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ENCLOSURE B

WCAP-16738-NP, Revision 0, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program," {NON-PROPRIETARY}

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

Westinghouse Non-Proprietary Class 3

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Revision 0

March 2007

Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program

Westinghouse

WCAP-16738-NP Revision 0

Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program

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March 2007

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

EXECUTIVE SUMMARY

Westinghouse performed analyses for the Leak-Before-Break (LBB) of Kewaunee Power Station (KPS) primary loop piping in 1987. The results of these analyses were documented in WCAP-11411 Revision 1 (Reference 1-2) and WCAP-11619 (Reference 1-3), which were approved by the NRC in a letter dated February 16, 1988 (Reference 1-4).

Westinghouse performed another analysis to support Power Uprate program in WCAP-16040-P Revision 0 (Reference 1-12). This current report (WCAP-16738) demonstrates compliance with LBB technology for the Kewaunee reactor coolant system piping due to the steam generator replacement, Tavg operating window, power uprate and the License Renewal programs. The report documents the plant specific geometry, operating parameters, 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, fracture toughness considering thermal aging was determined for each heat of material for the fully aged condition (applicable for the license renewal period). Information from References 1-2 and 1-3 is used for this evaluation.

This report includes the temperature, pressure and loadings generated as a result of the changes due to the steam generator replacement, Tavg operating window, power uprate and License Renewal Programs.

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 (10) 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.

The effects of the steam generator replacement, Tavg operating window, the power uprate and License Renewal Programs on the continued applicability of LBB for the reactor coolant loop piping at the KPS have been evaluated. It is demonstrated that the previous LBB conclusions still remains valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the structural design basis of the KPS due to the steam generator replacement, Tavg operating window, power uprate and the License Renewal programs.

1.0 INTRODUCTION

1.1 PURPOSE

This report applies to the KPS Reactor Coolant System (RCS) primary loop piping. It is intended to demonstrate that for the specific parameters of the KPS, RCS primary loop pipe breaks need not be considered in the structural design basis due to the steam generator replacement, Tavg operating window, power uprate and for the License Renewal programs. The approach taken has been accepted by the Nuclear Regulatory Commission (NRC) (Reference 1-5).

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 1-6). 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 1-7).

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 1-8 and 1-9). The results from the LLNL study were released at a March 28, 1983, ACRS Subcommittee meeting. These studies, which are applicable to all Westinghouse plants east of the Rocky Mountains, determined the mean probability of a direct LOCA (RCS primary loop pipe break) to be 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 1-6) 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 1-10) 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 1-5).

1.3 SCOPE AND OBJECTIVES

The general purpose of this investigation is to demonstrate leak-before-break for the primary loops in KPS on a plant specific basis due to the steam generator replacement, Tavg operating window, the power uprate and for the License Renewal programs. The recommendations and criteria proposed in Reference 1-11 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 a governing location. Determine fatigue crack growth. Show that a through-wall crack will not result.

4. Postulate a through-wall flaw at the governing (critical) 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. 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 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 KPS consistent with the NRC position for exemption from consideration of dynamic effects.

It should be noted that the terms "flaw" and "crack" have the same meaning and are used interchangeably. "Governing location" and "critical location" are also used interchangeably throughout the report.

1.4 REFERENCES

1-1 WCAP-7211, Revision 5, "Proprietary Information and Intellectual Property Management Policies and Procedures," November, 2005.

1-2 WCAP-11411, Revision 1, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee," April 1987 (Westinghouse Proprietary Class 2).

1-3 WCAP-11619, "Additional Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee," October 1987 (Westinghouse Proprietary Class 2).

1-4 USNRC Docket No. 50-305, dated February 16, 1988, "Application of Leak-Before-Break Technology as a Basis for Kewaunee Power Station Steam Generator Snubber Reduction".

1-5 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.

1-6 WCAP-9283, "Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March, 1978.

1-7 Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, November 10, 1981.

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

1-9 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

1-10 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.

1-11 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.

1-12 WCAP-16040-P Revision 0,"Power Uprate Project Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report " February 2003.

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 1400 reactor-years, including 16 plants each having over 30 years of operation, 10 other plants each with over 25 years of operation, 11 plants each over 20 years of operation and 12 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 feed water 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.

Primary Water Stress Corrosion Cracking (PWSCC) occurred in V. C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld. It should be noted that this susceptible material is <u>not</u> found at the KPS primary loop piping.

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

An 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 exceedance of the vibration limits. Field measurements have been made on a number of plants during hot functional testing, including plants similar to Kewaunee. 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 weld thickness of the pipe are 34.58 in. and 2.69 in., respectively, as shown in the figure. The normal stresses at the weld location 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 normal 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 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 geometry and operating temperature of the cross-over leg and the cold leg are different than the hot leg, locations other than highest stressed location were examined taking into consideration both fracture toughness and stress. 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 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:

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

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:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \tag{3-2}$$

where,

M_X = X component of moment, Torsion

M_Y = Y component of bending moment

M_Z = Z component of bending moment

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 is 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 weld locations of interest (see Figure 3-2) are given in Table 3-2.

3.5 REFERENCE

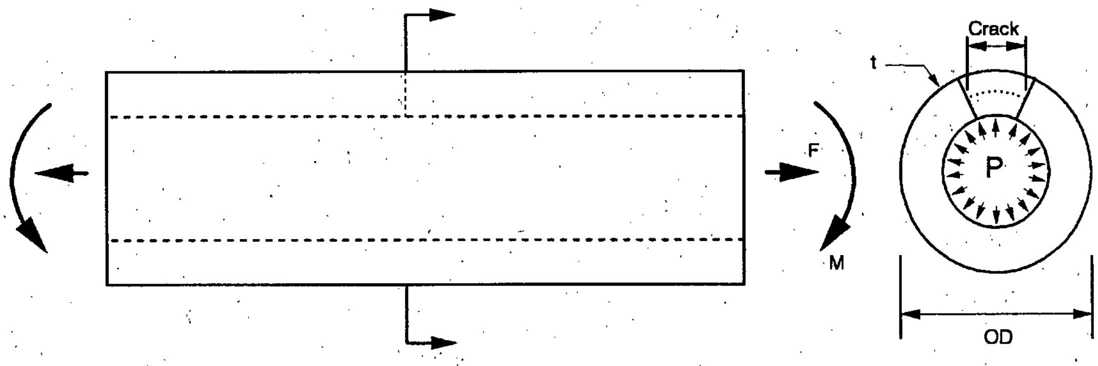
3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

Table 3-1 Dimensions, Normal Loads and Normal Stresses for Kewaunee					
Location[a]	Outside Diameter (in)	Minimum Thickness (in)	Axial Load[b] (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	34.58	2.69	1487	23329	17.21
2	34.58	2.69	1487	5733	8.39
3	36.96	2.88	1573	12371	10.17
4	36.96	2.88	1696	2805	6.65
5	36.96	2.88	1692	2663	6.58
6	36.96	2.88	1686	2626	6.54
7	36.96	2.88	1710	1310	6.08
8	36.96	2.88	1710	2903	6.74
9	36.96	2.88	1765	5354	7.92
10	32.80	2.55	1358	4660	8.34
11	32.80	2.55	1358	3426	7.62
12	32.80	2.55	1356	4923	8.49

a. See Figure 3-2
b. Included Pressure

Table 3-2	Faulted Loads and Stresses for Kewaunee		
Location[a,b]	Axial Load[c] (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	1656	25034	18.69
2	1656	7183	9.75
3	1957	14779	12.41
4	1752	4043	7.34
5	1769	3664	7.24
6	1762	3302	7.07
7	1719	1855	6.34
8	1719	3574	7.04
9	1774	6726	8.51
10	1422	6691	9.80
11	1422	3763	8.08
12	1409	5692	9.16

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



OD[a] = 34.58 in
t[a] = 2.69

Normal Loads[a]		Faulted Loads[b]	
Force[c]:	1487 kips	Force[c]:	1656 kips
Bending moment:	23329 in-kips	Bending moment:	25034 in-kips

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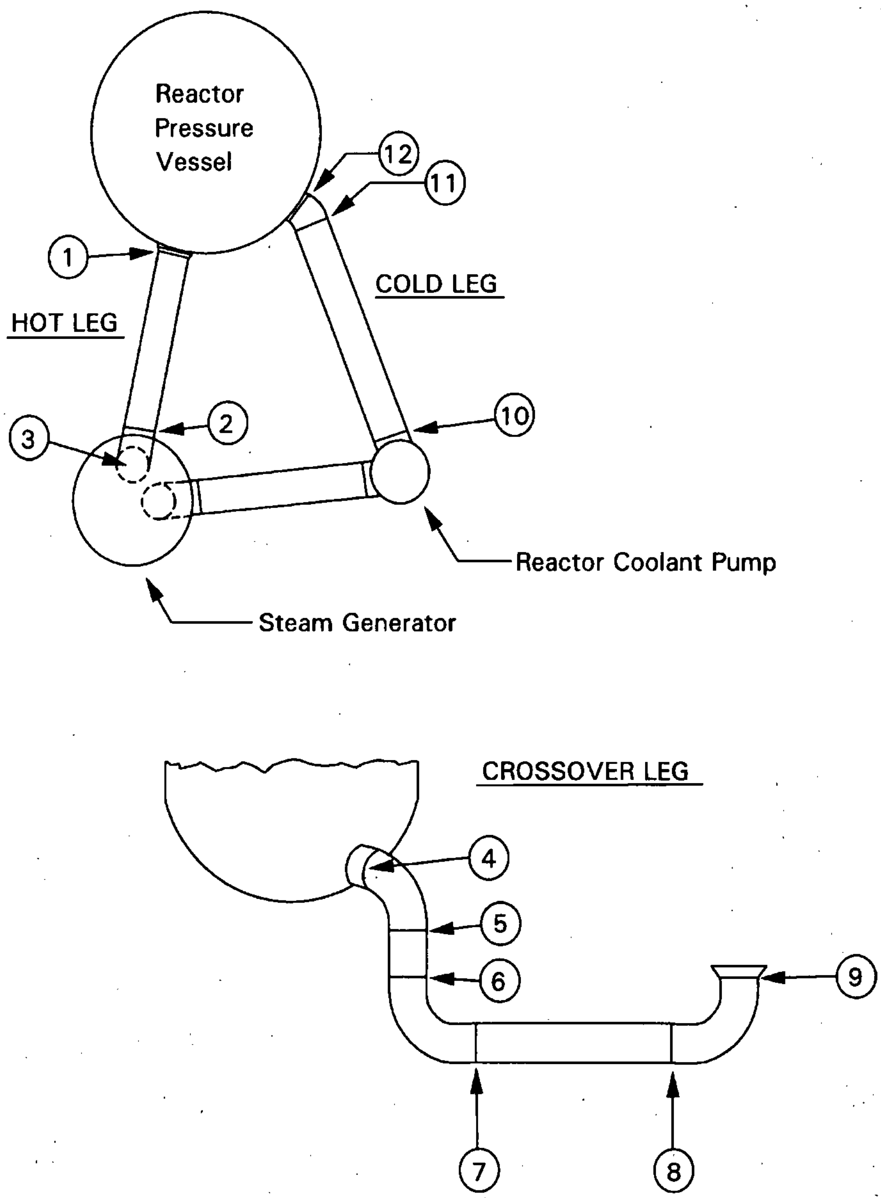
[a] See Table 3-1
[b] See Table 3-2
[c] Includes the force due to a pressure of 2250 psia

Figure 3-1 Hot Leg Coolant Pipe



HOT LEG
Temperature: 607°F Pressure: 2250 psia

CROSS-OVER LEG
Temperature: 544°F Pressure: 2250 psia

COLD LEG
Temperature: 544°F Pressure: 2250 psia
(Note: Temperature are rounded off)

Figure 3-2 Schematic Diagram of Kewaunee Primary Loop Showing Weld Locations

4.0 MATERIAL CHARACTERIZATION

4.1 PRIMARY LOOP PIPE AND FITTINGS MATERIALS

The primary loop pipe and the fittings material for KPS is SA351 CF8M.

4.2 TENSILE PROPERTIES

The Certified Materials Test Reports (CMTRs) for KPS primary loop piping 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.

Material properties at 607°F and 544°F are needed for the leak before break analysis. The representative properties at 607°F and 544°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 4-1). Code tensile properties at 607°F and 544°F were obtained by interpolating between the 500°F, 600°F and 650°F tensile properties. Ratios of the code tensile properties at 607°F and 544°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 SA351 CF8M at 607°F and 544°F.

The average and lower bound yield strengths and ultimate strengths are given in Table 4-2. The ASME Code moduli of elasticity values are also given, and Poisson's ratio was taken as 0.3.

4.3 FRACTURE TOUGHNESS PROPERTIES

The pre-service fracture toughness (J) of both forged and cast stainless steels of interest are in terms of J_{IC} (J at Crack Initiation) have been found to be very high at 600°F (see figure 4-1). Fracture toughness values for forged materials are even higher. However, cast stainless steel is susceptible to thermal aging during service. Thermal aging of cast stainless steel results in 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 fracture toughness values shown in WCAP-11411 Revision 1 (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 KPS primary loop pipe and elbow fitting material are available from CMTRs and are provided in Table B-1 of WCAP-11411 Revision 1 (Reference 4-2) and also reproduced in Table 4-3 and Table 4-4 of this report.

The following equations are taken from Reference 4-4:

The saturation room temperature (RT) impact energies of the cast stainless steel materials were determined from the chemical compositions. δ_c (ferrite content) in percent volume were obtained from Table B-1 of WCAP-11411 Revision 1 (Reference 4-2) and were used in this analysis.

For CF8M steel with < 10% Ni, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from

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

where the material parameter ϕ is expressed as

$$\phi = \delta c\,(Ni + Si + Mn)^2(C + 0.4N)/5.0 \qquad (4\text{-}2)$$

and from

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

For CF8M steel with > 10% Ni, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from

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

where the material parameter ϕ is expressed as

$$\phi = \delta c\,(Ni + Si + Mn)^2(C + 0.4N)/5.0 \qquad (4\text{-}5)$$

and from

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

The saturation J-R curve at RT, for static-cast CF8M steel is given by

$$J_d = 16(Cv_{sat})^{0.67}(\Delta a)^n \qquad (4\text{-}7)$$

For centrifugally cast CF8M steel, it is given by

$$J_d = 20(Cv_{sat})^{0.67}(\Delta a)^n \quad (4\text{-}8)$$

where the exponent n for CF8M steel is expressed as

$$n = 0.23 + 0.08 \log_{10} (Cv_{sat}) \quad (4\text{-}9)$$

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

The saturation J-R curve at 290°C (554°F), for static-cast CF8M steel is given by

$$J_d = 49 (Cv_{sat})^{0.41}(\Delta a)^n \quad (4\text{-}10)$$

For centrifugally cast CF8M steel, it is given by

$$J_d = 57(Cv_{sat})^{0.41}(\Delta a)^n \quad (4\text{-}11)$$

where the exponent n for CF8M steel is expressed as

$$n = 0.23 + 0.06 \log_{10} (Cv_{sat}) \quad (4\text{-}12)$$

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

[

]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 strength for the weld materials is much higher at the temperature[1]. Therefore, weld regions are less limiting than the cast base material.

In the fracture mechanics analyses that follow, the fracture toughness properties given in Table 4-5 will be used as the criteria against which the applied fracture toughness values will be compared.

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-11411, Revision 1, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee," April 1987 (Westinghouse Proprietary Class 2).

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.

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

Table 4-1 Measured Tensile Properties for Kewaunee Primary Loop Piping System					
		At Room Temperature		At 650°F	
Heat Number	Location	YIELD (PSI)	ULTIMATE (PSI)	YIELD (PSI)	ULTIMATE (PSI)
A3391234C	Cold Leg Pipe	43000	80750	24800	68000
A3557890A&B	Cold Leg Pipe	37900	80400	24200	66250
A3626789A	Cold Leg Pipe	43100	78850	22900	62250
A362012345A	Hot Leg Pipe	44950	84900	26900	66250
A-351567890A&B	Hot Leg Pipe	39950	80000	24500	64000
A-355123456-A	Hot Leg Pipe	43950	88100	26200	73500
B-223012345	Crossover Pipe	40950	82900	24500	66500
A-352123456A	Crossover Pipe	43000	88750	30400	72750
B-1853A	Crossover Pipe	44000	75325	23100	62250
B-2256A&B	Crossover Pipe	38460	77900	22900	65000
35794-3	Cold Leg Elbow	41400	84900	N/A	N/A
36668-1	Cold Leg Elbow	44400	85700	N/A	N/A
34027-1	Crossover Elbow	35350	73750	N/A	N/A
33712-2	Crossover Elbow	47220	86220	N/A	N/A
37034-1	Crossover Elbow	42100	85200	N/A	N/A
37429-1	Crossover Elbow	43800	85100	N/A	N/A
36896-3	Crossover Elbow	41500	82000	N/A	N/A
37523-5	Crossover Elbow	41700	84900	N/A	N/A
34837-2	Crossover Elbow	34500	72500	N/A	N/A
39792-2	Crossover Elbow	45700	85700	N/A	N/A
35794-2	Crossover Elbow	43600	85000	N/A	N/A
36572-1	Crossover Elbow	48500	88400	N/A	N/A
33801-1	Hot Leg Elbow	40800	81800	N/A	N/A
38757-2	Hot Leg Elbow	45900	86200	N/A	N/A

N/A = Not available

Table 4-2 Mechanical Properties for Kewaunee Materials at Operating Temperatures				
Material	**Temperature (°F)**	**Average Yield Strength (psi)**	**Lower Bound**	
			Yield Stress (psi)	**Ultimate Strength (psi)**
SA351 CF8M	607	25117	23222	62250
	544	26001	24039	62250
Modulus of Elasticity				
$E = 25.26 \times 10^6$ psi at 607°F				
$E = 25.58 \times 10^6$ psi at 544°F				
Poisson's *ratio*: 0.3				

Table 4-3 Chemistry and Fracture Toughness Pipe Properties of the Material Heats of Kewaunee

Heat Number	%Ni	%C	%Mn	%Cr	%Si	%Mo	%N	δc^{1}	ϕ^{2}	$C_{Vsat1}{}^{3}$	$C_{vsat2}{}^{4}$	$C_{vsat}{}^{5}$	n^{6}	$J_{IC}{}^{7}$

a,c,e

Notes:

- * Hot Leg
- ** Cross-over Leg
- *** Cold Leg

[1]From Reference 4-2
[2]From Equations 4-2 and 4-5
[3]From Equations 4-1 or 4-4
[4]From Equations 4-3 or 4-6
[5]Minimum of C_{Vsat1} and C_{Vsat2}
[6]From Equation 4-12
[7]J_{IC} (converted in in-lb/in^2) Values obtained by extrapolating between Room Temperature (RT) and 554°F values

Table 4-4 Chemistry and Fracture Toughness Elbow Properties of the Material Heats of Kewaunee

Heat Number	%Ni	%C	%Mn	%Cr	%Si	%Mo	%N	δc[1]	ϕ[2]	C_{Vsat1}[3]	C_{vsat2}[4]	C_{vsat}[5]	n[6]	J_{IC}[7]

a,c,e

Notes:

[1]From Reference 4-2
[2]From Equations 4-2 and 4-5
[3]From Equations 4-1 or 4-4
[4]From Equations 4-3 or 4-6
[5]Minimum of C_{Vsat1} and C_{Vsat2}
[6]From Equation 4-12
[7]J_{IC} (converted in in-lb/in^2) Values obtained by extrapolating between Room Temperature (RT) and 554°F values

* Hot Leg
** Cross-over Leg
*** Cold Leg

Table 4-5 Fracture Toughness Properties for Kewaunee Primary Loops for Leak-Before-Break Evaluation at Critical Locations				
Location	**J_{IC} (in-lb/in^2)**	**T_{mat} (non-dimensional)**	**J_{MAX} (in-lb/in^2)**	**Heat Number**

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 critical locations (governing locations). 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 KPS primary loop piping. 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. In addition critical locations for the crossover leg and the cold leg were selected for the analysis. Location 9 was selected which has the highest stress critical location in the crossover leg. In the cold leg, location 10 has the highest stress and therefore location 10 is the critical location in the cold leg. At these locations worst-case material properties for piping and/or elbow fittings were applied. It is thus concluded that the enveloping locations in KPS primary loop piping for which LBB methodology is to be applied are locations 1, 9, and 10. The tensile properties and the allowable toughness for the critical locations are shown in Tables 4-2 and 4-5.

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 6-2 was used to estimate the critical pressure, Pc, 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 6-2). 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, where P_o is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

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

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

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [ ]^{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 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 4-2) 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 flaw sizes.

The Kewaunee 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 flaw sizes) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [

]a,c,e

6-2 M. M, El-Wakil, "Nuclear Heat Transport, international Textbook Company," New York, N.Y, 1971.

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	4.29
9	7.57
10	7.10

a,c,e

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

a,c,e

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

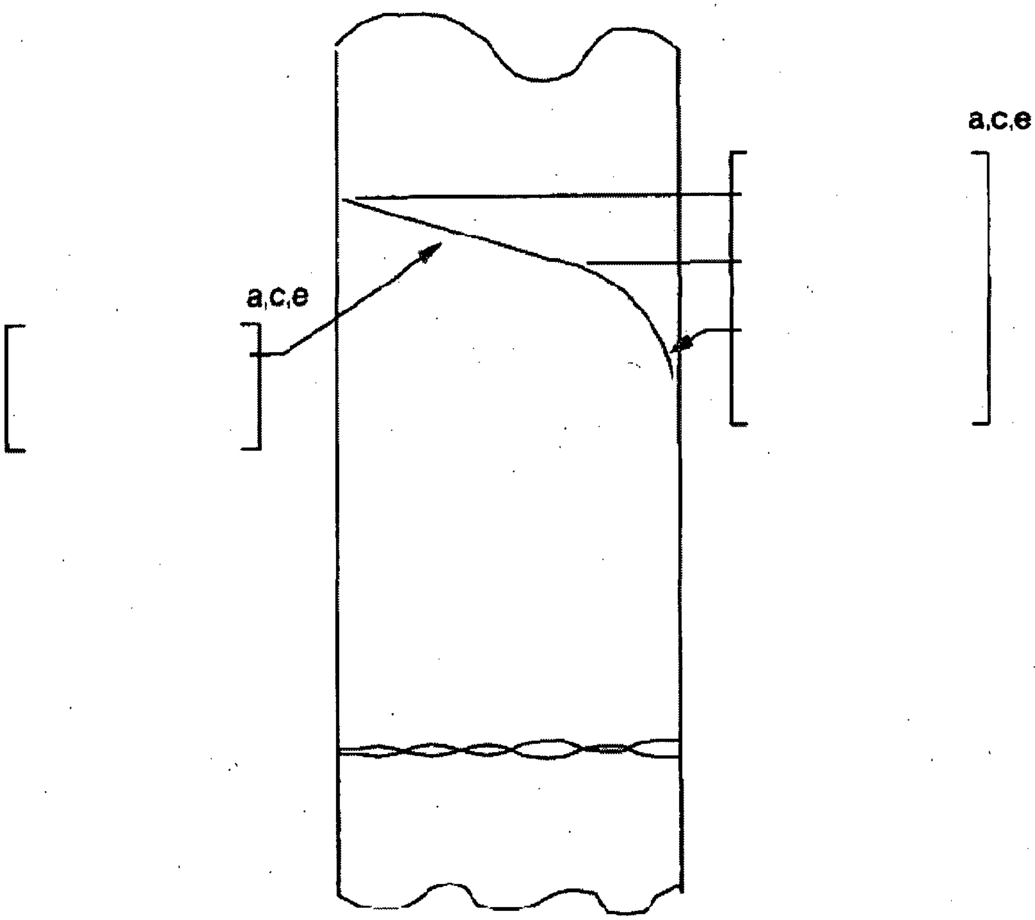


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

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 final 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 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 = $0.5 (\sigma_y + \sigma_u)$ = 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

Stability analyses were performed at the governing 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 were used. 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. The postulated flaw size was 2 times (for flaw size margin of 2) the leakage flaw size established in section 6.0 (see Table 6-1). Evaluations were performed at the critical locations identified in Section 5.1. The results of the elastic-plastic fracture mechanics J-integral evaluations are given in Table 7-1.

A stability analysis based on limit load was performed for these locations as described in Section 7.2. The field weld process type, at locations 1, 9, and 10, are used conservatively as SMAW. The "Z" factor correction for SMAW (Reference 7-3) is follows:

$$Z = 1.15\ [1.0 + 0.013\ (OD-4)] \text{ for SMAW}$$

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 1.61 for location 1. The Z factor was 1.64 for location 9. The Z factor was 1.58 for location 10. 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 flaw sizes are also presented on the same table.

7.4 REFERENCES

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

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

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

Table 7-1 Stability Results for Kewaunee Based on Elastic-Plastic J-Integral Evaluations

a,c,e

N/A= Not Applicable; T_{app} is not applicable since $J_{app} < J_{IC}$; * 2 times the leakage flaw size.

Table 7-2 Stability Results for Kewaunee Based on Limit Load

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)
1	26.43	4.29
9	43.51	7.57
10	37.40	7.10

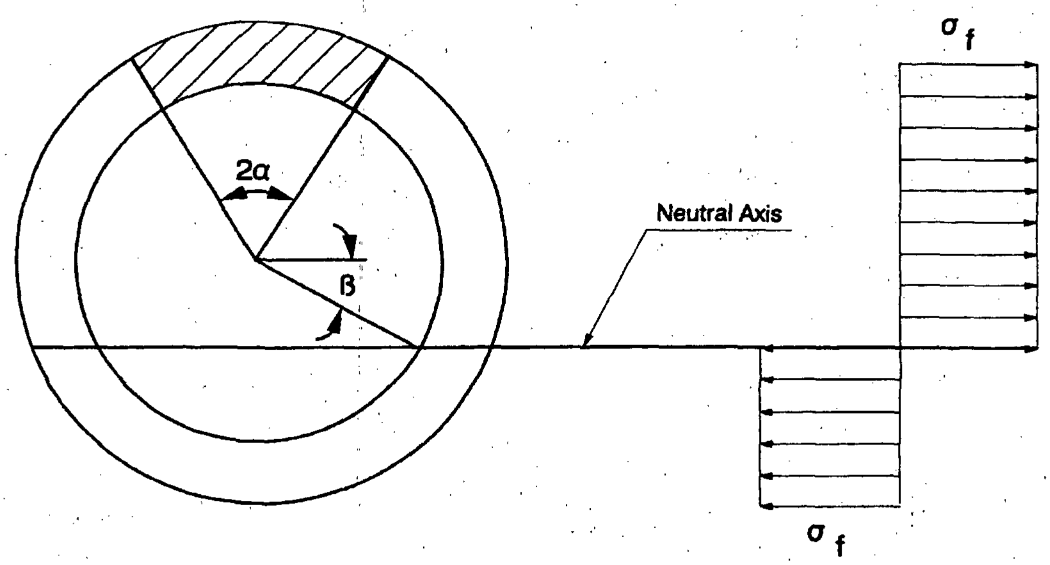


Figure 7-1 [ ]^{a,c,e} Stress Distribution

a,c,e

OD = 34.58 in.	σ_y = 23.22 ksi	F = 1656 kips
t = 2.69 in.	σ_u = 62.25 ksi	M = 25034 in-kips

SA351-CF8M with SMAW Weld

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

a,c,e

OD = 36.96 in. σ_y = 24.04 ksi F = 1774 kips

t = 2.88 in. σ_u = 62.25 ksi M = 6726 in-kips

SA351-CF8M with SMAW Weld

Figure 7-3 Critical Flaw Size Prediction - Cross-over Leg at Location 9

a,c,e

[ ]

OD = 32.80 in. σ_y = 24.04 ksi F = 1422 kips

t = 2.55 in. σ_u = 62.25 ksi M = 6691 in-kips

SA351-CF8M with SMAW Weld

Figure 7-4 Critical Flaw Size Prediction - Cold Leg at Location 10

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 for Kewaunee (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 finite element stress analysis was carried out for the inlet nozzle safe end region. The normal, upset, and test conditions were considered. Table 8-1 summarizes the transients and cycles for Kewaunee. The circumferentially oriented surface flaws were postulated at 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 8-1. A compilation of data for austenitic stainless steel in a PWR water environment was presented in Reference 8-2, and it was found that the effect of the environment on the crack growth rate was very small. From the information it was estimated that the environmental factor should be conservatively set at [ ]^{a,c,e} in the crack growth equation from Reference 8-1.

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 is very small,

[ ]^{a,c,e}

Fatigue crack growth analysis results shown in Table 8-2 are also valid for 60-year plant life (license renewal period) assuming the transients and cycles for 40-year plant life are valid for 60-year plant life. This assumption is typical for Westinghouse plants.

8.1 REFERENCES

8-1 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.

8-2 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 Reactor Coolant System Operating Transients (40-Year Plant Life*)		
Number	**Transient Identification**	**Number of Cycles**
1	Heatup at 100°F/hr Cooldown at 100°F/hr (Pressurizer 200°F/hr)	200 200
2	Plant Loading at 5% of Full Power/min Plant Unloading at 5% of Full Power/min	18300 18300
3	Step Load Increase of 10% of Full Power Step Load Decrease of 10% of Full Power	2000 2000
4	Large Step Load Decrease in Load (with Steam Dump)	200
5	Loss of Load (without immediate Turbine or Reactor Trip)	80
6	Loss of power (Blackout with natural circulation in the Reactor Coolant System)	40
7	Loss of Flow (Partial Loss of Flow, one pump only)	80
8	Reactor Trip From Full Power	400
9	Turbine Roll Test	10
10	Hydrostatic Test Conditions: a. Primary Side Hydrostatic Test before initial Startup at 3107 psig b. Secondary Side Hydrostatic Test before initial Startup at 1356 psig	5 5
11	Primary Side Leak Test	50
12	Accident Conditions: a. Reactor Coolant Pipe Break c. Steam Pipe Break	1 1
13	Steady-State Fluctuations- the Reactor Coolant average Temperature for Purpose of Design is assumed to Increase and Decrease a Maximum of 6°F in 1 Minute. The Corresponding Reactor Coolant Pressure Variation is Less than 100 psi.	Infinite

* also assumed to be applicable for 60-year plant life.

Table 8-2 Fatigue Crack Growth at [ ]a,c,e (40-year and 60-year)		
FINAL FLAW (in.)		
Initial Flaw	Ferritic Steel	Stainless

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 [ ]a,c,e

9.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability 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 of 1 on loads using the absolute summation of faulted load combinations is satisfied.

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Kewaunee

Location	Leakage Flaw Size	Critical Flaw Size	Margin
1	4.29 in.	26.43$^{\alpha}$ in.	6.1$^{\alpha}$
1	4.29 in.	8.58$^{\beta}$ in.	>2.0$^{\beta}$
9	7.57 in.	43.51$^{\alpha}$ in.	5.7$^{\alpha}$
9	7.57 in.	15.14$^{\beta}$ in.	>2.0$^{\beta}$
10	7.10 in.	37.40$^{\alpha}$ in.	5.2$^{\alpha}$
10	7.10 in.	14.20$^{\beta}$ in.	>2.0$^{\beta}$

$^{\alpha}$ based on limit load

$^{\beta}$ based on J integral evaluation (note: critical flaw size for J integral evaluation was postulated as 2 times the leakage flaw size in order to satisfy a margin on flaw size of 2. Since J applied is higher than J allowable (see Table 7-1) the flaw size margin is >2).

10.0 CONCLUSIONS

This report justifies the elimination of RCS primary loop pipe breaks from the structural design basis for the KPS due to the steam generator replacement, Tavg operating window, power uprate and the License Renewal programs 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.

   Note: Primary Water Stress Corrosion Cracking (PWSCC) occurred in V. C. Summer reactor vessel hot leg nozzle, Alloy 82/182 weld. It should be noted that this susceptible material is not found at the KPS primary loop piping.

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 Kewaunee 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 (fully aged) 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 KPS primary loop piping. All the recommended LBB margins are satisfied. The effect of steam generator replacement, Tavg operating window, power uprate and the License Renewal programs on the continued applicability of LBB for the reactor coolant loop piping at the KPS has been evaluated. It is demonstrated that the previous LBB conclusions still remains valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the structural design basis of the KPS due to the steam generator replacement, Tavg operating window, power uprate and the License Renewal programs.

APPENDIX A: LIMIT MOMENT

[

]a,c,e

a,c,e

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

ENCLOSURE C

Affidavit for Westinghouse Report WCAP-16738-P, Rev. 0, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program"

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Our ref: CAW-09-2621

July 27, 2009

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16738-P, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-09-2621 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Dominion Energy Kewaunee, Inc.

Correspondence with respect to the proprietary aspects of the Application for Withholding Proprietary Information from Public Disclosure or the Westinghouse Affidavit should reference this letter, CAW-09-2621, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: George Bacuta (NRC OWFN 12E-1)

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co.; 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) (letter and Affidavit only)
D. L. Rogosky
R. J. Morrison

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 27th day of July, 2009

Sharon L. Markle

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16738-P, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program" (proprietary), dated March 2007 for submittal to the Commission, being transmitted by Dominion Energy Kewaunee, Inc. letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Kewaunee Power Station is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of leak-before-break.

This information is part of that which will enable Westinghouse to:

(a) Provide technical support for the application of leak-before-break technology.

(b) Assist the customer in the licensing of leak-before-break as the design basis.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of using leak-before-break technology as the plant design basis.

(b) Westinghouse can sell support and defense of leak-before-break analyses.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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