluated using the pipe rupture protection criteria outlined in subsections 3.6.1 and 3.6.2.

This appendix provides a leak-before-break analysis for the applicable piping systems. Table 3B-1 provides a list of AP600 leak-before-break piping systems.

## 3B.1 Leak-Before-Break Criteria for AP600 Piping

The methodology used for leak-before-break analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2). The steps are:

- Evaluate potential failure mechanisms
- Perform bounding analysis

## 3B.2 Potential Failure Mechanisms for AP600 Piping

In high-energy piping, there are material degradation mechanisms that could adversely affect the integrity of the system as well as its suitability for leak-before-break analysis. The following lists potential degradation (or "failure") mechanisms:

- Erosion-corrosion induced wall thinning
- Stress corrosion cracking (SCC)
- Water hammer
- Fatigue
- Thermal aging
- Thermal stratification
- Other mechanisms

The stainless steel piping is fabricated of SA312TP316LN or SA312TP304L material. The type 304L material is used in the accumulator discharge lines. The main feedwater piping is fabricated of SA335P11 (low alloy steel). The main steam piping is fabricated of SA333 Grade 6. The welds are made by the gas tungsten arc welding (GTAW) method.

The various degradation mechanisms are discussed in the following subsections.

### 3B.2.1 Erosion-Corrosion Induced Wall Thinning

**Primary Loop Piping**

Wall thinning by erosion and erosion-corrosion effects does not occur in the primary loop piping because SA312TP316LN austenitic stainless steel material is highly resistant to these effects. The coolant velocity in the AP600 primary loop is about 43 feet per second, which is lower than the velocity in operating Westinghouse-designed pressurized water reactors. The bend radii in the AP600 hot and cold legs are greater than the bend radii used in the crossover legs of operating plants. There is no record of erosion-corrosion induced wall thinning in the primary loops of operating plants.

**Auxiliary Stainless Steel Piping**

Wall thinning by erosion-corrosion effects does not occur in the auxiliary stainless steel piping because SA312TP316LN and SA304TP304L austenitic stainless materials are highly resistant to these effects. The coolant velocity in these systems is lower than in comparable system velocity in operating Westinghouse-designed pressurized water reactors. There is no record of erosion-corrosion induced wall thinning in the stainless steel piping of operating plants.

**Main Steam Line**

Main steam lines in the AP600 are fabricated from SA333 Grade 6 Carbon steel. Erosion-corrosion induced wall thinning is not expected in the main steam line. Extensive work has been done investigating erosion-corrosion in carbon steel pipes. The main steam line has low susceptibility to erosion due to the relatively high operating temperature. Susceptibility is also low due to the high quality steam in the main steam line.

**Main Feedwater Line**

The feedwater line is fabricated from SA335 P11 low alloy steel. The water chemistry and flow velocities in the feedwater line are controlled to limit the potential for erosion and corrosion. The feedwater piping material has enhanced erosion resistance compared to materials traditionally used in feedwater lines. The alloy steel was modeled utilizing Electric Power Reshcarch Institute's (EPRI's) "CHECMATE" (Reference 3) program to determine erosion-corrosion rates based on AP600 chemistry controls. "CHECMATE" is an EPRI developed computer code which quantifies expected erosion-corrosion rates based on chemistry, material, fluid conditions, and piping configurations. The calculated wear rates provide significant margin for the proposed feedwater line for the 60-year plant life. The corrosion-resistant material used for the feedwater piping extends from the steam generator to the common header pipe in the turbine building.

Based on the above discussion, erosion-corrosion induced wall thinning does not have an adverse effect on the integrity of the AP600 leak-before-break piping systems.

### 3B.2.2 Stress Corrosion Cracking

Stress corrosion cracking is not expected to occur in the AP600 piping systems because the three conditions necessary for stress corrosion cracking to take place are not present. If any of these three conditions is not present, stress corrosion cracking will not take place. The three conditions are:

- There must be a corrosive environment.
- The material itself must be susceptible.
- Tensile stresses must be present in the material.

**Primary Loop Piping**

During plant operation, the reactor coolant water chemistry is monitored and maintained within specific limits (see subsection 5.2.3 for a discussion of reactor coolant chemistry). Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking. The major water chemistry control standards are included in the plant operating procedures as a condition for plant operation.

The key to avoidance of a corrosive environment is control of oxygen. During normal power operation, oxygen concentration in the reactor coolant system is controlled to extremely low levels by controlling charging flow chemistry and maintaining a hydrogen overpressure in the reactor coolant at specified concentrations. Halogen concentration is controlled by maintaining concentrations of chlorides and fluorides within the specified limits. During plant operations, the likelihood of stress corrosion cracking in the primary loop piping systems is very low.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (for example, sulfides, sulfites, and thionates). Pipe cleaning standards prior to operation and careful water chemistry control during plant operation are applied to prevent the occurrence of a corrosive environment. Before being placed in service the piping is cleaned. During flushes and preoperational testing, water chemistry is controlled according to written specifications. Standards on chlorides, fluorides, conductivity, and pH are included in the guidelines for water for cleaning the piping.

The SA312TP316LN austenitic stainless steel chosen for the AP600 is resistant to stress corrosion cracking in a low- or no-oxygen environment. The "L" grades of austenitic stainless steel contain low carbon (less than 0.035 weight percent) which mitigates sensitization.

Design tensile stresses in the reactor coolant loop are within the ASME Code, Section III allowables. Residual tensile stresses are expected in the welds and such stresses are not considered when designing by the ASME Code, Section III because these stresses are self-equilibrating and do not affect the failure loads. The residual stresses should not be more severe than for the operating Westinghouse pressurized water reactor plants (which have not experienced stress corrosion cracking in the primary loop).

The material used for buttering nozzles at the stainless-to-carbon steel safe ends is a high nickel alloy. The nickel-chromium-iron alloy selected and qualified for this application is not susceptible to primary water stress corrosion cracking.

**Auxiliary Stainless Steel Piping**

The discussion above regarding the necessary conditions for primary loop piping stress corrosion cracking is also applicable to the other stainless steel piping of the primary system.

The SA312TP316LN/SA312TP304L austenitic stainless steel chosen for the auxiliary stainless steel piping of the AP600 is resistant to stress corrosion cracking in a low- or no-oxygen environment. The "L" grades of austenitic stainless steel contain low carbon (less than 0.035 weight percent) which mitigates sensitization.

Design tensile stresses in the other stainless steel piping are within the ASME Code, Section III allowables. Residual tensile stresses are expected in the welds; however, the residual stresses should not be more severe than for the operating Westinghouse pressurized water reactor plants (which have not experienced stress corrosion cracking in the auxiliary stainless steel piping).

**Main Steam Line and Main Feedwater Line**

The main steam piping is constructed from ferritic steel. Stress corrosion cracking in ferritic steels commonly result from a caustic environment. A source of a caustic environment in the main steam piping would be moisture carryover from the steam generator. However, the secondary side water treatment utilizes all volatile treatment. All volatile treatment effectively precludes causticity in the steam generator bulk liquid environment. For some operating plants prior to implementing all volatile treatment, the phosphate water treatment caused a caustic chemical imbalance resulting in stress corrosion cracking of steam generator tubing. Under all volatile treatment water treatment conditions, there is no instance of caustic stress corrosion cracking on the ferritic steam lines indicating no significant caustic carryover. The operating secondary side chemistry precludes stress corrosion cracking on the ferritic main steam line.

Stress corrosion cracking is not expected to occur in the main feedwater line piping because of control of the oxygen to very low levels. There has been no experience with stress corrosion cracking in feedwater lines in operating plants of Westinghouse design. The operating secondary side chemistry precludes stress corrosion cracking on the main feedwater line.

Based on the above discussion, stress corrosion cracking does not have an adverse effect on the integrity of AP600 leak-before-break piping systems.

### 3B.2.3 Water Hammer

**Primary Loop Piping**

The reactor coolant loop is designed to operate at a pressure greater than the saturation pressure of the coolant, thus precluding the voiding conditions necessary for water hammer to occur. The reactor coolant primary system is designed for Level A, B, C, and D (normal, upset, emergency, and faulted) service condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening have been

considered in the system design. Other valve and pump actuations cause relatively slow transients with no significant effect on the system dynamic loads.

To provide dynamic system stability, reactor coolant parameters are controlled. Temperature during normal operation is maintained within a narrow range by control rod positioning. Pressure is controlled within a narrow range for steady-state conditions by pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle. The operating transients of the reactor coolant system primary loop piping are such that significant water hammer loads are not expected to occur.

**Auxiliary Stainless Steel Piping**

The passive core cooling system and automatic depressurization system are designed to minimize the potential for water hammer induced dynamic loads. Design features include:

- Continuously sloping core makeup tank and passive residual heat exchanger inlet lines to eliminate local high points
- Inlet diffusers in the core makeup tanks to preclude adverse steam and water interactions
- Vacuum breakers in the discharge lines of the automatic depressurization valves connected to the pressurizer

The AP600 pressurizer spray control valve is similar to what is used in the operating plants. There is no history of water hammer caused by the spray control valve.

The normal residual heat removal system isolation valves are slow closing valves, identical to operating plants, and therefore would not be a source of water hammer.

These features minimize the potential of water hammer in the auxiliary stainless steel piping system.

**Main Feedwater Line**

The feedwater piping, steam generator design details, and other component details in the feedwater system are designed to minimize the potential and severity of water hammer within the feedwater piping. The following addresses each aspect of the design incorporated to minimize water hammer.

Steam Generator Design: The AP600 steam generator design benefits from investigation of water hammer events and the resulting design changes developed to address the events (References 4 through 8).

- Top discharge feed flow through spray tubes (similar to J-tubes) from the feedring reduces the potential of void formation when the steam generator level drops below the

feedring level. Previous steam generator feedring designs had incorporated bottom discharge holes that permitted feedring draining whenever the steam generator level dropped below the feedring.

- Separate startup feedwater and main feedwater nozzles are incorporated to provide for only heated feedwater from the deaerator entering the steam generator via the main feedwater line

- Feedwater nozzle design incorporates a welded thermal liner attached to the feedwater nozzle forging to form a positive seal to limit the potential for feedring drainage and therefore void formation within the feedring. Previous designs had included a "close fit" but not a complete seal at the connection to the nozzle forging. The welded thermal liner design has no leak paths within the steam generator through which the water can drain from the feedring.

Feedwater piping design: The AP600 feedwater piping layout has incorporated features to limit void formation and water hammer initiation.

- A downward facing elbow is connected to the steam generator nozzle and thus complies with industry recommendations to minimize the horizontal feedwater piping connected to the steam generator. The short horizontal section minimizes amount of steam void which can form.

- The main feedwater piping inside containment continuously rises to the steam generator providing for natural venting of the steam generator in the event a steam void is formed.

- Long straight piping runs in the feedwater line are limited.

Component and system design selection:

- A major cause of water hammer problems in pressurized water reactor feedwater systems has been control valve instability. These instabilities resulted from factors such as oversized valve, unbalanced valve trim, damage to valve components, and incompatibility of the feedwater control valve with the rest of the feedwater system. These problems are minimized on AP600 by the following:

  - The specification of specialized valve trim to avoid instability

  - The use of variable speed feedwater pumps to reduce the demands on the control valve requirements

  - Reduced control requirements on the main feedwater control valve by the use of a startup feedwater line that provides feedwater flow control from either the startup feedwater pump or the main feed pump at lower feed demand (power) levels.

- Main feedwater control valve positioning during normal operation is the function of the plant control system (see subsection 7.7.1.8) using a refinement of a standard three-element control scheme. The control scheme provides greater steam generator level stability and thus reduces potential feedwater transients.

- Rapid closure of some types of feedwater check valves may potentially cause water hammer in main feedwater lines. The controlled closure check valve specified for the AP600 main feedwater lines limits the magnitude of the closing loads generated by valve closure caused by depressurization of the feedwater line upstream of the check valve.

- Feedwater delivered to the main feedwater line is drawn only from the deaerator. The heated feedwater is normally at least 250°F and helps reduce the possibility of water hammer.

- Startup feedwater is piped directly to the steam generator. This feature helps prevent the need to introduce cold water directly into the main feedwater and thus minimizes the chances of steam water counterflow or steam bubble collapse type of water hammer events.

- Rapid resumption of feedwater flow to the steam generators is accomplished in the AP600 design. Numerous options are available to maintain or restore steam generator level with the feedwater system design. Based on the flow demand signal and level of feedwater isolation either the main feedwater pump(s) or the startup feedwater pumps can adequately provide level control. If there is no engineered safeguard features feedwater isolation signal present, the main feedwater pumps will provide adequate steam generator inventory control, via the main feedwater line or the startup feedwater line. If a main feedwater isolation signal exists then either the main feedwater pump(s) or the startup feedwater pump(s) provides startup feedwater flow via the startup feedwater line.

The above design provisions make the potential for steam generator water hammer in the feedwater line extremely low. However, with consideration for the main feedwater and steam generator design features, the susceptibility of the main feedwater line inside containment for water hammer has been evaluated. The most common historic causes were evaluated as well as the relevant modes of operation for susceptibility to the appropriate water hammer mechanisms (Reference 4). The limiting anticipated and unanticipated events were evaluated. The results of the analysis demonstrate that the system is acceptable for leak-before-break application.

**Main Steam Line**

The steam lines are not subject to water hammer by the nature of the fluid transported. The following system design provisions address concerns regarding steam hammer within the main steam line and identify the significant dynamic loads included in the main steam piping design.

- Design features that prevent water slug formations are included in the system design and layout. In the main steam system, these include the use of drain pots and the proper sloping of lines.

- The operating and maintenance procedures that protect against a potential occurrence of steam hammer include system operating procedures that provide for slowly heating up (to avoid condensate formation from hotter steam on colder surfaces), operating procedures that caution against fast closing of the main steam isolation valves except when necessary, and operating and maintenance procedures that emphasize proper draining.

- The stress analyses for the safety-related portion of the main steam system piping and components include the dynamic loads from rapid valve actuations, including actuation of the main steam isolation valves and the safety valves.

Based on the above discussion, water hammer does not have an adverse effect on the integrity of AP600 leak-before-break piping systems.

## 3B.2.4 Fatigue

### Low-Cycle Fatigue

Low-cycle fatigue due to normal operation and anticipated transients is accounted for in the design of the piping system. The Class 1 piping systems comply with the fatigue usage requirements of the ASME Code, Section III. The Class 2 and 3 piping systems comply with the stress range reduction factors of the ASME Code, Section III.

A fatigue evaluation at the main feedwater nozzle equivalent to ASME Class 1 piping is performed. Also, a fatigue crack growth analysis at the main feedwater nozzle is performed.

Due to the nature of operating parameters, main steam line piping ~~systems~~ (Class 2) and the Class 3 portion of the accumulator piping, are not subjected to any significant transients to cause low-cycle fatigue.

Based on the above discussion, low-cycle fatigue is not a concern of AP600 leak-before-break piping systems.

### High-Cycle Fatigue

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 (see subsection 5.4.2). The loads from reactor coolant pump vibrations are minimized by criteria for pump shaft vibrations during hot functional testing and operation. During operation, an alarm signals when the reactor coolant pump vibration is greater than the limits.

Main feedwater pump vibration is isolated from the leak-before-break feedwater line inside containment via the piping and equipment supports.

With these precautions taken, the likelihood of leakage due to fatigue in piping systems evaluated for leak-before-break is very small.

### 3B.2.5 Thermal Aging

**Stainless Steel Piping**

Piping used in the reactor coolant loop and other auxiliary lines are wrought stainless steel materials, rather than cast materials, so that thermal aging concerns are not expected for the AP600 piping and fittings. The welds used in the assembly of the AP600 are gas tungsten arc welds (GTAW). These welds are essentially as resistant to the effects of thermal aging as the base metal materials. This is due to the typically low ferrite contact in welds which results in minimal impact from thermal aging. Based on this information, thermal aging of weld materials and piping used in the AP600 is not an issue.

**Main Steam and Main Feedwater Lines**

The main steam and main feedwater piping systems do not have cast materials. The welding process used on these lines is also gas tungsten arc weld (GTAW).

There are no thermal aging concerns for the carbon steel piping of the main steam line and the alloy steel of the main feedwater piping.

The material used for the main steam and main feedwater piping systems is not susceptible to dynamic strain aging effects.

### 3B.2.6 Thermal Stratification

Leak-before-break analyses include consideration of the loads and stresses due to thermal stratification.

Thermal stratification occurs only in a pipe that has a susceptible geometry and low flow velocities. A temperature difference between the flowing fluid and stagnant fluid is also a prerequisite.

The design of piping and component nozzles in the AP600 includes provisions to minimize the potential for and the effects of thermal stratification, cycling, and striping, pursuant to actions requested in several NRC bulletins, as discussed below.

**Primary Loop Piping**

Thermal stratification in the reactor coolant loops resulting from actuation of passive safety features is evaluated as a design transient. Stratification effects due to both Level B and Level D service conditions are considered. The criteria used in the evaluation of the stress in the loop piping due to stratification is the same as that applicable for other Level B and Level D service conditions.

**Auxiliary Stainless Steel Piping**

Pursuant to the actions requested in NRC Bulletin 88-11, the pressurizer surge line is analyzed to demonstrate that the applicable requirements of the ASME Code, Section III are met. This analysis includes consideration of plant operation, thermal stratification, and thermal striping using temperature distributions and transients developed from experience on existing plant monitoring programs.

Pursuant to the actions requested in NRC Bulletin 88-08 (cracking in piping connected to reactor coolant systems due to isolation valve leakage), a systems review of the AP600 piping was performed in accordance with the criteria provided in subsection 3.9.3.1.2.

The candidate leak-before-break lines do not contain unisolable sections ~~of the following line which is evaluated for leak before break has been reviewed and is~~ susceptible to adverse stresses as described in Bulletin 88-08:

~~*Pressurizer spray lines from the cold legs to the pressurizer*~~
~~These lines have a controlled bypass flow around the pressurizer spray valves which maintains the piping at approximately cold leg temperature to prevent thermal shock transients in the spray piping and pressurizer spray nozzle. This flow is expected to stratify with pressurizer steam in the piping near the pressurizer. The stratification loads are considered in the analysis.~~

The unisolable sections of the following lines which are evaluated for lead-before-break have been reviewed and are not susceptible to adverse stresses as described in NRC Bulletin 88-08:

***Direct vessel injection lines from the reactor vessel nozzle up to the accumulator injection valves, core makeup injection valves, in-containment refueling water storage tank injection valves and normal residual heat removal injection valves***
A pressure differential capable of forcing leakage flow into the reactor pressure vessel does not exist, therefore leakage is not a concern.

***Core makeup lines from the cold legs to the core makeup tanks***
These lines will be essentially at reactor coolant system temperature due to convective currents. The potential for leakage is not a concern for these lines, since hot leakage from the reactor coolant system would be entering a hot section of piping.

***Passive residual heat removal (PRHR) line from the hot leg, through the passive residual heat removal heat exchanger, and to the steam generator channel head***

The potential for leakage through the isolation valves is not a concern for the piping extending from the reactor coolant system hot leg connection to the passive residual heat removal heat exchanger inlet, since hot leakage from the reactor coolant system would be entering a hot section of piping. Leakage exiting the passive residual heat removal heat exchanger would not be a concern since the cooled leakage would be entering a cold section of piping. This leakage would then heat up in the piping directly below the steam generator. Any amount of leakage is expected to be small, since the pressure differential across the isolation valves is about 30 psi (the difference between the hot leg and reactor coolant pump suction pressures). Activation of the passive residual heat removal system following a plant scram is not a concern, since stratification will not occur due to the high flow velocity in the passive residual heat removal return flow line.

***Automatic depressurization stage 4 lines from the hot legs to the stage 4 depressurization valves***

Leakage is not a concern since the squib valves are leaktight and other potential leakage flow paths have double isolation.

***Pressurizer safety line from the pressurizer to the safety valve***

This line is steam filled and will not experience stratified loadings.

***Automatic depressurization stage ~~1,~~ 2 and 3 lines from the pressurizer to the depressurization valves***

Leakage is not a concern since double isolation exists in all potential leakage flow paths.

***Normal residual heat removal suction lines from the hot legs to the isolation valves***

The piping from the hot legs to the isolation valves is expected to be essentially at the hot leg temperature during 100 percent power due to turbulent penetration and convective currents which heat the line. Isolation valve leakage is not a concern since hot leakage from the reactor coolant system would be entering a hot section of piping.

***Pressurizer spray lines from the cold legs to the pressurizer***

These lines have a controlled bypass flow around the pressurizer spray valves which maintains the piping at approximately cold leg temperature to prevent thermal shock transients in the spray piping and pressurizer spray nozzle. This flow is expected to stratify with pressurizer steam in the piping near the pressurizer. The stratification loads are considered in the analysis.

**Main Steam Line**

The steam lines are not subjected to thermal stratification by the nature of fluid transported.

### Main Feedwater Line

Thermal stratification is prevented in the main feedwater line based on the flow rate limitations within the main feedwater and startup feedwater line and the flow control stability for feedwater control. Low feedwater flow duty is provided by the startup feedwater line while higher feedwater flow rates are provided and controlled via the main feedwater line. Subsection 10.4.7.2.3 provides details of the automatic switchover between main and startup feedwater lines. The switchover between the feedwater lines occurs above a minimum flow rate to prevent thermal stratification for limiting temperature deviations.

Main feedwater control valve positioning during normal operation is the function of the plant control system (see subsection 7.7.1.8). The control scheme enhances steam generator level stability and thus reduces potential feedwater thermal stratification resulting from temporary low flow transients.

For additional information about augmented inspection of the feedwater line see subsection 3B.8.

For additional information about stratification, refer to subsection 3.9.3.

Based on the above discussion, thermal stratification does not have an adverse effect on the integrity of AP600 leak-before-break piping systems.

## 3B.2.7 Other mechanisms

The pipe evaluated for leak-before-break does not operate at temperature for which creep fatigue must be considered. Creep fatigue is a concern for ferritic steel piping operation at temperatures above 700°F and for austenitic stainless steel operation above 800°F.

Pipe degradation or failure by indirect causes such as fires, missiles, and component support failures is precluded by criteria for design, fabrication, inspection, and separation of potential hazards in the vicinity of the safety-related piping. The structures, larger pipe, and components in the vicinity of pipe evaluated for leak-before-break are safety-related and seismically designed or are seismically supported if nonsafety-related.

Cleavage type failures are not a concern for systems operating temperature and material used in the stainless steel piping systems. The material used in the main steam and main feedwater lines are highly ductile and resistant to cleavage type failure at operating temperatures. The resistance to failure have been demonstrated by material fracture toughness tests.

## 3B.3 Leak-Before-Break Bounding Analysis

The methodology used for performing the bounding analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2).

Bounding leak-before-break analysis for the applicable AP600 piping systems is performed. The analysis criteria and development techniques of the bounding analysis curves (BAC) are described below. The bounding analysis curve allows for the evaluation of the piping system in advance of the final piping analysis, incorporating leak-before-break considerations early in the piping design process. The leak-before-break bounding analysis curve is used to evaluate critical points in the piping system. A minimum of two points are required to develop the bounding analysis curve. One point for the low normal stress case and the other point for the high normal stress case. If variations in pipe size, material, pressure or temperature occur for a specific piping system, an additional bounding analysis curve is generated. These points meet the following margins for leak-before-break analysis: (References 1 and 2).

- Margin of 10 on leak detection capability
- Margin of 2 on flaw size
- Establish margin of 1 on load by using absolute combination method of maximum loads

### 3B.3.1 Procedure for Stainless Steel Piping

#### 3B.3.1.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- ~~Obtain p~~Piping materials - 316LN/304L, Type 304L is used for the accumulator discharge line

- ~~Determine n~~Normal operating temperature

- ~~Determine n~~Normal operating pressure

- ~~Determine p~~Pipe outside diameter

- ~~Determine p~~Pipe thickness

- ~~Determine~~ The number of bounding analysis curves needed for each analyzable piping system is determined by a review of the combinations of the following parameters:

  ~~Parameters~~

  - Pipe size
  - Pipe schedule
  - Operating pressures (100 percent power and maximum stress condition)
  - Operating temperatures (100 percent power and maximum stress condition)

### 3B.3.1.2 Pipe Physical Properties ~~Calculation Steps~~

The physical and metallurgical properties for each of the lines are determined in the following manner

- ~~Calculate m~~Minimum wall thickness is calculated at the weld counterbore
- ~~Calculate~~ The area (A) and section modulus (Z) are calculated using minimum wall thickness
- ~~Obtain~~ The yield strength is the ASME ~~(Reference 9)~~ Code, Section III (Reference 9) minimum value, at temperature of interest
- ~~Obtain~~ The ultimate strength is the ASME ~~(Reference 9)~~ Code, Section III (Reference 9) minimum value, at temperature of interest
- ~~Obtain~~ The modulus of elasticity is the ASME Code, Section III (Reference 9) at temperature of interest

### 3B.3.1.3 Low Normal Stress Case (Case 1)

To determine the first point of the bounding analysis curve the following steps are used.

- Calculate axial force $F_P$ (for normal operating pressure)

- Assume a lower magnitude of bending stress. The magnitude selected is a very small number that is lower than the expected minimum bending stress.

- Calculate bending moment = (bending stress) x (section modulus)

- Calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability (for 0.5 gpm leak detection capability, this is 10 x 0.5 = 5 gpm)

- Perform the stability analysis using the limit load methodology to obtain the critical flaw size. For AP600 piping systems, there is no cast material and the weld process is gas tungsten arc welds (Z factor is 1.0 since weld process is gas tungsten arc welds, Reference 1.)

  - Determine the maximum loads for a critical flaw size of twice the leakage flaw size. The margin of 2 on flaw size is satisfied.

- Calculate the low normal stress and corresponding maximum stress by using:

$$\text{Stress} = \frac{\text{Axial Force}}{\text{Area}} + \frac{\text{Bending Moment}}{\text{Section Modulus}} \tag{3B-1}$$

### 3B.3.1.4 High Normal Stress Case (Case 2)

To determine the other endpoint of the bounding analysis curve the following steps are used.

- Axial force $F_p$ is calculated as above ~~(~~for normal operating pressure~~)~~
- Assume a higher magnitude of bending stress to get higher bending moment. The magnitude of bending selected such that the corresponding maximum stress generated is close to the flow stress.
- Calculate bending moment = (bending stress) x (section modulus)
- Repeat leakage flaw size and stability calculations as outlined for the low normal stress case above

Note: For an intermediate point, calculation steps are the same as low normal or the high normal case.

### 3B.3.1.5 Develop the Bounding Analysis Curve

- For Case 1, normal and maximum stresses are established.
- For Case 2, normal and maximum stresses are established.
- Plot these two points with normal versus maximum stress. The curve is generated by joining these two points in a straight line. More than two points may be used if desired, to obtain a smooth curve fit between the calculated points. A typical curve is shown in Figure 3B-1.

### 3B.3.2 Procedure for Non-Stainless Steel Piping

The procedure to develop the bounding analysis curve for the ~~(~~Alloy steel for main feedwater lines and carbon steel for main steam lines~~)~~ is similar to that for the stainless steel and is described below.

### 3B.3.2.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- ~~Determine p~~Piping materials
- ~~Determine n~~Normal operating temperature
- ~~Determine n~~Normal operating pressure
- ~~Determine p~~Pipe outside diameter
- ~~Determine p~~Piping thickness

The number of bounding analysis curves needed for each analyzable piping system is determined by a review of the combinations of the following parameters:

~~Parameters~~

- Pipe size
- Pipe schedule
- Operating pressures (100 percent power and maximum stress condition)
- Operating temperatures (100 percent power and maximum stress condition)

### 3B.3.2.2 Calculations Steps

- ~~Calculate~~ The minimum wall thickness is calculated at the weld counterbore
- ~~Calculate~~ The area (A) and section modulus (Z) are calculated using minimum wall thickness
- ~~Obtain t~~The material yield strength, ultimate strength, modulus of elasticity, stress-strain curves, and J-R curves are determined from the material tests

### 3B.3.2.3 Low Normal Stress Case (Case 1)

To determine the first point of the bounding analysis curve the following steps are used.

- Calculate axial force $F_p$ (for normal operating pressure)
- Assume a lower magnitude of bending stress
- Calculate bending moment = (bending stress) x (section modulus)
- Calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability (for 0.5 gpm leak detection capability, this is 10 x 0.5 = 5 gpm)
- Stability analysis
  - Perform J-integral analysis
  - Determine the maximum loads for a critical flaw size of twice the leakage flaw size by satisfying the stability criteria. The margin of 2 on flaw size is satisfied.
- Stability criteria
  - $J_{applied} \leq J_{IC}$

- If $J_{applied} > J_{IC}$ then
- $J_{applied} < J_{max}$ and $T_{applied} < T_{mat}$

- Calculate the low normal stress and corresponding maximum stress by using:

$$\text{Stress} = \frac{\text{Axial Force}}{\text{Area}} + \frac{\text{Bending Moment}}{\text{Section Modulus}}$$

### 3B.3.2.4 High Normal Stress Case (Case 2)

To determine the other endpoint of the bounding analysis curve the following steps are used.

- Axial force $F_p$ is calculated above (for normal operating pressure)
- Assume a higher magnitude of bending stress to get higher bending moment
- Calculate bending moment = (bending stress) x (section modulus)
- Repeat leakage flaw size and stability calculations as outlined for the low normal stress case above

Note: For an intermediate point, calculation steps are the same as low normal or the high normal case.

### 3B.3.2.5 Develop the Bounding Analysis Curve

Follow steps as outlined for the stainless steel case in subsection 3B.3.1.5.

### 3B.3.3 Evaluation of Piping System Using Bounding Analysis Curves

To evaluate the applicability of leak-before-break, the results of the pipe stress analysis are compared to the bounding analysis curve. The cCritical location ~~for each bounding analysis curve~~ is the location of highest maximum stress as determined by the pipe stress results. ~~Determine~~ A comparison is made with the applicable bounding analysis curves ~~needed~~ for ~~each~~ the analyzable piping systems. :As outlined in 3B.3.1.1 and 3B.3.2.1, bounding analysis curves are calculated for different combinations of pipe size, pipe schedule, operating pressures, operating temperatures.

The bounding analysis curves are used during the layout and design of the piping systems to provide a design that satisfies leak-before-break criteria. In addition, the Combined License holder compares the results of the as-built piping analysis reconciliation to the bounding analysis curves to verify that the fabricated piping systems satisfies leak-before-break criteria.

See subsection 3.6.4.2 for the Combined License information item associated with this verification.

~~Parameters~~

- ~~Pipe size~~
- ~~Pipe schedule~~
- ~~Operating pressures~~
- ~~Operating temperatures~~
- ~~Loading conditions~~

~~Determine~~ At the critical location, load combinations for the maximum stress calculation use the (absolute sum method.) The load combinations include the following combinations. ~~Examples:~~

(1) |Pressure| + |Deadweight| + |Thermal (100% Power)| + |Safe Shutdown Earthquake|

(2) |Pressure| + |Deadweight| + |Thermal (100% Power)| + |Valve Thrust Maximum*|

(3) |Pressure| + |Deadweight| + |Thermal Maximum*|

(4) |Pressure| + |Deadweight| + |Pipe Break**|; for main feedwater line only

(5) |Pressure| + |Deadweight| + |Thermal (100% Power)| + |Water Hammer Loads***|; for main feedwater line only

* Level A and Level B of ASME Code load conditions. Valve thrust maximum includes anticipated water hammer events resulting from rapid valve closure or opening, including pressurizer safety valve opening (Level C). Thermal maximum includes applicable stratification loads.
** Main feedwater pipe break in the turbine building
***Includes unanticipated water hammer events (vapor pocket collapse during feedwater refilling is the limiting unanticipated event).

~~Determine corresponding~~ The normal stress is calculated using the (algebraic sum method) at critical location and the following load combination.

(1) Pressure + Deadweight + Thermal (100% Power)

### 3B.3.3.1 Calculation of Stresses

The stresses due to axial loads and bending moments are calculated by the following equation:

where:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{3B-2}$$

$\sigma$ = stress
F = axial load
M = bending moment
A = cross-sectional area
Z = section modulus

The bending moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_Y^2 + M_Z^2} \tag{3B-3}$$

where,

M = bending moment for required loading
$M_Y$ = Y component of bending moment
$M_Z$ = Z component of bending moment
The Y and Z-axes are lateral axes to the X-axis which is the axial axis

The axial load and bending moments for the normal case and maximum case are computed by the methods shown below.

### 3B.3.3.2 Normal Loads

The normal operating loads are calculated by the following equations:

$$F = F_{DW} + F_{Th} + F_P \tag{3B-4}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{Th} \tag{3B-5}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{Th} \tag{3B-6}$$

The subscripts of the above equations represent the following load cases:

DW = deadweight

Th = normal thermal expansion (100 percent power)
P = load due to internal pressure

The method of combining loads is often referred to as the ***algebraic sum method***.

Calculate the normal stress at the critical location.

### 3B.3.3.3 Maximum Loads

For the maximum case, the ***absolute summation method*** of load combination is applied which results in higher magnitude of the combined loads. Since stability is demonstrated using these loads, the leak-before-break margin on loads is satisfied. An ***example*** of the absolute summation expressions are shown below:

$$F = |F_{DW}| + |F_{Th}| + |F_{P}| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3B\text{-}7)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{Th}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3B\text{-}8)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{Th}| + |(M_Z)_{SSEINERTIA}| + |(M_z)_{SSEAM}| \quad (3B\text{-}9)$$

where subscripts SSE, Inertia and AM mean safe shutdown earthquake, inertia and anchor motion respectively.

### 3B.3.3.4 Bounding Analysis Curve Comparison

To compare the stress results with the bounding analysis curve the following process is followed. ~~Calculate t~~The normal and maximum stress at the critical location are calculated by using the~~se~~ loads defined in subsection 3B.3.3. Plot the normal stress versus maximum stress on the bounding analysis curve for the specified system. If the point is on or below the bounding analysis curve, the leak-before-break analysis and margins are satisfied. If the point falls above the bounding analysis curve, the leak-before-break analysis criteria are not satisfied and the pipe layout or support configuration needs to be revised to meet the leak-before-break bounding analysis. Figure 3B-1 shows a typical bounding analysis curve.

### 3B.3.4 Bounding Analysis Results

Table 3B-2 shows a summary of piping systems and corresponding bounding analysis figures. Figures 3B-2 to 3B-40 show the bounding analysis curves. The curves satisfy the margins as indicated in subsection 3B.3.

## 3B.4 Differences in Leak-Before-Break Analysis for Stainless Steel and Ferritic Steel Pipe ~~Class 1 and Class 2 Systems~~

The significant difference between leak-before-break analysis performed for the stainless steel (Class 1 and Class ~~3~~2) systems and the ferritic steel in the Class 2 systems is in the stability analysis. In the case of stainless steel ~~Class 1~~ systems, stability analyses are performed by limit load approach ~~since the piping material is stainless steel~~. In the ferritic steel ~~Class 2~~ systems, stability analyses are performed by J-integral approach.

## 3B.5 Differences in Inspection Criteria for Class 1, 2, and 3 Systems

Class 1, 2 and 3 systems are subjected to in-service inspection requirements from ASME Code, Section XI. For Class 1 piping, terminal ends and dissimilar metal welds are volumetrically inspected, along with other locations, to total 25 percent of the welds. For Class 2 piping, the requirement is to volumetrically inspect the terminal ends and other locations to total 7.5 percent of the welds. For Class 3 systems (the only Class 3 piping is in the accumulator line which is always at room temperature), the system receives periodic visual examinations in conjunction with pressure testing. These requirements were developed by ASME Code, Section XI consistent with the different safety classes of these systems.

The leak-before-break evaluations are based on the ability to detect a potential leaking crack; not the ability to find cracks by inservice inspections. The criteria or methods of the leak-before-break evaluations are the same for ASME Code Class 1, 2, and 3.

## 3B.6 Differences in Fabrication Requirements of ASME Class 1, ~~and~~ Class 2, and Class 3 Piping

The significant difference among ~~between~~ Class 1, 2 and 3 ~~2 large diameter~~ seamless pipe occurs in the nondestructive examination requirements. The Class 1 seamless pipe examination requirements include an ultrasonic testing examination, whereas Class 2 and 3 do~~es~~ not. In addition, the Class 1 examination requirements for a circumferential butt welded joint include ~~a~~ radioagraphic testing and magnetic particle or liquid penetrant examination where Class 2 does not. The examination requirements for Class 2 pipe require radiographic examination of the welds and normally Class 3 pipe does not. As noted in subsection 3.2.2.5, for Class 3 lines required for emergency core cooling functions, radiography will be conducted on a random sample of welds. The Class 3 leak-before-break lines are included in the lines that are radiographed.

For the fabrication of welds in the Class 1, ~~and~~ Class 2 and Class 3 pipes there is no significant differences.

The~~se~~ differences in fabrication and nondestructive examination requirements do not affect the leak-before-break analyses assumptions, criteria, or methods.

## 3B.7 Monitoring of Unanticipated Dynamic Loads in the Feedwater Lines

Instrumentation for monitoring unanticipated dynamic loads in the feedwater lines inside containment will be provided in the first plant.

## 3B.8 Augmented In-Service Inspection at the Main Feedwater Nozzles Connected to the Steam Generators

Augmented in-service inspection (100 percent volumetric inspection every 10 years of inspection interval) at the weld connecting the piping to the steam generator feedwater nozzles ~~connected to the steam generators~~ will be performed.

## 3B.9 References

1. Standard Review Plan 3.6.3, "Leak Before Break Evaluation Procedures," Federal Register, Volume 52, Number 167, Friday, August 28, 1987; Notice (Public Comment Solicited), pp. 32626-32633.

2. NUREG-1061, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, (prepared by the Pipe Break Task Group), November 1984.

3. "CHECMATE (TM) Computer Program User's Manual," EPRI NSAC-145L, April 1989.

4. EPRI Report EPRI NP-6766 "Water Hammer Prevention, Mitigation and Accommodation," July 1992.

5. NUREG/CR-2069 "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants," May 1992.

6. NUREG-1190, "Loss of Power and Water Hammer Event at San Onofre, Unit 1 on November 21, 1985," January 1986.

7. NUREG-0918, "Prevention and Mitigation of Steam Generator Water Hammer Events in PWR Plants."

8. NUREG-0800/ABS 10-2 (Branch Technical Paper), "Design Guidelines for Avoiding Water Hammers in Steam Generators."

9. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components; Division 1 - Appendices," 1989 Edition, July 1, 1989.

Table 3B-1

## AP600 LEAK-BEFORE-BREAK SCOPE OF PIPING SYSTEMS

| Number | System | Pipe Diameter (Inch -Nominal) | Material |
|---|---|---|---|
| 1 | Primary Loop | 31 ID & 22 ID | SA312 Type 316 LN |
| 2 | Main Steam A & B | 32 | SA333 Grade 6 |
| 3 | Main Feedwater A & B | 16 | SA335 P11 |
| 4 | Pressurizer Surge Line | 18 | SA312 Type 316 LN |
| 5 | Automatic Depressurization System Stage 2, 3, and Safety | 6, 8, & 14 | SA312 Type 316 LN |
| 6 | Normal Residual Heat Removal | 10, 12, & 20 | SA312 Type 316 LN |
| 7 | Passive Residual Heat Removal Return | 10 | SA312 Type 316 LN |
| 8 | Passive Residual Heat Removal Supply/Automatic Depressurization System Stage 4 (West) | 10 & 12 | SA312 Type 316 LN |
| 9 | Automatic Depressurization System Stage 4 (East) | 10 & 12 | SA312 Type 316 LN |
| 10 | Direct Vessel Injection A & B<br>(Accumulator Discharge line) | 6 & 8<br>8 | SA312 Type 316 LN<br>or Type 304 L |
| 11 | Core Makeup Tank A & B | 8 | SA312 Type 316 LN |
| 12 | Pressurizer Spray | 4 | SA312 Type 316 LN |
| ~~13~~ | ~~Automatic Depressurization System Stage 1~~ | | |

Table 3B-2

## AP600 LEAK-BEFORE-BREAK BOUNDING ANALYSIS SYSTEMS AND CORRESPONDING FIGURES

| System | Figure Number |
|---|---|
| Primary Loop | 3B-2, 3B-3, 3B-4, 3B-5, 3B-6, 3B-7 |
| Main Steam A & B | 3B-8, 3B-9 |
| Main Feedwater A & B | 3B-10 |
| Pressurizer Surge Line | 3B-11, 3B-12 |
| Automatic Depressurization System Stage 2,3/Safety | 3B-13, 3B-14, 3B-15 |
| Normal Residual Heat Removal | 3B-16, 3B-17, 3B-18 |
| Passive Residual Heat Removal Return | 3B-19, 3B-20, 3B-21, 3B-22 |
| Passive Residual Heat Removal Supply/Automatic Depressurization System Stage 4 (West) | 3B-16, 3B-17, 3B-23, 3B-24 |
| Automatic Depressurization System Stage 4 (East) | 3B-16, 3B-17, 3B-23, 3B-24 |
| Direct Vessel Injection A & B | 3B-25, 3B-26, 3B-27, 3B-28, 3B-29, 3B-30, 3B-31, 3B-32, 3B-33, 3B-34 |
| Core Makeup Tank A & B | 3B-27, 3B-35 |
| Pressurizer Spray | 3B-36, 3B-37, 3B-38, 3B-39 |
| ~~Automatic Depressurization System Stage 1~~ | ~~3B-40~~ |