WCAP-14560
Revision 1

TECHNICAL JUSTIFICATION FOR ELIMINATING LARGE PRIMARY LOOP PIPE RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR THE BYRON AND BRAIDWOOD UNITS 1 AND 2 NUCLEAR POWER PLANTS

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EXECUTIVE SUMMARY

The original structural design basis of the reactor coolant system for the Commonwealth Edison Company Byron and Braidwood Units 1 and 2 Nuclear Power Plant required consideration of dynamic effects resulting from pipe break and that protective measures for such breaks be incorporated into the design. Subsequent to the original Byron and Braidwood design, additional concern of asymmetric blowdown loads was raised as described in Unresolved Safety Issue A-2 (Asymmetric Blowdown Loads on the Reactor Coolant System) and Generic Letter 84-04 (Reference 1-1). However, research by the NRC and industry coupled with operating experience determined that safety could be negatively impacted by placement of pipe whip restraints on certain systems. As a result, NRC and industry initiatives resulted in demonstrating that Leak-Before-Break (LBB) criteria can be applied to reactor coolant system piping based on fracture mechanics technology and material toughness. The Byron and Braidwood Units 1 and 2 primary loop piping analyses by Westinghouse for the application of LBB was documented in WCAP-10553 (Reference 1-2) and approved by the NRC letter dated October 28, 1985 (Reference 1-3). By letter dated October 28, 1985, the NRC stated that:

> "In a letter dated September 28, 1984, Commonwealth Edition Company (CECO) requested an exemption from a portion of the requirements of General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50. You provided the Westinghouse report "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Byron Units 1 and 2 and Braidwood Units 1 and 2, "WCAP-10554 (Westinghouse Non-Proprietary) and WCAP-10553 (Westinghouse Proprietary) as an enclosure to this letter which serves as the technical basis in support of the request. The Westinghouse report addresses the "leak-before-break" concept as an alternative to providing protective devices against the dynamic effect of postulated rupture in the primary coolant loops. Your submittal dated June 28, 1985 provided a value-impact analysis associated with the exemption request and requested that a partial exemption to GDC-4 be granted for the first two cycles of operation for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (the facilities). Your letter dated August 14, 1985 withdrew, without prejudice, the request for the exemption for Byron Station, Unit 1.

> On the basis of the Staff's evaluation of these submittals the Commission has granted your exemption request for periods ending at the completion of the second refueling outage of each of the facilities, pending the outcome of the Commission's ongoing rulemaking on this subject. The staff has also received your September 25, 1985 letter requesting amendments to the construction permits for Byron Station, Unit 2, and Braidwood Station, Units 1 and 2. The exemption granted will become effective upon the date of issuance. The enclosed exemption is being forwarded to the office of the Federal Register for publication, accordingly."

This report demonstrates compliance with LBB technology for the Byron and Braidwood reactor coolant system piping based on the latest criteria. The report documents the plant specific geometry, loading, and material properties used in the fracture mechanics evaluation. Mechanical properties were determined at operating temperatures. Since the piping systems include cast stainless steel fittings, fracture toughnesses considering thermal aging were determined for each heat of material.

Based on loading, pipe geometry and fracture toughness considerations, enveloping critical locations were determined at which leak-before-break crack stability evaluations were made. Through-wall flaw sizes were found which would cause leak at a rate of ten times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the primary loops.

It is concluded that dynamic effects of reactor coolant system primary loop pipe breaks need not be considered in the structural design basis of the Byron and Braidwood Units 1 and 2 Nuclear Power Plants.

Revision 1 is to modify the second paragraph of section 2.3 and to add reference 2-3.

The revisions are identified by vertical lines in the column.

SECTION 1.0
INTRODUCTION

1.1 Purpose

This report applies to the Byron and Braidwood Units 1 and 2 Reactor Coolant System (RCS) primary loop piping. It is intended to demonstrate that for the specific parameters of the Byron and Braidwood Units 1 and 2 Nuclear Power Plants, RCS primary loop pipe breaks need not be considered in the structural design basis. The approach taken has been accepted by the Nuclear Regulatory Commission (NRC) (Reference 1-1).

1.2 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 1-4). That topical report employed a deterministic fracture mechanics evaluation and a probabilistic analysis to support the elimination of RCS primary loop pipe breaks. That approach was then used as a means of addressing Generic Issue A-2 and Asymmetric LOCA Loads.

Westinghouse performed additional testing and analysis to justify the elimination of RCS primary loop pipe breaks. This material was provided to the NRC along with Letter Report NS-EPR-2519 (Reference 1-5).

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 (References 1-6 and 1-7). 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 4.4×10^{-12} 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 1-4) 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 1-1) 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. In a more formal recognition of Leak-Before-Break (LBB) methodology applicability for PWRs, the NRC appropriately modified 10 CFR 50, General Design Criterion 4, "Requirements for Protection Against Dynamic Effects for Postulated Pipe Rupture" (Reference 1-8).

1.3 Scope and Objective

The general purpose of this investigation is to demonstrate leak-before-break for the primary loops in Byron and Braidwood Units 1 and 2. The recommendations and criteria proposed in Reference 1-9 are used in this evaluation. These criteria and resulting steps of the evaluation procedure can be briefly summarized as follows:

1) Calculate the applied loads. Identify the location at which the highest stress occurs.

2) Identify the materials and the associated material properties.

3) Postulate a surface flaw at the governing location. Determine fatigue crack growth. Show that a through-wall crack will not result.

4) Postulate a through-wall flaw at the governing location. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the pipe is subjected to normal operating loads. A margin of 10 is demonstrated between the calculated leak rate and the leak detection capability.

5) Using faulted loads, demonstrate that there is a margin of 2 between the leakage size flaw and the critical size flaw.

6) Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer or low and high cycle fatigue.

7) For the materials actually used in the plant provide the properties including toughness and tensile test data. Evaluate long term effects such as thermal aging where applicable.

8) Demonstrate margin on applied load.

This report provides a fracture mechanics demonstration of primary loop integrity for the Byron and Braidwood Units 1 and 2 Plants consistent with the NRC position for exemption from consideration of dynamic effects.

Several computer codes are used in the evaluations. The main-frame computer programs are under Configuration Control which has requirements conforming to NRC's Standard Review Plan 3.9.1 (Reference 1-10). The fracture mechanics calculations are independently verified (benchmarked).

1.4 References

1-1 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.

1-2 WCAP-10553, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design basis for the Byron Units 1 and 2 and Braidwood Units 1 and 2," May 1984.

1-3 Nuclear Regulatory Commission Docket Nos. STN 50-455, STN 50-456, and STN 50-457, dated October 28, 1985, Letter from B. J. Youngblood, Chief Licensing Branch No. 1 Division of Licensing, NRC, to Dennis L. Farrar, Director of Nuclear Licensing, Commonwealth Edison Company.

1-4 WCAP-9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March, 1978.

1-5 Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, November 10, 1981.

1-6 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated April 25, 1983.

1.4 References(cont)

1-7 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

1-8 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288-41295.

1-9 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday August 28, 1987/Notices, pp. 32626-32633.

1-10 Nuclear Regulatory Commission, Standard Review Plan Section 3.9.1, "Special Topics for Mechanical Component," NUREG-0800, Revision 2, July 1981.

1-11 WCAP-7211, Revision 3, "Energy Systems Business Unit Policy and Procedures for Management, Classification, and Release of Information," March, 1994.

SECTION 2.0
OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 Stress Corrosion Cracking

The Westinghouse reactor coolant system primary loops have an operating history that demonstrates the inherent operating stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking (IGSCC)). This operating history totals over 800 reactor-years, including five plants each having over 18 years of operation and 15 other plants each with over 13 years of operation.

In 1978, the United States Nuclear Regulatory Commission (USNRC) formed the second Pipe Crack Study Group. (The first Pipe Crack Study Group (PCSG) established in 1975 addressed cracking in boiling water reactors only.) One of the objectives of the second PCSG was to include a review of the potential for stress corrosion cracking in Pressurized Water Reactors (PWR's). The results of the study performed by the PCSG were presented in NUREG-0531 (Reference 2-1) entitled "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants." In that report the PCSG stated:

> "The PCSG has determined that the potential for stress-corrosion cracking in PWR primary system piping is extremely low because the ingredients that produce IGSCC are not all present. The use of hydrazine additives and a hydrogen overpressure limit the oxygen in the coolant to very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustic, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the primary systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress corrosion cracking has been reported in the primary piping or safe ends of any PWR."

During 1979, several instances of cracking in PWR feedwater piping led to the establishment of the third PCSG. The investigations of the PCSG reported in NUREG-0691 (Reference 2-2) further confirmed that no occurrences of IGSCC have been reported for PWR primary coolant systems.

As stated above, for the Westinghouse plants there is no history of cracking failure in the reactor coolant system loop. The discussion below further qualifies the PCSG's findings.

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. 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 material in the system, applicable ASME Code rules, fracture toughness, welding, fabrication, and processing.

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 (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. 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 in the ppb range 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. Thus during plant operation, the likelihood of stress corrosion cracking is minimized.

2.2 Water Hammer

Overall, there is a low potential for water hammer in the RCS since it is designed and operated to 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 are relatively slow transients with no significant effect on the system 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 accounted for 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 8.0.

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 through periodic monitoring by plant personnel through instrumentation located in the Byron and Braidwood Auxiliary Electrical Equipment Room. Field measurements have been made on a number of plants during hot functional testing, including plants similar to Byron and Braidwood Units 1 and 2. Stresses in the elbow below the reactor coolant pump resulting from system vibration 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. During preoperational testing at Byron and Braidwood, vibration levels of the Reactor Coolant System were measured and determined by Commonwealth Edison to be significantly below the Westinghouse acceptance criteria (Reference 2-3).

2.4 References

2-1 Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531, U.S. Nuclear Regulatory Commission, February 1979.

2-2 Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, U.S. Nuclear Regulatory Commission, September 1980.

2.4 References(cont)

2-3 Letter # CAW-5738; " Pre-Operational loop Testing," from Westinghouse (W. E. Kortier) to J. D. Deress of Commonwealth Edison Company dated May 19, 1983.

SECTION 3.0
PIPE GEOMETRY, LOADS AND STRESSES

3.1 Introduction

The general approach is discussed first. As an example a segment of the primary coolant cold leg pipe is shown in Figure 3-1. The as-built outside diameter and minimum wall thickness of the pipe are 32.14 in. and 2.215 in., respectively, as shown in the figure. The normal stresses at the weld locations are from the load combination procedure discussed in Section 3.3 whereas the faulted loads are as described in Section 3.4. The components for normal loads are pressure, dead weight and thermal expansion. An additional component, Safe Shutdown Earthquake (SSE), is considered for faulted loads. As seen from Table 3-2, the highest stressed location in the entire loop is at Location 11 at the reactor coolant pump outlet nozzle to pipe weld. This highest stressed location is a load critical location and is one of the locations at which, as an enveloping location, leak-before-break is to be established. Essentially a circumferential flaw is postulated to exist at this location which is subjected to both the normal loads and faulted loads to assess leakage and stability, respectively. The loads (developed below) at this location are also given in Figure 3-1.

Since the elbows are made of cast stainless steel, thermal aging must be considered (Section 4.0). Thermal aging causes in lower fracture toughness of the cast materials; thus, location other than the load critical location must be examined taking into consideration both fracture toughness and stress. The enveloping location so determined is called the toughness critical location. One most critical location is identified after the full analysis is completed. Once loads (this section) and fracture toughnesses (Section 4.0) are obtained, the load critical and toughness critical locations are determined (Section 5.0). At these locations, leak rate evaluations (Section 6.0) and fracture mechanics evaluations (Section 7.0) are performed per the guidance of Reference 3-1. Fatigue crack growth (Section 8.0) and stability margins are also evaluated (Section 9.0).

All the weld locations including load critical location and toughness critical location for evaluation are shown in Figure 3-2.

3.2 Calculation of Loads and Stresses

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

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad (3\text{-}1)$$

where,

σ	=	stress
F	=	axial load
M	=	bending moment
A	=	pipe cross-sectional area
Z	=	section modulus

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

$$M = \sqrt{M_Y^2 + M_Z^2} \qquad (3\text{-}2)$$

where,

M	=	bending moment for required loading
M_Y	=	Y component of bending moment
M_Z	=	Z component of bending moment

The axial load and bending moments for leak rate predictions and crack stability analyses are computed by the methods to be explained in Sections 3.3 and 3.4.

3.3 Loads for Leak Rate Evaluation

The normal operating loads for leak rate predictions are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \tag{3-3}$$
$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} + (M_Y)_P \tag{3-4}$$
$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} + (M_Z)_P \tag{3-5}$$

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

DW = deadweight
TH = normal thermal expansion
P = load due to internal pressure

This method of combining loads is often referred as the algebraic sum method (Reference 3-1).

The loads based on this method of combination are provided in Table 3-1 at all the locations identified in Figure 3-2.

3.4 Load Combination for Crack Stability Analyses

In accordance with Standard Review Plan 3.6.3 (Reference 3-1) the absolute sum of loading components can be applied which results in higher magnitude of combined loads. If crack stability is demonstrated using these loads, the LBB margin on loads can be reduced from √2 to 1.0. The absolute summation of loads are shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \tag{3-6}$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_P| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \tag{3-7}$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_P| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \tag{3-8}$$

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

The loads so determined are used in the fracture mechanics evaluations (Section 7.0) to demonstrate the LBB margins at the locations established to be the governing locations. These loads at all the locations of interest (see Figure 3-2) are given in Table 3-2.

3.5 References

3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

Table 3-1

Normal Loads and Normal Stresses for Byron and Braidwood Units 1 and 2

Location[a]	Outside Diameter (in)	Minimum Thickness (in)	Axial Load[b] (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	33.90	2.345	1470	17909	16.76
2	33.90	2.345	1472	917	6.87
3	34.21	2.500	1472	8635	10.60
4	37.56	3.175	1616	14023	9.86
5	37.57	3.180	1638	5161	6.66
6	36.20	2.495	1737	4014	8.50
7	36.20	2.495	1624	3846	7.99
8	36.20	2.495	1673	663	6.64
9	36.20	2.495	1673	2898	7.72
10	37.57	3.180	1770	7052	7.74
11	32.14	2.215	1404	5481	10.50
12	32.14	2.215	1404	5191	10.30
13	32.14	2.215	1404	3444	9.10
14	32.14	2.215	1389	4770	9.94
15	32.43	2.360	1387	4937	9.38

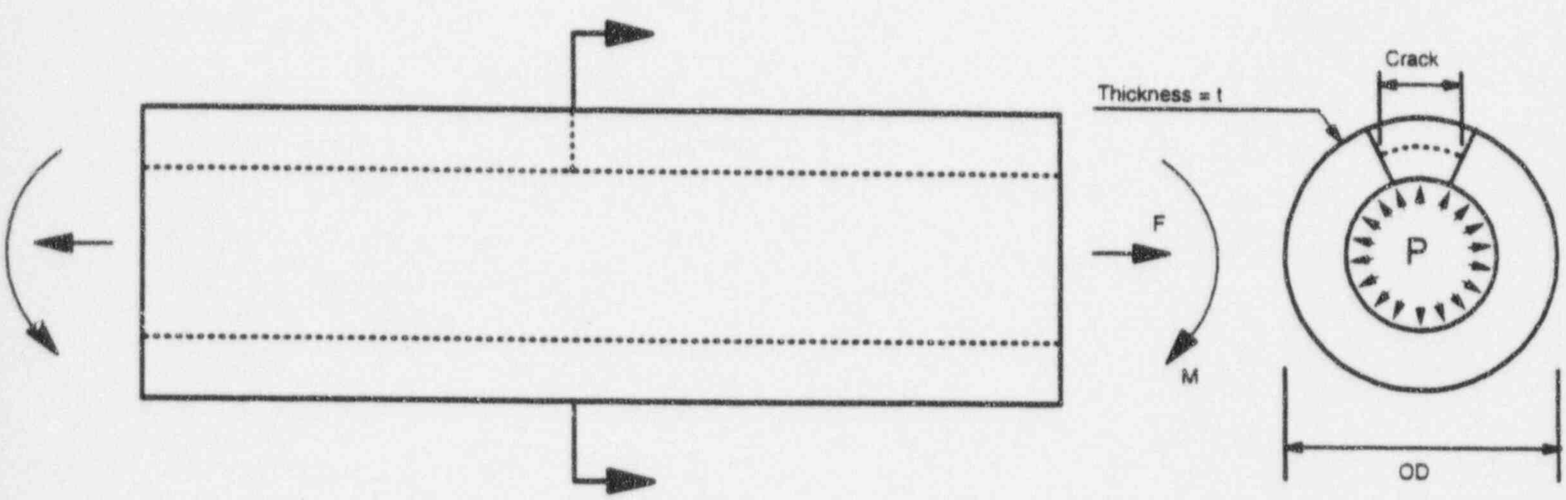
a See Figure 3-2
b Includes pressure

Table 3-2

Faulted Loads and Stresses for Byron and Braidwood Units 1 and 2

Location[a]	Axial Load[b] (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	2917	30318	30.21
2	2892	9965	18.24
3	2573	22713	22.67
4	2686	31649	19.46
5	2095	30190	17.17
6	1977	22177	18.12
7	1984	12980	13.74
8	1988	10434	12.53
9	1953	13057	13.66
10	2017	39666	20.42
11	2275	31856	32.77
12	2275	28561	30.51
13	2306	19245	24.27
14	2293	13283	20.12
15	2226	15558	19.93

a See Figure 3-2
b Includes pressure



$OD^a = 32.14$ in
$t^a = 2.215$ in

Normal Loads[a]		Faulted Loads[b]	
force[c]:	1404 kips	force[c]:	2275 kips
bending moment:	5481 in-kips	bending moment:	31856 in-kips

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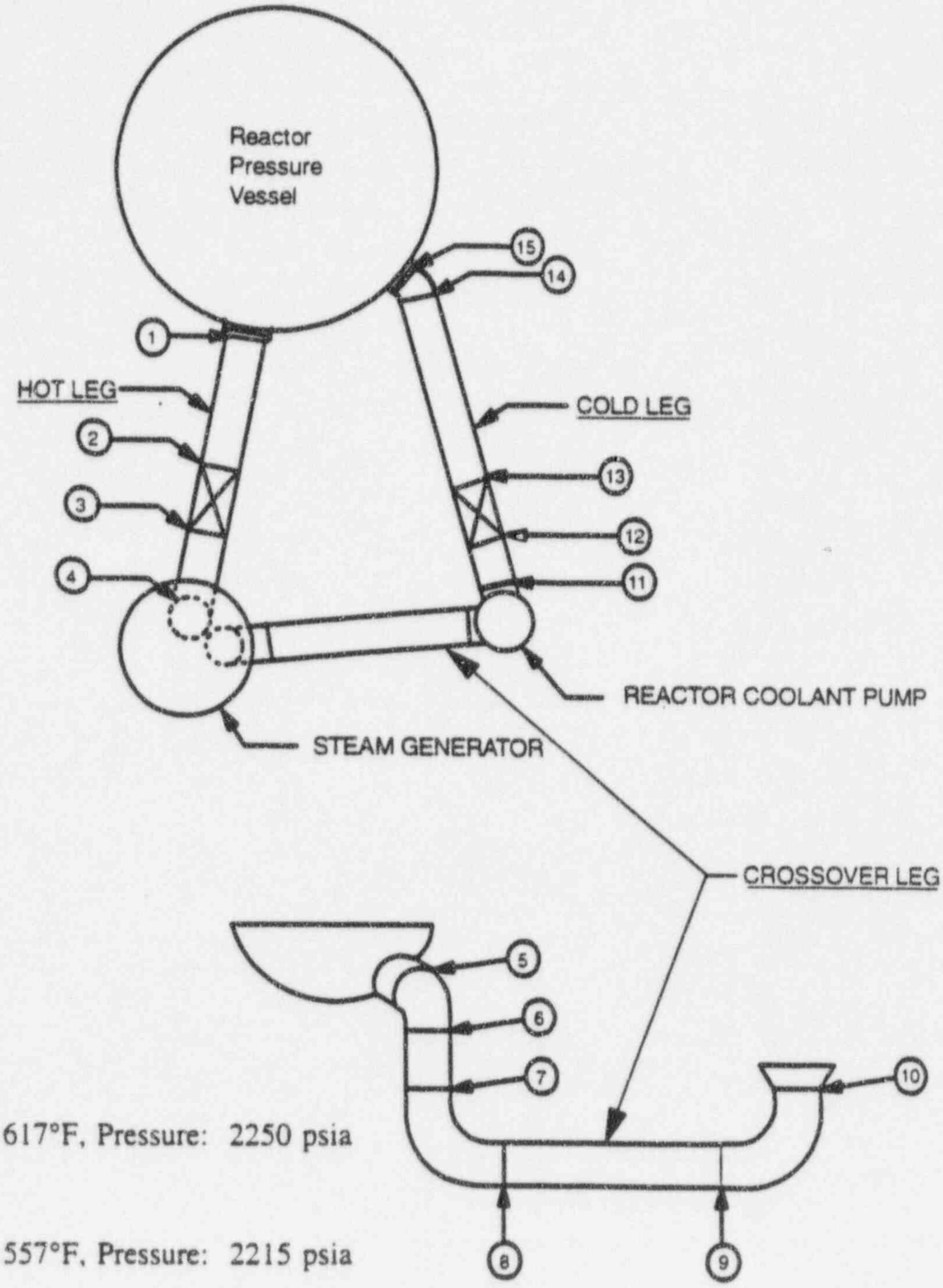
[a] See Table 3-1
[b] See Table 3-2
[c] Includes the force due to a pressure of 2305 psia

Figure 3-1 Cold Leg Coolant Pipe



HOT LEG

Temperature 617°F, Pressure: 2250 psia

CROSSOVER LEG

Temperature 557°F, Pressure: 2215 psia

COLD LEG

Temperature 557°F, Pressure: 2305 psia

Figure 3-2 Schematic Diagram of Byron and Braidwood Units 1 and 2 Primary Loop Showing Weld Locations

SECTION 4.0
MATERIAL CHARACTERIZATION

4.1 Primary Loop Pipe, Fittings, and Weld Materials

The primary loop piping material for Byron and Braidwood Units 1 and 2 is SA376 Gr. 304N. The elbows are made of SA351 CF8A material. The weld processes used were TIG and SMAW combination.

4.2 Tensile Properties

The Pipe Certified Materials Test Reports (CMTRs) for Byron and Braidwood Units 1 and 2 were used to establish the tensile properties for the leak-before-break analyses. The CMTRs include tensile properties at room temperature for each of the heats of material. These properties are given in Table 4-1 through Table 4-4 for piping and in Table 4-5 through Table 4-8 for elbow fittings.

For the SA376 Gr. 304N material, the representative properties at 557°F were established from the tensile properties at room temperature given in Table 4-1 through 4-4 by utilizing Section III of the 1989 ASME Boiler and Pressure Vessel Code (Reference 4-6). Code tensile properties at 557°F were obtained by interpolating between the 500°F and 600°F tensile properties. Ratios of the code tensile properties at 557°F to the corresponding tensile properties at room temperature were then applied to the room temperature tensile properties given in Table 4-1 through 4-4 to obtain the plant specific properties for SA376 Gr. 304N at 557°F.

The Elbow Fittings Certified Materials Test Reports (CMTRs) for Byron and Braidwood Units 1 and 2 were used to establish the tensile properties for the leak-before-break analyses. The CMTRs for elbow fittings include tensile properties at room temperature for each of the heats of material. These properties are given for Byron and Braidwood Units 1 and 2 in Table 4-5 through Table 4-8.

For the SA351 CF8A material, the representative properties at 617°F were established from the tensile properties at room temperature given in Table 4-5 through 4-8 by utilizing Section III of the 1989 ASME Boiler and Pressure Vessel Code. Code tensile properties at 617°F were established by interpolating between the 600°F and the 650°F tensile properties. Ratios of the code tensile properties at 617°F to the corresponding properties at room temperature were then applied to the room temperature properties given in Table 4-5 through 4-8 to obtain the plant specific representative properties for SA351 CF8A at 617°F.

The average and lower bound yield strengths and ultimate strengths are given in Table 4-9. The ASME Code modulus of elasticity are also given, and Poisson's ratio was taken as 0.3.

For leak-before-break fracture evaluations of the toughness critical location the true stress-true strain curve for SA351 CF8A at 617°F must be available. This curve was obtained using the Nuclear Systems Materials Handbook (Reference 4-1). The lower bound true stress-true strain curve is given in Figure 4-1.

4.3 Fracture Toughness Properties

The pre-service fracture toughnesses of both forged and cast stainless steels of interest here have in terms of J_{Ic} been found to be very high at 600°F. Typical results for a cast material are given in Figure 4-2. J_{Ic} is observed to be over 2500 in-lbs/in^2. Forged materials are even higher. However, 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 somewhat 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 4-2. In that report, fracture toughness results were presented for a material [

]^{a,c,e} The effects of the aging process on the end-of-service life fracture toughness are further discussed in Appendix B.

End-of-service life toughnesses for the heats are established using the alternate toughness criteria methodology of Reference 4-5 (Appendix B). By that methodology a heat of material is said to be as good as [ ]^{a,c,e} if it can be demonstrated that its end-of-service fracture toughnesses equal or exceed those of [

]^{a,c,e}. The fracture toughness value for Byron and Braidwood Units 1 and 2 loops at the toughness critical location, as established in Appendix B, is given in Table 4-10.

Available data on aged stainless steel welds (References 4-2 and 4-3) indicate that J_{Ic} values for the worst case welds are of the same order as the aged material. However, the slope of the J-R curve is steeper, and higher J-values have been obtained from fracture tests (in excess of 3000 in-lb/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 temperature[a]. Therefore, weld regions are less limiting than the cast material.

It is thus conservative to choose the end-of-service life toughness properties of [ ]^{a,c,e} as representative of those of the welds. Also, such pipes and fittings have an end-of-service life calculated room temperature Charpy U-notch energy, (KCU), greater than that of [ ]^{a,c,e} are also conservatively assumed to have the properties of [ ]^{a,c,e}.

In the fracture mechanics analyses that follow, the fracture toughness properties given in Table 4-10 will be used as the criteria against which the applied fracture toughness values will be compared.

Forged stainless steel piping such as SA376 Gr. 304N does not degrade due to thermal aging. Thus fracture toughness values well in excess of that established for the cast material and welds exist for this material throughout service life and are not limiting.

4.4 References

4-1 Nuclear Systems Materials Handbook, Part I - Structural Materials, Group 1 - High Alloy Steels, Section 2, ERDA Report TID 26666, November, 1975.

4-2 WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for W NSSS," W Proprietary Class 2, November 1983.

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[a] In the report all the applied J values were conservatively determined by using base metal strength properties.

4.4 References(cont)

4-3 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.

4-4 Appendix II of Letter from Dominic C. DiIanni, NRC to D. M. Musolf, Northern States Power Company, Docket Nos. 50-282 and 50-306, December 22, 1986.

4-5 Witt, F.J., Kim, C.C., "Toughness Criteria for Thermally Aged Cast Stainless Steel," WCAP-10931, Revision 1, Westinghouse Electric Corporation, July 1986, (Westinghouse Proprietary Class 2).

4-6 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 4-1

Measured Tensile Properties for Byron Unit 1 Primary Loop Piping Material SA376 Gr. 304N

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)
Hot Leg	L1360 Ser. 14480	45900	89900
Hot Leg	L1360 Ser. 14480	48500	91400
Hot Leg	L1360 Ser. 14482	43500	85700
Hot Leg	L1360 Ser. 14482	46200	89200
Hot Leg	K2980 Ser. 12585Y	42200	84900
Hot Leg	K2980 Ser. 12585Y	43000	87200
Hot Leg	L1360 Ser. 14481	44700	87900
Hot Leg	L1360 Ser. 14481	48200	90700
Cold Leg	K2980 Ser.11044Y	42200	84900
Cold Leg	K2980 Ser.11044Y	43700	88100
Cold Leg	L1336 Ser.14459	43400	86100
Cold Leg	L1336 Ser.14459	41700	81600
Cold Leg	K3660 Ser.14458	45400	89900
Cold Leg	K3660 Ser.14458	47200	91700
Cold Leg	K3723 Ser.14477X	45000	87400
Cold Leg	K3722 Ser.16100	41200	84900
Cold Leg	K3722 Ser.16100	41400	86600
Cold Leg	K3723 Ser.14477Y	47700	92400
Cold Leg	L1334 Ser.14457	40800	84800
Cold Leg	L1334 Ser.14457	41400	87400
X-Over Leg	L1359 Ser.14467Y	46200	89600
X-Over Leg	L1359 Ser.14467Y	50900	93400
X-Over Leg	L1334 Ser.14465Y	44700	88900
X-Over Leg	L1334 Ser.14465Y	44900	84400
X-Over Leg	L1336 Ser.14466Z	41400	83600
X-Over Leg	L1336 Ser.14466Z	43200	87400
X-Over Leg	L1336 Ser.14466W	41400	83600

Table 4-1 (Cont.)

Measured Tensile Properties for Byron Unit 1 Primary Loop Piping Material SA376 Gr. 304N

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)
X-Over Leg	L1334 Ser.14465X	44700	88900
X-Over Leg	L1336 Ser.14466X	43200	87400
X-Over Leg	L1334 Ser.14465W	44900	84400

Table 4-2

Measured Tensile Properties for Byron Unit 2 Primary Loop Piping Material SA376 Gr. 304N

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)
Hot Leg	K3660 Ser. 14483	47000	89900
Hot Leg	"	47500	90900
Hot Leg	K3660 Ser. 14484	45700	89900
Hot Leg	"	47000	88900
Hot Leg	K3660 Ser. 14485	45200	88400
Hot Leg	"	50000	91400
Hot Leg	K3660 Ser. 14486	46200	88700
Hot Leg	"	48500	90200
Cold Leg	K2980 Ser.11044U	42200	84900
Cold Leg	K3810 Ser. 16116	42200	85900
Cold Leg	"	42700	86400
Cold Leg	K2980 Ser.11044W	43700	88100
Cold Leg	L1360 Ser.14478	43700	86400
Cold Leg	"	50400	92200
Cold Leg	K3810 Ser.16115	42500	85900
Cold Leg	"	42900	85700
Cold Leg	K3722 Ser.16099	41200	84900
Cold Leg	"	43400	85900
X-Over Leg	K3722 Ser.15691Y	43500	87200
X-Over Leg	L1359 Ser.14467Z	46200	89600
X-Over Leg	"	50900	93400
X-Over Leg	K3723 Ser.15692Z	45000	87400
X-Over Leg	"	48000	91400
X-Over Leg	L1334 Ser.14465Z	44700	88900
	"	44900	84400
X-Over Leg	K3722 Ser.15691X	43700	86900

Table 4-2 (Cont.)

Measured Tensile Properties for Byron Unit 2 Primary Loop Piping Material SA376 Gr. 304N

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)
X-Over Leg	L1359 Ser.14467W	46200	89600
X-Over Leg	L1359 Ser.14467X	50900	93400

Table 4-3

Measured Tensile Properties for Braidwood Unit 1 Primary Loop Piping Material SA376 Gr. 304N

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)
Hot Leg	L3384 Ser. 27223	42700	84100
Hot Leg	L3384 Ser. 27224	42700	84100
Hot Leg	L3384 Ser. 27225	42700	84100
Hot Leg	L3387 Ser. 27229Z	42900	83600
Cold Leg	J6199 Ser. 21741Z	39900	82700
Cold Leg	J6199 Ser. 21741Y	39900	82700
Cold Leg	J6200 Ser. 27146Z	42400	80000
Cold Leg	J6200 Ser. 27146Y	42400	80000
Cold Leg	J6200 Ser. 27145Z	42400	80000
Cold Leg	J6200 Ser. 27145Y	42400	80000
Cold Leg	J6200 Ser. 27144Z	42400	80000
Cold Leg	J6200 Ser. 27146Y	42400	80000
X-Over Leg	L3385	41200	83900
X-Over Leg	L3385	41200	83900
X-Over Leg	L3386	42200	85900
X-Over Leg	L3386	42200	85900

Table 4-4

Measured Tensile Properties for Braidwood Unit 2 Primary Loop Piping Material SA376 Gr. 304N

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)
Hot Leg	J6343 Ser. 27698Z	41200	83900
Hot Leg	J6343 Ser. 27698Y	41200	83900
Hot Leg	J6349 Ser. 27700Z	39900	82400
Hot Leg	J6349 Ser. 27700Y	39900	82400
Cold Leg	J6344 Ser. 27672Y	41700	84600
	"	42400	84400
Cold Leg	J6344 Ser. 27673Z	41700	84600
	"	42400	84400
Cold Leg	J6344 Ser. 27672X	41700	84600
	"	42400	84400
Cold Leg	J6344 Ser. 27672Z	41700	84600
	"	42400	84400
Cold Leg	J6347 Ser. 27674Y	41900	84100
Cold Leg	J6347 Ser. 27675Y	41900	84100
Cold Leg	J6347 Ser. 27675X	41900	84100
Cold Leg	J6347 Ser. 27674Z	41900	84100
X-Over Leg	J6342 Ser. 27645Y	40400	83600
X-Over Leg	J6341 Ser. 27642Y	41800	84900
X-Over Leg	L3386	42600	87000
X-Over Leg	L3386	42600	87000

Table 4-5

Measured Room Temperature Tensile Properties for Byron Unit 1 Primary Loop Elbow Fittings

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)	Material
Hot Leg	97192-2	35600	84950	SA351-CF8A
Hot Leg	02158-1	36400	81600	SA351-CF8A
Hot Leg	00763-5	37750	82300	SA351-CF8A
Hot Leg	99556-2	38200	84050	SA351-CF8A
Cold Leg	02588-4	38500	79750	SA351-CF8A
Cold Leg	97577-7	35500	80800	SA351-CF8A
Cold Leg	04252-2	43050	86050	SA351-CF8A
Cold Leg	04303-2	42250	93100	SA351-CF8A
X-Over Leg	01064-2	38500	82700	SA351-CF8A
X-Over Leg	99599-3	41350	86200	SA351-CF8A
X-Over Leg	03109-1	39400	82350	SA351-CF8A
X-Over Leg	01285-2	39100	85450	SA351-CF8A
X-Over Leg	02231-1	39250	85800	SA351-CF8A
X-Over Leg	04506-2	40600	85200	SA351-CF8A
X-Over Leg	02772-1	40300	83100	SA351-CF8A
X-Over Leg	04171-1	40000	82800	SA351-CF8A
X-Over Leg	99386-1	41950	88400	SA351-CF8A
X-Over Leg	02930-1	40000	82300	SA351-CF8A
X-Over Leg	02727-1	39850	83950	SA351-CF8A
X-Over Leg	07128-1	44400	89350	SA351-CF8A

Table 4-6

Measured Room Temperature Tensile Properties for Byron Unit 2 Primary Loop Elbow Fittings

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)	Material
Hot Leg	05337-1	40600	89700	SA351-CF8A
Hot Leg	04342-3	40350	85400	SA351-CF8A
Hot Leg	03755-2	39250	83300	SA351-CF8A
Hot Leg	03692-2	41450	86950	SA351-CF8A
Cold Leg	04567-1	38050	84200	SA351-CF8A
Cold Leg	04640-3	37900	82850	SA351-CF8A
Cold Leg	04640-4	37900	82850	SA351-CF8A
Cold Leg	04606-2	42250	86300	SA351-CF8A
X-Over Leg	04567-1	38050	84200	SA351-CF8A
X-Over Leg	04603-2	39900	84400	SA351-CF8A
X-Over Leg	04817-2	37900	79850	SA351-CF8A
X-Over Leg	04640-1	37900	82850	SA351-CF8A
X-Over Leg	02576-1	38950	82650	SA351-CF8A
X-Over Leg	04146-1	43950	91450	SA351-CF8A
X-Over Leg	04050-1	38500	82700	SA351-CF8A
X-Over Leg	04377-1	41100	86750	SA351-CF8A
X-Over Leg	02833-1	39250	82550	SA351-CF8A
X-Over Leg	12426-1	39550	84600	SA351-CF8A
X-Over Leg	04932-1	41500	87800	SA351-CF8A
X-Over Leg	12669-1	42100	83200	SA351-CF8A

Table 4-7

Measured Room Temperature Tensile Properties for Braidwood Unit 1 Primary Loop Elbow Fittings

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)	Material
Hot Leg	04270-3	39700	87400	SA351-CF8A
Hot Leg	04171-2	40000	82800	SA351-CF8A
Hot Leg	03652-2	36100	78700	SA351-CF8A
Hot Leg	09464-2	37900	78850	SA351-CF8A
Cold Leg	04606-1	42250	86300	SA351-CF8A
Cold Leg	04641-3	37900	82850	SA351-CF8A
Cold Leg	04464-1	40000	87050	SA351-CF8A
Cold Leg	04641-2	37900	82850	SA351-CF8A
X-Over Leg	02772-2	40300	83100	SA351-CF8A
X-Over Leg	03002-1	36250	84800	SA351-CF8A
X-Over Leg	04567-2	38050	84200	SA351-CF8A
X-Over Leg	04568-2	43200	87650	SA351-CF8A
X-Over Leg	04431-1	37000	79350	SA351-CF8A
X-Over Leg	04252-1	43050	86050	SA351-CF8A
X-Over Leg	14936-1	37000	81550	SA351-CF8A
X-Over Leg	15822-2	37150	78300	SA351-CF8A
X-Over Leg	13670-1	36100	79450	SA351-CF8A
X-Over Leg	16141-1	38350	79050	SA351-CF8A
X-Over Leg	02801-1	41100	82200	SA351-CF8A
X-Over Leg	18524-1	40600	86000	SA351-CF8A

Table 4-8

Measured Room Temperature Tensile Properties for Braidwood Unit 2 Primary Loop Elbow Fittings

Component	Heat Number	Yield Room Temp. (psi)	Ultimate Room Temp (psi)	Material
Hot Leg	04407-1	39400	86900	SA351-CF8A
Hot Leg	04253-1	39100	85700	SA351-CF8A
Hot Leg	04146-2	43950	91450	SA351-CF8A
Hot Leg	03804-2	36550	78900	SA351-CF8A
Cold Leg	04567-4	38050	84200	SA351-CF8A
Cold Leg	04407-1	39400	86900	SA351-CF8A
Cold Leg	07006-1	37900	86400	SA351-CF8A
Cold Leg	11431-2	39100	85950	SA351-CF8A
X-Over Leg	16778-1	38350	81350	SA351-CF8A
X-Over Leg	17743-2	38450	82950	SA351-CF8A
X-Over Leg	15271-1	38950	81700	SA351-CF8A
X-Over Leg	16875-1	36550	79900	SA351-CF8A
X-Over Leg	13674-1	40600	80200	SA351-CF8A
X-Over Leg	18683-1	39950	83850	SA351-CF8A
X-Over Leg	18264-1	39750	84050	SA351-CF8A
X-Over Leg	18587-1	38800	84350	SA351-CF8A
X-Over Leg	04640-2	37900	82850	SA351-CF8A
X-Over Leg	05337-3	40600	89700	SA351-CF8A
X-Over Leg	07006-3	37900	86400	SA351-CF8A
X-Over Leg	07006-2	37900	86400	SA351-CF8A

Table 4-9

Mechanical Properties for Byron and Braidwood Units 1 and 2 Materials at Operating Temperatures

a,c,e

Table 4-10

Fracture Toughness Properties for Byron and Braidwood Units 1 and 2 Primary Loops for Leak-Before-Break Evaluation at Critical Location

a,c,e

a.c.e

Figure 4-1 Representative Lower Bound True Stress - True Strain Curve for SA351 CF8A at 617°F

a.c.e

Figure 4-2 Pre-Service J vs. Δa for Cast Stainless Steel at 600°F

a,c,e

Figure 4-3 **J Vs. Δa at Different Temperatures for Aged Material [ ]^{a,c,e} (7500 Hours at 400°C)**

SECTION 5.0
CRITICAL LOCATIONS AND EVALUATION CRITERIA

5.1 Critical Locations

The leak-before-break (LBB) evaluation margins are to be demonstrated for the limiting locations (governing locations). Candidate locations are designated load critical locations or toughness critical location as discussed in Section 3.0. Such locations are established based on the loads (Section 3.0) and the material properties established in Section 4.0. These locations are defined below for Byron and Braidwood Units 1 and 2. Table 3-2 as well as Figure 3-2 are used for this evaluation.

Load Critical Location

The highest stressed location for the SA376 Gr. 304N straight pipes is Location 11 at the reactor coolant pump outlet nozzle to pipe weld. Furthermore, since it is on a straight pipe, it is a high toughness location.

Toughness Critical Location

Low toughness locations are the elbows. All the elbows for the Byron and Braidwood Units 1 and 2 primary loop as indicated in page B-1 of Appendix B, exceed the toughness of [ ].[a,c,e] Hence , for all the elbows the toughness allowables are[ ][a,c,e] The highest stressed elbow is at location 3 where the temperature is 617°F. Location 3 is most limiting since it has the highest stress and the temperature is also higher. It is thus concluded that the enveloping location is 3. The allowable toughness for the critical location are shown in Table 4-10.

5.2 Fracture Criteria

As will be discussed later, fracture mechanics analyses are made based on loads and postulated flaw sizes related to leakage. The stability criteria against which the calculated J and tearing modulus are compared are:

(1) If $J_{app} < J_{Ic}$, then the crack is stable;

(2) If $J_{app} \geq J_{Ic}$, but, if $T_{app} < T_{mat}$ and $J_{app} < J_{max}$, then the crack is stable.

Where: J_{app} = Applied J
J_{Ic} = J at Crack Initiation
T_{app} = Applied Tearing Modulus
T_{max} = Material Tearing Modulus
J_{max} = Maximum J value of the material

These criteria apply to the toughness critical locations. For critical locations, the limit load method discussed in Section 7.0 is used.

SECTION 6.0
LEAK RATE PREDICTIONS

6.1 Introduction

The purpose of this section is to discuss the method which is used to predict the flow through postulated through-wall cracks and present the leak rate calculation results for through-wall circumferential cracks.

6.2 General Considerations

The flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. For long channels where the ratio of the channel length, L, to hydraulic diameter, D_H, (L/D_H) is greater than [ ]^{a,c,e}, both [

]^{a,c,e}.

6.3 Calculation Method

The basic method used in the leak rate calculations is the method developed by [

]^{a,c,e}

The flow rate through a crack was calculated in the following manner. Figure 6-1 from Reference 6-1 was used to estimate the critical pressure, Pc, for the primary loop enthalpy condition and an assumed flow. Once Pc was found for a given mass flow, the [ ]^{a,c,e} was found from Figure 6-2 (taken from Reference 6-1). For all cases considered, since [ ]^{a,c,e} Therefore, this method will yield the two-phase pressure drop due to momentum effects as illustrated in Figure 6-3, Po is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

ΔP_f = [-]^{a,c,e} (6-1)

where the friction factor f is determined using the [ ]^{a,c,e} The crack relative roughness, ε, was obtained from fatigue crack data on stainless steel samples. The relative roughness value used in these calculations was [ ]^{a,c,e}

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [momentum pressure drop calculated using the Fauske model]^{a,c,e} to obtain the total pressure drop from the primary system to the atmosphere. That is, for the primary loop

Absolute Pressure - 14.7 = [ ]^{a,c,e} (6-2)

for a given assumed flow rate G. If the right-hand side of equation 6-2 does not agree with the pressure difference between the primary loop and the atmosphere, then the procedure is repeated until equation 6-2 is satisfied to within an acceptable tolerance which in turn leads o correct flow rate value for a given crack size.

6.4 Leak Rate Calculations

Leak rate calculations were made as a function of crack length at the governing locations previously identified in Section 5.1. The normal operating loads of Table 3-1 were applied, in these calculations. The crack opening areas were estimated using the method of Reference 6-2 and the leak rates were calculated using the two-phase flow formulation described above. The average material properties of Section 4.0 were used for these calculations.

The flaw sizes to yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 6-1. The flaw sizes so determined are called <u>leakage flaws</u>.

The Byron and Braidwood Units 1 and 2 RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaws) are determined which yield a leak rate of 10 gpm.

6.5 References

6-1 [

]a,c,e.

6-2 Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," Section II-1, NUREG/CR-3464, September 1983.

6-3 WCAP-9558 Rev. 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," Westinghouse Proprietary Class 2, June 1981.

Table 6-1
Flaw Sizes Yielding a Leak Rate of 10 gpm
at the Governing Locations

[ ] a,c,e

a.c.e

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

a,c,e

Figure 6-2 [ ]^{a,c,e} Pressure Ratio as a Function of L/D

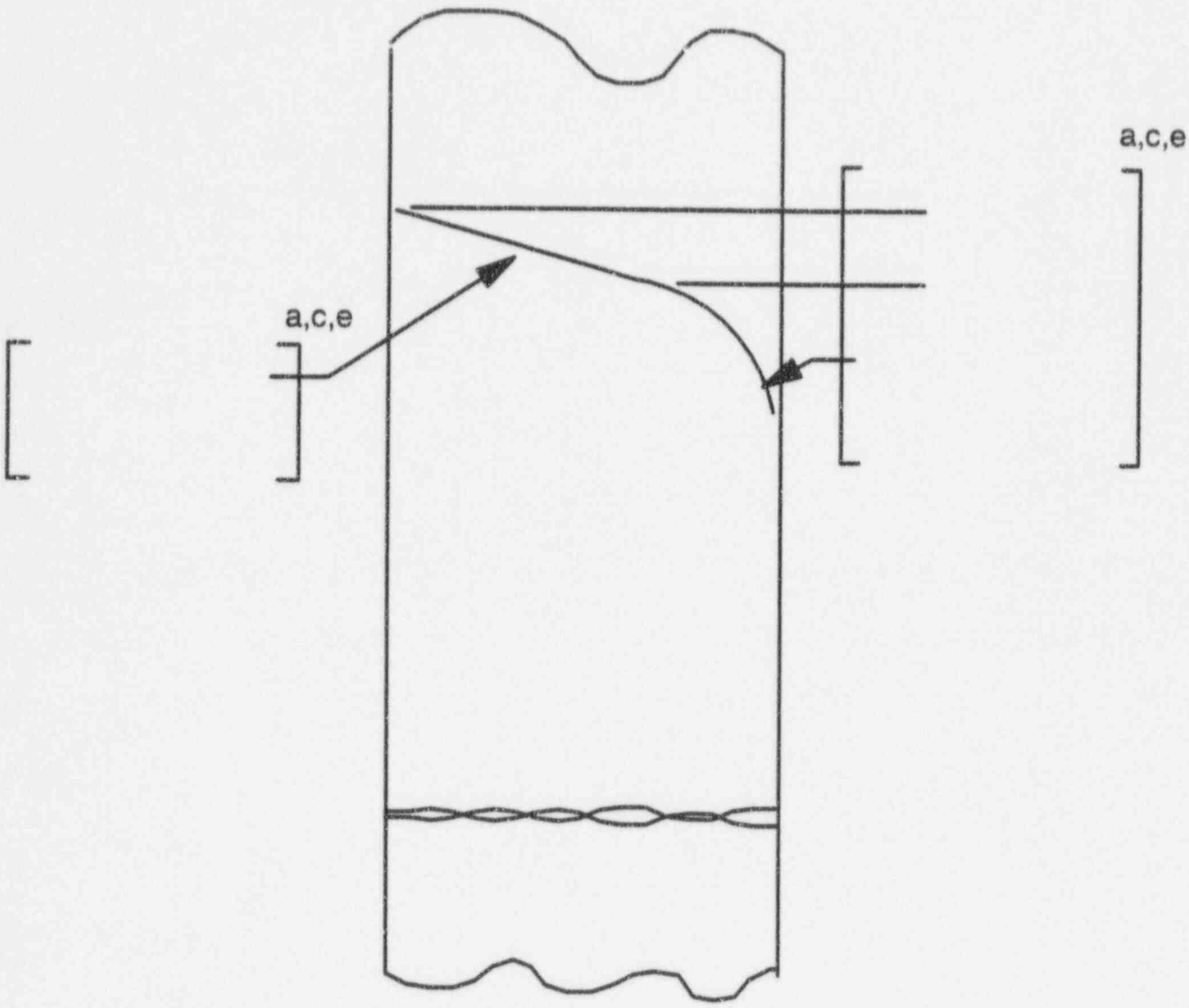


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

SECTION 7.0
FRACTURE MECHANICS EVALUATION

7.1 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. The local 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_{Ic} 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 the J_{Ic} 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:

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

where:

T_{app}	=	applied tearing modulus
E	=	modulus of elasticity
σ_f	=	0.5 ($\sigma_y + \sigma_u$) (flow stress)
a	=	crack length
σ_y, σ_u	=	yield and ultimate strength of the material, respectively

Stability is said to exist when ductile tearing occurs if T_{app} is less than T_{mat}, the experimentally determined tearing modulus. Since a constant T_{mat} is assumed a further restriction is placed in J_{app}. J_{app} must be less than J_{max} where J_{max} is the maximum value of J for which the experimental T is greater than or equal to the T_{mat} used.

As discussed in Section 5.2 the local crack stability will be established by the two-step criteria:

(1) If $J_{app} < J_{Ic}$, then the crack will not initiate.

(2) If $J_{app} \geq J_{Ic}$, but, if $T_{app} < T_{mat}$

and $J_{app} < J_{max}$, then the crack is stable.

7.2 Global Failure Mechanism

Determination of the conditions which lead to failure in stainless steel should be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One method for predicting the failure of ductile material is the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 7-1) 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 limit moment for such a pipe is given by:

[ ]^{a,c,e}

where:

[

]^{a,c,e}

σ_f = $0.5\ (\sigma_y + \sigma_u)$ (flow stress), psi

[ ]^{a,c,e}

[

]^{a,c,e}

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect the limit moment. Good agreement was found between the analytical predictions and the experimental results (Reference 7-1).

For application of the limit load methodology, the material, including consideration of the configuration, must have a sufficient ductility and ductile tearing resistance to sustain the limit load.

7.3 Results of Crack Stability Evaluation

Stability analyses were performed at the critical locations established in Section 5.1. The elastic-plastic fracture mechanics (EPFM) J-integral analyses for through-wall circumferential cracks in a cylinder were performed using the procedure in the EPRI fracture mechanics handbook (Reference 7-2).

The lower-bound material properties of Section 4.0 were applied (see Table 4-9). The fracture toughness properties established in Section 4.3 and the normal plus SSE loads given in Table 3-2 were used for the EPFM calculations. Evaluations were performed at the toughness critical locations identified in Section 5.1. The results of the elastic-plastic fracture mechanics J-integral evaluations are given in Table 7-1.

The critical locations were also identified in Section 5.1. A stability analysis based on limit load was performed for these locations as described in Section 7.2. The welds at these locations are TIG and SMAW combination. The "Z" factor correction for SMAW was applied (Reference 7-3) as follows:

$$Z=1.15[1.0+0.013(OD-4)]$$

where OD is the outer diameter of the pipe in inches.

The Z-factors were calculated for the toughness critical and load critical locations. The Z factors were 1.60 and 1.57 for locations 3 and 11 respectively. The applied loads were increased by the Z factors and plots of limit load versus crack length were generated as shown in Figures 7-2 and 7-3. Table 7-2 summarizes the results of the stability analyses based on limit load. The leakage size flaws are also presented on the same table.

7.4 References

7-1. Kanninen, M. F., et. al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks," EPRI NP-192, September 1976.

7-2. Kumar, V., German, M. D. and Shih, C. P., "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI Report NP-1931, Project 1237-1, Electric Power Research Institute, July 1981.

7-3. Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

Table 7-1

Stability Results for Byron and Braidwood Units 1 and 2 Based on Elastic-Plastic J-integral Evaluations

		Fracture Criteria			Calculated Values	
Location	**Flaw Size (in)**	**J_{Ic} (in-lb/in^2)**	**T_{mat}**	**J_{max} (in-lb/in^2)**	**J_{app} (in-lb/in^2)**	**T_{app}**

a,c,e

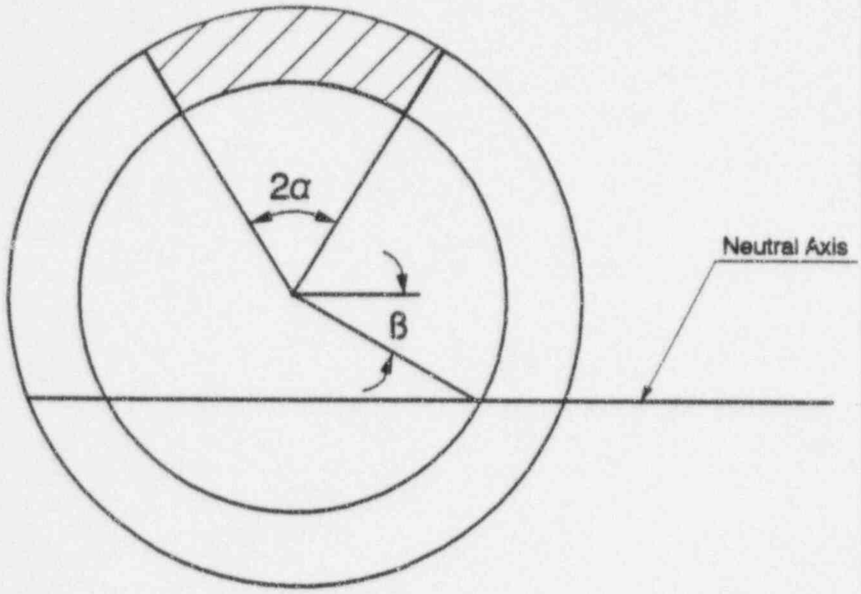
[ ]

Table 7-2

Stability Results for Byron and Braidwood Units 1 and 2 Based on Limit Load

Location	Flaw Size (in.)	Leakage Flaw Size(in)

a,c,e



a,c,e

Figure 7-1 [ ]^{a,c,e} Stress Distribution

a.c.e

OD = 34.21 in σ_y = 21.57 ksi F_a = 2573 kips
t = 2.50 in σ_u = 66.30 ksi M_b = 22713 in-kips

SA351 CF8A Material With SMAW Weld

Figure 7-2 Critical Flaw Size Prediction - Hot Leg at Location 3

a,c,e

OD = 32.14 in	σ_y = 23.11 ksi	F_a = 2275 kips
t = 2.215 in	σ_u =70.35 ksi	M_b = 31856 in-kips

SA376 Gr. 304N Material With SMAW Weld

Figure 7-3 Critical Flaw Size Prediction - Cold Leg at Location 11

SECTION 8.0
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 (see Location [ ]^{a,c,e} of Figure 3-2). 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 can be expected to show less than 10% variation.

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. A summary of generic applied transients is provided in Table 8-1. Circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in three different locations, as shown in Figure 8-1. Specifically, these were:

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

Fatigue crack growth rate laws were used [ ]^{a,c,e} The law for stainless steel was derived from Reference 8-1, 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:

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

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

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

[

]^{a,c,e}

[ ]$^{-a,c,e}$

where: [ ]^{a,c,e}

where ΔK is the stress intensity factor range.

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 8-2, and shows that the crack growth is very small, [ ]^{a,c,e}

8.1 References

8-1 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.

8-2 [

]^{a,c,e}

8-3 [

.]^{a,c,e}

Table 8-1

Summary of Reactor Vessel Transients

Number	Typical Transient Identification	Number of Cycles
	<u>Normal Conditions</u>	
1	Heatup and Cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200
2	Load Follow Cycles (Unit loading and unloading at 5% of full power/min)	18300
3	Step load increase and decrease	2000
4	Large step load decrease, with steam dump	200
5	Steady state fluctuations	10^6
	<u>Upset Conditions</u>	
6	Loss of load, without immediate turbine or reactor trip	80
7	Loss of power (blackout with natural circulation in the Reactor Coolant System)	40
8	Loss of Flow (partial loss of flow, one pump only)	80
9	Reactor trip from full power	400
	<u>Test Conditions</u>	
10	Turbine roll test	10
11	Hydrostatic test conditions	
	Primary side	5
	Primary side leak test	50
12	Cold Hydrostatic test	10

Table 8-2

Typical Fatigue Crack Growth at
[ ]a,c,e (40 years)

FINAL FLAW (in.)			
Initial 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 8-1 Typical Cross-Section of [ ]^{a,c,e}

a,c,e

[ ]

Figure 8-2 Reference Fatigue Crack Growth Curves for [ ]^{a,c,e}

a.c.e

Figure 8-3 **Reference Fatigue Crack Growth Law for [ ]^{a,c,e} in a Water Environment at 600°F**

SECTION 9.0
ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability and fracture toughness evaluations of Sections 7.1, 7.2 and 7.3 are used in performing the assessment of margins. Margins are shown in Table 9-1.

In summary, at all the critical locations relative to:

1. Flaw Size

   At location 3 using faulted loads obtained by the absolute sum method, a margin of more than 2 exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

   At location 11 using faulted loads obtained by the absolute sum method, a margin of 1.91 exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

2. Leak Rate - A margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.

3. Loads - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method.

Table 9-1

Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Byron and Braidwood Units 1 and 2

Location	Leakage Flaw Size	Critical Flaw Size	Margin

a,c,e

[a] based on limit load

[b] based on J integral evaluation

SECTION 10.0
CONCLUSIONS

This report justifies the elimination of RCS primary loop pipe breaks from the structural design basis for the Byron and Braidwood Units 1 and 2 as follows:

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

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

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

d. Ample margin exists between the leak rate of small stable flaws and the capability of the Byron and Braidwood Units 1 and 2 reactor coolant system pressure boundary Leakage Detection System.

e. Ample margin exists between the small stable flaw sizes of item d and larger stable flaws.

f. Ample margin exists in the material properties used to demonstrate end-of-service life (relative to aging) stability of the critical flaws.

For the critical locations flaws are identified that will be stable because of the ample margins described in d, e, and f above.

Based on the above, it is concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis of the Byron and Braidwood Units 1 and 2 Nuclear Power Plant.

APPENDIX A
LIMIT MOMENT

[

]^{a,c,e}

a,c,e

Figure A-1 Pipe with a Through-Wall Crack in Bending

APPENDIX B

TOUGHNESS CRITERIA FOR BYRON AND BRAIDWOOD UNITS 1 AND 2 CAST PRIMARY LOOP COMPONENTS

All of the individual cast piping components of the Byron and Braidwood Units 1 and 2 primary loop piping satisfy the original [ ]^{a,c,e} criteria (Reference 4-5). [

]^{a,c,e}

Table B-1

Chemistry and Fracture Toughness Properties of the Material Heats of Byron Unit 1

a,c,e

Table B-1 (Cont.)

a,c,e

Table B-2

Chemistry and Fracture Toughness Properties of the Material Heats of Byron Unit 2

a,c,e

Table B-2 (Cont.)

a,c,e

Table B-3

Chemistry and Fracture Toughness Properties of the Material Heats of Braidwood Unit 1

a,c,e

Table B-3 (Cont.)

a,c,e

Table B-4

Chemistry and Fracture Toughness Properties of the Material Heats of Braidwood Unit 2

a,c,e

Table B-4 (Cont.)

a,c,e

a,c,e

ATTACHMENT A

WCAP-14559 Revision 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants"- Proprietary Version