WCAP-12559

KEWAUNEE STEAM GENERATOR
TUBESHEET CREVICE INDICATIONS
RETURN TO POWER REPORT

APRIL, 1990

Westinghouse Energy Systems

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RETURN TO POWER REPORT

APRIL, 1990

PREPARED FOR WISCONSIN PUBLIC SERVICE

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* The information in Sections 3.2, 5, and 7.2 was provided by Wisconsin Public Service.

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1.0 INTRODUCTION

This report provides information to support the return to power of the Kewaunee Nuclear Power Plant with indications of tube degradation within the steam generator tubesheet crevice region remaining in service. An evaluation of tube integrity and radiological consequences demonstrates that subsequent operation will provide adequate safety. An evaluation using the guidance of NSAC-125 demonstrates that an unreviewed safety question does not exist.

1.1 Background

During the 1990 Kewaunee refueling outage, a bobbin coil eddy current inspection of hot leg tubesheet crevices of the Kewaunee steam generators was performed. This base scope inspection revealed a number of tubes which exhibited signals from the 400/100 Khz differential mix channel. Subsequent rotating pancake coil (RPC) eddy current inspection of the tubesheet crevices of the tubes exhibiting such signals revealed the presence of single or multiple, axially oriented indications. These indications, although not accurately sizeable by RPC, were conservatively classified as axial cracks. In general, the bobbin coil differential mix response from the bobbin was of large enough volume that the data analysts had no difficulty in detecting and reporting them.

One tube, however, exhibited a bobbin coil differential signal barely above the noise level. This tube also exhibited axial indications from the RPC inspection in the tubesheet crevice region. To provide an assessment of the potential number of such indications the RPC program was continued in the crevice region. More than 300 tubes were inspected by RPC in the tubesheet crevice region. Approximately 20% of the additional tubes inspected by RPC revealed the presence of axially oriented indications which could not be identified in the bobbin coil differential mix when that data was rereviewed. These indications were located five inches or more below the top of the tubesheet. Several tubes with indications were judged to be less severe and will remain in service. These tubes are intended to be used to track the growth or change in indication in subsequent inspections. Additionally, it can be projected that similar indications exist in unsleeved tubes which were not inspected by RPC.

A comparison of bobbin and RPC results of more than 400 tubes seemms to demonstrate that RPC indications at or above the top of the tubesheet will be detected by the bobbin probe. It is the crevice RPC indications intentionally remaining in service and the projected indications in unsleeved tubes not inspected by RPC which are the subject of this report.

1.2 Evaluation Approach

In order to evaluate the effect of the tube degradation within the crevice, the flow rate of water through a crevice from a through wall crack must be considered. The evaluation demonstrates that a tube with such degradation can not burst and that the restriction offered by the crevice limits the flow rate from a through wall crack. The administrative operating leak rate limit establishes the maximum length of a single, through wall crack which may be inservice during normal operation. For the purposes of the analysis, all of the remaining indications projected to remain in service are postulated to have developed a depth where they would be opened up during a steam line break. The total calculated primary to secondary flowrate under postulated steamline break conditions from this assumption is less than the maximum primary to secondary leak rate acceptable during a steam line break. The maximum acceptable leak rate is established based on a conservatively modified radiological evaluation of the steam line break analysis in the USAR.

1.3 Tube Degradation Characterization

A review of axial indication profiles of many RPC traces from the crevice region basically shows that these indications cover some length of the tube axially. These indications generally taper (in terms of volume) from a null base condition to a higher volume along the axial length and then taper back to the base condition. Bobbin coil differential signals are generated only when there is a reasonably abrupt change in volume (or depth) along the axial length of an indication. Because the differential coil pair are self-comparing within a very short axial distance, any indication that has a gradually changing volume (or depth) may not be detected in the differential mode. The RPC inspection was the primary probe for identifying the axial indications. Further laboratory studies may identify techniques, such as signal drift in the absolute mode, to aid in the identification of indications using the bobbin coil probe. For the purposes of the tube integrity evaluation in this report the indications are considered to be axial oriented cracks based on the RPC indication features.

2.0 TUBE INTEGRITY ANALYSIS

This section provides the Kewaunee tube integrity evaluation for crack indications within the tube to tubesheet crevice.

2.1 Tube Integrity Requirements

General Design Criteria 14, 15, and 31 of 10 CFR Part 50 Appendix A specify the design requirements for protection against abnormal leakage, rapidly propagating failure, and gross rupture of the reactor coolant pressure boundary. The tube integrity requirements are defined by Reg. Guide 1.121 together with satisfaction of USAR Chapter 14 accident analyses for allowable radiological consequences. The tube integrity analysis presented in this section demonstrates that the plant will remain within the guidelines of the General Design Criteria for RCS integrity.

For thinned or unthinned tubes with partial or through wall cracks, the Reg. Guide 1.121 criteria can be summarized as:

- Cracks should not burst under accident conditions

- The maximum permissible length of the largest single crack should have a burst pressure greater than 3 times normal operation differential pressure

- The Plant Technical Specification leakage rate should be less than the leakage rate determined for the largest permissible crack

The USAR accident analyses evaluate the radiological consequences to satisfy 10CFR20 and 10CFR100 criteria. For accidents such as a Steam Line Break (SLB), the release of radioactivity is the result of tube leakage in the steam generators. Thus the tube integrity requirements must provide that the SLB leakage limits are satisfied. These leak rate criteria are developed in Section 3 of this report and results in a requirement to limit SLB leakage to 190 gpm.

In the case of cracking within a tubesheet crevice or a tube support plate (TSP) intersection, testing has shown that tube burst does not occur within the TSP even with nominal tube to plate gaps. Within a tubesheet or TSP crevice, all burst requirements of Reg. Guide 1.121 are inherently satisfied by the reinforcement provided by the tubesheet or TSP. However, the USAR chapter 14 leakage requirements for radiological consequences during accident conditions must still be satisfied.

Based on the above requirements, the governing requirement for tube integrity of the Kewaunee tubesheet crevice indications is to show that the SLB leakage limit of 190 gpm is satisfied. This evaluation is given in Sections 2.3 to 2.6.

2.2 Tube Burst Considerations

A tube within the tubesheet cannot rupture because of the support provided by the tubesheet. This support is a result of the tube expansion under high pressure to engage the tubesheet. The nominal diametral annular gap between the unexpanded tube and tubesheet is [ ]^{a,c} inch whereas the radial displacement for an unrestrained and undegraded tube under high pressure would be on the order of 0.080 inch at the time of rupture. A degraded tube would deflect more readily prior to rupture (albeit at a lower pressure). Therefore, the plastic, radial deformation of the tube allows the tube to expand to the tubesheet where the support needed to prevent rupture of the tube is provided.

This result has been demonstrated in laboratory tests of degraded and undegraded tubing pressurized within a test fixture collar that simulates the presence of a tubesheet or support plate. For all tests, rupture of the tubing occurred outside of the section of tubing supported by the collar at a pressure level consistent with the condition of the tube outside of the collar. This result was the same even for tubes with degradation of equal or greater severity in the section of tubing within the collar. Figure 2-1 illustrates this point. Burst tests were conducted for tubes with collars where the annular gap between tube and collar was varied from nominal clearance to various levels of squeezing of the tube (denting). Electro discharge machining (EDM) was used to produce through wall slits (cracks) in the tube that continued through the collar section and extended beyond the collar (exposed crack). The burst test results are shown in the figure to be consistent with unsupported tube burst capability as a function of only the exposed crack length.

2.3 Crevice Leakage Model

This section describes the model that was developed for calculating leakage from a crack located within a crevice. The model adapted an existing code for this purpose. Results of some of the calculations performed are included in the next section.

Code Overview

The following is a brief discussion on the flow and pressure drop characteristics assumed in the model.

The analytical model assumes one-dimensional flow and accounts for crack entrance pressure losses, tube wall friction, and flashing. As the flow enters the crack from the primary side it encounters a sudden reduction in flow area. This change in flow area results in a pressure change which is modeled by an empirically based discharge coefficient. Beyond the vena contracta and point of attachment to the crack wall, flashing and friction predominate. The flashing of liquid to vapor generates an acceleration type pressure drop. The combination of surface roughness and number of turns in the flow path determines the friction loss. The overall pressure drop, therefore, is given by the sum of pressure losses due to area contraction, acceleration, and friction. This pressure drop determines the pressure at the exit of the crack. For non critical flow conditions, the exit pressure equals the secondary side pressure. For critical flow conditions, however, the exit pressure will be higher than the secondary side pressure. Obviously, the crack leakage flow will depend on whether or not critical flow conditions exist.

Critical flow is evaluated according to Henry's non-equilibrium formulation (1, 2). This method accounts for non-equilibrium effects due to finite evaporation rates. This is expected to be particularly important in flow through short cracks, where the fluid transit time is short. In essence, this approach corrects for the deviation between the measurements (non-equilibrium) and the flow rates predicted by the homogeneous equilibrium model.

Essential to the prediction of leakage flow rate is the evaluation of crack opening area. The crack opening area is determined by the crack length times the crack opening displacement (C.O.D.) which is calculated from equations as developed by Paris-Dugdale (3). The Paris-Dugdale formulation is based on the methods of linear elastic fracture mechanics which require the stress intensity factor for each problem. The effect of yielding near the crack tip is incorporated by plastic zone corrections in which the crack opening displacement is replaced by an effective C.O.D. As such, the C.O.D. is a function of mean tube radius, tube wall thickness, pressure difference, crack length, Young's Modulus, and flow stress (1/2 * (yield + ultimate stresses)). Thus this model adjusts the crack opening as appropriate for normal operation and steam line break pressure differentials.

Application in Crevices

Since the program already contains a flow resistance model, this model was used to reflect the flow resistance of the packed crevice outside the crack. The code was first used to calculate the crack width with unrestricted flow at the crack outlet. These calculations determined the crack dimensions for a range of crack lengths. The crack dimensions were calculated for normal operating and steam line break conditions.

The flow resistance of a packed crevice was simulated by an assumed annular gap between the tube outside surface and the crevice packing. For a given crack, the equivalent added friction loss was calculated in terms of equivalent length to hydraulic diameter ratios (L/D). This was added to the crack resistance and the code was rerun to determine the flow rate when the crack was located within a crevice. Since the crevice is not well defined, sensitivity was assessed with respect to annular gap thickness and depth of the crack below the tubesheet surface.

2.4 Leakage Analysis Results

This section describes results from a crevice leakage sensitivity study to assess the effects of crevice gaps, crack length and depth of the crack in the crevice. These results are compared with data from a Kewaunee leaking tube in 1989 for which both bobbin coil and RPC inspection data are available.

The Kewaunee tubesheet crevices are packed with sludge which reduces leak rates compared to an open crevice. Crevice packing is supported by visual examinations of tubes pulled from Kewaunee and of the tubesheet tube hole after pulling of one of the tubes. Difficulties in pulling tubes from Kewaunee, as indicated by very high pull forces, also indicate the likelihood of tightly packed crevices. In addition, the presence of small dents at the top of the tubesheet can be attributed to hard magnetite with tight gaps. For this assessment, the packed crevices are analytically simulated by small annular gaps of 1.0 and 0.1 mils as compared to the nominal tube to tubesheet radial gap of [ ]^{a,c} (10 mils used for analyses).

The dependence of leak rates on crack length for 0.2, 0.4 and 0.7 inch through wall cracks are also evaluated. Crevice cracks located 5 and 10 inches below the top of the tubesheet were assessed. The results indicate only about a 20% difference inpredicted flow rates between cracks located 5 and 10 inches below the top of the tubesheet and these results are not further discussed in this report.

Figure 2-2 shows the results for crevice leakage versus crack length under normal operating primary to secondary pressure differentials. The results indicate that a crevice of 10 mils behaves essentially as an open crevice. A crevice of 1 mil shows significant reduction in leakage for crack lengths greater than 0.4 inch. Further decreases in the effective gap such as the 0.1 mil gap case show large reductions in the leak rates. Similar trends are shown in Figure 2-3 for SLB pressure differentials.

Both Figures 2-2 and 2-3 show an asymptotic leak rate as crack length increases above about 0.7 inches for gaps of 1 mil or smaller. For the 1 mil gap case with large crack lengths, the leak rate approaches 0.35 gpm (500 gpd) for normal operation and 0.5 gpm for SLB conditions. In these cases, the tight gap restriction to leakage flow is more important than the restriction of the flow through the crack such that leak rates are bounded for long through wall cracks.

Figure 2-4 shows the ratio of SLB to operating leak rates for the 1 mil crevice gap. This ratio is less than 2 for tight crevices of 1 mil or less and crack lengths greater than 0.4 inch. This result together with the asymptotic, limited leakage for large cracks indicate limited leak rates for tight crevices even under SLB conditions.

Kewaunee had a plant leak rate that peaked at 0.08 gpm (118 gpd) prior to the 1989 refueling and inspection outage. The leakage was traced to tube R32C29 in S.G. B which had cracks within the tubesheet crevice. Bobbin coil inspection results showed depths ranging from 84% about 4 inches above the hot leg tube end to 98% at 13 inches. The tube was also inspected using the RPC probe. Figure 2-5 shows a relatively small crack at ~ 4" above the tube end compared to the large cracks shown in Figures 2-6 and 2-7 at higher elevations in the crevice. The crack indications of Figures 2-6 and 2-7 are typical of through wall cracks. Typical pulled tube results for ODSCC crack morphologies indicate short cracks separated by ligaments rather than very long through wall cracks. The RPC profiles of Figures 2-6 and 2-7 show apparent through wall cracks exceeding 2 inches in length.

Even with the multiple, very long cracks indicated by the RPC profiles, the tube leaked at only 0.0B gpm. This result supports the trends from the above analytical results which show leak rates approximately independent of crack length for tight crevice gaps. The actual leak rate corresponds to a gap of about [ ]^{a,c,e} for the analysis results at large (> 0.7") crack lengths. Thus the [ ]^{a,c,e} model can be conservatively applied to estimate Kewaunee crevice leak rates based on the 1989 leakage experience.

2.5 Kewaunee Crevice Leakage Assessment

Based on the above results, the [ ]^{a,c,e} leakage model can then be applied to relate normal operation leakage limits to the allowable number of tubes with through wall cracks for the SLB leakage limit of 190 gpm. Utilizing Figure 2-2 for leakage from a through wall crack in one tube, an operating leak limit can be related to a corresponding crack length. From Figure 2-3, the crack length can be related to the SLB leak rate for a single tube with an assumed through wall crack. Table 2-1 shows the relation between operating leak limit, crack length and SLB leak rate for operating limits of 100, 200 and 500 gpd. For example, a 200 gpd operating leak limit corresponds to a crack length of [ ]^{a,c,e} which would lead to a SLB leak rate of 0.22 gpm.

The Table 2-1 results can be used to determine the allowable number of tubes with through wall cracks for specified operating and SLB leakage limits. It is assumed that only the longest through wall crack contributes to the operating leak limit. Although it is assumed that the shorter cracks do not contribute to operating leakage, the postulated short cracks are assumed to be the same length for estimating SLB leakage as the crack length causing the operating leakage. This assumption allows for opening of crack ligaments between operating and SL8 pressure differentials.

Table 2-2 shows the relationship between the operating leak limit and the allowable number of tubes with postulated through wall cracks per S.G. for SLB leak limits of 1 and 190 gpm. Since a SLB at Kewaunee would empty only one steam generator, the allowable number of tubes with cracks is to be greater than the projected indications in one steam generator. The B steam generator has the larger level of degradation so it is the limiting case and the numbers presented in this evaluation use the projected number of indications in the B steam generator. For the Kewaunee SLB limit of 190 gpm, it can be seen that an operating leak limit of 200 gpd results in an acceptable number of 860 tubes with through wall cracks in the tubesheet region with packed crevices. Based on similar analyses for an open crevice model, a 200 gpd operating leak limit would permit more than 300 tubes per S.G. with through wall cracks. Extrapolating the percentage of tubes inspected by RPC that were found to have crack indications to all unsleeved tubes, reaults in a projection that about 285 tubes in Kewaunee S.G. 8 might have crack indications. Thus, the 860 allowable tubes with through wall cracks for the planned 200 gpd operating leakage limit conservatively bounds the 285 tubes projected to have potential crack indications.

2.6 Tube Integrity Conclusions

Kewaunee had negligible leakage (~ 2 gpd) during the operating cycle prior to the 1990 refueling outage. This experience together with tube plugging during the outage indicates that no leaking tubes are being returned to service.

Prior Kewaunee operating experience with tube leakage indicates that the tubesheet crevices are packed. This is based on the fact that none of the prior tube leaks, which are attributable to tubesheet crevice cracks, exceeded the Technical Specification limit of 500 gpd. In particular, tube leakage prior to the 1989 refueling outage peaked at 118 gpd and bobbin coil and RPC profiles for the leaking tube indicate greater than two inches of total crack length that is through wall. This Kewaunee leakage experience can be conservatively bounded by the crevice leakage model for a [ ]^{a,c} crevice gap.

The governing requirement for tube integrity evaluations of cracks confined to crevices is that leakage be limited, under postulated accident conditions to acceptable levels for radiological consequences. This conclusion results from the fact that cracks within steam generator tube crevices cannot burst. The reinforcing effects of the tubesheet or tube support plates fulfill the Reg. Guide 1.121 margins against tube burst. The evaluations of Section 3 of this report show an acceptable SLB leakage limit of 190 gpm which is applied as the limit for leakage assessments in this study.

The [ ]^{a,c} gap, crevice leakage model was applied to the planned Kewaunee operating leakage limit of 200 gpd. This was applied to assess the allowable number of tubes with postulated through wall cracks that can leak under Steam Line 8reak accident conditions without exceeding the SL8 leakage limit of 190 gpm. This evaluation shows the acceptability, within the SL8 leakage limit, of approximately 860 tubes with through wall cracks within the tubesheet crevice. This acceptable number of tubes with cracks is well in excess of the projected number of crack indications of any detectable depth remaining in service in the Kewaunee steam generators. The operating leak rate limits will provide margins for significantly more tubes leaking under SLB conditions than at operating conditions.

The simulation of the packed crevice conditions by the [ ]^{a,c} gap leakage model show that leakage rates are relatively insensitive to crack length for cracks longer than about 0.4 inch. For this model, operating leak rates are bounded by about 0.35 gpm and SLB leakage by about 0.5 gpm for large crack lengths. In addition, the ratio of SLB to operating leak rates is less than two for crack lengths greater than 0.4 inch. These trends support limited leakage expectations for postulated accident conditions.

3.0 LEAK RATE CRITERIA

3.1 Review of Accident Analyses

The existing licensing basis as documented in the accident analyses of chapter 14 of the Kewaunee Nuclear Power Plant USAR has been reviewed to determine the limiting condition for primary to secondary leakage during the postulated accident conditions. The accident analyses which explicitly consider transfer of activity from the primary to secondary systems are the steam line break and steam generator tube rupture. Of these, the analysis of steam line break has the greatest primary to secondary pressure differential and was therefore selected as limiting. Consistent with the licensing basis of Kewaunee the effect of additional primary to secondary leakage on the dose assessment does not have to be considered for the remaining postulated accidents. Additionally it is noted that the primary to secondary pressure differential for the steam line break is larger than any other accident analyzed. The effect of secondary to primary leakage during a large or small break LOCA is judged not to have a significant effect on the system response.

3.2 USAR SLB Analysis

3.2.1 Method of Analysis

The Steam line break analyses for the Kewaunee Nuclear Power Plant (KNPP) are described in USAR Section 14.2. Four combinations of break sizes and plant conditions were evaluated for the KNPP.

The case selected for a conservative evaluation was:

The complete severance of the main steam pipe downstream of the flow restrictor, with the plant at no load conditions, and both reactor coolant pumps running.

This case is bounding for the purpose of this study because:

a) The severance of the pipe upstream of the flow restrictor limits the break to inside containment. For a conservative calculation of radiological consequences, a break outside of containment, downstream of the flow restrictor, was selected.

b) The plant at no load conditions provides for the most severe cooldown and maximum steam release, and hence, radiation release.

c) The assumption of offsite power provides for both reactor coolant pumps running, which will maximize any primary to secondary leakage during the steam line break analysis.

It is the intent of this safety analysis to assess the steam line break accident under the conditions of significant primary to secondary leakage. The analysis is performed in a stepwise manner building upon the current Kewaunee licensed analysis for the steam line break accident. In this report, licensed analysis results are identified as "USAR". The case of steam line break with significant leakage is analyzed as a steam line break combined with tube rupture. Results from the analysis of the steam line break steam generator tube rupture multiple accident are identified by "USARSGTR".

It is assumed that the increase in primary to secondary leakage takes place in the steam generator with the broken steam line and is initiated at the time of maximum differential pressure between the primary and secondary systems.

The system transient is analyzed with the DYNODE-P computer code which is the current NRC approved WPSC safety analysis methodology for this accident assessment. Using the current licensed analysis pressure transient results, a steam generator tube break area is calculated to be 7.91×10^{-4} ft^2 at the time of maximum differential pressure between primary and secondary. The leak flow rate caused by this tube break area is 190 gallons per minute. This is conservative compared to the Westinghouse analysis which sets the break flow at 190 gpm at a pressure differential of 2650 psi. The maximum pressure differential experienced by the faulted generator is on the order of one full power pressure differential, due to the concurrent primary system cooldown. The steam generator with the broken steam line experiences this tube break at 10 seconds into the steam line break accident.

3.2.2 Thermal Hydraulic Response

Transient parameter results are plotted in Figures 3-1 through 3-9. Each figure shows a comparison of the "USAR" licensed analysis result to the "USARSGTR," steam line break steam generator tube rupture, multiple accident analysis result.

From the results comparisons, it is evident that most parameters are insensitive to this magnitude of steam generator tube leakage, thus indicating the overall transient is dominated by the effects of the steam line break. For example, at 60 seconds into the transient, the steam break flow for the "USARSGTR" case is only 0.3% greater than the "USAR" reference analysis result. Changes in pressurizer liquid volume, reactivity, core temperature, steam generator level and pressure are also shown to be insignificant. The largest difference observed is in the RCS pressure which, prior to 60 seconds transient time, is less than the USAR reference analysis pressure by approximately 150 psi. Core heat flux also changes slightly but is still well below 10% of nominal core heat flux.

The minimum departure from nucleate boiling ratio (MDNBR) for the steam line break event occurs at a transient time when primary system pressure is in the 800 psia range. By this time in the "USARSGTR" case, pressure is nearly the same as the reference case and heat flux is slightly less than the reference analysis. Therefore, the MDNBR for "USARSGTR" is not changed significantly and remains well above the 1.3 DNBR limit. The steam line break at the exit of the steam generator, upstream of the flow restrictor, has a more severe heat flux and DNBR transient and bounds the steam line event being analyzed here.

Steam break flows do not change appreciably from the licensed analysis and thus total steam release in the "USARSGTR" case will be nearly identical to the licensed analysis. The radiological consequences of the steam line break with a steam generator tube leakage are evaluated in more detail in Section 3.3 of this submittal.

3.2.3 Failure Analysis

The postulation of leakage from primary to secondary systems during the steam line break is included in the original analysis presented in the KNPP USAR. The assumption for dose consequences was an instantaneous transfer of 1% failed fuel inventory to the faulted steam generator. The reanalysis of the system response verified that no new or unexpected challenges to plant systems will occur due to specifically accounting for a leakage on the order of 190 gallons per minute during the SLB event. Therefore, the postulation of additional equipment failures beyond that assumed in the USAR is unwarranted.

Furthermore, the probability of such events is extremely small. The probability of additional failures resulting in blowdown of both steam generators is equally remote and is not increased by steam generator leakage. Each steam line in the KNPP has a fast closing stop valve with a downstream check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails (single failure). In order to blowdown both steam generators, it would require a failure of two MSIVs or one MSIV and a nonreturn check valve depending on break location.

A failure analysis was performed to determine the probability of a main steam line break occurring with the blowdown of both steam generators. This analysis was performed in order to quantify the likelihood of main steam line breaks that could challenge the primary to secondary boundary of both steam generators.

The analysis considered both steam line breaks inside and outside containment. For breaks inside containment, the two main isolation valves on each steam line and the stop check valve on the faulted steam line are available to prevent the blowdown of both steam generators. For breaks outside containment, the two main isolation valves are available to prevent the blowdown of both steam generators.

A fault tree was developed to model the probability of having a main steam line break with the blowdown of both steam generators (figures 3-10 and 3-11). The model uses 5.4E-4/year as the probability of the initiating main steam line break, which is an accepted value used in past industry PRAs. The model then factors in the failure rates of the various components to determine the final probability. The failure of the main isolation valves due to common cause is also considered.

The overall probability of a main steam line break occurring with the blowdown of both steam generators was calculated to be 3E-7. The main cutsets for this model are included in Table 3-1. The references for the model's assumed values are also included in Table 3-1.

3.2.4 Summary of SLB Analysis

The steam line break analysis presented in the KNPP USAR bounds the case of a MSLB concurrent with a primary to secondary leakage of approximately 190 gallons per minute. The original USAR assumptions made remain inviolate both from a systems and single failure analysis as well as from a dose assessment standpoint.

The difference in system response is inconsequential as is evidenced by the system response comparisons discussed above. The inclusion of a primary to secondary leak does not, therefore, impact the accident scenario, event timing, or conclusions presented in the USAR. Therefore, a new or different type of accident is not created by primary to secondary leakage.

Total steam release to the environment did not appreciably change and the total curies released would be less for the case of a 190 gpm leak than assumed in the USAR licensing basis of 100% of primary activity with 1% failed fuel. Thus, the consequences are not increased over that previously analyzed.

The accident analysis assumes the proper functioning of some portions of the safety systems.

Based on the failure analysis presented in Section 3.2.3, the probability of occurrence of a malfunction of equipment important to safety is not increased by the postulation of a quantified primary to secondary steam generator leak.

3.3 Radiological Evaluation

The accidents addressed in the Kewaunee USAR that consider primary-to-secondary leakage in the offsite dose calculation include SG tube rupture and steam line break. Of these, the steam line break is most limiting with regard to leakage.

The evaluation of steam line break in the USAR has no explicit primary to secondary leak rate because the assumption is made that all of the primary coolant activity is transferred into the secondary side of the steam generator in the faulted loop. The radioactive material transfer to the steam generator included in the USAR SLB analysis inherently bounds an evaluation using a finite leak size since a primary to secondary leak can not transfer the entire activity in the coolant to the steam generator. However, this essentially infinite leak rate assumption does not provide a useful means of evaluating a condition that may result in a leak. Hence, a radiological evaluation was performed to determine the maximum allowable steam generator primary-to-secondary leak rate following a steam line break.

The evaluation was based on the assumptions used in the steam line break analysis of record presented in USAR Section 14.2.5. The salient assumptions include primary coolant activity corresponding to one percent fuel defects and an iodine decontamination factor of 0.1, for the steam generator in the faulted loop. The offsite dose acceptance criteria was assumed to be 30 rem thyroid, i.e. 10 percent of the 10 CFR 100 guideline. Iodine spiking is not addressed in the Kewaunee USAR Chapter 14 analysis or in the Technical Specifications. Hence, it can be concluded that iodine spiking is not considered in the current Kewaunee licensing basis. The estimated allowable leak rate based on a two hour dose at the site boundary is 1900 gpm, and based on an eight hour dose at the low population zone boundary is 9900 gpm. The leak rate based on the site boundary dose is clearly more limiting and will be used as the basis for the allowable leak rate determination.

The USAR analysis uses a value of 0.1 for the iodine decontamination factor. This is inconsistent with current practices and a more conservative value of 1.0 is used to determine the allowable leak rate for the evaluation of the tubesheet crevice indications. The allowable leak rate for this evaluation based on the assumptions noted above is 190 gpm. It is noted that this radiological evaluation was done solely as a means to provide evaluation criteria for the RPC indications and does not represent an effort to change the licensing basis of the plant.

4.0 EMERGENCY RESPONSE GUIDELINES

The Westinghouse Owners Group (WOG) was originally formed to respond to the post-TMI actions required by the NRC. One of the major contributions of the WOG has been the development of Emergency Response Guidelines (ERGs) for Westinghouse designed nuclear steam supply systems. These guidelines were developed using analysis of the system response of reference plants to various scenarios including primary to secondary leakage during a steam line break. The ERGs were validated by a program documented in Reference 4. The NRC has reviewed the ERGs and issued Safety Evaluation Reports documenting the review. The ERGs of interest for primary to secondary leakage are discussed in References 5, 6, and 7. The ERGs are entered any time that a reactor trip or safety injection occurs or is required, or any time a complete loss of all AC power on the AC emergency busses takes place.

The ERGs of interest are those dealing with steam line break and steam generator tube rupture. The steam generator tube rupture is considered since it is the limiting case of primary to secondary leakage. Since a main steam line break or a steam generator tube rupture causes a plant transient that results in a reactor trip and safety injection, the ERGs would be entered and used to bring the plant to a safe shutdown condition. Specific ERGs have been developed to address a number of SGTR and SLB situations including multiple tube ruptures, multiple steam line breaks (up to and including all steam generators), tube ruptures coincident with loss of coolant, tube ruptures coincident with steam line break (in either the same or different steam generators) and other events both within and beyond the plant design bases. An integral part of the ERG development has been the validation of the procedures on plant simulators. For example, a total of 16 different scenarios were used in the generic validation of the Revision 1 ERGs, including the rupture of more than one tube, the rupture of tubes in different steam generators, and tube rupture with a secondary side break.

Primary to secondary leakage during a secondary break is well within the bounds of the guidance, training and validation provided by the ERGs. The operator response and training to such an event is therefore not an unreviewed safety question. Other events such as steam line breaks in all steam generators with or without an associated steam generator tube rupture are additionally covered by the ERGs.

5.0 OPERATING LEAK CONSIDERATIONS

5.1 Leak Rate Monitoring and Trending

The steam generator tube leakage concern has been reviewed in conjunction with other programs WPSC has implemented to minimize the probability of a steam generator tube failure at the Kewaunee Nuclear Power Plant. The KNPP developed an enhanced primary to secondary leak rate monitoring program in accordance with NRC Bulletin 88-02. This program was found acceptable to the NRC as stated in a letter to C. R. Steinhardt from J. G. Gitter dated March 22, 1989.

In summary, this program utilizes the condenser air ejector radiation monitor (R-15) with alarm capabilities as the primary method to detect steam generator primary to secondary leakage. The R-15 monitor is sensitive enough to detect leakage which correlates to less than a ten (10) gallon per day (gpd) leak.

The accepted leak rate monitoring program requires, when the R-15 monitor indicates an upward trend, the leak rate calculation procedure (RC-C-88) to be performed to determine the primary to secondary leak rate. If the leak rate is determined to be less than 10 gallons per day and remains stable, sample acquisition and analysis will remain on a weekly basis. If the leak rate is determined to be between 10 and 40 gallons per day, condenser air ejector samples will be taken and analyzed daily to determine the leak rate. If the calculated leakage rate is between 40 and 80 gallons per day, air ejector sampling and leak rate determinations will be performed once every eight (8) hours. If the leak rate is greater than 80 gallons per day, the sampling and leak rate calculations will be performed once every four (4) hours. A summary of the frequency of leak rate determinations is presented in Table 5-1.

The results of these leak rate calculations will be utilized to estimate the amount of time available prior to the leak rate exceeding 200 gpd.

In addition to the routine and enhanced monitoring noted above, plant parameters sensitive to primary to secondary leak rate will be graphically trended. These parameters include: total reactor coolant system leakage, reactor coolant system activity, condenser air ejector gas radioactivity monitor response, and steam generator blowdown system liquid radioactivity monitor response.

These trends will be issued weekly and parameters updated consistent with the surveillance frequency. This information will be distributed to all Kewaunee departments and conspicuously posted in the Control Room to increase personnel awareness of potential primary to secondary leakage.

5.2 Administrative Limit Implementation

The objective of these additional leak rate monitoring and trending measures is to provide assurance that the plant is placed in hot shutdown in accordance with normal shutdown procedures prior to exceeding 200 gpd. Although the current Technical Specification Limit of 500 gpd could be supported by the analysis, implementation of an administrative limit of 200 gpd will provide additional margin of safety. This 200 gpd leakage rate will remain in effect until the 1991 refueling outage at which time it will be reevaluated based on the results of the 1991 steam generator inspection. This leakage limit will be incorporated into the following plant procedures prior to achieving initial criticality from the 1990 refueling outage.

A-RC-36D Reactor Coolant Leak
SP 36-082 Reactor Coolant System Leak Rate Check

5.3 Operations Training and Procedures

The discovery of crack-like indications in steam generator tubes in the tubesheet crevice area has given rise to a concern over a postulated main steam line break coincident with primary to secondary tube leakage. One facet of this concern is the ability of KNPP operations personnel to recognize such an event and respond appropriately, in order to bring the plant to a safe shutdown condition and terminate any radioactive releases. The following discussion demonstrates KNPP's readiness in this regard.

The KNPP Integrated Plant Emergency Operating Procedures (IPEOP's) were developed from the Westinghouse Owner's Group Emergency Response Guidelines, Revision 1A. This set of procedures, which has been reviewed by the NRC, provides explicit procedural guidance for event sequences with a probability as low as 10^{-8} per reactor-year (for the initiating event and combined functional failure probability). A main steam line break coincident with a tube rupture is one event, amoung others, for which explicit procedural guidance is provided. For the purposes of this discussion, a tube rupture is defined as primary-to-secondary leakage greater than the make-up capability of the charging system. It does not mean the catastrophic failure of a steam generator tube.

Specifically, the E-3 and ECA-3 series of IPEIOP's provide explicit procedural guidance for operator response to tube failures in combination with other LOCA's or secondary side breaks. Additionally, these procedures provide the operator with appropriate guidance to diagnose the event <u>and</u> to respond to numerous equipment problems (single failures).

The specific procedural sequence an operator would take in response to a main steam line break coincident with a tube rupture would be as follows:

E-0 Reactor Trip or Safety Injection

The operator enters E-0 upon a reactor trip or safety injection signal to verify that the reactor is shutdown and that Engineered Safety Features are operating as designed, and to diagnose the event. At step 22 in E-0, the operators would diagnose a faulted steam generator, and transition to E-2.

E-2 Faulted Steam Generator Isolation

E-2 provides the operator with the guidance needed to isolate the faulted steam generator. After isolation of the steam generator, additional diagnosis is performed. The operator would identify the ruptured steam generator at this point, and transition to E-3.

E-3 Steam Generator Tube Rupture

E-3 provides the operator with common recovery actions for any steam generator tube rupture. There are numerous transition points in E-3 which would cause the transition to ECA-3.1. For example, at step 13, if the ruptured steam generator pressure is less than the 300 psig, the transition to ECA-3.I would be made.

ECA 3.1 Steam Generator Tube Rupture With Loss of Reactor Coolant - Subcooled Recovery Desired

ECA 3.1 provides actions to cooldown and depressurize the RCS to cold shutdown conditions while minimizing loss of RCS inventory and voiding in the RCS. If RWST inventory or the ruptured steam generator level become a concern, the operator would transition to ECA-3.2, which minimizes RCS inventory loss with less concern for voiding in the RCS.

These procedures were developed based on detailed, bounding analyses performed for a "reference plant". These analyses and subsequent validation efforts have shown that the procedures provide the necessary guidance to bring the reactor to and maintain it in a safe shutdown condition.

The KNPP operations personnel have received extensive classroom and simulator training on the IPEOP's, and specifically on the E-3 and ECA-3 series of procedures. The classroom training has included, over the past several years, an Accident and Transient Analysis Course, developed specifically for KNPP and based on KNPP parameters and procedures, and additional classroom instruction on the KNPP IPEOP's and associated background documents.

In particular, the E-3 and ECA-3 series of procedures were discussed in detail during the series of lectures C-CI through C-C3 for the licensed operator requalification program given during the first quarter of 1988. IPEOP background documents, although not the tube rupture series, were discussed again in the first quarter of 1989 requalification sessions.

Finally, and most notably, the E-3 and ECA-3 series of procedures and background information were presented as part of the licensed operator requalification training program (lectures D-Cl through D-C3) during the first quarter of the present year (January 2 through April 6).

Typically, the classroom lectures are followed by simulator exercises which coordinate with the lecture. For example, during the most recent requalification training sessions, two different scenarios involving a main steam line break with a coincident tube rupture were performed by the licensed personnel.

The first was a simulated 0.6×10^5 lbm/hr steam line break (this corresponds approximately to a stuck open safety valve) with a 1000 gpm steam generator tube rupture (SGTR). This was given as a followup to the classroom lecture which covered this scenario. The second scenario was similar, a 1.75×10^5 lbm/hr steam line break with a 500 gpm SGTR. Each crew's performance was evaluated on this scenario.

As a supplement to this training, additional simulator scenarios performed during the current requalification training cycle have included three different steam generator tube rupture events, two main steam line breaks with a steam generator tube rupture in the opposite steam generator, and a main steam line break event.

Based on the detailed procedural guidance and extensive operator training in this area, Wisconsin Public Service has concluded that a hypothetical main steam line break concurrent with a tube rupture would not pose an unacceptable challenge to the skills of KNPP operations personnel.

Nevertheless, to provide additional assurance as to the training of KNPP operations personnel to respond to such a hypothetical event, WPSC will provide additional classroom instruction for licensed operations personnel emphasizing the information in this submittal and operator actions necessary to place the plant in a safe shutdown condition. Simulator scenarios, including multiple failures, will be used to further enhance operator training for this postulated event. This classroom instruction and simulator training will be given during the first series of licensed operator requalification training starting in September, 1990.

Additionally, this information is to be placed in required reading for operations and STA personnel.

6.0 PULLED TUBE INVESTIGATION PLAN

6.1 Areas of Interest

At the end of March, 1990 Wisconsin Public Service removed two hot leg tube segments from Steam Generator B of the Kewaunee Power Plant for examination. Both pulled tubes, tube R11-C9 and tube R4-C81, included only tubing from the tubesheet crevice and top of the tubesheet regions. No tube section from a tube support plate region was removed. The program described in Sections 6.1 and 6.2 is for the examination of these two tubes with the examination concentrating on the indications found by field or laboratory eddy current examination.

Tube R11-C9 was removed because of eddy current indications observed within the tubesheet crevice region. Two axial indications, approximately 1.25 inches long with an undefined depth, were present. The depth was judged to be one of the deepest indications found by the RPC inspection.

Tube R4-C81 was removed because of eddy current indications at the tubesheet top region. An axial indication and volumetric, pit-like indications were observed by RPC inspection.

The intent of the examination is to characterize any defects by both nondestructive and destructive examination techniques, to characterize any surrounding deposits which may help constitute the defect chemical environment, and to provide a defect mechanism definition. Since the tubing was highly elongated during the tube pull, it is presently judged unlikely that any leak rate testing, relevant to safety transients, will be performed.

6.2 Investigation Methods and Activities

The base scope examination program provides for the nondestructive examination of the two pulled tubes and for the destructive examination of one area on each tube (a tube tubesheet crevice region from Tube R11-C9 and the tubesheet top region from Tube R4-C81).

Nondestructive examinations will document the as-received condition of the tube pieces, confirm the field eddy current inspection results in the laboratory, and nondestructively examine the tubes with a variety of techniques to establish the nature and location of the indications. During the nondestructive portion of the examination, activities will include:

* Visual and macroscopic (up to 30X) examination.
* Profilometry and/or dimensional characterization of areas which may have experienced deformation.
* Radiography.
* Photography.
* A review of field eddy current data for the pulled tubes.
* A laboratory eddy current inspection of the pulled tube sections if the sections have not been excessively deformed by the tube pulling operation.
* A review of the Welch Allyn video tapes of the tubesheet crevice and tubesheet top regions of the pulled tubes.

Destructive examinations will study the specific areas of interest, determine the composition of surface deposits, characterize any tube degradation, and assess the extent of any tube degradation. Specific subtasks will include:

* Review nondestructive examination results and establish a plan for the destructive examination.
* Perform a SEM (scanning electron microscopy) examination
* Analyze the undisturbed OD deposits using EDS (energy dispersive spectrometry).
* Metallographic examinations to characterize any degradation. Plating of metallographic specimens may be performed, if appropriate, to preserve surface deposits in order to characterize deposit morphology and the distribution of impurities in any more detailed chemical analyses, such as electron probe microanalyses. Metallographic examinations will include a characterization of the carbide distribution. Hardness testing also will be performed to characterize the tube.

* If judged appropriate to the examination, SEM fractography of the main crack may be partially substituted for metallography.
* If judged appropriate to the examination, sections of the tubing, may be deformed to open-up any ID or OD surface cracks. Crack networks would be mapped by visual examination and photography.
* Perform modified Huey test to characterize sensitization level of the tubing.

Depending upon the base scope work effort results, additional examination will be considered as appropriate.

The results of the laboratory analysis will be submitted to the NRC as they become available. It is anticipated that these results will be available for submittal during October 1990 (approximately mid-cycle). In addition, the results will be reviewed in comparison with this safety analysis so that the analysis assumptions remain valid.

7.0 FUTURE RPC INSPECTIONS

7.1 Development of Analyst Guidelines

In conjunction with the tube pull, other techniques for identifying these generally undetected signals will be evaluated. Some thoughts are that drift of the bobbin signal when in the absolute mode may be an indicator of slowly changing axially oriented indications. This, or other, techniques, if supported by results of the tube pull will be incorporated into future analysis guidelines to allow identification of tubesheet crevice regions that may contain axially oriented indications.

7.2 RPC Inspection Plan Guidelines

During the 1991 refueling outage WPSC will perform an inspection of 100% of the unsleeved/unplugged tubes in the hot leg tubesheet area using the rotating pancake coil inspection technique. The results of the laboratory analysis described in Section 6.0 will be reviewed and, as applicable, used to modify the steam generator inspection program. In order to resolve the long term concern associated with the crack indications in the tube sheet crevice region, WPSC will continue to pursue the steam generator tube sleeving option. Currently, we plan to sleeve additional tubes during the 1991 outage using 27 inch sleeves. WPSC will investigate the feasibility of installing additional sleeves during future outages.

8.0 SAFETY EVALUATION

This evaluation is written to assess the impact on the safe operation of the Kewaunee Nuclear Power Plant of steam generator tubes remaining in service with indications of tube degradation. The criteria of 10CFR 50.59 are used to evaluate whether subsequent operation is an unreviewed safety question.

8.1 Introduction

During the Spring 1990 refueling outage at the Kewaunee Nuclear Power Plant eddy current inspections using a rotating pancake coil (RPC) found indications of steam generator tube degradation in the tubesheet crevice region which were not readily discernible using the standard bobbin coil eddy current probe. As a consequence an expanded RPC inspection found similar indications in several tubes. An evaluation of the indications was made and the indications judged to be most significant were removed from service. Some of the indications were allowed to remain in service to provide a means to assess the growth rate of the degradation during future inspections. In addition to the indications known to remain in service some of the tubes not inspected by RPC are projected to have a condition which would produce an indication if inspected by RPC. Based on the rate of occurrence in inspected tubes, approximately 285 tubes with these known and postulated RPC indications will remain in service in the steam generator with the largest number. An evaluation has been made to assess the impact of leaving the known and postulated indications in service.

8.2 Regulatory Basis

General Design Criteria 14, 15, and 31 of 10 CFR Part 50 Appendix A specify the design requirements for protection against abnormal leakage, rapidly propagating failure, and gross rupture of the reactor coolant pressure boundary. The NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes", issued for comment, addresses tubes with through-wall cracking. The tube integrity analysis demonstrates that the plant will remain within the guidelines of the General Design Criteria for RCS integrity. The Regulatory Guide utilizes safety factors on loads for tube burst and collapse that are consistent with Section III of the ASME Code. Per paragraph C.3.d (1) of R.G. 1.121, the analytical and loading criteria applicable to tubes with through-wall cracks in thinned or unthinned tubes are:

1. Through-wall cracks in minimum thickness tubes should not propagate and result in tube rupture under accident conditions.

2. The maximum permissible crack length of the largest single crack should be such that the burst pressure is at least 3.0 times the normal operation pressure differential.

3. The leakage rate limit under normal operation set forth in the plant technical specifications should be less than the leakage limit determined for the largest permissible crack.

The existing steam line break analysis in the Kewaunee USAR assumes all the coolant activity is transferred to the secondary side and provides a bounding evaluation for the release of radioactive material during an accident as a result of primary to secondary leakage through the steam generator tubes.

8.3 Evaluation

The evaluation of the tubesheet crevice indications considers the structural strength of the tube and the projected leak rate against a steam line break, leak rate criteria. An administrative leak rate limit of 200 gpd supports operation of the plant.

Existing tube repair/plugging criteria, i.e., the current application of R.G. 1.121, apply throughout the tube length and do not take into account the reinforcing effect of the tubesheet. The presence of the tubesheet constrains the tube and complements its integrity in that region by precluding tube deformation beyond the diameter of the tube hole in the tubesheet.

Tube Burst Capability Discussions

The steam generators at Kewaunee have tubes which were mechanically expanded into the tubesheet for a short length at the bottom of the tubesheet. In the remaining depth of the tubesheet above the expanded portion of the tube there is a small annular gap between the tube outside surface and the tubesheet hole. Tubes with through wall axial cracking that is confined to within the tubesheet thickness can not burst as a result of that degradation due to support provided by the tubesheet. The tube to tubesheet annular gap limits the amount of expansion which a degraded tube can undergo to less than that required for a tube burst for any type or extent of degradation. Tube degradation occurring within the tubesheet crevice region can not result in a tube burst condition because of the support provided by the tubesheet.

Tube Leakage Considerations

Although tubes are not expected to burst within the tubesheet under SLB conditions, it cannot be assured that the cracks will not leak during the SLB. Primary to secondary leakage has always been assumed in the Kewaunee licensing basis as evidenced by the assumption of 1% failed fuel inventory being present in the faulted generator for dose assessment considerations. Thus, the maximum amount of leakage permissible during SLB is used as a basis to establish the number of acceptable through wall cracked intersections. Making the conservative assumption that all projected RPC indications grow through wall as a result of the steam line break, the maximum leak rate for each indication can be determined. Dividing this maximum leak rate per indication into the total permissible leak rate during the SLB determines the acceptable number of indications in a steam generator tube bundle that may remain in service and still result in acceptable radiological consequences.

During the operating history of the Kewaunee steam generators the crevices between the tubes and tubesheet have filled with deposits. The deposits are primarily in the form of magnetite and are packed into the crevice very tightly. The evaluation of maximum leakage rates makes the assumption that the leakage is restricted by the presence of deposits in the tubesheet crevice. The leakage is restricted compared to a crack in a free span section of a tube or in an open crevice. Calculations of the effect of a tight crevice have been performed and demonstrate acceptable leak rate for cracks in the crevice region in a large number tubes. The flow restriction used in the analysis has been validated by an evaluation of a tube leak due to a through wall crack in the crevice region found during a previous outage.

To determine the acceptable number of potential cracks remaining in service, the length of individual cracks must be defined. Since a relationship between crack length and leakage rate during normal operation can be established, an administrative leak rate can be established which will limit the length of any through wall crack inservice. The basis of the leak rate analysis is that one indication has grown to a through wall crack sufficient in length to have a leak rate just less than the administrative limit. Furthermore all of the remaining indications are assumed to have grown to be cracks which would open up during a steam line break yet are not contributing to the operating leakage. For the leak rate analysis all of the tubes with projected indications are assumed to leak during steam line break conditions at the same rate as the lead crack which caused the operating leakage.

Radiological Evaluation

The steam line break documented in the Kewaunee USAR assumed that all of the coolant activity was transferred to the secondary system at the start of the accident and did not consider a specific primary to secondary leak rate. Because of this, a radiological evaluation was performed to determine the maximum allowable steam generator primary to secondary leak rate following a steam line break. The evaluation was based on, with one conservative exception, the assumptions used in the steam line break analyses of record presented in USAR Section 14.2.5. The salient assumptions include primary coolant activity corresponding to one percent fuel defects deposited into the steam generator in the faulted loop. The USAR assumes an iodine plate out factor of 0.1. The analysis for the leak rate evaluation used no iodine decontamination factor. The offsite dose acceptance criteria used was 30 rem thyroid, i.e., 10 percent of the 10 CFR 100 guideline. The estimated allowable leak rate which resulted is 190 gpm.

With an allowable SLB leakage rate of 190 gpm, the acceptable number of through wall cracks in the tubesheet crevice region, corresponding to an administrative operating leak rate of 200 gpd, is 860. The projected number of RPC indications of tube degradation in unsleeved tubes is 285 in the steam generator with the larger number of indications. This number of indications is an acceptable number.

8.4 ASSESSMENT OF UNREVIEWED SAFETY QUESTION

The safety significance of both known and potential indications indications of tube degradation in the tubesheet crevice region has been evaluated using the guidance of NSAC-125 and does not represent an unreviewed safety question on the basis of the following justification.

1. Will the probability of an accident previously evaluated in the FSAR be increased?

   No. The accidents of interest are steam line break and steam generator tube rupture. The probability of a SLB is independent of steam generator tube integrity and has been shown to be small. Steam generator tubes with through wall cracking that is confined to within the tubesheet do not burst during normal operation or postulated accident conditions even with nominal tube to tubesheet annular gaps. The criteria of R.G. 1.121 for tube burst are inherently satisfied, even for through-wall cracks, due to the presence of the tubesheet. Therefore, a single tube rupture event is not expected to occur. Therefore the probability of a steam generator tube rupture has not been increased.

2. Will the consequences of an accident previously evaluated in the FSAR be increased?

No. Although tubes are not expected to burst within the tubesheet, it cannot be assured that the cracks will not leak during the Chapter 14.0 accidents discussed in the Kewaunee USAR. As previously noted, the accidents affected by primary-to-secondary leakage and steam release to the environment are SG Tube Rupture (SGTR), and steam line break (SLB). Of these, SLB is most limiting relative to the potential for off-site doses. It has been shown that the projected number of tubesheet indications would not adversely affect these Chapter 14.0 radiological analyses. The leakage postulated in the conservative analysis has been shown to be bounded by the original licensing assumptions. In addition, the conservative dose assessment presented in Section 3.3 demonstrates continued conservatism with respect to 10 CFR 100 limits.

3. May the possibility of an accident which is different than already evaluated in the FSAR be created?

No. As demonstrated in section 3.2, the SLB behavior is not significantly affected by specifically accounting for primary to secondary leakage. Due to the reinforcing effect of the tubesheet, neither a single or multiple tube rupture event would be expected in the Kewaunee steam generators during all plant conditions. The safety issue associated with tubesheet crevice indications which may represent through-wall degradation is primary to secondary leakage during normal, upset, and accident conditions. The implementation of a more restrictive leak rate limit of 200 gpd is expected to preclude the potential for excessive leakage during subsequent plant operation.

4. Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

No. The presence of localized degradation in the steam generator tubes is not expected to affect the overall safety and functional requirements of the Kewaunee steam generator tube bundles. The steam generator tube bundle will continue to sustain with recommended margins, the loads during normal operation and the various postulated accident conditions without loss of safety function. The function of other safety related equipment is not affected.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

   No. The worst case consequences which could occur during plant operation is primary-to-secondary leakage during normal operating and plant transient conditions. It has been shown, for the limiting case of a postulated steam line break event, that the radiological consequences of leakage from the tubesheet crevice indications in the Kewaunee steam generators are acceptable, i.e., the consequences do not exceed a small fraction of 10 CFR 100 limits.

6. May the possibility of a malfunction of equipment important to safety different than already evaluated in the FSAR be created?

   No. As discussed above in response to questions 1, 3, and 4, the steam generator tubes will continue to sustain their overall tube bundle integrity requirements.

7. Will the margin of safety as defined in the bases to any technical specification be reduced?

   No. As indicated within the above evaluation, the conclusions provided within the USAR for steam generator tube integrity remain valid because acceptance criteria are met. Even under the worst case conditions, the growth of the tubesheet crevice indications to a through wall crack could not lead to a steam generator tube rupture and that the most limiting effect would be a possible increase in leakage following a steam line break event. The bases for Technical Specification 3.1.D, RCS Leakage, is not altered. In addition, this consequence of increased leakage has been evaluated for the Kewaunee steam generators conservatively assuming that each indication which remains in service could represent a through wall crack. It has been determined that the number of indications involved would not result in radiological consequences in excess of a small fraction of 10 CFR 100 limits. The bases for the iodine limit in Technical Specification 3.4.A.4 are not altered.

8.5 CONCLUSIONS

Operation of the steam generators in the Kewaunee Nuclear Power Plant for the fuel cycle starting in April, 1990 with known and projected steam generator tube indications in the tubesheet crevice does not represent an unreviewed safety question in accordance with 10 CFR 50.59 criteria.

9.0 REFERENCES

1) R. E. Henry, "Two Phase Critical Discharge of Initially Saturated and Subcooled Liquids," Nuclear Science and Engineering, 41, 1970, p336-343.

2) R. E. Henry and H. K. Fauske, "Two Phase Critical Flow of One Component Mixtures in Nozzles, Orifices, and Short Tubes," ASME Transactions, Journal of Heat Transfer, May, 1971, p179.

3) Tada, Paris, and Irwin, The Stress Analysis of Cracks, Handbook, Paris Productions and Del Research Corp., 1985.

4) WCAP 10599, Emergency Response Guidelines Validation Program Final Report, June 1984.

5) Supplemental Safety Evaluation of the Basic (Revision 0) Version of the Westinghouse Owners Group Emergency Response Guidelines, March 11, 1985.

6) Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulations in the Manner of Westinghouse Owners Group Emergency Response Guidelines, December 26, 1985.

7) WOG Emergency Response Guidelines Supplemental Safety Evaluation of Revision 1, July 7, 1986.

TABLE 2-1

LEAK RATES FOR SINGLE THROUGH WALL CRACK IN ONE TUBE (1.0 mil gap)

OPERATING LEAK LIMIT	CORRESPONDING SINGLE CRACK LENGTH	SLB LEAK RATE
	[ ]a,c	[ ]a,c
0.07 GPM (100 GPD)		
0.14 (200 GPD)		
0.35 (500 GPD)		

TABLE 2-2

ALLOWABLE NUMBER OF TUBES WITH THROUGH WALL CRACKS (1.0 mil gap)

SLB LEAK LIMIT	OPERATING LEAK LIMIT	ALLOWABLE NO. TUBES PER S.G.
		[ ]a,c
1 GPM	100 GPD	
	200	
	500	
190 GPM	100	
	200	
	500	

TABLE 3-1

FT IDENT	COMP	FAILURE MODE	PROBABILITY	IMPORTANCE	REFERENCE
06-MSIVO-----CM	CM	MSIVS FAIL (OC) DUE TO COMMON CAUSE	9.60E-04	88.88	EPRI NP-5613
* 06-PP-MSLA---RP	XX	LARGE STEAM/FEED LINE BREAK IE FREQUENCY	1.25E-04	50.00	PAST PRAs
* 06-PP-MSLB---RP	XX	LARGE STEAM/FEED LINE BREAK IE FREQUENCY	1.25E-04	50.0	PAST PRAs
06-AV-MS1B---FC	AV	MSIV FAILS TO CLOSE	1.09E-02	11.04	IEEE 500
06-AV-MS1A---FC	AV	MSIV FAILS TO CLOSE	1.09E-02	11.04	IEEE 500
06-MSIVI-----CM	CM	MSIVS FAIL (IC) DUE TO COMMON CAUSE	9.60E-07	0.09	EPRI NP-5613
06-CV-NRV1A--FC	CV	CV FAILS TO CLOSE	1.00E-03	0.00	NUREG 4550
06-CV-NRV1B--FC	CV	CV FAILS TO CLOSE	1.00E-03	0.00	NUREG 4550

* The comination of these initiating events in the model results in an overall initiating event probability for a MSLB of 5.0E-4/year.

TABLE OF CUTSETS

CUT SETS FOR GATE G0001 WITH A CUTOFF PROBABILITY OF 1.00E-22

1.	1.20E-07	06-PP-MSLB---RP	06-MSIVO-----CM	
2.	1.20E-07	06-PP-MSLA---RP	06-MSIVO-----CM	
3.	1.49E-08	06-AV-MS1B---FC	06-AV-MS1A---FC	06-PP-MSLB---RP
4.	1.49E-08	06-PP-MSLA---RP	06-AV-MS1B---FC	06-AV-MS1A---FC
5.	1.20E-10	06-MSIVI-----CM	06-PP-MSLB---RP	
6.	1.20E-10	06-PP-MSLA---RP	06-MSIVI-----CM	

ARITHMETIC SUM OF CUTSET PROBABILITIES = 2.699E-07

Table 5-1

Leak Rate Determination Frequency

as a Function of Leak Rate

Actual Leak Rate (L) Determined	Required Frequency of Leak Rate Determination (RC-C-88)
$L < 10$ gallons per day	Weekly
$10 \leq L < 40$ gallons per day	Daily
$40 \leq L < 80$ gallons per day	Once every 8 hours
≥ 80 gallons per day	Once every 4 hours

Figure 2-1

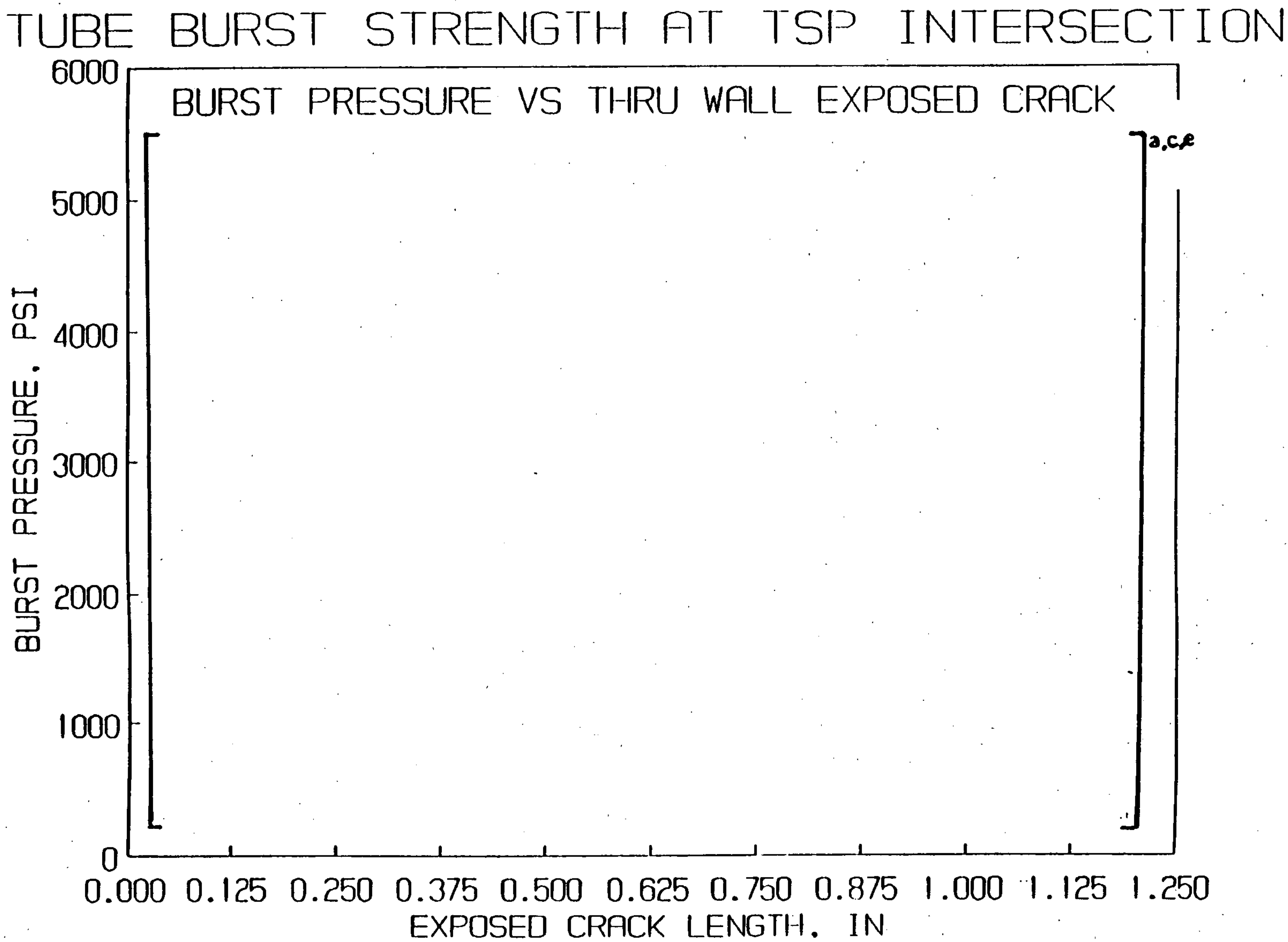

TUBE BURST STRENGTH AT TSP INTERSECTION
BURST PRESSURE VS THRU WALL EXPOSED CRACK
a,c,e
BURST PRESSURE, PSI
6000
5000
4000
3000
2000
1000
0
0.000 0.125 0.250 0.375 0.500 0.625 0.750 0.875 1.000 1.125 1.250
EXPOSED CRACK LENGTH, IN

Figure 2-2

EFFECT OF TUBE SHEET CREVICE ON CRACK FLOW

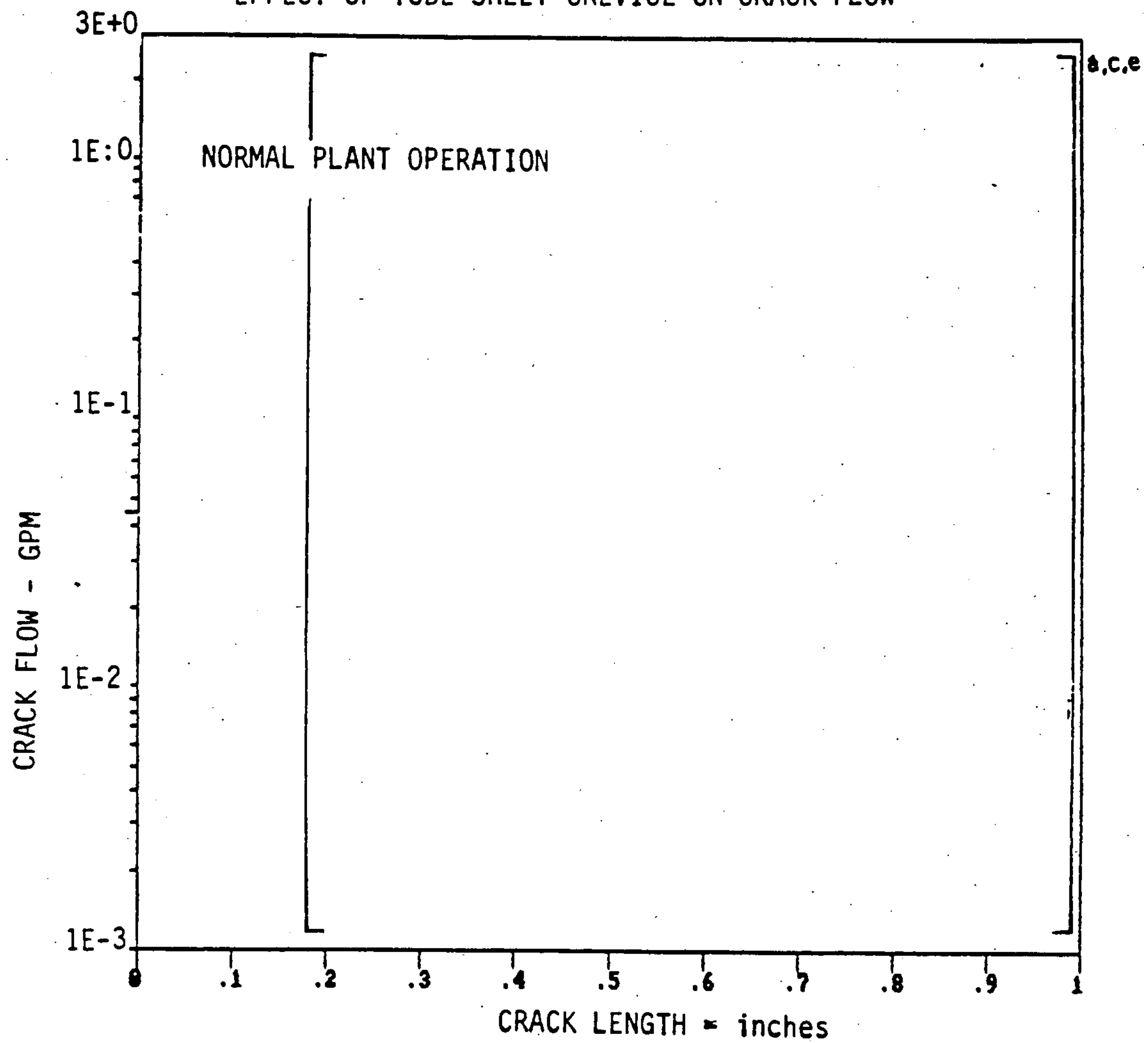
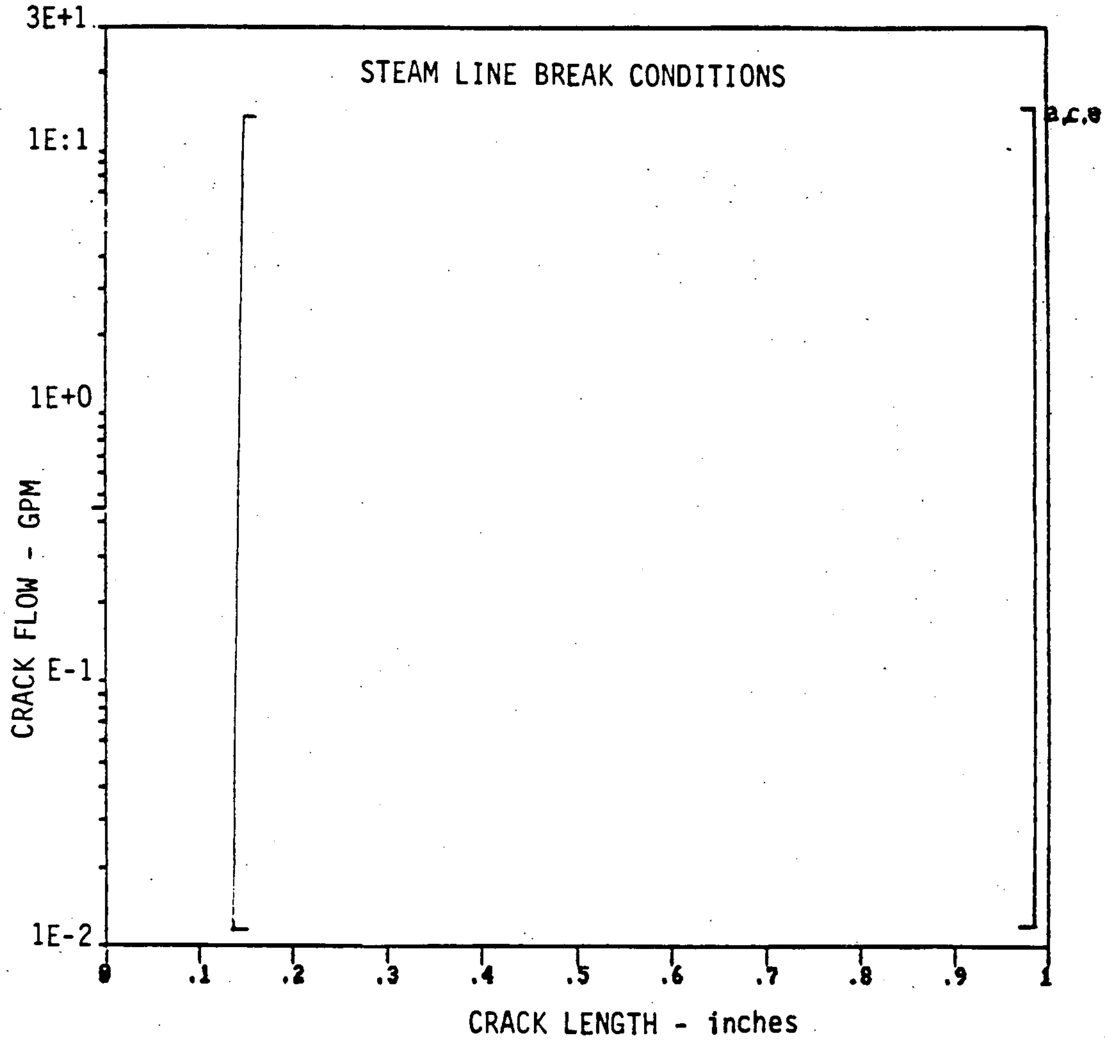
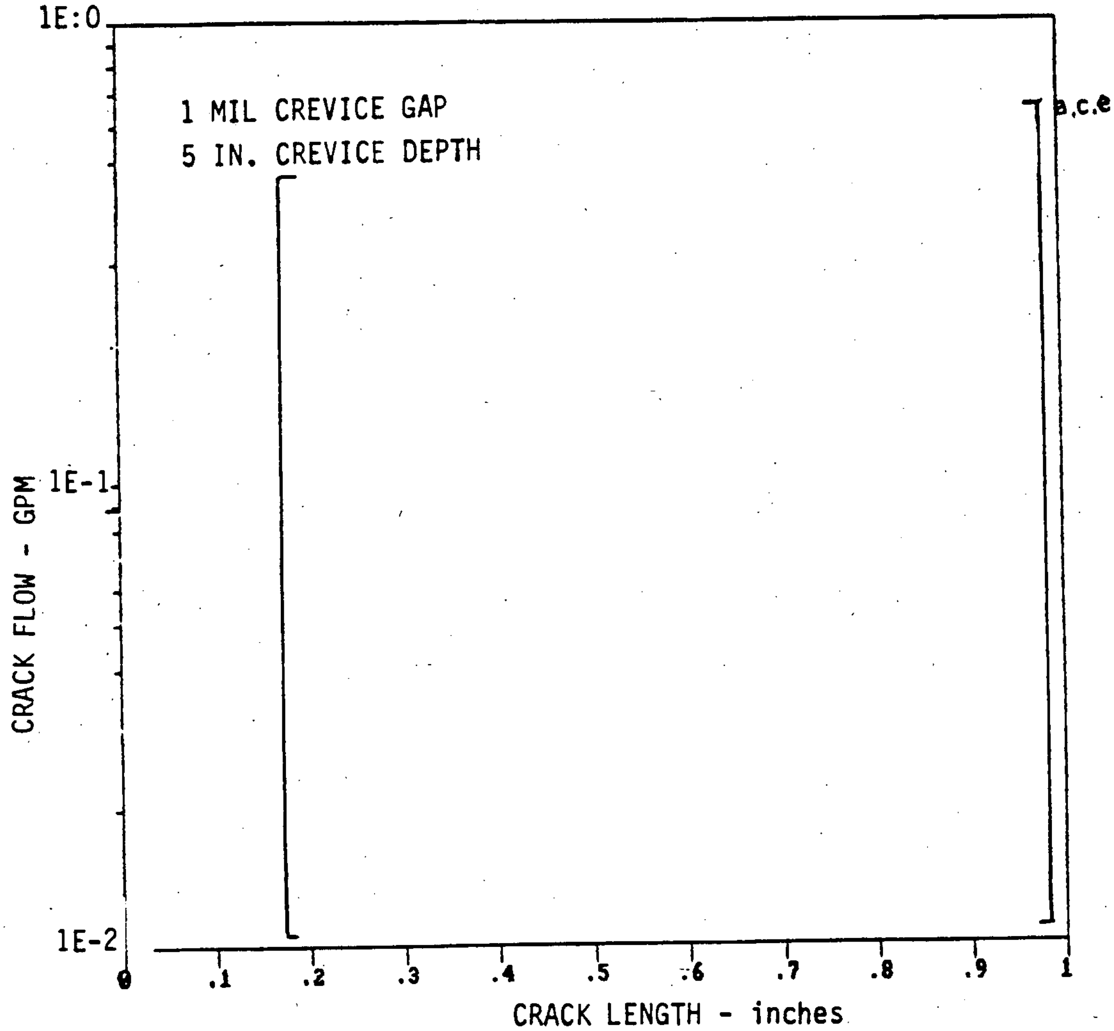


Figure 2-3
EFFECT OF TUBE SHEET CREVICE ON CRACK FLOW



Fiqure 2-4

EFFECT OF TUBE SHEET CREVICE ON CRACK FLOW



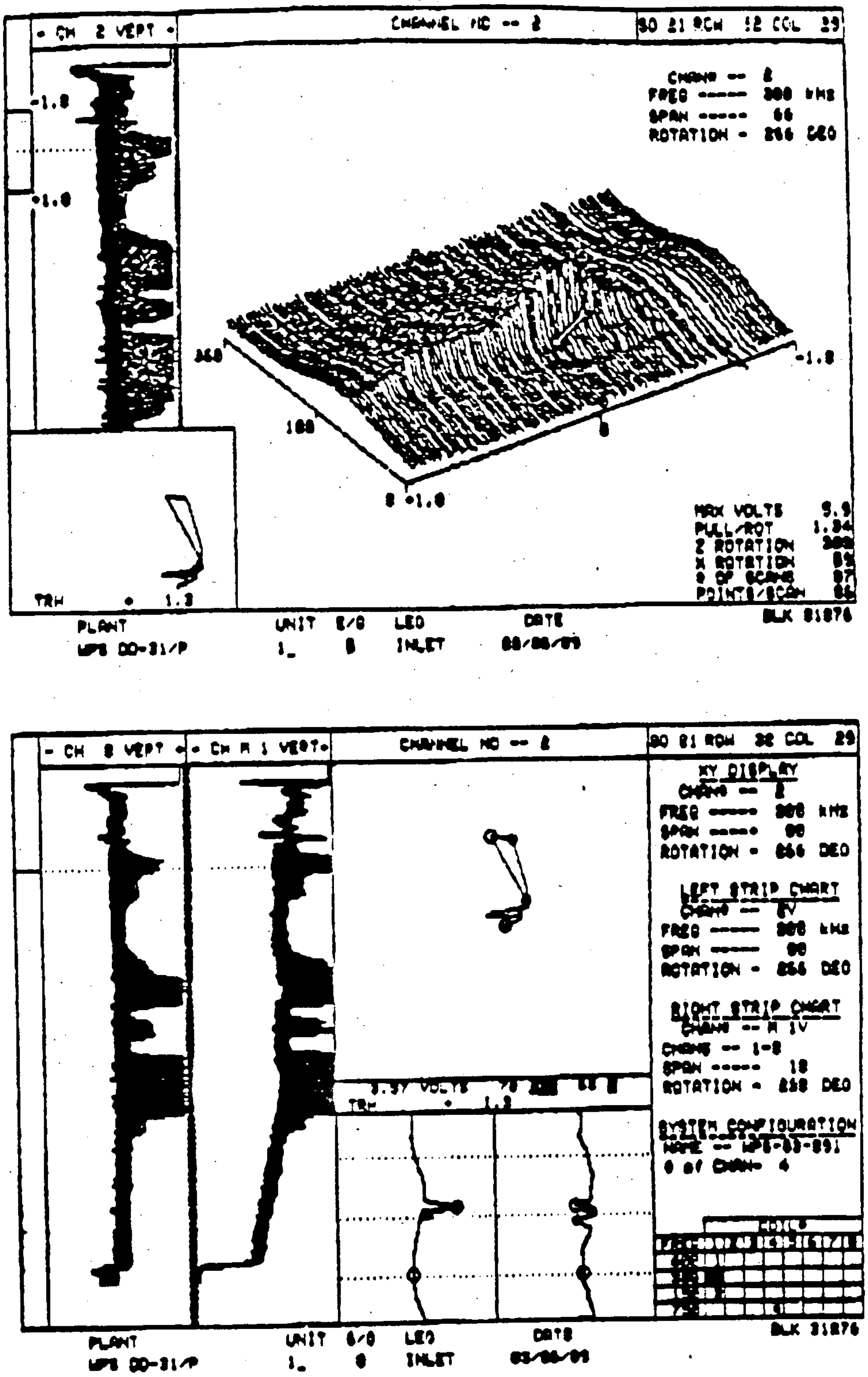


Figure 2-5. RPC Crack Profile at ~ 4" above TEH

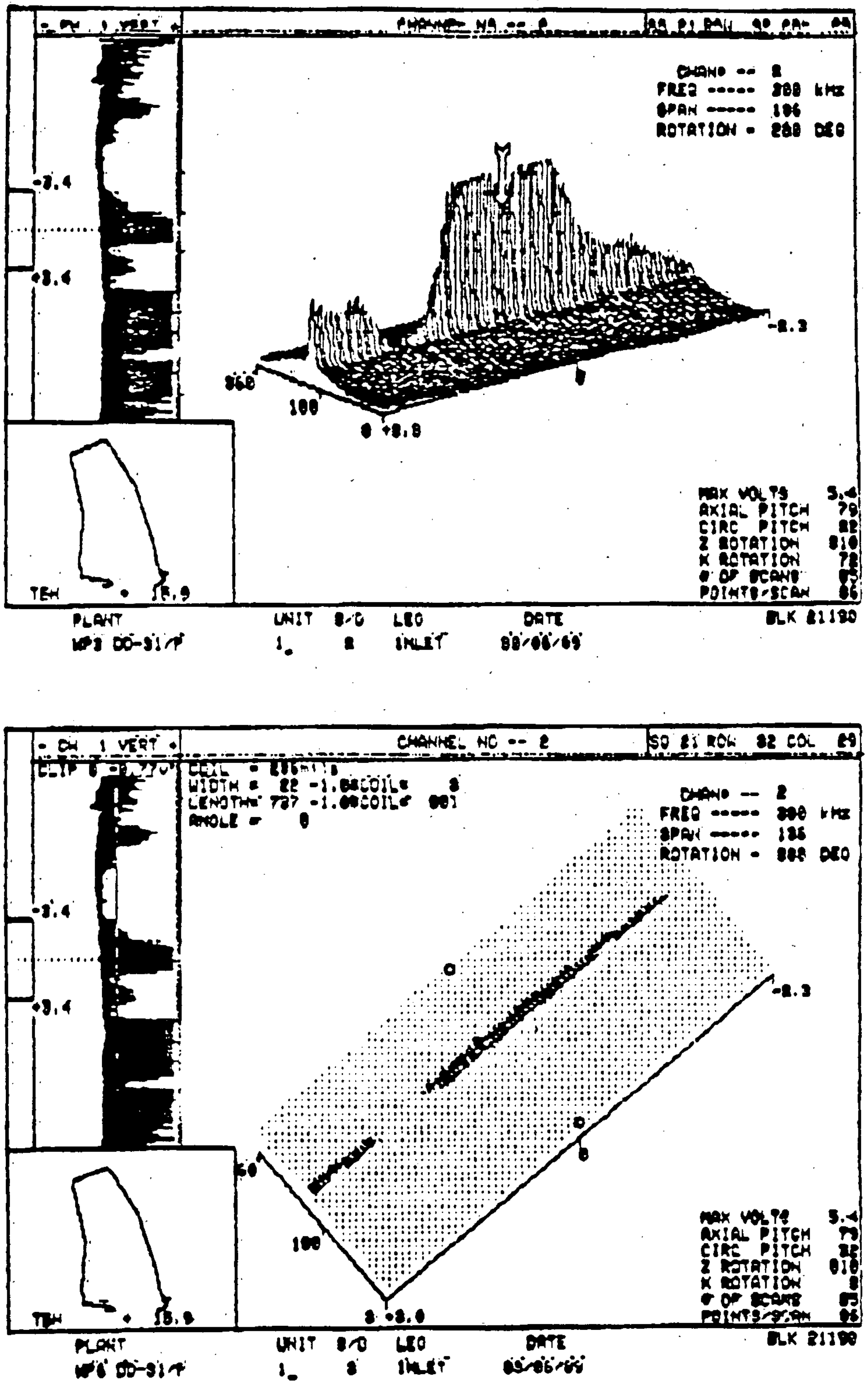
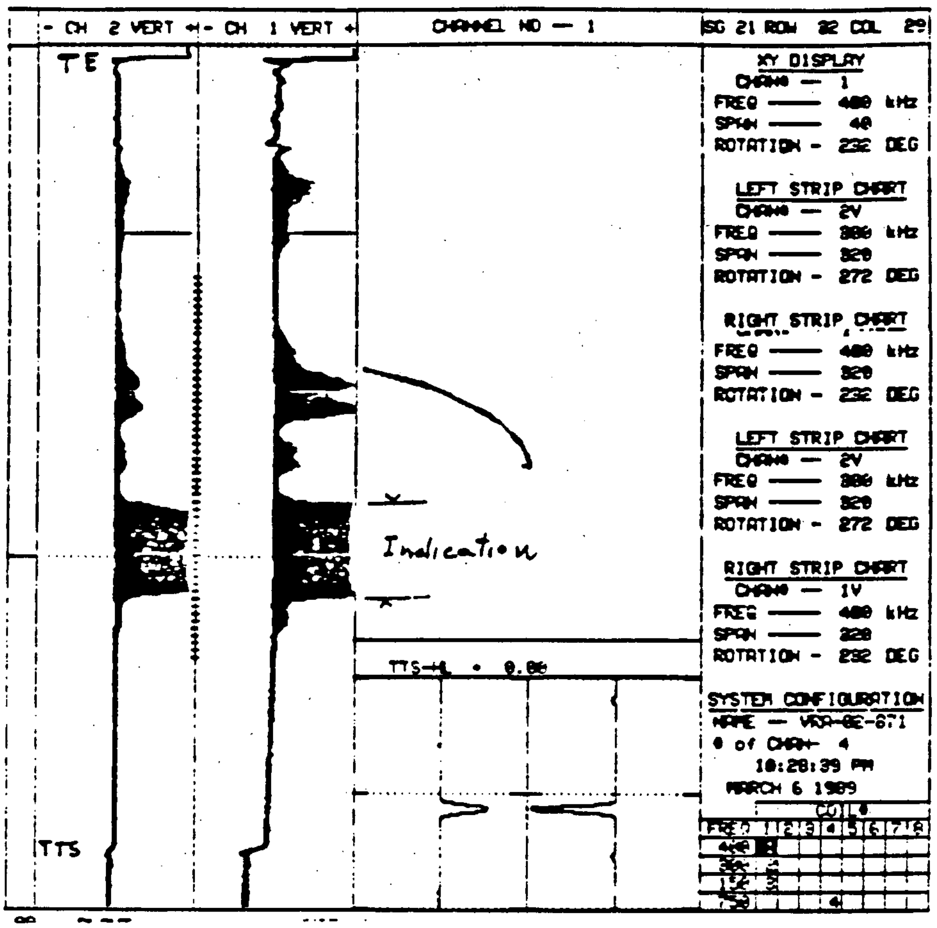


Figure 2-6. RPC Crack Profile at ∿ 9" above TEH



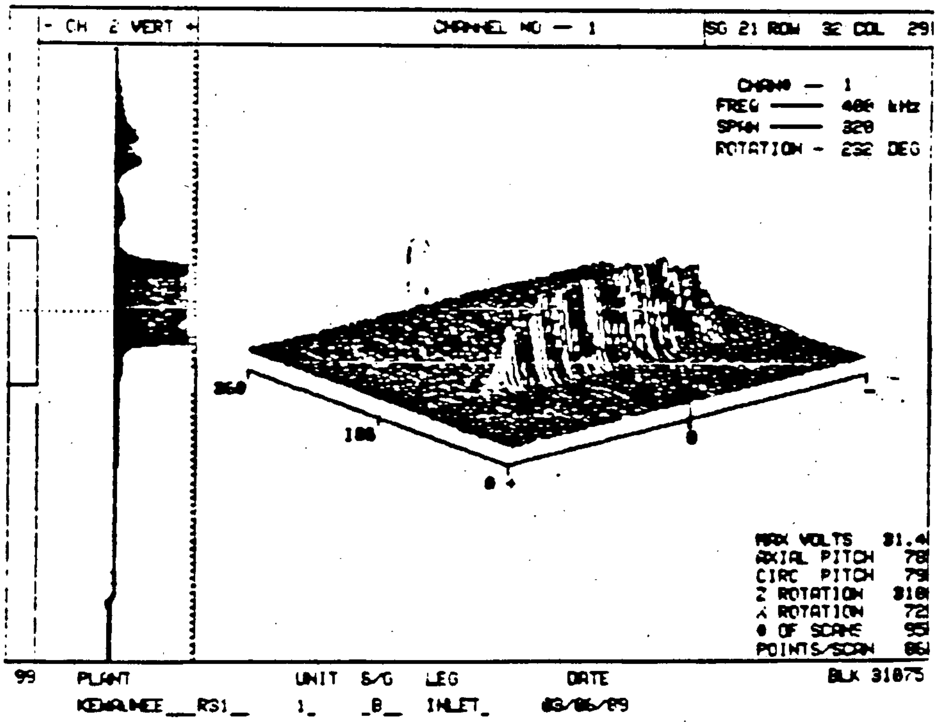
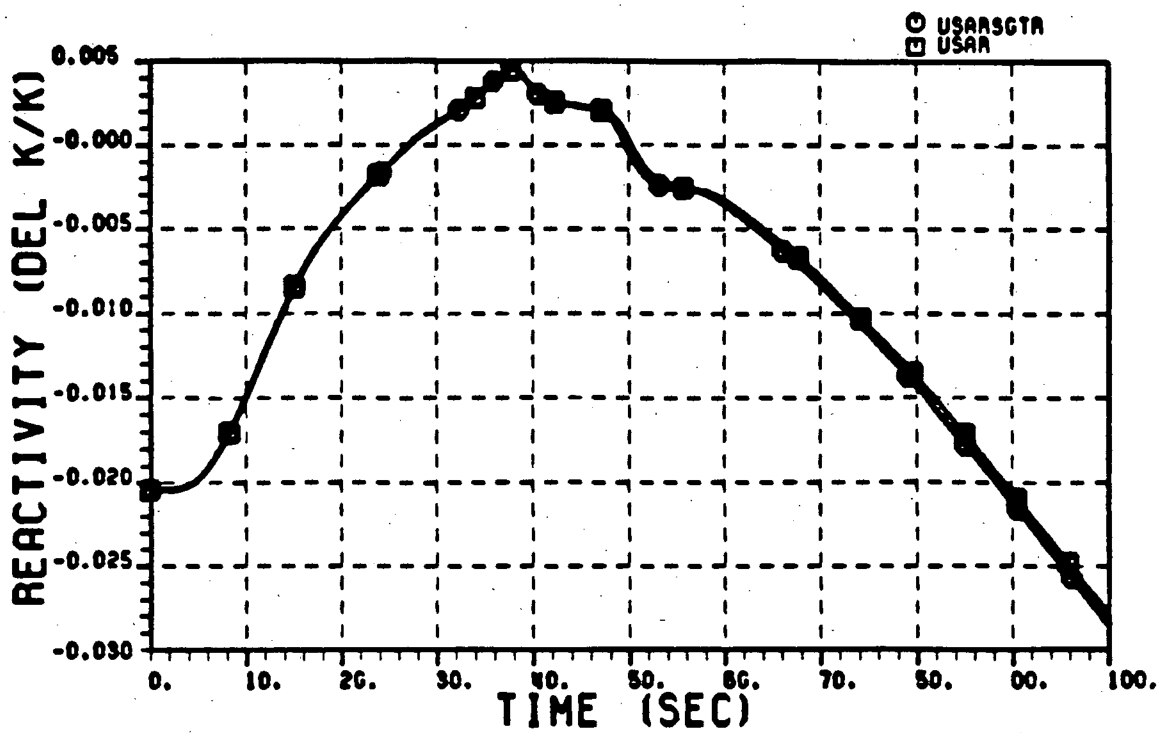


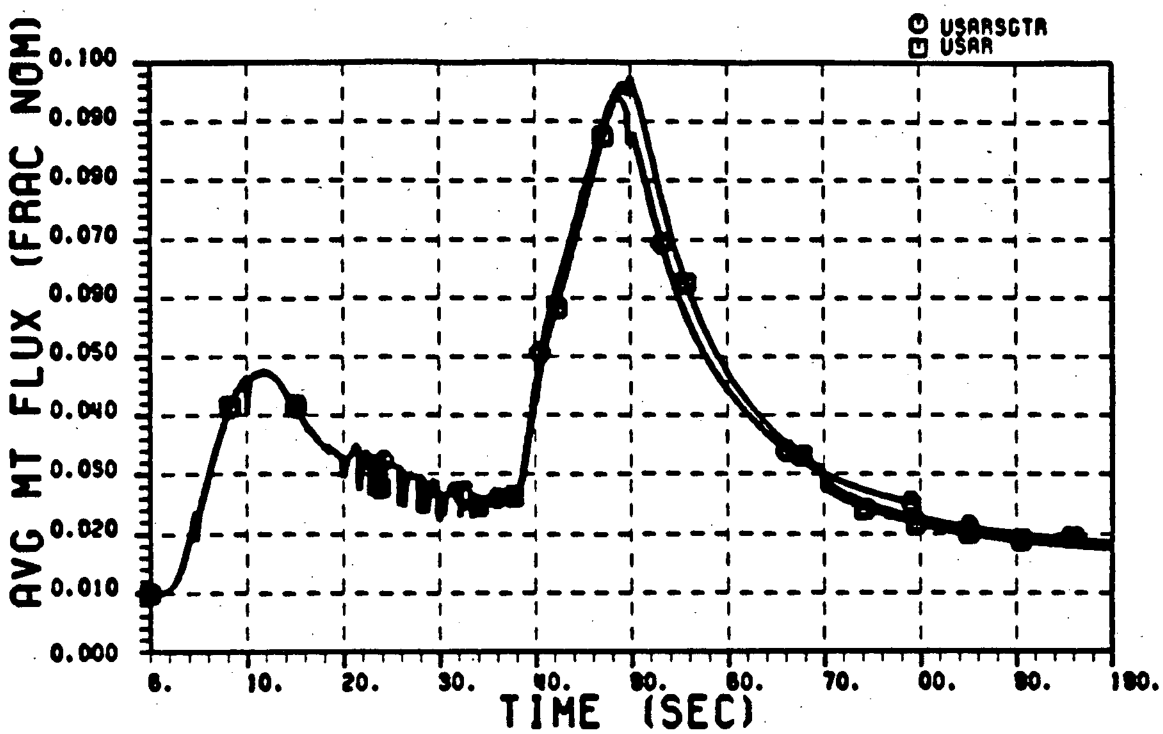
Figure 2-7. RPC Crack Profile at ~ 13" above TEH



STEAMLINE BREAK ANALYSIS

BREAK DOWNSTREAM OF FLOW RESTRICTOR

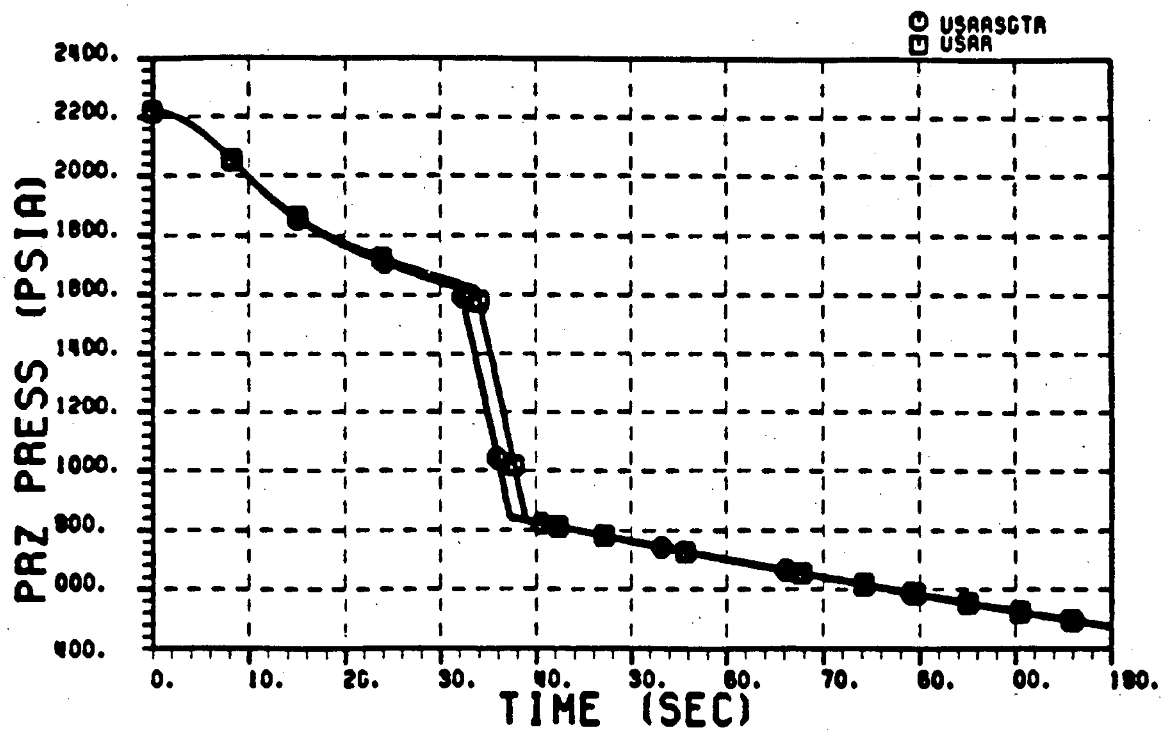
FIGURE 3-1 REACTIVITY



STEAMLINE BREAK ANALYSIS

BREAK DOWNSTREAM OF FLOW RESTRICTOR

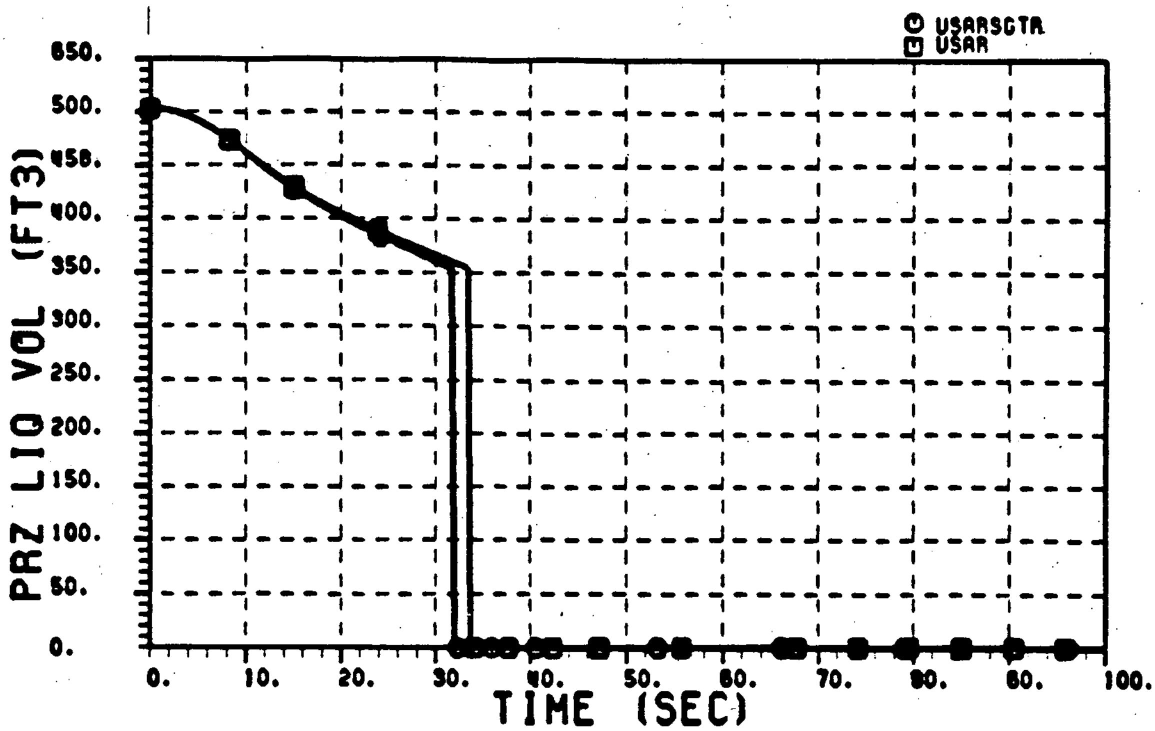
FIGURE 3-2   AVERAGE HEAT FLUX



STEAMLINE BREAK ANALYSIS

BREAK DOWNSTREAM OF FLOW RESTRICTOR

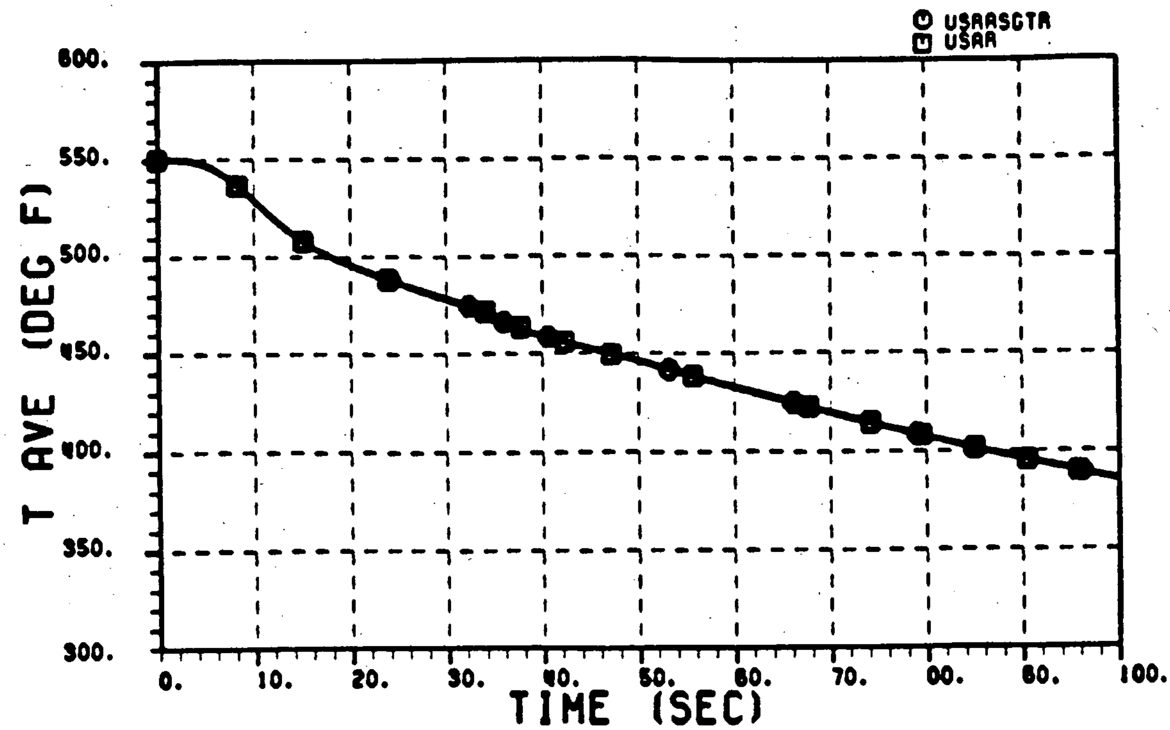
FIGURE 3-3 PRESSURIZER PRESSURE



STEAMLINE BREAK ANALYSIS

BREAK DOWNSTREAM OF FLOW RESTRICTOR

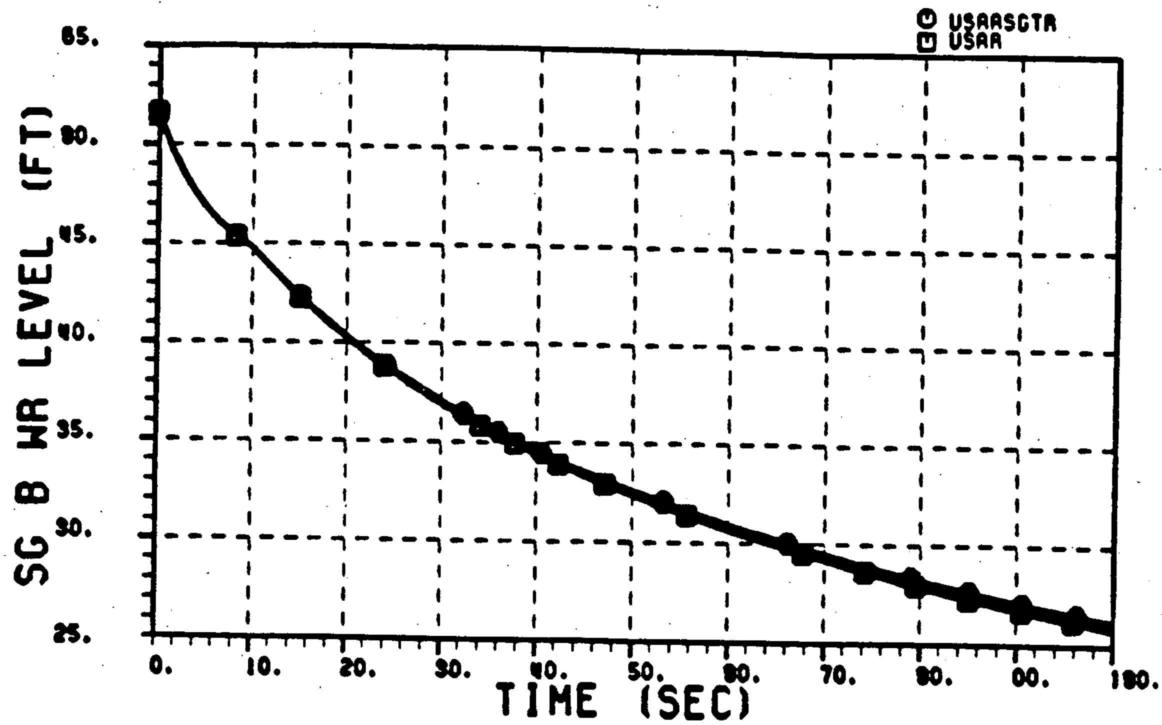
FIGURE 3-4 PRESSURIZER LIQUID VOLUME



STEAMLINE BREAK ANALYSIS

BREAK DOWNSTREAM OF FLOW RESTRICTOR

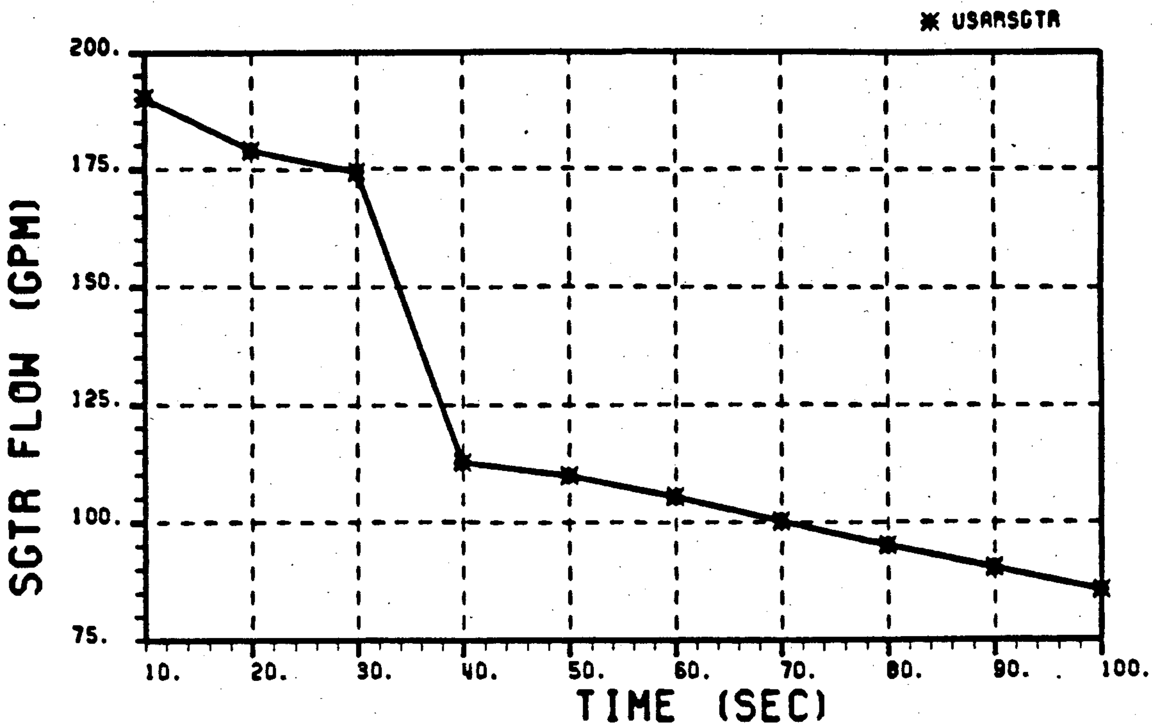
FIGURE 3-5 REACTOR AVERAGE TEMPERATURE



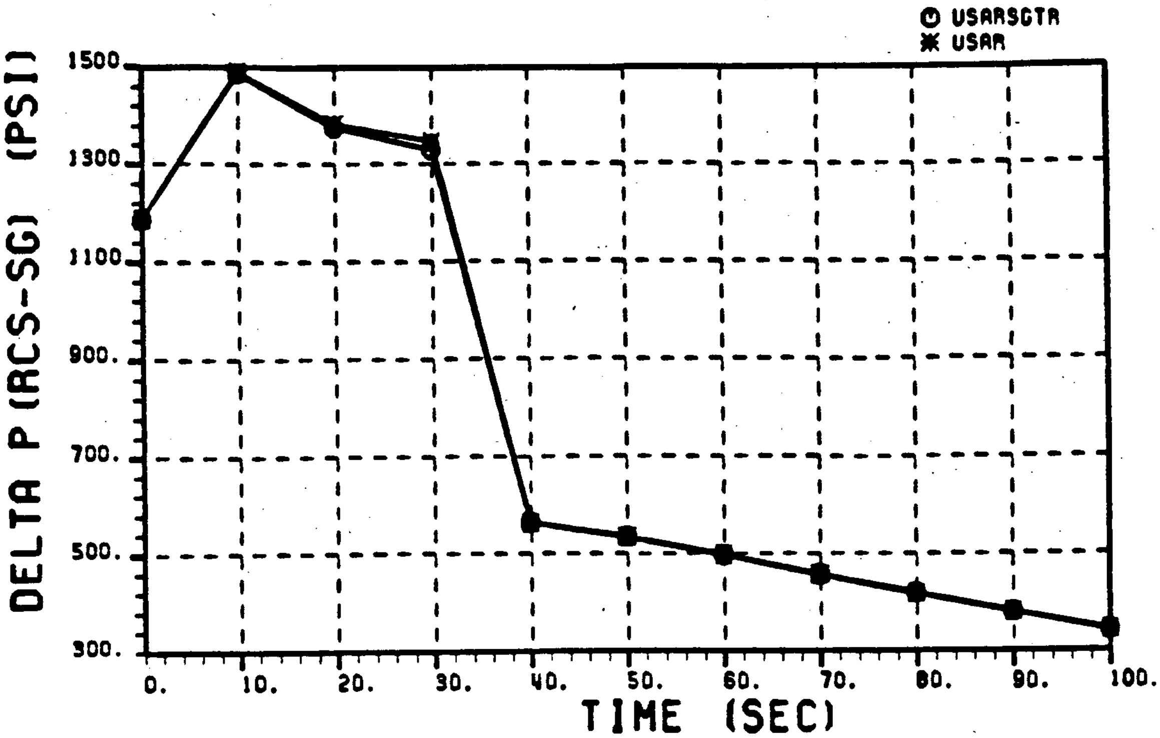
STEAMLINE BREAK ANALYSIS

BREAK DOWNSTREAM OF FLOW RESTRICTOR

FIGURE 3-6 STEAM GENERATOR WATER LEVEL



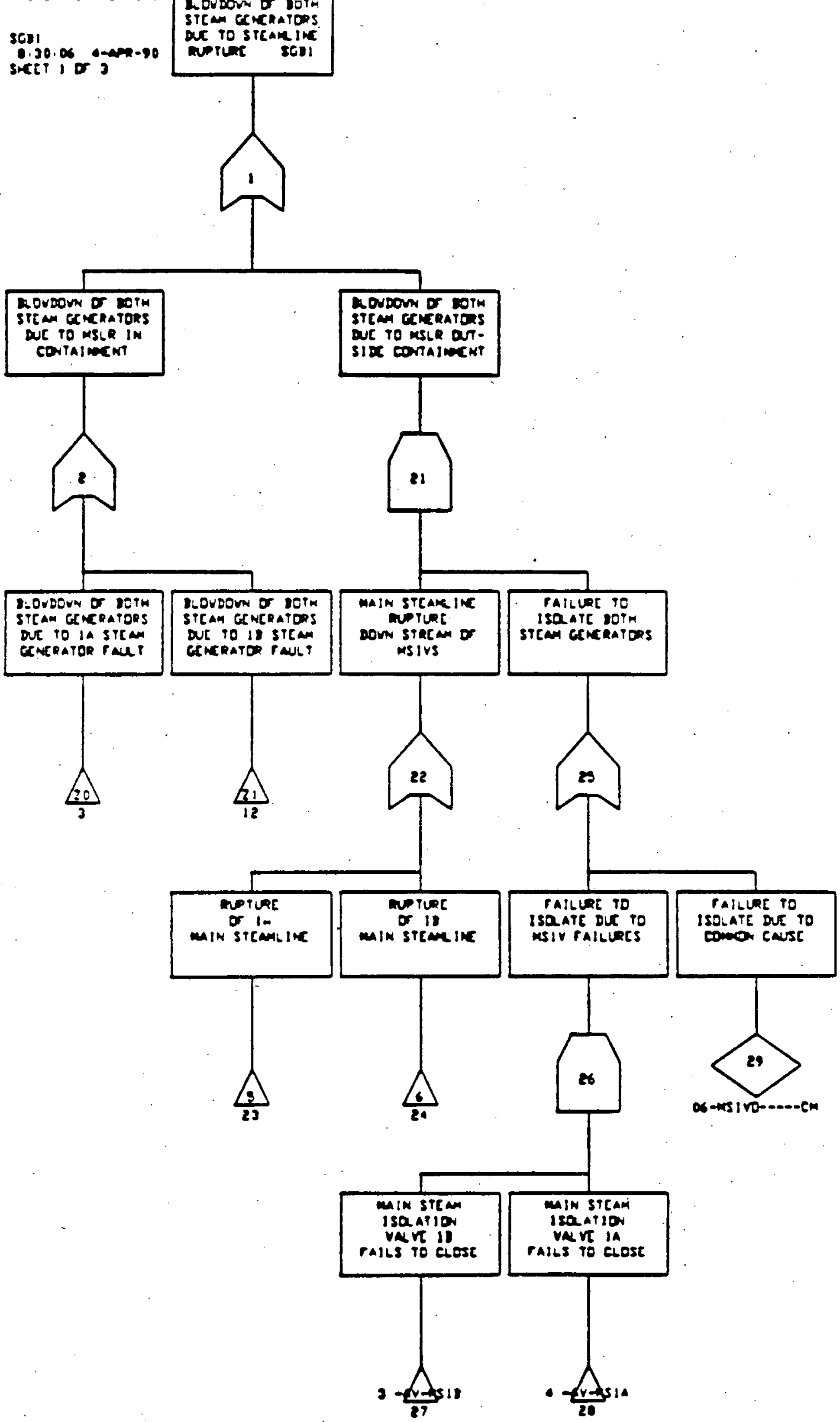
STEAMLINE BREAK ANALYSIS
BREAK DOWNSTREAM OF FLOW RESTRICTOR
FIGURE 3-8 PRIMARY TO SECONDARY LEAKAGE



STEAMLINE BREAK ANALYSIS

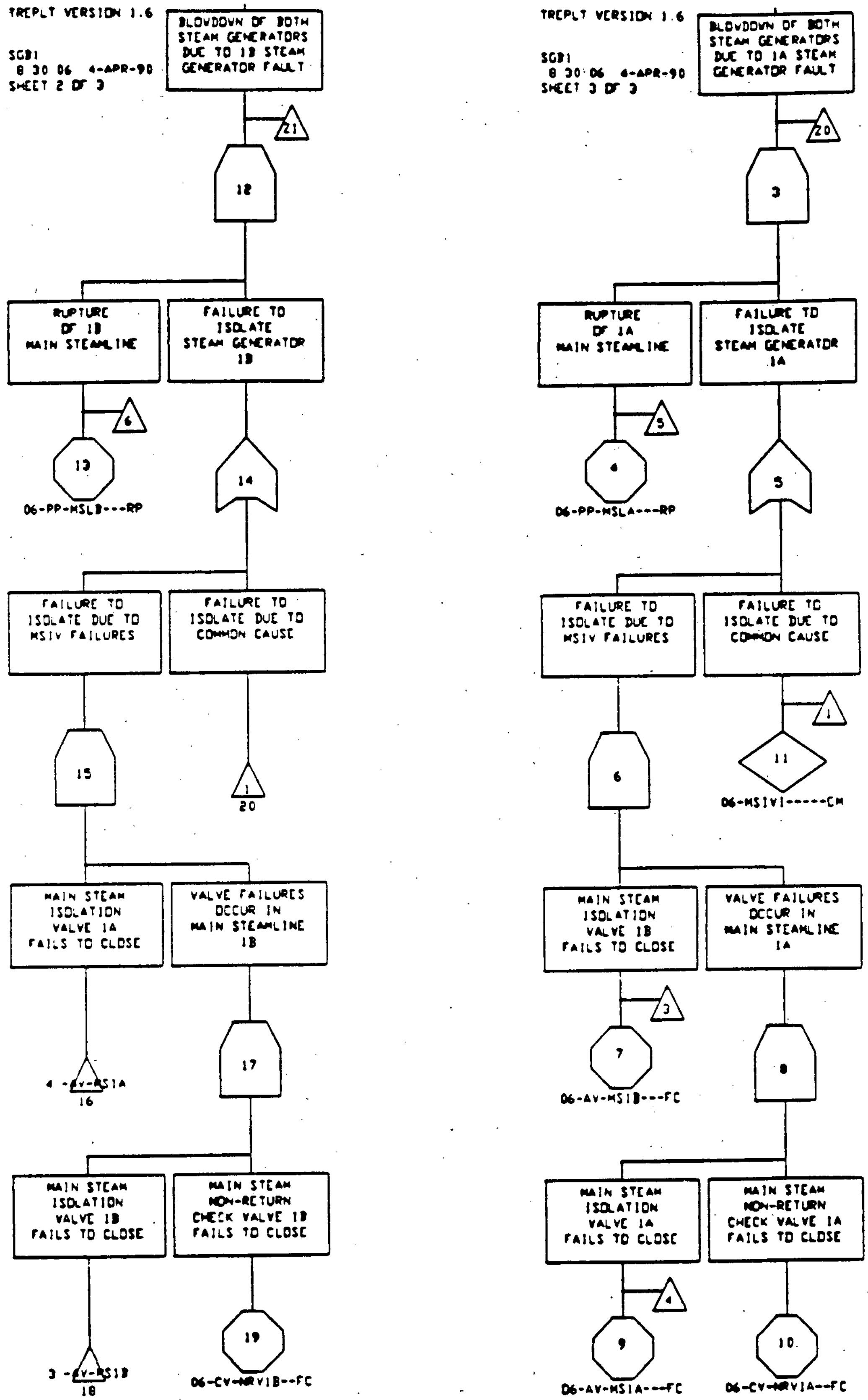
BREAK DOWNSTREAM OF FLOW RESTRICTOR

FIGURE 3-9 PRIMARY TO SECONDARY PRESSURE DIFFERENTIAL



STEAM GENERATOR BLOWDOWN FAULT TREE

FIGURE 3-10



STEAM GENERATOR BLOWDOWN FAULT TREE

FIGURE 3-11