WESTINGHOUSE CLASS 3

WCAP-11984

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TECHNICAL JUSTIFICATION FOR ELIMINATING LARGE PRIMARY LOOP PIPE RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR WATTS BAR UNITS 1 AND 2

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SHEET — REV 1 UNIT 1 AND 2

ISSUED FOR ECN ——

APRROVED

This approval does not relieve the Contractor from any part of his responsibility for the correctness of design, details and dimensions.

Date: NOV 30 1988

TENNESSEE VALLEY AUTHORITY

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1.0 INTRODUCTION

1.1 Purpose

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

1.2 Background

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

Westinghouse performed additional tests and analyses to justify the elimination of RCS primary loop pipe breaks. As a result of this effort, WCAP-9558, Revision 2, and WCAP-9787, (references 1-2 and 1-3) were submitted to the NRC.

The NRC funded research through Lawrence Livermore National Laboratory (LLNL) to address this same issue using a probabilistic approach. As part of the LLNL research effort, Westinghouse performed extensive evaluations of specific plant loads, material properties, transients, and system geometries to demonstrate that the analysis and testing previously performed by Westinghouse and the research performed by LLNL applied to all Westinghouse plants. 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 1-1) 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 1-4) 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 LBB methodology applicability for PWRs, the NRC appropriately modified (reference 1-6) 10CFR50, General Design Criterion 4, "Requirements for Protection Against Dynamic Effects for Postulated Pipe Rupture."

1.3 Scope and Objective

The general purpose of this investigation is to demonstrate leak-before-break for the primary loops in Watts Bar Units 1 and 2. The criteria and resulting steps of the evaluation procedure (reference 1-5) 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. Demonstrate a margin of 10 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.

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) Provide the material properties including toughness and tensile test data. Justify that the properties used in the evaluation are representative of the plant specific material. Evaluate long term effects such as thermal aging where applicable.

Westinghouse has performed fracture mechanics evaluations, a determination of leak rates from through-wall cracks, a fatigue crack growth evaluation, and an assessment of margins to demonstrate that primary loop rupture may be eliminated as the structural design basis for Watts Bar Units 1 and 2.

This report provides details of the evaluations to demonstrate primary loop integrity for the Watts Bar 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 Standard Review Plan 3.9.1. The fracture mechanics calculations are independently verified (benchmarked).

1.4 References

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

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

1-3 WCAP-9787, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation", Westinghouse Proprietary Class 2, May 1981.

1-4 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.

1-5 Report of the U.S. Nuclear Regulatory Commission Piping Review Committee -- Evaluation of Potential for Pipe Breaks, NUREG 1061, Volume 3, November 1984.

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.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 Stress Corrosion Cracking

The Westinghouse reactor coolant system primary loop and connecting Class 1 lines 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). This operating history totals over 450 reactor-years, including five plants each having over 17 years of operation and 15 other plants each with over 12 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 established in 1975 addressed cracking in boiling water reactors only.) One of the objectives of the second Pipe Crack Study Group (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 and connecting Class 1 lines 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 7.

High cycle fatigue loads in the system would result primarily from pump vibrations. These are minimized by restrictions placed on shaft vibrations during hot functional testing and operation. During operation, an alarm signals the exceedance of the vibration limits. Field measurements have been made on a number of plants during hot functional testing, including plants similar to Watts Bar 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

The general analytical approach is discussed first. A segment of the primary coolant cold leg pipe, shown below to be limiting in terms of stresses, is sketched in figure 3-1. This segment is postulated to contain a circumferential through-wall flaw. The inside diameter and wall thickness of the pipe are 27.71 and 2.21 inches, respectively. The pipe is subjected to a normal operating pressure of 2305 psig. Figure 3-2 identifies the loop circumferential weld locations. The material properties and the loads at these locations resulting from deadweight, thermal expansion, pressure and safe shutdown earthquake (SSE) are indicated in table 3-1. As seen from this table, the junction of the cold leg to the reactor coolant pump (which is location 10 shown on figure 3-2) is the worst location for crack stability analysis based on the highest direct stress. At this location, the axial load (F_x) is 2008 kips (including axial force due to pressure), and the bending moment (M_b) is 23468 inch-kips.

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

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad (3.1)$$

where,

σ = stress

F = axial load

M = bending moment

A = metal cross-sectional area

Z = section modulus

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

$$M = \sqrt{M_Y^2 + M_Z^2} \qquad (3.2)$$

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 analysis are computed by the methods to be explained in sections 3.1 and 3.2.

3.1 Loads for Leak Rate Evaluation

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

$$F = F_{DW} + F_{TH} + F_P \qquad (3.3)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} + (M_Y)_P \qquad (3.4)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} + (M_Z)_P \qquad (3.5)$$

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

DW = deadweight

TH = normal thermal expansion

P = load due to internal pressure

The loads based on this method of combination are provided in table 3-2 for all the locations identified in figure 3-2.

3.2 Load Combination for Crack Stability Analysis Based on Absolute Summation

In accordance with draft Standard Review Plan 3.6.3 (reference 3-1) the following combination 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 results in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SI}| + |F_{SAM}| \tag{3.9}$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_P| + |(M_Y)_{SI}| + |(M_Y)_{SAM}| \tag{3.10}$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_P| + |(M_Z)_{SI}| + |(M_Z)_{SAM}| \tag{3.11}$$

In these equations F_{SI} and F_{SAM} represent the maximum seismic inertia force and the corresponding seismic anchor motion respectively.

Based on this method of combination, the loads at the highest stressed location (i.e. location 10 - pump outlet nozzle junction) are:

F_x = 2008 kips, M_b = 23468 in-kips.

These loads are used in the fracture mechanics evaluations to demonstrate the LBB margins at location 10. The loads at all the locations of interest are summarized in table 3-1. In section 4, location 10 will be shown to be the governing location considering the material properties and the loads of table 3-1.

3.3 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
WATTS BAR UNITS 1 AND 2 PRIMARY LOOP LOADS
(FAULTED CONDITION)

Weld Location	Axial Load (kips)	Bending Moment (in-kips)	Direct Stress $\sigma_x = Fx/A + M_b/Z$ (ksi)
1	1982	29049	25.63
2	1959	17012	18.46
3	2193	28446	22.09
4	1868	13297	13.54
5	1844	9233	11.49
6	1824	8093	10.86
7	1923	11508	12.88
8	1911	14754	14.41
9	1841	19550	18.16
10	2008	23468	25.80
11	1946	14843	20.22
12	1969	17430	21.46

Notes: 1. Effect of internal pressure is included.
2. The critical location is at weld 10.

TABLE 3-2

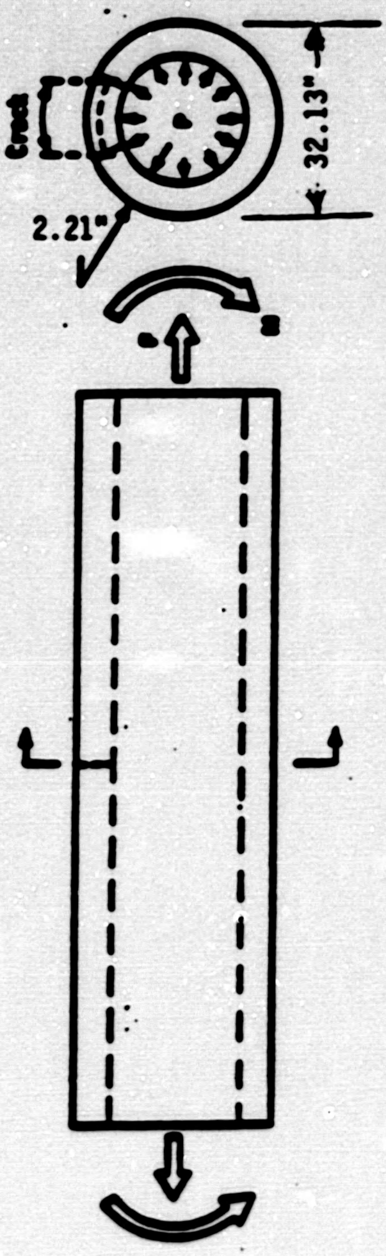
WATTS BAR UNITS 1 AND 2 PRIMARY LOOP LOADS

(NORMAL OPERATING CONDITION)

Weld Location	Axial Load (kips)	Bending Moment (in-kips)	Direct Stress $\sigma_x = Fx/A + M_b/Z$ (ksi)
1	1509	22546	19.76
2	1489	9265	11.89
3	1577	17047	14.24
4	1660	4428	8.46
5	1665	2321	7.47
6	1660	2353	7.46
7	1697	4895	8.83
8	1697	6979	9.84
9	1770	2201	7.81
10	1392	10711	14.06
11	1399	10027	13.63
12	1388	10991	14.23

NOTES: 1. Internal pressure is included

2. The critical location is at weld 10.



Normal Loads

F_x = 1392 kips

M_b = 10711 in-kips

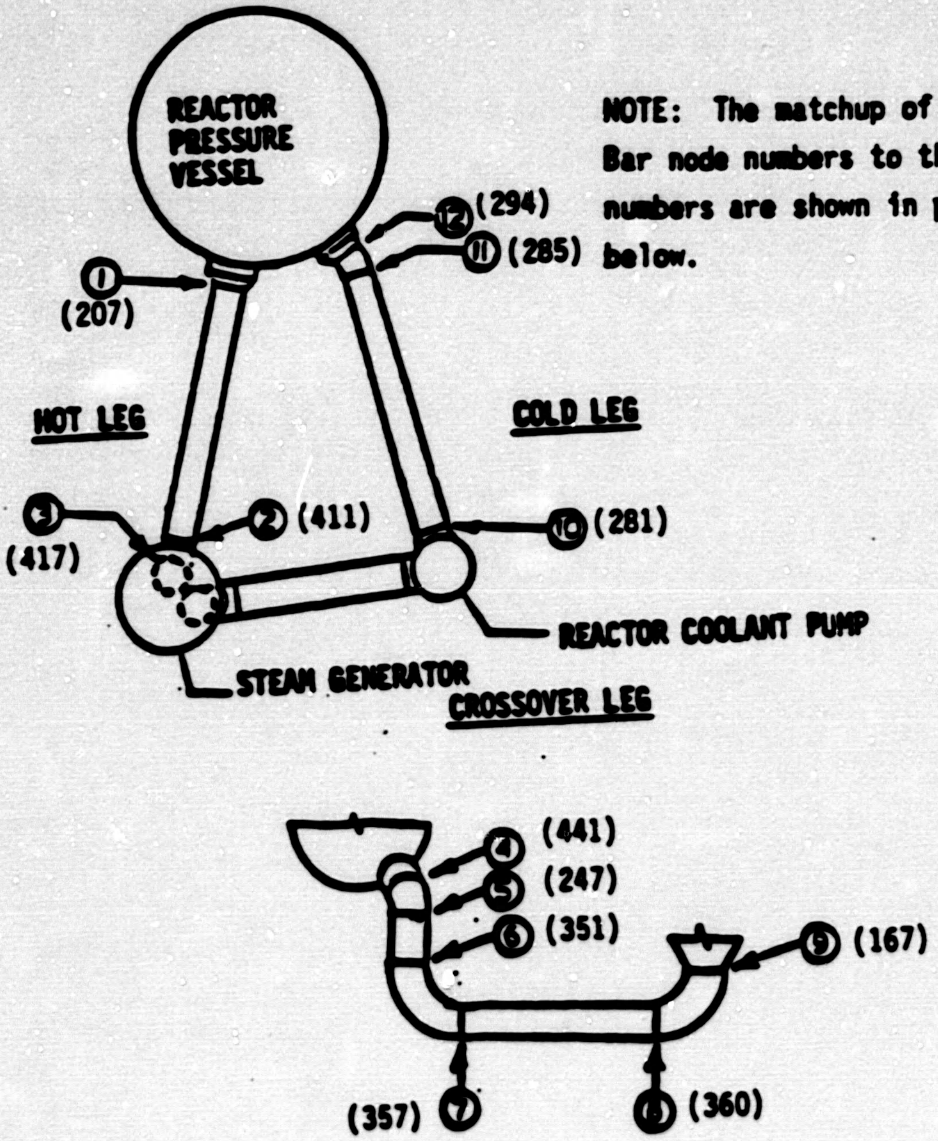
Faulted Loads

F_x = 2008 kips

M_b = 23468 in-kips

Figure 3-1. Reactor Coolant Pipe

NOTE: The matchup of the Watts Bar node numbers to the welds numbers are shown in parentheses below.



HOT LEG

Temperature: 618°F; Pressure 2250 psi

CROSSOVER LEG

Temperature: 558°F; Pressure 2250 psi

COLD LEG

Temperature: 558°F; Pressure 2305 psi

Figure 3-2. Schematic Diagram of RCL Showing Weld Identification

4.0 MATERIAL CHARACTERIZATION

4.1 Pipe, Fittings, and Weld Materials

The primary loop piping and fittings material for Watts Bar Units 1 and 2 is cast stainless steel SA-351-CF8A. Welds and fittings exist as indicated in figure 3-2. The piping is centrifugally cast while the fittings are statically cast. The field welds feature a gas tungsten arc weld (GTAW or TIG) root pass followed by shielded metal arc welding (SMAW) to completion. The shop welds are either SMAW or submerged arc (SAW) with a GTAW root pass. Weld repairs on shop welds would be either SMAW or GTAW. The welds have TP 308 stainless steel chemistry. No solution annealing was performed.

4.2 Tensile Properties

Plant specific material certifications were used to establish the tensile properties for the leak-before-break analyses. Tables 4-1 and 4-2 show the tensile properties of the materials at 650°F taken from the material certifications. The properties of these tables were used to obtain the minimum and average tensile properties of the materials at 650°F. These properties are shown in table 4-3 along with the ASME Code minimum properties which are included for a comparison. Tables 4-1 and 4-2 also include room temperature properties for information.

Table 4-4 shows the temperature and pressure conditions in the primary loop. At location 10, for example, the leak rate calculations are performed at normal operating temperature of 558°F and the crack stability analyses are also performed at 558°F. [

]a,c,e In brief, the following material properties are the ones used in the leak-before-break analyses set forth in this report.

Minimum SA351 CF8A Properties for Flaw Stability Analysis at Location 10 (558°F)

[ ]a,c,e

Yield Stress:
Ultimate Strength:
Poisson's Ratio:
Modulus of Elasticity:

Average SA351 CF8A Properties for Leak Rate Calculations at Location 10 (558°F)

[ ]a,c,e

Yield Stress:
Poisson's Ratio:
Modulus of Elasticity:

4.3 Fracture Toughness Properties

The pre-service fracture toughness of cast materials in terms of J have been found to be very high at 600°F. Typical results are given in figure 4-2 taken from reference 4-2. J_{Ic} is observed to be over 5000 in-lbs/in^2. 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 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 4-3. In that report, fracture toughness results were presented for a material [ ]^{a,c,e}). Toughness results were provided for the material in the full service life condition and these properties are also presented in figure 4-3 of this report for information. The J_{Ic} value for this material at operating temperature was [ ]^{a,c,e} in-lbs/in^2 at the end of life, and the maximum value of J obtained in the tests was in excess of [ ]^{a,c,e} in-lb/in^2. The tests of this material were conducted on small specimens and therefore rather short crack extensions occurred, (maximum extension 4.3 mm) so it is expected that higher J values would be sustained for larger specimens. Specifically, a value of 3000 in-lb/in^2 is acceptable (reference 4-4.) T_{mat} was [ ]^{a,c,e}.

The effect of the aging process on loop piping material for Watts Bar Units 1 and 2 is addressed in table 4-5, where the plant specific material chemistry for the loop materials is considered. The table shows that the degree of thermal aging expected by end-of-life is less than that produced in [ ]^{a,c,e}, since the minimum KCU is [ ]^{a,c,e} which is greater than the [ ]^{a,c,e}. (In reference 4-5, 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 toughness equals or exceeds that of [ ]^{a,c,e}. This toughness value for [ ]^{a,c,e}). Therefore the J_{Ic} values for end-of-life would be expected to be considerably higher than those reported in figure 4-3 (also see reference 4-6). In addition the tearing modulus would be greater than [ ]^{a,c,e}.

Available data on aged stainless steel welds (references 4-3 and 4-6) indicate the J_{Ic} values for the worst case welds are of the same order as the aged material. However, the slope of the J-R curve is steeper, and higher J-values have been obtained from fracture tests (in excess of 3000 in-lb/in^2). The applied value of the J-integral for a flaw in the weld regions will be lower than that in the base metal because the yield stress for the weld materials is much higher at temperature[a]. Therefore, weld regions are less limiting than the cast material.

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

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[a] In the report all $J_{applied}$ values were conservatively determined by using base metal strength properties.

It is therefore conservative to consider the [.   ]^{a,c,e} values of J_{Ic} = [   ]^{a,c,e} in-lbs/in^2, J_{max} = [   ]^{a,c,e} in-lbs/in^2, and T_{mat} = [   ]^{a,c,e} as the fracture toughness values applicable for fittings, piping and welds.

Since all the heats of piping and fittings in Watts Bar Units 1 and 2 pass the [   ]^{a,c,e} criteria, the location of highest stress is the governing location where leak-before-break margins must be demonstrated. From Table 3-1, the location of highest stress is location 10 (pump outlet nozzle junction). Thus, location 10 is the governing location.

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

4-3 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.

4-4 Letter: Dominic D. DiIanni, NRC to D. M. Musolf, Northern States Power Company, dated December 22, 1986, Docket No. 50-282 and 50-306.

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 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.

TABLE 4-1

MECHANICAL PROPERTIES AT ROOM TEMPERATURE AND 650°F
OF THE PRIMARY LOOP MATERIALS OF WATTS BAR UNIT 1

Heat No.	Product	Tensile	Yield	Tensile	Yield
		(at Room Temp, psi)		(at 650°F, psi)	
143334	Pipe	82,750	47,700		23,700
146271	Pipe	84,750	46,950		25,500
143335	Pipe	81,000	47,100		25,500
143336	Pipe	84,250	48,150		24,600
143309	Pipe	84,000	40,700		24,450
143340	Pipe	81,000	45,650		24,100
143311	Pipe	83,500	43,650		25,800
143312	Pipe	83,000	47,400		26,100
143279-1,2,3	Pipe	83,500	50,400		26,700
143280-1,2,3	Pipe	81,000	46,200		25,800
73477-1	Elbow	85,850	39,750		
73618-5	Elbow	81,800	37,000		
77445-1	Elbow	85,750	38,800		
70656-3	Elbow	82,350	41,050		
85305-1	Elbow	83,100	37,100		
82485-2	Elbow	89,800	43,300		
83976-1	Elbow	89,300	39,600		
84141-1	Elbow	81,550	37,750		
78807-1	Elbow	84,900	40,150		
80019-1	Elbow	77,700	35,150		
80585-1,2	Elbow	79,150	36,550		
87090-1	Elbow	83,000	39,250		
90643-1	Elbow	83,550	40,500		
92492-1	Elbow	82,950	39,100		
91777-1	Elbow	83,450	38,350		
92784-1	Elbow	86,050	42,250		
93262-1	Elbow	82,300	41,500		
93464-1	Elbow	83,650	40,150		
93609-1	Elbow	82,450	36,550	62,650	21,400

TABLE 4-2

MECHANICAL PROPERTIES AT ROOM TEMPERATURE AND 650°F OF THE PRIMARY LOOP MATERIALS OF WATTS BAR UNIT 2

Heat No.	Product	Tensile	Yield	Tensile	Yield
		(at Room Temp, psi)		(at 650°F, psi)	
143337	Pipe	82,650	45,150		27,450
143338	Pipe	82,250	46,650		24,600
143339	Pipe	83,000	46,850		24,000
143340	Pipe	80,750	45,750		25,800
143313	Pipe	83,250	48,150		25,650
143314	Pipe	80,700	45,000		27,150
143315	Pipe	80,500	47,100		24,150
143316	Pipe	78,750	42,750		22,800
143283-1,2,3	Pipe	78,000	49,200		23,700
143284-1,2,3	Pipe	82,000	48,750		25,550
81434-2	Elbow	85,050	40,000		
73618-4	Elbow	81,800	37,000		
76941-1	Elbow	83,650	40,150		
77639-1	Elbow	86,600	41,250		
85204-1	Elbow	81,700	37,600		
85403-1	Elbow	79,900	35,800	60,950	21,000
86907-1	Elbow	87,700	38,350	80,900	36,550
85106-1	Elbow	85,650	41,700		
77807-2	Elbow	84,900	40,150		
80195-1	Elbow	82,650	35,100	58,250	20,100
80195-2	Elbow	82,650	35,100	58,250	20,100
80643-1	Elbow	87,400	38,650		
92229-1	Elbow	83,950	38,800		
91138-1	Elbow	79,850	36,400	65,700	25,650
92684-1	Elbow	84,050	39,250		
92528-1	Elbow	80,500	38,200		
92852-1	Elbow	83,200	40,600		
89064-1	Elbow	86,100	39,750		
93278-1	Elbow	82,700	36,850		
89448-1	Elbow	81,200	36,100	62,200	25,450

TABLE 4-3
SA351-CF8A MATERIAL PROPERTIES AT 650°F

	s_y (ksi)	s_u (ksi)	
ASME Code	21.0	65.2	a,c,e

TABLE 4-4

PRESSURE AND TEMPERATURE CONDITIONS
IN THE PRIMARY LOOP FOR WATTS BAR UNITS 1 AND 2

Hot Leg	Temperature:	618°F
	Pressure:	2250 psig
Crossover Leg	Temperature:	558°F
	Pressure:	2250 psig
Cold Leg	Temperature:	558°F
	Pressure:	2305 psig

TABLE 4-5
CHEMISTRY PROPERTIES AND END-OF-LIFE KCU
FOR WATTS BAR UNITS 1 AND 2

Watts Bar Unit 1

%NI	%C	%MN	%N	%CR	%SI	%MO	%CB	CR(E)	NI(E)	CR/NI	DELTA	FRACTURE TOUGHNESS (KCU)

a,c,e

TABLE 4-5 (cont.)
CHEMISTRY PROPERTIES AND END-OF-LIFE KCU
FOR WATTS BAR UNITS 1 AND 2

%NI	%C	%MN	%N	%CR	%SI	%MO	%CB	CR(E)	NI(E)	CR/NI	DELTA	FRACTURE TOUGHNESS (KCU)

a,c,e

TABLE 4-5 (cont.)
CHEMISTRY PROPERTIES AND END-OF-LIFE KCU
FOR WATTS BAR UNITS 1 AND 2

%NI	%C	%MN	%N	%CR	%SI	%MO	%CB	CR(E)	NI(E)	CR/NI	DELTA	FRACTURE TOUGHNESS (KCU)

a,c,e

3293s-101988:10

TABLE 4-5 (cont.)
CHEMISTRY PROPERTIES AND END-OF-LIFE KCU
FOR WATTS BAR UNITS 1 AND 2

%NI	%C	%MN	%N	%CR	%SI	%MO	%CB	CR(E)	NI(E)	CR/NI	DELTA	FRACTURE TOUGHNESS (KCU)

a,c,e