WCAP-10690

TECHNICAL BASES FOR ELIMINATING LARGE PRIMARY LOOP PIPE RUPTURE AS A STRUCTURAL DESIGN BASIS FOR CALLAWAY AND WOLF CREEK PLANTS

October 1984

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1.0 INTRODUCTION

1.1 Purpose

This report applies to the Callaway and Wolf Creek plants reactor coolant system primary loop piping. It is intended to demonstrate that specific parameters for the two plants are enveloped by the generic analysis performed by Westinghouse in WCAP-9558, Revision 2 (Reference 1) (i.e., the reference report) and accepted by the NRC (Reference 2).

1.2 Scope

The current structural design basis for the Reactor Coolant System (RCS) primary loop requires that pipe breaks be postulated as defined in the approved Westinghouse Topical Report WCAP-8082 (Reference 3). In addition, protective measures for the dynamic effects associated with RCS primary loop pipe breaks have been incorporated in the designs of the Callaway and Wolf Creek plants. However, Westinghouse has demonstrated on a generic basis that RCS primary loop pipe breaks are highly unlikely and should not be included in the structural design basis of Westinghouse plants (see Reference 4). In order to demonstrate the applicability of the generic evaluations to the Callaway and Wolf Creek plants, Westinghouse has performed a comparison of the loads and geometry of the Callaway and Wolf Creek plants with the envelope parameters used in the generic analyses (Reference 1), a fracture mechanics evaluation, a determination of leak rates from through-wall cracks, a fatigue crack growth evaluation, and an assessment of margins.

1.3 Objectives

The conclusions of WCAP-9558, Revision 2 (Reference 1) support the elimination of RCS primary loop pipe breaks for the Callaway and Wolf Creek units. In order to validate this conclusion the following objectives must be achieved.

a. Demonstrate that the parameters of the Callaway and Wolf Creek plants are enveloped by generic Westinghouse studies.

b. Demonstrate that margin exists between the critical crack size and a postulated crack which yields a detectable leak rate.

c. Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of the Callaway and Wolf Creek plants.

d. Demonstrate that fatigue crack growth is negligible.

1.4 Background Information

Westinghouse has performed considerable testing and analysis to demonstrate that RCS primary loop pipe breaks can be eliminated from the structural design basis of all Westinghouse plants. The concept of eliminating pipe breaks in the RCS primary loop was first presented to the NRC in 1978 in WCAP-9283 (Reference 5). That Topical Report employed a deterministic fracture mechanics evaluation and a probabilistic analysis to support the elimination of RCS primary loop pipe breaks. This approach was then used as a means of addressing Generic Issue A-2 and Asymmetric LOCA Loads.

Westinghouse performed additional tests and analyses to justify the elimination of RCS primary loop pipe breaks. As a result of this effort, WCAP-9558, Revision 2, WCAP-9787, and Letter Report NS-EPR-2519 (References 1, 6, and 7) were submitted to the NRC.

The NRC funded research through Lawrence Livermore National Laboratory (LLNL) to address this same issue using a probabilistic approach. As part of the LLNL research effort, Westinghouse performed extensive evaluations of specific plant loads, material properties, transients, and system geometries to demonstrate that the analysis and testing previously performed by Westinghouse and the research performed by LLNL applied to all Westinghouse plants including Callaway and Wolf Creek (References 8 and 9). The results from the LLNL study were released at a March 28, 1983 ACRS Subcommittee meeting. These studies, which are applicable to all Westinghouse plants east of the Rocky Mountains, determined the mean probability of a direct LOCA (RCS primary loop pipe break) to be 10^{-10} per reactor year and the mean probability of an indirect LOCA to be 10^{-7} per reactor year. Thus, the results previously obtained by Westinghouse (Reference 5) were confirmed by an independent NRC research study.

Based on the studies by Westinghouse, LLNL, the ACRS, and the AIF, the NRC completed a safety review of the Westinghouse reports submitted to address asymmetric blowdown loads that result from a number of discrete break locations on the PWR primary systems. The NRC Staff evaluation (Reference 2) concludes that an acceptable technical basis has been provided so that asymmetric blowdown loads need not be considered for those plants that can demonstrate the applicability of the modeling and conclusions contained in the Westinghouse response or can provide an equivalent fracture mechanics demonstration of the primary coolant loop integrity.

This report will demonstrate the applicability of the Westinghouse generic evaluations to Callaway and Wolf Creek units.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

The Westinghouse reactor coolant system primary loop has an operating history that demonstrates the inherent stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 400 reactor-years, including five plants each having 15 years of operation and 15 other plants each with over 10 years of operation.

2.1 Stress Corrosion Cracking

For the Westinghouse plants, there is no history of cracking failure in the reactor coolant system loop piping. For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, a susceptible material, and a corrosive environment (Reference 10). Since some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as other materials in the system, applicable ASME Code rules, fracture toughness, welding, fabrication, and processing.

The environments known to increase the susceptibility of austenitic stainless steel to stress corrosion are (Reference 10): oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. External cleaning for Class 1 stainless steel piping includes patch tests to monitor and control chloride and fluoride levels. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications.

Requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is expected to be less than 0.005 ppm by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. This is assured by controlling charging flow chemistry and specifying proper wetted surface materials.

2.2 Water Hammer

Overall, there is a low potential for water hammer in the RCS since its design and operation preclude the voiding condition in normally filled lines. The reactor coolant system, including piping and primary components, is designed for normal, upset, emergency, and faulted 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 are considered in the system design. Other valve and pump actuations result in relatively slow transients with no significant effect on the system's dynamic loads. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by control rod position; pressure is controlled by pressurizer heaters and pressurizer spray also within a narrow range for steady-state conditions. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping are such that no significant water hammer can occur.

2.3 Low Cycle and High Cycle Fatigue

Low cycle fatigue considerations are taken into account in the design of the piping system through the fatigue usage factor evaluation to show compliance with the rules of Section III of the ASME Code. A further evaluation of the low cycle fatigue loadings was carried out as part of this study in the form of a fatigue crack growth analysis, as discussed in Section 6.

High cycle fatigue loads in the system would result primarily from pump vibrations. These are minimized by restrictions placed on shaft vibrations during hot functional testing and operation. During operation, an alarm signals the exceedance of the vibration limits. Field measurements have been made on a number of plants during hot functional testing, including plants similar to Callaway and Wolf Creek. Stresses in the elbow region below the reactor coolant pump have been found to be very small, between 2 and 3 ksi at the highest. These stresses are well below the fatigue endurance limit for the material and would also result in an applied stress intensity factor below the threshold for fatigue crack growth.

3.0 PIPE GEOMETRY AND LOADING

The loop weld locations for Callaway and Wolf Creek are identified in Figure 3-1. The material properties and the loads at these locations resulting from deadweight, thermal expansion, and Safe Shutdown Earthquake (SSE) are provided in Table 3-1. The primary loop material is cast type SA-351-CF8A. As seen from this table, the junction of the hot leg and the reactor vessel outlet nozzle (Location 1) is the most limiting location for crack stability analysis based on the highest stress due to the combined pressure, deadweight, thermal expansion, and SSE loadings. A segment of the primary coolant hot leg pipe is sketched in Figure 3-2. This segment is postulated to contain a circumferential through-wall flaw. This location will be referred to as the critical location. The inside diameter and the wall thickness of the pipe are 29.2 and 2.37 inches, respectively. At this location, the axial force (F_x) and the bending moment (M_b) are [$]^{a,c,e}$ (including the axial force due to pressure) and [$]^{a,c,e}$, respectively. The pipe is subjected to a normal operating pressure of 2235 psig. The method for calculating the loads found in Table 3-1 is described below.

The axial force F and transverse bending moments, M_y and M_z, are chosen for each static load (pressure, deadweight, and thermal) based on elastic-static analyses for each of these load cases. These pipe load components are combined algebraically to define the equivalent pipe static loads F_s, M_{ys}, and M_{zs}. Based on elastic SSE response spectra analyses, amplified pipe seismic loads, F_d, M_{yd}, M_{zd}, are obtained. The maximum pipe loads are obtained by combining the static and dynamic load components as follows:

$$F_x = |F_s| + |F_d|$$

$$M_b = \sqrt{M_y^2 + M_z^2}$$

where:

$$M_y = |M_{ys}| + |M_{yd}|$$

$$M_z = |M_{zs}| + |M_{zd}|$$

The corresponding geometry and loads used in the reference report (Reference 1) are as follows: inside diameter and wall thickness are 29.0 and 2.5 inches; axial load and bending moment are [ ]^{a,c,e} The outer fiber stress for Callaway and Wolf Creek is [ ]^{a,c,e} while in the reference report it is [ ]^{a,c,e} This demonstrates conservatism in the reference report which makes it more severe than the Callaway and Wolf Creek analyses.

The normal operating loads (i.e., algebraic sum of pressure, deadweight, and 100 percent power thermal expansion loading) at the critical location, i.e., the junction of the hot leg and the reactor vessel outlet nozzle, are as follows:

F = [ ]^{a,c,e} (including internal pressure)
M = [ ]^{a,c,e}

The calculated and allowable stresses for ASME Code Section III, NB-3600 equation 9 (faulted, i.e., pressure, deadweight, and SSE) and equation 12 (thermal) at the critical location are as follows:

Equation Number	Calculated Stress (ksi)	Allowable Stress (ksi)	Ratio of Calculated/ Allowable
	a,c,e		a,c,e
9F	[ ]	53.4	[ ]
12	[ ]	53.4	[ ]

TABLE 3-1

PRIMARY LOOP DATA FOR CALLAWAY AND WOLF CREEK

Faulted Loads

a,c,e

a Includes internal pressure

b Critical Location

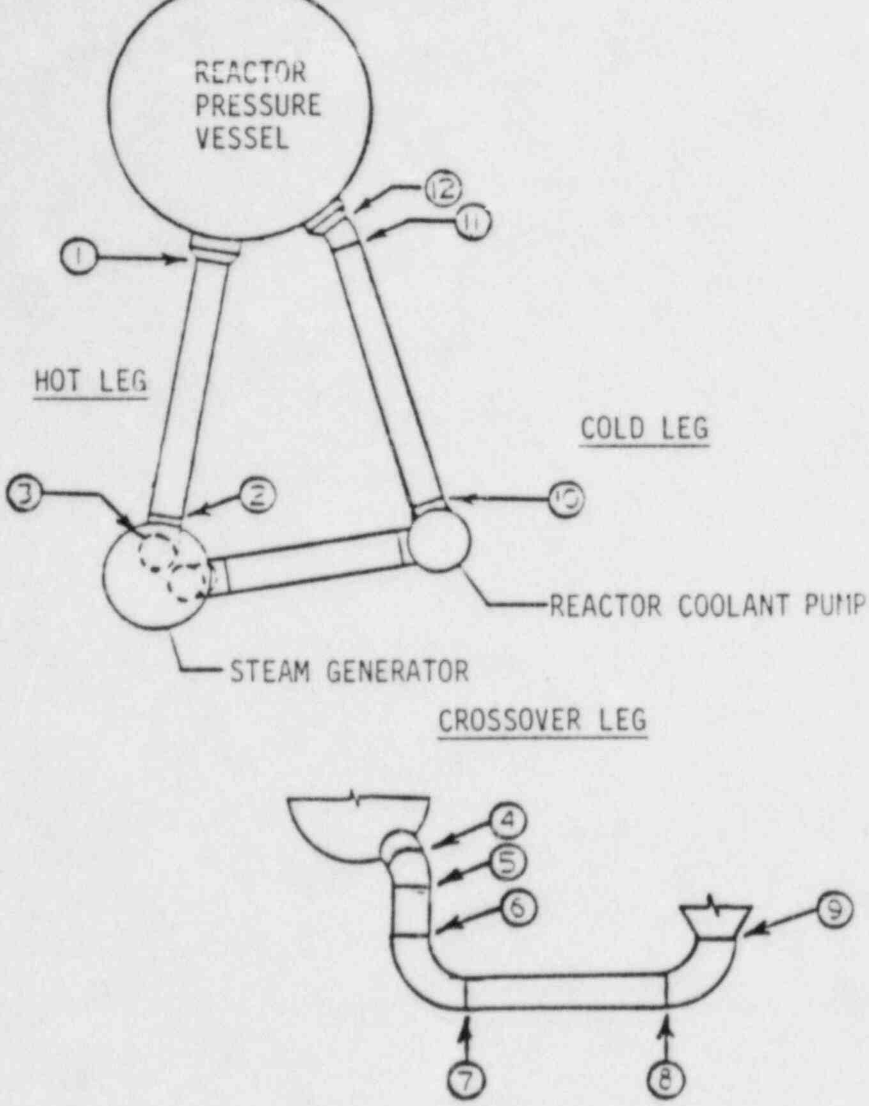


FIGURE 3-1 Schematic Diagram of Primary Loop Showing Weld Locations for Callaway and Wolf Creek Units

HOT LEG

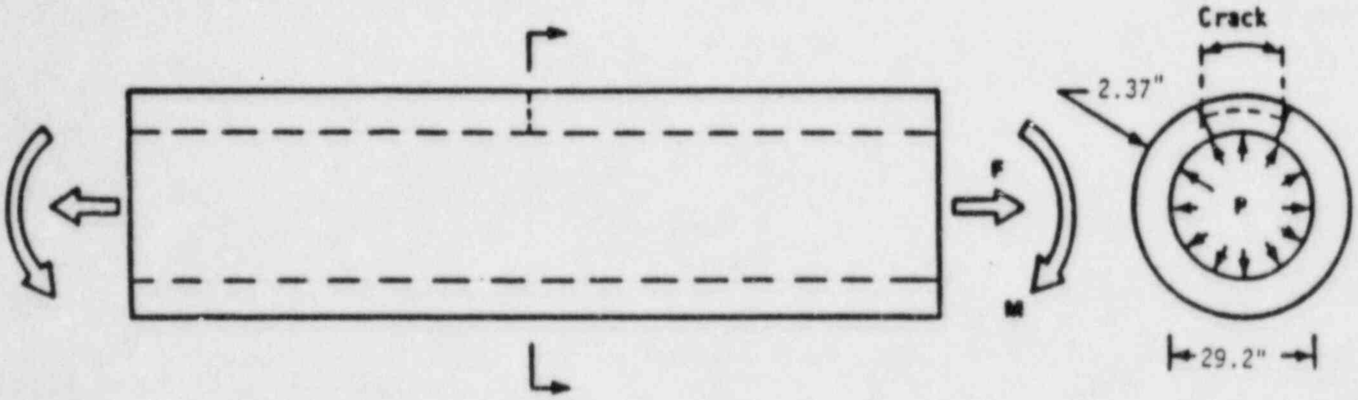
Temperature: 619°F; Pressure: 2235 psig

CROSSOVER LEG

Temperature: 558°F; Pressure: 2190 psig

COLD LEG

Temperature: 558°F; Pressure: 2290 psig



P = 2,235 psig

[ ]^{a,c,e}

FIGURE 3-2 Reactor Coolant Pipe

4.0 FRACTURE MECHANICS EVALUATION

4.1 Global Failure Mechanism

Determination of the conditions that lead to failure in stainless steel must be done with plastic fracture mechanics methods because of the large amount of deformation accompanying fracture. A conservative method for predicting the failure of ductile material is the [

$]^{a,c,e}$ This methodology has been shown through a large number of experiments to be applicable to ductile piping and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring [

$]^{a,c,e}$ (Figure 4-1) when loads are applied. The detailed development is provided in Appendix A for a through-wall circumferential flaw in a pipe with internal pressure, axial force, and imposed bending moments. The [$]^{a,c,e}$ for such a pipe is given by:

$$[\qquad\qquad\qquad]^{a,c,e}$$

where:

$$\left[\qquad\qquad\qquad\qquad \right]^{a,c,e}$$

[

]^{a,c,e}

The analytical model described above accurately accounts for the piping internal pressure as well as the imposed axial force as they affect [ ]^{a,c,e} Good agreement was found between the analytical predictions and the experimental results (Reference 11).

4.2 Local Failure Mechanism

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension, and finally crack instability. Depending on the material properties and geometry of the pipe, flaw size, shape and loading, the local failure mechanisms may or may not govern the ultimate failure.

The stability will be assumed if the crack does not initiate at all. It has been accepted that the initiation toughness measured in terms of J_{IN} (i.e., J_{Ic})[a] from a J-integral resistance curve is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than J_{IN} of the material, then the crack will not initiate. If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation:

---

a The notation J_{IN} instead of J_{Ic} was used in Reference 1 to designate the value of the J-integral at crack initiation; the J_{IN} notation will be used in this report in keeping with Reference 1.

$$T_{app} = \frac{dJ}{da} \frac{E}{\sigma_f^2}$$

where:

T_{app} = applied tearing modulus
E = modulus of elasticity
σ_f = [$]^{a,c,e}$ (flow stress)
a = crack length

[

$]^{a,c,e}$

In summary, the local crack stability will be established by the two-step criteria:

$$J < J_{IN}$$

$$T_{app} < T_{mat}, \text{ if } J \geq J_{IN}$$

4.3 Material Properties

The piping system materials in the primary loops of the Callaway and Wolf Creek units are cast stainless steel (SA-351-CF8A) and associated welds. The tensile and flow properties at the critical location, the hot leg, and the reactor vessel outlet nozzle junction are given in Table 3-1.

The fracture properties of CF8A cast stainless steel have been determined through fracture tests carried out at 600°F and reported in Reference 12. This reference shows that J_{IN} for the base metal ranges from [$]^{a,c,e}$ for the multiple tests carried out.

Cast stainless steels are subject to thermal aging during service. This thermal aging causes an elevation in the yield strength of the material and a degradation of the fracture toughness, the degree of degradation being proportional to the level of ferrite in the material. To determine the effects of thermal aging on piping integrity, a detailed study was carried out in Reference 13. In that report, fracture toughness results were presented for a material representative of [

]^{a,c,e} Toughness results were provided for the material in the fully aged condition and these properties are also presented in Figure 4-2 of this report. The J_{IN} value for this material at operating temperature was approximately [ ]^{a,c,e} and the maximum value of J, designated J_{max}, obtained in the tests was in excess of [

]^{a,c,e} The tests for this material were conducted on small specimens and therefore rather short crack extensions resulted. (Maximum extension was 4.3 mm.) Therefore it is expected that higher J values would be sustained for larger specimens. The effect of the aging process on loop piping integrity for Callaway and Wolf Creek is addressed in Table 4-1, where the plant specific material chemistry for the loop materials is considered. This table shows that the degree of thermal aging expected by end-of-life is less than that produced in [ ].^{a,c,e} Therefore the J_{IN} values for the Callaway and Wolf Creek units at end-of-life would be expected to be considerably higher than those reported for [ ]^{a,c,e} in Figure 4-2 (also see Reference 14). In addition, the tearing modulus for the Callaway and Wolf Creek material would be greater than [

]^{a,c,e}

Available data on stainless steel welds indicate that the J_{IN} values for the worst case welds are on the same order as the aged material, but the slope of the J-R curve is steeper, and higher J-values have been obtained from fracture tests (in excess of 3000 in-lbs/in^2). The applied value of the J-integral for a flaw in the weld regions will be lower than that in the base metal because the yield stress for the weld materials is much higher at the operating temperature. Therefore, weld regions are less limiting than the cast material.

4.4 Results of Crack Stability Evaluation

Figure 4-3 shows a plot of the [ ]^{a,c,e} as a function of through-wall circumferential flaw length in the [ ]^{a,c,e} of the main coolant piping. This [ ]^{a,c,e} was calculated for Callaway and Wolf Creek using data for a pressurized pipe at 2235 psig with an axial force of [ ]^{a,c,e} operating at 619°F with [ ]^{a,c,e} properties.

The maximum applied bending moment of [ ]^{a,c,e} can be plotted on this figure, and used to determine a critical flaw length, which is shown to be [ ].^{a,c,e} This is considerably larger than the [ ]^{a,c,e} reference flaw used in Reference 1.

J-integral calculations were performed in Reference 1. Based on the calculations it was shown that a [ ]^{a,c,e}

The pipe under present investigation is 2.37 inches thick with a 29.2 inch inside diameter. These dimensions are within [ ]^{a,c,e} The axial load used in the present case is [ ]^{a,c,e} than that used in Reference 1. However, the [ ]^{a,c,e} percent of the moment load used in Reference 1. The maximum outer fiber stress for Callaway and Wolf Creek is only [ ]^{a,c,e} of that of Reference 1. [ ]^{a,c,e}

On this basis it is judged that the conclusions of Reference 1 are applicable to the Callaway and Wolf Creek primary loops. Specifically, it can be concluded that a postulated [ ]^{a,c,e} through-wall flaw in the Callaway and Wolf Creek loop piping will remain stable from both a local and global stability standpoint.

Actually, for the Callaway and Wolf Creek loads, the applied J was calculated by the [ ]^{a,c,e} method for the [ ]^{a,c,e} inch flaw. A yield strength of [ ]^{a,c,e} was used. For the maximum load, the applied J was [ ]^{a,c,e} which is less than J_{max} for the [ ]^{a,c,e}.

Based on available [ ]^{a,c,e} for a [ ]^{a,c,e} through-wall flaw, the applied J was estimated using a yield strength of [ ]^{a,c,e} ksi. The purpose of the evaluation was to investigate the crack stability for a postulated flaw smaller in size than the [ ]^{a,c,e} reference flaw since, as seen later, leak rates are very large for such a long flaw. For the Callaway and Wolf Creek maximum moment of [ ]^{a,c,e} the maximum applied J was estimated to be [ ]^{a,c,e} for the [ ]^{a,c,e} flaw.

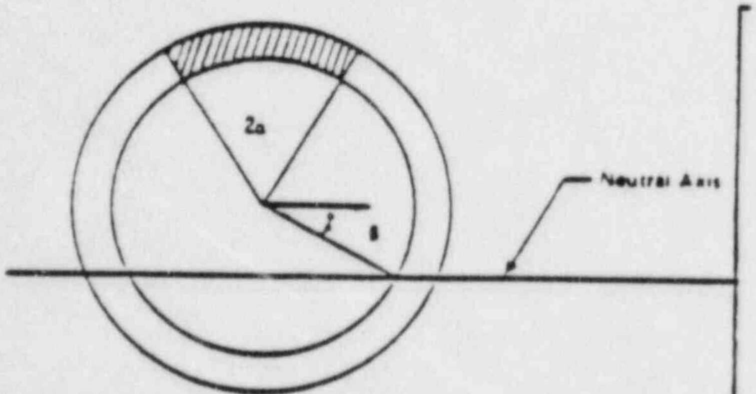
The applied tearing modulus, T_{app}, was calculated for both flaws using the methodology of Reference 1 and was less than [ ]^{a,c,e} which is significantly less than T_{mat} for even the worst case [ ]^{a,c,e} material of Reference 13.

FIGURE 4-1

CHEMICAL AND PHYSICAL PROPERTIES OF CALLAWAY AND WOLF CREEK PRIMARY LOOP MATERIAL

MATERIAL: SA-351-CF8A

a,c,e



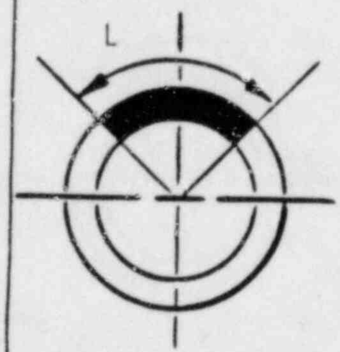
a,c,e

FIGURE 4-1 [ ]^{a,c,e} Stress Distribution

[ ]^{a,c,e}

FIGURE 4-2 J-Δa Curves at Differen emperatures, Aged Material [ ]^{a,c,e}
(7500 Hours at 400°C)

a,c,e



FLAW GEOMETRY

OD = 33.94 in.
t = 2.37 in.
p = 2235 psig
Temp = 619°F

FIGURE 4-3 CRITICAL FLAW SIZE PREDICTION

5.0 LEAK RATE PREDICTIONS

Leak rate estimates were performed by applying the normal operating bending moment of [ ]^{a,c,e} in addition to the normal operating axial force of [ ]^{a,c,e} to the hot leg pipe containing a postulated [ ]^{a,c,e} through-wall flaw. The crack opening area was estimated using the method described in Reference 15. The leak rate was calculated using the two-phase flow formulation described in Reference 1. The computed leak rate was [ ]^{a,c,e} In order to determine the sensitivity of leak rate to flaw size, a through-wall flaw [ ]^{a,c,e} in length was postulated. The calculated leak rate was [ ]^{a,c,e}

The Callaway and Wolf Creek plants have RCS pressure boundary leak detection systems which are consistent with the guidelines of Regulatory Guide 1.45 of detecting leakage of 1 gpm in one hour. Thus, for the [ ]^{a,c,e} inch flaw, a factor in excess of 93 exists between the calculated leak rate and the criteria of Regulatory Guide 1.45. Relative to the [ ]^{a,c,e} inch flaw, a factor of over 40 exists.

6.0 FATIGUE CRACK GROWTH ANALYSIS

To determine the sensitivity of the primary coolant system to the presence of small cracks, a fatigue crack growth analysis was carried out for the [$]^{a,c,e}$ region of a typical system [$]^{a,c,e}$ This region was selected because crack growth calculated here will be typical of that in the entire primary loop. Crack growths calculated at other locations are expected to be within 10%.

A [$]^{a,c,e}$ of a plant typical in geometry and operational characteristics to any Westinghouse PWR System. [

$]^{a,c,e}$ All normal, upset, and test conditions were considered and circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in three different locations, as shown in Figure 6-1. Specifically, these were:

Cross Section A: [ ]^{a,c,e}
Cross Section B:
Cross Section C:

Fatigue crack growth rate laws were used [

$]^{a,c,e}$ The law for stainless steel was derived from Reference 16, with a very conservative correction for the R ratio, which is the ratio of minimum to maximum stress during a transient. For stainless steel, the fatigue crack growth formula is:

$$\frac{da}{dn} = (5.4 \times 10^{-12})\, K_{eff}^{\;4.48} \qquad \text{inches/cycle}$$

where $K_{eff} = K_{max}\,(1-R)^{0.5}$

$R = K_{min}/K_{max}$

[

]a,c,e

[ ] a,c,e

where: [ ] a,c,e

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 6-1. The results show that the crack growth is very small, regardless [ ]a,c,e

TABLE 6-1

FATIGUE CRACK GROWTH AT [ ]a,c,e (40 years)

INITIAL FLAW (IN)	FINAL FLAW (IN)		
	[ ]a,c,e	[ ]a,c,e	[ ]a,c,e
0.292	0.31097	0.30107	0.30698
0.300	0.31949	0.30953	0.31626
0.375	0.39940	0.38948	0.40763
0.425	0.45271	0.4435	0.47421

[ ]a,c,e

FIGURE 6-1 Typical Cross-Section of [ ]a,c,e

CRACK GROWTH RATE, da/dN (MICRO-INCHES/CYCLE)

a,c,e

FIGURE 6-2 Reference Fatigue Crack Growth Curves For [ ]a,c,e

[ ]a.c.e

FIGURE 6-3 Reference Fatigue Crack Growth Law for [ ]a,c,e in a Water Environment at 600°F

7.0 ASSESSMENT OF MARGINS

In Reference 1, the maximum design moment was [ ]^{a,c,e} whereas, the maximum moment as noted in Section 3.0 of this report is [ ]^{a,c,e}. The calculated applied J [ ]^{a,c,e} as compared with the value of [ ]^{a,c,e} in Reference 1. Furthermore, Section 4.3 shows the testing of fully aged material with chemistry more limiting than that existing in Callaway and Wolf Creek cast piping extended to J values of [ ]^{a,c,e}.

As shown in Section 3.0, margins of factors of greater than 3 exist between calculated and ASME Code allowable faulted condition and thermal stresses.

Referring to Section 4.3, the estimated tearing modulus for Callaway and Wolf Creek cast stainless steel piping in the fully aged condition is at least [ ].^{a,c,e} T_{app} for the reference flaw as taken from Reference 13 is [ ]^{a,c,e} and is certainly less than the [ ]^{a,c,e} calculated for the [ ]^{a,c,e} flaw in Section 4.4. Consequently, a margin on local stability of at least 3 exists relative to tearing.

In Section 4.4, it is seen that a [ ]^{a,c,e} flaw has a J value at maximum load of [ ]^{a,c,e} which is, of course enveloped by the J_{max} of Reference 1 and the value for aged material. For the [ ]^{a,c,e} flaw the J at maximum load is [ ]^{a,c,e} which is also enveloped by the J_{max} of Reference 1 and the value for aged material. In Section 4.4, the critical flaw size using [ ]^{a,c,e} methods is calculated to be [ ]^{a,c,e} Based on the above, the critical flaw size will, of course, exceed [ ]^{a,c,e}

In Section 5.0, it is shown that a flaw of [ ]^{a,c,e} would yield a leak rate in excess of [ ]^{a,c,e} while for a [ ]^{a,c,e} flaw, the leak rate is [ ].^{a,c,e} Thus, there is a margin of at least 3 between the flaw size that gives a leak rate well exceeding the criterion of Regulatory Guide 1.45 and the "critical" flaw size of [ ]^{a,c,e}

In summary, relative to

1. Loads

   a. Callaway and Wolf Creek are enveloped both by the maximum loads and J values in Reference 1 and the J values employed in testing of fully aged material.

   b. At the critical location, a margin of 3 on both faulted conditions and thermal stresses exist relative to ASME Code allowable values.

2. Flaw Size

   a. A margin of at least 3 exists between the critical flaw and the flaw yielding a leak rate of [ ]^{a,c,e}

   b. A margin exists of at least 3 relative to tearing.

   c. If [ ]^{a,c,e} is used as the basis for critical flaw size, the margin for global stability would exceed 3.7 when compared to the reference flaw.

3. Leak Rate

   A margin in excess of 90 exists for the reference flaw ([ ]^{a,c,e}) between calculated leak rates and the criteria of Regulatory Guide 1.45.

8.0 CONCLUSIONS

This report has established the applicability of the generic Westinghouse evaluations which justify the elimination of RCS primary loop pipe breaks for the Callaway and Wolf Creek plants as follows:

a. The loads, material properties, transients, and geometry relative to the Callaway and Wolf Creek RCS primary loops are enveloped by the parameters of WCAP-9558, Revision 2 (Reference 1) and WCAP-10456 (Reference 13).

b. Stress corrosion cracking is precluded by the use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

c. Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations.

d. The effects of low and high cycle fatigue on the integrity of the primary piping are negligible.

e. A large margin exists between the leak rate of the reference flaw and the criteria of Reg. Guide 1.45.

f. Ample margin exists between the flaw chosen for leak detectability and the critical flaw.

g. Ample margin exists in the material properties used to demonstrate end-of-life (relative to aging) stability of the reference flaw.

The reference flaw will be stable throughout reactor life because of the ample margins in e, f, and g above and will leak at a detectable rate which will assure a safe plant shutdown.

Based on the above, it is concluded that RCS primary loop pipe breaks should not be considered in the structural design basis for Callaway and Wolf Creek.

9.0 REFERENCES

1. WCAP-9558, Rev. 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," Westinghouse Proprietary Class 2, June 1981.

2. USNRC Generic letter 84-04, Subject: "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops", February 1, 1984.

3. WCAP-8082 P-A, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," Class 2, January 1975.

4. Letter from Westinghouse (E. P. Rahe) to NRC (R. H. Vollmer), NS-EPR-2768, dated May 11. 1983.

5. WCAP-9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March, 1978.

6. WCAP-9787, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation", Westinghouse Proprietary Class 2, May 1981.

7. Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, November 10, 1981.

8. Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated April 25, 1983.

9. Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

10. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors", USNRC, September 1980.

11. Kanninen, M. F., et. al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks", EPRI NP-192, September 1976.

12. Bush, A. J., Stouffer, R. B., "Fracture Toughness of Cast 316 SS Piping Material Heat No. 156576, at 600°F", Westinghouse R D Memo No. 83-5P6EVMTL-M1, Westinghouse Proprietary Class 2, March 7, 1983.

13. WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping For Westinghouse NSSS," Westinghouse Proprietary Class 2, November 1983.

14. Slama, G., Petrequin, P., Masson, S. H., and Mager, T. R., "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Casting and Welds", presented at SMiRT 7 Post Conference Seminar 6 - Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, August 29/30, 1983, Monterey, CA.

15. NUREG/CR-3464. "The Application of Fracture Proof Design Methods using Tearing Instability Theory to Nuclear Piping Postulating Circumferential Through Wall Cracks" 1983.

16. Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment", Trans. ASME Journal of Pressure Vessel Technology, Vol. 101, Feb. 1979.

17. [

]^{a,c,e}

18. [

]^{a,c,e}

APPENDIX A

[ ]a,c,e

+a,c,e

FIGURE A-1 Pipe with a through-wall crack in bending