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Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant Units 1 and 2 for the Power Uprate and License Renewal Program

Westinghouse

WCAP-14439-NP Revision 0

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Note: The proprietary version of this report is documented in WCAP-14439-P Revision 2.

D. C. Bhowmick

September 2003

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EXECUTIVE SUMMARY

The original structural design basis of the reactor coolant system for the Wisconsin Electric Power Company Point Beach Units 1 and 2 Nuclear Power Plants 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 Point Beach 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 1-2). 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. Generic analyses by Westinghouse for the application of LBB for specific plants were documented in response to Unresolved Safety Issue A-2 and approved for Point Beach in the NRC letter dated May 6, 1986 (Reference 1-3). By letter dated May 6, 1986, the NRC stated that:

> "By letter dated May 30, 1985, you requested an exemption from the requirements of 10 CFR 50 Appendix A, General Design Criterion (GDC) 4, to eliminate the consideration of large reactor coolant system primary loop pipe breaks in the structural design basis of the Point Beach Nuclear Plant Units 1 and 2. You requested this exemption as directed by staff guidance contained in Generic Letter 84-04, dated February 1, 1984, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops." The Safety Evaluation contained in the generic letter concluded that an acceptable technical basis had been provided so that the asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owner's Group plants listed (two of which were Point Beach Units 1 and 2) provided certain conditions were met. Your May 30, 1985 exemption request provided information showing that the first of these conditions was not applicable to Point Beach Units 1 and 2 and that the second condition was met for both units.
>
> Subsequent to your submittal, the Commission has published in the Federal Register on April 11, 1986 (51 FR 12502) its final rule, "Modification of General

> Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures." This change to the rule allows use of leak-before-break technology for excluding from the design basis the dynamic effects of postulated ruptures in primary coolant loop piping in pressurized water reactors (PWRs). The new rule is effective May 12, 1986.
>
> Based upon the Commission's issuance of the final rule modifying GDC 4 of Appendix A to 10 CFR Part 50, the staff has determined that the exemption requested in your May 30, 1985 application is no longer needed."

Westinghouse performed a LBB analysis for the Point Beach Units 1 and 2 primary loop piping in 1996 and the results of the analysis were documented in WCAP-14439 (Reference 1-4). The report demonstrates compliance with LBB technology for the Point Beach reactor coolant system piping based on a plant specific analysis. Westinghouse has performed a revised LBB evaluation for the Point Beach Units 1 and 2 primary loop piping to incorporate the license renewal and power uprate programs. The purpose of this revised report is to document the results of the revised evaluation results and as well as to provide a general update of the original WCAP-14439. This revision 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, end of life (60 year) fracture toughness considering thermal aging was determined for each heat of material.

This Report includes the temperature, pressure and loadings generated as a result of the Point Beach Units 1 and 2 power uprate program.

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 a 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, using the plant specific transients and cycles, fatigue crack growth for the 60 years was shown to be acceptable for the primary loops. All the recommended LBB margins (margin on leak rate, margin on flaw size and margin on loads) are satisfied.

It is therefore concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis of the Point Beach Units 1 and 2 Nuclear Power Plants for the license renewal program and for the power uprate program.

1.0 INTRODUCTION

1.1 PURPOSE

This report applies to the Point Beach Nuclear Plant Units 1 and 2 Reactor Coolant System (RCS) primary loop piping. It is intended to demonstrate that for the specific parameters of the Point Beach Nuclear Plant Units 1 and 2, RCS primary loop pipe breaks need not be considered in the structural design basis for a 60 year plant life and for the power uprate program. The Nuclear Regulatory Commission (NRC) (Reference 1-2) has accepted the approach taken.

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 1-5). 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, 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 1-6).

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 1-7 and 1-8). 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 1-5) 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 1-2) 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. 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 1-9).

1.3 SCOPE AND OBJECTIVES

The general purpose of this investigation is to demonstrate leak-before-break for the primary loops in Point Beach Units 1 and 2 on a plant specific basis for a 60 year plant life and for the power uprate program (i.e., an uprate of up to 10.4% reactor power). The recommendations and criteria proposed in Reference 1-10 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 locations at which the highest stress occurs.

2. Identify the materials and the associated material properties.

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

4. Postulate a through-wall flaw at the governing locations. 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 flaw size and the critical flaw size.

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.

8. Demonstrate margin on applied load.

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

The computer codes that are used in this evaluation for leak rate and fracture mechanics calculations have been validated and used for all the LBB applications by Westinghouse.

1.4 REFERENCES

1-1 WCAP-7211, Revision 4, "Energy Systems Business Unit Policy and Procedures for Management, Classification, and Release of Information," January 2001.

1-2 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.

1-3 Nuclear Regulatory Commission Docket #'s 50-266 and 50-301 Letter from G. E. Lear, Director PWR Project Directorate #1 Division of PWR Licensing-A, NRC, to C. W. Fay, Vice President Nuclear Power Department Wisconsin Electric Power Company.

1-4 WCAP-14439, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Units 1 and 2 Nuclear Power Plants," December 1996.

1-5 WCAP-9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March 1978.

1-6 Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, and November 10, 1981.

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

1-8 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

1-9 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.

1-10 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.

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 1100 reactor-years, including 5 plants each having over 30 years of operation, 14 plants each with over 25 years of operation, 12 other plants each with over 20 years of operation and 8 plants each over 15 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 a result of the recent issue of Primary Water Stress Corrosion Cracking (PWSCC) occurring in the V. C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld is being currently investigated under the EPRI Material Reliability Project (MRP) Program. It should be noted that the susceptible material under investigation is not found in the primary loop piping at the Point Beach Nuclear Plant Unit 1. The Alloy 82 weld exists between the Steam Generators (SG) nozzle-to-safe end weld locations with Alloy 152 buttering on the inside surfaces at the Point Beach Nuclear Plant Unit 2 and the welding was performed in the shop under controlled conditions and with improved welding technology. Alloy 152 weld material, in contact with the primary coolant, is considered to have improved PWSCC resistance. Since the Alloy 82 weld in

the primary loop piping for Point Beach Unit 2 is not exposed to the primary coolant, PWSCC should not be a concern.

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; 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 has 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

An assessment 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 operation. During operation, an alarm signals the exceedence of the vibration limits. Field measurements have been made on a number of plants during hot functional testing, including plants similar to Point Beach 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.

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.

3.0 PIPE GEOMETRY AND LOADING

3.1 INTRODUCTION TO METHODOLOGY

The general approach is discussed first. As an example a segment of the primary coolant hot leg pipe is shown in Figure 3-1. The as-built outside diameter and minimum wall thickness of the pipe are 34.21 in. and 2.50 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 1 at the reactor vessel outlet nozzle to pipe weld (see Figure 3-2). This highest stressed location is a <u>load critical location</u> 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 results in lower fracture toughness; thus, locations other than the load critical locations must be examined taking into consideration both fracture toughness and stress. The enveloping locations so determined are called <u>toughness critical locations</u>. The three most critical locations are identified after the full analysis is completed. Once loads (this section) and fracture toughness (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 for evaluation are those 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:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{3-1}$$

where,

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

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

$$M=(Mx^{2}+My^{2}+Mz^{2})^{0.5} \tag{3-2}$$

where,

M	=	moment for required loading
M_x	=	x component of moment
M_y	=	y component of bending moment
M_z	=	z component of bending moment

NOTE: x-axis is along the centerline of the pipe.

The axial load and 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:

$$F = F_{DW} + F_{TH} + F_{P} \tag{3-3}$$

$$M_x = (M_x)_{DW} + (M_x)_{TH} \tag{3-4}$$

$$M_y = (M_y)_{DW} + (M_y)_{TH} \tag{3-5}$$

$$M_z = (M_z)_{DW} + (M_z)_{TH} \tag{3-6}$$

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. The as-built dimensions are also given.

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:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3\text{-}7)$$

$$M_x = |(M_x)_{DW}| + |(M_x)_{TH}| + |(M_x)_{SSEINERTIA}| + |(M_x)_{SSEAM}| \quad (3\text{-}8)$$

$$M_y = |(M_y)_{DW}| + |(M_y)_{TH}| + |(M_y)_{SSEINERTIA}| + |(M_y)_{SSEAM}| \quad (3\text{-}9)$$

$$M_Z = |(M_z)_{DW}| + |(M_z)_{TH}| + |(M_z)_{SSEINERTIA}| + |(M_z)_{SSEAM}| \quad (3\text{-}10)$$

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

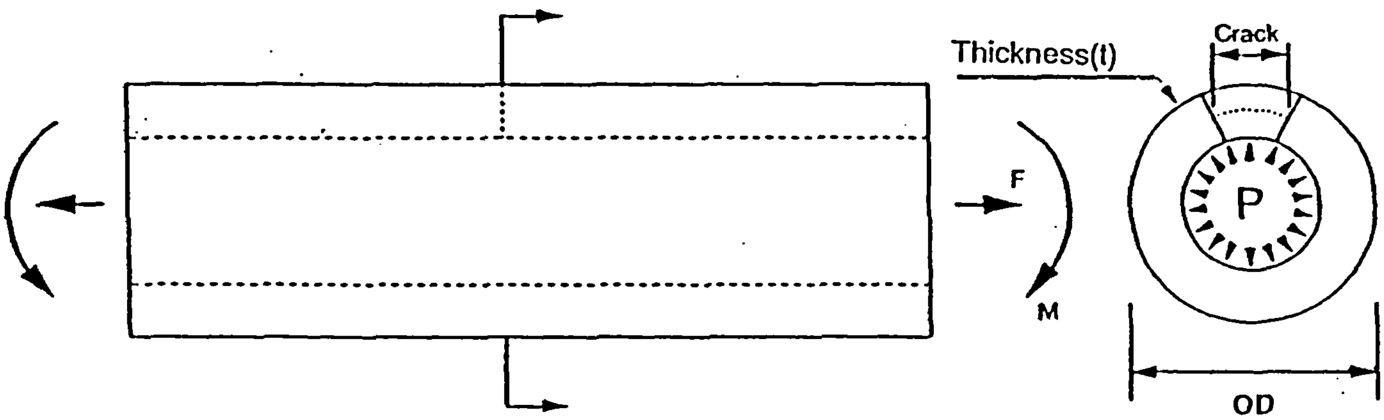
3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

Table 3-1 Dimensions, Normal Loads and Normal Stresses for Point Beach Units 1 & 2					
Location[a]	Outside Diameter (in)	Minimum Thickness (in)	Axial Load[b] (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	34.21	2.500	1,412	13,165	12.82
2	34.21	2.500	1,412	4,102	7.90
3	37.75	3.270	1,585	6,424	6.76
4	37.63	3.210	1,713	1,310	5.41
5	36.56	2.675	1,710	1,447	6.65
6	36.56	2.675	1,707	1,609	6.71
7	36.56	2.675	1,717	1,215	6.57
8	36.56	2.675	1,713	2,449	7.10
9	37.63	3.210	1,767	4,786	6.83
10	32.46	2.375	1,361	2,804	7.85
11	32.46	2.375	1,361	4,111	8.68
12	33.56	2.925	1,362	5,182	7.45

a. See Figure 3-2
b. Included Pressure

Table 3-2 Faulted Loads and Stresses for Point Beach Units 1 & 2			
Location[a,b]	Axial Load[c] (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	1,715	20,688	18.12
2	1,715	6,080	10.19
3	1,943	10,696	9.29
4	1,794	11,387	9.30
5	1,785	8,438	10.02
6	1,781	6,895	9.32
7	1,798	5,856	8.92
8	1,803	9,070	10.36
9	1,793	11,098	9.19
10	1,399	7,378	10.92
11	1,426	14,069	15.29
12	1,401	17,254	13.66

a. See Figure 3-2
b. See Table 3-1 for dimensions
c. Includes Pressure

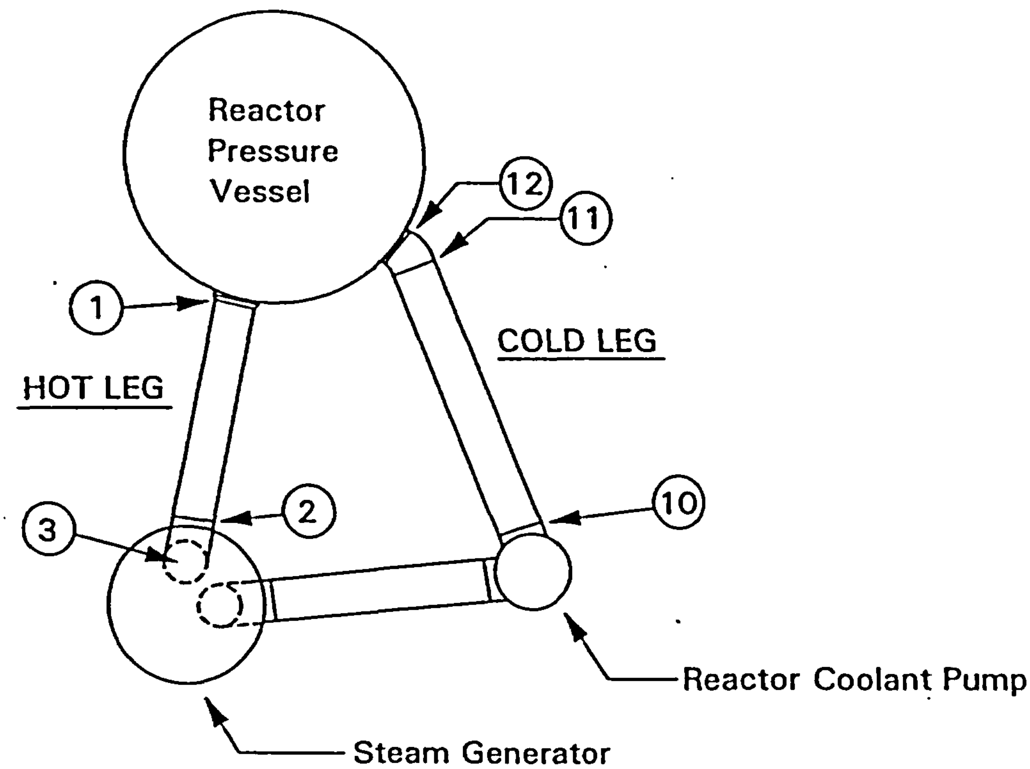


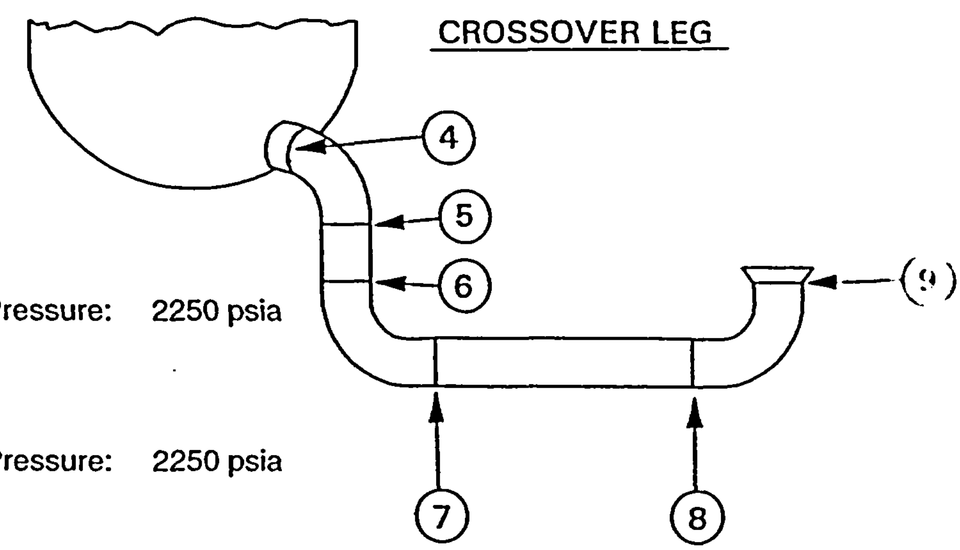
OD[a] = 34.21 in
t[a] = 2.500 in

Normal Loads[a]		Faulted Loads[b]	
force[c]:	1,412 kips	force[c]:	1,715 kips
moment:	13,165 in-kips	moment:	20,688 in-kips

[a] See Table 3-1
[b] See Table 3-2
[c] Includes the force due to a pressure of 2250 psia

Figure 3-1 Hot Leg Coolant Pipe





HOT LEG

Temperature 605.5°F, Pressure: 2250 psia

CROSS-OVER LEG

Temperature 541.1°F, Pressure: 2250 psia

COLD LEG

Temperature 541.4°F, Pressure: 2250 psia

Figure 3-2 Schematic Diagram of Point Beach Units 1 and 2 Primary Loop Showing Weld Locations

4.0 MATERIAL CHARACTERIZATION

4.1 PRIMARY LOOP PIPE AND FITTINGS MATERIALS

The primary loop piping is A376 TP316 and the elbow fittings are A351 CF8M for the Point Beach Units 1 and 2.

4.2 TENSILE PROPERTIES

The piping Certified Materials Test Reports (CMTRs) for Point Beach Units 1 and 2 were used to establish the tensile properties for the Leak-Before-Break analysis. The CMTRs include tensile properties at room temperature and/or at 650°F for each of the heats of material. These properties are given in Tables 4-1, 4-2 for piping and in Tables 4-3 and 4-4 for elbows fittings.

For the A376 TP316 material, the representative properties at 605.5°F were established from the tensile properties at 650°F given in Tables 4-1 and 4-2 by utilizing Section III of the 1989 ASME Boiler and Pressure Vessel Code (Reference 4-1). Code tensile properties at 605.5°F were obtained by interpolating between the 600°F and 650°F tensile properties. Ratios of the code tensile properties at 605.5°F to the corresponding tensile properties at 650°F were then applied to the 650°F tensile properties given in Tables 4-1 and 4-2 to obtain the plant specific properties for A376 TP316 at 605.5°F.

The Elbow Fittings Certified Materials Test Reports (CMTRs) for Point Beach Units 1 and 2 were used to establish the tensile properties for the Leak-Before-Break analysis. The CMTRs for elbow fittings include tensile properties at room temperature for each of the heats of material. These properties are given for Point Beach Units 1 and 2 in Tables 4-3 and 4-4.

For the A351 CF8M material, the representative properties at 605.5°F and 541.4°F were established from the tensile properties at room temperature given in Tables 4-3 and 4-4 by utilizing section III of the 1989 ASME boiler and pressure vessel code. Code tensile properties at 605.5°F and 541.4°F were established by interpolating between the 500°F, 600°F and the 650°F tensile properties. Ratios of the code tensile properties at 605.5°F and 541.4°F to the corresponding properties at room temperature were then applied to the room temperature properties given in Tables 4-3 and 4-4 to obtain the plant specific representative properties for A351 CRF8M at 605.5°F and 541.4°F. The material properties for the cross-over leg are used from 541.4°F instead of 541.1°F. The difference is insignificant and therefore acceptable.

The average and lower bound yield strengths and lower bound ultimate strengths are given in Table 4-5. The ASME code moduli of elasticity are also given in Table 4-5, and poisson's ratio was taken as 0.3.

4.3 FRACTURE TOUGHNESS PROPERTIES

The pre-service fracture toughnesses of cast stainless steels in terms of J_{Ic} have been found to be very high at 600°F. [

]^{a,c,e} However, cast stainless steel is susceptible to thermal aging at the reactor operating temperature, that is, about 290°C (550°F). Thermal aging of cast stainless steel results in embrittlement, that is, a decrease in the ductility, impact strength, and fracture toughness, of the material. Depending on the material composition, the Charpy impact energy of a cast stainless steel component could decrease to a small fraction of its original value after exposure to reactor temperatures during service.

The susceptibility of the material to thermal aging increases with increasing ferrite contents. The molybdenum bearing CF8M shows increased susceptibility to thermal aging. The end of life fracture toughness values calculated by Westinghouse methodology and shown in WCAP-14439 (Reference 4-2) were conservative and were not used for this current evaluation, an alternate method as described below was used to calculate the end of life toughness properties for the cast material.

In 1994, the Argonne National Laboratory (ANL) completed an extensive research program in assessing the extent of thermal aging of cast stainless steel materials. The ANL research program measured mechanical properties of cast stainless steel materials after they have been heated in controlled ovens for long periods of time. ANL compiled a data base, both from data within ANL and from international sources, of about 85 compositions of cast stainless steel exposed to a temperature range of 290-400°C (550-750°F) for up to 58,000 hours (6.5 years). From this database, ANL developed correlations for estimating the extent of thermal aging of cast stainless steel (References 4-3 and 4-4).

ANL developed the fracture toughness estimation procedures by correlating data in the database conservatively. After developing the correlations, ANL validated the estimation procedures by comparing the estimated fracture toughness with the measured value for several cast stainless steel plant components removed from actual plant service. The ANL procedures produced conservative estimates that were about 30 to 50 percent less than actual measured values. The procedure developed by ANL in Reference 4-4 was used to calculate the end of life fracture toughness values for this analysis. ANL research program was sponsored and the procedure was accepted (Reference 4-5) by the NRC.

The chemical compositions are available from CMTRs and are provided in Table 4-6.

$Cr_{(eq)}$= chromium equivalent; $Ni_{(eq)}$ = nickel equivalent; where $Cr_{(eq)}$ and Ni_{eq} are in percent weight

δ_c= ferrite in percent volume.

$Cr_{(eq)}$, $Ni_{(eq)}$ and δ_c values obtained from Reference 4-2 are shown in Table 4-6.

The following equations are taken from Reference 4-4.

For CF 8M steel with <10% Ni the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10} C_{Vsat} = 1.10 + 2.12 \exp(-0.041\phi) \qquad (4\text{-}1)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2 (C + 0.4N)/5; \qquad (4\text{-}2)$$

and from

$$\log_{10} C_{Vsat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si$$
$$-0.007Ni - 4.71(C + 0.4N) \qquad (4\text{-}3)$$

For CF 8M steel with >10% Ni, the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10} C_{Vsat} = 1.10 + 2.64 \exp(-0.064\phi) \qquad (4\text{-}4)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2 (C + 0.4N)/5 \qquad (4\text{-}5)$$

and from

$$\log_{10} C_{Vsat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si$$
$$-0.007Ni - 4.71(C + 0.4N) \qquad (4\text{-}6)$$

The saturation room temperature (RT) impact energies of the cast stainless steel materials were determined from the chemical compositions available from CMTRs and provided in Table 4-6.

The saturation J-R curve at 290°C (554°F), for static-cast CF 8M steel is given by

$$J_d = 49 (C_{Vsat})^{0.41} (\Delta a)^n \qquad (4\text{-}7)$$

$$n = 0.23 + 0.06 \log_{10} (C_{Vsat}) \qquad (4\text{-}8)$$

where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

[

][a,c,e]

[

][a,c,e]

The correlations presented in Reference 4-4 is applicable to cast stainless steels used in the U. S. Nuclear industry. The steels contain <25% ferrite in almost all cases. [

][a,c,e]

The results from the ANL Research Program indicate that the lower-bound fracture toughness of thermally aged cast stainless steel is similar to that of submerged arc welds (SAWs). 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[1]. Therefore, weld regions are less limiting than the cast material.

In the fracture mechanics analyses that follow, the fracture toughness properties given in Table 4-7 will be used as the criteria against which the applied fracture toughness values will be compared.

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

4.4 REFERENCES

4-1 ASME Boiler and Pressure Vessel Code Section III, "Rules for construction of Nuclear Power Plant Components; Division 1, Appendices." 1989 Edition, July 1,1989.

4-2 WCAP-14439, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Units 1 and 2 Nuclear Power Plants," December 1996.

4-3 O. K. Chopra and W. J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U. S. Nuclear Regulatory Commission, Washington, DC, May 1994.

4-4 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG-CR-4513, Revision 1, U. S. Nuclear Regulatory Commission, Washington, DC, August 1994.

4-5 "Flaw Evaluation of Thermally aged Cast Stainless Steel in Light-Water Reactor Applications," Lee, S.; Kuo, P. T.; Wichman, K.; Chopra, O.; Published in International Journal of Pressure Vessel and Piping, June 1997.

Table 4-1 Measured Tensile Properties (psi) for Point Beach Unit 1 Primary Loop Pipes						
			At Room Temperature		At 650°F	
HEAT NO.	SERIAL NO.	LOCATION	YIELD STRENGTH	ULTIMATE STRENGTH	YIELD STRENGTH	ULTIMATE STRENGTH
F0056	2627	Hot Leg	41900	82200	33100	72100
F0056	2627	Hot Leg	47600	90400	N/A	N/A
52154	1985Y	Hot Leg	32300	75000	24300	61000
52154	1985Y	Hot Leg	32400	75300	N/A	N/A
F0058	2630	Hot Leg	43500	86200	26600	71000
F0058	2630	Hot Leg	43900	85200	N/A	N/A
52154	1985X	Hot Leg	32400	75300	24300	61000
52154	1985X	Hot Leg	32300	75000	N/A	N/A
F0060	2633Y	Cold Leg	45000	89400	23100	74200
F0060	2632	Cold Leg	41600	86300	23100	74200
F0060	2632	Cold Leg	46900	89000	N/A	N/A
D8649	1475X	Cold Leg	32700	76400	20500	56700
D8649	1475X	Cold Leg	31700	75000	N/A	N/A
F0060	2633X	Cold Leg	41400	86100	23100	74200
51254	1983Y	X-Over Leg	34600	79300	20600	62200
V0246	1981	X-Over Leg	33200	75900	25300	68600
V0246	1981	X-Over Leg	31700	75500	N/A	N/A
51254	1983X	X-Over Leg	34000	75500	20600	62200
F0056	2626	X-Over Leg	40700	82200	22100	66800
F0056	2626	X-Over Leg	47000	87400	31800	74800

N/A= Not applicable

Table 4-2 Measured Tensile Properties (psi) for Point Beach Unit 2 Primary Loop Pipes						
			At Room Temperature		At 650°F	
HEAT NO.	**SERIAL NO.**	**LOCATION**	**YIELD STRENGTH**	**ULTIMATE STRENGTH**	**YIELD STRENGTH**	**ULTIMATE STRENGTH**
F0213	2890Y	Hot Leg	41900	82500	23700	69800
F0213	2890Y	Hot Leg	46500	84500	N/A	N/A
F0227	2897Y	Hot Leg	42500	85500	20900	66600
F0227	2897Y	Hot Leg	41800	86800	N/A	N/A
F0225	2896Y	Hot Leg	44000	86900	22100	68200
F0225	2896Y	Hot Leg	48500	88900	N/A	N/A
F0225	2895Y	Hot Leg	42000	85900	22100	68200
F0225	2895Y	Hot Leg	46000	88800	N/A	N/A
F0382	3131	Cold Leg	45700	86900	25300	72600
F0382	3131	Cold Leg	45000	90500	N/A	N/A
V0631	3123	Cold Leg	37000	77200	22000	60000
V0631	3123	Cold Leg	47700	83500	N/A	N/A
F0212	2867Y	X-Over Leg	43500	84200	34400	67000
F0212	2867Y	X-Over Leg	43900	82300	N/A	N/A
F0221	2865Y	X-Over Leg	39300	82600	22600	65200
F0221	2865Y	X-Over Leg	45500	89000	N/A	N/A
F0212	2887Y	X-Over Leg	43500	84200	34400	67000
F0212	2887Y	X-Over Leg	43900	82300	N/A	N/A
E1483	3362	X-Over Leg	43200	89700	23700	52100
E1483	3362	X-Over Leg	42900	89900	N/A	N/A

N/A= Not applicable

Table 4-3 Measured Room Temperature Tensile Properties for Point Beach Unit 1 Primary Loop Elbow Fittings

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp(psi)	Material Type
Hot Leg	04258-6	46500	90000	A351CF8M
Hot Leg	04099-1	55500	89000	A351CF8M
Cold Leg	06874-2	45000	87000	A351CF8M
Cold Leg	07242-1	45000	85500	A351CF8M
X-Over Leg	02710-2	53000	92000	A351CF8M
X-Over Leg	07392-1	43500	85500	A351CF8M
X-Over Leg	07896-1	43500	85500	A351CF8M
X-Over Leg	08849-1	45000	92000	A351CF8M
X-Over Leg	09439-1	48000	91500	A351CF8M
X-Over Leg	02259-1	49500	88500	A351CF8M
X-Over Leg	07599-1	55500	92500	A351CF8M
X-Over Leg	07848-1	51000	88500	A351CF8M
X-Over Leg	8325-1	40500	85000	A351CF8M
X-Over Leg	09216-1	49500	92500	A351CF8M

Table 4-4 Measured Room Temperature Tensile Properties for Point Beach Unit 2 Primary Loop Elbow Fittings

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp(psi)	Material Type
Hot Leg	10400-3	46500	89000	A351CF8M
Hot Leg	11007-1	45000	86000	A351CF8M
Cold Leg	11575-3	42000	82000	A351CF8M
Cold Leg	11556-4	43500	85000	A351CF8M
X-Over Leg	12660-1	42000	83000	A351CF8M
X-Over Leg	12547-1	43500	86500	A351CF8M
X-Over Leg	13174-1	45000	86000	A351CF8M
X-Over Leg	11575-1	40500	81500	A351CF8M
X-Over Leg	11897-1	42000	82000	A351CF8M
X-Over Leg	12476-2	40500	86000	A351CF8M
X-Over Leg	12584-1	42000	83000	A351CF8M
X-Over Leg	12476-3	42000	85500	A351CF8M
X-Over Leg	11632-1	37500	75000	A351CF8M
X-Over Leg	12012-1	42000	81500	A351CF8M

Table 4-5 Mechanical Properties for Point Beach Units 1 and 2 Materials at Operating Temperatures				
Material	Temperature (°F)	Average Yield Strength (psi)	Lower Bound	
			Yield Stress (psi)	Ultimate Strength (psi)
A376 TP316	605.5	25127	20796	52100
A351 CF8M	605.5	28262	23459	71786
	541.4	29283	24306	71786
Modulus of Elasticity				
E = 25.27x 10^6 psi, at 605.5°F				
E = 25.59 x 10^6 psi, at 541.4°F				
Poisson's ratio: 0.3				

a,c,e

a,c,e

[ ]

a,c,e

Figure 4-1 Pre-Service J vs. Δa for SA351 CF8M Cast Stainless Steel at 600°F

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 location or toughness critical locations as discussed in Section 3.0. Such locations are established based on the loads in Section 3.0 and the material properties established in Section 4.0. These locations are defined below for Point Beach Units 1 and 2. Table 3-2 as well as Figure 3-2 is used for this evaluation.

Critical Locations

The highest stressed location for the entire primary loop is at Location 1 (in the Hot Leg) (See Figure 3-2) at the reactor vessel outlet nozzle to pipe weld. Location 1 is the critical location for all the weld locations in the primary loop piping. Furthermore, since it is on a straight pipe with forged material, it is a high toughness location.

Toughness Critical Locations

Low toughness locations are at the ends of every elbow. In the case of the hot leg low toughness are at locations 2 and 3 (see figure 3-2 for locations). Location 2 governs since it has a higher stress than location 3. In the case of cross over leg the lowest toughness is at locations 8 and 9. Location 8 governs since it has a higher stress than location 9. In the case of cold leg the low toughness is at locations 11 and 12. Location 11 governs since it has a higher stress than location 12. It is thus concluded that the enveloping locations are 2, 8 and 11. The allowable toughness values for the critical locations are shown in Table 4-7.

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 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.

Where:

J_{app} = Applied J

J_{IC} = J at Crack Initiation

T_{app} = Applied Tearing Modulus

T_{mat} = Material Tearing Modulus

J_{max} = Maximum J value of the material

For critical locations, the limit load method discussed in Section 7.0 was also used.

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}.

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). Po is the operating pressure. 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. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

$$\Delta P_f = [\qquad]^{a,c,e} \qquad (6\text{-}1)$$

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} (Reference 6-2).

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [ ]^{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 to 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 6-3 and the leak rates were calculated using the two-phase flow formulation described above. The average material properties of Section 4.0 (see Table 4-5) were used for these calculations.

The flaw sizes (crack lengths) to yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 6-1. The flaw sizes (crack lengths) so determined are called leakage flaw sizes (crack lengths).

The Point Beach Units 1 and 2 RCS pressure boundary leak detection system capability is 1 gpm in 4 hours. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaw sizes) (crack lengths) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [

]^{a,c,e}.

6-2 WCAP-9558, Revision 2," Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated circumferential Through-wall Crack, Longitudinal Through-Crack in a Pipe," May 1981 (Westinghouse Proprietary Class 2).

6-3 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.

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm at the Governing Locations	
Location	**Leakage Flaw Size (in)**
1	5.41
2	7.61
8	7.99
11	6.88

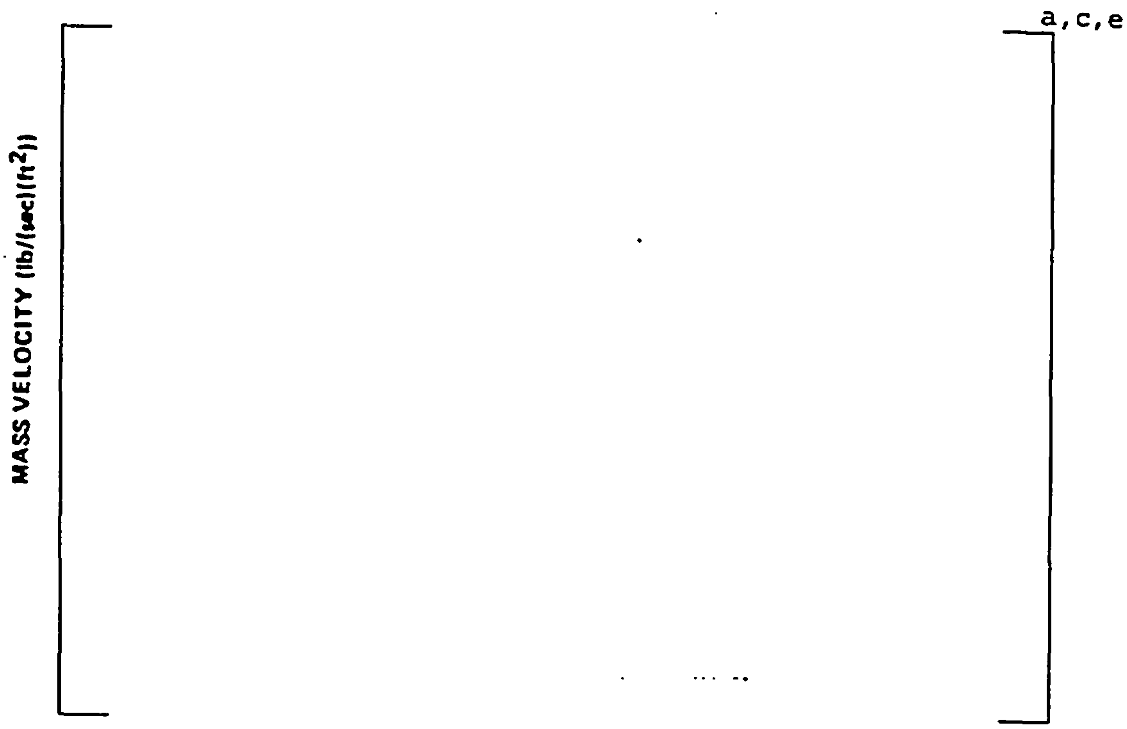


Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

a,c,e

CRITICAL PRESSURE RATIO (p_c/p_o)

LENGTH/DIAMETER RATIO (L/D)

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

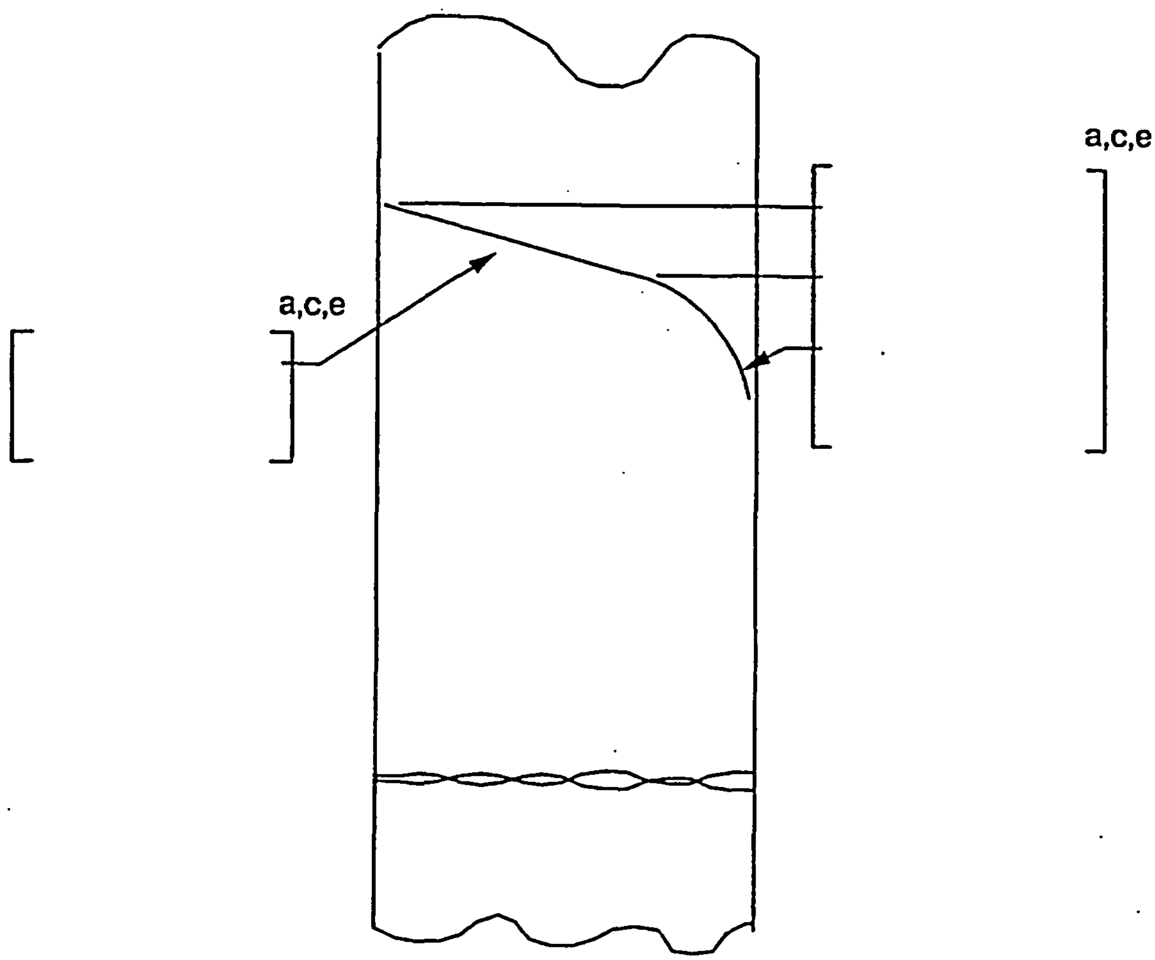


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

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 on 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 T_{app} used.

As discussed in Section 5.2 the local crack stability criteria is a two-step process:

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

(2) If $J_{app} > J_{Ic}$, 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 (σy + σu) (flow stress), psi

[

]^{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

Local Failure Mechanism:

[

]a,c,e

Global failure mechanism:

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 assumed conservatively as SAW weld (SAW gives highest "Z" factor correction). The "Z" factor correction for SAW was applied (Reference 7-5) as follows:

$$Z = 1.30\,[1.0 + 0.010\,(OD-4)]$$

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

The Z-factors were calculated for the critical locations, using the dimensions given in Table 3-1. The Z factor was 1.69 for locations 1 and 2. The Z factor was 1.72 for location 8. The Z factor was 1.67 for location 11. 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, 7-3 and 7-4. 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 Johnson, W. and Mellor, P. B., Engineering Plasticity, Van Nostrand Reinhold Company, New York, (1973), pp. 83-86.

7-3 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.

7-4 Irwin, G. R., "Plastic Zone Near a Crack and Fracture Toughness," Proc. 7th Sagamore Conference, P. IV-63 (1960).

7-5 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.

a,c,e

Note: T_{app} is not applicable since $J_{app} < J_{Ic}$

Table 7-2 Stability Results for Point Beach Units 1 and 2 Based on Limit Load

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)
1	20.28	5.41
2	38.13	7.61
8	40.92	7.99
11	30.75	6.88

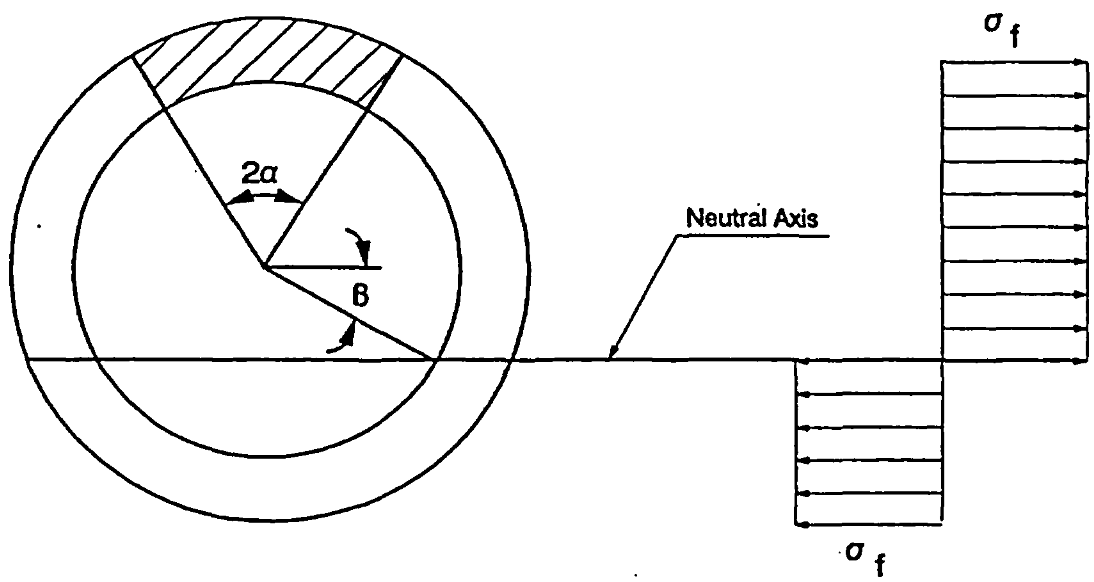


Figure 7-1 [ ]a,c,e Stress Distribution

a,c,e

OD = 34.21 in σ_y = 20.80 ksi F_a = 1715 kips

t = 2.50 in σ_u = 52.10 ksi M_b = 20688 in-kips

A376 TP316 Material with SAW Weld

Figure 7-2 Critical Flaw Size Prediction - Hot Leg at Location 1

a,c,e

OD = 34.21 in $\sigma_y = 23.46$ ksi $F_a = 1715$ kips

t = 2.50 in $\sigma_u = 71.79$ ksi $M_b = 6080$ in-kips

A351 CF8M Material with SAW Weld

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

a,c,e

$OD = 36.56$ in $\sigma_y = 24.31$ ksi $F_a = 1803$ kips

$t = 2.675$ in $\sigma_u = 71.79$ ksi $M_b = 9070$ in-kips

A351 CF8M Material with SAW Weld

Figure 7-4 Critical Flaw Size Prediction – Cross-Over Leg at Location 8

a,c,e

OD = 32.46 in σ_y = 24.31 ksi F_a = 1426 kips

t = 2.375 in σ_u = 71.79 ksi M_b = 14069 in-kips

A351 CF8M Material with SAW Weld

Figure 7-5 Critical Flaw Size Prediction – Cold Leg at Location 11

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

Location [ ]^{a,c,e} of Figure 3-2). This region was selected because crack growth calculated here would be typical of that in the entire primary loop. Crack growths calculated at other locations would be expected to show less than 10% variation.

A [ ]^{a,c,e}. The normal, upset, and test conditions were considered. A summary of Point Beach applied transients (Reference 8-1) for 60 year is provided in Table 8-1. Circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in two different locations, as shown in Figure 8-1. Specifically, these were:

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

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

Fatigue crack growth rate laws were used [

]^{a,c,e} The law for stainless steel was derived from Reference 8-2, a compilation of data for austenitic stainless steel in a PWR water environment was presented in Reference 8-3, and it was found that the effect of the environment on the crack growth rate was very small. From this information it was estimated that the environmental factor should be conservatively set at [ ]^{a,c,e} in the crack growth rate equation from Reference 8-2.

For stainless steel, the fatigue crack growth formula is:

[

]^{a,c,e}

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 for 60 year is very small, [ ]^{a,c,e}

8.1 REFERENCES

8-1 Westinghouse Report, "Power Uprate Project, Point Beach Nuclear Plant Units 1 and 2", April 2002, (Table 5.6-2).

8-2 James, L. A. and Jones, D. P., "Fatigue Crack Growth Correlations for Austenitic Stainless Steel in Air, Predictive Capabilities in Environmentally Assisted Cracking," ASME publication PVP-99, December 1985.

8-3 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.

Table 8-1 Summary of Reactor Vessel Transients		
Number	**Transient Identification**	**Number of Cycles**
	Normal conditions and Upset Conditions	
1	Heat up/Cool Down at 100° F/hr (pressurizer cool down 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	Feed water Cycling at Hot Standby	25000
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
10	Inadvertent Auxiliary Spray	10
	Test Conditions	
11	Primary Hydro @3106 psig	5
12	Primary Pressure Test @2485 psig	100
13	Secondary Hydrotest @1356 psig	10
14	Secondary Pressure Test @1085 psig	50
15	Primary-to-Secondary Leak Tests	30
16	Secondary-to-Primary Leak Tests	130

Table 8-2 Fatigue Crack Growth at [ ]a,c,e (60 years)		
FINAL FLAW (in.)		
[		
		]a,c,e

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 Carbon and Low Alloy Ferritic Steels

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 - Using faulted loads obtained by the absolute sum method, a margin of 2 or more 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 (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin on loads >1 using the absolute summation of faulted load combinations is satisfied.

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Point Beach Units 1 and 2			
Location	**Leakage Flaw Size**	**Critical Flaw Size**	**Margin**
1	5.41 in.	20.28[1] in.	3.7[1]
2	7.61 in.	38.13[1] in.	5.0[1]
2	7.61 in.	15.22[2] in.	>2.0[2]
8	7.99 in.	40.92[1] in.	5.1[1]
8	7.99 in.	15.98[2] in.	>2.0[2]
11	6.88 in.	30.75[1] in.	4.5[1]
11	6.88 in.	13.75[2] in.	>2.0[2]

[

][a,c,e]

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[1]based on limit load

[2]based on J integral evaluation

10.0 CONCLUSIONS

This report justifies the elimination of RCS primary loop pipe breaks from the structural design basis for a 60 year plant life and for the power uprate program of Point Beach 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 Point Beach 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, the Leak-Before-Break conditions are satisfied for the Point Beach Units 1 and 2 primary loop piping. All the recommended margins are satisfied. It is therefore, concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis of the Point Beach Units 1 and 2 Nuclear Power Plants for a 60 year plant life as part of the license renewal program and for the power uprate program.

APPENDIX A

LIMIT MOMENT

[

]a,c,e

a,c,e

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