

WCAP-14238

TECHNICAL JUSTIFICATION FOR ELIMINATING
LARGE PRIMARY LOOP PIPE RUPTURE AS THE
STRUCTURAL DESIGN BASIS FOR THE
TURKEY POINT UNITS 3 AND 4
NUCLEAR POWER PLANTS

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EXECUTIVE SUMMARY

The original structural design basis of the reactor coolant system for the Florida Power and Light Company (FPL) Turkey Point Units 3 and 4 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 Turkey Point 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. Generic analyses by Westinghouse for the application of LBB for specific plants was documented in response to Unresolved Safety Issue A-2 and approved for Turkey Point in NRC letter dated November 28, 1988 (Reference 1-10). By letter dated November 28, 1988, the NRC stated that:

"...an acceptable technical basis had been provided so that the asymmetric loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for certain plants, including Turkey Point Units 3 and 4, provided two conditions are satisfied. However, of the two conditions, only the one relating to the leakage detection system applies to Turkey Point Units 3 and 4. Specifically, the condition requires that leakage detection systems be sufficient to provide adequate margin to detect the postulated circumferential throughwall flaw utilizing the guidance of Regulatory Guide 1.45.....

We have reviewed your November 1, 1988 submittal and concur that the leakage detection systems at Turkey Point Units 3 and 4 satisfy the requirements in Generic Letter 84-04. This closes MPA [Multi-Plant Action] D-10 for Turkey Point Units 3 and 4. Turkey Point Units 3 and 4 primary loop piping also complies with the revised General Design Criteria 4 (GDC-4) of Appendix A to 10 CFR Part 50, and the dynamic effects of postulated primary loop pipe ruptures may be eliminated from the design basis."

This report demonstrates compliance with LBB technology for the Turkey Point reactor coolant system piping based on a plant specific analysis. 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 Turkey Point Units 3 and 4 Nuclear Power Plants.

SECTION 1.0 INTRODUCTION

1.1 Purpose

This report applies to the Turkey Point Units 3 and 4 Reactor Coolant System (RCS) primary loop piping. It is intended to demonstrate that for the specific parameters of the Turkey Point Units 3 and 4 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 1-2). 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 1-3).

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 1-4 and 1-5). 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 1-2) 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 1-6).

1.3 Scope and Objective

The general purpose of this investigation is to demonstrate leak-before-break for the primary loops in Turkey Point Units 3 and 4. While the Standard Review Plan which is documented in Reference 1-7 was not in existence at the time Turkey Point Units 3 and 4 were licensed, the recommendations and criteria proposed in Reference 1-7 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 at least 2 between the leakage size flaw and the critical size flaw.

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- 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 Turkey Point Units 3 and 4 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 1-8). 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-9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March, 1978.
- 1-3 Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, November 10, 1981.
- 1-4 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated April 25, 1983.
- 1-5 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

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- 1-6 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.
- 1-7 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.
- 1-8 Nuclear Regulatory Commission, Standard Review Plan Section 3.9.1, "Special Topics for Mechanical Component," NUREG-0800, Revision 2, July 1981.
- 1-9 WCAP-7211, Revision 3, "Energy Systems Business Unit Policy and Procedures for Management, Classification, and Release of Information," March, 1994.
- 1-10 Nuclear Regulatory Commission Docket #'s 50-250 and 50-251 Letter from G. E. Edison, Sr. Project Manager, NRC, to W. F. Conway, Sr. Vice President, Florida Power and Light, Subject: "Turkey Point Units 3 and 4, Generic Letter 84-04, Asymmetric LOCA Loads," dated 11/28/88.

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 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 Turkey Point Units 3 and 4. 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.

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SECTION 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 34.00 in. and 2.395 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. 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 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 toughness critical locations. Two most critical locations are 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).

The 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:

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

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:

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

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:

$$F = F_{DW} + F_{TH} + F_P \quad (3-3)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} + (M_Y)_P \quad (3-4)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} + (M_Z)_P \quad (3-5)$$

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. 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 $\sqrt{2}$ to 1.0. The absolute summation of loads are shown in the following equations:

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

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

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

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

Dimensions, Normal Loads and Normal Stresses for Turkey Point Units 3 and 4

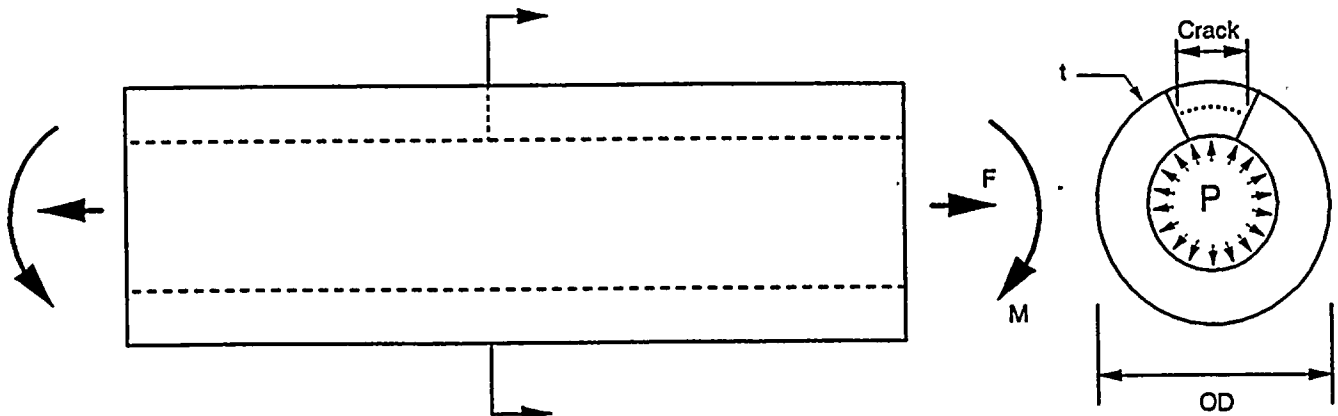
Location ^a	Outside Diameter (in.)	Minimum Thickness (in.)	Axial Load ^b (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	34.00	2.395	1341	23453	18.99
2	34.00	2.395	1341	11266	12.05
3	37.75	3.270	1578	17788	10.77
4	37.62	3.205	1715	3777	6.32
5	36.25	2.520	1657	4329	8.26
6	36.25	2.520	1653	4422	8.29
7	36.25	2.520	1719	781	6.81
8	36.25	2.520	1716	3712	8.19
9	37.63	3.208	1819	8794	8.44
10	32.25	2.270	1363	6861	10.95
11	32.25	2.270	1363	6878	10.96
12	33.56	2.930	1363	7759	8.75

^a See Figure 3-2^b Includes pressure

Table 3-2
Faulted Loads and Stresses for Turkey Point Units 3 and 4

Location^{a,b}	Axial Load^c (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	1877	24027	21.57
2	1876	11912	14.67
3	2007	20886	13.09
4	1745	5043	6.87
5	1824	5235	9.32
6	1821	4834	9.11
7	1743	1169	7.08
8	1745	4385	8.62
9	1832	10705	9.17
10	1434	9600	13.12
11	1435	8097	12.12
12	1430	9331	9.78

-
- ^a See Figure 3-2
^b See Table 3-1 for dimensions
^c Includes pressure



$$OD^a = 34.00 \text{ in}$$

$$t^a = 2.395 \text{ in}$$

Normal Loads^a

force^c: 1341 kips
bending moment: 23453 in-kips

Faulted Loads^b

force^c: 1877 kips
bending moment: 24027 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

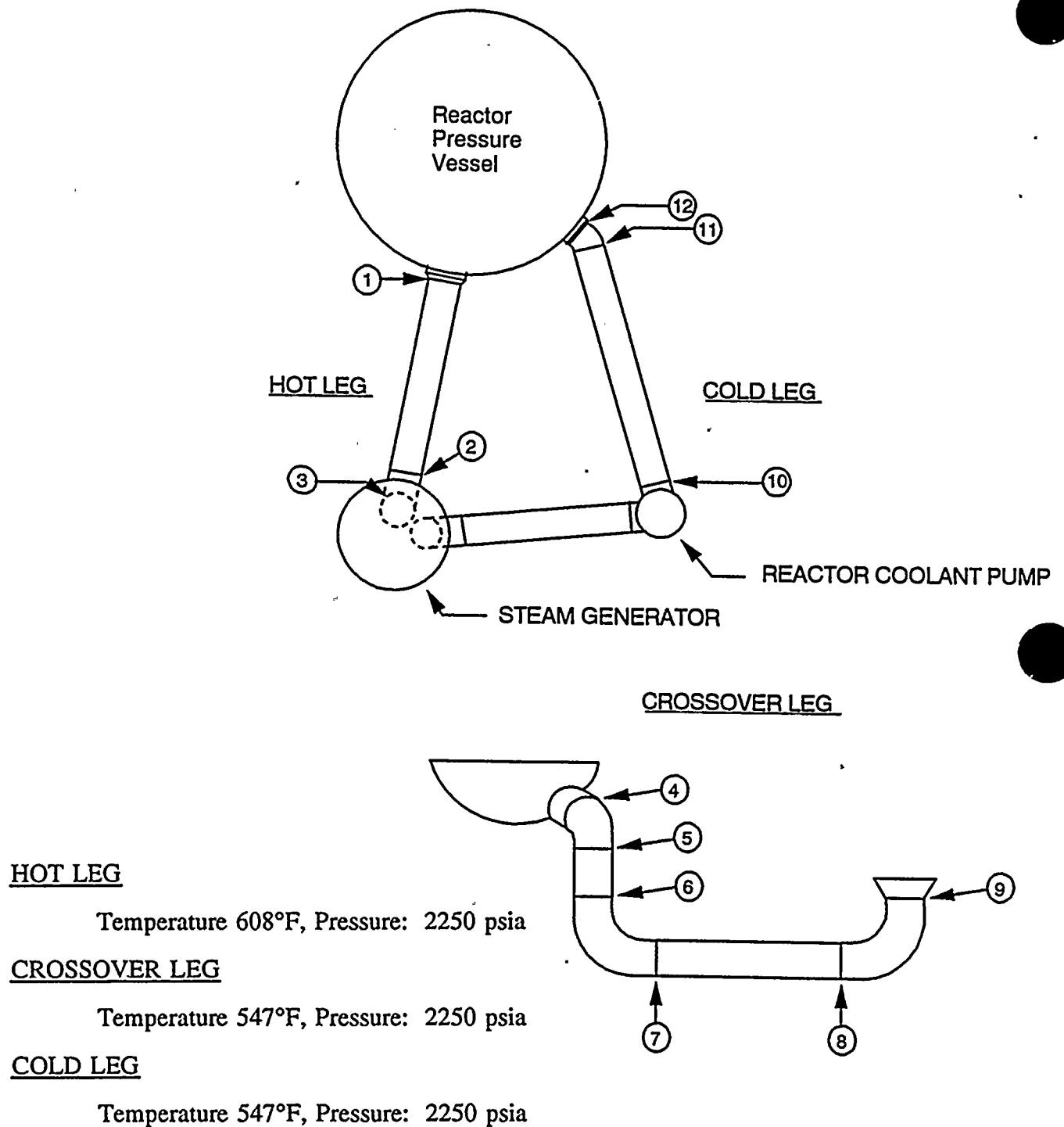


Figure 3-2

Schematic Diagram of Turkey Point Units 3 and 4 Primary Loop Showing Weld Locations

SECTION 4.0 MATERIAL CHARACTERIZATION

4.1 Primary Loop Pipe and Fittings Materials

The primary loop pipe materials are A376 TP316, and the elbow fittings are A351 CF8M.

4.2 Tensile Properties

The Pipe Certified Materials Test Reports (CMTRs) for Turkey Point Units 3 and 4 were used to establish the tensile properties for the leak-before-break analyses. 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 Table 4-1 for piping and in Table 4-2 for elbow fittings.

For the A376 TP316 material, the representative properties at 608°F were established from the tensile properties at 650°F given in Table 4-1 by utilizing Section III of the 1989 ASME Boiler and Pressure Vessel Code (Reference 4-6). Code tensile properties at 608°F were obtained by interpolating between the 600°F and 650°F tensile properties. Ratios of the code tensile properties at 608°F to the corresponding tensile properties at 650°F were then applied to the 650°F tensile properties given in Table 4-1 to obtain the plant specific properties for A376 TP316 at 608°F.

The Elbow Fittings Certified Materials Test Reports (CMTRs) for Turkey Point Units 3 and 4 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 Turkey Point Units 3 and 4 in Table 4-2.

For the A351 CF8M material, the representative properties at 608°F and 547°F were established from the tensile properties at room temperature given in Table 4-2 by utilizing Section III of the 1989 ASME Boiler and Pressure Vessel Code. Code tensile properties at 608°F and 547°F were established by interpolating between the 500°F, 600°F and the 650°F tensile properties. Ratios of the code tensile properties at 608°F and 547°F to the corresponding properties at room temperature were then applied to the room temperature properties given in Table 4-2 to obtain the plant specific representative properties for A351 CF8M at 608°F and 547°F.

The average and lower bound yield strengths and ultimate strengths are given in Table 4-3. The ASME Code moduli of elasticity are also given, and Poisson's ratio was taken as 0.3.

For leak-before-break fracture evaluations of the toughness critical locations the true stress-true strain curves for A351 CF8M at 608°F and 547°F must be available. These curves were obtained using the Nuclear Systems Materials Handbook (Reference 4-1). The lower bound true stress-true strain curves are given in Figures 4-1 and 4-2.

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 4-3. J_{Ic} is observed to be over 2500 in-lbs/in². 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 worst case fracture toughness values

for Turkey Point Units 3 and 4 loops at critical locations, as taken from Appendix B, are given in Table 4-4.

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/in²). 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} 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} are also conservatively assumed to have the properties of []^{a,c}.

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

Forged stainless steel piping such as A376 TP316 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.

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

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- 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 1989, Section III.

Table 4-1

Measured Tensile Properties for Turkey Point Units 3 and 4 Primary Loop Piping

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp (psi)	Yield Temp 650F (psi)	Ultimate Temp 650F (psi)	Material Type
Hot Leg	F0070 #2646	38200	79000	21000	60800	A-376TP316
Hot Leg	F0070 #2646	34100	78200	21000	60800	A-376TP316
Hot Leg	F0214 #2850	42000	83300	22400	62300	A-376TP316
Hot Leg	F0214 #2850	43000	82800	22400	62300	A-376TP316
Hot Leg	F0188 #2844	39200	83400	26100	67000	A-376TP316
Hot Leg	F0188 #2844	38900	84000	26100	67000	A-376TP316
Hot Leg	D8549 #1170	30100	75000	22100	63400	A-376TP316
Hot Leg	D8549 #1170	36900	79000	22100	63400	A-376TP316
Hot Leg	F0215 #2892X	43000	83000	21700	66800	A-376TP316
Hot Leg	F0215 #2892X	42000	84600	21700	66800	A-376TP316
Hot Leg	F0225 #2895X	42000	85900	22100	68200	A-376TP316
Hot Leg	F0225 #2895X	46000	88800	22100	68200	A-376TP316
Hot Leg	E-1493 #3356	42000	83100	25500	69200	A-376TP316
Hot Leg	E-1493 #3356	42600	84500	25500	69200	A-376TP316
Hot Leg	E-1485 #3352X	42300	85400	27500	72200	A-376TP316
Hot Leg	E-1485 #3352X	45400	87400	27500	72200	A-376TP316
Hot Leg	E-1482 #3355	40900	82700	24500	66200	A-376TP316
Hot Leg	E-1482 #3355	48600	88800	24500	66200	A-376TP316
Hot Leg	E-1483 #3357	42100	85000	23700	52100	A-376TP316
Hot Leg	E-1483 #3357	43000	85200	23700	52100	A-376TP316
Hot Leg	E-1490 #3348Y	46000	85700	23700	68600	A-376TP316
Hot Leg	E-1490 #3348Y	40900	83300	23700	68600	A-376TP316
Hot Leg	E-1490 #3347X	43100	86000	23700	68600	A-376TP316
Hot Leg	E-1490 #3347X	43000	82700	23700	68600	A-376TP316
Cold Leg	D8777 #2874X	35300	79200	24100	65600	A-376TP316
Cold Leg	D8777 #2874X	34900	78200	24100	65600	A-376TP316
Cold Leg	D8913 #2877	35100	79200	23100	64200	A-376TP316

Table 4-1 (cont)

Measured Tensile Properties for Turkey Point Units 3 and 4 Primary Loop Piping

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp (psi)	Yield Temp 650F (psi)	Ultimate Temp 650F (psi)	Material Type
Cold Leg	D8913 #2877	38100	79600	23100	64200	A-376TP316
Cold Leg	D8915 #2876	34100	78400	22500	63600	A-376TP316
Cold Leg	D8915 #2876	32800	78600	22500	63600	A-376TP316
Cold Leg	F0230 #2993	48000	88100	21700	70200	A-376TP316
Cold Leg	F0230 #2993	47000	90900	21700	70200	A-376TP316
Cold Leg	F0162 #2858	43700	83100	27600	69400	A-376TP316
Cold Leg	F0162 #2858	46200	84900	27600	69400	A-376TP316
Cold Leg	F0371 #3130	47000	88700	25600	72300	A-376TP316
Cold Leg	F0371 #3130	44000	88800	25600	72300	A-376TP316
Cold Leg	F0228 #2949	42200	86100	24700	69800	A-376TP316
Cold Leg	F0228 #2949	45000	89900	24700	69800	A-376TP316
Cold Leg	F0373 #3166	40100	86900	25600	71900	A-376TP316
Cold Leg	F0373 #3166	45500	91900	25600	71900	A-376TP316
Cold Leg	F0244 #2997	42900	85400	25400	69900	A-376TP316
Cold Leg	F0244 #2997	47500	88100	25400	69900	A-376TP316
Cold Leg	F0229 #2950	41000	85700	24500	68200	A-376TP316
Cold Leg	F0229 #2950	44400	86400	24500	68200	A-376TP316
Cold Leg	F0371 #3128	46800	78700	25600	72300	A-376TP316
Cold Leg	F0371 #3128	36900	84700	25600	72300	A-376TP316
Xover Leg	F0189 #2868Z	37700	80600	25200	70000	A-376TP316
Xover Leg	F0189 #2868Z	44100	91000	25200	70000	A-376TP316
Xover Leg	F0189 #2870X	41300	82200	25200	70000	A-376TP316
Xover Leg	F0189 #2870X	42900	84200	25200	70000	A-376TP316
Xover Leg	F0189 #2870Y	41300	82200	25200	70000	A-376TP316
Xover Leg	F0189 #2870Y	42900	84200	25200	70000	A-376TP316
Xover Leg	D8775 #2880	34100	75800	24000	60400	A-376TP316
Xover Leg	D8775 #2880	41100	77800	24000	60400	A-376TP316

Table 4-1 (cont)

Measured Tensile Properties for Turkey Point Units 3 and 4 Primary Loop Piping

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp (psi)	Yield Temp 650F (psi)	Ultimate Temp 650F (psi)	Material Type
Xover Leg	D8777 #2879	38100	78200	20200	60000	A-376TP316
Xover Leg	D8777 #2879	38500	78200	20200	60000	A-376TP316
Xover Leg	D8785 #2881	32900	78200	20400	57200	A-376TP316
Xover Leg	D8785 #2881	34100	79000	20400	57200	A-376TP316
Xover Leg	E-1485 #3361Y	38300	83900	23600	70700	A-376TP316
Xover Leg	E-1485 #3361Y	45900	91400	23600	70700	A-376TP316
Xover Leg	F-0215 #2892	40000	84800	21700	82300	A-376TP316
Xover Leg	F-0215 #2892	41900	87700	21700	82300	A-376TP316
Xover Leg	F-0221 #2866	41000	83900	21600	65200	A-376TP316
Xover Leg	F-0221 #2866	41500	83000	21600	65200	A-376TP316
Xover Leg	E-1484 #3360X	39600	83300	23600	63600	A-376TP316
Xover Leg	E-1484 #3360X	44500	88800	23600	63600	A-376TP316
Xover Leg	F-0212 #2887	43500	85100	23400	67400	A-376TP316
Xover Leg	F-0212 #2887	43200	88000	23400	67400	A-376TP316
Xover Leg	E-1484 #3360Y	39600	83300	23600	63600	A-376TP316
Xover Leg	E-1484 #3360Y	44500	88800	23600	63600	A-376TP316

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Table 4-2

Measured Room Temperature Tensile Properties for Turkey Point Units 3 and 4
Primary Loop Elbow Fittings

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp (psi)	Material Type
Hot Leg	11897-2	39000	76000	A351CF8M
Hot Leg	12121-2	43500	85000	A351CF8M
Hot Leg	12012-2	39000	77000	A351CF8M
Hot Leg	08368-1	43500	87500	A351CF8M
Hot Leg	08247-1	46500	88000	A351CF8M
Hot Leg	08586-1	40000	84000	A351CF8M
Cold Leg	16037-1	37500	75000	A351CF8M
Cold Leg	10865-2	42000	81500	A351CF8M
Cold Leg	12393-4	42000	83000	A351CF8M
Cold Leg	05872-2	49500	89000	A351CF8M
Cold Leg	05769-5	52500	93500	A351CF8M
Cold Leg	05715-2	51000	90000	A351CF8M
Xover Leg	10048-1	45000	87000	A351CF8M
Xover Leg	10763-1	44250	81000	A351CF8M
Xover Leg	10008-1	46500	88000	A351CF8M
Xover Leg	10804-1	43500	81000	A351CF8M
Xover Leg	10091-1	42000	86500	A351CF8M
Xover Leg	10128-1	46500	89000	A351CF8M
Xover Leg	10320-1	48000	91500	A351CF8M
Xover Leg	10563-1	43500	88000	A351CF8M
Xover Leg	10283-1	48000	91500	A351CF8M
Xover Leg	10243-1	46500	90000	A351CF8M
Xover Leg	10400-1	48000	91500	A351CF8M
Xover Leg	10644-1	49500	91000	A351CF8M
Xover Leg	09476-3	45000	88000	A351CF8M
Xover Leg	09027-1	48000	90500	A351CF8M
Xover Leg	06239-2	48000	88000	A351CF8M

Table 4-2 (cont)

**Measured Room Temperature Tensile Properties for Turkey Point Units 3 and 4
Primary Loop Elbow Fittings**

Component	Heat Num.	Yield Room Temp (psi)	Ultimate Room Temp (psi)	Material Type
Xover Leg	12511-2	46500	89000	A351CF8M
Xover Leg	12584-2	40500	83250	A351CF8M
Xover Leg	12357-7	42000	82500	A351CF8M
Xover Leg	12432-5	45000	86000	A351CF8M
Xover Leg	11518-1	42000	83750	A351CF8M
Xover Leg	12087-1	39000	79000	A351CF8M
Xover Leg	13498-3	45750	87250	A351CF8M
Xover Leg	14206-1	45000	86250	A351CF8M
Xover Leg	13378-3	42000	84250	A351CF8M
Xover Leg	15258-1	42000	83750	A351CF8M
Xover Leg	12660-2	42000	85000	A351CF8M
Xover Leg	12857-1	42000	83500	A351CF8M
Xover Leg	14048-1	40500	78500	A351CF8M
Xover Leg	14335-1	51000	88500	A351CF8M
Xover Leg	13133-2	45000	87000	A351CF8M

Table 4-3

**Mechanical Properties for Turkey Point Units 3 and 4 Materials
at Operating Temperatures**

Material	Temperature (°F)	Average Yield Strength (psi)	Lower Bound	
			Yield Stress (psi)	Ultimate Strength (psi)

a,c,e

Modulus of Elasticity

E = 25.26×10^6 psi. at 608°F

E = 25.56×10^6 psi. at 547°F

Poisson's ratio: 0.3

Table 4-4

**Fracture Toughness Properties for Turkey Point Units 3 and 4 Primary Loops for
Leak-Before-Break Evaluation at Critical Locations**

Location ^(a)	Heat No.	KCU (daJ/cm ²)	J _{IC} (in-lb/in ²)	T _{mat} (non-dim)	J _{max} (in-lb/in ²)	Comments	a,c,e

^(a) The locations are shown in Figure 3-2



Figure 4-1 **Representative Lower Bound True Stress - True Strain
Curve for A351 CF8M at 608°F**



Figure 4-2 **Representative Lower Bound True Stress - True Strain
Curve for A351 CF8M at 547°F**



Figure 4-3 Pre-Service J vs. Δa for SA351 CF8M Cast Stainless Steel at 600°F



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

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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 locations 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 Turkey Point Units 3 and 4. Table 3-2 as well as Figure 3-2 are used for this evaluation.

Load Critical Locations

The highest stressed location for the A376 TP316 straight pipes is Location 1 (See Figure 3-2) at the reactor vessel outlet nozzle to pipe weld. Furthermore, since it is on a straight pipe, it is a high toughness location.

Toughness Critical Locations

Low toughness locations are at the ends of every elbow. All the elbows except those indicated in page B-1 of Appendix B, exceed the toughness of [$J_{AC,0}$]. 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 higher stress than location 3. In the case of cross over leg and cold leg the lowest toughness is on cold leg at locations 11 and 12. Location 11 governs since it has higher stress and lowest toughness among all the elbow locations in cross over leg and cold leg. It is thus concluded that the enveloping locations are 2 and 11. The allowable toughness for the critical locations are shown in Table 4-4.

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;

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- (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

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, P_c , for the primary loop enthalpy condition and an assumed flow. Once P_c 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, P_o is the operating pressure. Now using the assumed flow rate, G , the frictional pressure drop can be calculated using

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

where the friction factor f is determined using the [\quad]^{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 [\quad]^{a,c,e}

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

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

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 6-2 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 leakage flaws.

The Turkey Point Units 3 and 4 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.

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

Location	Flaw Size (in)	a,c,e



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



Figure 6-2 [

] ^{a,c,e} Pressure Ratio as a Function of L/D

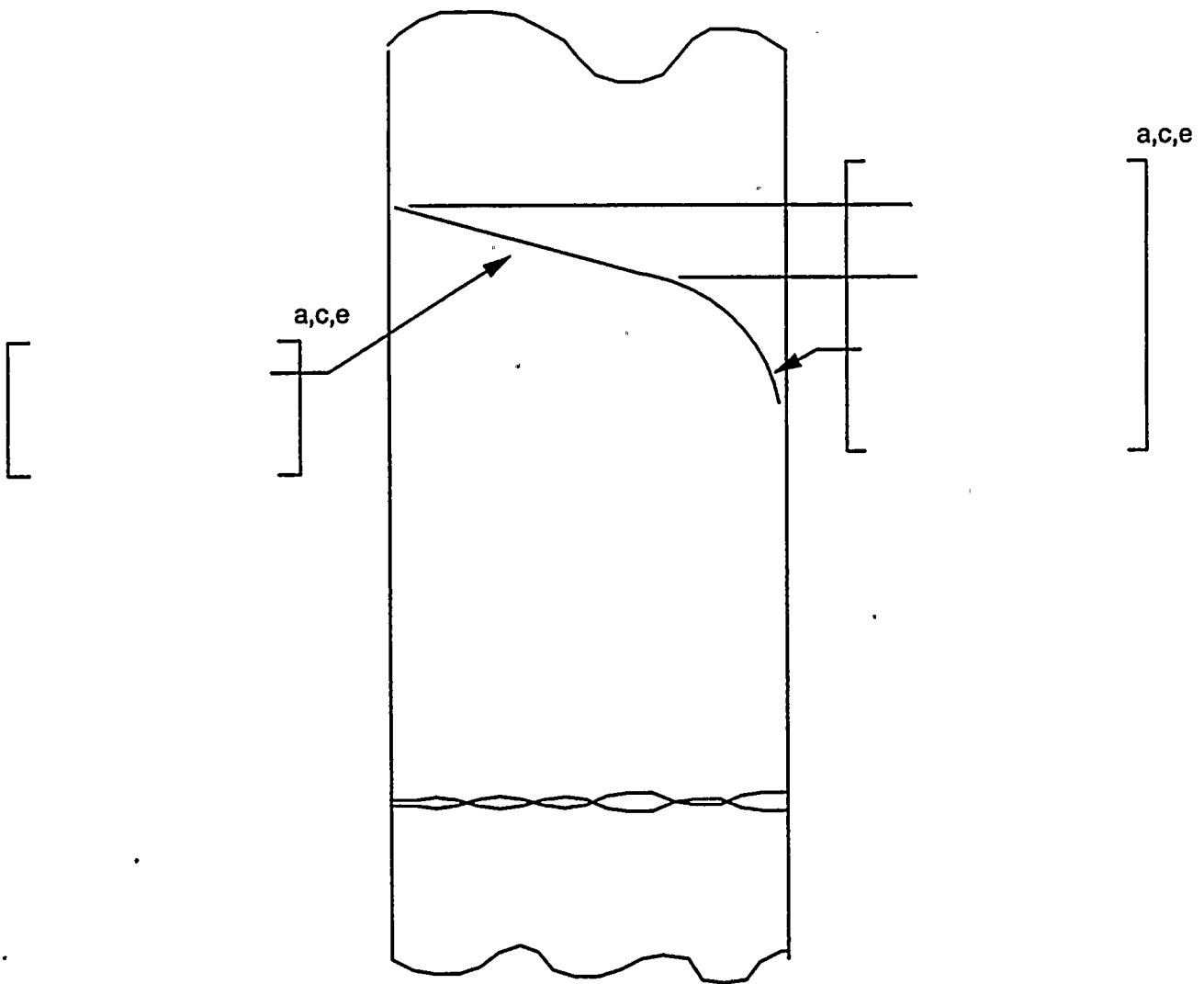


Figure 6-3 **Idealized Pressure Drop Profile Through a Postulated Crack**

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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:

$$[\quad]_{a,c,e}$$

where:

[

]_{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 4-3). 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 assumed conservatively as SAW weld (SAW gives highest "Z" factor correction). The "Z" factor correction for SAW was applied (Reference 7-3) 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 1.67 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 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 Turkey Point Units 3 and 4 Based on Elastic-Plastic
J-Integral Evaluations

Location	Flaw Size (in)	Fracture Criteria			Calculated Values	
		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

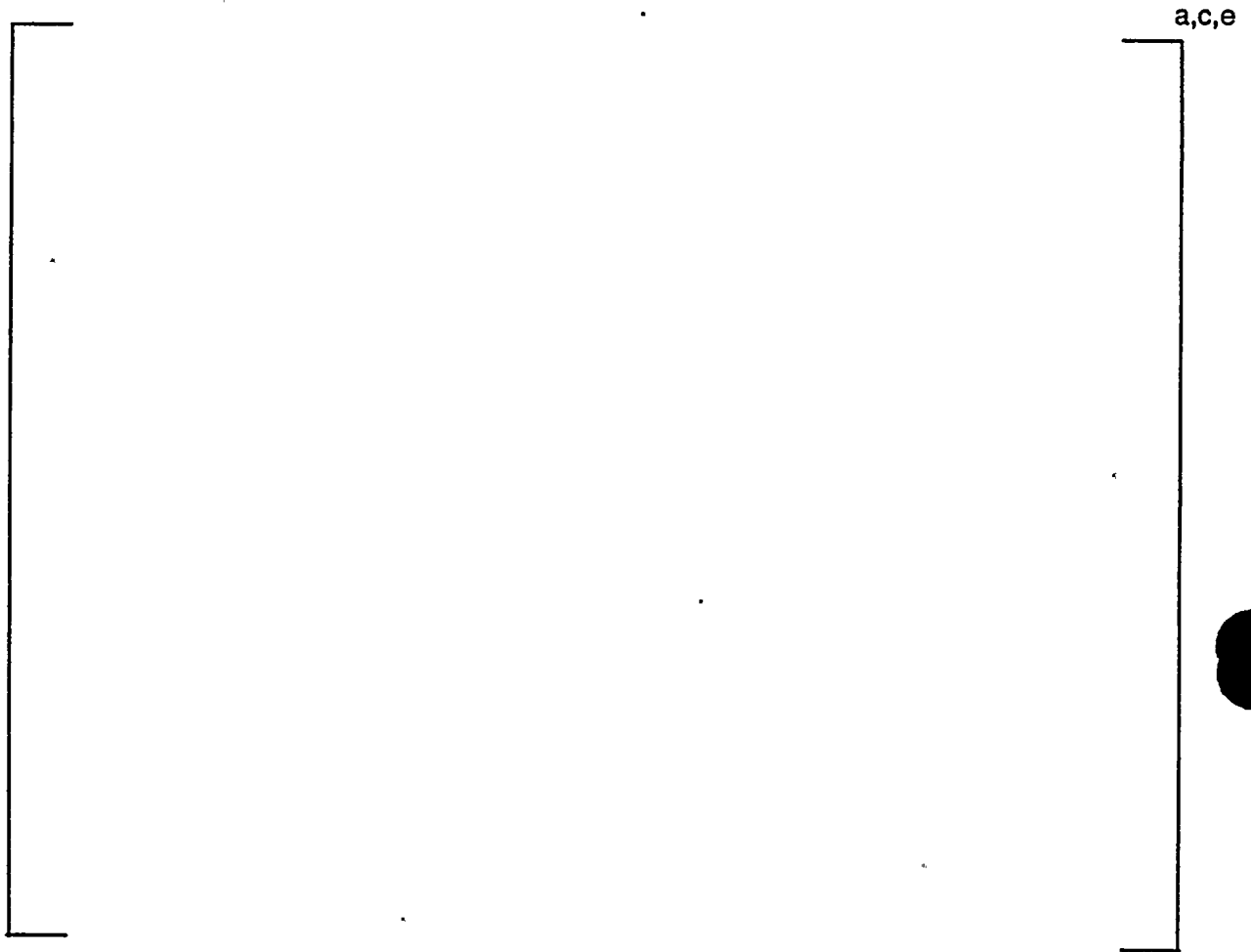
Table 7-2

Stability Results for Turkey Point Units 3 and 4 Based on Limit Load

<u>Location</u>	<u>Flaw Size (in.)</u>	<u>Leakage Flaw Size</u>	a,c,e
[]



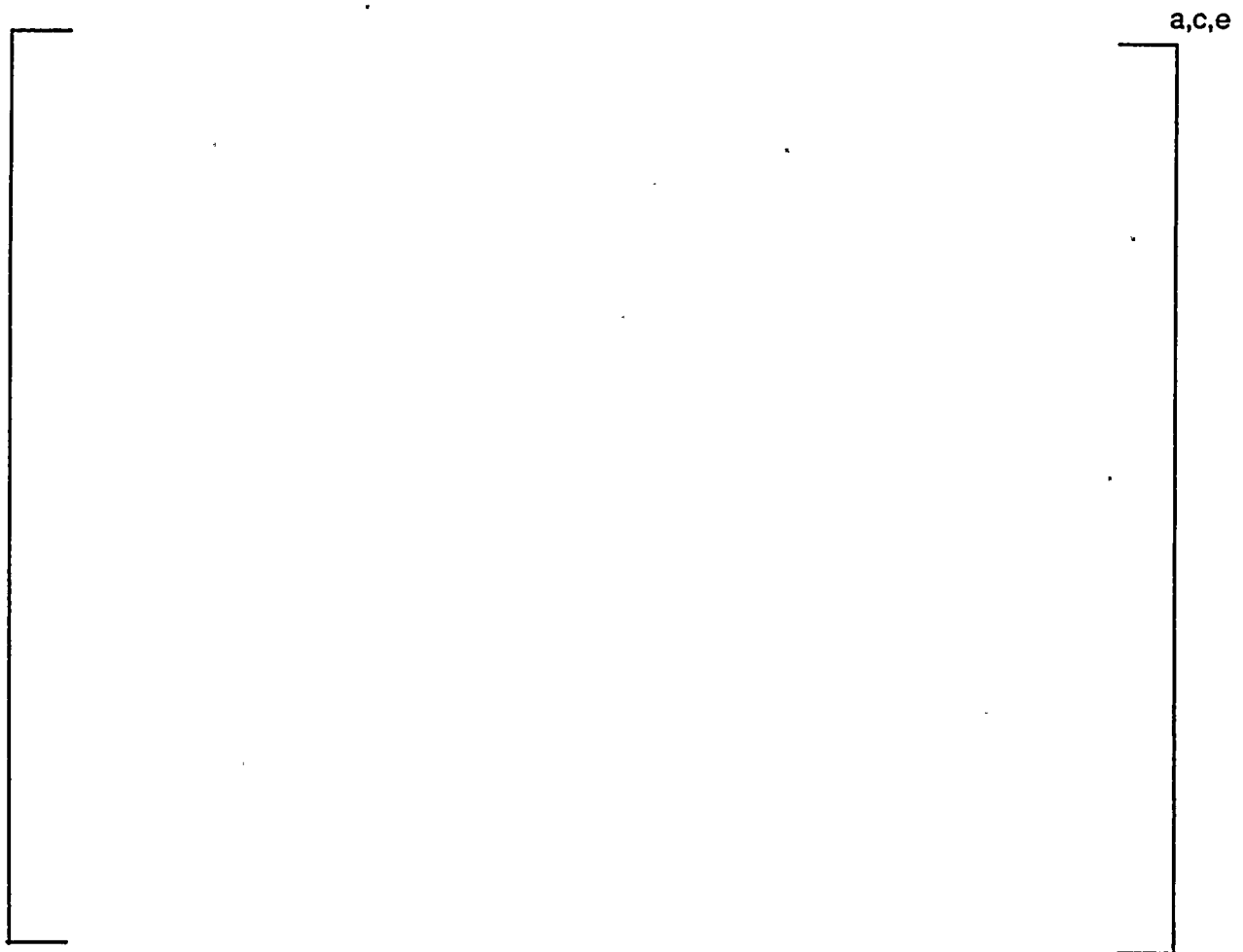
Figure 7-1 []^{a,c,e} Stress Distribution



OD = 34.00 in	$\sigma_y = 20.47$ ksi	$F_a = 1877$ kips
t = 2.395 in	$\sigma_u = 52.10$ ksi	$M_b = 24027$ in-kips

A376 TP316 Material With SAW Weld

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



OD = 34.00 in $\sigma_y = 23.44$ ksi $F_a = 1876$ kips
t = 2.395 in $\sigma_u = 71.79$ ksi $M_b = 11912$ in-kips

A351 CF8M Material With SAW Weld

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



OD = 32.25 in	$\sigma_y = 24.23$ ksi	$F_a = 1435$ kips
t = 2.27 in	$\sigma_u = 71.79$ ksi	$M_b = 8097$ in-kips

A351 CF8M Material With SMAW Weld

Figure 7-4 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
 $[\quad]^{a,c,e}$ (40 years)

FINAL FLAW (in.)			
Initial Flaw (in.)	$[\quad]^{a,c,e}$	$[\quad]^{a,c,e}$	$[\quad]^{a,c,e}$
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



Figure 8-1 Typical Cross-Section of [

]a,c,e



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

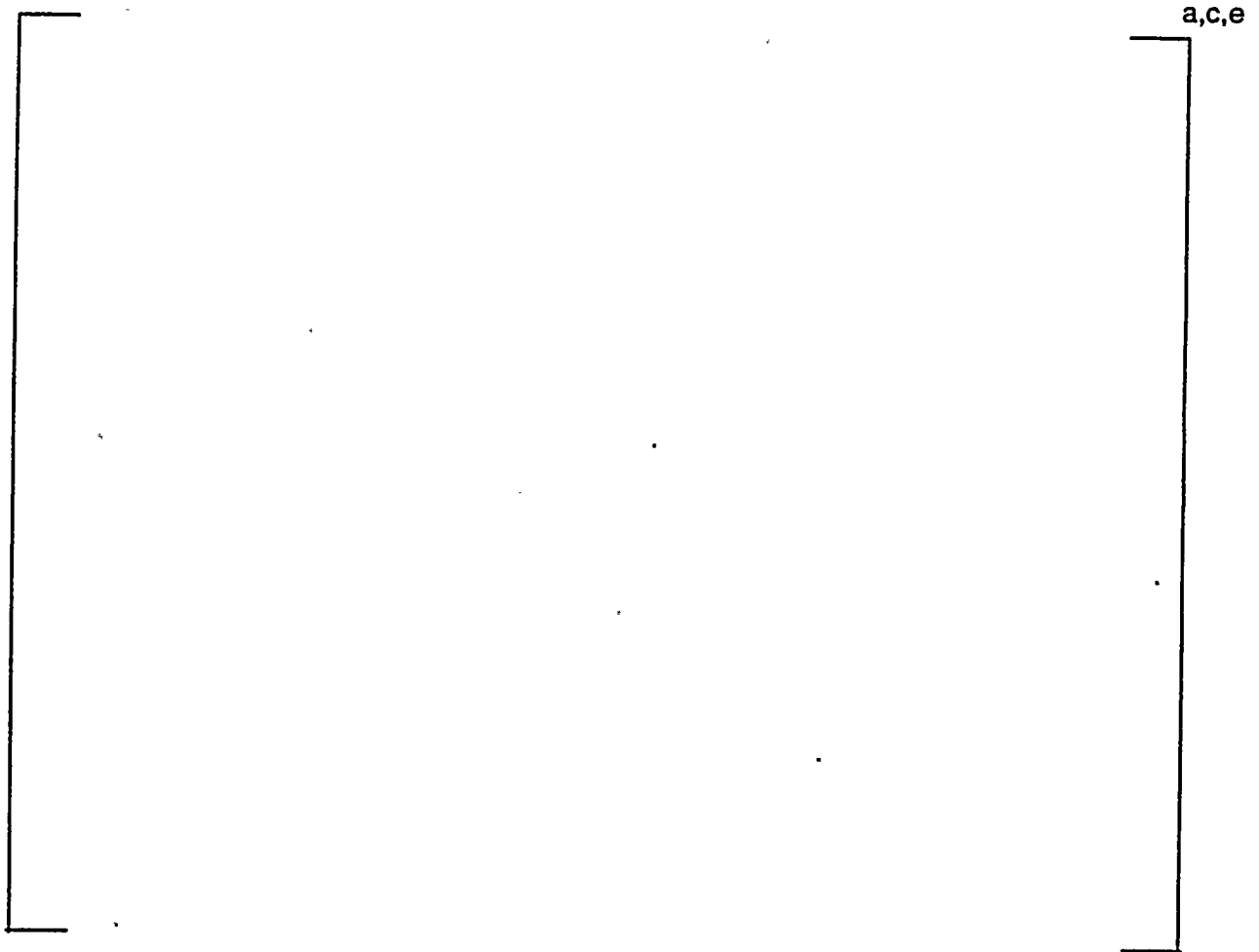


Figure 8-3

Reference Fatigue Crack Growth Law for [
a Water Environment at 600°F

] ^{a, c, e} in

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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 - 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 size flaw is stable).

Table 9-1

Leakage Flaw Sizes, Critical Flaw Sizes and
Margins for Turkey Point Units 3 and 4

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 Turkey Point Units 3 and 4 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 Turkey Point Units 3 and 4 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 Turkey Point Units 3 and 4 Nuclear Power Plants.

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APPENDIX A
LIMIT MOMENT

Jacc



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

APPENDIX B
TOUGHNESS CRITERIA FOR TURKEY POINT UNITS 3 AND 4
CAST PRIMARY LOOP COMPONENTS

All of the individual cast piping components of the Turkey Point Units 3 and 4 primary loops, do not satisfy the original []^{acc} criteria (Reference 4-5). []

] ^{acc}

Table B

Chemistry and Fracture Toughness Properties of the
Material Heats of Turkey Point Units 3 and 4

[Redacted content]

a,c,e
[Redacted content]

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a,c,e

