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A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of H. B. Robinson Unit 2 for the License Renewal program

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1 BACKGROUND AND OBJECTIVE

1.1 BACKGROUND

Periodic volumetric inspections of the welds of the primary loop pump casings of commercial nuclear power plants are required by Section XI of the ASME Boiler and Pressure Vessel Code (see Table IWB-2500-1, Examination Categories). These inspections are quite costly in terms of both dollars and radiation exposure (rem). To perform a volumetric inspection, complete disassembly of the pump is required. A lowering of the primary coolant water level most likely would be necessary which would, in turn, necessitate a complete core unload. Even then the volumetric inspection is very difficult. The pump casings are inspected twice prior to placing in service. When fabricated the castings are radiographed and liquid penetrant tested. After assembly, the welds are again radiographed and liquid penetrant tested. This in-shop examination is required per Section III of the American Society of Mechanical Engineer's (ASME) Boiler and Pressure Vessel Code. The pre-service inspection criteria are the same as the in-service inspection criteria. Since no significant mechanisms exist for crack initiation and propagation, these criteria requiring that all welded surfaces be volumetrically and surface examined may not be warranted. In recognition of these facts the ASME Code body approved Code Case N-481 which provides an alternative to the volumetric inspection requirement (Reference 1[*]).

The ASME Code Case, N-481 (Alternate Examination Requirements for Cast Austenitic Pump Casings), allows the replacement of volumetric examinations of primary loop pump casing welds with fracture mechanics based integrity evaluations (Item (d) of the code case) supplemented by specific visual inspections. It also requires that a report of the evaluation be submitted to the regulatory and enforcement authorities having jurisdiction at the plant site for review (Item (e) of the code case). A copy of the code case is given in Appendix A.

Following approval of Code Case N-481 by the ASME, the Westinghouse Owners Group sponsored the analyses required by the code case which are applicable to the various primary loop pump casing models found in Westinghouse design nuclear steam supply systems. This work is documented in WCAP-13045 (Reference 2). Specifically, stress analyses for loadings on the pump casings were performed to support the fracture mechanics analyses for postulated flaws. Compliance to Item (d) of ASME Code Case N-481 was demonstrated on a generic basis.

However, a plant specific evaluation to demonstrate safety and serviceability is required by Code Case N-481. Since there is a variety of pumps casing models, loads and materials as discussed in WCAP-13045, it was not feasible to qualify each plant of Westinghouse design specifically to the requirements of the code case. Rather, enveloping or bounding criteria were set up whereby a specific utility, in most cases, needs only to show that the primary loop pump casings fall under the umbrella established by the analyses. The U.S. Nuclear Regulatory

[*] See Section 9.0 for a listing of references.

Commission (U.S. NRC) has approved ASME Code Case N-481 in Revision 9 of Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI Division I," dated April 1992.

1.2 OBJECTIVE

It is the objective of this report to qualify the primary loop pump casings of the H. B. Robinson Unit 2 to Item (d) of ASME Code Case N-481 (Reference 1).

Revision 0 of this report was issued in April 2000 for the 40-year life of the plant. The objective of Revision 1 of the report is to validate the integrity of the H. B. Robinson Unit 2 primary loop pump casings to ASME Code Case N-481 for the 60-year plant life (as a part of the License Renewal Program).

Revision 1 is to revise cover page, pages 1-2, 4-1, 4-2, 4-3, 5-2, 6-1, 8-1, 9-1 and 9-2, Table 4-1, Table 5-2 and Table 6-1. The revision is identified by bar in the column on the right.

2 DESCRIPTION OF THE PRIMARY LOOP PUMP CASINGS OF H. B. ROBINSON UNIT 2

The primary loop pump casings of H. B. Robinson Unit 2 are Westinghouse Model 93 design. The pump casings are fabricated from SA351 CF8 cast stainless steel. A sketch of a typical pump casing of this type along with the weld locations is shown in Figure 2-1. This figure also contains typical dimensions.

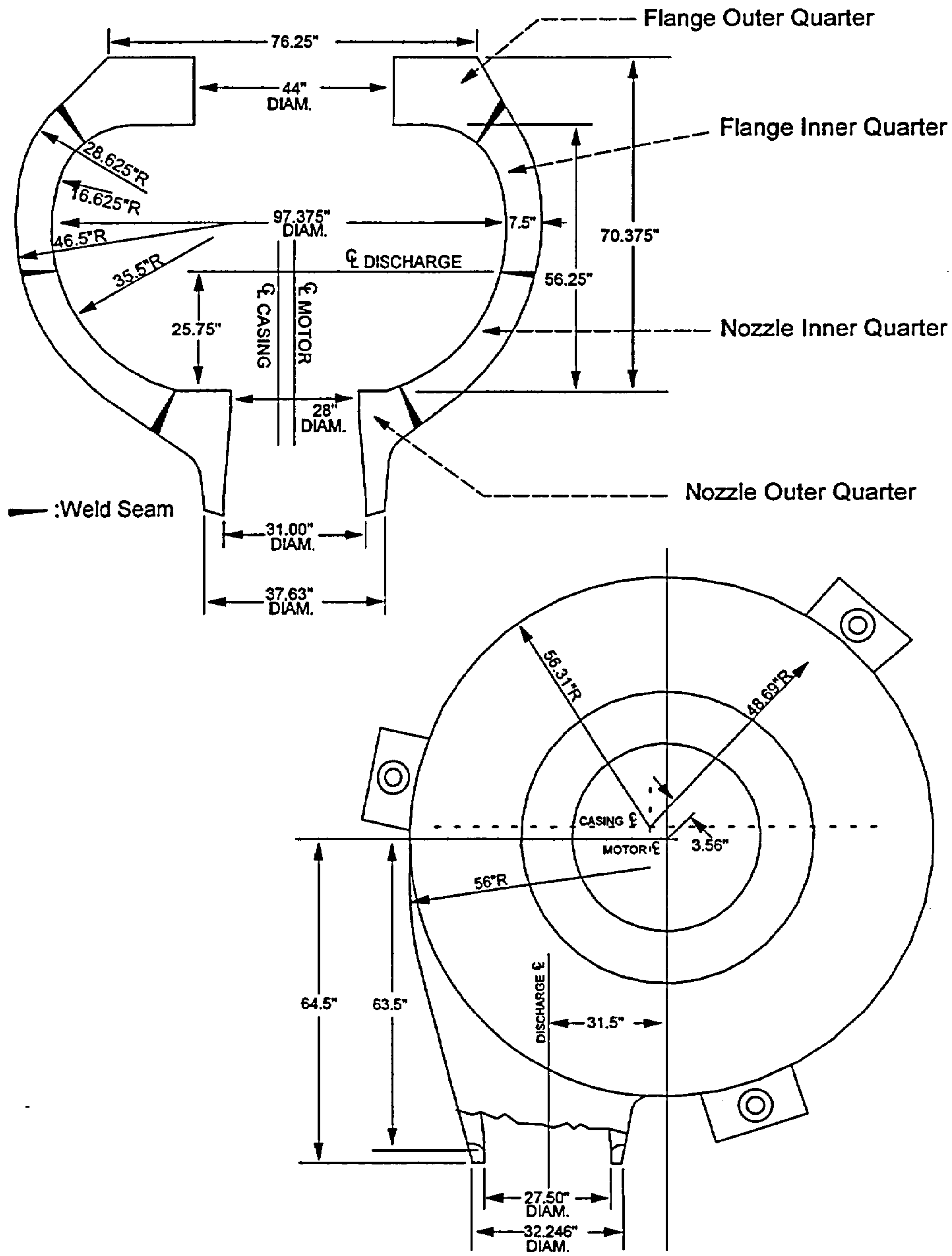

Figure 2-1 Dimensional Sketch of a Typical Model 93 Pump Casing with the Weld Seams Identified

3 LOADS ON THE PUMP CASING NOZZLES

In WCAP-13045 enveloping axial force and moment loadings on the inlet and outlet pump casing nozzles were applied in the three-dimensional finite element analyses of the WOG plant pump casings (Reference 2).

Normal Loads

The normal operating loads are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \qquad (3\text{-}1)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} + (M_Y)_P \qquad (3\text{-}2)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} + (M_Z)_P \qquad (3\text{-}3)$$

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.

Faulted Loads

The faulted loads are calculated by the absolute sum of loading components. The absolute summation of loads are shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSE}| \qquad (3\text{-}4)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_P| + |(M_Y)_{SSE}| \qquad (3\text{-}5)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_P| + |(M_Z)_{SSE}| \qquad (3\text{-}6)$$

where subscript SSE means Safe Shutdown Earthquake.

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

$$M = \sqrt{M_Y^2 + M_Z^2} \qquad (3\text{-}7)$$

where

M = bending moment for required loading

M_Y = Y component of bending moment

M_Z = Z component of bending moment

F = axial force

NOTE: X axis is along the centerline of pipe

Summary and Comparison of Loads

The faulted nozzles loads (i.e., the normal plus safe shutdown earthquake nozzle loads) for the H. B. Robinson Unit 2 pump casings are compared with the screening (i.e., enveloping) faulted loads. The normal loads are compared with screening normal loads for evaluating the loss-of-load condition. The normal and faulted loads utilized here are based on the information shown in Reference 10.

In Table 3-1 the normal operating loads obtained, as mentioned above, for H. B. Robinson Unit 2 are compared with the Level C screening nozzle loads (see Table 6-2 of WCAP-13045) which were used for evaluating the loss-of-load upset condition. The H. B. Robinson Unit 2 normal moment at the inlet nozzle is seen to be bounded by the corresponding Level C screening moment. The H. B. Robinson Unit 2 normal force at the inlet nozzle is not bounded by the corresponding Level C force. The H. B. Robinson Unit 2 normal moment at the outlet nozzle is seen to be bounded by the corresponding level screening moment. The H. B. Robinson Unit 2 normal force at outlet nozzle is not bounded by the screening loads. Also for normal case pressure of 2755 psia for H. B. Robinson Unit 2 is higher than the screening pressure of 2635 psig used in Reference 2 (See table 6-2 of WCAP-13045). Additional analyses were performed for the normal condition as shown in Section 5.0.

The H. B. Robinson Unit 2 faulted loads determined, as mentioned above, are compared in Table 3-2 to the Level A screening loads as defined in WCAP-13045 (see Table 6-2 of WCAP-13045). The H. B. Robinson Unit 2 plant faulted force and moments at the inlet and outlet nozzles are bounded by the faulted screening loads. No additional analysis is required for the Level A loading case. Analysis performed in Reference 2, is conservatively applicable for the H. B. Robinson Unit 2 Plant. The faulted temperature for the WCAP-13045 generic analysis is 550°F and the faulted temperature for H. B. Robinson is 554°F. The difference between the 550°F and 554°F Yield strength is 0.26% which is negligible (the Ultimate strength remains the same for 550°F and 554°F) and also there are plenty of margins available for loads and this difference of 4°F will have insignificant impact on the results and therefore is acceptable

Temperature and Pressure

Faulted Case

Temperature = 554°F (used 550°F for the evaluation)

Pressure = 2250 psia

Loss-of-Load (LOL) Case (Reference 3)

Temperature = 598°F

Pressure = 2755 psia

Table 3-1 Comparison of the Normal Loads for the Pump Casing Nozzles of H. B. Robinson Unit 2 with the Screening Level C, Normal Loads

Load	Temperature (°F)	Inlet Nozzle		Outlet Nozzle	
		Force (kips)	Moment (in-kips)	Force (kips)	Moment (in-kips)
H. B. Robinson Unit 2	598	2173	7461	1645	7524
Screening Level C	590	1900	23000	1400	8000

Note: As explained in WCAP-13045, the enveloping stresses were determined for the loss-of-load transient. This was conservatively assumed as the limiting Level C transient.

Table 3-2 Comparison of the Faulted Loads for the Pump Casing Nozzles of H. B. Robinson Unit 2 with the Level A, Faulted Screening Loads

Load	Temperature (°F)	Inlet Nozzle		Outlet Nozzle	
		Force (kips)	Moment (in-kips)	Force (kips)	Moment (in-kips)
H. B. Robinson Unit 2	554	1824	14809	1387	12469
Screening Level A	550	2000	40000	1800	20000

4 MATERIAL CHARACTERIZATION

4.1 TENSILE PROPERTIES

The ASME Code material tensile properties were conservatively used for H. B. Robinson Unit 2 to establish the tensile properties for the fracture mechanics analyses.

For the H. B. Robinson Unit 2, the properties at 550°F and 598°F were required for the analyses. The lower bound properties at 550°F and 598°F were established from the tensile properties of the Section III 1989 ASME Boiler and Pressure Vessel Code (Reference 4). Code tensile properties at 550°F and 598°F were obtained by interpolating between the 500°F and 600°F tensile properties.

The lower bound yield strengths and ultimate strengths at operating temperatures are given in Table 4-1. Modulus of elasticity values obtained from Reference 4 at 550°F and 598°F are also shown in Table 4-1. Poisson's Ratio used is 0.30.

For fracture evaluations the true stress-true strain curves for SA351 CF8 at the temperature of interest must be available. These curves were obtained from the Westinghouse tensile property database by using the information shown in Table 4-1. The lower bound true stress-true strain curves are given in Figures 4-1 and 4-2.

4.2 FRACTURE TOUGHNESS PROPERTIES AND CRITERIA FOR THE PUMP CASINGS

The H. B. Robinson Unit 2 pump casings are fabricated from SA351 CF8. This material has 304 stainless steel chemistry and is not extremely susceptible to thermal aging degradation. Values for the chemistry of each heat of material used in fabricating the H. B. Robinson Unit 2 pump casings are taken from the information shown Appendix A of WCAP-13045 (Reference 2). Predictions for fracture toughness values are based on the material chemistry content.

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]a,c,e

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 12 and 13).

ANL developed the fracture toughness estimation procedures by correlating data in the data base 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 13 was used to calculate the fracture toughness values for this analysis. ANL research program was sponsored and the procedure was accepted (Reference 14) by the NRC.

The chemical compositions are available from CMTRs and are provided in Table 4-1. The following equations are taken from Reference 13.

$$Cr_{eq} = Cr + 1.21(Mo) + 0.48(Si) - 4.99 = \text{chromium equivalent} \tag{4-1}$$

$$Ni_{eq} = Ni + 0.11(Mn) - 0.0086(Mn)^2 + 18.4(N) + 24.5(C) + 2.77 \tag{4-2}$$

Ni_{eq} = nickel equivalent

$$\delta_c = 100.3(Cr_{eq} / Ni_{eq})^2 - 170.72(Cr_{eq} / Ni_{eq}) + 74.22 \tag{4-3}$$

where the elements are in percent weight and δ_c is ferrite in percent volume.

The saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10} C_{Vsat} = 1.15 + 1.36 \exp(-0.035\phi) \tag{4-4}$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Cr + Si)(C + 0.4N) \tag{4-5}$$

and from

$$\begin{aligned} \log_{10} C_{Vsat} = {} & 5.64 - 0.006\delta_c - 0.185Cr + 0.273Mo - 0.204Si \\ & + 0.044Ni - 2.12(C + 0.4N) \end{aligned} \tag{4-6}$$

The saturation room temperature (RT) impact energies of the cast stainless steel materials were determined from the chemical compositions available from CMTRs and provided in Table 4-1. The saturation J-R curve at 290°C (554°F), for static-cast CF8 steel is given by

$$J_d = 102 (C_{Vsat})^{0.28} (\Delta a)^n \tag{4-7}$$

$$n = 0.21 + 0.09 \log_{10} (C_{Vsat}) \tag{4-8}$$

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

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]a,c,e

[

]a,c,e

a,c,e

N is assumed as 0.05

[1] from Equation 4-4
[2] from Equation 4-6
[3] minimum of items Cv_{sat}^1 and Cv_{sat}^2

a,c,e

Figure 4-1 Representative Lower Bound True Stress - True Strain Curve for SA351 CF8 at 550°F for H. B. Robinson Unit 2

a,c,e

Figure 4-2 Representative Lower Bound True Stress - True Strain Curve for SA351 CF8 at 598°F for H. B. Robinson Unit 2

5 STABILITY EVALUATIONS

5.1 SELECTION OF LOCATIONS FOR POSTULATED QUARTER THICKNESS CRACKS

In this section selection of flaw locations for evaluation per ASME Code Case N-481 are described. Three criteria for flaw location selections are applied as follows:

1. for each weld, a flaw will be located in the highest stressed region;
2. flaws will be located in regions of significant stress concentrations;
3. flaws will be located in welds not affected by discontinuities such as nozzles.

The selection of quarter thickness flaw locations and related information are given below for the finite element model.

5.2 FLAW LOCATIONS FOR THE MODEL 93 PUMP CASING

Seven locations were selected for postulating quarter thickness flaws in the Model 93 pump casing. The flaws so selected have a 6 to 1 aspect ratio as required by ASME Code Case N-481 with one exception. The exception is the flaw selected at the outlet nozzle knuckle. An aspect ratio is not defined but the crack front curvature is representative of the crack front curvature for a crack having a 6 to 1 aspect ratio. Also for the outlet nozzle knuckle, the depth of the crack is taken as one-fourth the nominal casing wall thickness, not one-fourth the distance from the nozzle knuckle to the nozzle crotch.

The seven flaws are identified in Figure 5-1 of this report and the detailed description is given in Table 9-1 of Reference 2. Three of the postulated flaws are on the outside surface. Three locations were selected based on high stresses in weld regions. Four were selected based on high stress concentrations and one selection was a nominally stressed weld location. One location (5-93) was selected for two reasons – high stresses in a weld and highest stressed location in the pump casing.

5.3 THE FINITE ELEMENT STRESS ANALYSIS MODELS OF THE PUMP CASINGS

Detailed stress analyses for Model 93 was performed in Reference 2. A large three-dimensional (3D) finite element model, containing the inlet and outlet nozzles, was developed for the pump casing.

Details of the finite element model are given in Figures 7-1 through 7-4 of Reference 2. For the complete 3D model, 3D isoparametric brick elements (20 nodes) and 3D isoparametric wedge elements (15 nodes) were used. Pipe extensions were made to the nozzles to allow continuous remote loadings of the nozzles.

For the Model 93 pump casing there are 1643 elements and 8729 individual nodes. Exterior details of the model are shown in Figures 7-1 and 7-2 of Reference 2. The coordinate directions are noted on Figure 7-1 (Reference 2). The support lugs are fully developed. The head is attached to the casing by a ring of small elements simulating the ring of bolts as indicated in Figure 7-1 (Reference 2). The pump shaft hole was introduced to add flexibility to the head. An interior half view containing the outlet nozzle is shown in Figure 7-3 (Reference 2). Various sections of the outlet nozzle region are shown in Figure 7-4 (Reference 2).

Using the finite element stress analysis results from Reference 2, plant specific through-wall stresses for H. B. Robinson Unit 2 at flaw locations were calculated for the normal loading condition.

5.4 STABILITY ANALYSIS

In Section 3.0, it was determined that the normal forces at the inlet and outlet nozzles of the H. B. Robinson Unit 2 are not bounded by the respective screening loads of WCAP-13045. Consequently, in this section, an analysis is conducted to determine whether enough margin is available to allow the H. B. Robinson Unit 2 pump casings to meet the stability criteria.

As explained in Section 10.0 of WCAP-13045, a postulated flaw is stable if either:

1. $J_{applied} < J_{Ic}$ or
2. If $J_{applied} > J_{Ic}$ then

   $T_{applied} < T_{material}$ and $J_{applied} < J_{max}$

The limiting material toughness values are listed in Section 4.0 of this report. They are:

[

]a,c,e

In WCAP-13045, $J_{applied}$ and $T_{applied}$ values are calculated using the screening loads and the minimum ASME Code mechanical properties for SA351 CF8 stainless steel. To determine if the flaw stability criteria is met for the H. B. Robinson Unit 2 pump casings for the normal conditions, these parameters can be recalculated using the H. B. Robinson Unit 2 loads and material properties. This evaluation is conducted at the critical locations described in WCAP-13045. These locations were 1-93, 2-93, 3-93, 4-93, 5-93, 6-93, and 7-93 and are shown in Figure 5-1.

While there are extensive solutions for surface flaws in structures (e.g., Reference 5) assuming linear elastic behavior, there is no procedure comparable to that of the EPRI fracture mechanics handbook (Reference 6) available for elastic-plastic considerations. There are stresses in pump casings well in excess of yield stress; thus EPFM procedures are necessary. Such solutions are developed in three steps as discussed below.

As a first step, a linear elastic fracture mechanics (LEFM) solution is obtained for a quarter thickness flaw having a six-to-one aspect ratio using the methodology of References 5 or other available solutions, as appropriate. For all but the nozzle knuckle the through-wall stress distribution is used and single curvature is accounted for (Reference 5). Actually this reduces the analysis to that of a cylinder. For the nozzle knuckle, the LEFM solution of Reference 7 applies. The knuckle-to-crotch stress distribution is used. As a second step, the stress intensity factor, K_I, so obtained as above, is then evaluated for elastic-plastic behavior in the following manner. An LEFM solution is obtained for a cylinder with the same dimensions as in the first step with a quarter thickness internal surface continuous circumferential flaw subjected to a constant stress loading. By interpolation, the constant stress level is then determined which produces the same K_I as in the first step. The stress so determined is called the equivalent stress. As a final step, J_{app} and T_{app} are found using the material curves (Figures 4-1 and 4-2) and properties given in Table 4-1 in conjunction with an EPFM model of the cylinder of the second step. That is, the cylinder is subjected to the equivalent stress using the EPFM solutions developed in Reference 6.

The temperature and dimensions associated with the postulated cracks are summarized in Table 5-1 for H. B. Robinson Unit 2. The equivalent stresses shown in Table 5-1 for loading level C are associated with the H. B. Robinson Unit 2 loading conditions. The equivalent stresses shown in Table 5-1 for loading Case A (faulted condition) are taken from Reference 2 and they are conservative for H. B. Robinson Unit 2.

Table 5-2 shows stability results for H. B. Robinson Unit 2. As shown in Table 5-2 all the stability criteria are met. Therefore, it is concluded that flaws postulated in the H. B. Robinson Unit 2 pump casing per Code Case N-481, when subject to the normal and faulted loadings are determined to be stable.

Table 5-1 Dimensions and Equivalent Stresses Associated with the Postulated Flaws in the Model 93 Pump Casings for the H. B. Robinson Unit 2 Plant

a,c,e

[1] Normal Loads

[2] Faulted Load

Table 5-2 Stability Results for the Model 93 Pump Casings of H. B. Robinson Unit 2 Plant

a,c,e

[1] Not Applicable, $J_{app} < J_{IC}$

a,c,e

Figure 5-1 Location of Flaws Postulated in the Pump Casing

6 FATIGUE CRACK GROWTH ASSESSMENT

6.1 INTRODUCTION

In the stability analyses presented in the Section 5, cracks are postulated at various locations in the pump casings. Such postulated cracks would be subject to the various cyclic conditions the pump casing experience. Thus, the sensitivity to cyclic loadings of postulated cracks in the pump casings was evaluated as a generic fatigue crack growth analysis for pump casing Model 93 in Section 12 of Reference 2.

The highest stressed location was chosen for the fatigue crack growth. This region is at Flaw 5-93. The postulated flaws are at the outlet nozzle knuckle in the plane of the weld. The stress contours for Level A loads are given in Figure 8-8 of Reference 2 and typify this location.

The generic transients considered for the fatigue crack growth are given in Table 12-2 of Reference 2. The fatigue crack growth results taken from Table 12-2 of Reference 2 are also given in Table 6-1. The maximum acceptable flaw size (0.30-in.) is seen to increase by less than [

]a,c,e

6.2 DISCUSSION AND CONCLUSIONS

The highest stressed location in the Models 93-pump casing has been evaluated for fatigue crack growth. The crack growth observed is bounding for other less severely stressed locations. For 40 year plant life, postulated crack depths initially well in excess of the maximum ASME code allowable remain well less than the flaw sizes shown to be stable in Section 5.

It is concluded that any reasonably sized flaws in the pump casings will exhibit only minimal crack extension during service life of 40 years, such flaws remaining well below the flaw sizes shown to be stable.

The transients and cycles of H. B. Robinson Unit 2 for 60 year plant life are the same as those of 40 year plant life and therefore fatigue crack growth results shown in Table 6-1 for 40-year are also applicable for 60-year.

Table 6-1 Fatigue Crack Growth for Postulated Flaws in the Outlet Nozzle Knuckle Region of the Model 93 Pump Casings

Initial Crack Depth (in.)	Crack Depth (in.) at End of Year			
	10	20	30	40 and 60
0.30[1]	0.32	0.34	0.37	0.40
0.50	0.53	0.56	0.61	0.66
0.80	0.85	0.89	0.94	0.99

1 The maximum acceptable depth of a flaw per Table IWB 3518-2 of Section XI of the ASME Code (1989 Edition).

7 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

7.1 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 900 reactor-years, including five plants each having over 20 years of operation and 15 other plants each with 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 8) 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 9) 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 failure in the reactor coolant system loop. The discussion below further qualifies the PCSG's findings.

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Since

some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as other material in the system, applicable ASME Code rules, fracture toughness, welding, fabrication, and processing.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are: oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications. Requirements on chlorides, fluorides, conductivity, and Ph are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is expected to be in the ppb range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. Thus during plant operation, the likelihood of stress corrosion cracking is minimized.

7.2 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; pressurizer heaters and pressurizer spray also within a narrow range for steady-state conditions control pressure. 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.

7.3 LOW CYCLE AND HIGH CYCLE FATIGUE

An assessment of the low cycle fatigue loadings was carried out as part of this study in the form of a fatigue crack growth assessment, as discussed in Section 6.0.

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. 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.

8 DISCUSSION AND CONCLUSIONS

This report provides an assessment of the primary loop pump casings of H. B. Robinson Unit 2 to the conditions of Item (d) of ASME Code Case N-481 (see Appendix A).

This evaluation considers actual H. B. Robinson Unit 2 fracture toughness values. Thus Item (d) (1) is satisfied.

Stress analyses of a representative primary loop pump casing are presented in WCAP-13045. (See also a description in Section 5.2 of this report.) This satisfies Item (d) (2).

The operating history of Westinghouse design primary loop pumps is reviewed in Section 7. This satisfies Item (d) (3). Flaws are postulated in the pump casings as described in Section 5.1 (also see Section 9.0 of WCAP-13045) satisfying Item (d) (4). One-quarter thickness reference flaws with a six-to-one aspect ratio are postulated consistent with Item (d) (5).

Comparisons of the loads of the H. B. Robinson Unit 2 pump casings with the screening loads of WCAP-13045 are presented in this report. The stability of the flaws postulated in the H. B. Robinson Unit 2 primary loop pump casings are established by evaluating the resulting $J_{applied}$ and $T_{applied}$ against the fracture toughness values noted in the discussion of Item (d) (1) (See Section 5.0). This satisfies Item (d) (6).

The preservice fracture toughness of cast stainless steels is very high. Thermal aging causes a reduction in the toughness. The effect of thermal aging has been evaluated in Section 4.0 of this report and Appendix A of WCAP-13045. No other mechanism is known to degrade the properties of the pump casings during service. Item (d) (7) is so satisfied.

It is concluded that the primary loop pump casings of H. B. Robinson Unit 2 are in compliance with Item (d) of ASME Code Case N-481 for 40-year and 60-year (as a part of the License Renewal program) plant life.

9 REFERENCES

1. Case N-481: "Alternate Examination Requirements for Cast Austenitic Pump Casings," Section XI, Division 1, Cases of ASME Boiler and Pressure Vessel Code, Approval Date: March 5, 1990.

2. F. J. Witt and J. F. Petsche, Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems, WCAP-13045, September 1991 (Westinghouse Proprietary Class 2).

3. Equipment Specification 676429 Revision 0 dated June 4, 1967, H. B. Robinson Unit 2 Reactor Coolant Controlled Leakage Pump.

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

5. Raju, I. S. and Newman, J. C., "Stress Intensity Factor Influence Coefficients for Internal and External Surface Cracks in Cylindrical Vessels," in Aspects of Fracture Mechanics in Pressure Vessels and Piping, ASME publication PVP. Vol. 58, 1982.

6. 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. EPRI-NP-719-SR, August 1978 and Errata for this report issued April 14, 1980.

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

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

10. Letter EDRE-SMT-00-018, " H. B. Robinson Reactor Coolant Pump Nozzle Loads," Dated February 7, 2000.

11. WCAP-7211, Revision 4, "Proprietary Information and Intellectual Proprietary Management Policies and procedures," January 2001.

12. 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.

REFERENCES (Cont'd)

13. 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.

14. "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.

APPENDIX A
CASES OF ASME BOILER AND PRESSURE VESSEL CODE

Approval Date: March 5, 1990
See Numerical Index for expiration and any reaffirmation dates.

Case N-481
Alternate Examination Requirements for Cast Austenitic Pump Casings
Section XI, Division 1

Inquiry: When conducting examination of cast austenitic pump casings in accordance with Section XI, Division 1, what examinations may be performed in lieu of the volumetric examinations specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

Reply: It is the opinion of the Committee that the following requirements shall be met in lieu of performing the volumetric examination specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

(a) Perform a VT-2 visual examination of the exterior of all pumps during the hydrostatic pressure test required by Table IWB-2500-1, Category B-P.

(b) Perform a VT-1 visual examination of the external surfaces of the weld of one pump casing.

(c) Perform a VT-3 visual examination of the internal surfaces whenever a pump is disassembled for maintenance.

(d) Perform an evaluation to demonstrate the safety and serviceability of the pump casing. The evaluation shall include the following:

(1) evaluating material properties, including fracture toughness values;

(2) performing a stress analysis of the pump casing;

(3) reviewing the operating history of the pump;

(4) selecting locations for postulating flaws;

(5) postulating one-quarter thickness reference flaw with a length six times its depth;

(6) establishing the stability of the selected flaw under the governing stress conditions;

(7) considering thermal aging embrittlement and any other processes that may degrade the properties of the pump casing during service.

(e) A report of this evaluation shall be submitted to the regulatory and enforcement authorities having jurisdiction at the plant site for review.