WESTINGHOUSE CLASS 3 (Non-Proprietary)

Westinghouse Energy Systems

Westinghouse Owners Group

OG-93-16

February 25, 1993

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Robert A. Hermann, Section Chief
Materials and Chemistry Engineering Branch
Office of Nuclear Reactor Regulation

Subject: Westinghouse Owners Group
Transmittal of Topical Report: WCAP-13565 Revision 1, Non-Proprietary, Entitled "Alloy 600 Reactor Vessel Head Adaptor Tube Cracking Safety Evaluation"

Dear Mr. Hermann:

Enclosed are ten (10) copies of WCAP-13565, Revision 1, entitled "Alloy 600 Reactor Vessel Head Adaptor Tube Cracking Safety Evaluation" (Non-Proprietary).

As with the original report, this revised report is provided for your information and use based upon a request made by the NRC during our joint meeting on this topic on November 20, 1992.

The safety conclusion provided in this document remains the same as Revision 0. This revision was necessary to incorporate the results of additional stress analysis work performed.

Very truly yours,

Lawrence A. Walsh

Lawrence A. Walsh, Chairman
Westinghouse Owners Group

/dac

enclosure

cc: Steering Committee (1L)
C.K. McCoy, Georgia Power (1L)
N.J. Liparulo (1L)
K.J. Voytell (1L)

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WESTINGHOUSE CLASS 3 (Non-Proprietary)

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WESTINGHOUSE CLASS 3 (Non-Proprietary)

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WCAP-13565
REV. 1

WESTINGHOUSE CLASS 3

ALLOY 600 REACTOR VESSEL HEAD ADAPTOR TUBE CRACKING SAFETY EVALUATION

February 1993

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Division
P. O. Box 355
Pittsburgh, Pennsylvania 15230

GENCL NO. 92-011 Rev. 1

Customer Reference No(s).

Westinghouse Reference No(s).
MUHP-5016

WESTINGHOUSE NUCLEAR SAFETY
GENERIC SAFETY EVALUATION CHECK LIST (GENCL)

1.) NUCLEAR PLANT(S): Westinghouse NSSS Plants

2.) SUBJECT (TITLE): Alloy 600 Reactor Vessel Head Adaptor Tube Cracking

3.) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59(b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10CFR50.59(a)(1)

3.1) Yes __ No X A change to the plant as described in the FSAR?
3.2) Yes __ No X A change to procedures as described in the FSAR?
3.3) Yes __ No X A test or experiment not described in the FSAR?
3.4) Yes __ No X A change to the plant technical specifications? (See Note on Page 2.)

4.) CHECK LIST - PART B - 10CFR50.59(a)(2) (Justification for Part B answers must be included on page 2.)

4.1) Yes __ No X Will the probability of an accident previously evaluated in the FSAR be increased?
4.2) Yes __ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
4.3) Yes __ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?
4.4) Yes __ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.5) Yes __ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.6) Yes __ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
4.7) Yes __ No X Will the margin of safety as described in the bases to any technical specification be reduced?

NOTES:

If the answer to any of the above questions is unknown, indicate under 5.) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 10CFR50.90.

5.) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

FOR FSAR UPDATE

Section: ________ Pages: ______ Tables: ________ Figures: _______

No FSAR Update Required

SAFETY EVALUATION APPROVAL LADDER:

Nuclear Safety Preparer:	J. S. Galembush	Date: 2/9/93
Nuclear Safety Verifier:	Gary O. Barrett G. O. Barrett	Date: 2/9/93
Nuclear Safety Group Manager:	H. A. Sepp	Date: 2/10/93

TABLE OF CONTENTS

SAFETY EVALUATION
POTENTIAL REACTOR VESSEL HEAD ADAPTOR TUBE CRACKING
WESTINGHOUSE NSSS PLANTS

1.0 BACKGROUND

In late September 1991 Westinghouse was informed by Electricite de France (EdF) of the discovery of a leaking reactor vessel head adapter penetration (Figure 3.4-1) at the Bugey 3 plant in France. Bugey 3 has been in commercial operation since 1979. The leak was found during a hydrotest associated with a 10 year in-service-inspection (ISI). The hydrotest was performed at approximately 3000 psi and 194 degrees F. The leak was discovered at core location H-14 (Penetration # 54), a peripheral full length CRDM location. The leak was located by using microphones attached to both the top and bottom heads of the reactor vessel. EdF determined that the leak rate was approximately 0.70 l/hr (0.003 gpm).

A visual examination performed at that time indicated the presence of longitudinal (axial) cracks in the I.D. of the head adaptor tube. The head adaptor tubes are manufactured from Alloy 600 material. The use of Alloy 600 material for the head adaptor tubes is common to both Framatome and Westinghouse plants.

A subsequent inspection of all 65 head adaptor tubes at Bugey 3 revealed axial cracks at two peripheral head adaptor locations. After finding the leak at Bugey 3, EdF performed examinations at two additional plants. Examination of 24 penetrations at Bugey 4 revealed axial cracks at eight peripheral head adaptor locations. Twenty-six penetrations were inspected at Fessenheim 1. This examination revealed axial cracking in one head adaptor. Based upon these subsequent inspections, EdF undertook an inspection program which encompassed all of their operating plants. To date, over 500 reactor vessel head adaptor penetrations have been inspected encompassing thirteen (13) European plants. Of these inspections, 26 penetrations have exhibited crack indications. Plant inspections are continuing in Europe.

At Bugey 3, EdF removed the penetration corresponding to core location H-14 for hot cell examination / root cause determination. The mechanism of the degradation (root cause) was identified by EdF as primary water stress corrosion cracking (PWSCC). Westinghouse has reviewed the available metallographic records and concurs with this conclusion. The weld-induced bending and ovality of the peripheral penetrations appears to be the initiating source of the stress which is promoting the degradation.

The purpose of this safety evaluation is to assess the continued safe operation of Westinghouse designed NSSS plants focusing on the likelihood of cracking, the characterization of any such potential cracking, the potential for leakage, and finally, the disposition of low alloy carbon steel wastage issues; in the knowledge that similarities do exist in the various plant designs between Westinghouse and European manufacturers.

This safety evaluation will provide the following elements:

1. A summary of the stress analysis focusing on the type of cracking that may be expected in the Alloy 600 material, and the stresses necessary for crack propagation.

2. A summary of the crack propagation analysis will be provided along with the background of the crack prediction method.

3. An assessment will be made of the Westinghouse Owners Group (WOG) plants with respect to penetration crack indication data from plant inspections at Ringhals, Beznau and various EdF plants. The key parameters for cracking will be compared against WOG plants.

4. A leakage assessment will be provided summarizing leak rate vs. crack size, and in postulating leaks for those few WOG plants for which leakage considerations may apply.

5. A vessel head wastage assessment will assess the process by which wastage may potentially occur and an estimate of allowable wastage will be provided.

2.0 LICENSING BASIS

The situation regarding the potential cracking of reactor vessel head adaptor tubes at Westinghouse designed NSSS plants represents a change to the normal plant configuration. Title 10 of the Code of Federal Regulations, Section 50.59 (10 CFR 50.59) allows the holder of a license authorizing operation of a nuclear power facility the capacity to evaluate these types of situations. Prior Nuclear Regulatory Commission (NRC) approval is not required to return the plant to power as long as the situation does not involve an unreviewed safety question or result in a change to the plant technical specifications incorporated in the license. It is, however, the obligation of the licensee to maintain a record of the change or modification to the facility, as a result of any given situation, to the extent that such a change impacts the FSAR. While this situation does not represent a change to the FSAR, 10CFR 50.59 further stipulates that these records shall include a written safety evaluation which provides the basis for the determination that the situation does not involve an unreviewed safety question. It is the purpose of this document to support the requirement for a written safety evaluation.

The scope of this document is limited to an evaluation of the potential cracking in reactor vessel head adaptor tubes centering on any effects this situation may have on existing plant equipment or any unreviewed safety questions that may be identified.

3.0 EVALUATION

3.1 PENETRATION STRESS ANALYSIS

Background

Initially, several 3D-elastic finite element analyses were performed to establish penetration stress magnitude and distribution. These analyses demonstrated that stresses caused by operational pressure and temperature (2250 psi and 600°F.) loads are not large enough to cause penetration tube ovality of the magnitude which has been measured in a number of plants (based on penetration I.D. diametral and profile measurements taken from irradiated and non-irradiated vessels). It was further determined, qualitatively, that the residual stresses in and near the weld region due to welding are significantly higher than those caused by operational loads. It was also determined that stresses experienced due to the welding fabrication processes of attaching the penetration to the vessel head exceed the yield strength of the Alloy 600 weld and penetration material at some locations.

It therefore became essential to perform stress analyses considering the inelastic mechanical properties of the penetrations, to more quantitatively define the stress field in the penetration.

Approach

The additional analyses not only have to be detailed enough to provide quantitative stress distributions, but also must envelope all WOG plant penetrations. To make sure that the 4-loop models are enveloping, a parametric study was performed to study the effect of a) vessel size and b) penetration location. The results indicated that the outermost penetrations of the 4-loop plant, having the largest weld-offset angle among the 2, 3 and 4 loop plants, are the highest stressed penetrations under operating loads as well as having the largest residual stresses. Therefore, it was concluded that the outermost penetrations of the 4-loop plants are the enveloping penetrations of all WOG plants.

Having determined that the 4-loop plants are appropriate to represent all WOG plants, three penetration models were built using temperature dependent elastic-plastic material properties. Three different radial locations were modeled as described below:

1. the center location (#0)

2. outermost location (e.g. penetration #78), and

3. next to the outermost location (e.g. penetration #65).

In the Westinghouse 4-loop plant, penetrations #65 and #78 are located radially from the vessel centerline at 59.8 and 64.5 inches respectively.

The models utilized 3-dimensional isoparametric brick and wedge elements. Taking advantage of symmetry through the vessel and penetration centerlines only half the penetration geometry plus the surrounding vessel are modeled. These models are shown in Figures 2-1, 2-2 and 2-3 of Reference 6.

The penetration tube, weld metal and buttering were modeled as Alloy 600 and the vessel head shell as carbon steel. Elements with elastic-plastic capabilities were incorporated in the weld region and surrounding elements in both the penetration tube and vessel head shell. The stress-strain material properties of the elastic-plastic elements representing Alloy 600 were derived from test data obtained using an actual Alloy 600 penetration material sample taken from the outermost penetration of an unirradiated plant. At this elevation in the reactor vessel, material irradiation effects are considered to be negligible. The curve used was a half-life cycle stress-strain curve (Cyclic stress-strain curve at one-half of the life of the penetration). Use of the cyclic stress-strain curve is expected to provide conservative upper bound levels for stress estimates in the penetration. Existing monotonic stress strain curves were used for the carbon steel elements.

To simulate the stress history of the penetration tube the following loading sequence was applied to each of the three models described. The stresses caused by each of the load cycles are stored and maintained in the model before the next load cycle is applied to simulate the effect of residual stress. This provides for the accumulation of plastic stresses.

1. Thermal load from first weld pass.

2. Thermal load from second weld pass.

3. Fabrication Shop Cold Hydrotest (@ 3170 psi)

   (a) Cold hydro-test loading
   (b) Cold hydro-test unloading

4. Field / Site Hydrotest (@ 3170 psi)

   (a) Cold hydro-test loading
   (b) Cold hydro-test unloading

5. Steady state operational loading

Loadings

It was found from the analysis that the welding process introduces high residual stresses in the penetration tube near the partial penetration weld. The welding process was simulated by adding the weld material to the model in layers and subsequently specifying the stress-free reference temperatures for the weld and surrounding elements so as to provide shrinkage in the weld (due to cooldown). The reference temperatures were "benchmarked" or adjusted to generate ovality levels in the penetration tube approximating those measured in actual penetrations while still maintaining welding temperatures within reasonable limits. Two methods were tried, the first used two consecutive layers of welding, the second used three. The difference in the results between the two methods was insignificant and the two layer method adopted as the approach to be used.

The stress developed in the penetration model after applying the first weld pass (residual stress) was maintained as the initial stress as the elements of the second weld pass were applied. The stresses induced by the welding simulation were large enough to cause plastic deformation in weld region of the model.

The cold hydro-test loading required applying an internal pressure to the model of 3107 psi and performing the analysis at a temperature of 150°F. This combined with the cold hydro-test unloading (the unloading step is designed to return the model to an unloaded condition) step simulates the conditions during the fabrication shop hydro-test. The combination of the two weld passes and the shop hydro conditions produced permanent set ovalities in the penetration comparable to field measured values. A second cold hydro-test loading and unloading was used to model the field hydro-test.

Adding the steady state operational loadings (i.e. pressure of 2250 psi and temperature of 600°F.) brings the model to its final stress condition at operation. It has been determined through analysis that adding additional operational steady state pressure and temperature loadings and unloadings would not significantly alter the stress state of the penetration tube. The stresses calculated at this point of the analysis provide an upper envelope of the steady state operating conditions in all the operational Westinghouse plants.

Stress Analysis Results - Outermost Penetration

The outward displacement of the vessel and penetration are shown in Figure 2-4 of Reference 6 for steady state operation. Note that the vessel displaces radially, and the tube displaces toward a position perpendicular to the head. Also, the gap opens slightly between the penetration and the vessel, on the vessel center line side of the penetration, as shown in Figure 2-4 of Reference 6. These results were obtained for an initial interference fit of 0.0 inches as fabricated, which is the minimum case for Westinghouse plants. A higher interference value would add a compressive stress to the steady state stresses, therefore, the 0.0 inch interference is conservative. As input to the potential leakage and wastage assessment, an average annular radial gap of 0.003 in. during plant operation represents a conservative estimate.

The hoop stresses at the inside surface of the penetrations are shown in Figure 2-5 and 2-6 of Reference 6. The highest stresses are found in a zone around the weld. The peak stresses fall along lines nearest the center of the vessel (centerside), and the side furthest away from the center (180° away) (hillside). Note that these two locations correspond to the locations where axial cracks have been found in service.

The axial stress distribution at the inside surface is shown in Figure 2-5 and 2-6 of Reference 6, which also shows that the highest axial stresses are along the weld. Note that the magnitude of axial axial stresses are less than the hoop component of stress. No axial cracks have been found in any penetrations in service.

The degree of plastic deformation in the penetration can be seen in the color contour stress plots shown in Figure 2-6 of Reference 6. Here we see that the most significant plastic deformation occurs at and below the centerline of the weld, extending down the sides of the tube at the upper and lower hillside locations, similar to the pattern of hoop stress in Figure 2-5 (Ref. 6).

The stresses decrease significantly above the weld, as may be seen qualitatively in Figure 2-5 (Ref. 6). Since the extent of crack propagation above the weld is of interest, the stresses at the inner and outer surfaces of both the inner and outer hillside location have been evaluated. Plots of both the axial and hoop stresses as a function of distance from the bottom of the penetration are also shown in Figure 2-5 of Reference 6. Note that the weld location is shown on each plot, and also note that the weld location is different for the hillside and center side locations.

The highest hoop stress at the I.D. surface exceeds the axial stress at the same location by a factor of approximately 1.4, which in turn corresponds approximately to the ratio of hoop to axial stress ratio of 1.6 reported to be obtained by field measurements of an actual penetration. This finding strongly supports the contention that axial cracks are the most likely orientation to be expected, which is consistent with all reported inspection findings.

Intermediate Penetration

The intermediate penetration analyzed is on a radius of 59.8 in. from the vessel centerline (i.e. penetration No. 65 in a four loop plant). The stress analyses were carried out in precisely the same manner as the outermost penetration.

Figure 2-8 and 2-9 of Reference 6 provide plots of the hoop, axial and Von Mises stresses for the steady state operating conditions.

Center Penetration

The stress analysis carried out for the center penetration was performed in the same manner as the outermost penetration. The maximum hoop stresses, 34.6 ksi, are higher than the corresponding axial stresses, 21.4 ksi. The overall stress magnitudes are lower than those of the outer penetrations. The stress contours along the inside and outside surfaces are shown in Figures 2-10 and 2-11 of Reference 6.

Bolt-Up Effects

An additional 2-dimensional, axisymmetric model was developed and used to determine the stress contribution in the closure head shell, due to bolt-up loads. Bolt-up loads are generated in the closure head shell by the rotation of the flanges caused by the tightening of the stud tensioners. The analysis evaluated the change in the stresses at a radial distance of 64.5 inches (penetration location #78) from the vessel centerline through the thickness of the shell. The results of the analysis indicate that the bolt-up stresses are negligible compared to the operational stresses. This analysis also demonstrates that the simple boundary conditions (symmetry) used in the 3-dimensional, elastic-plastic analysis provide suitable constraints to the problem.

Likelihood of Circumferential Cracks

The hoop stress is the dominant stress in all the head penetration locations attached by an angular weld, which leads to the conclusion that cracks will be oriented axially. This is consistent with the inspection findings from several plants. The hoop stress at the inside surface of the outermost penetrations exceeds the axial stress by a factor of approximately 1.4, which is approximately the same ratio reported from actual measurements on head penetrations in service. In all of these cases there is little or no likelihood of circumferentially oriented flaws in any of the head penetrations.

Summary and Conclusions

The stress analyses have shown that the stresses in the head penetrations are a strong function of the weld offset, or the angle of intersection between the penetration and the head. In Westinghouse plants, the four loop units have the largest weld offset, and therefore the outermost penetration location was chosen for a complete stress analysis. These results will be conservative for all WOG two and three loop plants, which have the same penetration dimensions, but smaller offsets.

For penetrations attached by an angular weld, the hoop stresses are greater than the axial stresses, which is consistent with the inspection findings in that the flaws are axially oriented. Furthermore, the locations of the maximum hoop stress correspond with the locations where cracks have been observed.

3.2 CRACK GROWTH ANALYSIS: FLAW TOLERANCE

Introduction

The goal of this work was to provide a quantitative measure of the tolerance of the head penetrations for the presence of a flaw. The mode of crack extension is primary water stress corrosion cracking, and therefore the loading of interest is the steady state operating condition. The crack growth law used in this study was obtained from a survey of the available literature on this material and environment, with the temperature effect on the growth rates based on a collection of crack growth information from both laboratory and field data.

Approach and Results

The results of the three dimensional stress analysis of the outermost head penetrations were used directly in the flaw tolerance evaluation. The maximum stress is the hoop stress, but there is a component of axial stress which makes the maximum principle stress at a small angle to the hoop direction. Cracks would be expected to be oriented perpendicular to the maximum principle stress, which, based on the stress results would be at a slight angle to the axial direction. The flaws which have been found inservice are nearly all longitudinally oriented, thus the hoop stress component was used in the crack growth calculations. Stress analyses have shown that the penetration location with the highest stress is the 4-loop vessel head outermost penetrations, so this location was chosen for analysis.

The crack growth evaluation for the "part-through" flaws was based on the stress distribution through the penetration wall at the location which corresponds to the highest stress along the inner surface of the penetration. The highest stressed location was found to be in the immediate vicinity of the weld.

The results of the calculated growth through the wall for surface flaws postulated in the highest stress location of all the penetrations are summarized in Figure 3.2-1. (The highest stress location was found to be just below the weld on the center side of the outermost penetration.) Figure 3.2-1 applies to surface crack locations below and near the weld region, while cracks above the weld would grow more slowly because of the lower stresses. Note that the predicted extension through the penetration thickness requires a number of years for the entire range of operating temperatures, regardless of the location.

Figure 3.2-1 presents the predicted crack growth for a "through-wall" flaw postulated to exist below the weld region on the center side location. Although there are various levels of ovality (and therefore residual stress) in the various penetrations, in the vicinity of the weld and below it on the center side location the total stresses exceed the yield stress of the material. This was found to be the highest stress location on any of the penetrations, so the crack growth calculated will be conservative for all other locations.

To make these crack growth calculations, the average hoop stress through the wall of the penetration was used. As the flaw propagates past the weld, the stresses in the penetration decrease and become compressive in some cases, so the crack will slow down and nearly stop once it reaches this location. The bottom of the centerside portion of the weld is located 7.2 inches from the bottom of the penetration, and is 2.2 inches in width at this location, so a crack length exceeding 9.4 inches will have the upper tip of the crack above the weld. The results for propagation above the weld are also shown in Figure 3.2-2, and show the growth slowing considerably above the weld, nearly stopping before reaching 11 inches.

On the lower hillside location the average hoop stresses are only slightly lower than those on the center side, and therefore the predicted crack growth is slightly lower. These results are shown in Figure 3.2-3. Note that the weld is located 3 inches from the bottom of the penetration, as shown on the figure. Figure 3.2-3 shows the results for a postulated crack growing from below the weld, and again shows crack growth slowing considerably above the weld, this time nearly stopping at a height of about six inches, 1.5 inches above the weld.

Summary and Conclusions

An extensive evaluation has been carried out to characterize the loadings and stresses which exist in the head penetrations of Westinghouse plants. Three-dimensional finite element models were constructed, and all known loadings on the penetrations were analyzed. These loadings included internal pressure and thermal expansion, and embodied the conservative assumption that there is zero interference between the penetration and the head. In addition, residual stresses due to the welding of the penetrations to the vessel head (as evidenced by the observed ovality) were considered, using an elastic-plastic finite element analysis.

Results of the analyses reported here are consistent with the axial orientation and location of flaws which have been found in service in a number of plants, in that the largest stress component is the hoop stress, and the peak stresses were found to exist in the circumferential locations farthest away from the center of the vessel. The most important loading conditions were found to be those which exist on the penetration for the majority of the time, which are the steady state loading and the residual stresses.

These stresses are important because the cracking which has been observed to date in operating plants has been determined to result from primary water stress corrosion cracking (PWSCC). These stresses were used in fracture calculations to predict the future growth of flaws postulated to exist in the head penetrations. Crack growth laws were developed specifically for the range of operating temperatures of the head for Westinghouse plants, based on information from the literature as well as a compilation of crack growth results for operating plants.

The crack growth predictions discussed in previous sections show that the future growth of cracks which might be found in the penetrations will in general be very slow, in that a number of years will be required for any significant extensions. It is concluded, therefore, that it is conservative to assume that no "through-wall" crack will grow to a length longer than 2 in. above the penetration to vessel weld.

It is appropriate to examine the safety consequences of an indication which might be found below the weld. The indication, even if it were to propagate through the penetration wall, would have no consequence at all, since the pressure boundary would not be broken, unless it were to propagate above the weld.

Further propagation of the indication would not change its orientation, since the hoop stresses in the penetration remain larger than the axial stresses as a crack might move up the penetration. Therefore, it is extremely unlikely that the head penetration would be severed as a result of any indications.

Any indication is unlikely to propagate very far up the penetration above the weld, because the hoop stresses decrease in this direction, and this will cause it to slow down, and perhaps even to stop before it reaches the outside surface of the head. This result from the stress analysis supports the conclusion that it is extremely unlikely that leakage of any magnitude will occur.

The high likelihood that the indication will not propagate beyond the head ensures that no catastrophic failure of the head penetration will occur. This is because the indication will be enveloped in the head itself, which precludes the opening of the crack and limits leakage. In order to produce a failure of the head penetration, the flaw would have to extend over 13 inches above the head, an extremely unlikely event.

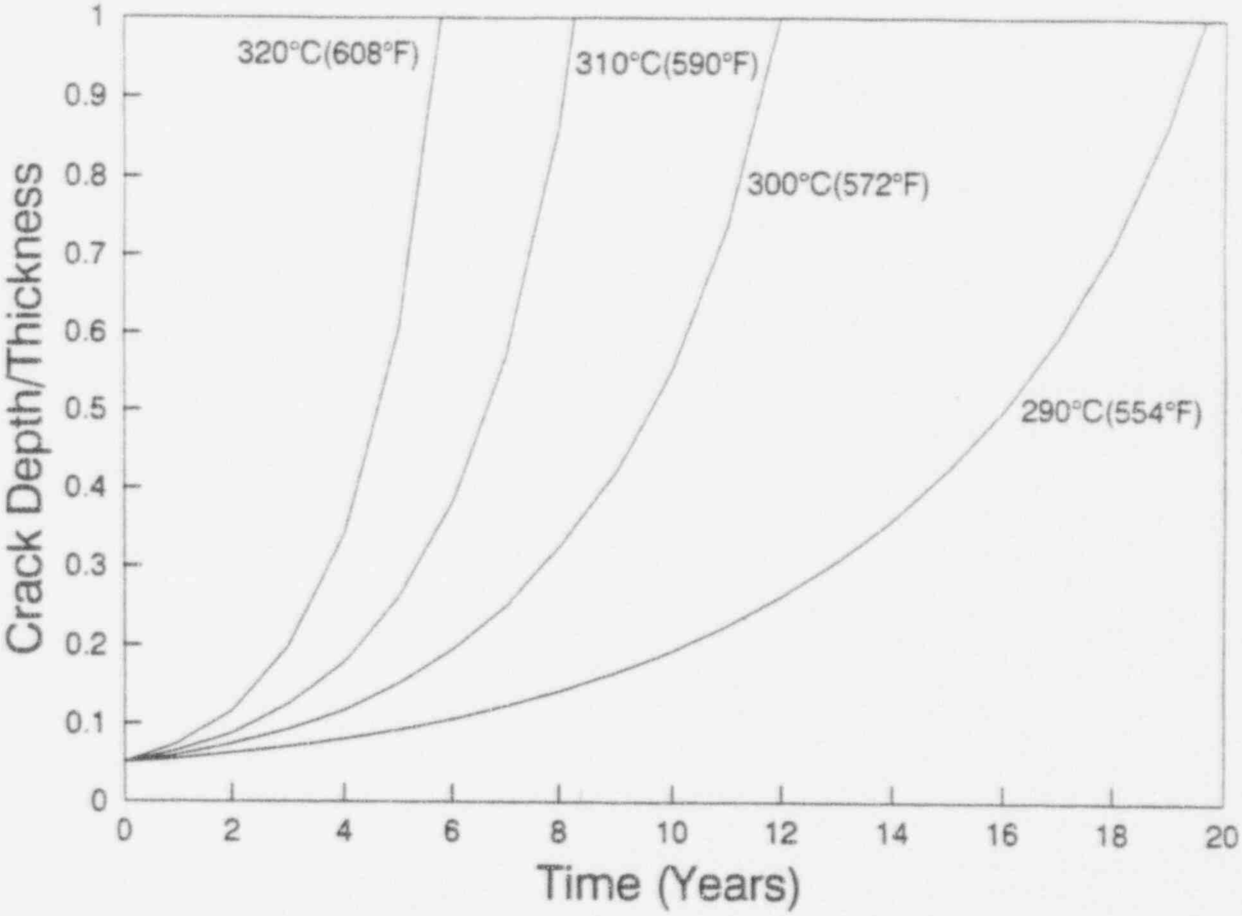


FIGURE 3.2-1

CRACK GROWTH PREDICTIONS FOR SURFACE FLAWS BELOW AND AT THE WELD REGION IN THE HEAD PENETRATIONS FOR A RANGE OF TEMPERATURES

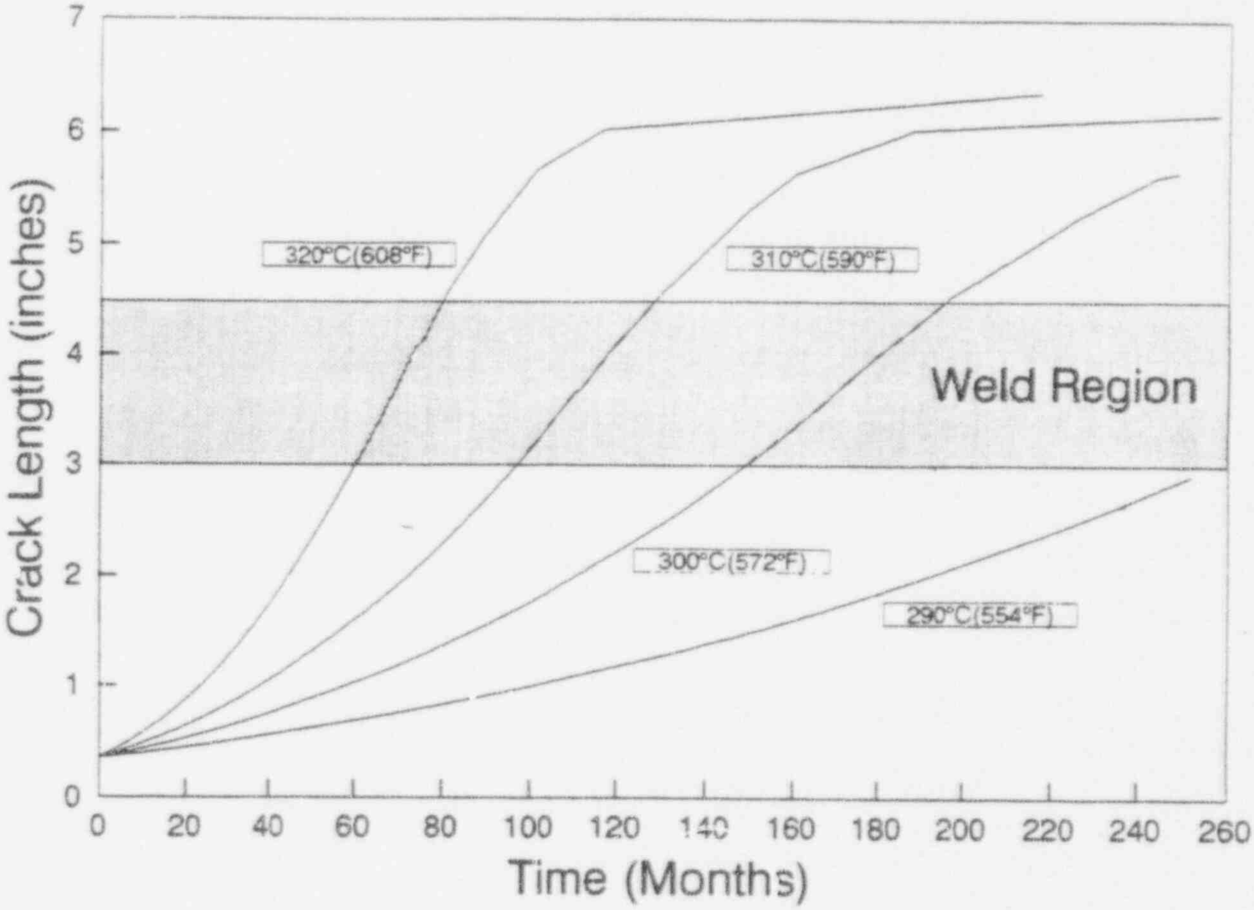


FIGURE 3.2-3

CRACK GROWTH PREDICTIONS FOR THROUGH-WALL FLAWS LOCATED ON THE LOWER HILLSIDE OF THE OUTERMOST HEAD PENETRATIONS, FOR A RANGE OF TEMPERATURES

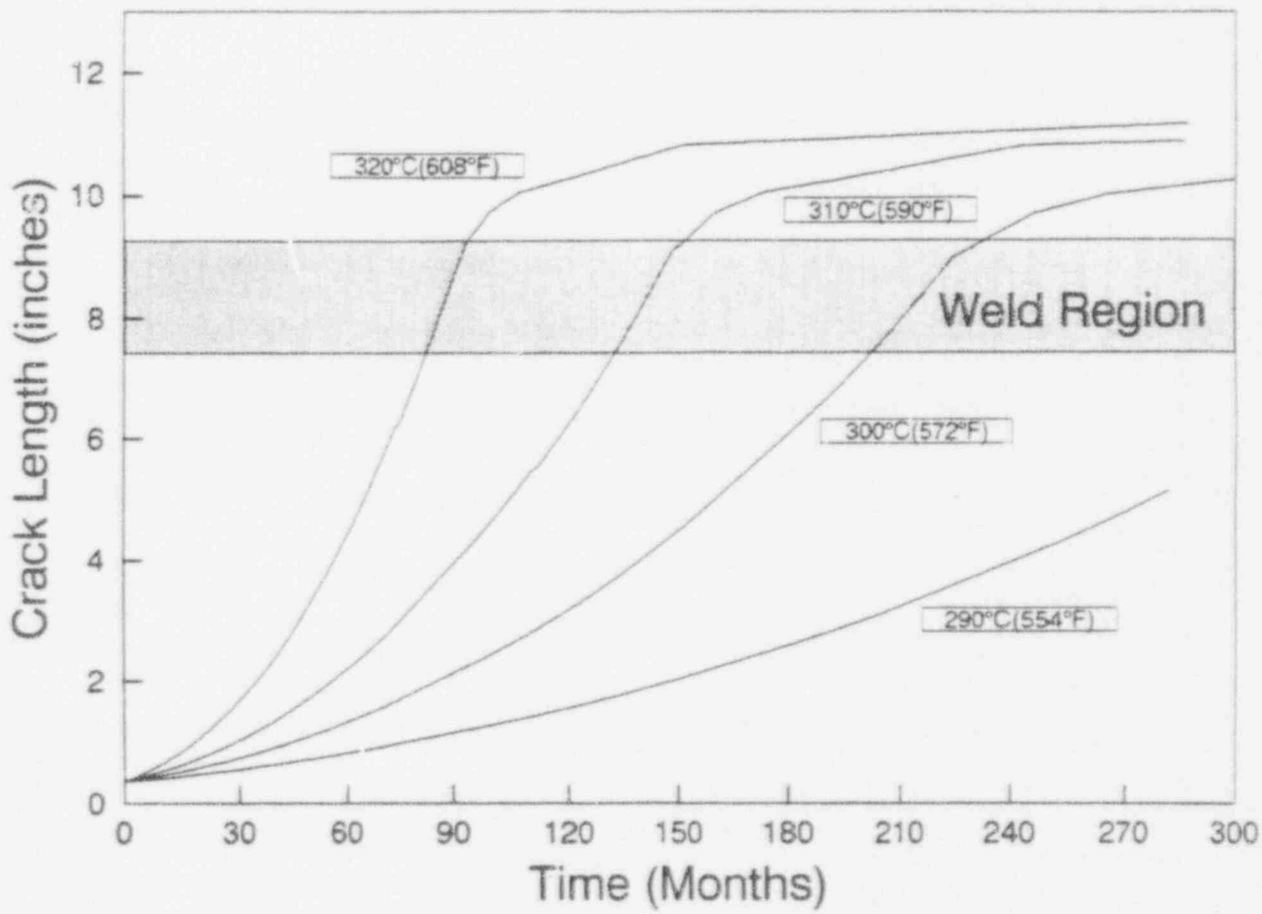


FIGURE 3.2-2

CRACK GROWTH PREDICTIONS FOR THROUGH-WALL FLAWS LOCATED AT THE CENTER SIDE OF THE OUTERMOST HEAD PENETRATIONS, FOR A RANGE OF TEMPERATURES

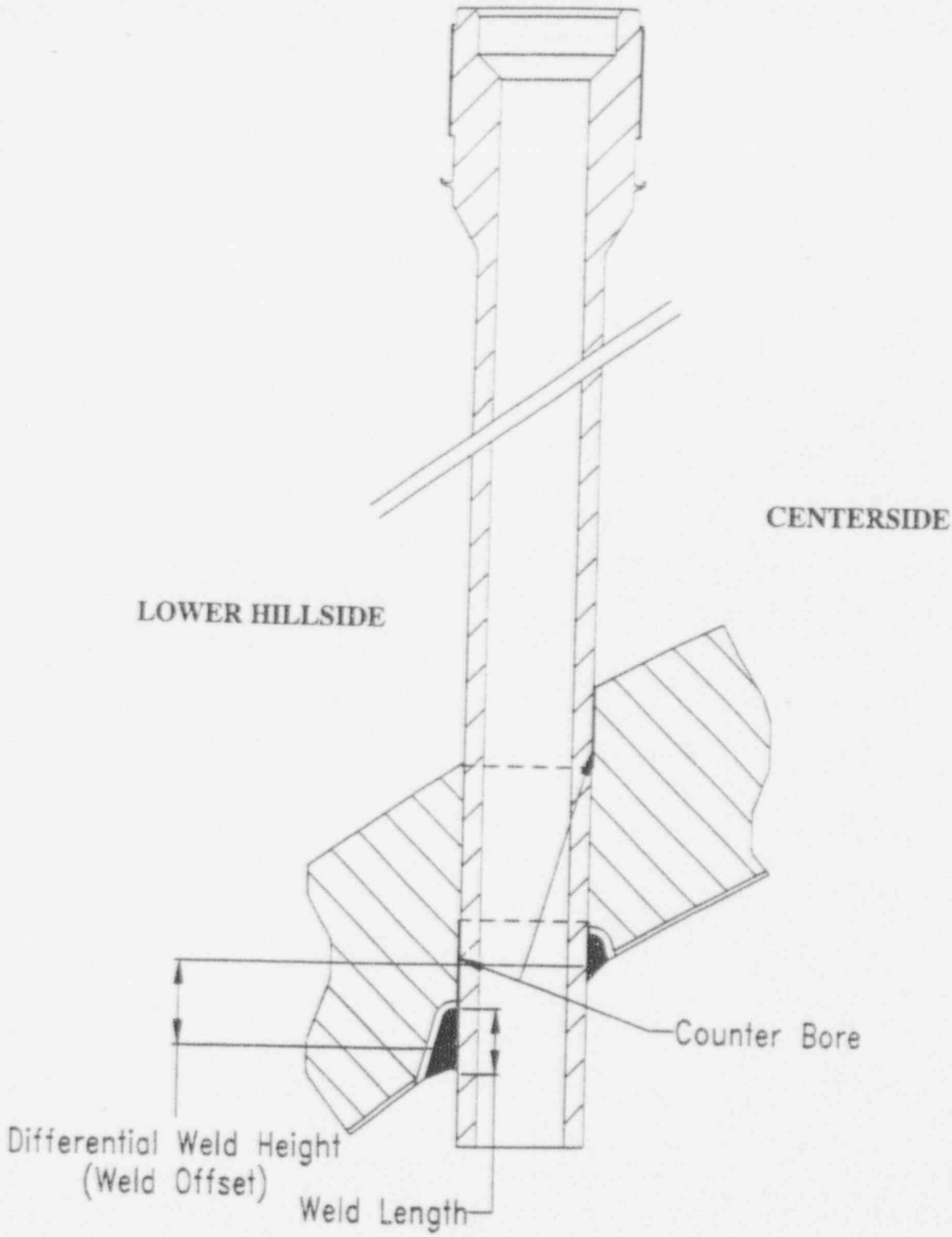


FIGURE 3.2-4

WELD LENGTH ON HEAD PENETRATION TUBE

3.3 ASSESSMENT OF WOG PLANTS

Using the stress and crack propagation analysis data and methodology provided above, an assessment of the potential for cracks in the 54 WOG plants was performed. This assessment is summarized below. Indication data from inspections performed at the Ringhals and Beznau plants is used to estimate the condition of the reactor vessel head penetrations in the WOG plants.

Review of Penetration Indication Data

A review of the inspection results from the Ringhals and Beznau plants has been performed. A summary of the inspection results is provided below.

Plant	Total No. of Penetrations	Penetrations Inspected	Penetrations W/ Indications
Ringhals 2	65	65	6
Ringhals 3	65	60	0
Ringhals 4	65	65	2
Beznau 1	36	22	2
Beznau 2	36	27	0

In the case of Ringhals 2, six penetrations were found to be cracked. The maximum depth of these cracks were reported to be 4 mm with a length of 16 mm. This plant had operated for approximately 108,400 hours at the time the measurements were taken. Two of the Ringhals 4 penetrations were found to be cracked. These cracks were judged to be very shallow with a maximum length of 7 mm. Ringhals 4 had operated for approximately 75,000 hours when the measurement was taken.

Indications were found on two of the penetrations at Beznau Unit 1. Depth was estimated to be less than 2 mm and the longest indication was estimated to be 28 mm. These indications were found after at least 157,000 hours of operation.

Applicability to WOG Plants

Westinghouse has completed a review of the cause of the cracking in French plants, and the key parameters associated with this situation for the Ringhals and Beznau plants, as well as selected French plants, and made a comparison of these key parameters to the WOG plants. Westinghouse has concluded that the mechanism of degradation in the reactor vessel head penetrations is primary water stress corrosion cracking (PWSCC). This conclusion was reached after an engineering review (Reference 1) of the data from the French plants. This review included consideration of metallographic records, material, environment, and penetration bending and ovality.

Several potential key factors which could impact susceptibility to PWSCC of the reactor vessel head penetrations were investigated (Reference 2). These included residual and operating stresses in the penetration, environment, material condition, temperature, and time of operation at temperature and pressure. The review of material condition included such parameters as microstructure, heat treatment, welding procedures, chemistry, yield strength, and machining/grinding. It was concluded that a relative susceptibility could be developed. The comparison of the key parameters between the Ringhals, Beznau, and French plants, to the WOG plants resulted in the conclusions that the Ringhals 2 data is representative of the WOG plants. Further, an assessment of the WOG plants can be made by comparing the relative susceptibility of those plants to Ringhals 2. The condition of the Beznau units fall within the boundaries of this model.

Estimate of Condition of WOG Plant Penetrations

An estimate of the condition of the WOG plants, relative to Ringhals 2, was developed. This estimate was made for each of these plants by comparison of the key parameters discussed above and application of the stress and crack propagation analyses discussed previously.

Ringhals 2 was reported to have a maximum 4 mm deep crack after approximately 108,400 hours (12.4 effective years) of operation. The head temperature during all of this time, except for an intermediate period of approximately 11,000 hours, was 605.6°F. As a result of plant operating conditions, the head temperature during the 11,000 hours was 579.9°F. An equivalent operation time of 100,130 hours (11.4 effective years) was calculated assuming the head temperature was a constant 605.6°F during the entire time of operation. This equivalent operation time was determined using the relationship:

$$1/t \propto A\sigma^{n}e^{-Q/RT} \qquad \text{(Reference 2)}$$

where:

A = A constant related to the material microstructure characteristics

σ^{n} = A constant related to the material stress

Q = 50,000 cal/mole

R = 1.987 cal/mole°K

T = head temperature (°K)

It is possible to make comparisons and draw conclusions relative to the WOG plants on the basis of the Ringhals 2 experience, by determining an equivalent operation time for the WOG plants, which has been corrected for the key parameters. By comparing this equivalent operation time for each plant, to the operation time of Ringhals 2 at the time the cracks were found, estimates can be made about the condition of the head penetrations in the WOG plants, relative to the condition of the Ringhals 2 head penetrations. The equivalent operation times were developed using the expression provided above to account for the key parameters, as described below.

Experimental evidence based on the tests conducted over the past several years suggests that PWSCC susceptibility of Alloy 600 is strongly sensitive to, among other factors, the microstructure of the material. It was further established that the extent of grain boundary coverage by carbides plays a very significant role in controlling the cracking behavior of this alloy. It is generally established that increasing the grain boundary coverage increases the resistance to PWSCC. In a recent systematic study of microstructure versus crack initiation time on Alloy 600, it has been shown that the crack initiation time can increase by a factor of five when the grain boundary carbide coverage is increased from zero to 100%. This observation is also found to be very consistent with the recent cracking experience on reactor vessel head penetrations in French plants where the microstructural examinations conducted on the material from nineteen penetrations which exhibited cracking, showed very little or no grain boundary carbide coverage in every case.

A microstructural study of eight samples of material representing typical WOG plant head penetration material was conducted to examine the grain boundary carbide coverage. This population of samples included two heats of material which are also representative of the Ringhals 2 material. The results showed that at least five of the eight samples exhibited good carbide coverage. Of the three samples which did not exhibit good carbide coverage, two were samples of the Ringhals 2 representative material. The results of these examinations were compared against data representative of the head penetration material in the French plants. A quantitative comparison of the grain boundary carbide coverage suggested a factor ranging from 3 to 5 increase in crack initiation time relative to the worst case microstructure for the French plants. Additionally, the data indicates that a factor of at least 3 could be applied to most of the WOG plants for crack initiation time, relative to Ringhals 2. However, in view of the relatively small sample size, a factor of 1 was conservatively used in the assessment of the WOG plants relative to Ringhals 2 and therefore the superior structure was not taken credit for in this evaluation.

Two key parameters were accounted for with the stress constant, σ. First, a review was made of material certifications in the Quality Assurance data packages for all of the plants to determine the yield stress of the head penetration material applicable for each plant. It is generally accepted that material yield stress is a factor in the susceptibility of a material to PWSCC, higher yield stress material being more susceptible. A constant determined by the ratio of the WOG plant yield stress to the Ringhals 2 yield stress, raised to the fourth power, was then determined for each WOG plant. A second constant was determined for each plant to account for the level of operational and residual stresses applicable to each WOG plant, as compared to the Ringhals 2 plant. Stress analysis and ovality measurements of the head penetrations have shown that the level of stress in the outer penetration (highest stressed) of a four loop plant is higher than that of a two or three loop plant. Since Ringhals 2 is a three loop plant, this second stress constant was determined for the four loop plants by taking the ratio of the ovality measured at four loop plants to the ovality measured at a three loop plant, raised to the fourth power.

Finally, a correction was made for the head operating temperature by determining the WOG plant "equivalent" operating time, if it were operating at the Ringhals 2 head temperature.

With all of the key parameters accounted for as described above, an equivalent operating time was determined for each of the WOG plants. These operating times were then compared to the operating time of Ringhals 2, at the time the 4 mm deep crack was discovered, corrected for the

11,000 hours of operation at the lower temperature as described above. Those plants with a lower number of equivalent operating hours are considered to be less susceptible to PWSCC than Ringhals 2. Likewise, those plants with a higher number of equivalent operating hours than Ringhals 2 may be more susceptible to PWSCC.

Results and Conclusions

It was concluded that in terms of relative susceptibility to PWSCC, the Ringhals 2 condition envelops 45 of the 54 WOG plants. It is therefore expected that none of these 45 WOG plants would have cracks deeper than 4 mm, if there are cracks present at all. Further, it is estimated based on the crack propagation analysis described in Section 3.2, a 4 mm crack would not propagate to become a through wall crack in any of these 45 plants within an additional time period of 14,000 hours (2 years of operation at 80% availability).

Conservatively, it was concluded from the assessment of relative susceptibility to PWSCC, that 9 of the WOG plants may be more susceptible than Ringhals 2. These results suggest that these plants may have partial penetration cracks deeper than the 4 mm crack found at Ringhals 2. In addition, conservative predictions indicate that through-wall cracks at or below the weld may exist in some of these plants. In these cases, however, crack extensions above the weld would be less than 1.0 inch at the end of an additional operational period of 24 months. It should be noted, as can be seen in Figure 3.2-3, crack growth rates decrease significantly above the weld. Section 3.5 of this evaluation summarizes the wastage assessments. Conclusions concerning plants with penetrations cracked above the weld are made in that section.

3.4 LEAK RATE CALCULATIONS FOR THE REACTOR VESSEL HEAD PENETRATION

The purpose of this analysis is to estimate the leak rate between the reactor vessel head penetration and the reactor vessel head when an axial through-wall crack is postulated in the former. A schematic of the vessel head penetration and vessel head portion associated with this region is shown in Figure 3.4-1. To determine the leak rate through this annulus, the leak rate through the head penetration crack is first estimated. This value is then compared to the leak rate through the annulus based on a choked flow rate. The lower of these two rates is taken as the leak through the annulus.

The clearance between the head penetration and the vessel is effected by the different thermal expansion coefficients of the base metal and the weld metal, internal pressure, and "as-fabricated" interference fit of the penetration.

The pressure inside the head penetration was 2250 psia. The calculations were performed that apply to the entire temperature range of operation for the WOG plants (550°F to 620°F). The outside radius and thickness of the tube are 2 in. and 0.622 in. respectively. In this analysis, the clearance between the head penetration and vessel is considered to be a variable parameter, and it is assumed to be uniform through the direction of the head penetration axis.

The leak rate through the area of concern is estimated using two systems models. The first leak rate is estimated through a postulated axial through wall crack in the head penetration. Second, a leak rate is calculated through the tight annulus corresponding to the clearance between the head penetration and the vessel. The water inside the head penetration is in the subcooled liquid phase. The leak rate through the postulated crack in the head penetration is estimated using a two phase flow model

(Reference 1). The leak rate through the clearance between the tube and the vessel is estimated using a single phase flow model (superheated steam). If subcooled liquid leaks from the postulated crack in the head penetration, the pressure will drop radically due to crack friction effects. The temperature, however, of the fluid would not drop as quickly as the pressure, since the crack is surrounded by a large heat source contained in a structure such as the vessel. Hence, the flow in the annulus would be single phase. This fluid injected through the postulated crack of the head penetration would change to steam and would leak through the tight annulus clearance of the vessel. The leak rate of steam is estimated using the choked flow rate of steam.

Mass conservation is assumed at the exit of the head penetration crack and at the exit of the vessel. Since the interaction of fluid flow resistance between the head penetration crack and the tight annulus clearance of the vessel is not known, an engineering assumption is made: If the flow rate through the head penetration is smaller than that through the annulus, the flow rate is considered to be the flow rate through the head penetration. Alternately, if the flow rate through the head penetration is larger than that through the annulus, then the flow rate is considered to be the flow rate at the annulus. This implies that flow resistance imposed on the flow rate through the head penetration crack is caused by the very tight clearance between the penetration and the vessel. This results in the reduction of the flow rate from the head penetration crack.

Using the engineering assumption described above, a numerical calculation is performed. The stagnation pressure in the clearance between the head penetration and the vessel is considered to be equal to the choked pressure at the exit of the head penetration crack. The stagnation temperature in the annulus is assumed to be the same as the vessel temperature (600°F). The ratios of heat capacity C_p/C_v are obtained from the Mollier diagram at the choked pressure and temperature. The choking pressure is a function of the crack geometry.

The crack opening area of the head penetration and leak rates are calculated for various circumferential crack lengths. These values are tabulated using both British and SI units. Mass flux for various axial crack lengths and heat capacity ratios are also given in both British and SI units.

The clearances in the axial direction are obtained using a finite element model. The leakage through these clearances are tabulated and provided along with tables showing calculated leak rates.

It can be concluded that generous margins exist for all the plotted leakage size flaws. The annular clearance between the head penetration and the reactor vessel during operation is estimated to be 0.003 inches. Therefore, since the crack extensions above the weld are expected to be less than 2.0 inch, as indicated in Section 3.3 of this evaluation, a review of Figure 3.4-2 provides that the maximum leakrate which can be expected is 0.70 gpm. Leakage in excess of 1.0 gpm is detectable in WOG plants.

For the case where the "as-fabricated" penetration interference fit is at the high end of the tolerance, the annulus between the head penetration O.D. and the corresponding vessel hole I.D. is reduced to very small levels during plant operating conditions. As such, the leakage will be smaller than the above value for penetration cracks lengths equal to that of the vessel head thickness. It is considered even less likely that cracks would grow to this length because of the compressive stress in the penetration. If crack growth beyond the outside surface of the head is considered, leakage will be detected prior to the crack reaching the critical flaw size of 13 inches in the penetration.

Table 3.4-1
Leak Rates Through a Range of Crack Lengths: Head Penetration

Crack Length CL	Crack Opening Area in^2 (cm^2)	Leak Rate lb/sec kg/sec	Choking Pressure psi (kg/cm^2)
1.0 in (2.54) cm	0.0004117 (0.002656)	0.0103 (0.004672)	381 (26.7874)
2.0 (5.08)	0.00246 (0.01587)	0.0968 (0.04391)	596 (41.9037)
3.0 (7.62)	0.00922 (0.05948)	0.5253 (0.2383)	846 (59.4807)
3.5 (8.89)	0.01622 (0.1046)	1.1115 (0.5042)	979 (68.8317)
4.0 (10.16)	0.02721 (0.1755)	2.3772 (1.0783)	1138 (80.0107)

Table 3.4-2
Crack Length, Mass Flux and Heat Capacity Ratio
in the Annulus Clearance Between Penetration and Vessel

Cl in.	$\underline{\dot{m}}/\underline{A}$ lb/in^2	K=Cp/Cv
1.0 (2.54) cm	4.7677 lb/sec in^2 (0.3352) kg/sec. cm^2	1.2885
2.0 (5.08)	7.4409 (0.5232)	1.2800
3.0 (7.62)	10.5354 (0.7407)	1.2709
3.5 (8.89)	12.1750 (0.85600)	1.2660
4.0 (10.16)	14.1285 (0.9933)	1.2600

Table 3.4-3
Leak Rate Through the Annular Clearance Between the Head Penetration and Vessel

δ	A	Crack Length				
		Cℓ = 1.0 in. (2.54) cm	2.0 (5.08)	3.0 (7.62)	3.5 (8.89)	4.0 (10.16)
0.0001 in (0.000254) cm	0.0012567 in^2 (0.008108)cm^2	0.0060 lb/sec (0.002722) kg/sec	0.009400 (0.004264)	0.01300 (0.005897)	0.0153 (0.006940)	0.01780 (0.008074)
0.0013 (0.003302)	0.01634 (0.10542)	0.07790 (0.03534)	0.1216 (0.05516)	0.1722 (0.07811)	0.1990 (0.09027)	0.2309 (0.1047)
0.0019 (0.004826)	0.02389 (0.15413)	0.1139 (0.05167)	0.1777 (0.08060)	0.2517 (0.1142)	0.2908 (0.1319)	0.3375 (0.1531)
0.0032 (0.008128)	0.04024 (0.25961)	0.1919 (0.08705)	0.2995 (0.1359)	0.4240 (0.1923)	0.4900 (0.2222)	0.5686 (0.2579)
0.0033 (0.008382)	0.04150 (0.26774)	0.1979 (0.08977)	0.3088 (0.14007)	0.4373 (0.1984)	0.5053 (0.2292)	0.5864 (0.2660)
0.0054 (0.013716)	0.06795 (0.43839)	0.3240 (0.1470)	0.5056 (0.2293)	0.7159 (0.3247)	0.8273 (0.3753)	0.9600 (0.4355)

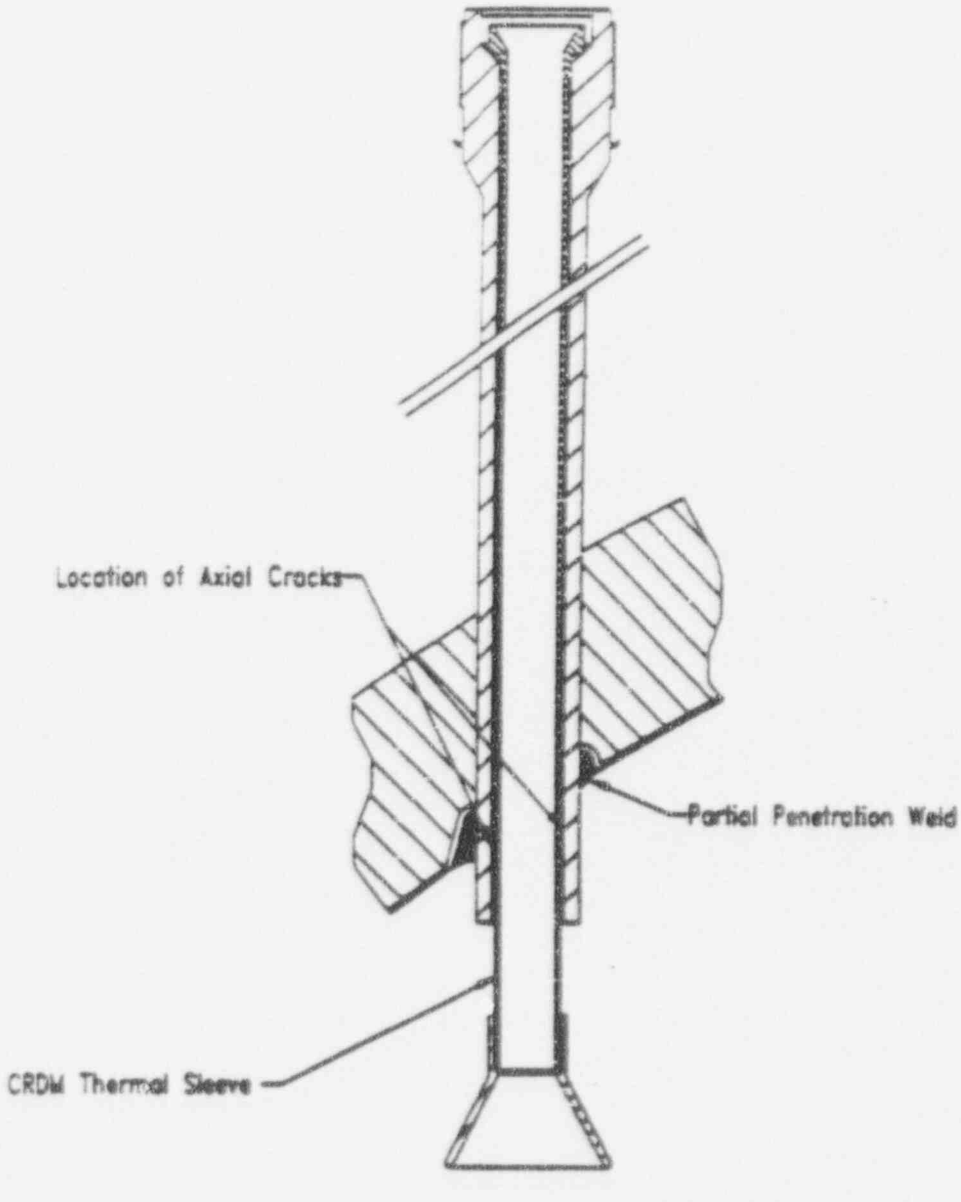
Numbers in the parentheses are SI units

Table 3.4-4 **Final Leak Rate at the Exit of Annulus Clearance Between Head Penetration and Vessel**								
Clearance δ(in)	$C\ell$ = 1.0 in.		2.0 in.		3.0 in.		3.5 in.	
	Q_v lb/sec	GPM*	Q_v lb/sec	GPM*	Q_v lb/sec	GPM*	Q_v lb/sec	GPM*
0.0001 (0.0025) mm	0.00600 (0.002722)**	0.043 (0.1633)***	0.0094 (0.004264)	0.06700 (0.2558)	0.013 (0.005897)	0.095 (0.3538)	0.0153 (0.006940)	0.11 (0.4164)
0.0013 (0.0330) mm	0.010261 (0.004654)	0.074 (0.2793)	0.09676 (0.04389)	0.700 (2.6334)	0.1722 (0.07811)	1.24 (4.6866)	0.1990 (0.09027)	1.43 (5.4160)
0.0019 (0.0483)	0.010261 (0.004654)	0.074 (0.2793)	0.09676 (0.04389)	0.700 (2.6334)	0.2517 (0.1142)	1.81 (6.8503)	0.2908 (0.1319)	2.1 (7.9144)
0.0032 (0.0813)	0.010261 (0.004654)	0.074 (0.2793)	0.09676 (0.04389)	0.700 (2.6334)	0.424 (0.1923)	3.05 (11.5396)	0.4900 (0.2222)	3.5 (13.3358)
0.0033 (0.0838)	0.010261 (0.004654)	0.074 (0.2793)	0.09676 (0.04389)	0.700 (2.6334)	0.4373 (0.1984)	3.14 (11.9016)	0.5053 (0.2292)	3.6 (13.7522)
0.0054 (0.1372)	0.010261 (0.004654)	0.074 (0.2793)	0.09676 (0.04389)	0.700 (2.6334)	0.52531 (0.2383)	3.78 (14.2968)	0.8273 (0.3753)	5.9 (22.5158)

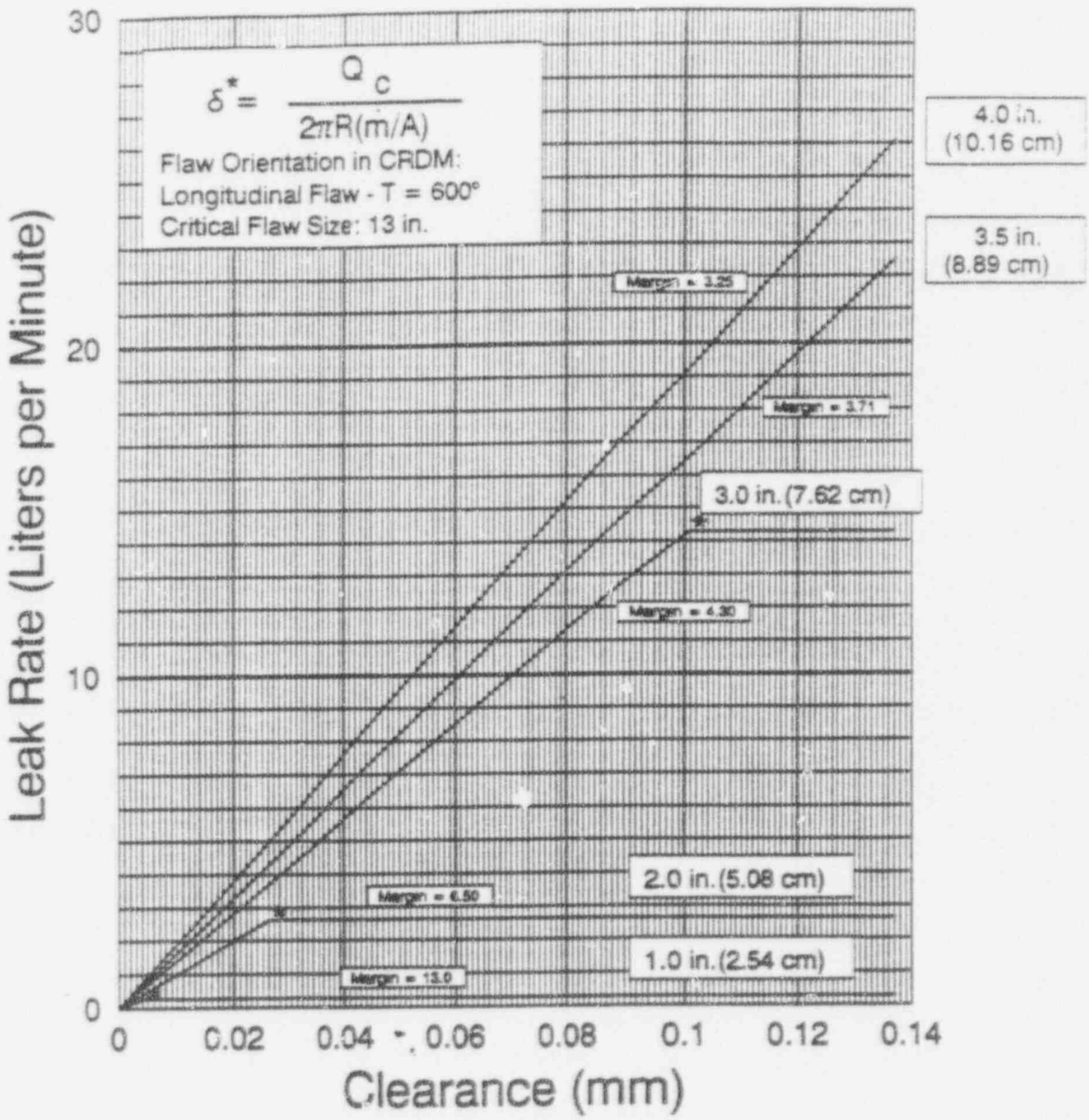
* at room temperature conditions
** kg/sec
*** liter/min.

Table 3.4-5 **Final Leak Rate at the Exit of Annulus Clearance Between Head Penetration and Vessel**		
Clearance δ(in)	$C\ell$ = 4.0 in.	
	Q_v lb/sec	GPM*
0.0001 (0.0025)	0.0178 (0.008074)	0.13 (0.4844)
0.0013 (0.0330)	0.2309 (0.1047)	1.7 (6.2842)
0.0019 (0.0483)	0.3375 (0.1531)	2.4 (9.1854)
0.0032 (0.0813)	0.5686 (0.2579)	4.1 (15.4750)
0.0033 (0.0838)	0.5864 (0.2660)	4.2 (15.9595)
0.0054 (0.1372)	0.9600 (0.4355)	6.9 (26.1274)

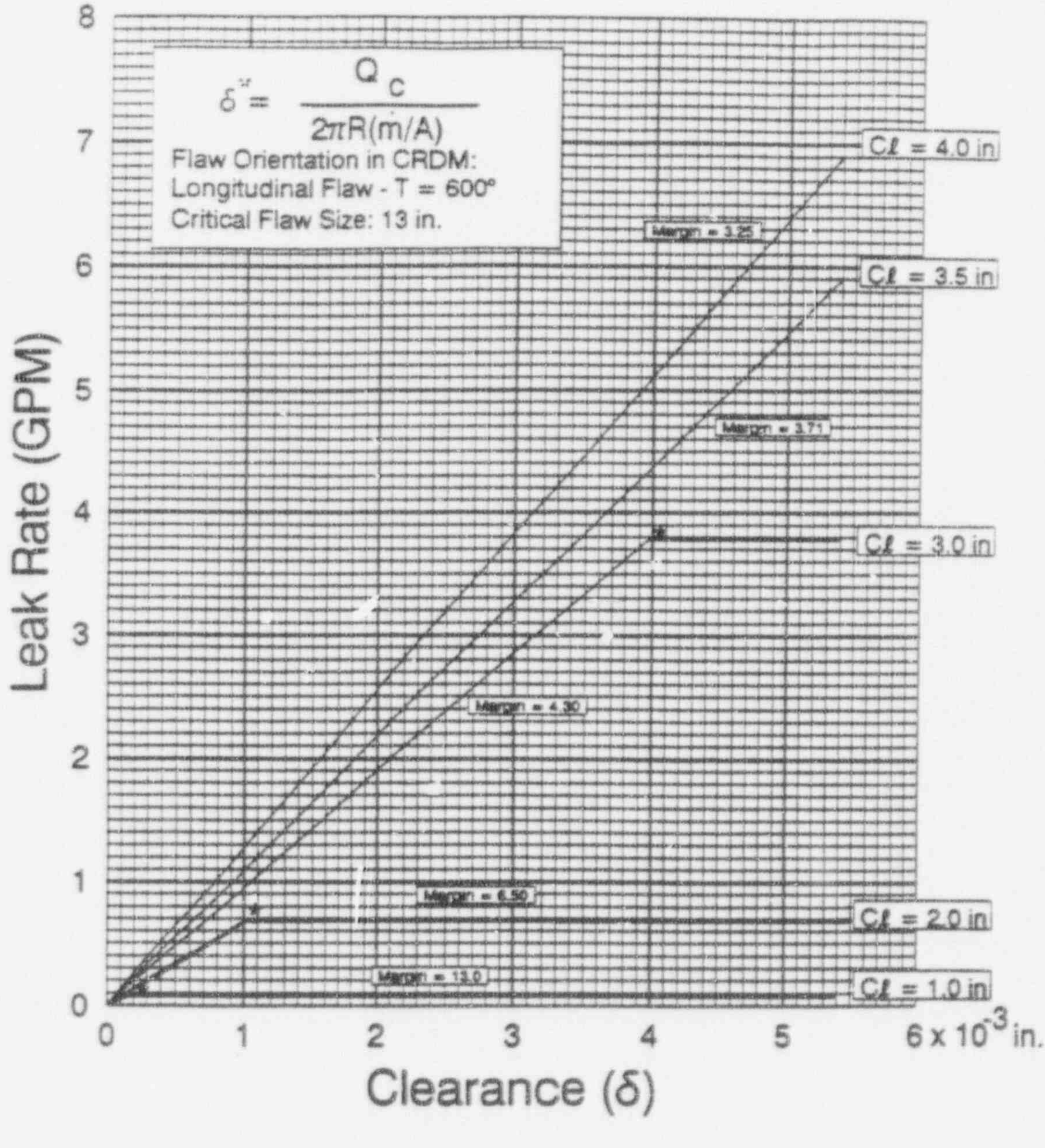
* at room temperature conditions



Head Penetration and Vessel Assembly
Figure 3.4-1



Leak Rate as a Function of Clearance and Crack Lengths in Terms of Liters per Minute at the Exit of Annulus Clearance Between Head Penetration and Vessel
Figure 3.4-2



Leak Rate as a Function of Clearance and Crack Lengths in Terms
of GPM at the Exit Annulus Clearance
Between Head Penetration and Vessel
Figure 3.4-3

3.5 REACTOR VESSEL HEAD WASTAGE ASSESSMENTS

General Technical Discussion

The purpose to this section is to conduct an assessment of the potential wastage (i.e. pitting, and wall thinning by general corrosion) of the reactor vessel head due to the leakage of the boric acid coolant through a postulated axial through wall crack in the Alloy 600 head penetration. The wastage assessment considered here is based on the existing wastage data obtained from the laboratory test programs conducted at Westinghouse and the results of a penetration mockup test conducted under a Combustion Engineering Owners Group (CEOG) program, Reference [15].

For the current wastage considerations, under steady state operating conditions, it is assumed that the top of the reactor head is maintained at approximately 500°F while the top of the insulation above the vessel head is maintained at approximately 150°F. Under these conditions, the coolant leak through the penetration will leave the penetration and the counter bore annulus in the form of (superheated) flashing steam leaving behind a "snow" of boric acid crystals in the crevice and at the top of the vessel head around the penetration. The majority of the boric acid crystals formed in the crevice are expected to be slowly pushed out to the top of the vessel head by the exiting steam. Westinghouse laboratory test results showed that boric acid crystals heated to 500°F contributed to no or negligible wastage of carbon steel. Due to the high temperature environment in the crevice region, the wastage in the crevice region due to steam moisture is expected to be minimal. Any occurrence of wastage at the vessel head would require a re-wetting mechanism of the dry boric acid crystals deposited on the top surface of the vessel head. This re-wetting mechanism would require a condition whereby as the steam escapes through and above the insulation, it encounters lower temperatures in the range of 150°F to 212°F where it starts to condense into moisture, a fraction of which would potentially find its way back to the vessel top surface through the available flow paths in the insulation. This, of course, could create a wetting condition of the boric acid crystals deposited on the top of the vessel head. Since the vessel head is maintained at near 500°F, any wetting is expected to be minimal if at all. Conservatively, a continuous leak could establish a wetting and dryout condition of the boric acid crystals on the top of the vessel head resulting in some wastage at the crevice mouth region.

Laboratory tests conducted at Westinghouse showed that aqueous boric acid solutions caused carbon steel to corrode at rates dependent on the concentration of boric acid in solution at any given temperature. Low concentrations (approximately 1500 ppm boron) produced corrosion rates on the order of 5 to 10 mils per month, whereas concentrations of 25% by weight of boric acid removed carbon steel from a specimen at a rate of approximately 400 mils per month at 200°F. A concentration of 25% by weight of boric acid is saturated at about 200°F. Galvanic corrosion between carbon steel and Inconel-600 appeared to contribute little to the carbon steel attack in aqueous boric acid solutions. The boric acid crystals, when heated, dehydrated to form B_2O_3, a glass-like substance which coated the specimens and apparently protected them from atmospheric oxidation. The glass-like material is, however, a potentially corrosive agent because it hydrolyzes in hot water to form boric acid. A 25% by weight of boric acid solution, de-aerated for 24 hours with a nitrogen sparge, produced an average corrosion rate of 250 mils per month. A 25% by weight of boric acid solution containing dissolved oxygen yielded a corrosion rate of about 400 mils per month. The presence of insulation was found to inhibit the corrosion of carbon steel at 200°F.

The space between the vessel head top surface and the insulation is expected to be in the temperature range of 500°F to 350°F (on a conservative basis) so that active steam condensation is not likely to occur here and the majority of steam is expected to escape through the clearance at the head penetration. This creates the situation where by the boric acid crystals are left below the insulation at the penetration while the majority of steam condensation is occurring above the insulation. Under this condition, only a fraction of condensed steam can reach the boric acid crystals located at the annulus of the affected penetration to re-wet the crystals and boil away, creating an intermittent wetting and dry out conditions. The above conditions are expected to significantly moderate the maximum observed wastage rate of 400 mils per month achievable in the laboratory under complete aqueous, concentrated, and full oxygenated condition of the boric acid at 212°F.

As indicated in the previous section, the leakage rate to be considered through the largest axial "through-wall" flaw expected in a WOG plant penetration, estimated at two inches in length, is approximately 0.7 gpm at an average annular gap of 0.003 inch. In comparison, the maximum leak rate expected for a "through-wall" flaw of approximately 1 inch is 0.074 gpm for the same annular gap. This value is consistent with the leak rates experienced in the CEOG test program for comparably sized cracks.

An assessment of realistic wastage rates achievable at a head penetration can be made from the results of the recent mockup test conducted by the CEOG to establish wastage rates due to the leakage at the bottom of a pressurizer. Key parameters of the CEOG test are; a) leak rates ranged from 0.026 gpm to 0.119 gpm, b) the diametral crevice clearances ranged from 0.0015 inch to 0.0099 inch, and c) the block (head) temperatures ranged from 351°F to 566°F. These parameters are consistent with the conditions found at the WOG reactor vessel closure penetrations. The relevant results of the test can be summarized as follows:

1. Although the maximum penetration rate (at the deepest pit) observed was 2.15 inch/year at a localized region, the maximum average penetration rate achieved was 0.0835 inch/year.

2. The maximum total metal loss rate (wastage volume) observed was 1.07 in^3/year.

3. The greatest damage occurred almost entirely where the leakage left the annulus.

The CEOG mockup test results are judged to represent a conservative estimate of the wastage rates which could be expected due to the leakage at the vessel head penetration. High condensation in the CEOG test is postulated, as compared to the head penetrations; 1) due to the close proximity of insulation in the test, and 2) it is postulated that exiting steam would have been expected to rise back to the pressurizer head surface rather than escaping away from the surface, due to the inverted geometry of the simulated pressurizer test configuration. The anticipated higher condensation would serve to maintain a relatively moist environment, i.e., rewetting, thus resulting in conservative wastage rates as compared to the head penetration geometry. Recall that leak rates range from 0.074 gpm to 0.7 gpm for maximum expected crack lengths of one to two inches depending on the circumferential position about the weld. Existing in plant leak detection capabilities are limited to 1.0 gpm or higher. Thus a flaw which results in leakage equal to or greater than 1.0 gpm can be detected and addressed appropriately. Thus, it is only necessary to associate a wastage rate with leak rates ranging from 0 to 1.0 gpm. Based on the conservatisms judged to be in the crack length determination, the leak rate assessment, and the CEOG test data, the 1.07 in^3/year metal loss rate was selected as an appropriate value for use over the 0.0 to 1.0 gpm leak rate.

Analysis of Reactor Vessel Head

Two dimensional finite element analyses of 2, 3, and 4 loop reactor vessel heads were performed to assess the impact to the structural integrity of the reactor vessel head of the wastage discussed above. As discussed above, the wastage rate considered was 1.07 in^3/year. Six years was chosen as the time period which the wastage at this rate would occur undetected. This would result in a total loss of approximately 6.4 in^3 of vessel head material. The CEOG test data indicates that the wastage would be very localized, in a very small area where the leak exits from the annulus between the reactor vessel head and the head penetration. As indicated by the head penetration stress analysis discussed in Section 2.0, and experience in the French plant at which leakage actually occurred, the leakage would exit the annulus on the up-hill side of the penetration. This is because the annulus between the reactor vessel head and the penetration tends to open on this side of the penetration as the reactor vessel is pressurized during plant operation.

Based on the CEOG test data, two defect shapes were postulated to umbrella the various possible wastage defect shapes that could occur. One shape considered was a defect approximately 2.0" wide radially X 1.0" wide circumferentially X 3.2" deep (into the thickness of the head). The second shape considered was a defect approximately 1.07" wide radially X 1.0" wide circumferentially and extends through the vessel head thickness to the head penetration to vessel head weld. The two dimensional finite element analysis was performed to assure the structural integrity of the vessel head is maintained. This calculation is an allowable alternative to the ASME Code Section III conservative sizing rules based on nozzle reinforcement and ligament efficiency calculations. For each plant size, analyses were performed on the reactor vessel head in the normal or as manufactured condition, as well as with the above described defects introduced, and comparisons of the results were made to draw conclusions on the impact of the wastage. The wastage defects were introduced into the models as described below.

Each model developed was a two dimensional axisymmetric model representing the reactor vessel head and including the effects of the head adapters, especially near the expected location of possible leakage; and an approximation of the effect of the vessel head flange. The input parameters for these models were taken from the vessel design reports representative of 2, 3 and 4 loop plants. The information required includes the inside radius of the head to the base metal, the thickness of the vessel head (base metal), the locations of the penetrations, the diameter of the penetrations, (all 4.0 inches) and information describing the configuration of the vessel head flange. The code used for evaluation of this model is the WECAN-Plus computer code which is proprietary to the Westinghouse Electric Corporation. Various finite elements are available for evaluation, with element 53, the "Two Dimensional Isoparametric", being selected. This is a 2-D Quad element and is considered with axisymmetric properties. Furthermore, midside nodes were also considered to refine the displacement function to consider quadratic edges (of the quads). A minimum of five elements were selected through the thickness of the head, providing for eleven integration points through the thickness.

The pitch between adjacent adapters was considered as 11.035 for the 2 loop plant and 11.973 inches for the 3 and 4 loop plants. The elements representing the base metal of the head were identified as material property #1. At each location of a penetration, starting with the one at the centerline of the vessel, and spaced according to the pitch dimension mentioned above, the material properties were identified as property #2. The material properties for the adapter region were modified to consider the hole penetrating the base material. The hole properties were adjusted based upon a ratio of the adaptor diameter (4 inches for all plants) to the adapter pitch (pitch of holes in head).

The wastage is introduced at the inside of an outer most adapter, starting at the outside of the vessel head surface. The material property for the wastage is identified as material #3. For the 4 loop size vessel head, several different defect geometries were evaluated to envelop the two defect sizes described above. These included a 2X2 element defect, a 1X4 element defect, a 2X4 element defect, a 1X5 element defect and a 2X5 element defect. It was concluded from the review of the results of these cases that the 2X2 element case and the 2X4 element case conservatively envelop the two representative defects described above. These two cases, along with the normal head configuration case (no defect) were also run for the 2 loop and 3 loop size heads.

The boundary condition was applied to this model at a node near the gasket seal of the head (junction with the vessel). Since the axisymmetric option was selected, the boundary condition at the centerline need not be specified. A uniform pressure loading of 2500 psi was applied to the inside surface of the head.

The stresses in the center of the ligament between the penetration hole with the defect and the adjacent penetration hole were compared for each of the three cases, for each size head. The largest increase in stress intensity between the normal head configuration and the head configuration with the postulated wastage defect is only 6.5%. With this minor increase, the general primary membrane stress intensities remain below the corresponding ASME Section III allowable stress intensity limit (S_m).

In the unlikely event that a leak would develop in a WOG plant reactor vessel head penetration, and continue undetected for a period of time of up to six years, the wastage that would result on the vessel head is expected to be local to the immediate area of the penetration. It is conservatively estimated that the low alloy steel in the vessel head would waste at an approximate rate of 1.07 in^3 per year or 6.4 in^3 after six years of undetected leakage. Analysis of the vessel head in this degraded condition concludes that the stresses remain within the ASME code allowables and therefore the structural integrity of the reactor vessel head would not be jeopardized. This conclusion is applicable to 2 loop, 3 loop, and 4 loop size reactor vessel heads.

4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

1. Continued plant operation with the situation as described in this evaluation does not increase the probability of an accident previously evaluated in the FSAR. Inasmuch as catastrophic failures of vessel head penetrations are not expected, and any postulated through wall crack

would only lead to a minimal amount of leakage, the accident scenarios as presented in the FSAR are not impacted. Concerning the question of wastage, this evaluation has shown that over six years of operation is possible without impacting plant safety even with undetected, leaking penetrations.

2. The consequences of an accident previously evaluated in the FSAR are not increased due to continued plant operation. The preceding safety evaluation has shown that the reactor coolant system is not challenged in such a way as to deleteriously affect continued operation. As described above, catastrophic failures of vessel head penetrations are not expected, and any postulated through wall crack would only lead to a minimal amount of leakage. Wastage issues, for the plants most susceptible to postulated through wall cracks, have been shown not to affect plant operability for over six years. Therefore, the conclusions presented in the FSAR remain valid such that no more severe consequences will result from an accident condition.

3. Continued plant operation will not create the possibility of an accident which is different than any already evaluated in the FSAR. No new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified. Therefore, the possibility of an accident different than any already evaluated is not created. The postulated leaks and wastage issues, evaluated herein, will not create an accident different than any previously evaluated in the FSAR.

4. Continued plant operation will not increase the probability of a malfunction of equipment important to safety. Potential cracking, postulated leaks and wastage issues, as presented in this evaluation, will not cause the malfunction of equipment important to safety.

5. Continued plant operation will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The preceding safety evaluation has concluded that this situation will not adversely affect the reactor coolant system in such a way as to affect the expected consequences of the malfunction of any equipment important to safety.

6. Continued plant operation will not create the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR. Catastrophic failures of vessel head penetrations are not expected, and any postulated through wall crack would only lead to a minimal amount of leakage. As such, the malfunction of equipment important to safety is not expected

7. The evaluation for the effects of continued plant operation with potentially cracked reactor vessel head adaptors has taken into account the applicable Technical Specifications. The preceding safety evaluation has concluded that design and safety information, as presented in the FSAR, remains bounding for all plant operational conditions. As such, the margin of safety, as defined in the bases to the Technical Specifications and demonstrated by the safety analyses, will not be reduced.

5.0 CONCLUSIONS

Based upon this evaluation and the engineering analyses and assessments performed pursuant to the Westinghouse Owners Group program regarding Reactor Vessel Head Adaptor Cracking, it is concluded that catastrophic failures of reactor vessel head adaptor tubes will not occur inasmuch as circumferential cracking is not expected to occur and any potential axial flaw will not propagate to the point at which it reaches a critical flaw size. Additionally, it is considered extremely unlikely that vessel wastage, as described in this evaluation, could continue undetected for a six year period. Further, the supplemental plant operating requirements stated in NRC Generic Letter 88-05 (Reference 10) requiring walkdown inspections looking for visible boric acid deposits reduce the likelihood that any such situation would remain undetected. Accordingly, it is concluded that this situation does not represent an unreviewed safety question per the definitions and requirements delineated in 10 CFR 50.59 (a)(2).

As determined from this evaluation, the wastage issues continue to be the most limiting factor. Catastrophic failures of head adaptors are not expected. Potential cracking is not expected to have progressed through wall above the attachment weld, but if such is postulated, the calculated leak rates are minimal. Though considered unlikely due to the reduced stress levels in the penetration tube above the weld region, if a through wall crack were to propagate to a significant level above the weld, leak rates would increase to a detectable level. Potential cracks are not expected to reach the postulated critical flaw size of 13 inches.

6.0 REFERENCES

1. "Root Cause of French Reactor Vessel Head Penetration Cracking," WOG Program MUHP-5016, MED-PCE-12224, June, 1992.

2. Reactor Vessel Closure Head Penetration Key Parameter Comparison, WCAP-13493, September, 1992.

3. Fauske, H. K., "Critical Two-Phase, Steam Water Flows," Proceedings of the Heat Transfer and Fluid Mechanics Institute, Stanford, California, Stanford University Press, 1961.

4. Shapiro, A. H., The Dynamics and Thermodynamics of Compressible Fluid Flow, Ronald Press.

5. Eiber, Pressure Vessels and Piping, Volume 94, pp. 99, 100.

6. WCAP-13525, Alloy 600 Reactor Vessel Head Penetration Cracking

7. WCAP-13426, "RV Closure Head Penetration Finite Element Stress Analysis"

8. Scott, P.M., "An Analysis of Primary Water Stress Corrosion Cracking in PWR Steam Generators," in Proceedings, Specialists Meeting on Operating Experience With Steam Generators, Brussels Belgium, Sept. 1991, pages 5, 6.

9. "Corrosion and Corrosion/Erosion Testing of Pressurizer Shell Material Exposed to Borated Water", CE-NPSD-648-P, April 1991.

10. NRC Generic Letter No. 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants", dated March 17, 1988.