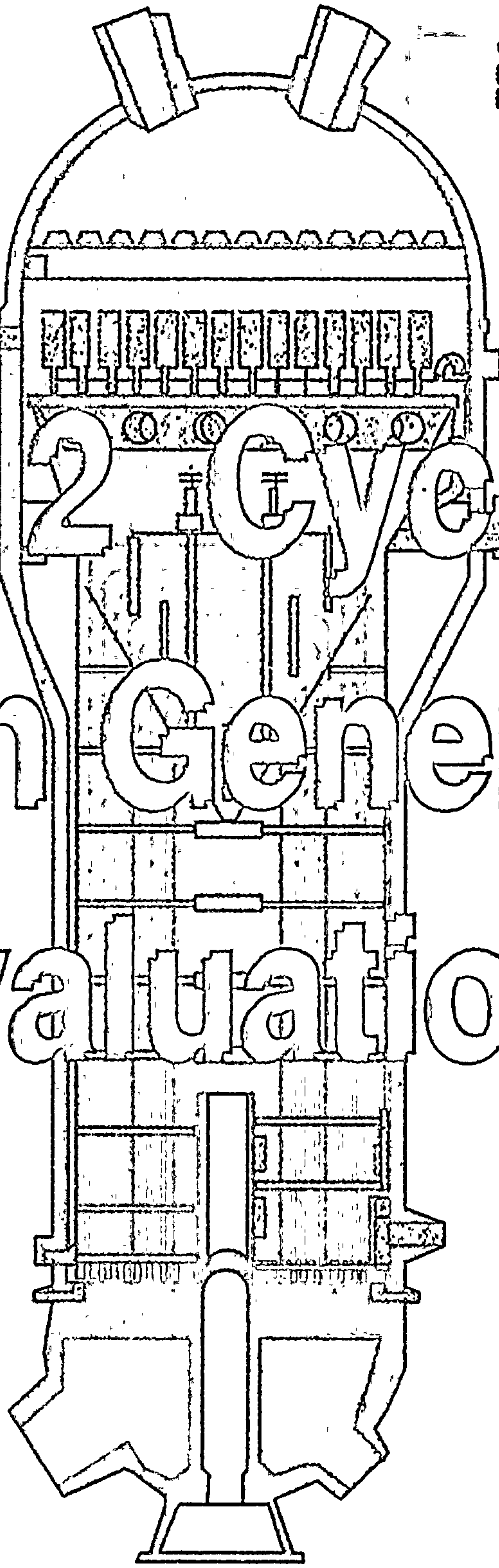
Palo Verde Nuclear Generating Station

Unit 2 Cycle 7 Steam Generator Evaluation



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PALO VERDE NUCLEAR GENERATING STATION

UNIT 2 CYCLE 7 STEAM GENERATOR EVALUATION

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I. EXECUTIVE SUMMARY

This report describes the efforts conducted by Arizona Public Service (APS) to address tubing degradation in the Unit 2 steam generators at the Palo Verde Nuclear Generating Station (PVNGS). The operational assessment supports continued Cycle 7 operation of Unit 2 until the next refueling outage. The refueling outage (U2R7) is scheduled for mid-September 1997, which represents 16.5 months of full power operation for Cycle 7.

The analyses provided in this report are similar to evaluations previously performed by APS, and presented to the USNRC Staff. The analytical framework and results have been validated by recent steam generator inspections during U1R6, U3R5 and U2R6. A condition monitoring assessment of Unit 2 Cycle 6 demonstrates the appropriateness of the PVNGS Steam Generator Management Program in assuring safe and consistent steam generator operation. The predicted end of Cycle 6 condition of the steam generators bounded the observed inspection results. A similar probabilistic run time analysis for Unit 2 Cycle 7 was performed by APTECH and reviewed by APS. The processes of crack initiation, crack growth and eddy current inspection were modeled via a Monte Carlo simulation. The operational assessment provides predicted end-of-cycle conditions for Unit 2.

The operational assessment results support at least 16.5 months of full power operation in Unit 2. The upper 95% confidence estimate of the conditional probability of tube burst for a postulated main steam line break is 3×10^{-4}. Therefore, the acceptance criteria of 10^{-2} is satisfied. The chance of a leaking crack during Cycle 7 is correspondingly remote at 4×10^{-5}. The analysis found that the performance of two successive inspections using the Plus Point Probe have significantly reduced the severity of undetected beginning of cycle degradation. It is also apparent by the analysis results, that remedial measures such as Thot reduction, chemical cleaning and improved secondary chemistry have substantially reduced degradation growth rates since 1993. Consequently, all of the analytical run time calculations performed for this report, strongly support full cycle operation in Unit 2.

This report also describes the PVNGS Steam Generator Management Program which provides a defense-in-depth approach to preventing challenges to nuclear safety. The program includes preventative remedial measures, comprehensive steam generator inspections and conservative plugging criteria. The program also incorporates strict administrative controls on primary-to-secondary leakage and RCS activity levels, enhanced radiological monitoring, emergency procedures improvements, and continued operator training. Based on the analyses and actions described in this report, APS concludes that the structural and leakage integrity of PVNGS Unit 2 steam generators will be maintained, and that Unit 2 can be safely operated until the scheduled refueling at the end of Cycle 7.

II. PROBLEM DESCRIPTION AND SAFETY ASSESSMENT

The purpose of this report is to describe the efforts conducted by APS to address the presence of ARC region[1] Outside Diameter Stress Corrosion Cracking (ODSCC) in the Unit 2 steam generators. Other degradation mechanisms found in Unit 2, such as tube support and loose part fretting wear, circumferential cracking at the tube sheet transition, and secondary side corrosion outside the ARC region were not explicitly considered in this assessment. Based on the results of comprehensive ECT examinations, these mechanisms exhibit slow growth and are, therefore, bounded by past analyses performed by APS (References 2, 8 and 10). From an analytical perspective, Cycle 7 operation is justified in Unit 2 based on the end of Cycle 6 steam generator inspection results. The assessment contained in this report is structured to support continued operation in Unit 2 for at least 16.5 months following U2R6 until the next scheduled refueling outage. The U2R7 refueling outage is scheduled for mid-September 1997 which represents a 16.5 month operating run for Unit 2 Cycle 7.

A. PVNGS Steam Generator Description

The PVNGS Nuclear Steam Supply System (NSSS) uses two recirculating steam generators which are vertical tube and shell heat exchangers approximately 68 feet in height with a steam drum diameter of 20 feet. The System80 steam generators were designed and fabricated by Combustion Engineering (CE), and are the only domestic operating units of this design. Each steam generator contains 11,012 Alloy 600 tubes which are 3/4 inch OD, and have a nominal wall thickness of 0.042" with an average heated length of 57.75 feet. The tubes were explosively expanded into the tubesheet for the entire tubesheet thickness. The tubing in Unit 2 was manufactured by Noranda to the requirements of ASME SB-167 as supplemented by CE specification requirements restricting carbon content to 0.05 percent and maximum yield strength of 55,000 psi. These requirements assured a relatively high temperature final anneal of 1806 °F. The tubes are arranged in rows, with all tubes in a given row having the same length. The rows are staggered, forming a triangular pitch arrangement. The shorter tubes, which have 180° bends, are at the center of the tube bundle in the first 18 rows. All subsequent rows have double 90° bends. The horizontal supports are of eggcrate design, while the

1. For brevity, the axial free span and support defects found in the upper bundle of the Units 1, 2 and 3 steam generators at PVNGS are referred to as ARC region ODSCC. The area of interest in the tube bundle has been previously defined in References 1 and 2.

bend and horizontal regions are supported by batwing and vertical lattice supports respectively. The supports are manufactured from 409 ferritic stainless steel. Diagrams of the steam generators, including flow paths are provided in Figures II-1, and II-2.

PVNGS Unit 2 commenced operation in September 1986, and at the end of Cycle 6, the steam generators had accumulated approximately 59,200 effective full power hours.

B. ARC Region ODSCC Description and Background

The ARC region freespan ODSCC phenomenon was first recognized in PVNGS Unit 2 during the ECT inspections conducted in the Spring of 1993 during the fourth refueling outage (U2R4). This mechanism had resulted in the rupture of a tube during power operation at the end of Cycle 4 in Unit 2. This event, and the subsequent root cause of failure assessment, was discussed in depth in the "Unit 2 Steam Generator Tube Rupture Analysis Report" submitted to the NRC staff as enclosure (2) to William Conway's letter 102-02569-WFC/JRP dated July 18, 1993. Based on the results of the root cause evaluation, which included information from laboratory examinations of tubes removed from the Unit 2 steam generators, APS concluded that a free span axial cracking phenomenon had developed in the upper tube bundle region of the Unit 2 SGs. The defects were determined to be outside diameter Intergranular Stress Corrosion Cracking (IGSCC) which formed due to a combination of contributing factors including: a susceptible region of high quality and contaminant concentration, tube-to-tube crevice formation, bridging ridge-like deposits, increased sulfate levels, and a caustic crevice pH. Additional factors, such as, less than standard metallurgical microstructures for High Temperature Mill Annealed (HTMA) tubing and cold working at the OD tube surface from manufacturing scratches, were also observed on some of the tube samples removed from Unit 2.

Since the Reference 1 submittal, APS has continued to aggressively assess this phenomenon with respect to the affected tube population, crack growth rates, ECT techniques and defect morphology. A total of 31 tubes were subsequently removed from Units 2 and 3, and the results are summarized in References 3-6. Thermal-hydraulic models have been developed and refined in an effort to define an empirical relationship between steam generator design and inspection results. APS developed a comprehensive steam generator degradation management program for all three PVNGS units as a result of the findings in Unit 2.

Design and operation similarities in the PVNGS units, required APS to consider that this corrosion mechanism might be transportable to Units 1 and 3. Consequently, the scope of ECT inspections have been adjusted since 1993 to verify that the technical basis and breadth of the phenomenon at work in Units 1 and 3, was similar to Unit 2. The ECT programs at Units 1 and 3 confirmed the presence of the free span ODSCC mechanism. The inspection programs and consequential run time assessments for all three PVNGS units have been discussed in detail in References 2, 8, 10 and 37. Recent discoveries of free span ODSCC at other CE model steam generators has been assessed by ABB-CE and APS (Reference 38). Based on this assessment and ECT testing at PVNGS, no changes in ECT scope are deemed necessary at this time. A detailed description of the ECT program conducted during U2R6 is provided in Section IV.

C. Safety Assessment

The safety significance associated with the operation of Unit 2 until the scheduled U2R7 refueling outage has been evaluated by APS. Since all SCC defects are plugged by APS on detection (i.e no sizing repair criteria is applied), the projected condition of the steam generator tubing is dependent on the size and characteristics of the undetected beginning-of-cycle defect population. To this end, APS has employed the best available inspection technology, to not only assure that no structurally significant defects are left in service, but also to lower the chance that large defects evolve during the following operating cycle. To demonstrate safe operation, the Unit 2 steam generator inspection results have been assessed statistically to determine the impact of leaving any undetected ARC region defects through the remainder of Cycle 7 operation. The results are then compared with established structural and leakage integrity acceptance criteria based on Regulatory Guide 1.121, NUREG 0844, and guidance, where applicable, from Generic Letter 95-05 and the Draft Steam Generator Rule and Regulatory Guide. The analyses reported in this evaluation conclude with high confidence that the conservative safety margins established in Regulatory Guide 1.121 are maintained, the conditional probability of tube burst is acceptably low and the possibility of leakage either operationally or accident induced is correspondingly remote.

In assessing the safety significance of the current tubing conditions in the PVNGS Unit 2 steam generators, APS has used the criteria defined in General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10CFR50 Appendix A in the development of the PVNGS Steam Generator Degradation Management Program described in Section III. These regulatory

criteria require appropriate inspection and testing of steam generators (RCS pressure boundary), leakage monitoring, and assurance of low probabilities of abnormal leakage, of rapidly propagating failure, and of gross rupture.

With regard to safety significance, APS has determined via the analyses presented in this report, that operation of the Unit 2 steam generators does not represent a significant increase in the probability or consequences of an accident previously evaluated in the PVNGS UFSAR. APS has determined that there is a low probability ($3x10^{-4}$) of generating a defect in excess of the structural margins specified in Regulatory Guide 1.121. The probability of any leak during Cycle 7 was calculated to be $4x10^{-5}$ as no instance of through-wall crack penetration was observed after 20,000 Monte Carlo simulations. The conditional probability for tube rupture given a MSLB was also calculated. No simulated bursts were generated after 10,000 simulations. These null results also provide an upper 95% confidence estimate of a $3x10^{-4}$ burst probability. This result satisfies the criteria set forth in NUREG-0844 and the 10^{-2} probabilistic criteria listed in Generic Letter 95-05 and the Draft Regulatory Guide, *Steam Generator Tube Integrity.* APS has also assessed leakage potential given a MSLB, and the impact of operation with ARC region defects on core damage probability. Tube leakage can occur when the maximum crack depth reaches 100% through-wall with structural integrity maintained. As stated previously, comprehensive ECT examinations, low observed crack growth rates and conservative plugging criteria have resulted in an analyzed probability of through-wall crack penetration of $4x10^{-5}$. Consequently, the analyses contained in Reference 10 with regard to leakage probability and core damage frequency are considered bounding for Unit 2 Cycle 7. The PRA results, from Reference 10, indicate that higher core damage risk would be incurred if a midcycle shutdown was performed to support steam generator inspections. Based on an even lower core damage probability (CDP), it is clear that continued operation of Unit 2 to the end-of-cycle (EOC) is preferable, from a risk perspective, to performing a midcycle shutdown and inspection.

Since probabilistic models contain a degree of uncertainty, additional actions have been taken by APS to assure safe plant operation. An improved leak rate monitoring program and administrative limits on primary-to-secondary leakage as described in References 2, 8 and 10 and are still in place. The program is designed to provide clear and unequivocal plant management support to commence orderly shutdown should leakage exceed very

stringent administrative limits of 50 gallons per day (GPD). APS has also endeavored to ensure that adequate staff, equipment and organizational resources are in place to implement this program, using a combination of radiation monitors and laboratory radiochemical analyses. The integrated leakage program at PVNGS is not only prescriptive, but preventative as well, as the program is supported by extensive steam generator inspections and conservative plugging criteria which ensure that all detected SCC defects are removed from service. These features provide additional assurance that an orderly shutdown would be conducted prior to a through-wall leak propagating to a rupture.

APS has also assessed the consequences of Unit 2 steam generator operation during Cycle 7 on any previously analyzed accidents. To further minimize the consequences of currently analyzed accidents, APS has taken measures to provide operations personnel with improved diagnostic tools and training. These measures include: event specific training of operations personnel for tube rupture events; improvements in leakage diagnostics via equipment upgrades including the implementation of N-16 monitors; and protocol upgrades to the Emergency Operating Procedures. These actions permit faster identification and isolation of the affected steam generator given an SGTR event.

APS has reviewed outside design basis events to assure the chance and consequences of an unanalyzed accident are not increased. A best estimate radiological assessment was performed by APS in Reference 9, which indicated that in the unlikely event of a main steam line break with consequential multiple tube ruptures, with current administrative limits on reactor coolant system dose equivalent iodine, the resulting offsite doses would be less than 10CFR100 limits. As indicated previously, the structural integrity analyses conducted by APTECH indicated that no tube ruptures occurred in the Unit 2 Cycle 7 run time simulation after 10,000 trials, for a loading condition of 3ΔP. As a result, the burst probability is less than $3x10^{-4}$ for lesser loading conditions such as MSLB, stuck open safety valve (SOSV) or normal operating conditions. This low default conditional probability of tube burst means that the probability of multiple tube bursts at postulated MSLB conditions is extremely low. If the default probability of burst is used to characterize a single event, then the conditional probability for two (2) bursts is $4.5x10^{-8}$. For three (3) or more the estimated probability is $4.5x10^{-12}$. Therefore, the probability of multiple tube ruptures either as an initiating event or the consequence of a MSLB is substantially less

than the $1x10^{-5}$ event probability used for Unit 2 Cycle 6 in Reference 10. Consequently, the risk assessment performed in Reference 10 is bounding, as it indicated that there was a negligible effect ($1.2x10^{-7}$) on the core damage probability (CDP) associated with continued operation of the Unit 2 steam generators. The baseline core damage frequency at PVNGS is $4.74x10^{-5}$ per reactor year, and therefore CDP was negligibly increased (approximately 0.4%) for consequential single and multiple tube ruptures due to the predicted propagation of ARC region axial cracks.

Finally, APS reviewed its steam generator program with regard to the margin of safety as provided in the PVNGS Technical Specifications. APS has developed a "defense in-depth" approach to managing steam generator tubing structural integrity. The analyses contained in this report indicate that the structural integrity margins specified in Regulatory Guide 1.121 have been satisfied to a high probability and confidence level. APS has considered that such probabilistic models contain a degree of uncertainty associated with the selected probability distributions. Therefore, additional actions have been taken by APS to assure safe plant operation. An improved leak rate monitoring program and administrative limits of 50 gpd for primary-to-secondary leakage, add significant margin over limits currently specified in both the PVNGS Technical Specifications and the recently regulated leakage limits contained in the voltage based repair criteria in Generic Letter 95-05. These conservative limits provide additional assurance that an orderly shutdown will be conducted prior to a through-wall leak propagating to a rupture.

APS therefore concludes that the measures and analyses reported in this evaluation provide reasonable assurance that there are no reductions in the required safety margins, that all issues associated with safety significance have been addressed and that PVNGS Unit 2 can be safely operated until the scheduled U2R7 shutdown.

III. PVNGS DEGRADATION MANAGEMENT PROGRAM

A. Background

Since the Unit 2 steam generator tube rupture in March 1993, APS has conducted five extensive ARC region inspections in Unit 2 (U2R4, U2M5-1, U2M5-2 U2R5 and U2R6), three inspections in Unit 1 (U1R4, U1R5 and U1R6) and four inspections in Unit 3 (U3M4, U3R4 U3M5 and U3R5). These inspections have provided considerable information on the behavior of ARC region ODSCC. Additionally, APS has performed three (3) tube removal and examination activities of 31 tubes from Units 2 and 3, ultrasonic testing (UT) in Units 1 and 2 and in-situ pressure testing in Units 1 and 2. Figure III-1 has been provided to summarize key timeline information for all three units including outage reference, cycle length and key operational and maintenance changes. By integrating the information obtained from activities conducted in all three units, APS has the tools to develop the elements of a PVNGS Steam Generator Degradation Management Program. The objective of the program is reliable and safe steam generator operation. This objective is achieved through improvements in secondary chemistry, primary temperature reduction, chemical cleaning, management of ARC region degradation through an aggressive ECT and plugging program, and continually updated comparisons of inspection results to predictive models. It is APS's position, that these actions now permit safe full cycle operation for all three PVNGS units

B. Objectives

The purpose of integrating planned outages, analytical evaluations, tube inspections and remedial/repair activities, is meeting the following safety objectives:

- Establish unit cycle lengths which are dependent on demonstrating compliance with the applicable General Design Criteria (GDC) in 10CFR50, applicable ASME Code criteria, and the structural integrity margins for steam generator tubing as defined in Regulatory Guide 1.121.

- Limit the safety issues associated with equipment and personnel resources, by avoiding the performance of simultaneous outages of the PVNGS Units.

- Perform an assessment of the safety consequences of performing midcycle outages versus proposed operation to planned refueling outages by assessing core damage risk as a function of graduated increases in inspection interval.

C. Program Plan

In order to satisfy these objectives, APS has implemented the following actions which meet or exceed similar industry and regulatory guidance for such a management program. These actions include:

- A comprehensive steam generator inspection program which significantly reduces the risk of leaving a significant defect(s) in service. Enhanced inspections (MRPC, Plus Point, UT) are conducted in the region affected by ARC degradation. The extent of MRPC and bobbin coil inspections performed by PVNGS, coupled with the use of new state of the art ECT data acquisition and analysis technology, has a significant impact in terms of preventing through-wall defects, and significantly lowers the probability of tube rupture for the specified period of operation. This position is supported by assessments performed in Reference 10, and as well as by sensitivity studies performed by EdF in Reference 16.

- APS applies plugging criteria more conservative than current PVNGS Technical Specifications. Before the steam generators are returned to service, a review by APS Engineering of all eddy current indications is conducted. All tubes with detected cracks, regardless of size or depth, are removed from service. As demonstrated in Reference 10, and supported by data presented in Reference 16, this action also minimizes the

potential for leakage and/or tube rupture, since no known crack defects are left in the steam generator at the beginning of the operating cycle.

- APS has implemented an integrated leakage detection and response program, using equipment and procedure upgrades, to permit plant operators to detect and respond to changes in steam generator primary-to-secondary leakage. The threshold for action is 10 gpd and the maximum allowable leak rate is 50 gpd. The program was established to provide reasonable assurance that the unit will be shutdown prior to a significant leak or steam generator tube rupture should tube degradation exceed expected values. Leakage detection and monitoring equipment has also been upgraded. PVNGS has installed N-16 monitors to provide supplemental and timely information to the operators.

- State-of-the-art probabilistic models have been developed for assessing operating cycle lengths and calculating tube structural margins. Plant specific NDE measured parameters and unit specific material properties have been incorporated, and crack initiation and growth predictions benchmarked, to ensure that the safety margins specified in Regulatory Guide 1.121 are maintained.

- Development of probabilistic leakage models to assess end of cycle leakage as a result of secondary overpressurization events. These models ensure that operation with ARC region degradation will not result in offsite releases in excess of 10CFR100 limits should a MSLB event occur.

- Development of a risk model to assess the impact on core damage probability for plant operation with degraded steam generator tubing. The calculation ensures that operation with

ARC region degradation represents a negligible impact on core damage probability.

- Primary temperature reductions of 10°F have been implemented in all three PVNGS units to take advantage of the temperature dependence of SCC growth rates. Stress corrosion cracking is a thermally activated process, and the effects of temperature reduction can be quantified for SCC mechanisms in terms of activation energy for an Arrhenius rate equation.

- APS has removed 31 tubes from service, and has conducted extensive NDE and destructive examination in an effort to determine causal effects of corrosion damage, and to provide substantial improvements in field ECT acquisition and interpretation.

- APS has implemented the industry recommended secondary chemistry controls to mitigate the initiation and propagation of secondary side IGA/SCC. The laboratory evidence from tubes removed from Unit 2 during U2M5-1 showed a favorable change in crack crevice chemistry tending towards neutral conditions. APS has established action levels for sulfate which are more restrictive than EPRI Guidelines, and procedures require reduced power operation if sulfate levels exceed 20 ppb or plant shutdown for levels exceeding 100 ppb.

- APS has conducted full bundle chemical cleaning in all three PVNGS units to remove contamination accumulation on the steam generator tubing. APS through tube pulls, ECT examination and thermal-hydraulic modeling has established a link between tube deposits and ARC region cracking. Chemical cleaning is intended to reduce or eliminate crack initiation sites

and slow the progression of crack growth. Recent inspection results indicate that these objectives may be realized.

- APS has established administrative limits on dose equivalent iodine levels in the reactor coolant system. The Technical Specification limits for initial primary system activity are 1.0 μCi/gm for steady state and 60 μCi/gm for transient or spiked limits. The PVNGS Administrative limits are 0.6 μCi/gm equilibrium and 12 μCi/gm transient. Therefore, should a non-isolatable main steam line break with multiple steam generator tube ruptures occur, the radiological consequences are estimated to be within 10CFR100 limits.

- APS is implementing significant Steam Generator modifications in an effort to provide a more favorable thermal hydraulic conditions. These modifications should reduce the number of tubes at risk within the ARC region.

It is APS's position that the implementation of these elements, as further described in this report, constitutes a defense-in-depth approach which ensures that adequate structural and leakage integrity is maintained for normal operating, transient, and postulated accident conditions. The steam generator management program at PVNGS is therefore consistent with the design bases defined in General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10CFR50 Appendix A.

IV. STEAM GENERATOR INSPECTION AND REPAIR

A. Introduction

The ability to detect and remove from service, critical ARC region defects, is a primary feature of the PVNGS Degradation Management Program. Previously unknown limitations of bobbin coil inspection techniques, used in the U2R3 steam generator inspections during the Fall of 1991, are considered to have contributed to the tube rupture event in Unit 2 in March 1993. Since 1993, APS has endeavored to improve these ECT techniques via improved technology, analyst training, industry-leading use of the Plus Point[1] MRPC Probe, and a large scale tube pull program which provided ECT-to-actual defect comparisons leading to the development of plant specific probability-of-detection (POD) curves. Additionally, APS has employed supplemental techniques such as ultrasonic testing (UT), and in-situ pressure testing when necessary.

The steam generator inspection programs conducted during U2R6 were performed to assure plant safety, and to assist in understanding the initiation, progression and scope of the ARC region ODSCC phenomenon. It is recognized by APS and the NSSS manufacturer, ABB-CE, that the design of the System80 steam generators creates an environment in the upper region of the tube bundle susceptible to ODSCC if other contributing factors such as alkaline secondary chemistry, crevice formation and material susceptibility are present. As such, APS has continued to refine the thermal-hydraulic analyses developed and discussed in Reference 1, to assist PVNGS Engineering in defining inspection scope. APS has also incorporated inspection experience from the six (6) steam generators from all three units in determining the breadth of the susceptible tube population and inspection extent. Expansion criteria consistent with Section C.1.3.4 of the draft Regulatory Guide, *Steam Generator Tube Integrity* has been applied since 1994. The inspection scope, expansion criteria, techniques and personnel qualification are described as follows.

1. Plus Point MRPC Probes are designed and manufactured by Zetec. The probe configuration combines the axial and circumferential coils in one gimbal-mounted surface riding coil shoe.

B. U2R6 Inspection Summary

A. Examination Scope

As in the previous inspections, the Unit 2R6 examination plan was developed in response to findings associated with previous eddy current examinations performed in Unit 2, as well as experience gained from recent inspections conducted in Units 1 and 3. For example, the purpose of 100% full length bobbin coil examinations is to provide general tube condition screening. The exam aids in assuring that a widespread pattern of pluggable flaws including wear and loose parts are not present. If such flaws are detected and exceed PVNGS Administrative Plugging Criteria, they are removed from service.

During U2R6, MRPC[1] (Plus Point) testing was performed in the high risk ARC region of the steam generators in search of corrosion defects similar to those found in the previous inspections. Typically, the MRPC probe inspected the section of tubing from the 07H horizontal eggcrate support to the second vertical strap support that the tube is supported by (See Figure IV-1). The upper bundle MRPC testing also included a sample program of tubing between columns 40-150 and rows 90 -110. This additional sample was performed in an effort to determine if corrosion damage is occurring outside the pre-defined 2500-2700 tube high risk ARC region.

In accordance with APS's response to Generic Letter 95-03, a MRPC test program was also performed on 100% of the hot leg tubesheet expansion transition locations. Additionally, a 5% MRPC sample of the cold leg tubesheet was performed to identify any circumferential indications in the expansion transition region similar to those found previously in the Units 1, 2, and 3 hot leg, and in cold leg locations at other CE units.

Also, based on previous PVNGS experience, 100% MRPC testing in the rows 1 and 2 short radius u-bends was conducted from 07H-07C to determine if

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1. Since 1994, the MRPC probe at PVNGS employs at least a 0.115" pancake coil and a Plus Point™ coil. The Plus Point coil is relied upon as the best detection technique available. Crack characterization is performed with the 0.115 coil based on PVNGS plant specific tube pulls.

corrosion damage in this region was present. In summary, the base scope for the examination included:

- Examine 100% of Steam Generator 21 (SG 21) and Steam Generator 22 (SG 22) using bobbin coil techniques.

- Examine ~2700 tubes in each of SG 21 and SG 22 from 07H-2nd vertical support (VS) using MRPC techniques. The MRPC probe included the Plus Point coil for defect detection and orientation. These tubes were selected in the area of interest for ARC region axial indications.

- Examine ~200 tubes in each of SG 21 and SG 22 from 07H-2nd VS using MRPC. These tubes were selected in areas between columns 40 - 150 and rows 90 - 110, and represented an additional sample based on ATHOS model results.

- Examine 100% of the hot leg tubesheet (TSH) expansion transition region in each of SG 21 and SG 22 using MRPC.

- Examine ~110 tubes in each of SG 21 and SG 22 from 07H-07C using MRPC. This examination included all the rows 1 and 2 short radius U-Bend tubes.

- Examine historical >20% bobbin wear indications in SG 21 and SG 22 using MRPC. This inspection was performed to determine if corrosion was present at these locations.

- Examine ~500 tubes in each of SG 21 and SG 22 of the cold leg tubesheet (TSC) expansion transition region using MRPC. This exam was performed to determine if cold leg corrosion was present.

The expansion criteria employed in Unit 2 included:

Axial Indications:

- Five (5) tube buffer zone in all directions using MRPC.
- MRPC of any Bobbin indications that exceed Palo Verde plugging Criteria.
- MRPC of all Bobbin I-codes including ADR's (absolute drift).

Circumferential Indications:

- Any circumferential indication in cold leg - expand to 100% MRPC of the cold leg tubesheet.

The exam description, the extent examined and the actual number of tubes analyzed, including expansions, are identified in Table IV-1. Figures IV-2 to IV-5 depict the scope inspected with respect to tube location on the tubesheet. In Figures IV-2 and IV-3, the tick marks identify regions inspected, including the random ARC inspection indicated by columns of tubes between column 60 and 120. While the filled-in areas indicate locations of defects. Figures IV-4 and IV-5 indicate areas inspected with Plus Point outside of the ARC region. No SCC defects were found at these locations.

Table IV-1 - Unit 2R6 ECT Inspection Summary

SCOPE DESCRIPTION		SG 21	SG 22
Exam Description	Extents	Analyzed	Analyzed
FULL LENGTH BOBBIN	TEC-TEH	10,592	9822
TUBESHEET (TSH) MRPC	TSH-TSH	10,592	9822
U-BEND MRPC	07H-2nd VS	2628	2495
LOW ROW U-BEND MRPC	07C-07H	112	124
RANDOM ARC MRPC	07H-2nd VS	158	168
TUBESHEET (TSC) MRPC	TSC-TSC	588	583
MRPC OF PREVIOUS >20% WEAR	VARIOUS	224	171
EXPANSION 1 (SPECIAL INTEREST I-CODES/PID)	VARIOUS	91	99
EXPANSION 2 (SAI, MAI BOUNDING)	VARIOUS	90	60

EXPANSION DESCRIPTION

EXPANSION 1 — This expansion is utilized to track the special interest MRPC performed to quantify or evaluate bobbin or previously called indications. This includes NQI, ADR, DSI, DTI, PLP[1], and other areas. PID (positive identification) were run to verify that tube identification is correct.

EXPANSION 2 — MRPC examinations bounding single axial indications (SAIs) to aid in determinations of additional SAIs in general area. This expansion was triggered by SAIs found in original MRPC scope.

Notes:

1. Non-Quantifiable Indications (NQI), Absolute Drift (ADR), Distorted Support Indication (DSI), Distorted Tubesheet Indication (DTI), Possible Loose Part (PLP).

B. Examination Results

A summary of the U2R6 examination results is provided in Table IV-2. Additionally, a ARC region summary Table is provided in Appendix B. The U2R6 inspection program resulted in the plugging of 99 tubes in SG 21 and 151 tubes in SG 22 as a result of ARC region corrosion. Figures IV-6 and IV-7 provide histograms of the ECT results in the ARC region as a function of 0.115 pancake coil voltage. The treatment of pancake coil voltage and voltage growth in determining the allowable run time in Unit 2 is described in Section V and Appendix A. It should be noted that the large number of zero (0) volt values represent defects too small to be observed by the pancake coil but detected by the Plus Point.

Table IV-2 Indication Summary U2R6

INDICATION CATEGORY	STEAM GENERATOR 21	STEAM GENERATOR 22
Cold Leg Corner Eggcrate Wear		
0% to 19%	2	0
20% to 29%	0	0
30% to 39%	0	0
40% to 100%	0	0
Eggcrate Wear		
0% to 19%	458	578
20% to 29%	163	305
30% to 39%	34	81
40% to 100%	4	6
Flow Dist Plate Wear		
0% to 19%	1	0
20% to 29%	0	0
30% to 39%	0	0
40% to 100%	0	1
Batwing Wear		
0% to 19%	665	757
20% to 29%	256	325
30% to 39%	73	132
40% to 100%	2	5
Vertical Strap Wear		
0% to 19%	291	351
20% to 29%	155	176
30% to 39%	69	55
40% to 100%	4	12

INDICATION CATEGORY	STEAM GENERATOR 21			STEAM GENERATOR 22		
Possible Loose Parts PLI PLP	2 0			1 2		
Axial Indications	orig	exp1	exp2	orig	exp1	exp2
TSH	1	na	na	6	na	na
01H	0	0	0	0	1	0
ARC Region	99	0	0	149	0	2
Circumferential Indications	8			3		
Single Volumetric Indications	42			40		

Notes: exp1, exp2 - refer to Expansion 1 and Expansion 2, orig - refers to original scope

C. Examination Techniques and Equipment

The PVNGS steam generator eddy current program has continued to evolve through the use of state-of-the-art equipment and technique development which incorporate lessons learned from PVNGS exams and tube pulls, as well as industry experience. The objectives of the APS program are to acquire data in a timely fashion while maintaining and/or improving the ability to detect and characterize flaws.

The eddy current examination for U2R6 was performed by Rockridge Technologies (formerly Conam Nuclear Inc.) using Zetec MIZ 30 digital data acquisition and analysis systems. The following frequencies were used for the tube examination(s):

Examination Frequencies	
Bobbin	MRPC
20 KHz	20 KHz
100 KHz	100 KHz
300 KHz	300 KHz
500 KHz	400 KHz

For the stated inspection scope, all tubing was examined with Zetec manufactured bobbin coil and MRPC style probes, either 0.610, 0.600, 0.590 0.580 or 0.560 inch diameter. Multiple configurations of Plus Point MRPC probes were used for the detection and characterization of axial and circumferential indications. Data acquisition was facilitated by using Zetec SM-22 manipulators with quad guide tubes and dual guide tubes in the hot leg and cold leg respectively in Steam Generators 21 and 22. A BWNT ROGER manipulator with a quad guide tube was also used in the hot legs of both of steam generators.

Fiber optic cable was used from the MIZ 30 containment location to the data acquisition room located at the PVNGS North Annex. Primary and Secondary analyses were performed remotely utilizing T-1 line technology. Primary Analysts were located in Benicia, California; Issaquah, Washington; and Lynchburg, Virginia. Secondary Analysts were located in San Clemente, California. The Primary and Secondary Resolution Analysts were located at PVNGS. Rockridge Technologies provided the data acquisition and primary data analysis. Anatec International, Inc. provided the secondary data analysis.

Each Level IIA individual from Rockridge Technologies and Anatec International, Inc. who performed data analysis was required to complete and pass a PVNGS site specific Eddy Current Data Analysis Course, as well as an associated performance examination with at least a 80% proficiency within the last year. All individuals performing data analysis were required to have EPRI QDA (Qualified Data Analyst) certification.

Plus Point MRPC

Due to the critical nature and extent of MRPC inspections at PVNGS, the primary objective in ECT technique and analysis evolution is to improve production speed, while at the same time maintaining or improving detection capability. In meeting these objectives the Zetec Plus Point MRPC Probe was utilized for the first time at PVNGS during the 1994 U3M5 inspections and has been subsequently used in all PVNGS inspections. The Plus Point Probe was originally developed by Zetec for surface examinations of reactor vessel welds. The probe was designed to reduce geometry and permeability effects. The coils are differentially paired within the same coil shoe and surface riding to reduce the effects of geometry. The standard MRPC probe design utilized during U2R6 employed a standard 0.115" diameter pancake coil, and a separate shoe contained the plus point coil. The 0.115" pancake coil response is utilized in assessing defect size and growth using a voltage correlation developed from tube pulls at PVNGS. The benefits of the Plus Point coil realized at PVNGS with regard to speed and detection are described in detail in Reference 10.

Data Quality

PVNGS Station Manual Procedure 73TI-9RC01, *Steam Generator Eddy Current Examinations*, provides requirements to assure data quality for ECT inspections. The procedure requires that calibration shall be made and recorded at the beginning and end of each optical disc or every four hours whichever comes first. For bobbin coil inspections, if this time is exceeded, all tubes examined after the four hours shall be re-inspected. For MRPC, the tubes shall be reviewed by a Level III analyst to verify the data.

APS recognizes that the useful life of ECT probes has been shown to vary significantly from probe to probe and cannot be predetermined. Probe life may be fewer than five (5) tubes or greater than 1000 tubes. As a result, the analysts are required to note if undesirable variations exist during examinations or calibrations. As further guidance, APS has developed measurable data quality acceptance criteria for bobbin and MRPC inspections. Parameters monitored include phase, amplitude, voltage, probe speed, electrical noise and data drop out. If a probe is determined to have become defective prior to post calibration, the data analysts

shall designate the data as BDA (Bad Data) for review by the resolution analysts. This review is documented according to procedure.

C. Defect Detection and Characterization

As described in Section V, the structural integrity analyses performed by APTECH apply MRPC signal response information to characterize defects and assess crack growth rate. As indicated throughout this report, all crack indications found during the ECT examinations conducted at PVNGS are reported and removed from service. This conservative plugging philosophy makes it inherently difficult to determine growth rates on detected defects from inspection to inspection. Additionally, eddy current technique changes, probe technology improvements, improved analyst training and reduction in signal interferences can impact detection and characterization. These factors must be studied and evaluated to determine the effects of these changes, and to normalize the inspection results.

1. Probability of Detection (POD)

The analyses described in Section V are dependent on the ability to characterize the undetected defect population remaining in the steam generators upon the completion of the ECT inspections, since all detected SCC defects are removed from service. Inspection results, to date, indicate that the bobbin coil technique is not sufficiently reliable for detection of low volume SCC defects due to tube pilgering interferences and geometry effects associated with tube ovalization in the bend region. As a result, APS has relied on the use of MRPC technology to improve defect detection thresholds. Since the U2R4 inspection, APS has removed 31 tubes from the Units 2 and 3 steam generators to provide accurate comparisons of actual defect depth with MRPC detection capability. A PVNGS specific probability of detection (POD) curve has been constructed from this data and presented to the USNRC Staff in References 2 and 8. This POD curve was considered to be reasonable when compared to the typical industry MRPC pre-1993 database.

Since the U2R4 inspection and tube pulls, APS has observed three distinct inspection transients or improvements in defect detection. The most notable inspection improvement observed by APS was implementation of the Plus Point

MRPC probe. Similar inspection transients due to the Plus Point have been observed at other nuclear facilities. APS has detailed the impacts of these inspection improvements in Reference 10 and calculated a Plus Point POD curve based on the observed transient results. Recently, pulled tube data for upper bundle cracking in a CE model steam generator has been made available. From this new data an actual Plus Point POD function was developed by APTECH using logistic regression. A comparison of the actual Plus Point POD and the "effective" POD used in Reference 10 showed good agreement (See Figure 5.5 of Appendix A). Therefore, the "effective" Plus Point POD curve has been utilized in the structural integrity analysis to simulate the inspections from U2R5-U2R7.

2. Defect Characterization

As stated previously, the conservative plugging philosophy employed by APS makes it inherently difficult to determine growth rates on detected defects from inspection to inspection. However, as part of the ECT analysis process of ARC region defect data, historical information was reviewed by the senior resolution analysts to determine if certain precursor signals from previous inspection data could be identified and voltage data obtained.

Since analyst variability exists with the selection of peak MRPC voltage, APS has strived to provide consistent measurements for the purpose of determining the voltage change for a particular flaw. Special care was taken to assess the effects of geometry, interfering signals and overall data quality. A lissajous graphic and C-scan hardcopy of each defect is generated and a final review of ARC region ECT calls for the Unit 2 data is conducted by the APS Level III analyst. Review of the ECT voltage data was also conducted by a Level III analyst on the APTECH Staff. The results of this review are summarized in Appendix B.

D. Metallurgical Examinations

Since the Unit 2 tube rupture event, APS has removed 31 tube sections from both the Unit 2 and Unit 3 steam generators. Laboratory examinations of these tubes have been conducted at Combustion Engineering, Babcock and Wilcock (B&W) and Westinghouse. The purpose of these examinations included:

- Evaluation of field eddy current testing (ECT) results against actual defect size and location
- Validation of ECT detectability thresholds
- Characterization of various ECT probe detection capabilities in the bend section tubing
- Burst test data correlation for structural integrity analysis
- Probability of detection (POD) database development
- Identification of mode of degradation
- Characterization of chemical environment
- Determination of microstructure attributes

The testing conducted on these tubes included:

- Receipt Inspections
- Visual Inspection and photography
- NonDestructive testing (Bobbin and MRPC, ultrasonic (UT) characterization)
- Dimensional Measurements (bend section radius, tube ovality)
- Deposit chemical analysis
- Swell testing of tubes without ECT corrosion indications
- Burst testing
- Scanning electron microscopy (SEM) of burst and crack extension areas
- Low Optical Microscopy of material cross sections

- Radial metallography to characterize surface intergranular corrosion
- Auger electron spectroscopy (AES) to determine crack tip chemistry
- X-Ray photoelectron spectroscopy (XPS) to determine crack tip chemistry
- Dual etch metallography to characterize tube microstructures
- Modified Huey testing to determine degree of tube sensitization
- Mechanical testing for tensile and yield strength properties and base metal chemical analysis

A detailed description of the test results has been provided to the USNRC Staff in References 1, 2 and 10. Data gained from these examinations has provided valuable input towards assessing eddy current capabilities, structural testing and analyses, and understanding the challenging ARC region degradation mechanism. The benefits of this effort have provided APS with a strong database of plant specific information. When supplemented by industry data and experience, it permits APS to implement the appropriate steam generator inspection and mitigation programs aimed at ensuring safe and reliable plant operation.

E. Steam Generator Modifications

APS has continued to assess the thermal-hydraulic conditions, which are believed to have resulted in a region of increased susceptibility to ODSCC in the PVNGS steam generators. As reported in Reference 1, APS has developed empirical relationships for predicting tube susceptibility in the vertical and square bend sections of the upper bundle region of the steam generator, based on calculated steam quality and mass flux (dryout). Using these relationships, thermal-hydraulic thresholds for dryout in the vertical section of the tubes and in the bend/horizontal section have been established. For the vertical sections of tubes, the dryout region represents a departure from nucleate boiling, and has been defined empirically by APS as a threshold deposit parameter. The deposit parameter consists of mass flux component (ρV) and a concentration factor for non-volatile impurities (1-X{quality}). Through computer modeling, APS has determined that ARC region crack occurs where the threshold value of the deposit parameter (ρV/1-X) exceeds 180 lbm/sq ft•sec. As part of its ongoing investigation, APS has found that the deposit parameter did not correlate as well for the cracks initiating in the bend or horizontal regions. APS and ABB-CE believe that contaminant concentrations in these areas, and the corresponding initiation of cracks, are related to a critical quality condition. By empirically correlating the location of the bend/horizontal defect, it has been shown that the threshold value for critical steam quality is 65%.

Recently, a number of other CE units have observed similar upper bundle corrosion damage. Although the thermal-hydraulic correlations developed by APS have been benchmarked in multiple inspections, APS is participating in a CE Owners Group (CEOG) root cause assessment to determine if additional adjustments should be made in the overall thermal hydraulic calculations and correlations. The CEOG root cause project will involve the use of neural networks to link, correlate and rank common causal factors.

APS has also performed an additional assessment of an experimental correlation referred to as the Zuber correlation (Reference 28) which relates the critical heat flux to quality (or void fraction) for low flow conditions. As shown in Figure IV-8, the bend and horizontal defects are located in a region of high void fraction with heat flux values in excess of the Zuber critical heat flux. This correlation agrees with the empirically determined threshold of 65% quality for the upper bundle bend region where free span cracks have been

observed in the PVNGS steam generators. Figures IV-9 and IV-10 provide iso-surface contour plots for both the deposit parameter and critical quality parameter for the PVNGS steam generators.

Finally, in 1996, APS contracted ABB-CE to conduct a complete review of the causative factors of free span axial cracking in the industry steam generators to compare the fabrication, thermal-hydraulic, and design similarities and differences associated with current CE and non-CE designed steam generators with free span SCC, with possible replacement steam generators. The study (Reference 38) compared well with previous APS efforts to define root cause and supported the thermal-hydraulic susceptibility criteria established by APS.

With the threshold criteria at PVNGS established and validated, the number of tubes subject to dryout for various modifications, operational changes, or combinations thereof, can be predicted and appropriate changes in ECT scope can be assessed. Based on further analyses performed by APS and ABB-CE, three (3) major modifications were designed and implemented in Unit 2 during U2R6. The steam generator modifications are described as follows:

Downcomer Feedwater on Hot Leg Side

The System80 steam generators were designed to introduce 10% of the feedwater flow to the cold side recirculating fluid. This original intent of this feature was to maximize the temperature difference between the cold leg sides of the primary and secondary fluids, thereby improving the thermal efficiency of the generators.

The modification, as shown in Figure IV-11, involves the replacement of the existing ring with a new feedring designed to deliver the downcomer feedwater flow to the hot side downcomer annulus. With the modification, the maximum bundle exit quality and computed deposit parameter values decrease and the hot side circulation ratio is increased. This outcome is achieved with a negligible decrease on thermal efficiency. The ARC region affected tube population is reduced by approximately 15%. The new

feedring was constructed of materials resistant to erosion/corrosion as an added benefit.

Separator Orifice Removal

Some of the steam separators in the PVNGS steam generators were originally installed with entrance orifices. The orifices were designed as a load leveling feature to improve separator efficiency. However, the location and design of the orifices contributed to the pressure drop in the recirculating loop thus reducing hot side flow rate through the steam generator. Analysis by APS found that the removal of the orifice plates reduced bundle exit quality and increased overall circulation ratio, thereby reducing the effect of the dryout condition in the upper bundle. The potential adverse affect of increasing separator loading and moisture carry-over was determined to be negligible. Consequently, the orifice plates originally installed in 62 of the 194 (per SG) steam separators were removed during U2R6 to reduce recirculating loop resistance.

Downcomer Shroud Modification.

The System80 steam generators are designed with a flow distribution plate (FDP) on the hot leg side. The FDP is located 16" above the tubesheet, and helps to distribute the incoming recirculating fluid more uniformly as it flows upward in the tube bundle. Operationally the FDP contributes to the ARC region phenomena as it produces additional flow resistance in the hot side recirculating loop, thus reducing recirculation. The new System80+ design by ABB-CE eliminated this feature, however, removal of the FDP at PVNGS would be an unrealistic field modification. As an alternative option, APS and ABB-CE designed a shroud modification involving the bypass of the FDP by cutting holes in the downcomer shroud above the flow distribution plate (See Figures IV-12 an IV-13).

These modifications improve the steam generator thermal-hydraulics and reduce tube corrosion susceptibility in the upper bundle region by reducing the maximum quality from 73.67% to 56.10%. This quality is more comparable to the earlier 3410 MWT CE plants and the newer Korean System80-modified steam generators. The number of

tubes in the dry-out region decreases from approximately 2700 tube to 1600 tubes. The analyses performed by APS and ABB-CE also demonstrate that these modifications do not have a significant impact on either flow induced tube vibration, structural integrity of the shroud, tubesheet sludge deposition or feedring water hammer susceptibility.

Unit 2 As-built Modification

The modification program in U2R6 was the second undertaking of this major in-situ modification of the PVNGS steam generators. The feedring modification as described previously, and depicted in Figure IV-11, was installed by APS/ABB-CE per design plans. The separator orifice plates were removed. The shroud modification was completed in SG 22 as planned, and the machining problems described in Reference 37 were eliminated. However, upon entry into SG 21, engineers from APS and ABB-CE found that the hot leg annulus gap was insufficient for the EDM tooling. APS elected to defer this modification.

As reported in Reference 37, APS contracted ABB-CE to analyze the as-built condition in Unit 3 to consider the effects of the imbalance on steam generator and plant operation (Reference 31). This assessment bounds the as-built condition in Unit 2. From the Unit 3 assessment, ABB-CE and APS concluded:

1. A core inlet temperature difference will not result

2. Impurity concentrations are expected to be lower in the as modified SGs than the as-designed, especially in regions where impurities are most concentrated (areas of highest quality at the top of the hot leg). It is anticipated that there will be a slight increase in blowdown impurity concentration. Associated with this increase will be a general improvement in the clean-up of the steam generators. Since SG 22 has had the most severe impact of ARC region defects, the as-built condition is considered more advantageous than if the reverse were true.

3. Table IV-3 provides a summary of original design, modification design and as-built thermal hydraulic parameters.

Table IV-3 Thermal Hydraulic Parameters

T/H Parameter	Original Design	As Designed Modification	As-built SG 21	As-built SG 22
Maximum Quality	74%	56%	68%	56%
Recirculation Ratio	3.08	3.72	3.36	3.72
Tubes in Dryout	3000	1600	2800	1600
Tubes above Q=65%	1600	0	500	0
Flow Stability Margin	< 0.75	0.78	< 0.75	0.78

Note: Flow Stability Margin is computed to determine if flow induced vibration could result in increased tube wear in the periphery locations. The threshold value which would indicate the onset of instability is 1.0.

F. Power Uprate

In Reference 32, APS proposed to increase the rated thermal power (RTP) of PVNGS 1, 2, and 3 by 2%. The analyses performed in support of the Technical Specification Amendment included an assessment of the impact to the PVNGS steam generators. For Unit 2, the increase in power was implemented on May 25, 1996. Table IV-4 provides a comparison of design, actual (pre-uprate) and predicted operating parameters at the higher RTP. The supporting analyses were performed for both reduced feedwater temperature and design feedwater temperature. ATHOS modeling indicates that reduced feedwater temperature results in improved thermal hydraulics in the ARC region (Reference 33). Operation at normal feedwater temperature results in a slightly poorer thermal hydraulic condition than shown for the as-built (modified SG) parameters given in Table IV-3, but increases megawatt output due to improved thermodynamic efficiency. The values in parentheses reflect operation at design feedwater temperature. Feedwater

temperature will be optimized during Cycle 7 to coincide with SG modification benefit and short term power needs.

Table IV-4 Plant Operating Parameters

	Design (VWO)	Prior to RTP Change	Increased RTP
Rated Thermal Power	3800Mwt	3800 Mwt	3876 Mwt
RCS Flow	164 Mlbm/hr	170 Mlbm/hr	170 Mlbm/hr
Hot Leg Temp	621 °F	611 °F	611°F
Cold Leg Temp	565 °F	555 °F	554 °F
Delta Temp	56 °F	56 °F	57 °F
SG Pressure	1070 psia	970 psia	970 psia
Feedwater Temp	445 °F	445 °F	420-445 (445°F)
Feedwater Flow	18.1 Mlbm/hr	16.9 Mlbm/hr	16.6 (17.2) Mlbm/hr
Secondary Side (VWO)	4030 Mwt	3817 Mwt	3899 Mwt
Main Turbine/Generator	1403 Mwe	1325 Mwe	1341 (1351) Mwe

Run Time Analysis Impacts

With respect to the operating run time analyses contained in this report, the key operating inputs of hot leg temperature and steam pressure (ΔP) were adjusted for worst case values. The increase in RTP coincides with a decrease in cold leg temperature from 555 °F to 554 °F. This one (1) degree reduction in temperature permits the maintenance of the current Thot of 611 °F. However, APS has found that the plugging asymmetry in the Unit 2 steam generators increases Thot to SG 22, from 0.5 - 2 degrees. Consequently the primary APTECH analysis was run with a bounding Thot assumption of 613°F. The independent RG 1.121 analysis was run at 611°F and 615°F to assess the impact of increased temperature. The analyses supported full Cycle 7 run time with the slight increases in Thot which may occur. An increase in secondary steam pressure (10-20 psia) may also occur. This increase would have the effect of reducing the operating ΔP, and therefore the allowable flaw sizes are bounded by the analysis inputs.

The analysis to increase RTP also reassessed allowable plugging limits. An increase in the allowable steam generator plugging limits to accommodate projected steam generator degradation was computed. Per the supporting analyses, for non-LOCA transients, up to 3000 total tubes may be plugged in both steam generators combined, not to exceed an asymmetry of 1000 tubes between the two steam generators. For ECCS LOCA transients, 2750 tubes may be plugged per steam generator.

V. STRUCTURAL AND LEAKAGE INTEGRITY ANALYSIS

A. Description and Background of PVNGS Structural Integrity Model

As described in References 2, 8, and 10, the PVNGS Degradation Management Program employs conservative analyses for assessing structural and leakage integrity in the PVNGS steam generators. The defense-in-depth analytical program developed by APS uses two (2) independent probabilistic methods to predict the end of cycle (EOC) condition of the steam generator tubing. Condition monitoring techniques as described in Reference 10 are used for benchmarking and validating past, present and future operating and inspection results. To date, the analytical approach described here and in previous PVNGS submittals to the USNRC, has provided good and conservative correlation between predicted and actual EOC conditions.

The primary model, for determining allowable operating time, estimates EOC structural margins in accordance with the guidance given in Regulatory Guide 1.121, Generic Letter 95-05 and the draft Regulatory Guide - *Steam Generator Tube Integrity.* The Unit 2 model was developed by APTECH with input and review by APS. The method of analysis, including results, are described in detail in Appendix A. The calculational framework is similar to previous assessments presented to the USNRC Staff. The model is a mechanistic simulation of the operating and inspection processes used at PVNGS. The simulation addresses the historical development of cracks, including, crack initiation and crack growth, the detection of cracks and the removal of detected SCC flaws from service. The initiation and growth of new cracks during a given cycle of operation are combined with the population of defects which have not been detected by eddy current inspections.

The model framework, as first presented for Unit 1 in Reference 2 in August 1994, was validated for Unit 1 in the U1R5 inspections conducted in April 1995. The model methodology has been refined, and has continued to conservatively predict EOC conditions in all three units. The model simulations have predicted with high confidence that structural margins would not be exceeded for either circumferential cracks in the tubesheet transition region or ARC region axial cracks. Subsequent inspections have

successfully confirmed the conservative nature of the analysis methods and results. The number of observable defects, when adjusted for inspection transients such as the use of Plus Point MRPC, have also indicated good correlation with model predictions. With each successive inspection, the model methodology has been refined. Industry and regulatory evolutions have also been incorporated in the analyses. The critical features of the model, as well as recent refinements, include:

1. APS and APTECH have continued to review PVNGS specific tube pull results and industry data to support the relevance of characterizing ARC region defects with MRPC signal response. The length and depth profiles of cracks in tubes pulled from Unit 2 have been analyzed to identify the critical regions of these profiles relative to burst pressures. Structurally significant crack lengths and depths have been defined. The structurally significant crack lengths have been correlated to crack lengths observed from the eddy current MRPC data. Average crack depths over the structurally significant crack lengths have been determined, and correlated with MRPC eddy current probe voltage readings.

2. As in References 2, 8 and 10, MRPC voltage changes have been analyzed to estimate crack growth rate via a correlating function developed from PVNGS tube pull data. Variations in the correlation of voltage with structurally significant depth have been included to reflect the effects of variable crack morphology. A new numerical study, requested by APS, is provided in Appendix A which assesses the factors which impact the extraction of crack growth rate information from ECT techniques that may contain significant measurement and detection uncertainties.

3. In contrast to earlier versions of the model, some concepts from Generic Letter 95-05 have been incorporated. Degradation growth rates are sampled directly from past observations without using a fitted analytical distribution. Also, negative voltage growth rates are treated as zero growth. If a crack survives detection in the simulated inspection a new

growth rate can be selected for the next cycle. The new growth rate may be the zero, equal to or different from the old growth rate. This simulation feature is consistent with actual plant operation, as changes in operation such as secondary chemistry, primary temperature, or maintenance, such as chemical cleaning could alter growth rates from one cycle to the next.

4. APTECH has included in their simulation, an additional probabilistic defect attribute not contained in previous PVNGS analyses. The additional attribute, the defect form factor, relates the crack maximum depth to average structural depth. The effect of a randomized form factor on the simulation outcome is to increase the number though-wall defects for large defect populations, thereby increasing the conservatism of the leakage integrity analysis.

A complete description of the primary model analysis conducted by APTECH is provided in Appendix A. In the following discussion, APS provides an informational summation of the APTECH analyses. The evaluation contains a condition monitoring assessment of the Reference 10 projected end of Cycle 6 steam generator conditions with the actual observations from U2R6. The key assumptions, methods, results and conclusions for the Unit 2 Cycle 7 operational assessment are also provided.

B. Unit 2 Cycle 6 Condition Monitoring Assessment

The APS structural and leakage integrity analyses are designed to provide several means of comparing steam generator inspection results with predictive modeling projections. This evaluation is the basis of condition monitoring as described in the draft Regulatory Guide, *Steam Generator Tube Integrity.* As in Reference 39 for Unit 3, the critical parameters for a PVNGS Condition Monitoring Assessment are: number of ARC region defects, crack length, crack depth and leakage monitoring.

1. Number of defects

For Unit 2 Cycle 6, APS has provided to the USNRC Staff a pre-outage prediction

(Reference 10) and a benchmarking verification (Appendix A) for the number of expected cracks in the U2R6 steam generator inspections. Figure V-1 is a reprint from Reference 10 of the projected number of Plus Point detected cracks for U2R6. The mean value is approximately 650 cracks. The total number of cracks detected in U2R6 was 286 cracks. The projection provided in Reference 10 was clearly conservative as the analysis overpredicted the actual number observed by a factor of two (2). APS and APTECH personnel have held several meetings to discuss a number of possible explanations for the large difference. Several of these included:

- POD effects
- Multi-population effects
- "Shutdown" of the initiation process
- Defect propagation rate effects

With regard to the POD effects, if the Plus Point POD under-estimated the benefit of the Plus Point probe, a larger number of small defects could have been observed. As stated in Reference 10, the POD curve was a simulated probability distribution. In Section IV, APS reported that recent pulled tube data for upper bundle cracking in a CE model steam generator has been made available. From this new data, an actual Plus Point POD function was developed by APTECH using logistic regression. A comparison of the actual Plus Point POD and the "effective" POD used in Reference 10 showed good agreement (See Figure 5.5 of Appendix A). Therefore, a POD effect is not substantiated by this independent validation.

As indicated in Section II, several root cause factors were identified during 1993-1994 tube pull analysis as contributing to aggressive ARC region degradation. These included: a susceptible region of high quality and contaminant concentration, tube-to-tube crevice formation, bridging ridge-like deposits, increased sulfate levels, and a caustic crevice pH. Additional factors, such as, less than standard metallurgical microstructures for High Temperature Mill Annealed (HTMA) tubing and cold working at the OD tube surface from manufacturing scratches were observed in the tubes removed from Unit 2. Not all of the tubes in the ARC region are suspected to realize all or most of these conditions. APS and APTECH have developed statistical models which suggests that several sub-populations of tubes with different initiation and

growth rates may exist within the ARC region. While this hypothesis cannot be validated, it can provide some insight into long term population evolution.

APS and APTECH also considered the possibility that remedial measures such as chemical cleaning, Thot reduction and improved secondary chemistry, may have "shutdown" the crack initiation process. However, model simulation of this assumption required an unreasonable time estimate of occurrence to show the observed effect, and therefore can not be substantiated at this time.

Finally, a review of an assumption that APS actions at PVNGS have had an effect on the defect propagation rate, provided a numerically compelling explanation for the reduced number of defects found in U2R6. The simulation performed for Unit 2 Cycle 6 in Reference 10, was modified to incorporate the observed Cycle 6 growth rate only for Cycle 6. All other modeling inputs and assumptions such as initiation function and prior Cycle growth rates were unchanged. Figure 5.13 of Appendix A graphically depicts this comparison. The results indicate good agreement with the actual number of defects observed in U2R6.

As stated in previous submittals, the APS structural integrity simulation model is a multiple cycle model, and therefore benchmarking is a critical part of simulation resolution and validation. Although benchmarking contains the results of the current inspection, it serves as a condition monitoring tool from a simulation verification perspective. As is shown in Figures 6.3 - 6.6 from Appendix A, the histograms for the last 4 inspections demonstrate the validity of the benchmark and the appropriateness of the model assumptions and methods.

Therefore, from a condition monitoring perspective, the analysis results with respect to the number of defects observed has been verified.

2. Crack Length

APS concludes that the axial crack lengths observed during U2R7 are bounded by the assessment contained in Reference 10 and Appendix A of this report. As reported in previous submittals to the USNRC, the crack length distribution used in the

[illegible]

[illegible]

structural and leakage integrity analyses is not a projection, but an input distribution of actual ECT results. This distribution combined with material property effects from randomized flow strength values, specific to the Unit 2 tubing, are treated probabilistically and defined in distributional terms as the allowable structural limit.

In 1994 (Reference 2), APS demonstrated via tube pulls, good correlation between structurally significant crack length (crack section where length and average depth over this length lead to the minimum predicted burst pressure) and the **0.115 pancake coil detected length.** Conversely, with the implementation of the Plus Point probe, APS analysis indicated that the shallow long crack contribution of the improved Plus Point POD was grossly overconservative in calculating tube burst probabilities. APS also found that end-of-cycle crack length distributions are very similar cycle to cycle and from PVNGS unit to unit. Consequently, condition monitoring can be performed one of two ways. Since many defects can only be detected by the Plus Point and not the 0.115 pancake coil, comparative assessments can be performed of sequential Plus Point length distributions. If subsequent Plus Point length distributions are bounded, the crack length distribution given as Figure 5.9 (reprinted as Figure V-2) of Reference 10 is considered valid. A further re-evaluation of recordable pancake coil lengths was also performed for U2R6 data. That data, shown in Figure 5.2 of Appendix A was compared directly with Figure V-2. The recordable pancake lengths are bounded by the input distribution. A single point comparison can be made for the frequency of defects greater than 0.8 inches. This length is considered the critical 100% through-wall crack length for tube rupture at MSLB conditions. Figure V-2 estimates that approximately 40% of the detected lengths exceed 0.8 inches. As indicated in Figure 5.2 of Appendix A, in U2R6 only 20% of the detected RPC crack lengths exceed this value, thereby demonstrating the conservatism associated with the conditional probability of tube burst calculation. As such, the length criterion is satisfied for Cycle 6 in Unit 2.

3. Voltage/Depth

Verification of Cycle 6 model output with respect to observed defect depth is performed three ways. The APS acceptance criteria for steam generator structural integrity requires a low probability of an end of cycle Regulatory Guide 1.121

exceedance. The value calculated for end of Cycle 6 in Unit 2 was reported in Reference 10 as 0.01. For criteria verification, ECT data is interrogated to determine by voltage, phase and length, if any defects exceed the Regulatory Guide 1.121 allowable flaw size limits. If ECT data is insufficient or questionable, or if MRPC voltage is greater than two (2) volts, APS has employed alternative techniques such as Ultrasonic Testing (UT) or in-situ pressure testing.

Although sizing of ARC region flaws with MRPC has not been industry qualified a voltage/average depth correlation has been developed by APS via tube pulls conducted in Unit 2. As reported in Section IV, APS has taken measures to reduce analyst and technique variability through training, performance demonstration and data quality requirements. Voltage measurements are reviewed the APS Level III to further reduce analyst variability.

The U2R6 results indicate that zero (0) structural integrity exceedances were recorded. The largest Unit 2 MRPC voltage recorded for a defect greater than 0.5 inches in length was 1.23 volts. The best estimate regression fit structural limit is 2.25 volts. Defects with voltages up to 3.07 volts have been in-situ tested in a 1994 Unit 2 inspection to 4260 psia with no leakage or burst.

Validation of the APS leakage integrity model also serves as a check of model input and output with respect to large defects. Monitoring of operational leakage has been identified as performance criterion in the Draft Regulatory Guide supporting the anticipated Steam Generator Rule. The analysis contained in Reference 10 estimated the probability of a through wall defect at 0.055. As described in References 2, 8 and 10, APS employs strict Administrative controls on steam generator primary to secondary leakage. Shutdown of the unit is required if leakage exceeds 50 gallons per day (GPD). APS has also installed and/or upgraded the PVNGS leakage monitoring equipment, including the installation of N-16 monitors, which would be particularly sensitive to leaks from upper bundle ARC region defects. No evidence of leakage was detected at the end of Cycle 6. None of the defects detected by ECT were estimated to be through wall indications. Therefore, the leakage integrity criterion has been satisfied and the possibility of large defects is considered remote.

Finally, the defect growth rate probability distribution function is the dominant stochastic variable in the APTECH based run time analysis. For the PVNGS steam generators, defect growth rates are computed using 0.115 pancake coil voltage growth statistics obtained from sequential inspections. Previous inspection data is re-evaluated for the presence of precursor signals. If possible, voltage values are recorded by the APS Level III for these signals. While the uncertainty in the voltage/depth correlation may be attributed to coil response to different defect morphologies, the success of past APS projections may be attributable to the consistency in voltage response of a particular defect as it evolves from initiation to detection. With respect to condition monitoring, the best comparison of an as-measured ECT parameter with model predictions is contained in Section 6.3 and Figure 6.7 of Appendix A. A simulation generated distribution of RPC voltage is compared with the observed voltage distribution from U2R6. The simulated distribution bounds the observed inspection results with good agreement. The simulation is a measure of the joint effects of the POD function and growth rate function employed in the PVNGS predictions.

The Cycle 6 condition monitoring assessment compared the key predictive parameters of defect quantity, length and voltage/depth. Based on this evaluation, APS concludes that the as-found condition of the Unit 2 steam generators, in U2R6, satisfies all structural and leakage performance design basis criterion for PVNGS.

C. Operational Assessment - Unit 2 Cycle 7

As in previous reports submitted to the USNRC regarding ARC region degradation, APS has performed an operational assessment to demonstrate that structural and leakage integrity criteria are satisfied for a proposed operating interval. Past operating cycles in Units 2 and 3 have been adjusted accordingly when inspection results did not support full cycle operation. However, as with Unit 3 (Reference 37) and Unit 1, the most current Unit 2 inspection results, when assessed using proven probabilistic techniques, support full cycle operation in Unit 2 for Cycle 7. The analysis results, when combined with the defense-in-depth philosophy of the PVNGS Steam Generator Degradation Management program, assure safe steam generator operation until the next scheduled refueling outage

in September 1997. The APTECH analysis methodology and results are presented in detail in Appendix A.

As in previous analyses, APS has defined PVNGS structural and leakage integrity acceptance criteria for steam generator tubing based on Regulatory Guide 1.121, NUREG 0844, and guidance, where applicable, from Generic Letter 95-05 and the Draft Steam Generator Rule and Regulatory Guide. The probabilistic based acceptance criteria provides high confidence that the conservative safety margins established in Regulatory Guide 1.121 are maintained and the conditional probability of tube burst is acceptably low. The criteria applied for Unit 2 include:

1. The conditional probability of tube burst at postulated accident conditions should not exceed 0.01 for ARC region axial cracking

2. The induced leak rate at a postulated accident event should not exceed the total charging pump capacity for the primary coolant system. Additionally this leak rate should not result in radiological consequences in excess of 10CFR 100 and GDC 19.

3. To provide consistency with past PVNGS submittals, the probability of a Regulatory Guide 1.121 exceedance is also calculated. This structural criteria assures that the steam generator tubing shall retain margins against burst which are consistent with the safety factors implicit with the stress limit criteria of the ASME Code Section III. That is, the probability of tube burst is estimated at the more limiting loading condition of three (3) times the normal operating pressure across the tube wall.

The following information summarizes key information used in support of the APS operational assessment.

1. Structural Integrity

The APTECH modeling approach, endorsed by APS, characterizes the structural integrity of steam generator tubing by performing a Monte Carlo simulation of the physical

processes of crack initiation, growth, eddy current detection and removal of all detected cracks from service. This approach, which is supported by a large plant specific database of pulled tubes, provides a reasonable physical characterization of critical cracks and crack growth. This characterization is key to the overall success of managing ARC region degradation, and in demonstrating that the proposed operational run times provide adequate safety margins. The success of this technique has been confirmed though a strong condition monitoring assessment program. The EOC conditions for Unit 2 Cycle 6, Unit 3 Cycle 5 and Unit 1 Cycle 5 have compared well with the PVNGS model predictions of low probability of RG 1.121 exceedance, through-wall leakers and numbers of detected defects. The U2R6 inspection results indicate a continued trend towards corrosion rate reduction, and no challenges to plant safety have been identified.

The probabilistic model used for Unit 2 is similar to the models described in References 2, 8 and 10, and is designed to simulate four basic processes:

Upper bundle crack initiation
Crack Growth
MRPC Inspection
Removal and repair of degraded tubes

The first step in the Monte Carlo process is the definition of defect attributes. Some attributes such as tube material properties, crack length and form factor (discussed below) remain constant throughout the simulation process. Conversely, crack growth rates are modeled such that crack progression can change from cycle to cycle. The development of a form factor is an important distinction from previous PVNGS analyses. The form factor relates the crack maximum depth to average structural depth. This factor provides for a broader range of defects which may leak, but not burst. Each of the attributes is obtained from sampling an appropriate probability distribution function.

Crack initiation times are selected from a Weibull distribution. The Weibull shape (slope) and scale parameters are based on the past history of reported indications as a function of operating time. Since reported indications have grown sufficiently to be detected by an eddy current inspection, the actual point of crack initiation must be at some earlier point in time. A constant time shift provides an adequate estimate of the crack initiation time. The magnitude of the shift then depends on the average crack growth rate. After a crack is

initiated, it is considered to grow at a constant through-wall rate until the next inspection. If a crack survives detection in the simulated inspection, a new growth rate can be selected for the next cycle. The new growth rate may zero, equal to or different from the old growth rate.

Crack growth rate is considered to be the dominant stochastic variable in the run time analysis. The crack growth probability function is derived from voltage growth statistics from sequential MRPC inspections. The rates of voltage change are converted to crack growth rates using:

$$\frac{\Delta D}{\Delta t} = \frac{\Delta V}{\Delta t} \times \frac{\Delta D}{\Delta V}$$

where:

ΔD = depth change
ΔV = voltage change
Δt = inspection interval

The slope of the depth/voltage function ($\Delta D/\Delta V$) is obtained from a linear regression of voltage and structural average depth from PVNGS tube pull data (References 2, 8 and 10). While the uncertainty in the voltage/depth correlation is large and may be attributed to coil response to different defect morphologies, the success of past APS/APTECH projections may be attributable to the consistency in voltage response of a particular defect as it evolves from initiation to detection.

As mentioned previously, all SCC indications are plugged on detection. Consequently, voltage growth values are dependent on the ability to review historical information and identify and size precursor signals. APS is concerned with analyst variability, and measurement error and uncertainty associated with this process. As indicated in Section IV, APS has strived to minimize analyst variability and provide consistent measurements, by requiring that a lissajous graphic and C-scan hardcopy of each defect is generated and

a final review of ARC region ECT calls for the Unit 2 data is conducted by the APS Level III analyst and APS Engineering. Since the overall depth of the defect population in Unit 2 has been reduced by successive Plus Point inspections, the percentage of paired pancake coil was correspondingly reduced. That is, the percentage of detected U2R6 defects which have interpretable precursor pancake coil data is lower. The growth rate data set used by APTECH consisted of 92 pairs of MRPC voltages (See Appendix B). APS requested that APTECH conduct a numerical study as to this impact of this condition. The study was designed to assess the factors which affect the extraction of limited growth rate information from inspection results which contain significant measurement and detection uncertainties. The study presented in Appendix A supported the use of paired pancake coil voltage to define growth rates. Although a similar correlation of tube pull based voltage-to-average depth correlation does not exist for Plus Point voltages, the established improvement in crack detection of the Plus Point was used as an additional validation. Consequently, APTECH confirmed that neglecting unpaired voltage data points, brought about principally by detection issues, is not unconservative. The crack growth distribution function for Unit 2 is shown in Figure 5.8 of Appendix A.

Crack length and tubing material properties are assigned to a crack at initiation and are considered intrinsic properties of the crack. The flow stress distribution (yield plus ultimate strength) is a normal distribution fitted from as-built Certified Material Test Reports (CMTRs) for the Unit 2 steam generator tubing. As stated previously, a comparative assessment of the eight ARC region inspections since the Unit 2 tube pulls indicate that the crack length distribution remains fairly constant unit to unit and inspection to inspection. This observation is consistent with data for other plants and types of degradation. While inspection results indicate that run time has little effect on the overall length distribution, the number of cracks at a given length does depend on the cycle run time.

The ratio of maximum crack depth to structurally significant crack depth along with the structurally significant crack length, characterizes the crack morphology when axial degradation in conservatively modeled as a single planar crack. The depth ratio is defined in the PVNGS analysis as the form factor. The treatment of form factor can have an impact on the assessment of the pressure and leakage bearing capacity of axial cracks with some through wall penetration. The distribution function for defect form factor and a

complete description of crack morphology considerations is described in Appendix A.

The probability of detection (POD) function described in Appendix A is used to perform a simulated inspection. The process is straight forward. The crack depth at EOC is known from the crack initiation time, the total time available for growth, and the past selected crack growth rates. A uniformly-distributed random number is selected between 0 and 1. If the random number is below the POD curve at the depth of interest, the crack is detected, otherwise, it is missed and remains in service. In an effort to match predicted EOC detected cracks for Unit 2 Cycle 6, APTECH developed a simulated POD for the Plus Point probe due to the lack of industry pulled tube data. Since 1995, pulled tube data for free-span axial cracks has become available. From the new data, an industry Plus Point POD function was developed using logistic regression, and compared with the Unit 2 simulation generated POD function. The good agreement of the POD functions validates the use of the simulated POD function used in Reference 10. To maintain consistency, the model simulation uses a PVNGS pancake coil POD for the inspection simulations through U2M5-2 and the PVNGS Plus Point POD for U2R5, U2R6 and the projected U2R7.

Upon detection, the length, depth, form factor and tensile properties associated with the crack are used to compute a burst pressure. This determines if a tube burst at main steam line break could occur, or if RG 1.121 structural limits have been exceeded. Design basis burst pressure calculations for PVNGS are described in Appendix A and documented in Reference 34. The calculation shows that crack depths required for bursting at main steam line break (MSLB) or three times normal operation pressure (3ΔP) differential are large enough to virtually assure detection by MRPC. Hence, as the program flags detected indications, there are no undetected structural limit exceedances.

Simulation of the overall process for the desired run time history is repeated up to 10,000 times to obtain reasonable estimates of the probability of a tube burst given a postulated main steam line break at EOC and to develop a distribution function for the number of RG 1.121 structural limit exceedances.

The model as described above was exercised using Monte Carlo simulation techniques described in Appendix A. The probability of exceeding Regulatory Guide 1.121 structural margins has been computed by APTECH as a function of run time in Cycle 7. After 16.5 months of operation, the probability of more than one RG 1.121 exceedance was

estimated to be 3 $\times 10^{-4}$. The structural integrity model was run for MSLB conditions. 10,000 model trials were conducted and no simulated bursts were encountered. Thus null results yielded a calculated conditional probability of tube burst for a postulated main steam line break of 3×10^{-4} and therefore an acceptance criteria of 10^{-2} is satisfied. The probability of multiple tube burst at MSLB conditions is extremely low at 4.5×10^{-8}. These positive results were further examined for reasonableness by assessing the extreme value distribution of simulated structural depths. The analysis indicated a low probability of a defect exceeding 50% through wall (See Figure 6.1 of Appendix A). This result is considered to be an outcome of successive Plus Point inspections and a positive effect of remedial measures taken by APS to reduce crack growth rates.

With regard to leakage integrity, over 20,000 Monte Carlo run time simulations were processed, and no instance of through-wall crack penetration was observed. The analysis provided as upper 95% confidence estimate of the probability of any leak during Cycle 7 of 4×10^{-5}. This result when compared to the PVNGS design basis calculations for accident induced leakage exceeds the acceptance criteria established by APS in Reference 2, and demonstrate with high confidence that steam generator tube leakage integrity in Unit 2 can be maintained until end of Cycle 7.

As a further measure of assuring that the modeling approach applied by APS/APTECH addresses the issues associated with crack growth and inspection uncertainties, the USNRC staff has requested during a public meeting, that PVNGS statistical models for predicting steam generator tube integrity be benchmarked against previous inspection findings. One benchmark of the probabilistic model is the comparison of the predicted versus observed number of defect indications. Since the model is probabilistic, there is no single prediction of the number of indications at a given inspection. However, some outcomes are more likely than others, and this is illustrated by the histograms of Figures 6.3 to 6.6 in Appendix A. The actual number of indications observed match up well with the most likely calculated number of indications for combined U2R4/U2M5-1, U2M5-2, U2R5, and U2R6 inspections.

Additionally second benchmark technique was employed for this analysis. A simulation generated distribution of 0.115 pancake coil voltage is compared with the observed voltage distribution from U2R6. The simulated distribution bounds the observed

inspection results with good agreement. This simulation is a particularly good measure of the joint effects of the POD function and growth rate function employed in the PVNGS predictions.

2. Structural Integrity - Independent Assessment

As with analyses presented in References 2, 8 and 10, and in accordance with the guidance given in 10CFR50 Appendix B, APS has employed the use of an independent alternative calculational method as a means of performing a check of the results generated from the primary structural model. APTECH was also retained by APS to perform this independent assessment. The computer code, input assumptions and analysis methodology are sufficiently diverse to assure independence. These independent assessments have been performed previously for the Unit 1, 2, and 3 steam generator evaluations with excellent agreement. This section provides a discussion of the modeling assumptions and results for the independent assessment.

A. Model Description

Currently, APS utilizes the BALIFE computer code (Reference 21) developed by APTECH to perform long term statistical analyses for evaluating the effects of repair and remedial measures on steam generator life. The BALIFE code applies rigorous Bayesian reliability estimation methods to the prediction of failure frequencies from life- or age-to-failure data. The code allows the user to develop a "prior" distribution, such as a set of Weibull curves and slopes, to establish a baseline for a typical industry damage mechanism. Plant specific data can then be used to calculate a "likelihood" function which provides the probability of specific values of the Weibull slope in light of PVNGS historical failure data. Finally, a "posterior" distribution is calculated as the product of the "prior" and "likelihood" functions. The code is independent of the computer code used in the primary model and has been verified by APTECH by comparing the BALIFE and exact solutions for several "textbook" classical and Bayesian problems, and by constant benchmarking against service failure and cracking data.

The BALIFE code has also been utilized by APS for assessing the probability of Regulatory Guide exceedances (RGEs) as a self-check of more detailed structural integrity analyses, as in the case of Units 1 2, and 3, in References 2, 10, and 8

respectively. The same calculational framework was employed by APTECH for predicting EOC conditions for Unit 2 Cycle 6. The current analysis is devoted to estimating an upper bound to the chance of ARC region defects exceeding the structural margins required by Regulatory Guide 1.121 during Cycle 7. As a new feature, the analysis has been modified to estimate probability of tube rupture at MSLB conditions for further comparisons with the critical structural criteria assessed in the primary model.

The results of this assessment have been compared with the corresponding calculation of condition burst probability determined by the primary structural integrity model. As with past reports, this independent assessment is in good agreement with the sophisticated mechanistic model. The key assumptions, inputs and model improvements developed for this analysis are summarized below.

B. BALIFE Model Assumptions and Input

I. The BALIFE Code, Version 5.051 has been used as in previous analyses with no changes. The Code incorporates an improved "Nuclear Inspection Option" which permits the user to assess varying inspection samples on the tube population of interest. This is considered useful when considering random, subsequent or repeat sample ECT inspections for the specific steam generators or the input of industry trends where different inspection samples are routine. This computer program has been benchmarked against the previous ten inspections in Units 1, 2, and 3, with reasonably accurate results for the subject ARC region cracks and other tube crack modes. Like the primary model, the BALIFE Code overpredicted by a factor of two (2) the number of indications found in U2R6. As reported in Reference 10, APS and APTECH did not consider the benefits of chemical cleaning and Thot reduction. Nor has the model previously considered the possibility of crack initiation shutdown or multiple-population statistics in the structural integrity assessment.

As a result, none of these effects are explicitly modeled in BALIFE. Instead, the model is updated at each inspection to incorporate detected

crack frequency. The code compensates for smaller than expected crack frequencies, as was observed in U2R6, by adjusting Weibull parameters accordingly. Unlike the primary model, the BALIFE code assesses crack frequency and consequential predictions on a per steam generator basis. The calculated probability values are then statistically combined to assess the potential of Regulatory Guide 1.121 exceedance for the Unit. This value is then compared to the primary model results.

II. In Reference 8, a separate MRPC probability of detection (POD) analysis was performed by APTECH, independent of a Packer Engineering POD curve provided in Reference 8, from the PVNGS plant specific tube pull data. The best estimate APTECH POD curve from Reference 8 is used in the Unit 2 BALIFE baseline analysis. The effects of various possible inspection improvements were then calculated by altering the best-estimate POD values.

III. Results from previous BALIFE predictions for Unit 2 and 3 show that ARC region ODSCC has a Weibull slope or β value range of between 2.65 and 4.5. This compares well with industry data. Different scale parameters (θ, the age at which 63.2% of the affected population has failed) were calculated directly from inspection data. These calibrated θ values were based on damage categories related to ECT detection levels, as defined below. For example, different categories were established for MRPC, Bobbin "I" codes and bobbin depth calls as a means for tracking crack growth and θ values.

IV. As the model is designed to track the damage classification via the inspection results, certain inspection data assumptions are used. In Appendix B the inspection results for U2R6 and look-up data from U2R5 are listed. The results list the number of tubes with ARC region defects detected by two (2) types of probes, Plus Point and 0.115 pancake coil. As documented in Reference 10, the 0.115 pancake has been demonstrated to generally be less sensitive to ARC region defects than the Plus Point coil. In all inspections conducted at PVNGS all tubes detected by 0.115 pancake were detected by Plus Point. Approximately

60% of the Plus Point calls were observed by 0.115 pancake. In order to maintain a reasonable comparison between a Plus Point influenced pancake coil detection and a blind pancake coil inspection, any tube with pancake voltage indication less than 0.2 volts would is considered as Plus Point detectable only.

C. Inspection Model Assumptions and Input

The BALIFE modeling approach employed by APTECH was supplemented to include an inspection/repair capability for ARC region defects. The following assumptions were used in the simulation:

I. Tube Population

- At the start of operation for each steam generator, each of the 2700 unplugged tubes was susceptible to the subject arc region cracks. Only eventual plugging can immunize a tube against this failure mode.

- The 2700 tubes are assumed to come from the same statistical population. This is done in spite of much evidence of preferred cracking sites within the ARC region, especially in Unit 2, SG 22. The use of only one tube population is very conservative in that it exaggerates the calculated probability of an RGE. The single-population model ignores the fact that all or most tubes in the worst ARC population subsets have already been plugged.

II. Damage Classification for Each Tube

- At any stage in the life of an unplugged tube, its worst crack belongs to one of six damage categories:

    a. Nonexistent and waiting to be initiated. Conservatively, the chance of plugging such a tube is assumed to be zero (i.e., no false calls).

b. Plus Point-size. These cracks are defined to be so small as to be possibly detected only by Plus Point MRPC. The probability of a miss is quantified below and defined as POM(pp)=1-POD(pp).

c. Bobbin-size. These cracks are defined to be large enough to be possibly detected by *either* Plus Point MRPC *or* Bobbin or both. This probability of a miss by *both* inspections is quantified below and defined as POM(Bobbin)= 1-POD(Bobbin).

d. RGE-size. These cracks are defined to exceed the size defined in Regulatory Guide 1.121. They are large enough to be possibly detected by *either* Plus Point MRPC *or* Bobbin or both. This probability of a miss by *both* inspections is quantified below and defined as POM(RGE)=1-POD(RGE).

e. Leak or burst given steam line break conditions. This is a new classification allowing us to compute a probability of tube burst at SLB conditions This provides a second independent check of the primary model not provided in previous APS submittals.

f. Leak or burst under normal operation. Computes the probability of a tube burst at a normal operating ΔP of 1267 psi.

- As any tube ages, its worst crack moves from category one toward category six and plugging is the only way to halt this process.
- Any tube found to be damaged is immediately plugged.

III. Inspection Simulation

- From the POD curve (Figure V-3), a given MRPC inspection has a probability, POM(RGE) = 6% of missing a crack (conversely a 94% probability of detection) which exceeds Regulatory Guide 1.121 based on the POD curve.

- Each single inspection has a probability POM(Bobbin) = 15% assigned for missing a crack that is "normally" detectable by Bobbin inspection.

- Each single inspection has a probability POM(pp) = 40% of missing damage that is normally detectable by MRPC inspection (35-40% through-wall). This 40% value is especially conservative in that the value remained unchanged from Reference 2 despite demonstrated improvements realized from the Plus Point.

IV. Crack Growth

- BALIFE is used to calculate a no-inspection build-up of various size cracks. From this a calculation, defect life factors can be generated and average number and age of cracks are predicted. As mentioned previously, any detected defect is removed from service. Within any inspection interval, the average number of tubes which move from a damage category (e.g., MRPC size) to the next higher damage category (Bobbin size) is directly proportional to the number of tubes within the lower damage category at the start of the interval. The life factors used in the APTECH analysis are contained in Appendix A and are expressed as multiples of lifetime to a Plus Point sized crack and normalized for a Weibull slope of five (5).

V. Unit 2 Forecast from Statistical Combination of SG 21 and SG 22 Forecasts

- The RGE analysis was first run for each steam generator. To make the calculations for RGEs in Unit 2, it is assumed that the number of RGEs

in SGs 21 and 22 are independent Poisson-distributed variables. So steam generator "failure" rates were added to obtain best estimates for Unit 2. To calculate confidence bounds for Unit 2, the variances associated with the steam generator RGE confidence interval ranges were also added. (In statistics, the variance is defined as the standard deviation squared.)

VI. Future EFPY Buildup and Damage Exposure

- *No credit* was taken for primary temperature reduction from 621°F to 611°F or chemical cleaning. In other words, damage is assumed to correlate with EFPY through the Bayesian Weibull model. EFPY is assumed to increase with future calendar time at past rates. However the model was run at a base case of 611°F and a worst case of 615°F to account for possible temperature fluctuations associated with plugging asymmetry and power uprate. The temperature increase is accounted for via an exposure rate increase of 1.17 times past rates.

D. Results

In past analyses, APS has established a reasonable acceptance criteria for demonstrating that Regulatory Guide 1.121 structural margins will be maintained for EOC conditions at PVNGS. This criteria is supplemental to the performance criteria specified by the USNRC Staff for demonstrating steam generator tube integrity (conditional probability of tube burst of 10^{-2}). The Regulatory Guide 1.121 acceptance criteria previously applied by APS is defined as:

> *There must be a 90% probability that one or fewer tubes will be expected to violate Regulatory Guide 1.121 limits for axial cracks during the specified operating period.*

Fifty-four different cases were run by APTECH and the estimates are summarized in Appendix A. Based on the results, the probability of one (1) or fewer RGEs after 16.5 months of operation is $4x10^{-4}$ or a 99.96% chance of no more than 1 tube exceeding

Regulatory Guide 1.121 structural limits. The confidence level associated with this upper bound forecast is 90%. The best-estimate forecast shows a 99.98% chance of one or less RGE in Unit 2 for Cycle 7. The 90% upper bound tube burst probability at MSLB is 2.6×10^{-4}. A complete list of the results from the independent assessment is provided in Appendix A.

E. Comparison with Primary Model Results

The results of the independent evaluation have been compared with the primary analyses for the operating run proposed in this evaluation. The independent model, as in the previous analyses, is a Bayesian approach relying on historical ECT results, whereas the primary model simulates, via Monte Carlo techniques, the mechanics of crack evolution. The structural margin assessments of the two markedly different techniques are in close agreement. The conditional probability of tube burst at MSLB conditions from the independent model is estimated to be 2×10^{-4} at an upper 90% confidence value. This corresponds extremely well with the 3×10^{-4} value at 95% confidence estimated by the primary model. This unique method of independent verification provides additional confidence in the analytical approaches and the treatment of uncertainties.

3. Leakage Integrity Assessment

A. Introduction

The Draft Regulatory Guide, *Steam Generator Tube Integrity,* specifies that the total potential accident induced leak rate and consequential radiological consequences should be assessed relative to a performance criteria for leakage integrity. A design basis steam generator leakage model for PVNGS was initially developed by APTECH and APS for Unit 3 Cycle 5 and Unit 2 Cycle 6. The model and results are discussed in detail in References 8 and 10. The model predicts end of cycle (EOC) primary to secondary leakage under faulted loads by probabilistic methods using a Monte Carlo numerical simulation of deterministic models for crack opening area, and statistical distributions for material strength and through-wall crack lengths. The analysis follows a mechanistic approach whereby the beginning of cycle (BOC) flaw distributions of undetected defects are projected over the operating cycle to give the EOC probability

distribution for through-wall cracks (leakers) should a main steam line break (MSLB) occur. The probability distribution function (PDF) is developed from an evaluation of the progression of ARC region ODSCC during the cycle, and an analysis of ligament integrity under MSLB loads to establish the number and through-wall extent of leaking defects at EOC. The PDF is calculated via the primary structural model. A design basis deterministic model for leakage from an axial through-wall crack with variable crack aspect ratio formed the basis of the Monte Carlo leakage model. The leakage calculation and the associated fluid mechanics model was first developed for Unit 3 Cycle 5 and presented to the USNRC Staff in Reference 8.

The following information is a description of the modeling development and format used in PVNGS leakage integrity analyses. As indicated in Appendix A, after 20,000 computer simulations, no instances of through-wall cracks were observed for Unit 2 Cycle 7. Reference 10, by comparison, found that in 5,000 simulations for Unit 2 Cycle 6 operation, a total of 475 instances of through-wall cracks were observed. Further computer time for Unit 2 was deemed unnecessary, as the lack of the development of a single through-wall indication in 20,000 run time simulations demonstrates that leakage is not an issue. At the very least, the Unit 2 Cycle 6 leakage analysis results are bounding. As reported in the condition monitoring assessment, at the end of Unit 2 Cycle 6, there was no detectable leakage and no indication of a through-wall defect upon the completion of the U2R6 ECT inspections.

Appendix A also reports on the future evolution of leakage prediction for the run time models conducted by APTECH. The primary changes consist of a more conservative treatment of crack opening area, input of form factor and an calculational input and output format more consistent with the requirements of the Draft Regulatory Guide. Some comparisons with industry data are also provided in Appendix A. As mentioned previously, the probability of through wall crack penetration is very remote. Therefore further changes in the PVNGS design basis leakage calculations using the methodology described in Appendix A were unwarranted. For consistency, a description of the current design basis leakage model for PVNGS is described as follows.

B. Leakage Integrity Acceptance Criteria

The Draft Regulatory Guide, Steam Generator Tube Integrity, specifies that the total potential accident induced leak rate and consequential radiological consequences should be assessed. The performance criteria is satisfied if this leak rate does not exceed normal charging pump capacity and the associated radiological consequences do not exceed 10CFR100 limits and are in accordance with GDC 19. The PVNGS accident induced leakage limits and associated assumptions are addressed in Section 15.1.5 of the PVNGS UFSAR. Based upon the review of the UFSAR assessment it has been determined that a primary-to-secondary leak rate could increase to 6 gpm without the associated radiological consequences exceeding 10CFR100 limits. The APS probability of leakage model acceptance criteria was defined as the demonstration of at least 95% probability that leakage from EOC steam generator tubing conditions, with consequential MSLB, will remain below the UFSAR limit of 6 gpm. It should also be noted that an upper bound limit of 6 gpm is well below the make-up capacity of one charging pump at PVNGS. Charging pump capacity is 44 gpm per pump.

C. Leakage Integrity Model Description

The deterministic leakage model for MSLB conditions was developed from the PICEP computer code (Reference 12). The PICEP program was developed by EPRI for performing leak before break evaluations for reactor piping and steam generator tubing. The leakage algorithm in PICEP is based on two-phase flow for subcooled and saturated liquid discharge through a crack. A schematic illustration of the two-phase flow model used in PICEP to represent the flow through a cracked tube was presented in Reference 10. The critical flow equations are based on a modified Henry's homogeneous non-equilibrium critical flow model. Non-equilibrium "flashing" mass transfer between liquid and vapor phases, fluid friction due to surface roughness, and convergent flow paths are modeled. The model for the leak before break analysis was validated with data discussed in References 13-15.

The leak rate will depend on several parameters including flaw length, crack opening area, tube differential pressure and fluid properties. Other parameters that affect flow rate, such as surface roughness and irregular or nonplanar crack surfaces are

conservatively accounted for in the leak rate model. The model will determine the flow through a freespan crack under MSLB conditions as a function of crack length, crack opening displacement, and crack aspect ratio as defined by the ratio of exit to inlet crack lengths (ie. ℓ_{OD}/ℓ_{ID}).

To allow leak rate calculations to be solved rapidly in the Monte Carlo simulations, key PICEP output was fitted with regression equations. PICEP leak rates were calculated for many combinations, covering all crack opening areas and aspect ratios of interest. The crack opening area varied from zero to 0.1 square-inch and the aspect ratio was varied between 1 and 20. The regression equations selected for Monte Carlo modeling fit these computed leak rate values with an average error of less than 2%. The regression fit was conservatively biased so as not to underestimate a PICEP leak rate by more than 3%. Details of the leakage model development and the regression equations model are given in Reference 29. To ensure a conservative leakage model, assumptions were made that are reasonable and conservative for predicting flow through a crack that will exaggerate the rate of flow. These include:

- Maximum MSLB primary differential pressure was assumed.
- A nominal crack surface roughness of 2E-4 inch is assumed for SCC surfaces per the PICEP manual.
- The crack faces are conservatively taken as flat (i.e. Nonzig-zag).
- The crack opening area in the model development is conservatively based on a rectangular opening equal to the crack opening displacement times the crack length.

To verify the behavior of the leakage model (i.e., flow assumptions and regression fit), a comparison was made between the PICEP/APS model developed by APTECH for Palo Verde, and the Hernalsteen (LABOLEAK) model which is based on a single-phase flow approximation as discussed in Reference 19. This comparison is shown in Reference 10 where the leak rate Q was plotted as a function of crack opening area (A_C). The PICEP/APS model is observed to be conservative over the range of interest in A_C.

D. Fluid Conditions

For the design basis accident conditions, the largest ΔP in the tube will occur during a postulated main steam line break. Per PVNGS UFSAR Section 15.1.5, the maximum primary pressure following a main steam line break is 2400 psia. This peak pressure occurs at the start of the event at 100% power with a concurrent loss of offsite power. The secondary side pressure is conservatively assumed to be at vacuum conditions caused by the instantaneous loss in secondary pressure during the event. This combination of high peak primary pressures with vacuum conditions on the secondary side yield the largest possible ΔP across the tube wall, and therefore, the highest hoop stress and flow conditions, for design basis or faulted events. The fluid conditions used in the leakage model are therefore:

$p_i = 2400$ psia
$p_o = 0$ psia
$T = 593\ ^{\circ}F$

Where T is based on the average of the hot leg (inlet) and cold leg (outlet) temperatures.

E. Crack Opening Area

The crack opening area (A_c) calculational method was conservatively selected from a comparison of three crack opening displacement models; namely, Erdogan solution (Reference 12), Tada/Kumar solution (Reference 15) and the Hernalsteen model (Reference 19). A plot of these three crack opening area models was provided in Reference 10. It was determined that the Erdogan model, which is contained in PICEP, is too limited and produces smaller Ac values at larger crack lengths (Reference 15). The Tada/Kumar and Hernalsteen models give similar results for A_c, with the latter being slightly more conservative at the crack lengths of interest. The Hernalsteen model was therefore used for computing A_c in the probabilistic analysis.

F. Crack Distributions

As in Reference 8, the distributions for through-wall crack ID and OD lengths are calculated from the crack growth simulation data from the primary structural model. During a meeting with the USNRC Staff on July 12, 1995, APS indicated that the possibility of mechanical breakthrough without burst would be included in the Unit 2 Cycle 6 leakage model (Reference 10). In the leakage model, the assumption of maintaining a semi-elliptical crack shape but constant crack length as growth occurs in the depth direction is reasonable and consistent with the structural integrity Monte-Carlo simulation model. Hence, following maximum crack depths in excess of the wall thickness permits calculation of the through-wall crack lengths. It was postulated that some population of crack geometries, mechanical breakthrough may occur during a MSLB without leading to a full tube burst. That is the case for deep cracks whose total length is less than the critical through-wall crack length for burst.

The Framatome burst equation does not extrapolate to the true burst pressure for a through-wall crack. This burst pressure is given by the EPRI equation given in Reference 26, which has received a full industry review. For very deep cracks, the Framatome equation does correlate with the onset of local but not necessarily global fracture. Reference 10 contained a plot of burst pressure versus relative crack depth for steam generator tubes containing stress corrosion cracks approximately 1.1 inches in length. The data was taken from NUREG/CR-2336. The test was terminated upon loss of the pressurizing medium. A full tube burst was not required to terminate the test. The Framatome equation together with the EPRI equation for burst pressure for through-wall cracks provide a good definition of the combinations of crack lengths and depths where local breakthrough, but not full burst will occur. At breakthrough, the through-wall crack length is taken as equal to the total crack length and this is input to leak rate calculations.

G. Probabilistic Method

The analysis technique as described in Reference 10, is based on a Monte Carlo numerical solution of the deterministic models for crack opening area and corresponding leakage which employ statistical distributions as input for material

strength (σ_y and σ_u) and through-wall crack lengths (l_{OD} and l_{ID}). The baseline Monte Carlo evaluation for crack leakage rate is based on 20 million unit simulations of combinations of conservatively simulated OD and ID through-wall crack lengths and material strengths. The leak rates for any and all simulated cracks in both steam generators is combined to give the total leakage for the affected unit.

H. Leakage Assessment

As indicated in Appendix A, after 20,000 computer simulations no instances of through-wall cracks were observed for Unit 2 Cycle 7. Reference 10, by comparison, found that in 5,000 simulations for Unit 2 Cycle 6 operation, a total of 475 instances of through-wall cracks were observed. Further computer time for Unit 2 was deemed unnecessary, as the lack of the development of a single through-wall indication in 20,000 run time simulations demonstrates that leakage is not an issue. At the very least, the Unit 2 Cycle 6 leakage analysis results are bounding. As indicated in the condition monitoring assessment, at the end of Unit 2 Cycle 6, there was no detectable leakage and no indication of a through-wall defect upon the completion of the U2R6 ECT inspections. Thus, it is highly likely (i.e., greater that 98% probability per Reference 10) that the maximum leakage computed at EOC for Unit 2 under worst case MSLB conditions will remain below the applicable 10CFR100 limits for off-site doses. The performance criteria specified in the Draft Regulatory Guide are satisfied as there is an extremely low probability that a MSLB will result in accident induce leakage in excess PVNGS charging pump capacity, and that the radiological consequences will not exceed the allowables of 10CFR100.

4. Operational Assessment Summary

The operational assessment conducted for Unit 2 Cycle 7 satisfies the established structural and leakage criteria for PVNGS. The operational assessment results support at least 16.5 months of full power operation in Unit 2. The upper 95% confidence estimate of the conditional probability of tube burst for a postulated main steam line break is 3×10^{-4} and therefore an acceptance criteria of 10^{-2} is satisfied. The chance of any leaking crack during Unit 2 Cycle 7 is correspondingly remote at 4×10^{-5}. The analysis found that the performance of two successive inspections using the Plus Point Probe have significantly

reduced the severity of undetected beginning of cycle degradation. It is also apparent by the analysis results that remedial measures such as Thot reduction, chemical cleaning and improved secondary chemistry have substantially reduced degradation growth rates since 1993. Consequently, all of the analytical run time calculations performed for this report, strongly support full cycle operation in Unit 2.

VI. OPERATIONAL RESPONSE

A. Background

The Draft Regulatory Guide, *Steam Generator Tube Integrity*, describes primary-to-secondary leakage monitoring as an important defense-in-depth measure which can assist plant operators in monitoring overall tube integrity during plant operation. Since 1994, APS has implemented an integrated leakage detection and response program, using equipment and procedure upgrades, to permit plant operators to detect and respond to changes in steam generator primary-to-secondary leakage. The program was established to provide reasonable assurance that the unit will be shutdown prior to a significant leak or steam generator tube rupture should tube degradation exceed expected values. The program is designed to provide clear and unequivocal plant management support to commence orderly shutdown should leakage exceed very stringent administrative limits. APS has also endeavored to ensure that adequate staff, equipment and organizational resources are in place to implement this program, using a combination of radiation monitors and laboratory radiochemical analyses. The integrated leakage program at PVNGS is not only prescriptive, but preventative as well, as the program is supported by extensive steam generator inspections and conservative plugging criteria which ensure that all detected SCC defects are removed from service. A description of how PVNGS operational response is supported by integrating inspection, repair, leakage monitoring, and operator training is provided below.

B. Inspection and Repair

The objective of the inspection and repair program at PVNGS is to support structural and leakage integrity. Industry studies and observations indicate that the use of enhanced NDE techniques and conservative plugging criteria can reduce the likelihood of forced shutdown due to primary-to-secondary leakage.

Inspection Program

A comprehensive steam generator inspection program reduces the risk of leaving a significant defect(s) in service. Enhanced inspections (100% bobbin, MRPC,

Plus Point) as summarized in Section IV were conducted in the region affected by ARC degradation. As demonstrated by assessments performed in this report, and as indicated by sensitivity studies performed by APS in Reference 10 and EdF in Reference 23, the level of MRPC and bobbin coil inspections and industry leading use of new ECT data acquisition and analysis technology is statistically significant in terms of preventing through-wall defects and significantly lowers the probability of tube rupture or operationally induced leakage for the specified period of operation.

Plugging Criteria

APS employs an administrative plugging criteria which is significantly more restrictive than required in the PVNGS Technical Specifications, particularly in regard to crack (SCC) defects which are plugged on detection. The current plugging criteria is given in Table VII-1. Before the steam generators are returned to service, a review is conducted by APS Engineering of all eddy current indications. As previously stated, all tubes with detected cracks are removed from service regardless of flaw size or depth. As demonstrated in Section V, and supported by data presented by EdF in Reference 23, this action also reduces the potential for leakage and/or tube rupture, since no known crack defects are left in the steam generator.

Table VI -1 - PVNGS Administrative Plugging Criteria

Mechanism	Plugging Criteria	Basis
Stay Cylinder Batwing Wear	≥ 20% wear	Rapid wear for these supports has been observed at several CE plants
Cold Leg Corner Wear	≥ 20% wear	Rapid wear has led to inservice leaks at PVNGS
Support Wear	≥ 35% wear for tubes previously examined with no indications detected	Per PVNGS fretting wear curves, tubes exhibiting this wear rate could exceed structural margins in following operating cycle
Support Wear	≥ 40% wear for tubes which had previous indications of > 10%	Technical Specification 4.4.4.4.a.6

Mechanism	Plugging Criteria	Basis
Axial Cracks	All detected/suspected cracks	Conservative criteria based on NDE capabilities
Circumferential Cracks	All detected/suspected cracks	Conservative criteria based on NDE capabilities. All tubes removed from service are stabilized
Volumetric Indications	Engineering and ISI evaluation for ≤ 40% through-wall	Possible corrosion growth is monitored and defects exhibiting noticeable change are removed from service
Loose Part Wear	All locations with detectable wear for unretrievable parts	APS Loose Part Study - All tubes removed from service for loose part wear are stabilized

C. Leakage Monitoring and Procedural Control

Monitoring Equipment Upgrades

As reported in References 2 and 17, APS has implemented upgrades in leakage monitoring equipment to provide plant operators with enhanced diagnostic tools. These upgrades have been implemented in all three PVNGS Units. The specific upgrades include:

Steam Generator Blowdown Radiation Monitors (RU-4 & RU-5)

The blowdown monitors currently monitor the downcomer flowstreams rather than hot leg blowdown to provide greater sensitivity for detection and response to primary to secondary leakage.

Condenser Vacuum Exhaust Monitor (RU-141)

The CVE Monitor has been changed to an in-line monitor, and a graph of monitor reading verses leak rate is trended.The alert setpoints for the CVE Monitor were decreased to four (4) times background to provide earlier alarms to plant operators in the event of increasing primary leakage.

N-16 Monitors

N-16 monitoring systems have been permanently installed in all three units to provide an additional diagnostic tool for primary-to-secondary leakage detection. The design incorporates sodium iodide crystal detectors connected to the existing radiation monitoring system. The detectors are located on the main steam lines (4 per unit) in the Turbine Building and are connected to the condenser exhaust high-range effluent monitor (RU-142), which had been abandoned in place when the condenser vacuum exhaust was routed to the plant vent. Existing control room alarm and indication capabilities were utilized. Addition information regarding changes in procedures, training and use of the N-16 monitors are described later.

Procedural Upgrades

The current leak rate limits at PVNGS were developed in response to the Unit 2 tube rupture event in March 1993. An administrative maximum shutdown limit was set at 50 gpd, well below the Technical Specification limit of 720 gpd and EPRI guidelines and Draft Regulatory Guide levels of 150 gpd. It should be noted, that a review of the available data prior to the tube rupture event determined that these administrative limits would have resulted in an orderly shutdown prior to the Unit 2 tube rupture at the end of Cycle 4.

The administrative limits, as well as, a leak rate hierarchy are proceduralized in PVNGS Station Manual Procedure 74DP-9ZZ05, Abnormal Occurrence Checklist. The hierarchy includes:

1. Leak Rate < 10 GPD: Monitor and perform accurate leak determinations shiftly. Inform the Control Room of all leak rate calculation results. Additionally, the CVE Monitor is trended shiftly. If an increase of three-fold is observed, Step 2 is entered. Finally, alarm setpoints for RU-141, as well as RU-4 and -5 are evaluated daily.

2. Leak Rate 10-50 GPD: Monitor and perform accurate leak determinations shiftly. Inform the Control Room of all leak rate calculation results. Additionally, if a increase in leak rate by 50% is observed within a 24 hour period or less, or the leak rate is greater than 25 GPD, formation of an evaluation team is required to address continued operation. Additional monitoring of RU-141, RU-4 and RU-5 is also required.

3. Leak Rate > 50 GPD: Following verification of the calculated leak rate by the most readily available method outlined in 74DP-9ZZ05, the Shift Supervisor initiates a plant shutdown, and then informs plant management.

N-16 Monitor Enhancements

As a result of an INPO audit regarding the use of N-16 monitors, significant changes have been incorporated in PVNGS procedures, training and setpoints regarding the use of N-16 monitors. These changes include:

- Changes to the PVNGS Station Manual Procedure 74DP-9ZZ05, *Abnormal Occurrence Checklist*, to monitor N-16 trends with a steam generator leak < 50 gallons per day, as well as to evaluate indication and alarm setpoints for the N-16 monitors.

- Changes to PVNGS Station Manual Procedure 43AO-3ZZ08, *Steam Generator Tube Leak*, to include the N-16 monitors as a means of identifying the affected steam generator. Cautions and classroom and simulator training have been incorporated to alert operators to the possible effects of "shine" from the affected steam lines to monitors on the unaffected steam lines.

- Changes to PVNGS Station Manual Procedure 74RM-9EF41, *Radiation Monitoring System Alarm Response*, to include a cross check of the N-16 monitors when alarms are received on the blowdown radiation monitors. Cautions have also been

added to this procedure to alert personnel to the effects of "shine".

- Calculations have been performed using the ATHOS thermal-hydraulic model to assess N-16 monitor response to varying leak rates at various leak locations (See Figure VI-1) as well as different reactor power levels (Reference 24).

- An engineering/chemistry evaluation has been completed to determine optimum setpoints for the N-16 monitors.

APS believes the enhancements to the leak rate program provide the operators with information and direction to recognize a leak prior to break, regardless of whether the defect is axial or circumferential. In 1995 and 1996, APS representatives met with EdF officials as part of a technology transfer program. Based on these discussions, APS believes that the overall PVNGS steam generator program to be comparable to the programs instituted by EdF. The defect management program at EdF of comprehensive eddy current inspections, preventative plugging criteria and strict leakage limits ($\cong$ 32 GPD) has resulted in 600 reactor years without tube rupture. Based on a 10^{-2} frequencies, six (6) significant leakage events could have occurred within this operational time frame. As stated previously, an added conservatism is provided with the PVNGS philosophy, that all detected cracks are removed from service, whereas EdF plants are permitted to operate with known SCC defects in service.

D. Operator Training

The Draft Regulatory Guide also describes the need for licensees to develop training scenarios for leakage events including tube rupture. As stated in Reference 2, extensive simulator training of operations personnel for tube rupture events as well as upgrades to the Emergency Operating Procedures permit faster identification and isolation of the affected steam generator. Improvements in operator response assure that in the unlikely event of a main steam line break with consequential multiple tube ruptures, the resulting offsite doses are maintained less than 10CFR100 limits.

Since the March 1993 Unit 2 tube rupture, PVNGS has developed the following operator training material. Industry Events Training provided to operations personnel details the Unit 2 SGTR event itself. This training was followed by three (3) simulator scenarios.

NUS08 Re-creation of the Unit 2 SGTR event with an emphasis on procedure changes implemented since March 1993, including N-16 monitor response to a SGTR (pre and post trip). Additional emphasis is placed on event impacts to the Auxiliary Steam System.

NUS22 Simulates a SGTR with a stuck open Main Steam Safety Valve. Similar to NUS08, emphasis was placed on functional changes to the plant procedures. Event impact on N-16 monitor response, square root extractor and low flow cutouts on the HPSI flow indicators were emphasized.

NUS28 Simulates a SGTR with a Loss of Offsite Power. In this scenario the ability to recognize an SGTR without the use of radiation monitors was emphasized. As with the other simulator scenarios, procedure changes were reviewed as well as discussions regarding HPSI low flow cutouts.

The training program changes have been in place since the third quarter of 1993. Upon implementation of the training in the new simulator, APS identified a previously unknown transient effect. The new simulator model for the SGTR event indicated that steam generator pressure increased during refill, with the auxiliary feed and MSIV's isolated. This phenomenon was categorized as a compression effect, and this unexpected transient response resulted in four crew failures. Based on the discovery of this condition, APS implemented additional training to provide operators with the approaches to adjust for the compression effect. As stated previously, additional training was also provided on the radiation shine effects associated with N-16 monitors. This updated training was provided in both the classroom and the simulator.

VII. SUMMARY

The analyses and evaluations contained in this report demonstrate that the operating, inspection and repair program for the Unit 2 steam generators permit safe operation of Unit 2 for the remainder of Cycle 7. The progression of ARC region degradation in Unit 2 is such, that further restrictions on cycle lengths are not required. The ability to manage the corrosion mechanisms in the PVNGS steam generators is a primary safety and strategic objective. The comprehensive actions completed by APS to achieve these objectives are summarized below:

- Primary temperature reductions of approximately 10°F have been implemented in all three PVNGS units to take advantage of the temperature dependence of SCC growth rates. Stress corrosion cracking is a thermally activated process, and the effects of temperature reduction can be quantified for SCC mechanisms in terms of activation energy for an Arrhenius rate equation.

- APS has removed 31 tubes from service, and has conducted extensive NDE and destructive examination in an effort to determine causal effects of corrosion damage, and to provide substantial improvements in field ECT acquisition and interpretation.

- APS has implemented the industry recommended secondary chemistry controls and performed full bundle chemical cleaning in all PVNGS steam generators to mitigate the initiation and propagation of secondary side IGA/SCC. The laboratory evidence from tubes removed from Unit 2 during U2M5-1 show a favorable change in crack crevice chemistry tending towards neutral conditions. APS has established action levels for sulfate

which are more restrictive than EPRI guidelines, that require reduced power operation or shutdown for sulfate levels as low as 20 ppb.

- APS has incorporated an integrated operational response program, utilizing equipment and procedure upgrades, to provide plant operators the ability to detect and respond to changes in steam generator primary-to-secondary leakage, and shutdown the unit prior to a significant leak or steam generator tube rupture should unexpected tube degradation exceed expected values. A number of improvements to the N-16 monitors and their use have been implemented during 1994-1995. The integrated leakage program at PVNGS is considered prescriptive, as well as preventative, as the program is supported by extensive steam generator inspections and conservative plugging criteria.

- APS, with support from APTECH, has developed state-of-the-art probabilistic models for assessing operating cycle lengths which maintain the safety margins specified in Regulatory Guide 1.121 and satisfy the performance criteria outlined in the proposed Steam Generator Rule.

- APS has developed a probabilistic leakage model to assess EOC leakage as a result of secondary overpressurization events. The model demonstrates that operation with ARC region degradation will not result in offsite releases in excess of 10CFR100 limits should a MSLB event occur during Cycle 7.

- APS has developed a risk model to assess the impact on core damage probability for plant operation with degraded steam generator tubing. The calculation is based on the output from the primary structural model. The Unit 2 model (Reference 10) indicated that operation with ARC region degradation in Unit 2

during Cycle 6 represented a negligible impact on core damage probability. The calculated impact was in fact lower in risk than the performance of an additional midcycle outage in Unit 2. The model for Unit 2 Cycle 6 is considered to be bounding for Unit 2 Cycle 7.

These actions are all part of a defense-in-depth approach employed by APS, to provide reasonable assurance that PVNGS Unit 2 can be safely operated until the next scheduled refueling shutdown for further steam generator inspections. This approach also includes actions and assessments which consider the remote possibility of transient defect growth or the presence of an unknown defect mechanism resulting in a large leakage event. An additional best estimate analysis was previously performed and submitted to the USNRC in Reference 9, which indicated that in the unlikely event of a main steam line break without or with consequential single or multiple tube ruptures, with the current administrative limits on reactor coolant system dose equivalent iodine, the resulting offsite doses are estimated to be less than 10CFR100 limits. Additionally, APS has continued to update training and conduct simulator testing of operations personnel for tube rupture events, and has developed upgrades to the Emergency Operating Procedures which permit faster identification and isolation of the affected steam generator.

It is therefore APS's position that the implementation of the elements of the PVNGS Degradation Management Program, as described in this report, constitutes a conservative and comprehensive approach which ensures that adequate structural and leakage integrity is maintained for normal operating, transient and postulated accident conditions for Unit 2 Cycle 7, consistent with General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10CFR50 Appendix A.

VIII. REFERENCES

1. *Unit 2 Steam Generator Tube Rupture Analysis Report,* Submitted to USNRC by letter, 102-02569-WFC/JRP dated July 18, 1993

2. *Status of PVNGS Steam Generator Activities, August 1994,* submitted to the USNRC by letter 102-03083-WLS/AKK/JRP dated August 11, 1994

3. *Examination of Palo Verde-3 Steam Generator Tubes Removed During the 1994 Outage, Final Report,* V-PENG-TR-004, dated July 1994

4. *Examination of Palo Verde Unit 2 Steam Generator Tubes,* M. Blaszkiewicz and W. A. Myers, Westinghouse, July 1994

5. ABB-CE Report TR-MCC-272, *Examination of Steam Generator Tubes Removed form Palo Verde Unit 2 in 1993*

6. *Palo Verde-2 Steam Generator Tube Bend Region Examination and Metallurgical Evaluation,* V-Peng-TR-005, Volumes 1-3, January 1995

7. Letter, *Mid-Cycle Outage Schedule,* 102-03010-WLS/RAB/JRP, dated June 20, 1995

8. *Unit 3 Steam Generator Evaluation, May 1995,* submitted to USNRC via letter 102-03364-WLS/SAB/JRP dated May 19, 1995

9. Letter, *Steam Generator Tube Evaluation,* 102-02585-WFC/TRB/RAB dated July 25, 1993

10. Sweeney, K., *Palo Verde Nuclear Generating Station Unit 2 Steam Generator Evaluation, August 1995,* submitted to the USNRC by letter 102-03465-WLS/SAB/JRP dated September 7, 1995

11. *Unit 3 Steam Generator Inspection Report, U3R4, July 1994,* Submitted to USNRC by letter 102-03082-WLS/RAB/JRP dated August 11,1994

12. EPRI NP-3596SR, *PICEP - Pipe Crack Evaluation Program,* Revision 1, December 1987

13. *Documentation of Verification Information for the PICEP Program,* APTECH Calculation AES-C-1893-6, October 1993

14. *Leak Rate Model for Axial Flaws in Freespan Tubing Under Faulted Loads,* APTECH Document AES-C-2402-1, April 1994

REFERENCES (cont)

15. EPRI NP-6897-L, *Steam Generator Tube Leakage Experiments and PICEP Correlations,* July 1990

16. Pitner P., Riffard T., *Application of Probabilistic Fracture Mechanics to Optimize the Maintenance of PWR Steam Generator Tubes,* September 1993

17. APS Study, 13-NS-B03, *Impact of Unit 2 Steam Generator Accelerated Crack Growth Phenomenon on Unit 1 and 3 Cycle 4 Core Damage Probability,* August 1993

18. APS Study 03-NS-B54, *Evaluation of SGTR Risk associated with operating Unit 3 until U3R5 due to axial crack propagation in the Unit 3 Steam Generators,* April 1995

19. EPRI NP-6864-L, *PWR Steam Generator Tube Repair Limits,* Revision 2, August 1993

20. *Probabilistic Analysis for Leak Rate at End-of-Cycle Under MSLB for Freespan Axial Cracks,* APTECH Document AES-C-2402-2, April 1995

21. Besuner, P. M, *BALIFE Version 5.05*1, April 1995

22. Besuner, P.M, *Development and Verification of Combined Analysis of Transformer and Boiler Tube Failures,* Volumes 1 and 2, Final Report, prepared for New England Electric, March 26, 1988

23. Regulatory Guide 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes,* Issued for Comment, August 1976

24. USNRC Generic Letter 95-05, *Voltage-Based Repair Criteria for Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,*

25. USNRC Letter dated October 19, 1994, *Summary of Meeting Held on August 22, 1994 to Discuss Steam Generator Issues*

26. Keating, R.K., Hernalsteen, P. and Begely, J.A., *Burst Pressure Correlation for Steam Generator Tubes with Through-wall Axial Cracks,* Draft Report to EPRI, Westinghouse Electric Corporation, SG-95--2-00, dated February, 1995

27. USNRC Letter dated August 22, 1995, *Summary of Meeting held on July 12, 1995, to Discuss Steam Generator Issues*

REFERENCES (cont)

28. Blanchat, T.K., *New Tube Bundle Heat Transfer Correlation and Flow Regime Maps for a Once Through Steam Generator*, Master of Science Thesis, December 1988

29. *Leak Rate Model for Axial Through-wall Cracks in Freespan Tubing Under Faulted Loads*, APTECH Document AES-C-2402-1, June 6, 1995

30. Report, *Palo Verde Unit 3 Steam Generator 2 Thermal-Hydraulic Analysis with Design Modifications*, ABB-CE Document No. CR-9419-CSE95-1116 Rev 0, August 15, 1995

31. Report, *Steam Generator Modification Project Report on the Operational Issues of Partially Modified Steam Generators of Unit 3*, V-PENG-ER-005, Revision 0, November 1995

32. *Proposed Amendments to Facility Operating Licenses and to Technical Specifications and Various Bases, Related to Power Uprate*, submitted to the USNRC by Letter 102-03578-WLS/AKK/GAM dated January 5, 1996

33. Memorandum, *Steam Generator Corrosion Effects of Stretch Power and Steam Generator Modifications*, 433-00263-RFS/JT, dated April 18, 1996

34. APTECH Calculation, *Allowable Axial Flaw Sizes for Steam Generator Tubes in Palo Verde Unit 2*, Calculation No. AES-C-1893-5, APTECH Report Number 93031893-1Q, Revision 2 dated June 1996

35. Draft Regulatory Guide x.xx, *Steam Generator Tube Integrity*

36. Letter, Jim Begley (APTECH) to Kevin Sweeney (APS), *Additional Information Regarding Report AES96072812-1 Rev 1*, AP-PIT-DEC-012,dated December 27, 1996

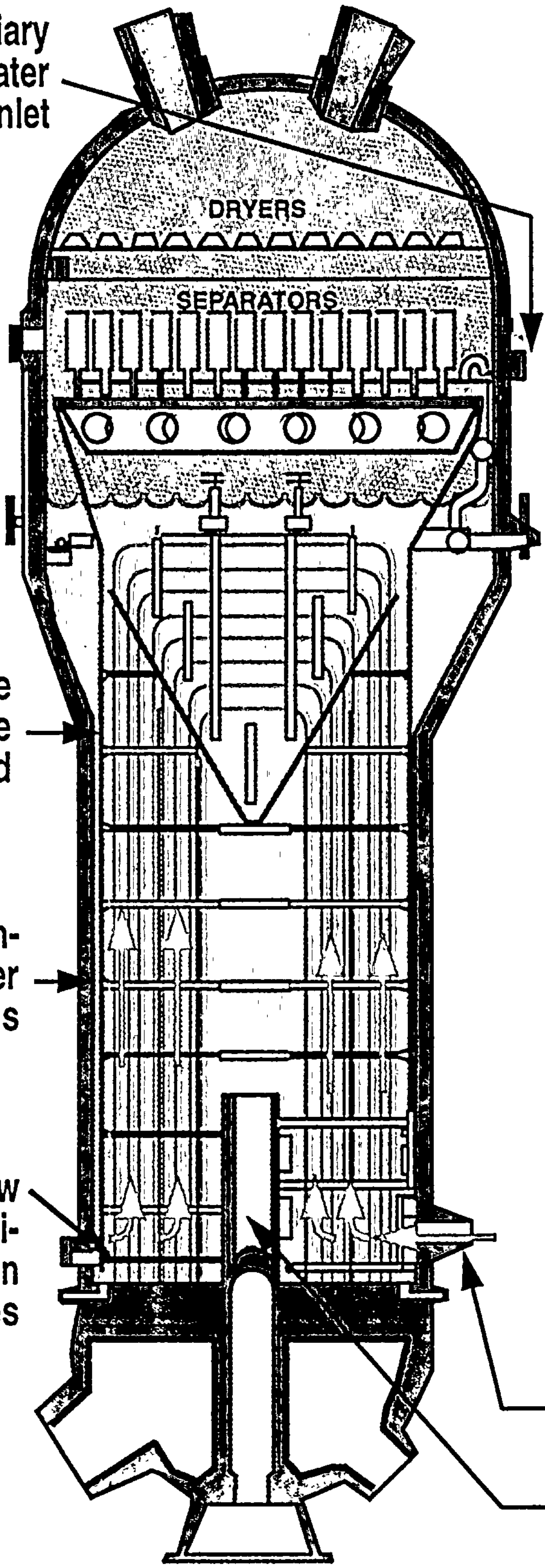
37. Sweeney, K., *Palo Verde Nuclear Generating Station Unit 3 Cycle 6 Steam Generator Evaluation, July 1996*, submitted to the USNRC by letter 102-03748-WLS/AKK/JRP dated August 6, 1996

38. *Report on Steam Generator Free Span Axial Cracking*, Rev 00, ABB Combustion Engineering Nuclear Operations, September 1996

39. Letter to USNRC, *Response to Request for Additional Information Regarding Unit 3 Cycle 6 Steam Generator Evaluation Report*, 102-03822-JML/AKK/JRP dated November 27, 1996

Integral Economizer Steam Generator Axial Flow

Side Elevation



System 80 Steam Generator Design and Operating Parameters

Plant rating	3817 MWt
Primary inlet temp:	
Original	621.2 F°
Current	611.0 F°
Primary outlet temp.	564.5 F°
Primary pressure	2250 Psia
Primary flow rate (per SG)	82×10^6 lb/hr.
Steam pressure	1070 Psia
Steam flow rate (per SG)	8.59×10^6 lb/hr.
Feedwater temp.	450 F°
Number of tubes	11,000
Tube diameter	0.750 inches
Tube wall thickness	0.042 inches
Tube material	Inconel 600 (Ni-Cr-Fe)
Overall length	67 feet
Steam drum diameter	247 inches
Evaporator shell diameter	189.5 inches
Weight (dry)	1,585,000 lbs.

FIGURE II-1

SG-3B

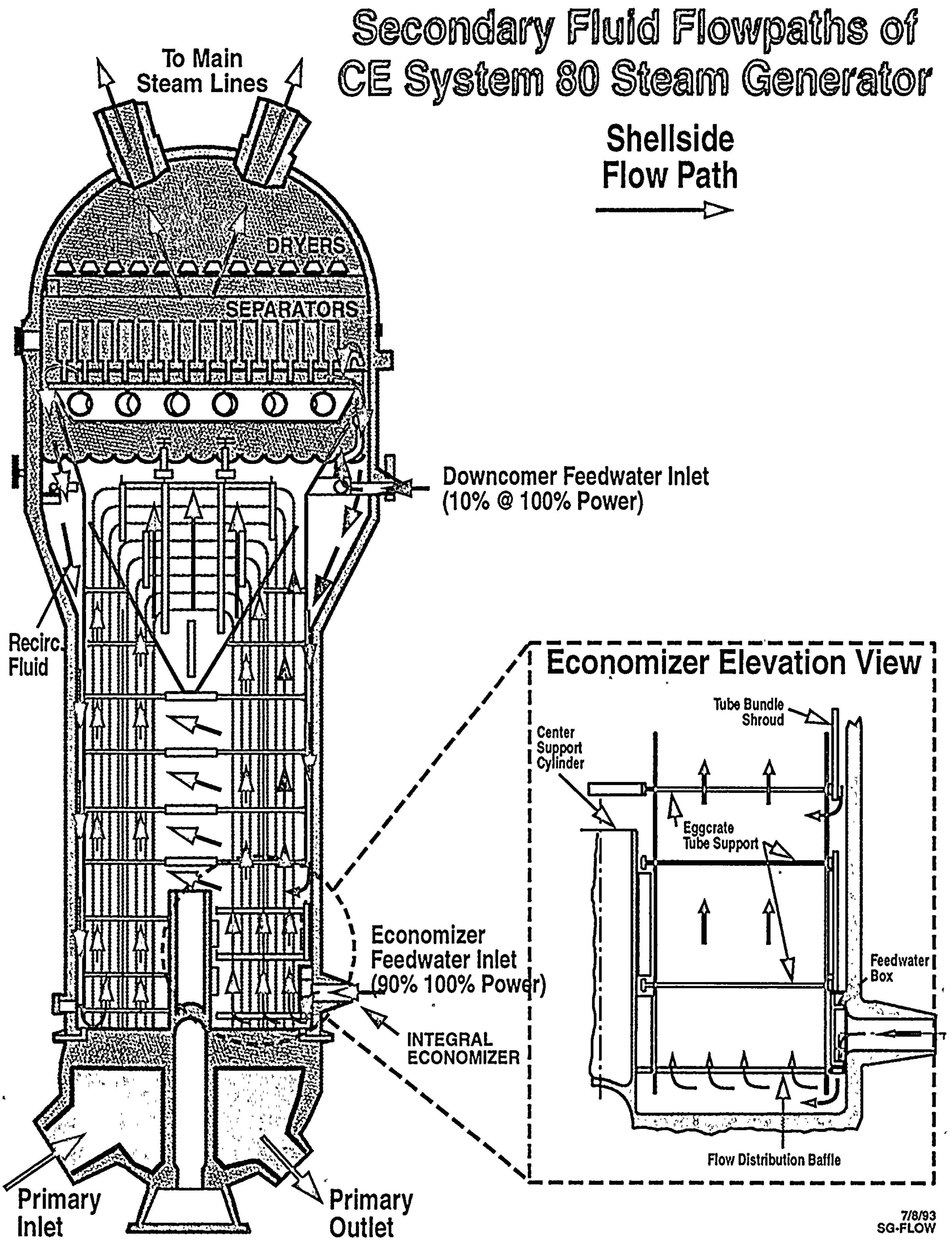

Secondary Fluid Flowpaths of
CE System 80 Steam Generator
Shellside
Flow Path
To Main
Steam Lines
DRYERS
SEPARATORS
Downcomer Feedwater Inlet
(10% @ 100% Power)
Recirc.
Fluid
Economizer Elevation View
Tube Bundle
Shroud
Center
Support
Cylinder
Eggcrate
Tube Support
Feedwater
Box
Flow Distribution Baffle
Economizer
Feedwater Inlet
(90% 100% Power)
INTEGRAL
ECONOMIZER
Primary
Inlet
Primary
Outlet
7/8/93
SG-FLOW

FIGURE II-2

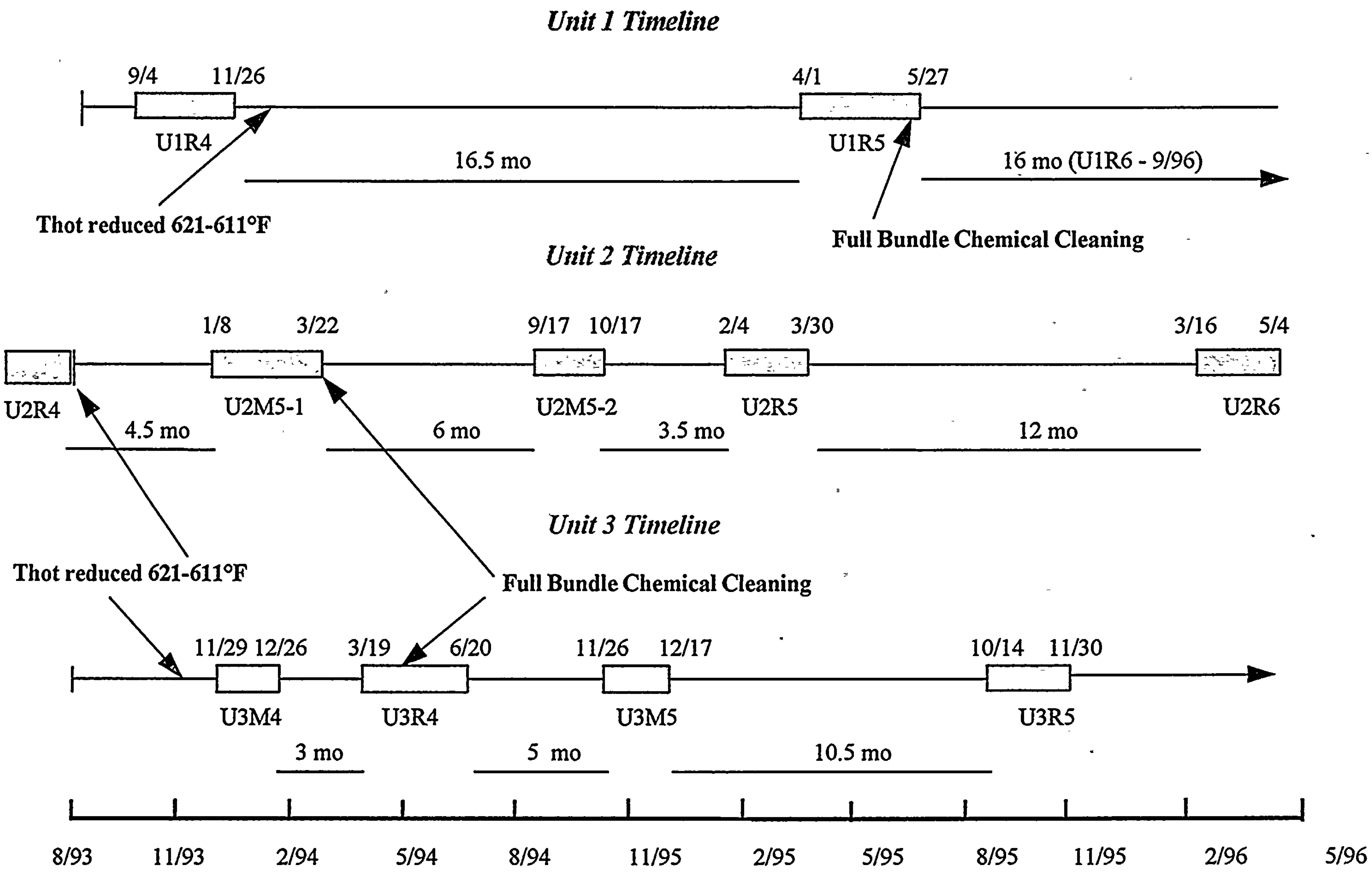
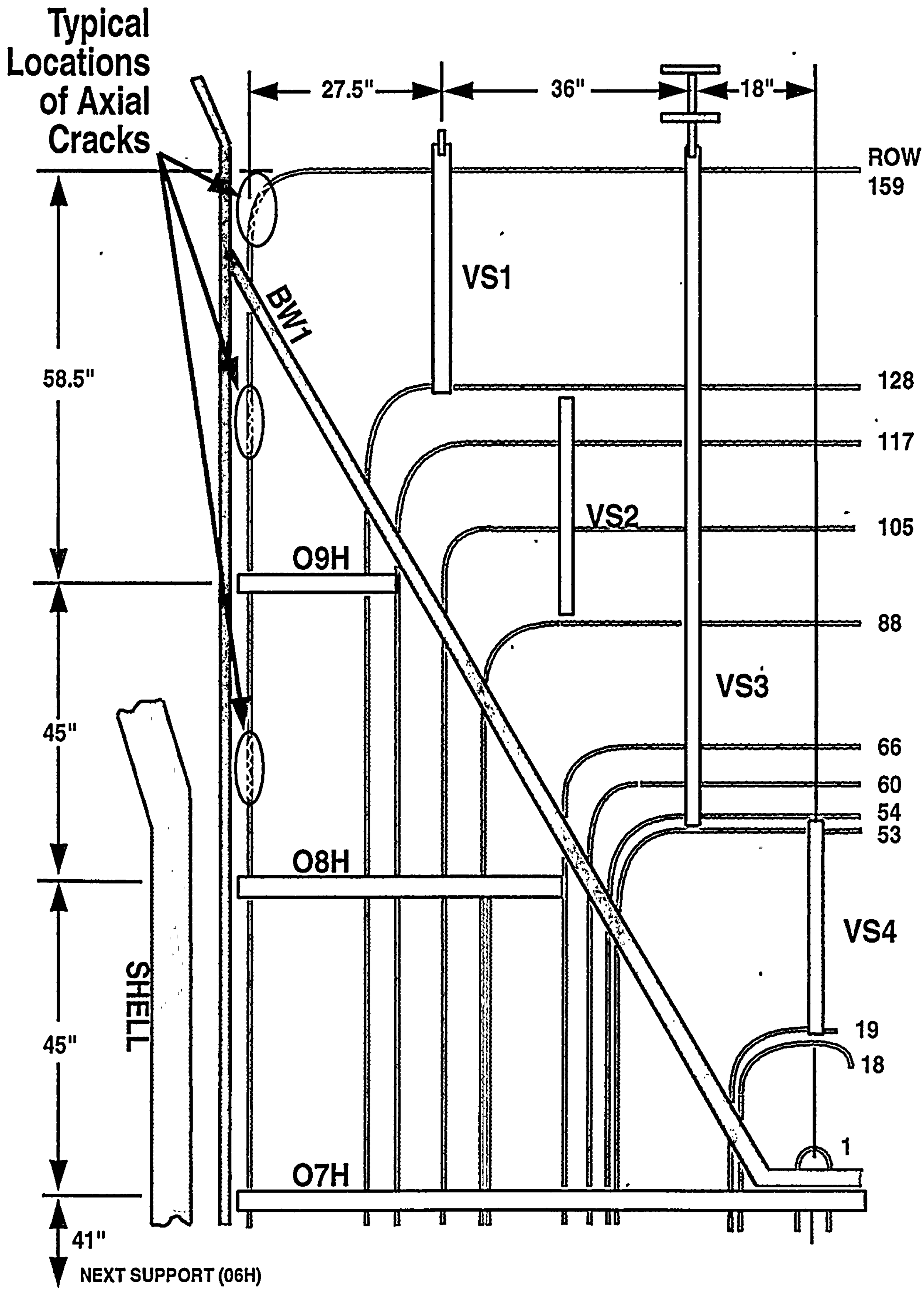

Unit 1 Timeline
9/4
11/26
4/1
5/27
U1R4
U1R5
16.5 mo
16 mo (U1R6 - 9/96)
Thot reduced 621-611°F
Full Bundle Chemical Cleaning
Unit 2 Timeline
1/8
3/22
9/17
10/17
2/4
3/30
3/16
5/4
U2R4
U2M5-1
U2M5-2
U2R5
U2R6
4.5 mo
6 mo
3.5 mo
12 mo
Unit 3 Timeline
Thot reduced 621-611°F
Full Bundle Chemical Cleaning
11/29
12/26
3/19
6/20
11/26
12/17
10/14
11/30
U3M4
U3R4
U3M5
U3R5
3 mo
5 mo
10.5 mo
8/93
11/93
2/94
5/94
8/94
11/95
2/95
5/95
8/95
11/95
2/96
5/96

Figure III-1

Location of Unit 2 Axial Cracks



SQ_14.IMA

FIGURE IV-1

03/96, ARIZONA PUBLIC SERVICE, PALO VERDE, UNIT 2

STEAM GENERATOR: 21
OUTAGE DATA SET : CURRENT
Indication Location: 01H -2.00 to 01C -2.00 AND Percent: MAI, SAI

DATE: 04/11/96
TIME: 08:40:47

STAYS ▲

PLUGGED 420 ✕ MAI 8 ◆ SAI 91 ◆

EXAM MULT SCOPE 2780 ╲

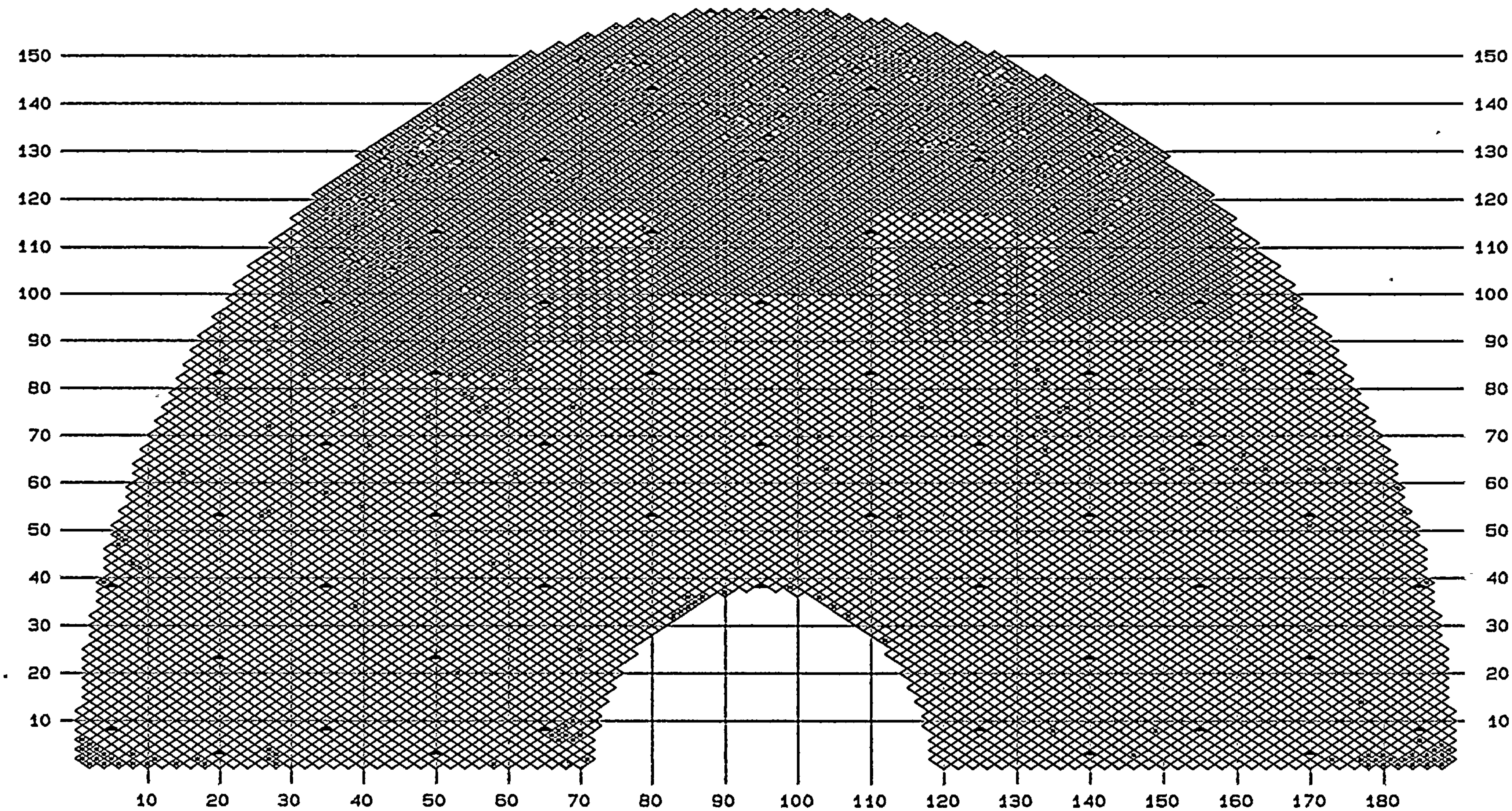


FIGURE IV-2

ROCKRIDGE TECHNOLOGIES

03/96, ARIZONA PUBLIC SERVICE, PALO VERDE, UNIT 2

STEAM GENERATOR: 22
OUTAGE DATA SET : CURRENT
Indication Location: 01H -2.00 to 01C -2.00 AND Percent: MAI, SAI

DATE: 04/11/96
TIME: 08:43:37

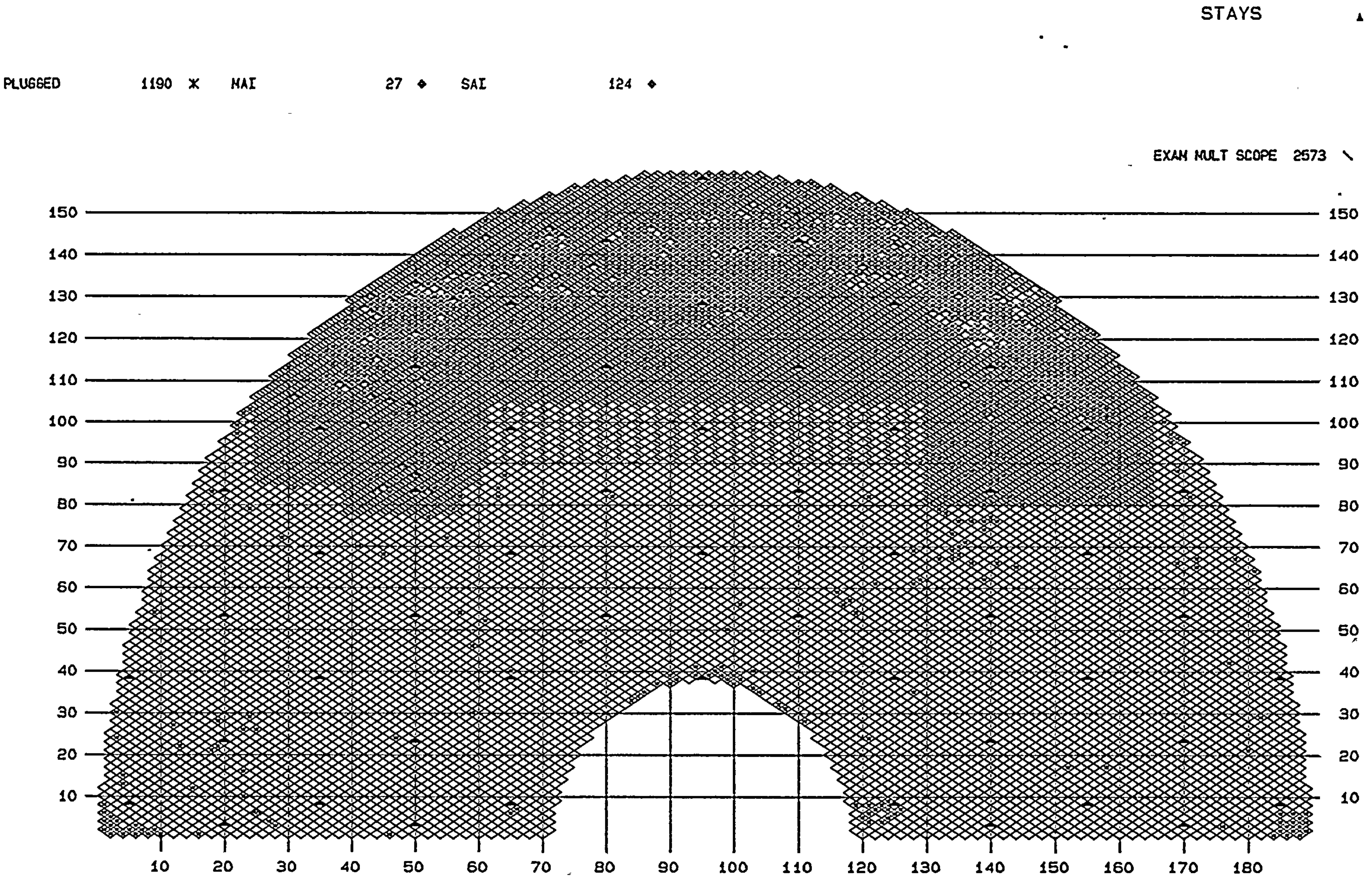


FIGURE IV-3

ROCKRIDGE TECHNOLOGIES

03/96, ARIZONA PUBLIC SERVICE, PALO VERDE, UNIT 2

STEAM GENERATOR: 21
PREVIOUS WEAR - RC PROGRAM NON-ARC

DATE: 04/11/96
TIME: 11:01:40

CRITERIA: TUBES EVALUATED/EXAMINED IN SELECTED SCOPE STATUS GROUPS

STAYS ▲

PLUGGED 420 X

EVALUATED 222 ◆ EXAMINED 0 ◆

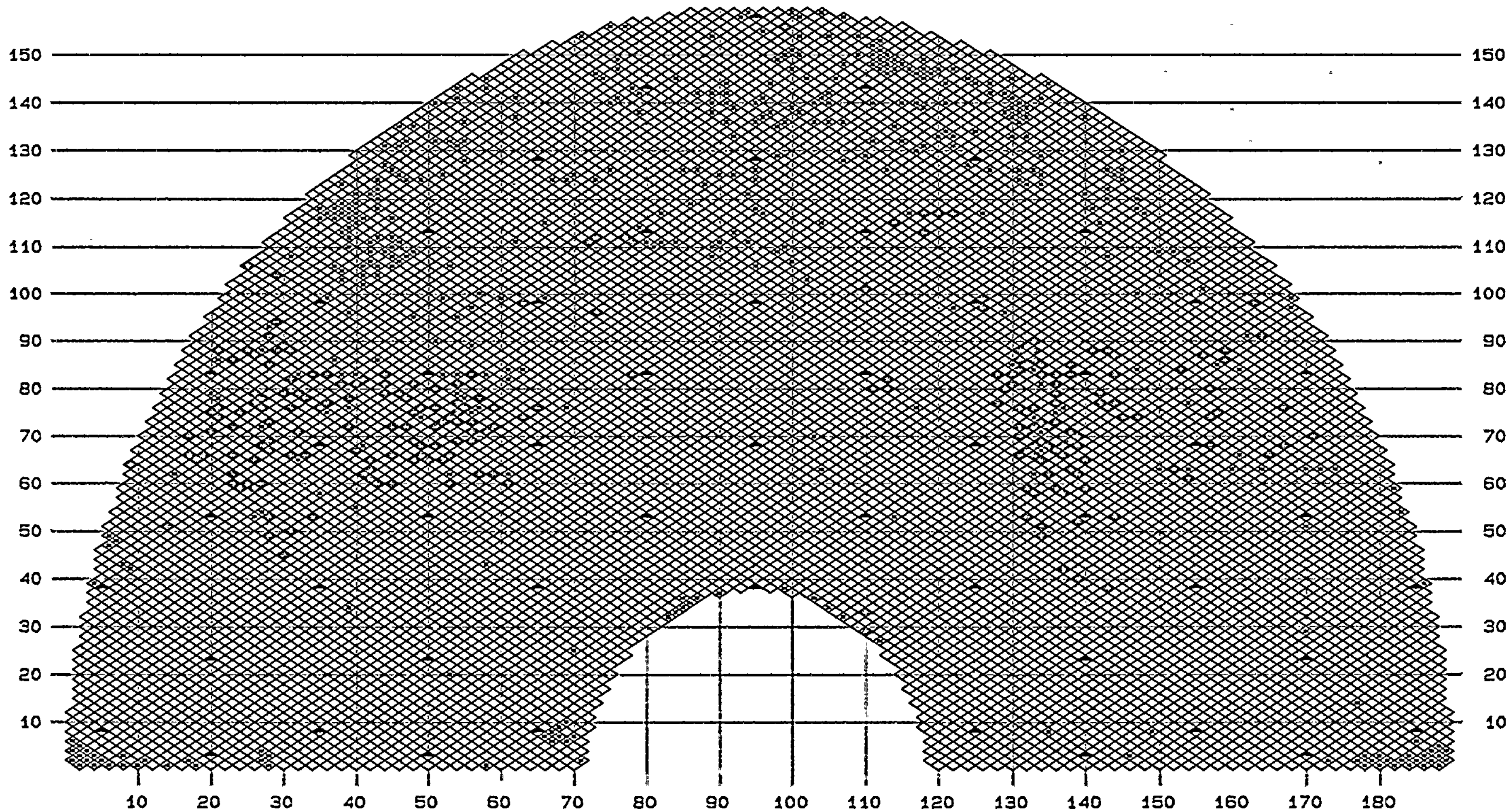


FIGURE IV-4

ROCKRIDGE TECHNOLOGIES

03/96, ARIZONA PUBLIC SERVICE, PALO VERDE, UNIT 2

STEAM GENERATOR: 22
PREVIOUS WEAR - RC PROGRAM NON-ARC

DATE: 04/11/96
TIME: 13:39:22

CRITERIA: ALL TUBES EVALUATED/EXAMINED

. STAYS ▲

PLUGGED 1190 X EVALUATED 171 ◆ EXAMINED 0 ◇

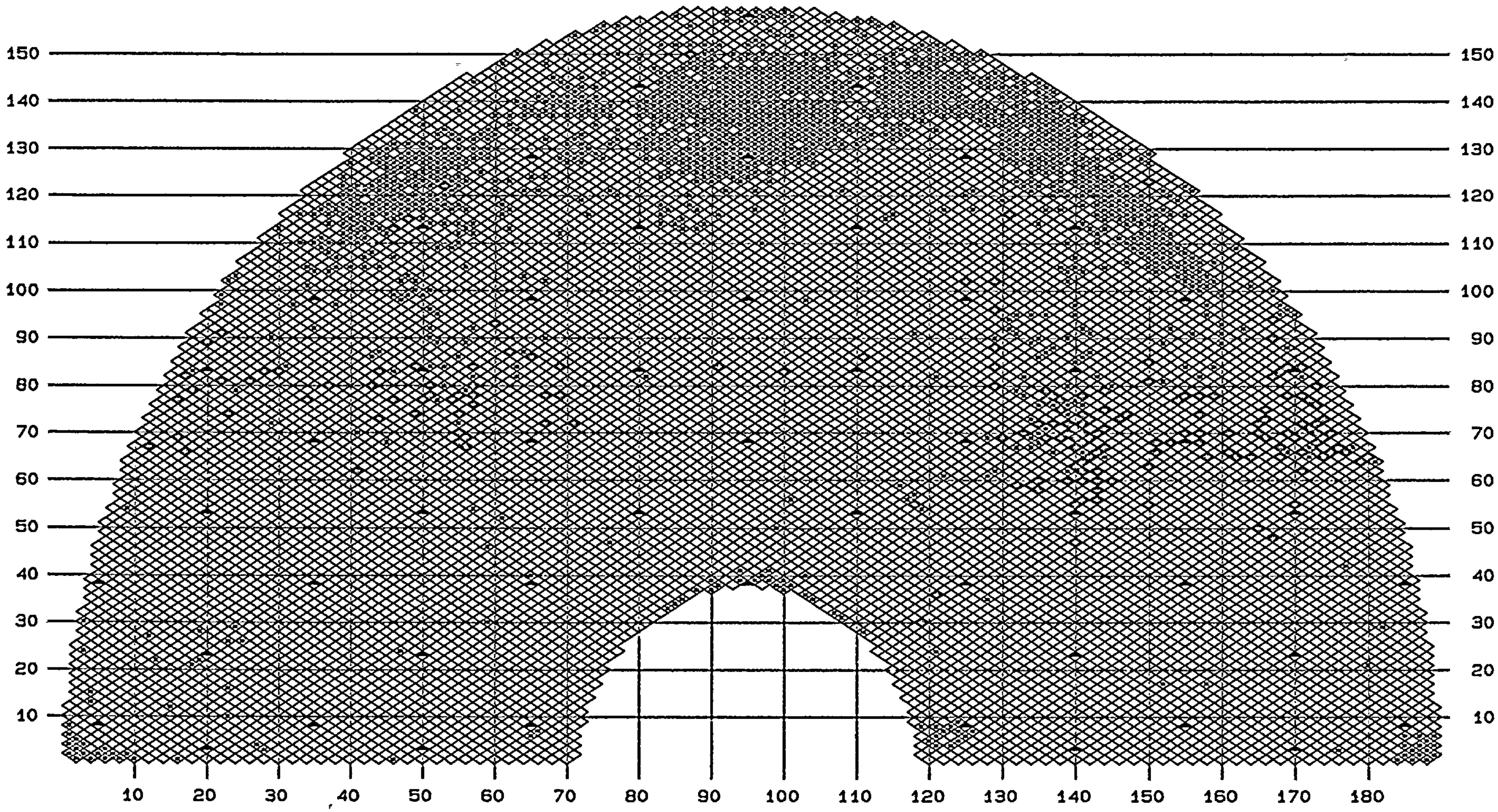


FIGURE IV-5

ROCKRIDGE TECHNOLOGIES

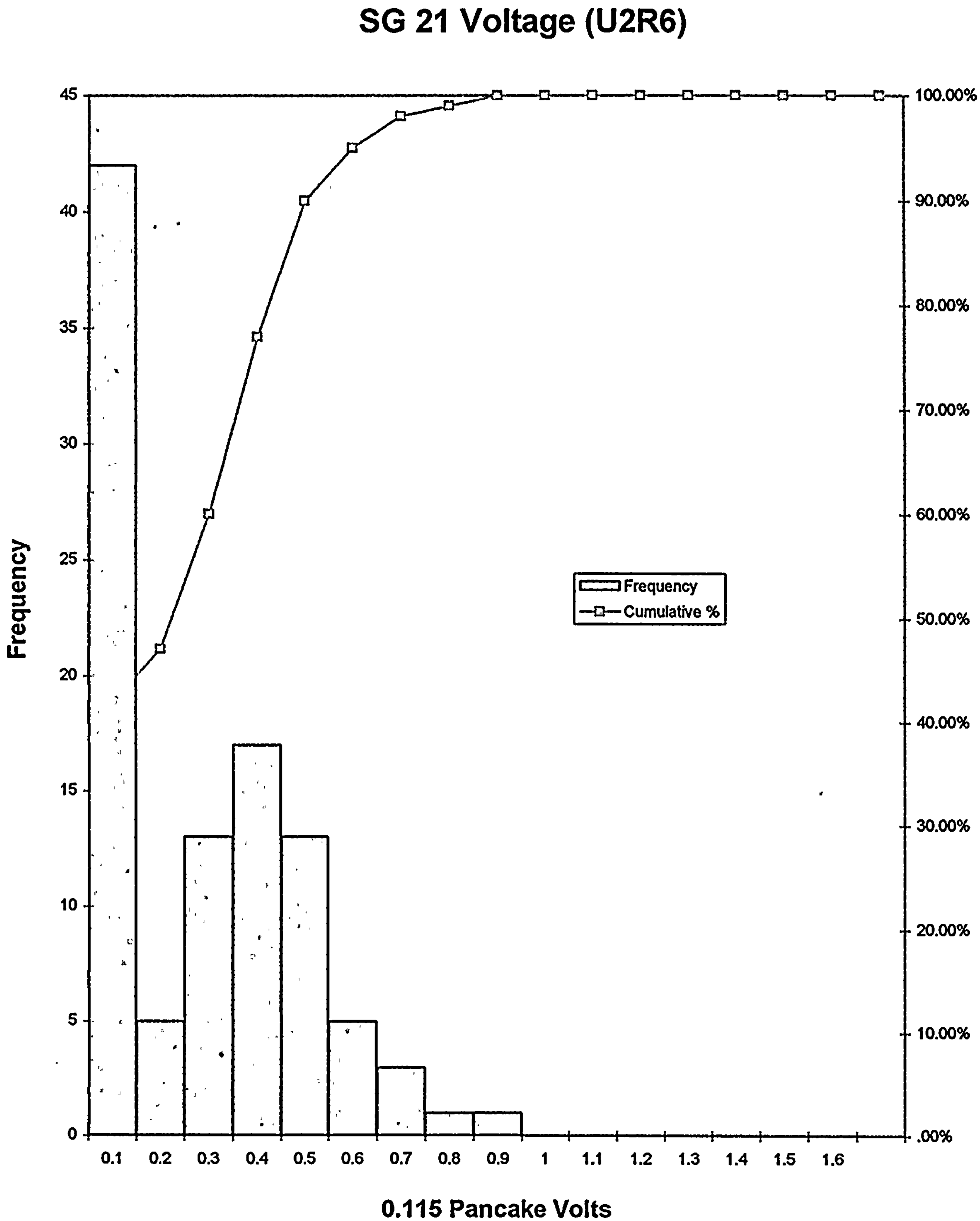

SG 21 Voltage (U2R6)
Frequency
0
5
10
15
20
25
30
35
40
45
100.00%
90.00%
80.00%
70.00%
60.00%
50.00%
40.00%
30.00%
20.00%
10.00%
.00%
Frequency
Cumulative %
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
1.1
1.2
1.3
1.4
1.5
1.6
0.115 Pancake Volts

Figure IV-6

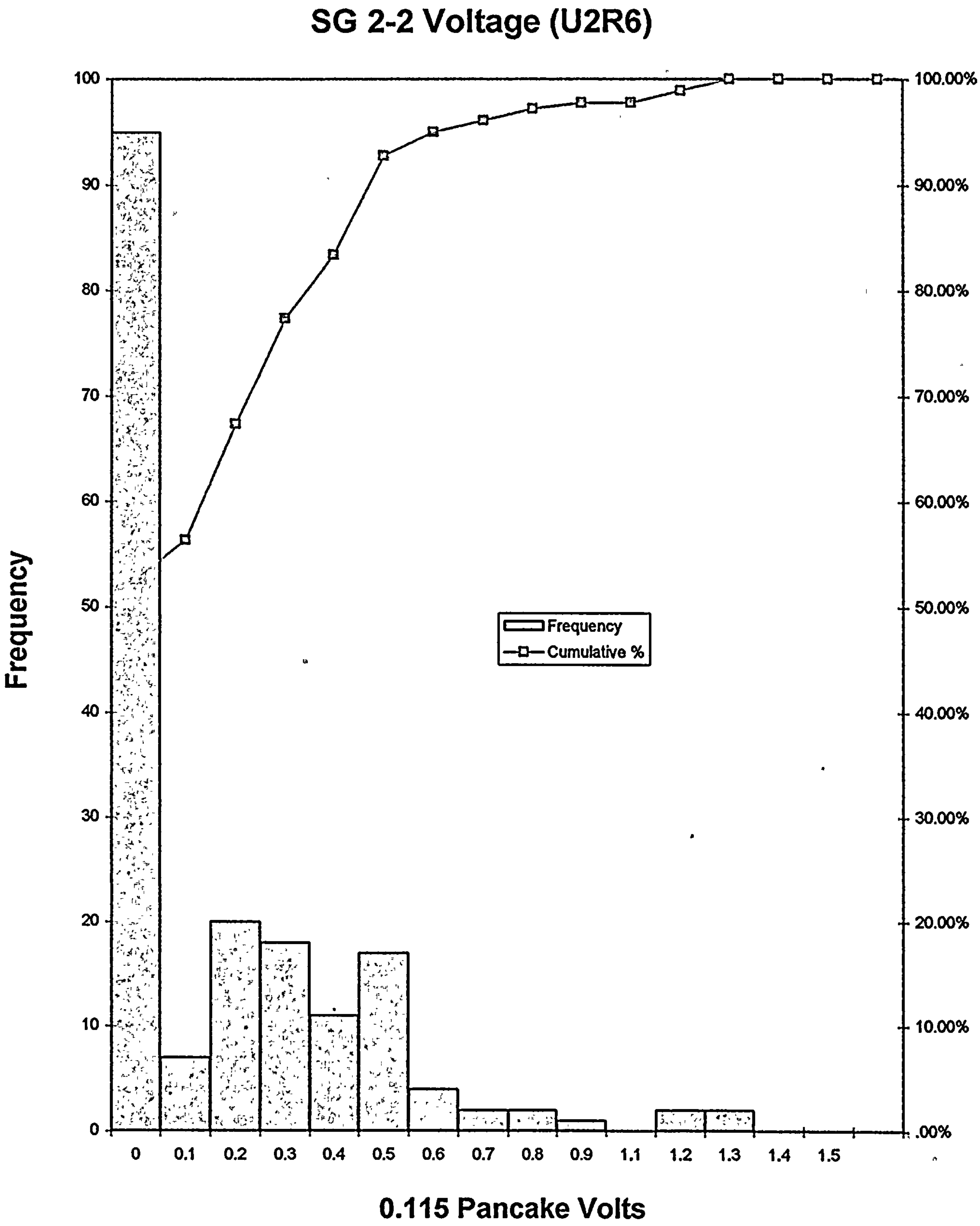
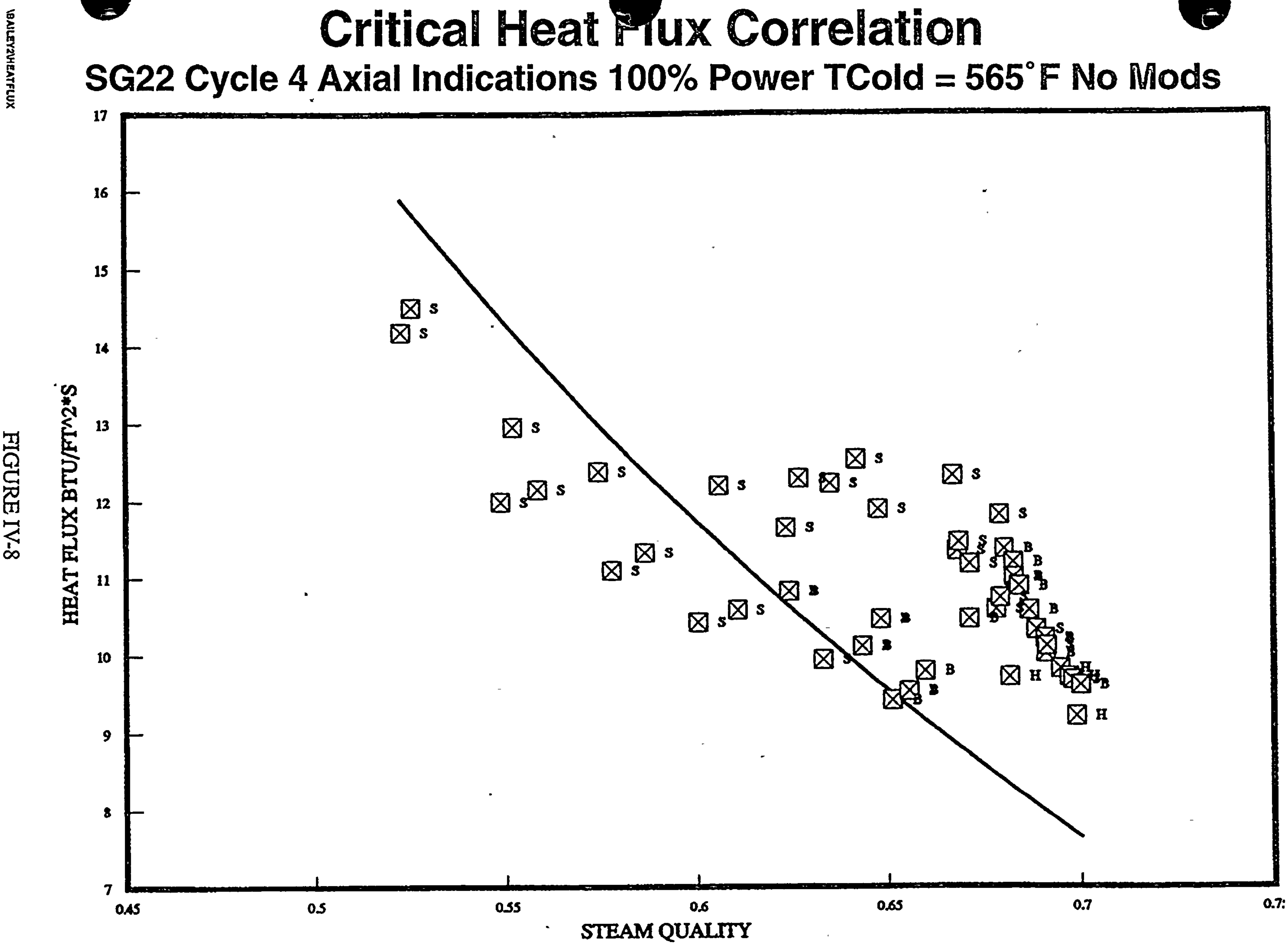

SG 2-2 Voltage (U2R6)
Frequency
Frequency
Cumulative %
0.115 Pancake Volts
100
90
80
70
60
50
40
30
20
10
0
100.00%
90.00%
80.00%
70.00%
60.00%
50.00%
40.00%
30.00%
20.00%
10.00%
.00%
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.1
1.2
1.3
1.4
1.5

Figure IV-7

Critical Heat Flux Correlation

SG22 Cycle 4 Axial Indications 100% Power TCold = 565°F No Mods



S = STRAIGHT, B = BEND, H = HORIZONTAL

FIGURE IV-8

100% POWER

Tcold = 555 deg F

As Designed PVNGS Steam Generator

CRITICAL QUALITY >= .65

CRITICAL DEPOSIT PARAMETER >= 180

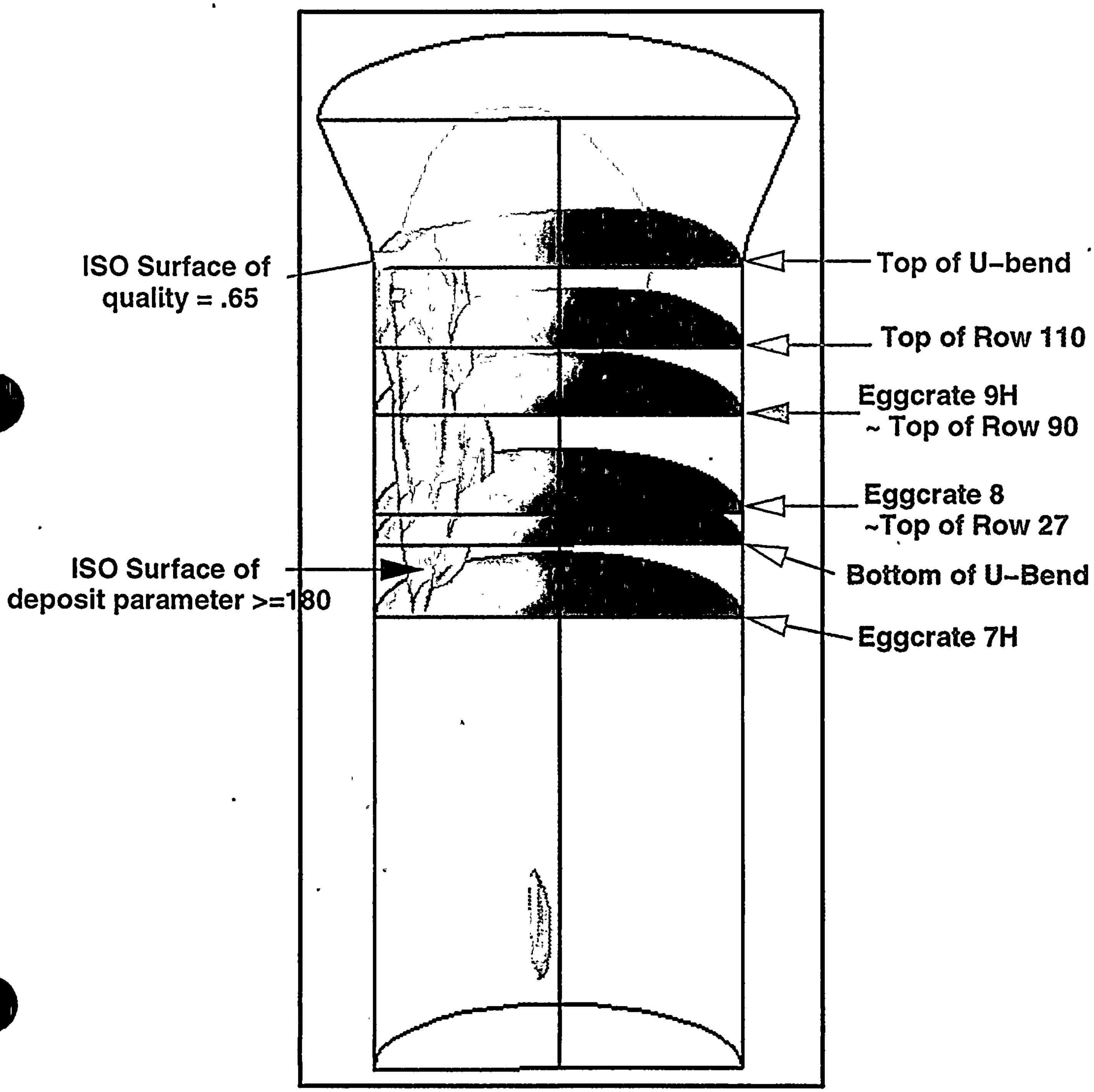


FIGURE IV-9

100% POWER

Tcold = 555 deg F

As Designed PVNGS Steam Generator

MAX CONTOUR OF CRITICAL QUALITY >= .65
occurs between 09H and top of U-bend

MAX CONTOUR OF CRITICAL DEPOSIT PARAMETER >= 180
occurs between 08H and 09H

Iso surface of deposit parameter >= 180

Iso surface of quality >= .65

overlap occurs from 09H to top of U-bend

FIGURE IV-10

Feedring Extension to Hot Leg

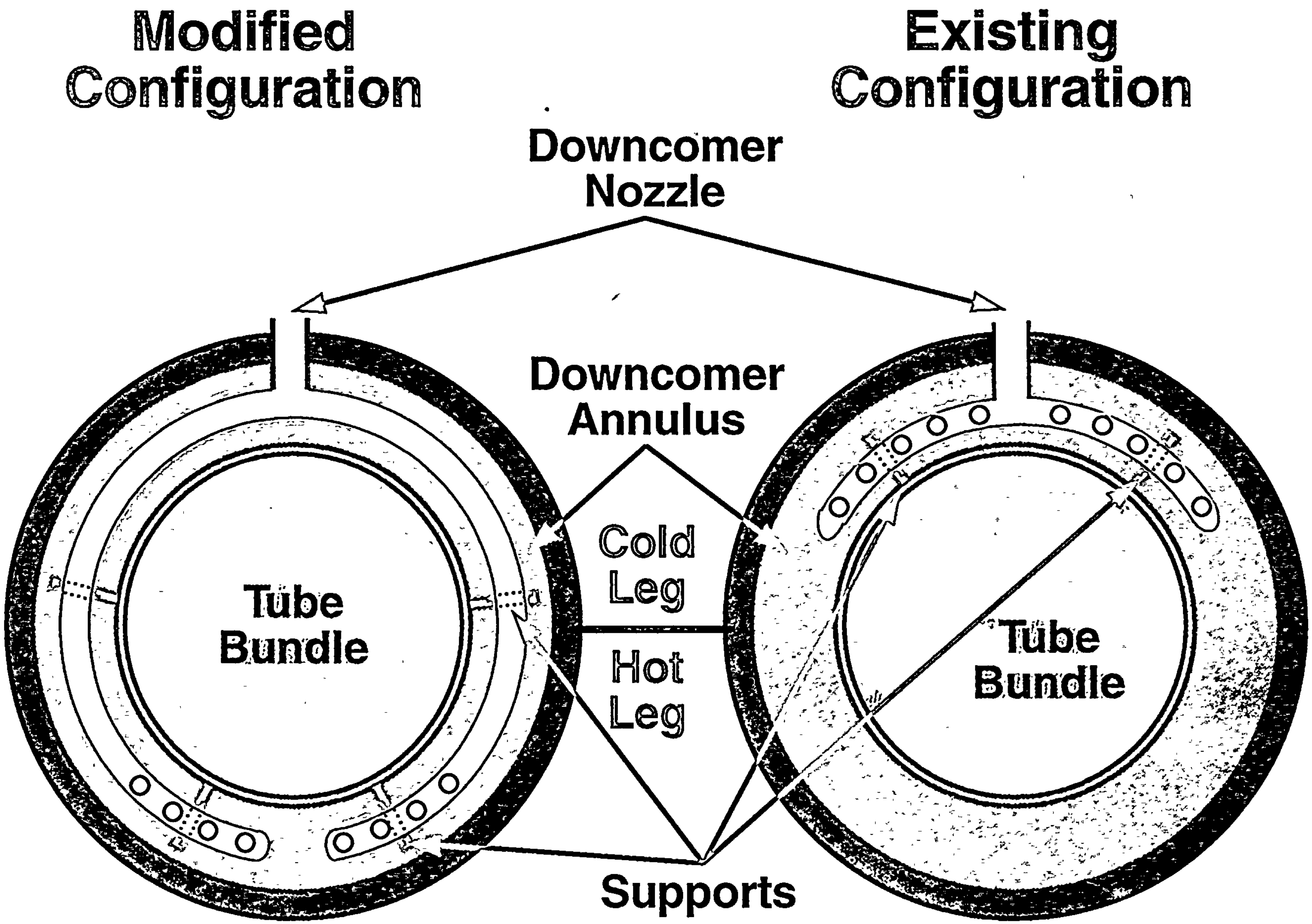
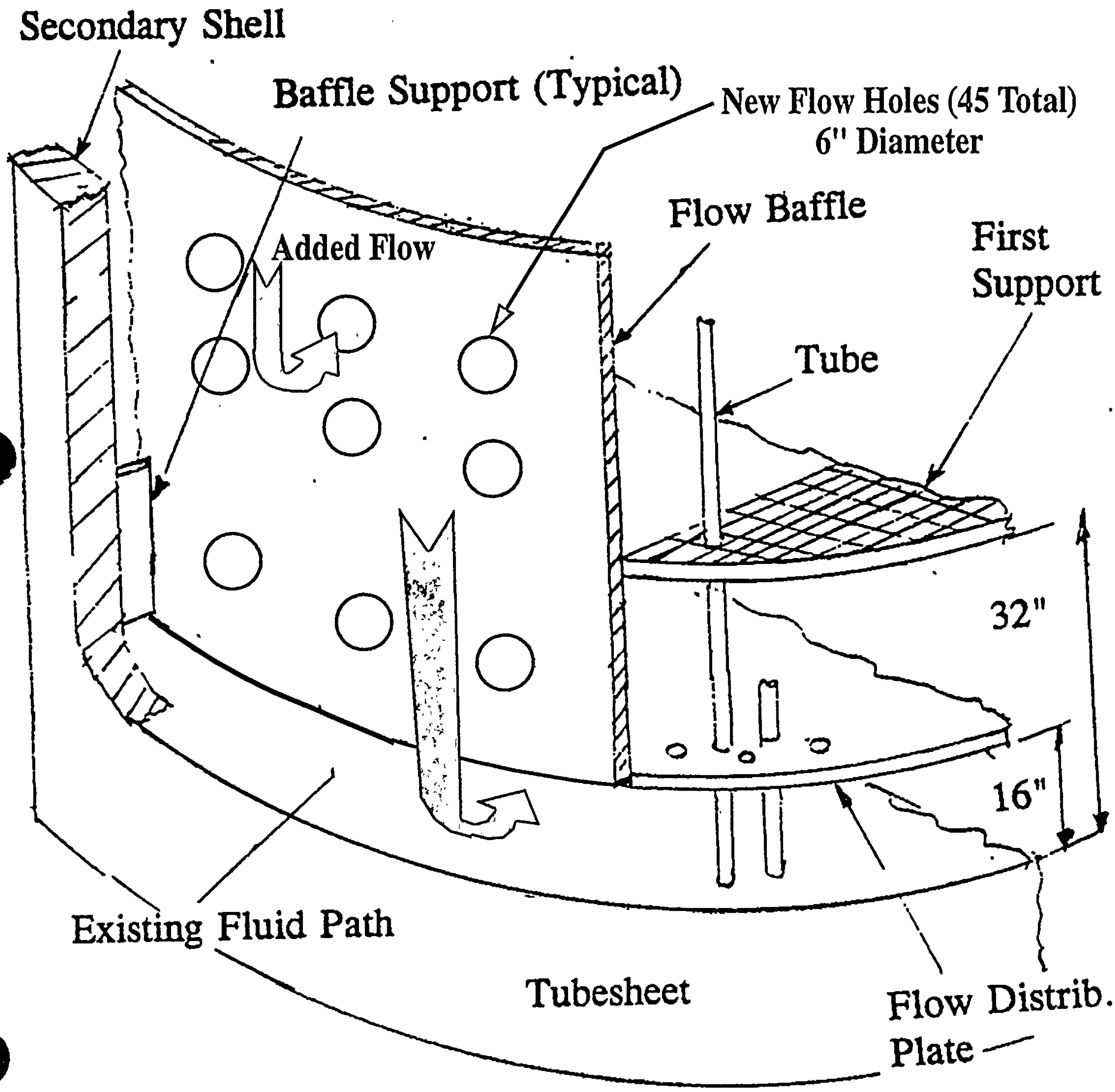


FIGURE IV-11

LEY2\FEEDRING

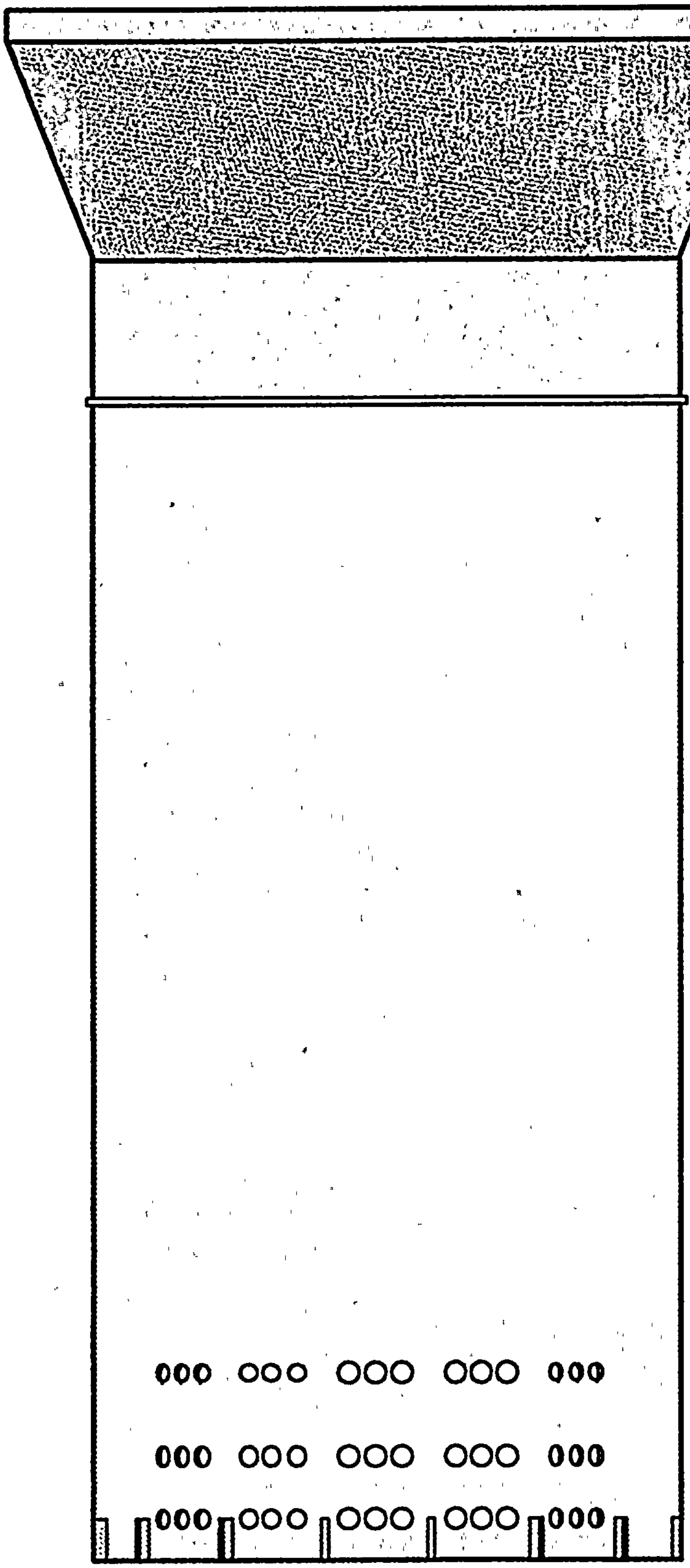
Steam Generator Secondary Hot Side Recirculating Fluid Entrance Cutaway of Modified Region



\PRATT\SG2.IMA

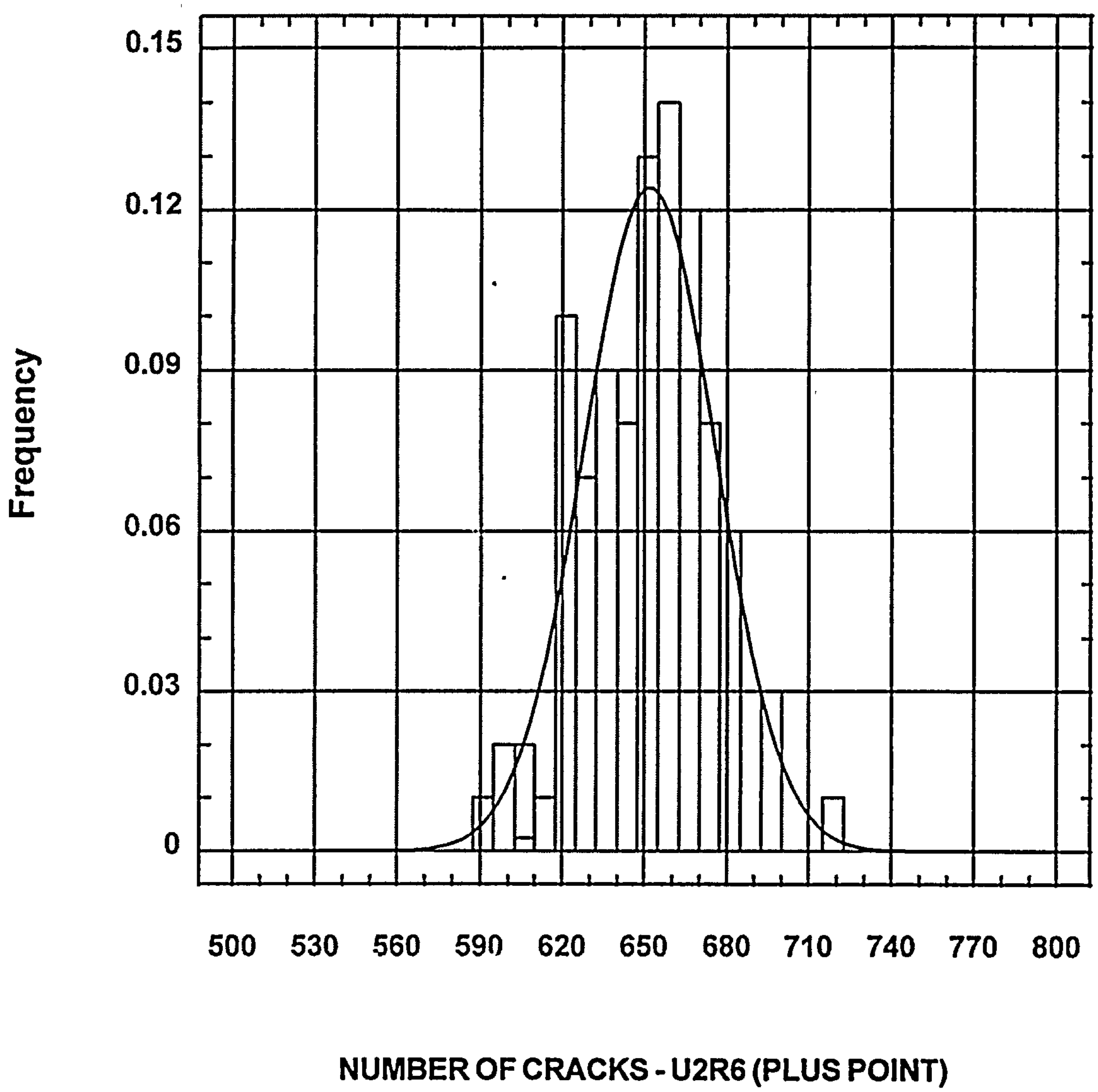
FIGURE IV-12

Palo Verde Steam Generator Shroud Modification Hole Configuration



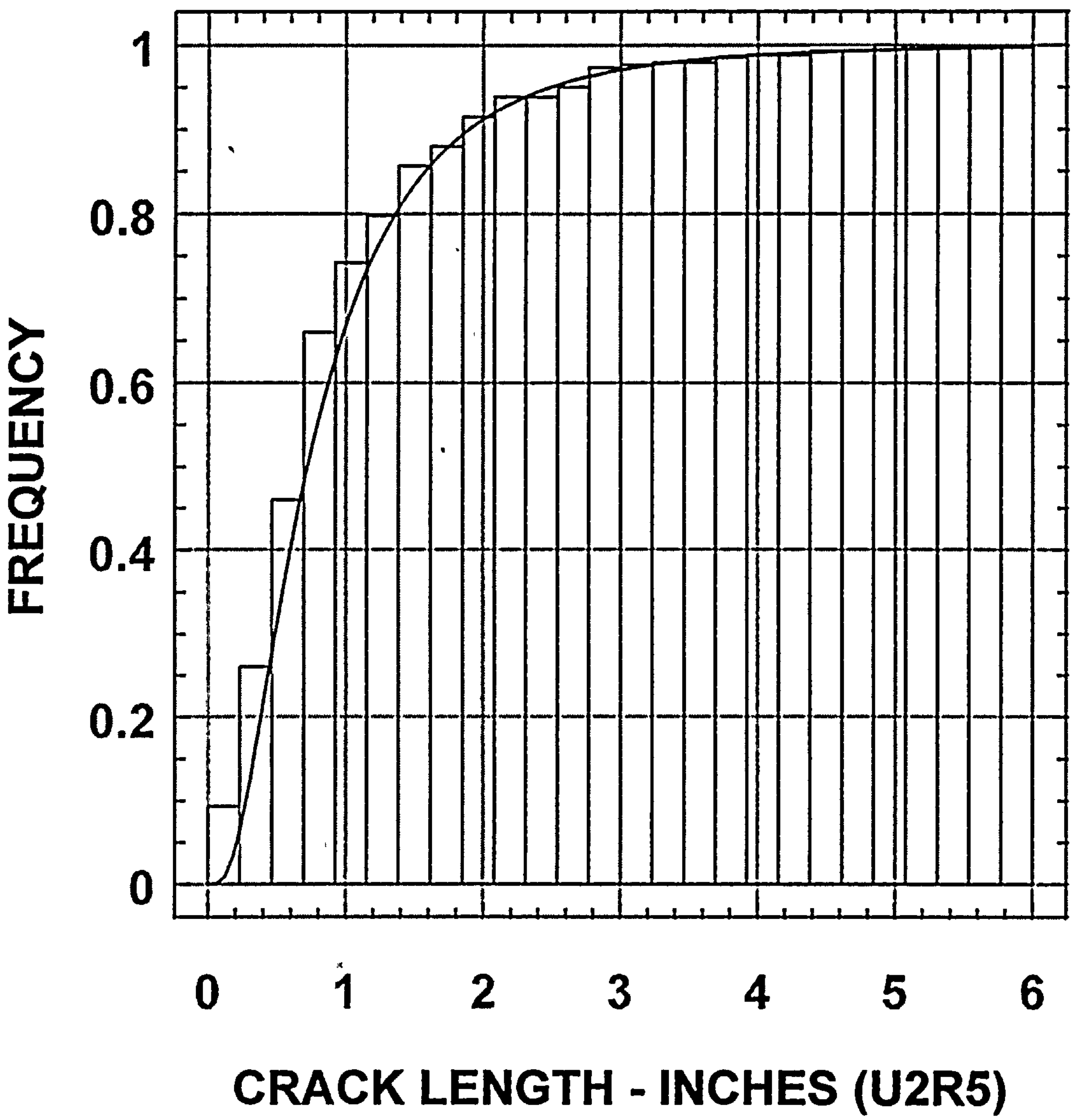
7/20/95
/BAILEY3/SHROUD-1

FIGURE IV-13



Reprint from Figure 5.7 of Reference 10

Figure V-1



DISTRIBUTION OF U2R5 CRACK LENGTHS

FIGURE V-2

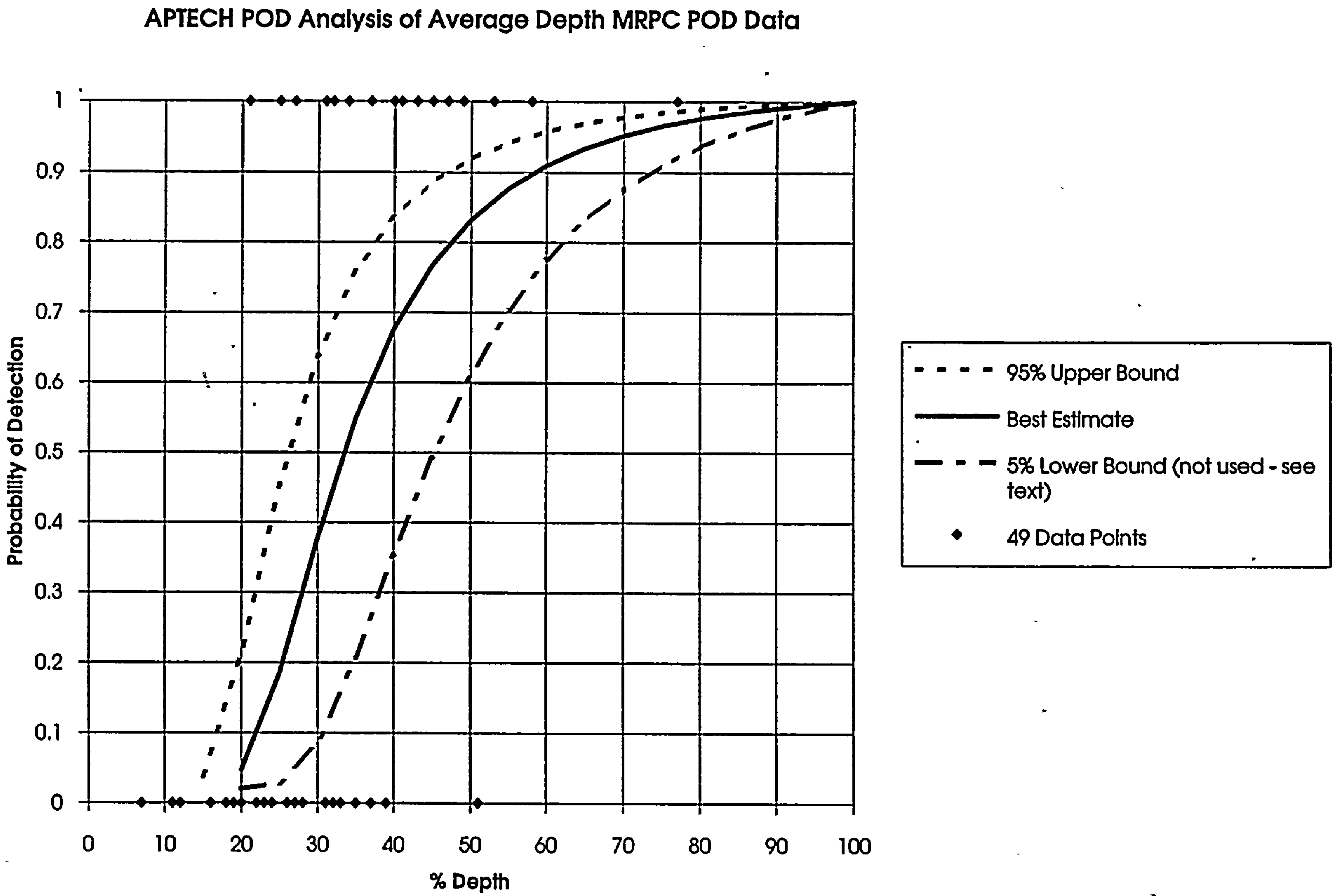

APTECH POD Analysis of Average Depth MRPC POD Data
Probability of Detection
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
10
20
30
40
50
60
70
80
90
100
% Depth
95% Upper Bound
Best Estimate
5% Lower Bound (not used - see text)
49 Data Points

FIGURE V-3

Leak Locations Input For ATHOS N-16 Calculation

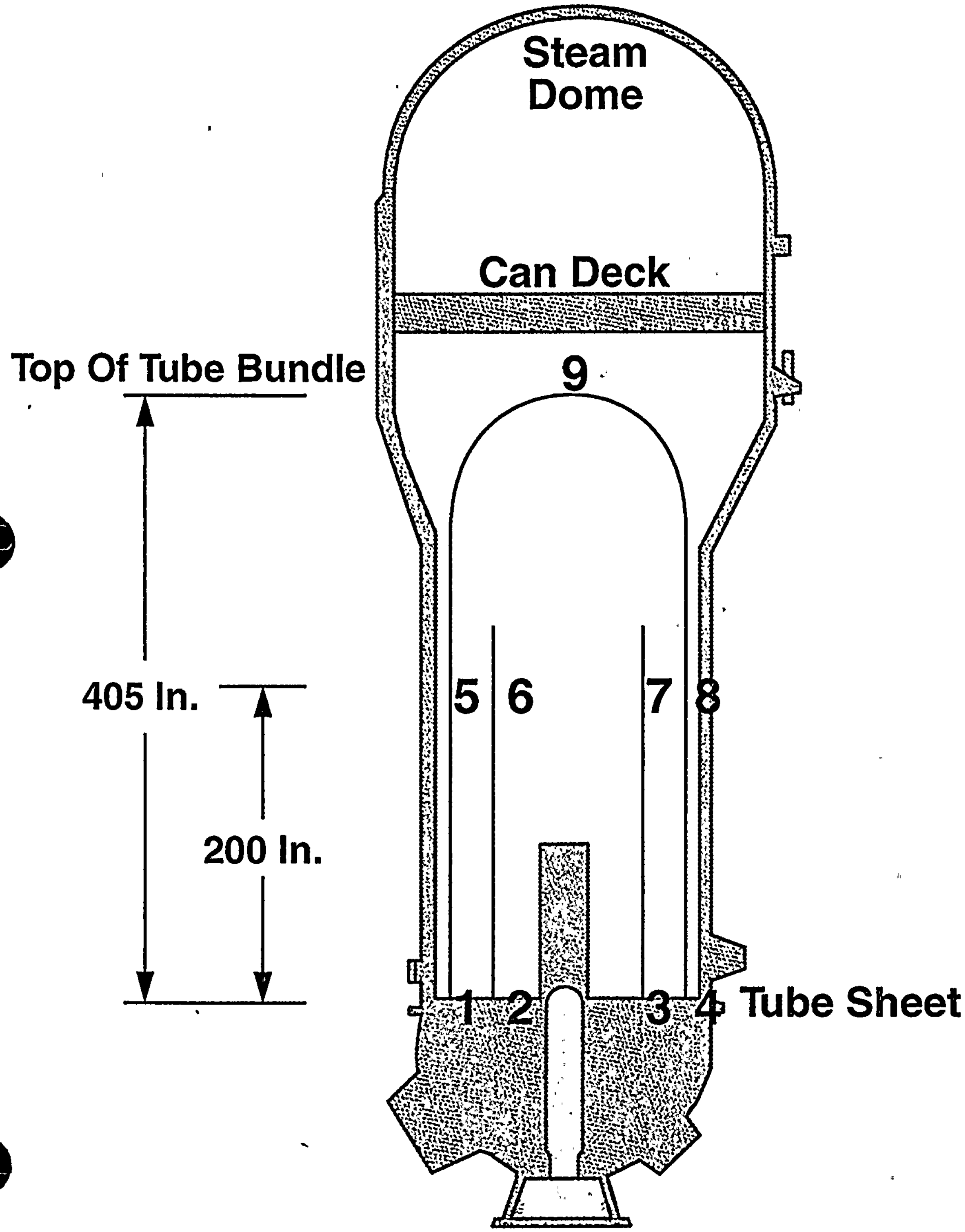


FIGURE VI -1

APPENDIX A

Palo Verde Unit - 2
Run Time Analysis
Regarding the Impact
of Upper Bundle Corrosion
Degradation During Cycle 7

APTECH Engineering Services

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AES 96072812-1-1
Revision 1
December, 1996

PALO VERDE UNIT 2 RUN TIME ANALYSIS REGARDING THE IMPACT OF UPPER BUNDLE CORROSION DEGRADATION DURING CYCLE 7

Prepared by

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J. A. Begley
S. D. Brown
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Prepared for

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TABLE OF CONTENTS

EXECUTIVE SUMMARY

Axial corrosion degradation of steam generator tubing in the upper bundle region of Palo Verde Unit 2 was evaluated. The effect of this corrosion degradation on the structural behavior and leakage integrity of tubing was determined based on pulled tube data and past and present eddy current inspection results. End of cycle conditions were projected for the next operating period using a Monte Carlo simulation model. The model simulates the physical processes of crack initiation, growth, detection of degradation by eddy current inspection and repair or removal from service of degraded tubing for multiple cycles of operation. Projections made in past applications of this model to Unit 2 have been proven to be accurate or conservative.

The projected axial corrosion degradation of tubing for the next cycle of operation was used to compute the conditional probability of tube burst under postulated steam line break conditions after 16.5 months of operation. An upper 95% confidence estimate of this probability is 3×10^{-4}. The chance of any leaking crack during Cycle 7 is correspondingly remote at 4×10^{-5}. Two eddy current inspections using the Plus Point probe have markedly reduced the severity of any undetected beginning of cycle degradation. Remedial measures have substantially reduced degradation growth rates. These two effects dramatically reduce the probability of axial corrosion degradation having any significant impact on the structural or leakage behavior of tubing in the next cycle of operation even for run times much longer than the current estimate of 16.5 months.

SECTION 1
INTRODUCTION

The most recent of five eddy current inspections of steam generator tubing at Palo Verde Unit 2 shows an ongoing process of corrosion degradation in the upper bundle region. As confirmed by previously pulled tubes from this unit,[1,2,3] the current indications are attributed to outer diameter stress corrosion cracking and intergranular attack. The cumulative number of tubes involved in both generators exceeds one thousand[4,5]. Until the most recent outage (Spring 1996), the progression rates in terms of number of defects involved and depth growth rates have been high by comparison with the less affected Unit 3 steam generators.[6,7]

The most recent inspection showed a significant reduction, however, in both progression rates. The number of defects observed was less than 50% of that expected and the observed defect growth rates were attenuated to levels comparable to those observed in the two most recent Unit 3 inspections.

A probabilistic run time model was employed to make projections of end of cycle conditions regarding the numbers of cracks and their lengths and depths. This forms the input for structural and leak rate evaluations. The model is based on the physical processes of crack initiation and growth. The probability of detecting cracks during eddy current inspections is explicitly treated. Hence the possibility of undetected cracks remaining in service for several cycles is considered. The undetected crack population forms the beginning of cycle condition. The Monte Carlo simulation model runs cycle by cycle and is benchmarked by comparing projected end of cycle numbers

of indications versus actual observations. This model has been used with excellent success for both Units 2 and 3.[5,6,7]

Some improvements have been added to the current version of the simulation program. Growth rates for a given location can vary from one cycle to the next. If a crack escapes detection in one cycle, its growth rate may be different in the next cycle. Additionally, growth rates can be sampled directly from past observations without using a fitted analytic distribution. These refinements provide better projections and a better match with an occasional observation of a very deep crack.

The following sections describe the general approach of structural evaluation, the simulation model and the analysis input parameters. The simulation results are then presented. Some attention is given to the distribution of the largest crack depths and lowest burst pressure predictions in a given simulation of cycle 7 operation. These extreme value distributions effectively illustrate the risk of finding a low burst pressure or the chance of wall penetration and thus leakage.

SECTION 2

STRUCTURAL AND LEAKAGE CONSIDERATIONS

The focus of probabilistic structural integrity and leakage evaluations presented in subsequent sections is on a postulated main steam line break accident. This is consistent with a new draft Regulatory Guide[8] in support of steam generator rule making. The appropriate limiting accident case pressure differential is 2400 psid for Palo Verde Unit 2. Leakage and bursting are evaluated at this loading severity. The conditional probability of burst at postulated accident conditions should not exceed 0.05 when all tube degradation mechanisms are considered. The leak rate at a postulated accident event at end of cycle should not exceed the total charging pump capacity of the primary coolant system, provided that radiological dose consequences do not exceed General Design Criteria 19 and 10 CFR Part 100 guidelines at this leak rate.

In addition, probability of tube burst under the more limiting Regulatory Guide 1.121 structural requirements are considered. The loading imposed is three times the normal operating pressure across the tube wall. In the case of Palo Verde Unit 2, the differential pressure is 3800 psid. Leakage computations were not performed at this loading.

The following paragraphs describe the basis of structural integrity analysis for axial degradation. Characterizations of crack morphologies are presented. Leak rate calculations are summarized. These concepts form the framework of the probabilistic models used to project leak rate and burst behavior.

From the perspective of leak rate and burst strength calculations, all axial corrosion degradation is idealized as single planar cracks. This is conservative in the sense that the strengthening and leak limiting effects of ligaments between crack segments in crack arrays are neglected. The entire spectrum of IGA/ODSCC degradation morphologies can be represented as arrays of axial and circumferential cracks of varying diffuseness. Leak and burst calculations are based on the single planar crack extreme of the spectrum of possible morphologies. Only the ligaments in the depth direction are considered to provide strengthening and leak limiting effects.

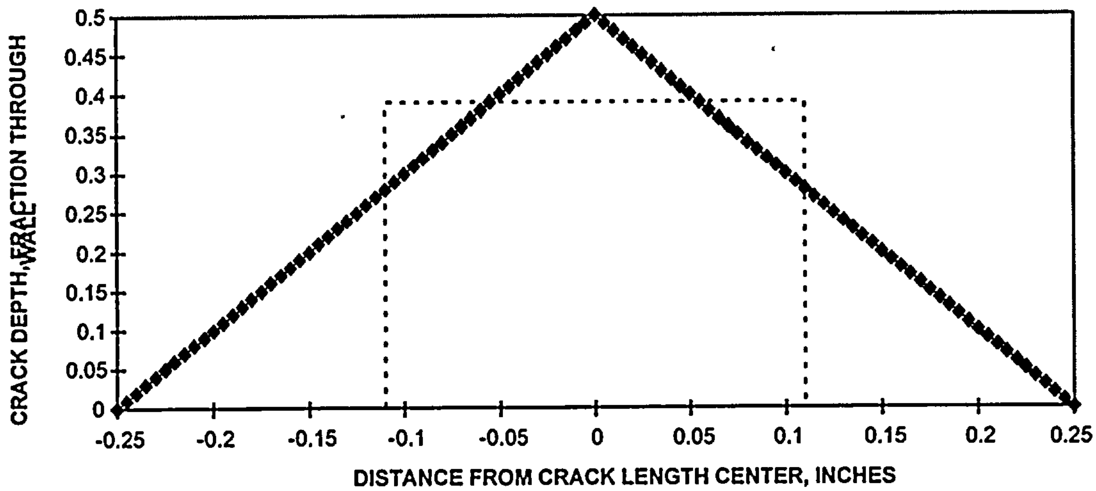
As in past APTECH models of axial cracking,[5,7,9,10] the Structural Minimum Method of computing the burst pressure associated with an arbitrary crack depth profile is utilized. Figure 2.1 illustrates this approach using a triangular crack shape. A selected portion of the total crack profile is first considered. The average crack depth over this selected length is calculated. This length and associated average depth is used as input to the Framatome axial partial through wall burst equation.[11] Burst pressures for successively larger portions of the total crack profile are computed. There exists some section of the total profile with a minimum burst pressure. The length of this critical section of the total crack is termed the structurally significant length and the average depth over the structurally significant length is termed the structurally significant depth. Figure 2.1 illustrates the minimum burst pressure calculation which identifies the critical section of the triangular crack profile. The dotted rectangle shows the structurally significant length and the associated structurally significant depth.

Pulled tube burst data from Palo Verde Unit 2 first validated the Structural Minimum Method and the use of the Framatome equation.[5] See Figure 2.2. Almost all calculated burst pressures are conservative with respect to measured burst pressures. This is mostly due to neglecting the

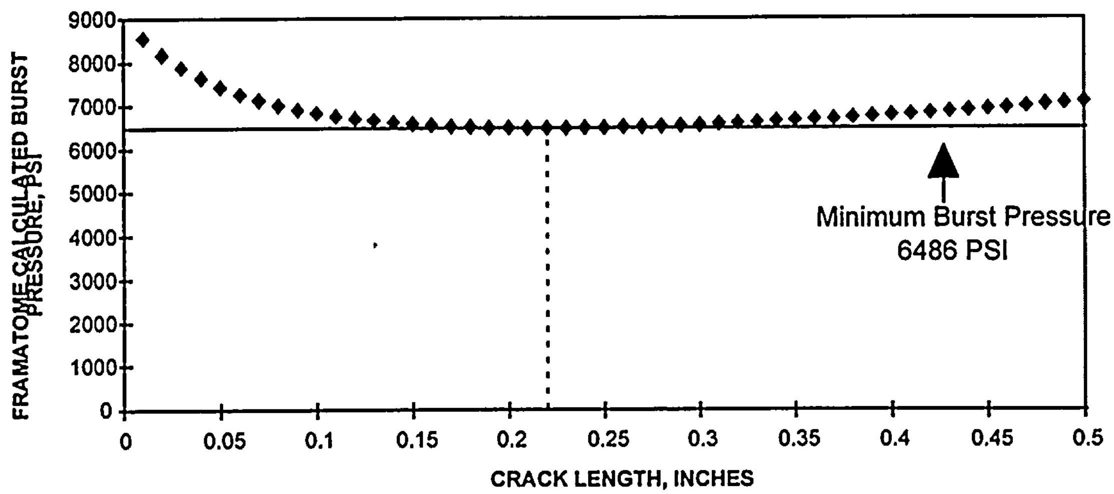
strengthening effects of ligaments between axial crack segments and only considering ligaments in the depth direction. Burst tests of laboratory specimens with EDM machined slots support this contention. Figure 2.3 illustrates the machined slots shapes. Figure 2.4 shows that use of the Framatome equation with the Structural Minimum Method provides excellent predictions of actual burst behavior for long, deep cracks. The predicted behavior of very short cracks is overly conservative. A long crack is shown to be on the order of 1.5 inches. It is important to use the Framatome equation and the Structural Minimum Method to characterize the behavior of long cracks which penetrate the wall.

Figure 2.5 illustrates the relationship between maximum crack depth and structurally significant depth for pulled tubes from Palo Verde Unit 2. The ratio of these depths together with the Structural Minimum Method define a crack shape. In probabilistic analyses, crack shapes were obtained by sampling the pulled tube data of Palo Verde Unit 2.

Figure 2.6 shows that distributions of crack lengths needed for probabilistic analyses of axial degradation can be obtained from RPC eddy current results. Data from Palo Verde Unit 2, where destructive examinations verified structurally significant crack lengths, shows that the RPC indicated lengths are usually longer, and sometimes much longer, than the structurally significant lengths. Hence, use of the distribution of RPC crack lengths in probabilistic analyses of axial degradation is conservative. The RPC crack length distribution is more adverse than the distribution of structurally significant lengths.



(A) TRIANGULAR CRACK PROFILES



(B) CALCULATED BURST PRESSURE VERSUS LENGTH OF CENTRAL CRACK SECTION

Figure 2.1 ILLUSTRATION OF THE STRUCTURAL MINIMUM METHOD

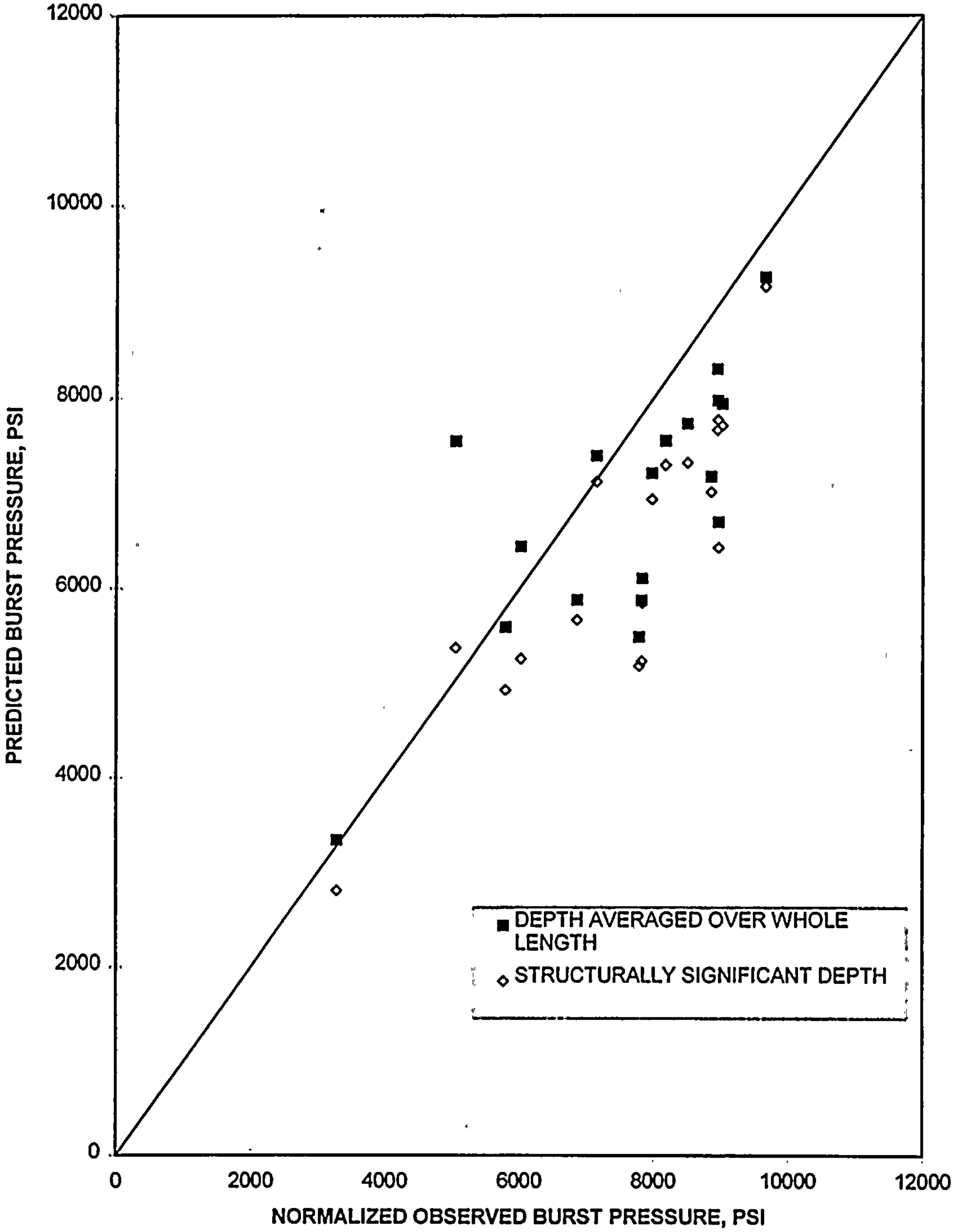


Figure 2.2 PREDICTED BURST PRESSURE VERSUS NORMALIZED OBSERVED BURST PRESSURE, PVNGS UNIT 2 PULLED TUBE DATA

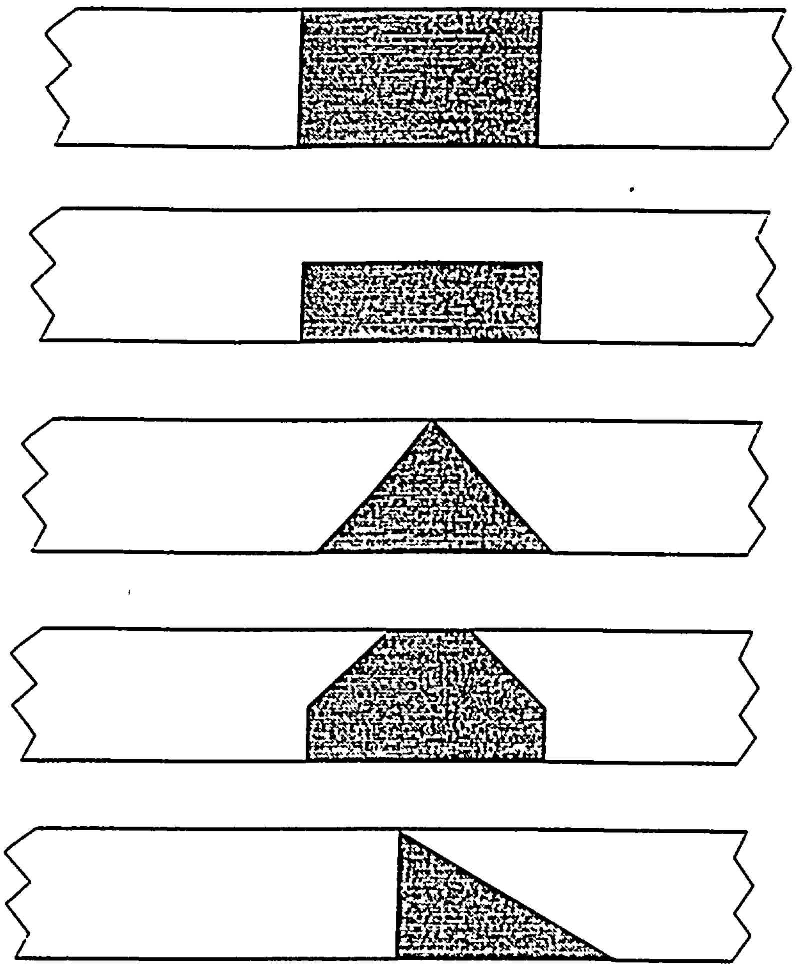


Figure 2.3 CRACK SHAPES INCLUDED IN TEST PROGRAM

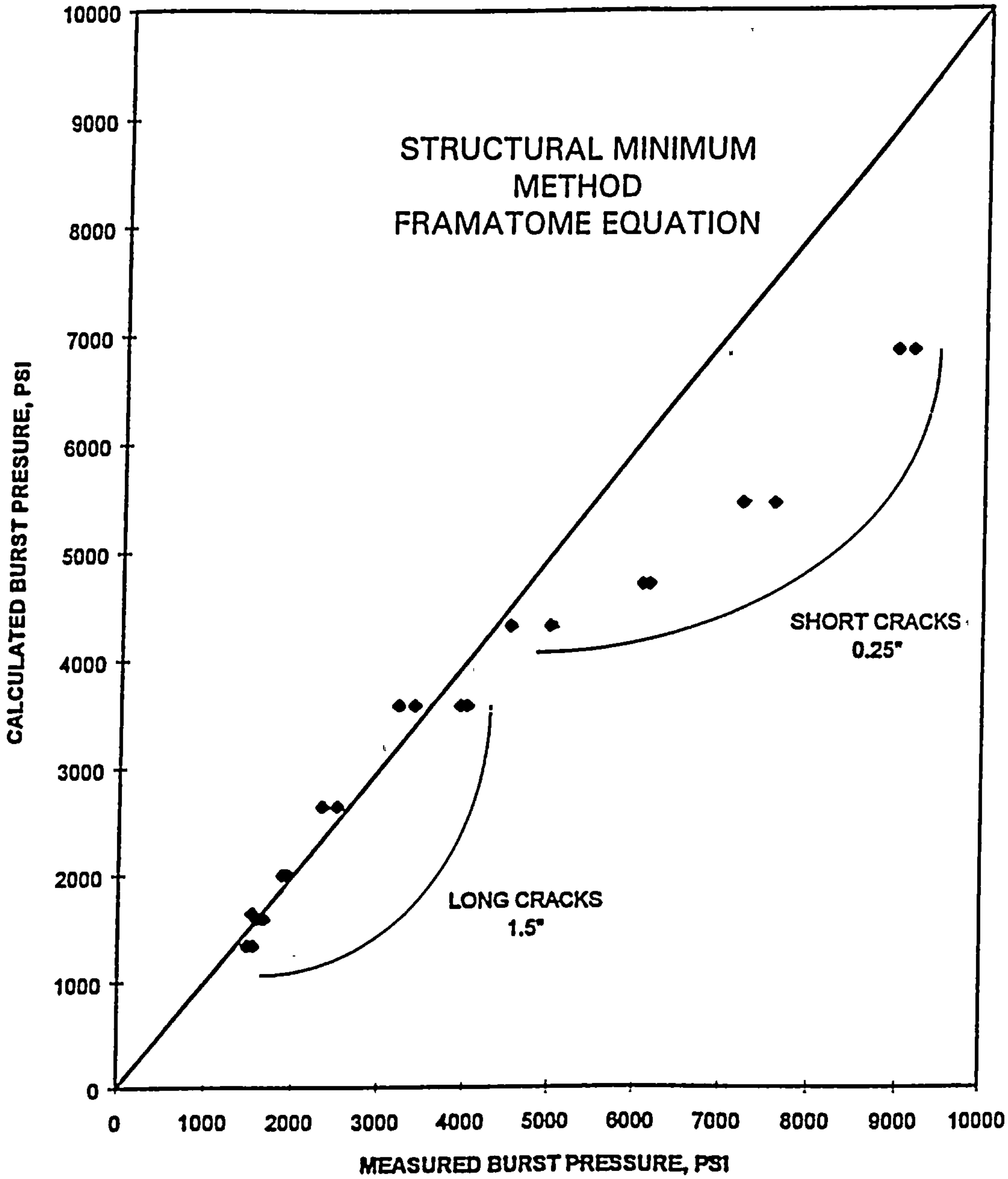


Figure 2.4 RESULTS OF EDM SLOT MACHINED SAMPLE BURST TEST PROGRAM

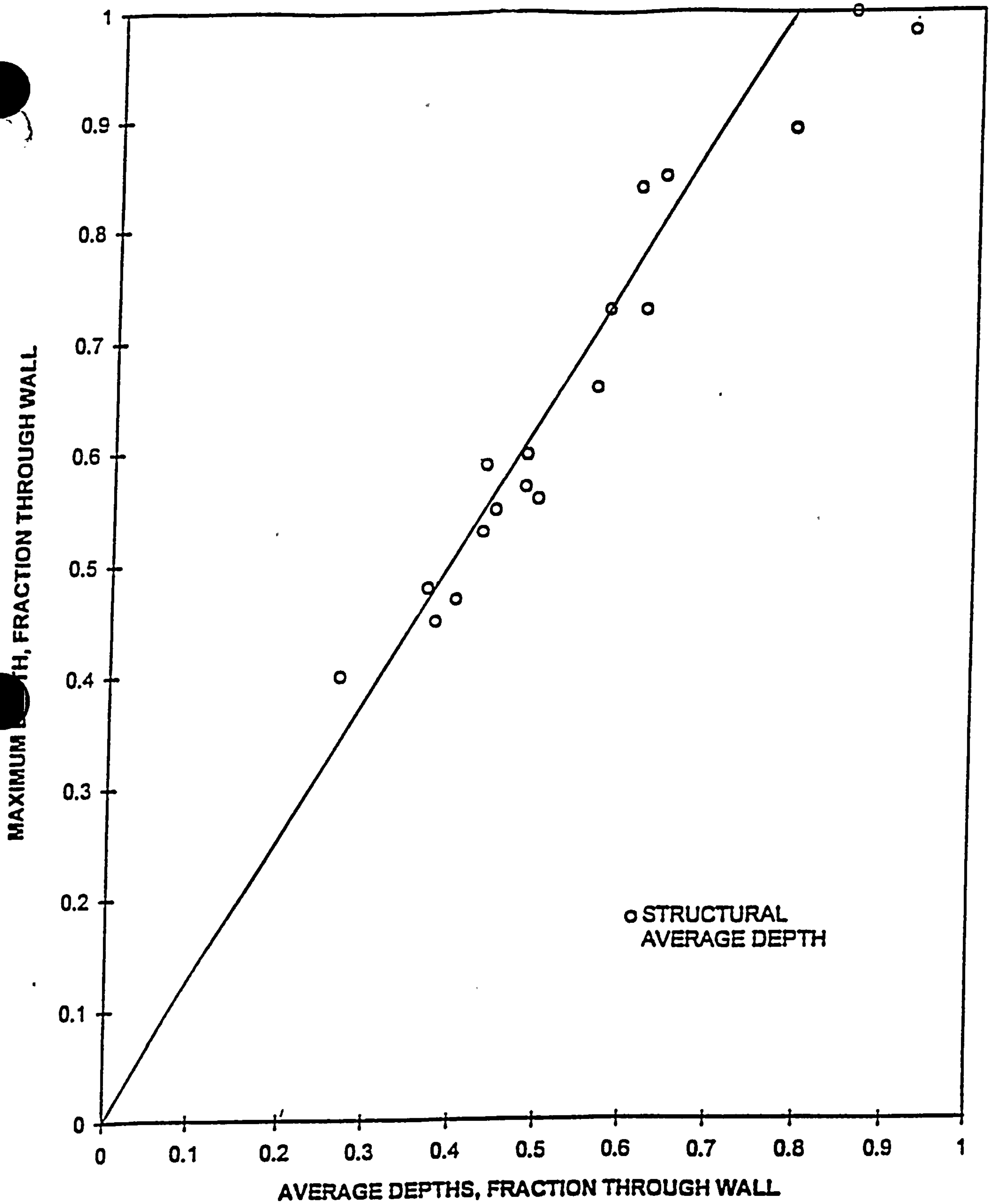


Figure 2.5 MAXIMUM CRACK DEPTH VERSUS STRUCTURAL CRACK DEPTH, PVNGS UNIT 2 PULLED TUBE DATA

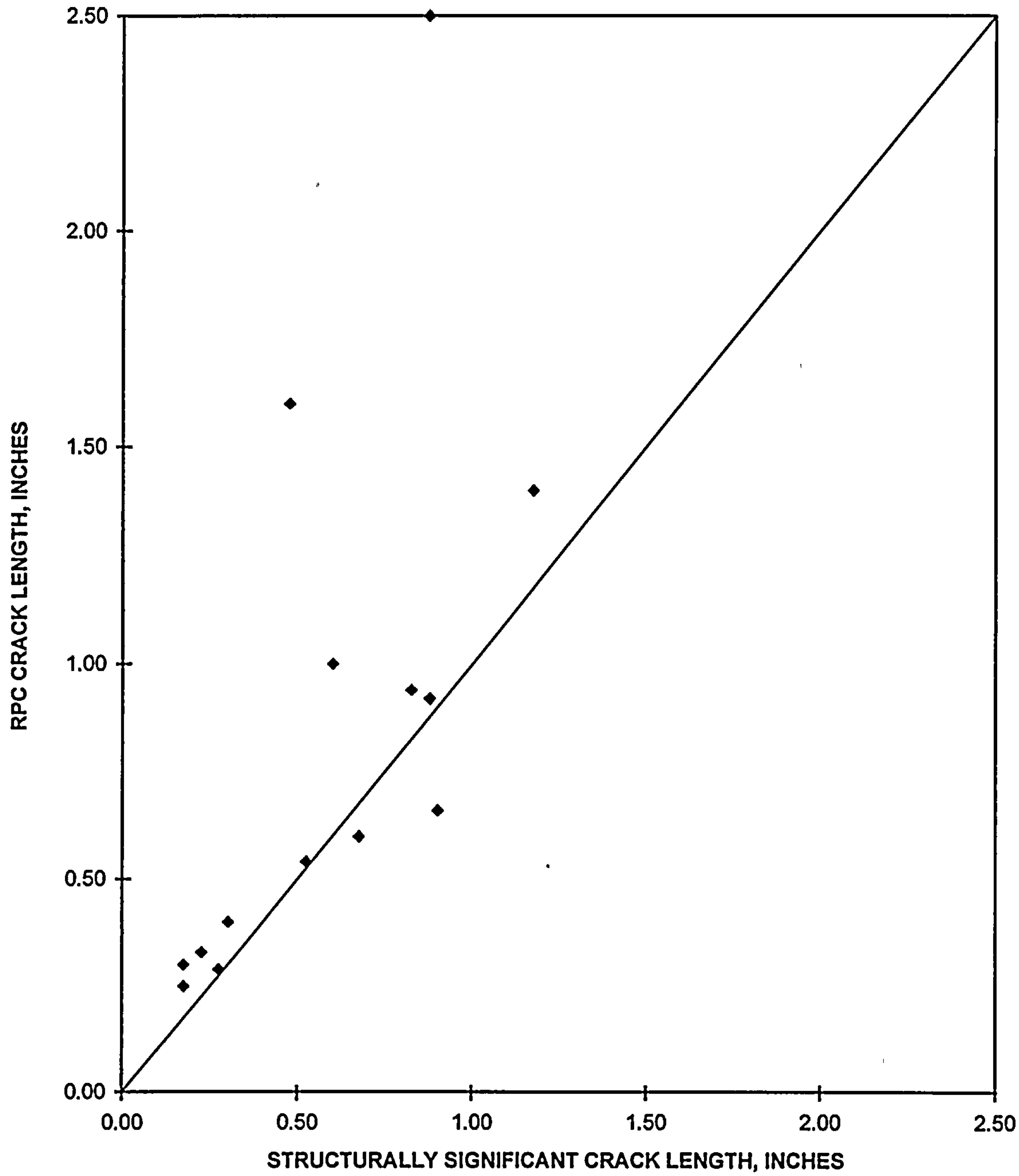


Figure 2.6 RPC CRACK LENGTH VERSUS STRUCTURALLLY SIGNIFICANT CRACK LENGTH, PVNGS UNIT 2

SECTION 3
PROBABILISTIC MODEL FOR STRUCTURAL INTEGRITY

The probabilistic model for the evaluation of run time at Palo Verde Unit 2 is described in this section. The model evaluates the probability of burst under selected differential pressure loadings for up to five sequential inspections. Essential features of the model include the ability to include the effects of variations in operating temperature, crack propagation rates, and detection hardware/procedure changes on a cycle-by-cycle basis. This permits a continuous benchmarking over a long period of plant operation and can result in a more thorough understanding of the degradation process with reference to operational parameters.

The probabilistic model for axial cracking with repair on detection is a modified version of the APTECH model used for the Palo Verde analyses of IGA/ODSCC at freespan locations in Units 2 and 3.[5,6,7] The probabilistic model consists of a Monte Carlo simulation of the processes of crack initiation, growth, eddy current inspection and repair/removal of degraded tubes. The current APTECH model simulates the behavior of up to 10,000 individual cracks for five operating periods. Larger populations are accommodated by dividing the steam generator into several risk groups, each of which can be individually simulated.

The actual simulation process is shown in Figure 3.1, which describes the execution of one Monte Carlo trial. The first step in the process is that of defect tagging in which the attributes that define the nature of the defect for the entire period of analysis, are determined. These defect attributes include:

- Initiation time
- Growth rates for each operating period

- Tube material properties (flow stress)
- Defect characteristic length
- Defect Form Factor

Each of these attributes is obtained by sampling from an appropriate probability distribution function. An important distinction between the present and previous models used for Palo Verde analysis is the incorporation of an additional probabilistic defect attribute in the first step in the Monte Carlo process. This additional attribute is the form factor which relates the crack maximum depth to the average structural depth. As with other attributes, such as defect characteristic length and material properties, the form factor remains with the simulated defect throughout the simulation trial. The effect of a randomized form factor on the simulation outcome is to increase the number of through wall defects for large defect populations. The derivation of the form factor probability distribution function and implications regarding the range of defect shapes are discussed in Section 5 and Appendices B and C.

The second step in the simulation process is to compute the size of the defect at each inspection time. The result of this step is a matrix of crack structural depths (D_{ij}) corresponding to time values (T_j) where the index j denotes the j^{th} inspection and the index i denotes the i^{th} defect. At this point in the simulation the characteristics necessary to determine the structural integrity of each simulated defect have been obtained.

The third step in the simulation process is to numerically inspect each defect to determine at which inspection the defect is detected and removed from service. This is accomplished by random selection using the appropriate eddy current POD function. In the current model the POD function can be changed for each operating cycle simulating actual revisions to the inspection process. A detection/repair state identity matrix is developed (ID_{ij}) in which a

value of 0 denotes an undetected defect, a value of 1 denotes a detected defect and a value of 2 denotes a repaired defect. For the repair-on-detection simulation, a value of 2 is assigned for the following inspection immediately on detection.

The final step in the simulation process is the identification of defects which have progressed to the point where tube burst or leakage can be expected. For a given inspection (j), each defect has three important attributes, which in combination, determine the structural and leakage integrity:

- Structural average depth (D_{ij})
- Structural length
- Maximum depth ($4/\Pi \times D_{ij}$)

Two attributes determine if a defect is to be counted as a burst in the simulation. These are the critical through wall crack length and the critical structural depth for the defect which has been assigned a characteristic structurally significant length. A defect is counted as a burst in the j^{th} inspection if three conditions are met:

- The defect structural depth is greater than or equal to the critical depth
- The defect characteristic length is greater than or equal to the critical through wall length
- The defect is undetected or newly detected

Defects which are too short to cause a complete burst, but of sufficient depth, are counted in a separate category which affects leakage integrity. These defects can "pop-through" under accident loading to their full structural length, and are assumed to behave in this manner. The characteristics of each defect of this type are stored in a special output file for use in separate leakage calculations.

The third class of important defect is the stable through wall defect for which the maximum computed depth exceeds 100% of the tube wall

thickness. The characteristics of each defect of this type are stored in the same output file as those of the "pop-through" defects for subsequent leakage analysis. Figure 3.1 shows the logic structure for the defect classification process.

At this point, one trial of the Monte Carlo simulation is complete. The key information available from this stage of the simulation includes numbers of:

- Defects detected
- Bursts predicted
- Pop-through defects predicted
- Stable leakage defects predicted

The probabilistic run-time analysis consists of many thousands of repeated trials which are used to obtain probability distribution functions for the frequencies and magnitudes of defects which can comprise the structural and/or leakage integrity of a steam generator.

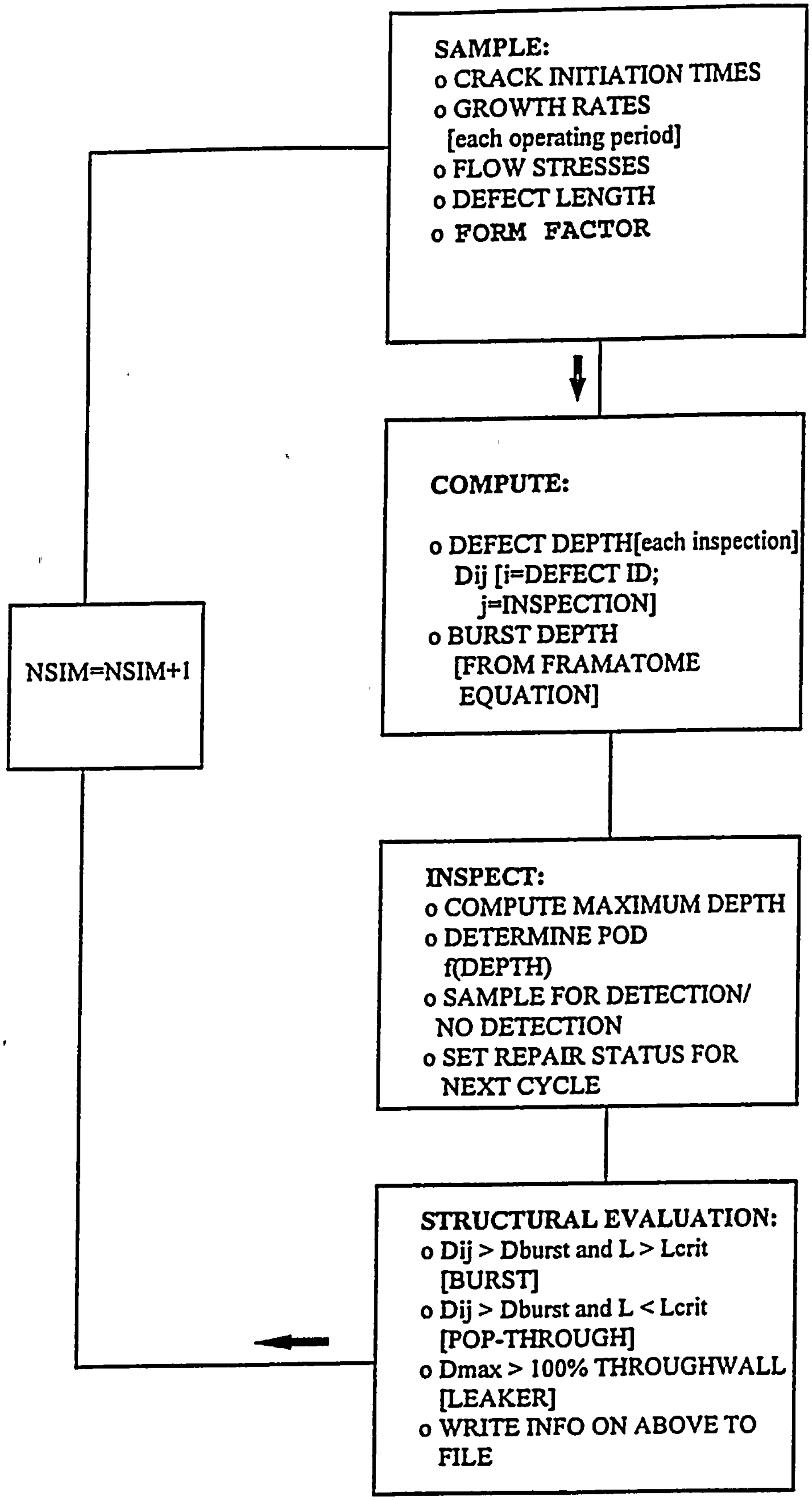


Figure 3.1 SIMULATION SCHEMATIC FOR REPAIR ON DETECTION MODEL

SECTION 4

PROBABILISTIC MODEL FOR LEAKAGE INTEGRITY

The probabilistic leakage integrity model used in the Palo Verde Unit 2 analysis was developed and used previously only in the repair-on-sizing (ROS) model for Plant F[10]. The model incorporates variations in morphology for axial degradation in a manner consistent with the implementation in the structural integrity model described in Section 2. For a given structurally significant depth, there exists a range of possible crack shapes. Propagation of these sampled shapes was used to define the extent of through wall penetration needed for leak rate calculations. The following sections describe the development of a leak rate sampling distribution for a given crack and the overall simulation method used to evaluate total leakage under steam line break conditions.

4.1 SIMULATION PROCESS FOR LEAKAGE EVALUATION

The repair on detection structural integrity simulation provides an output file containing information describing every through wall crack encountered in the simulation process. In addition, a similar description is provided for each defect which can "pop-through" under MSLB loading. The information for each defect includes:

- Structural average depth
- Maximum depth
- Structural length
- Flow stress

The frequency and character of the defects determine the probability of various levels of leakage under postulated MSLB conditions at end of cycle. The probabilities associated with various amounts of leakage are computed using the LEAKMC algorithm described in Figure 4.1. The LEAKMC

algorithm is a straightforward Monte Carlo simulation which computes the number and magnitude of leaks in a complete steam generator. The number of simulated leaking defects in the entire steam generator is obtained by first sampling from a binominal distribution using the average level frequency of leak computations and the number of equivalent levels in the steam generator. In the Palo Verde Unit 2 analysis a single level model is used.

4.2 DEVELOPMENT OF THE LEAKAGE SAMPLING DISTRIBUTION

The Monte Carlo simulation model produces an output file describing all cracks which were found to have penetrated the tube wall thickness at end of cycle. In addition, cracks determined to have "popped through" the wall at steam line break conditions but not burst are listed. While crack growth is considered in terms of growth of structurally significant crack depth, different crack shapes are sampled to determine if wall penetration occurs and, if so, the length of the through wall portion of the crack. The means of describing crack shapes and the sampled shape distributions are described in Appendix B. Appendices B and C combine to provide details considered relative to crack shapes and the extent of wall penetration caused by both corrosion degradation and mechanical tearing.

Leakage calculations are based on computed through wall crack lengths at end of cycle and the sampled tensile properties for the crack of interest. Leak rate calculations using the PICEP code were performed as part of an industry-wide EPRI program on circumferential cracking. Regression analysis of PICEP results led to leak rate equations as a function of crack length and crack opening area. Equations were developed for room temperature pressurization, typical normal operating conditions and bounding steam line break conditions.

At typical normal operating conditions in the vicinity of 600°F the leak rate, Q, is given by:

$$Q = (a + b \exp [c (A/L)^{0.564}])Ap^{N}$$

$$N = (1 - d \exp [eA^{4.259}])$$

where

Q = leak rate (gpm at 70°F)

A = crack opening area (inches^2)

L = total crack length (inches)

p = primary pressure - secondary pressure (psi)

a = 24.02751

b = -23.90614

c = -57.78132

d = 0.602285

e = 0.981541

At steam line break conditions the leak rate is given by:

$$Q = (c + b \exp [c (A/L)^{0.451} + d (A/L)]) Ap^{1.333}$$

where Q, p, A and L are as above and:

a = 0.0137492

b = -0.0138257

c = -18.076667

d = -395.8772

Crack opening areas are computed using a formula from the Ductile Fracture Handbook.[12] A plastic zone correction is used with a double iteration step. This was found to provide an excellent match with measured crack opening areas for Alloy 600 tubing containing axial cracks as well as EDM slots.

The reasonableness of the leak rate calculations is illustrated by Figure 4.2. Calculated and measured leak rates at normal operating conditions are

plotted for comparison. The dotted lines are the leak rate calculations for axial cracks in 0.750 inch and 0.875 inch diameter tubing. The heavy solid line represents data for fatigue cracks in both sizes of tubing.[13] Other data[14] from fatigue cracked specimens are shown as solid diamonds. Finally, data[14] on laboratory stress corrosion specimens is shown as open square symbols.

As expected, smooth fatigue cracks exhibit higher leak rates than stress corrosion cracks of the same axial through wall length. Both the fatigue cracked and stress corrosion cracked specimens exhibit minimal as cracked, crack opening areas. Care must be taken in examining the validity of benchmarking leak rate data. Laboratory produced cracked specimens, either fatigue cracked or stress corrosion cracked, can easily be blunted to large, as cracked, crack opening areas which are totally unrealistic compared to cracks produced in service.

Calculated leak rates are shown to be comparable to measured leak rates for specimens with sharp fatigue cracks. Fatigue cracks produce a conservative upper leak rate compared to service produced cracks. The calculated leak rates are somewhat lower than the measured leak rates for fatigue cracked specimens, yet calculated leak rates are a good upper bound to data from the laboratory stress corrosion cracked specimens.

Figure 4.2 is a highly satisfactory illustration of appropriate leak rate calculations. If only stress corrosion samples are used for benchmarking, three substantial problems are likely to develop. The first problem is matching the as-cracked openings of laboratory specimens to service produced cracks. The as-cracked opening is very sensitive to the method of specimen preparation. The second problem is a tendency to try to match calculated leak rates to the average measured leak rates using adjustable analysis parameters. This only serves to accentuate the scatter. The third

problem is akin to the second. Even if the crack morphology is known exactly from destructive examination after leak testing, it is very difficult to predict actual measured leak rates. Crack tortuosity is difficult to treat, and, more importantly, mechanical rupture of small ligaments during testing makes it very difficult in some cases to identify the correct leaking through wall crack length. Outliers in calculated leak rates for stress corrosion cracks are often due to selecting an incorrect leaking crack length. The addition of leak rate data from sharp fatigue cracked specimens provides a firm basis to evaluate leak rate calculations under well defined conditions and construct reasonable upper bound leak rates.

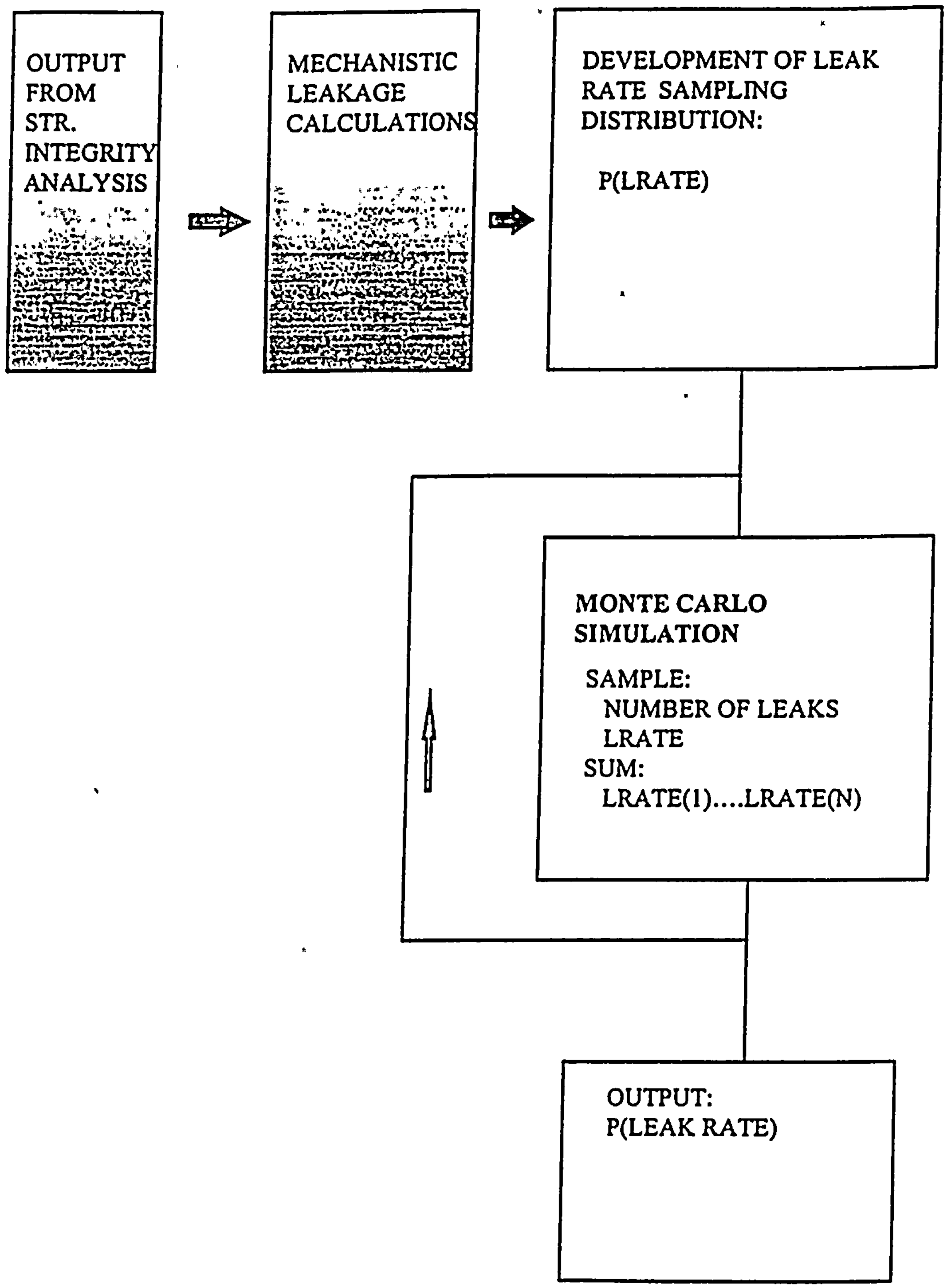


Figure 4.1 SCHEMATIC OF LEAKMC ALGORITHM

LEAK RATE VERSUS CRACK LENGTH

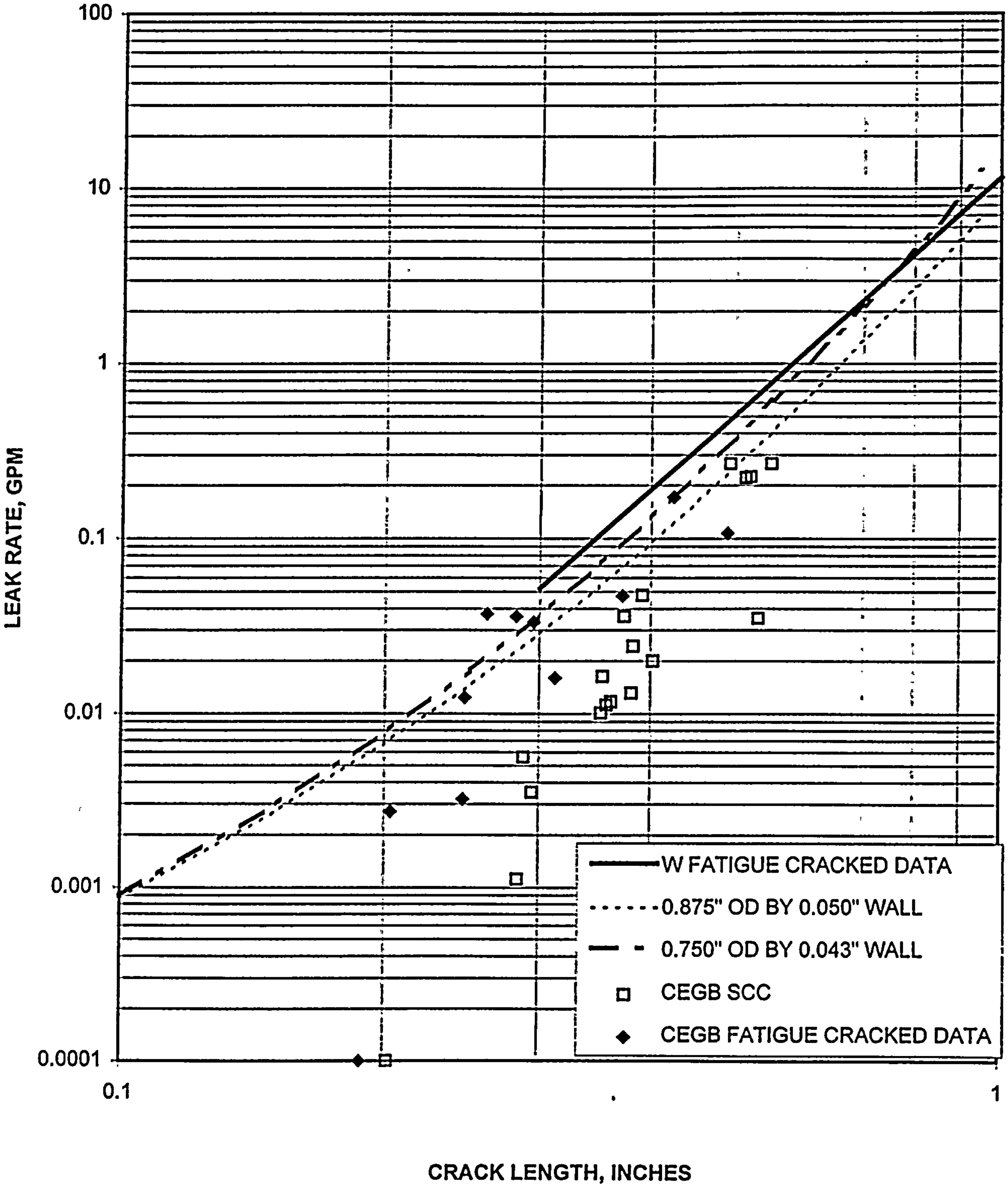


Figure 4.2 CALCULATED AND MEASURED LEAK RATES FOR AXIAL CRACKS IN ALLOY 600 TUBING AT NORMAL OPERAITNG CONDITIONS

SECTION 5
ANALYSIS INPUT

This section documents the input parameters for the probabilistic structural integrity model with the exception of plant operational parameters such as inspection times and operating temperatures. These are documented in the structural and leakage evaluation section. The required input for the structural integrity model includes probability distribution functions for:

- Tubing mechanical properties
- Defect length (EOC)
- Probability of detection
- Defect growth rates
- Defect initiation

There is no additional input for the leakage integrity model.

5.1 TUBING MECHANICAL PROPERTIES

The actual yield strength and ultimate tensile strength values were available for Palo Verde Unit 2 steam generator tubing. Room temperature test results were adjusted to account for the variation of flow strength with temperature. A normal distribution was fitted to this data and this provided the tube strength input to probabilistic calculations. Figure 5.1 shows the probability distribution function used for Unit 2 steam generator tubing in the simulation. This is identical to the normal distribution used in the previous Unit 2 analysis.[5]

5.2 DEGRADATION LENGTH DISTRIBUTION

The distribution of crack length judged to be the most appropriate for the present analysis is based on the RPC data from the most recent (R6) upper bundle inspection at Palo Verde Unit 2. A 0.115 inch diameter rotating

pancake probe was used. This probe supplies the structurally significant crack length. About 20% of the crack lengths are greater than 1.0 inch. This provides a reasonable long crack effect, but not the grossly over conservative shallow long crack contribution of the plus point probe. A log-normal distribution was found to best describe the RPC lengths. The axial length distribution used in the simulation is shown in Figure 5.2

5.3 PROBABILITY OF DETECTION

Five upper bundle inspections have been performed at Palo Verde Unit 2. The most recent two inspections have been performed using the Plus Point probe as the primary defect detection device. In the previous Unit 2 analysis[5] no pulled tube data was available from which a Plus Point specific POD function could be calculated. There was, however, a very significant inspection transient at U2R5 from which an "effective" Plus Point POD could be determined. The available data set consisted of defects (all detected by the Plus Point) for which approximately 60% were detected by the RPC probe. A single parameter POD function was developed and iteratively adjusted as shown in Figure 5.3 until the actual U2R5 inspection transient was replicated. The "effective" Plus Point POD function is compared with that used for RPC inspections in Figure 5.4.

Pulled tube data for upper bundle cracking in Plant C was recently made available. From this new data set an actual Plus Point POD function was developed using logistic regression and is compared to the previous POD function in Figure 5.5.

The good agreement between the POD functions provides an independent validation of the "effective" Plus Point POD used in the previous Unit 2 analysis. In the present analysis the RPC POD is used is the simulation of

inspections through U2M5-2 and the "effective" Plus Point POD is used in simulation of inspections U2R5-U2R7.

5.4 DEFECT GROWTH RATES

The defect growth rate probability distribution function is the dominant stochastic variable in a run time analysis. Even in the case of a repair-on-sizing application (ROS), where larger defects can remain in service, the vast majority of simulated defects which produce leakage or degraded burst pressure estimates result from extremes in defect growth rate.

For Palo Verde Units 2 and 3, defect growth rates are computed using RPC voltage growth statistics obtained from sequential inspections. The rates of voltage change are converted to crack growth rates using:

$$\frac{\Delta D}{\Delta t} = \frac{\Delta V}{\Delta t} \bullet \frac{\Delta D}{\Delta V} \qquad \text{Eq. 5.1}$$

where:

ΔD = depth change
ΔV = voltage change
Δt = inspection interval

The slope of the depth/voltage function $\left(\frac{\Delta D}{\Delta V}\right)$ is obtained from a linear regression of the data shown in figure 5.6. The variation in the slope is significant. This is accommodated in the development of the growth rate statistics by modelling $\Delta D/\Delta V$ as a random variable. The apparent growth rate probability density function is, therefore, the convolution of random variables representing voltage growth rate and $\Delta D/\Delta V$. The variation in the depth/voltage slope can be interpreted as particular paths followed by the confounding variables of length, crack shape and crack face ligaments as crack growth occurs. Different slopes are, therefore, reflections of various evolutions of crack morphology with time.

5.4.1 Evaluation of Cycle 6 Growth Rates

The development of the growth rate distribution function for Palo Verde Unit 2 was based on RPC data obtained during the U2R6 inspection. The data set consists of 92 pairs of RPC voltages. A histogram of this data is shown in Figure 5.7.

The overall growth rate distribution is obtained by convolution using Equation 5.1 as shown in Figure 5.8. The resulting growth rate probability distribution function for positive growth rates is shown also in Figure 5.8 in cumulative form. As can be seen the average growth rate from this data is 1-2% through wall/EFPY. The extreme growth rate is less than 10% through wall/EFPY.

The protocol for growth rate simulation in the Palo Verde Unit 2 Cycle 7 simulation is to sample directly from the ΔV data and to apply a independently sampled $\Delta D/\Delta V$ value. All non-positive growth rates are treated as zero growth in this simulation.

5.4.2 Comparison of Cycle 5 and Cycle 6 Growth Rates

The apparent defect growth rates for Palo Verde Unit 2 have decreased substantially between operating Cycles 5 and 6. This can be seen in Figure 5.9 which compares distribution functions for both observation periods. This observation is consistent with the relative proportion of negative growth points in the two data sets. A partial explanation for this behavior is related to the observation period for the two data sets. In the case of the Cycle 5 data, much of the set was obtained from U2M5-2 - U2R5, a period of only 0.3 EFPY. This has the effect of maximizing the impact of NDE uncertainty relative to the 1.0 EFPY observation period for Cycle 6 data.

5.4.3 Comparison with Other Data

A comparison of defect growth rates computed for Palo Verde Units 2 and 3 is shown in Figure 5.10.

The most recent growth rates from the two units are comparable suggesting similar local chemistry and defect character. In both cases defect propagation rates are those expected for intergranular attack (IGA).

5.5 DEFECT INITIATION

The process of defect initiation is simulated by the Weibull time-to-failure model given by:

$$P(t) = 1 - \exp\{-(t/b)^N\} \qquad \text{Eq. 5.2}$$

where:

P(t) = proportion of tubes experiencing cracking by time t

N = shape parameter

b = scale parameter

The parameters for the basic Weibull model are obtained from observations of the number of defects appearing in sequential observations. These, of course, represent data from cracks which have progressed well beyond initiation into the detectable regime. A relatively simple time delay function has been implemented in previous analyses[1,4] and had proven consistent with observations. The actual time delay varied from 2 to 4 EFPY depending on the crack growth rate distribution.

A much more sophisticated approach to the development of a defect initiation function was used in the prior Unit 2 analysis.[5] A process was developed using the full simulation to jointly optimize the Weibull shape and

setback parameters. The process is shown schematically in Figure 5.11 and in terms of the optimization surface in Figure 5.12. This resulted in a much less severe Weibull shape parameter estimate than that obtainable purely from inspection results. The resulting agreement between model predictions and observations was excellent for all prior inspections.

The projections for U2R6 are shown in Figure 5.13 and overpredict the actual number observed by a factor of 2. A number of possible explanations for this large discrepancy were considered. Among these were:

- POD effects
- "Shutdown" of initiation process
- Defect propagation rate effects

The first of these possibilities was not substantiated by the independent validation of the "effective" Plus Point POD function. The "shutdown" of the initiation process required an unreasonable time estimate of occurrence to show the observed result. The effect of defect propagation rate, however, provided a numerically compelling explanation. This explanation is consistent with the remedial measures applied to Unit 2. Chemical cleaning and improved chemistry control have apparently had some beneficial effect on corrosion degradation growth rate. A marked improvement in the behavior of corrosion degradation in Unit 2 is not unexpected in view of the wide ranging array of applied remedial measures.

The prior Unit 2 simulation was modified to incorporate the observed U2R5-U2R6 growth rate distribution function only for the last simulated cycle. The number of defects predicted is compared with those observed in Figure 5.13. On the basis of this agreement, the initiation function for Palo Verde Unit 2 remains unchanged for the present analysis.

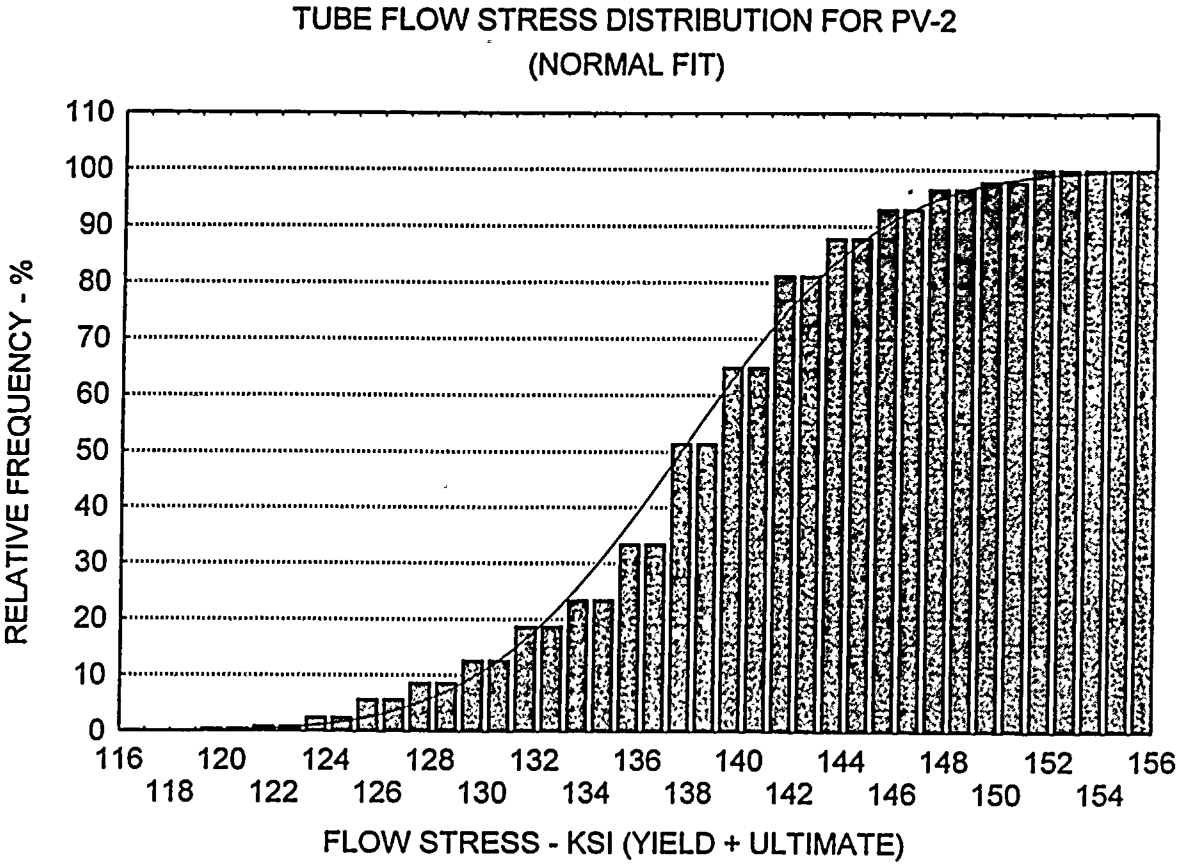


Figure 5.1 DISTRIBUTION FUNCTION OF FLOW STRESS

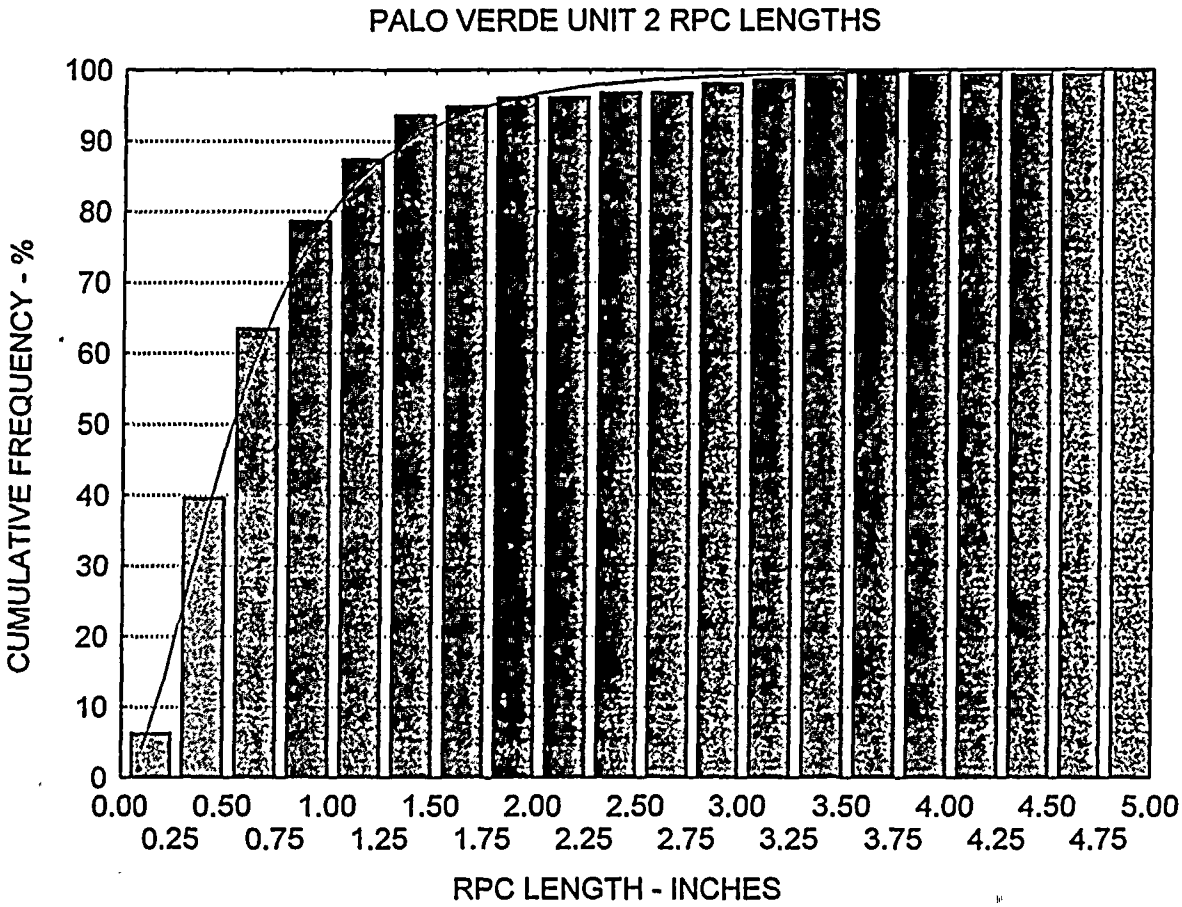


Figure 5.2 DISTRIBUTION FUNCTION OF EOC DEFECT STRUCTURAL LENGTHS

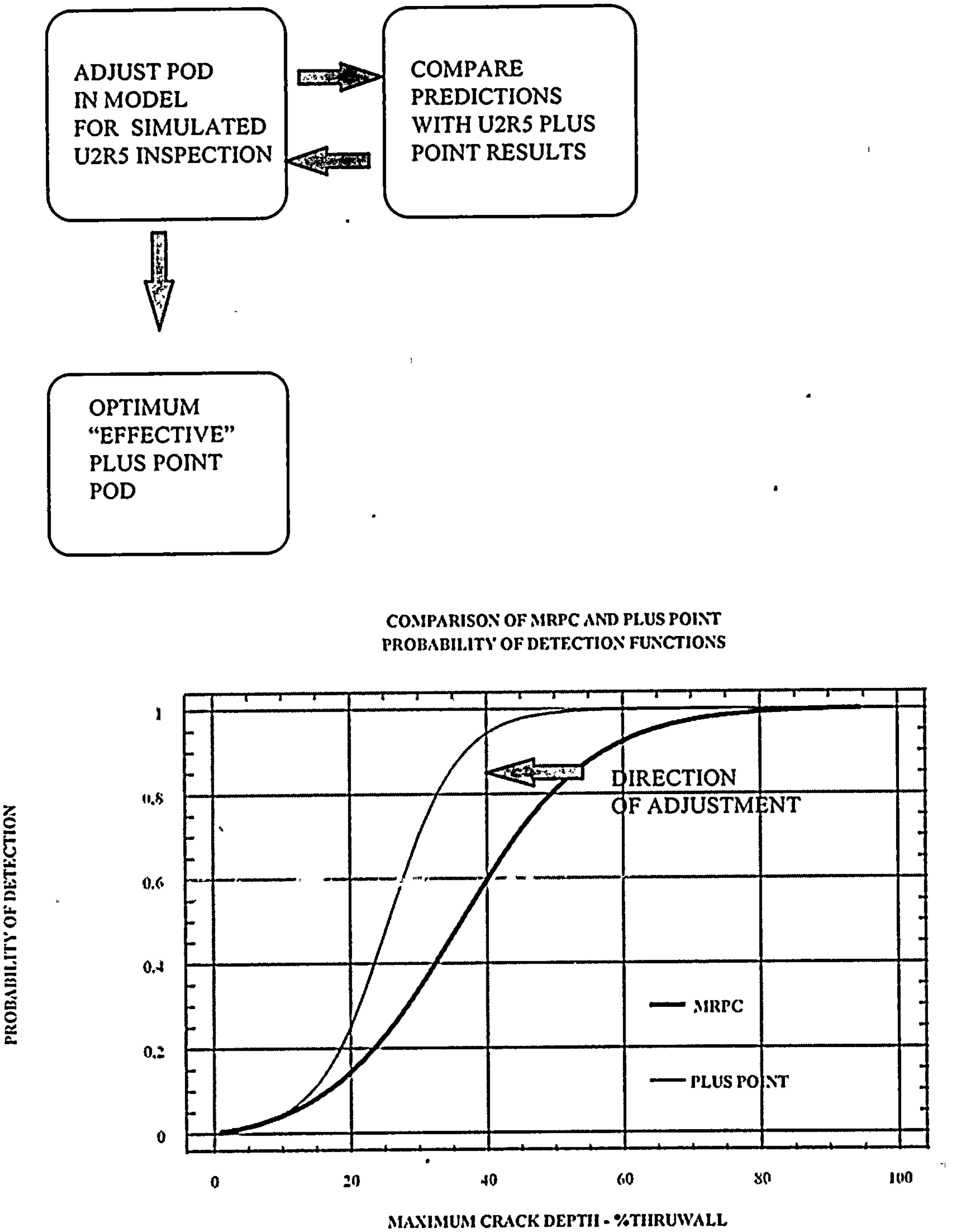


Figure 5.3 SCHEMATIC OF PLUS POINT POD OPTIMIZATION PROCESS

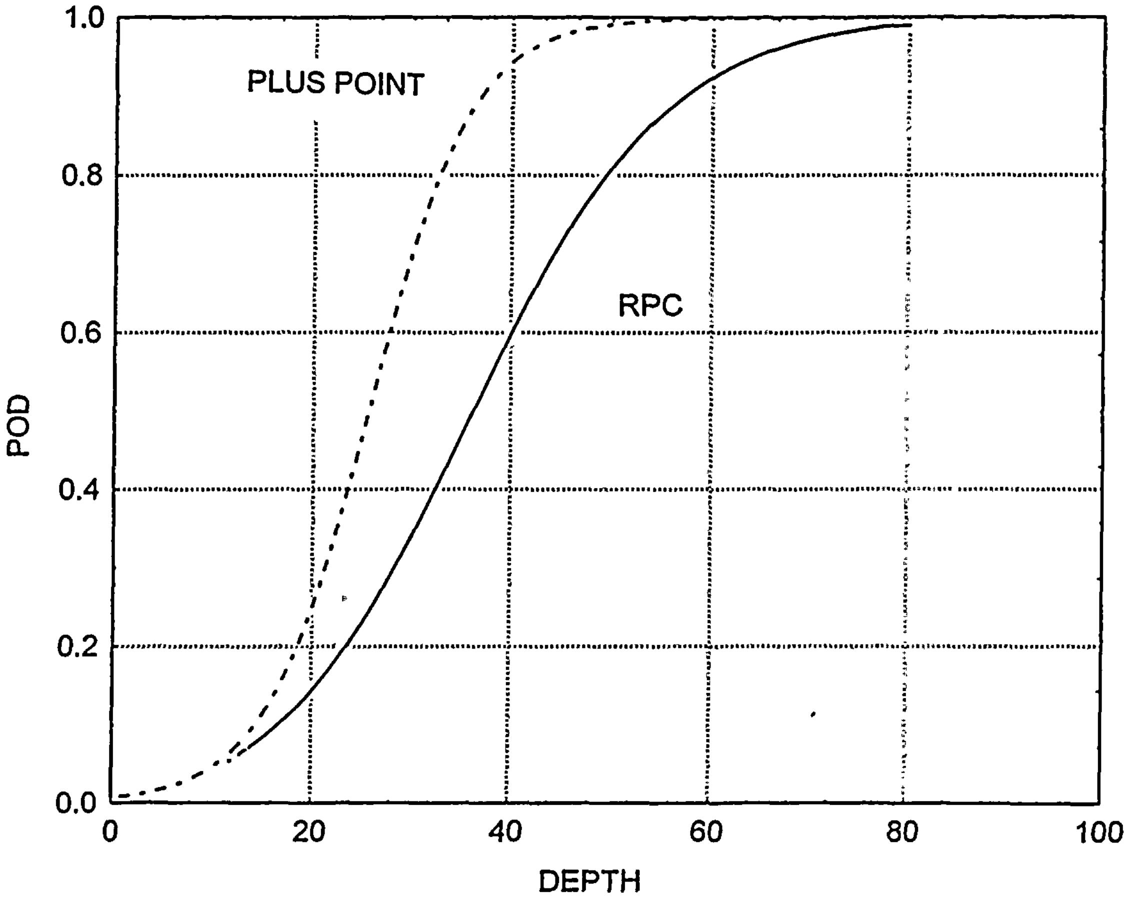


Figure 5.4 COMPARISON OF PLUS POINT AND RPC POD FUNCTIONS USED IN PALO VERDE SIMULATION

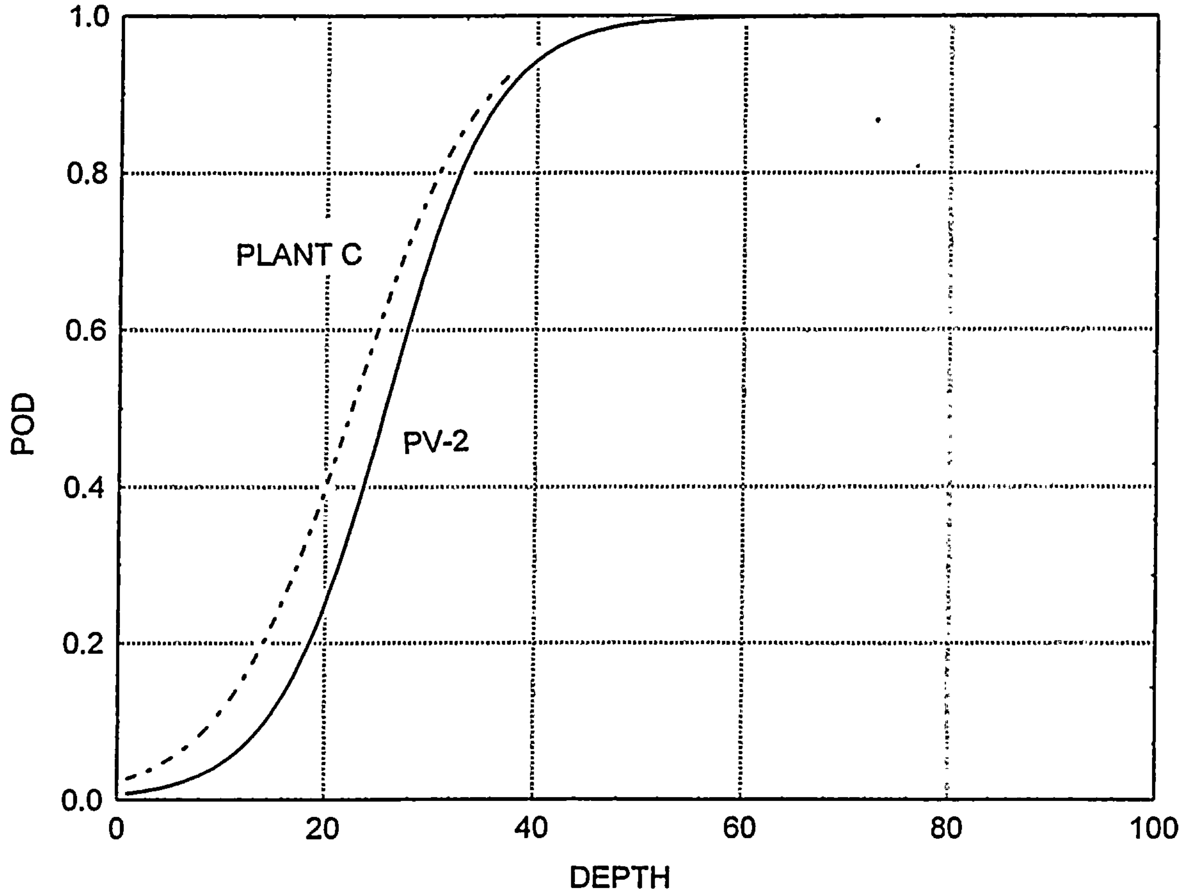


Figure 5.5 COMPARISON OF EFFECTIVE PLUS POINT POD AND PLANT C PLUS POINT POD

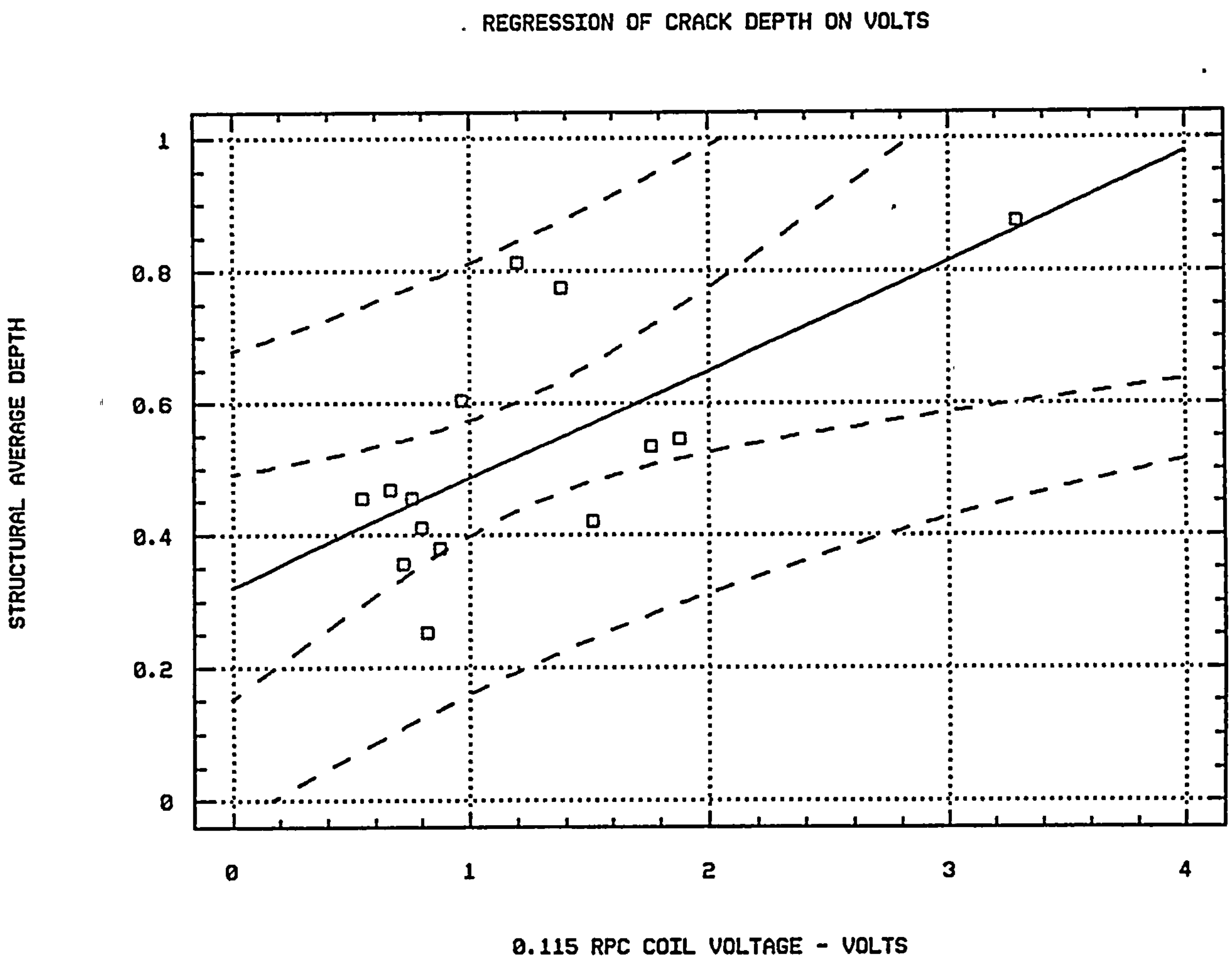


Figure 5.6 STRUCTURAL AVERAGE DEPTH VERSUS RPC VOLTS

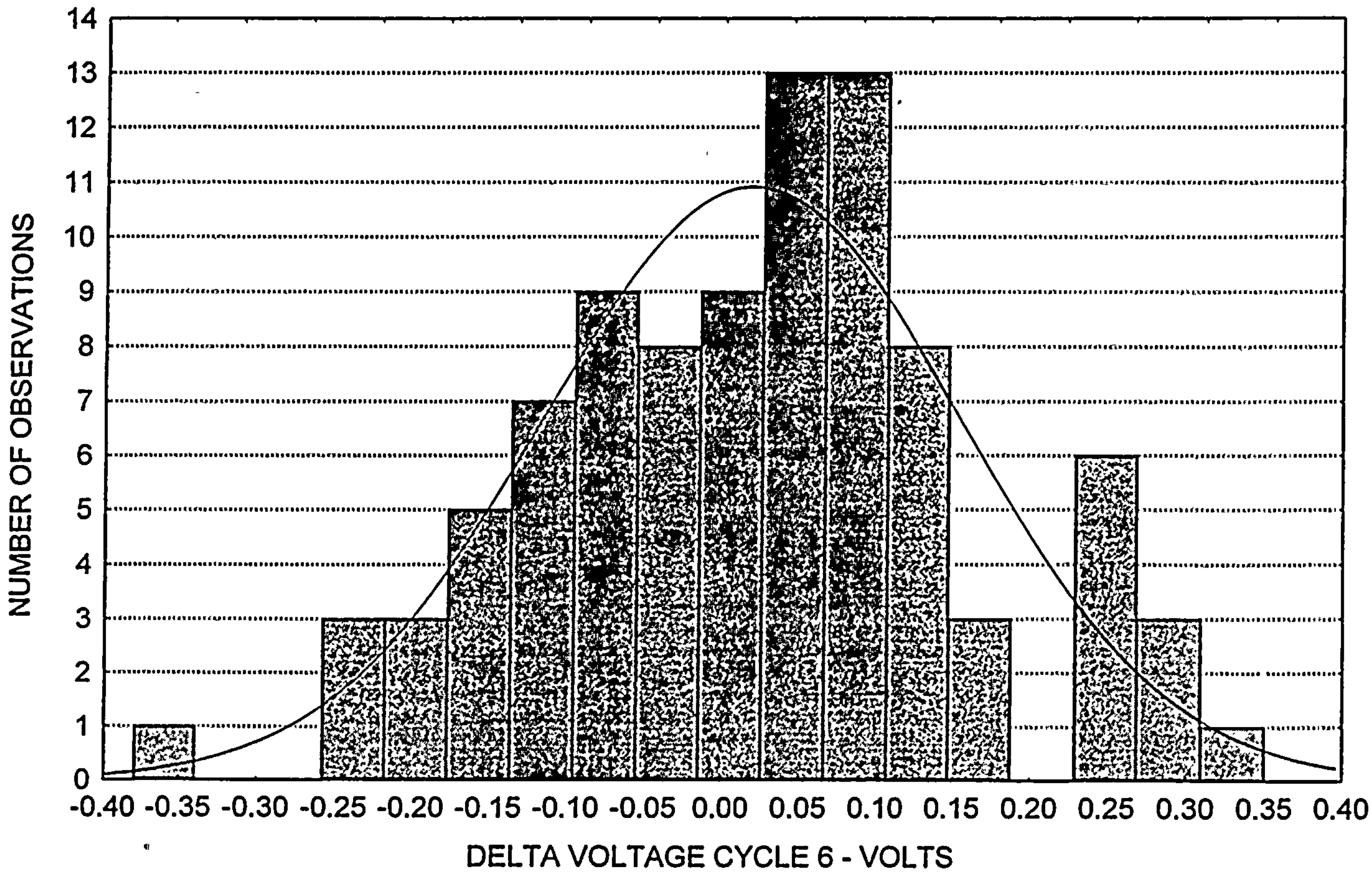


Figure 5.7 PALO VERDE UNIT 2 CYCLE 6 VOLTAGE GROWTH HISTOGRAM

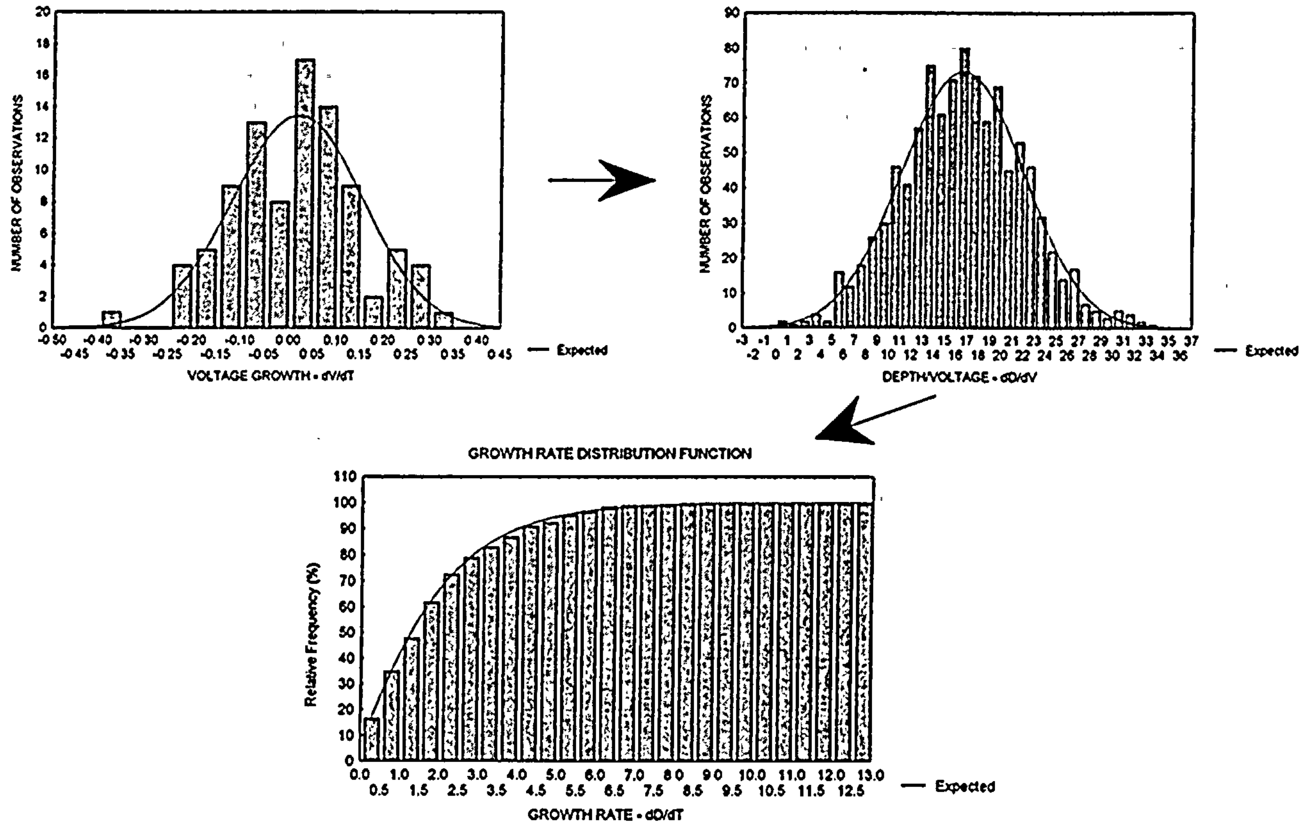


Figure 5.8 COMPUTATION OF GROWTH RATE DISTRIBUTION FOR CYCLE 6

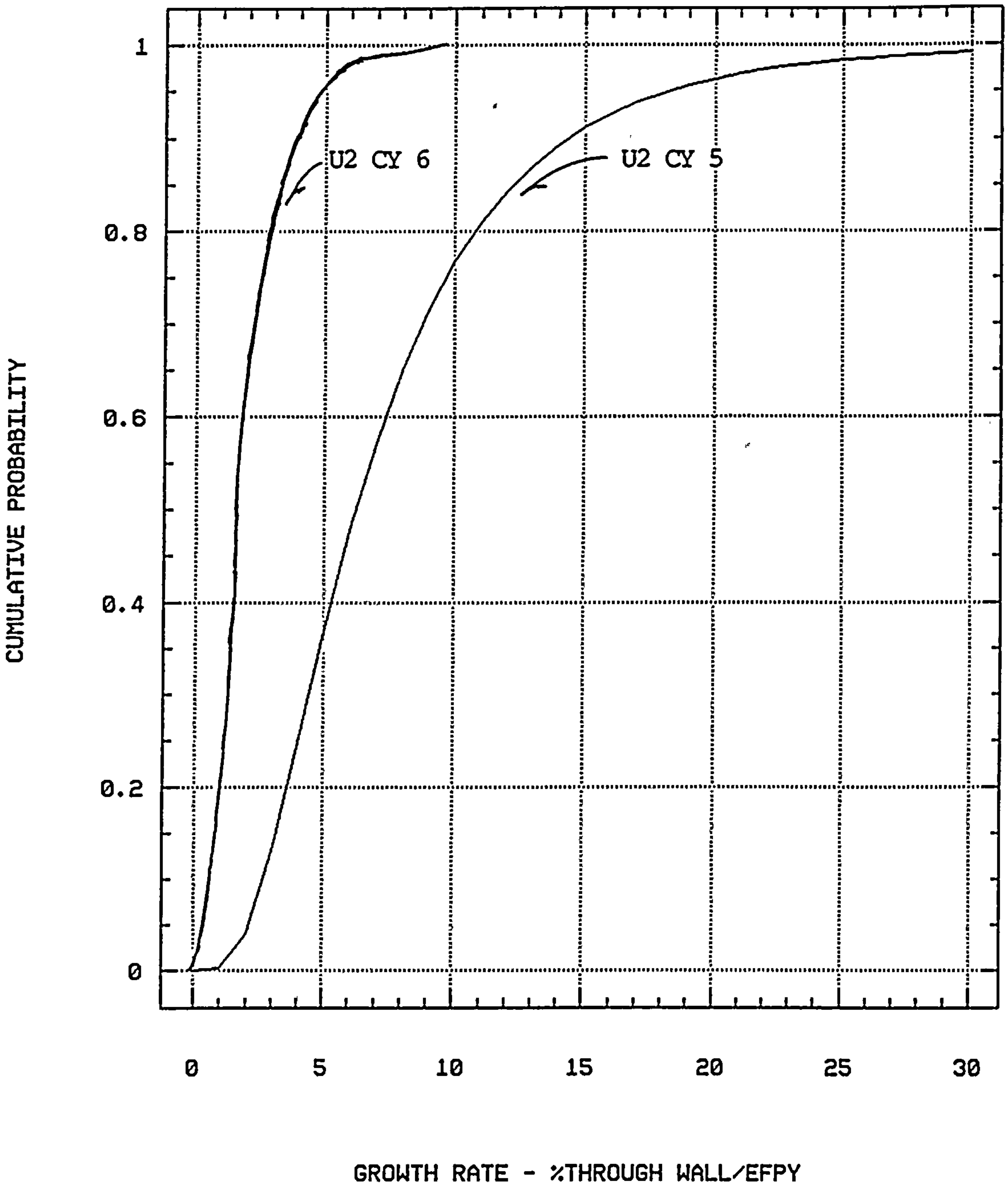


Figure 5.9 COMPARISON OF PALO VERDE CYCLE 5 AND CYCLE 6 GROWTH RATE DISTRIBUTIONS

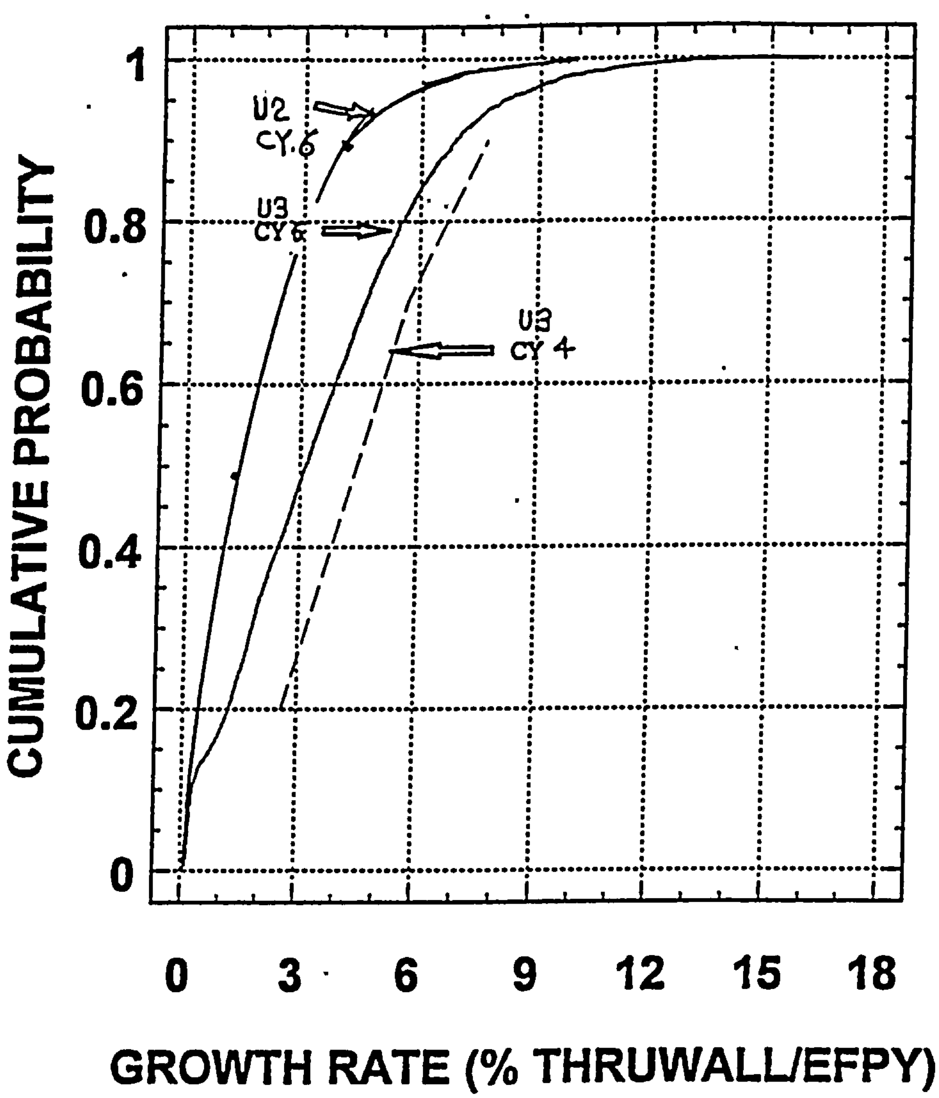
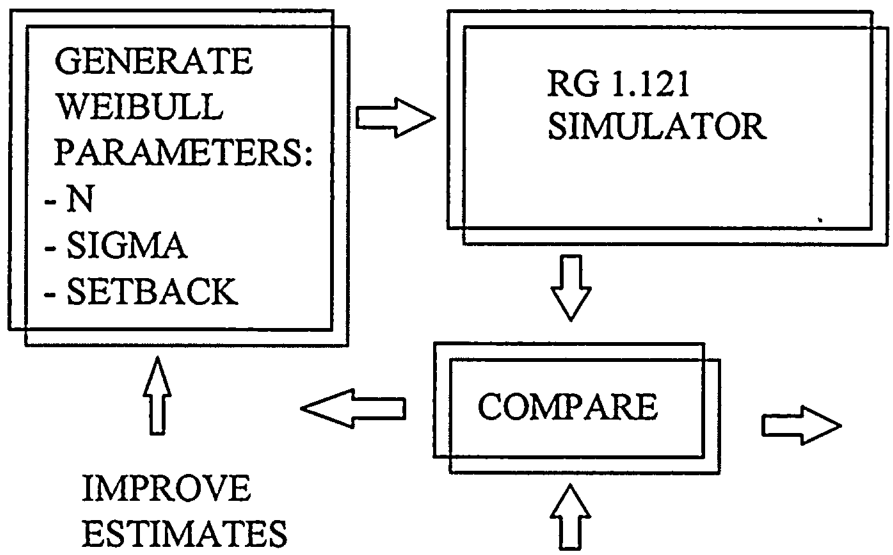


Figure 5.10 COMPARISON OF UNIT 2 AND UNIT 3 GROWTH RATE DISTIRBUTIONS



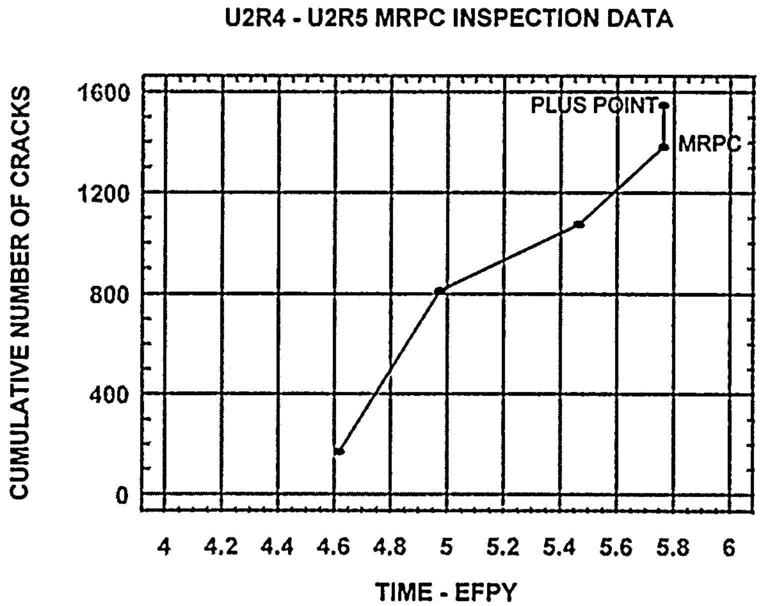


Figure 5.11 OPTIMIZATION PROCESS FOR INITIATION FUNCTION

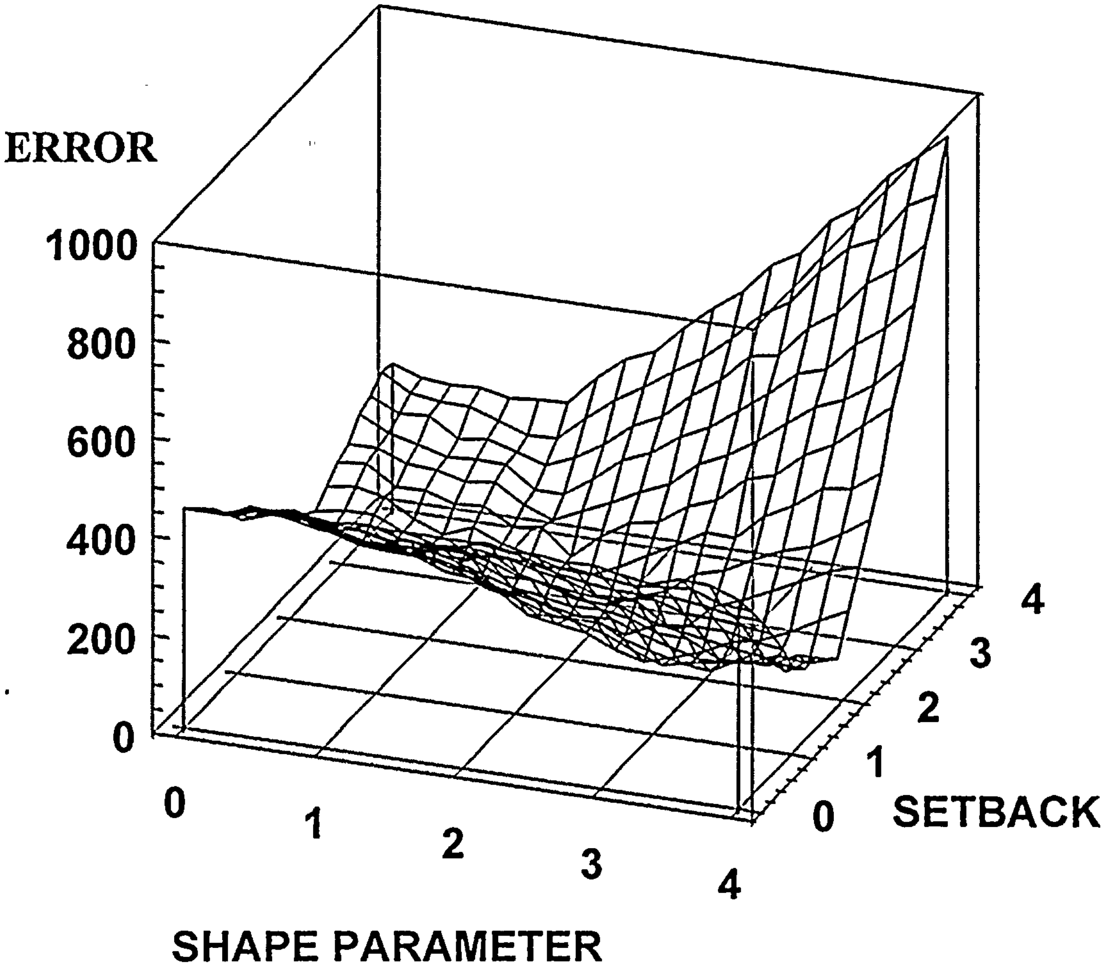
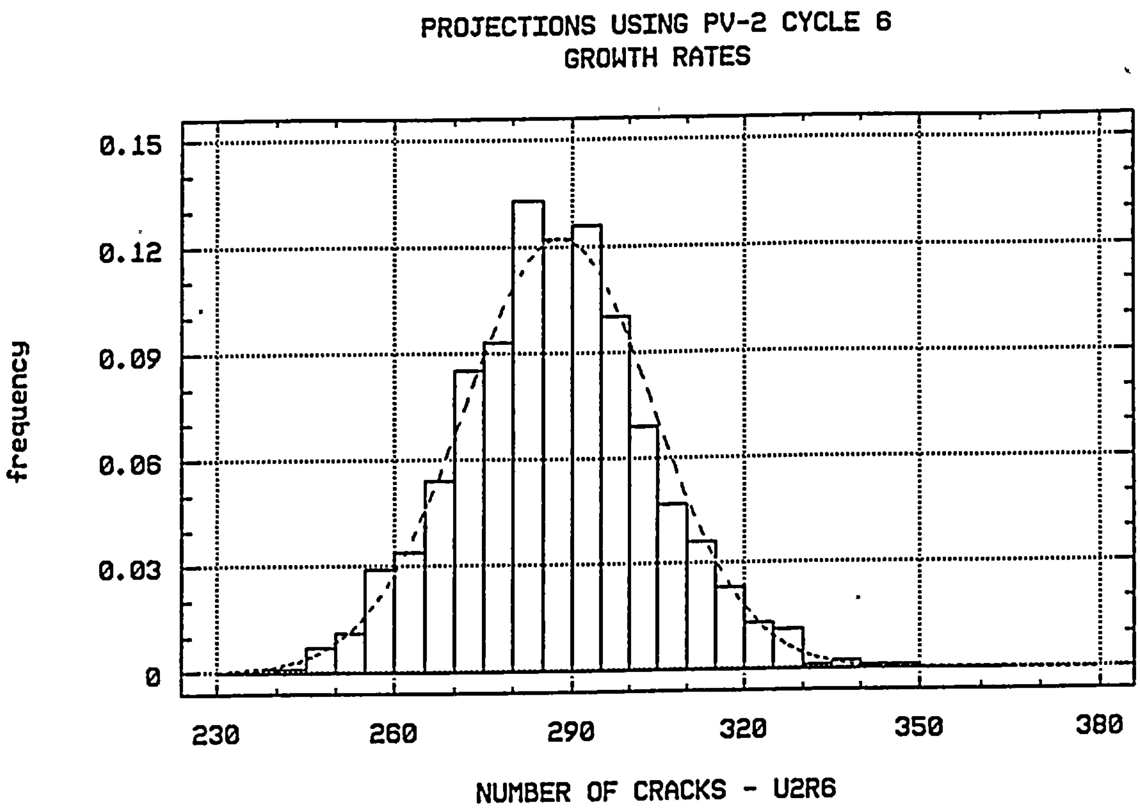


Figure 5.12 OPTIMIZATION SURFACE FOR INITIATION FUNCTION PARAMETERS



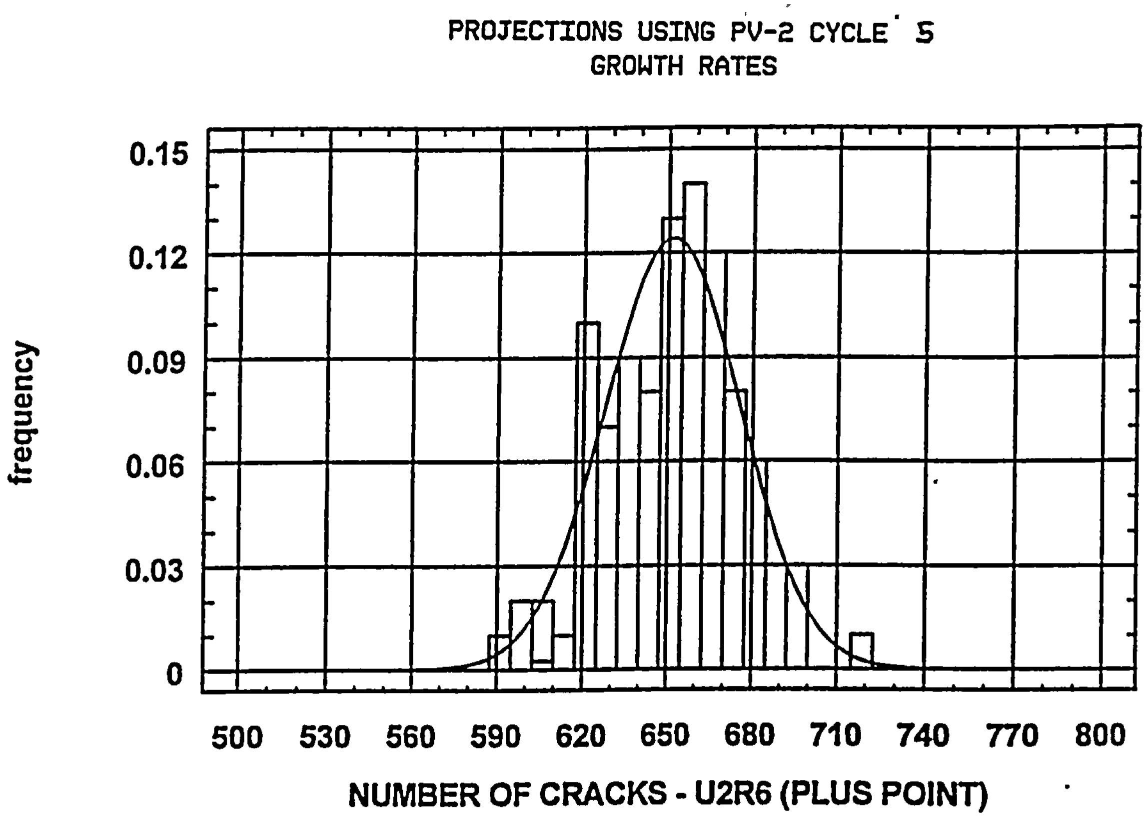


Figure 5.13 DEFECT PROJECTIONS FOR PALO VERDE UNIT 2 CYCLE 6

SECTION 6

STRUCTURAL AND LEAKAGE MARGIN EVALUATION

This section describes the use of the repair on detection, ROD, model to evaluate the probabilities of tube rupture under steam line break and RG 1.121 loadings. In addition, benchmarking of simulation results against U2R6 observations is included.

6.1 DESCRIPTION OF SIMULATION

The simulation performed in this analysis consists of five inspections and operating periods which is the maximum number possible in the present ROD model. In almost all respects the simulation is identical to that performed for Cycle 6.

There are important exceptions. The first is the use of the actual Cycle 6 growth rate distribution for the last two cycles of operation. The second is the use of a double RPC inspection at the simulated U2M5-1 outage to accommodate the inability to explicitly model the U2R4 outage. Table 6.1 shows the operating times, temperatures, POD functions, and crack growth rate distributions used for each operating period/inspection.

6.2 STRUCTURAL INTEGRITY EVALUATION

The structural integrity simulation was performed for a 16.5 month run time using the ROD simulator. Two cases were run each consisting of 10,000 trials. The first case had a differential pressure of 2400 psid (steam line break). The second case had a differential pressure of 3800 psid corresponding to RG 1.121 loading. No simulated bursts were encountered in either case. These null results give a best estimate burst probability of

7×10^{-5} for both steam line break and RG 1.121 conditions. The corresponding upper 95% confidence estimates is 3×10^{-4} for both conditions. These favorable results can best be understood by examining the extreme value distribution for simulated defect structural depths shown in Figure 6.1. As can be seen from the figure, the probability of any defect exceeding 50% through wall is an exceedingly rare event. This is a reasonable outcome given the low growth rates for Unit 2 defects in Cycle 6 and the efficacy of three sequential Plus Point inspections. Projected numbers of observed defects are shown in Figure 6.2.

As in the past, an independent assessment of structural margins has been performed. This work was performed by P. Besuner and is included as Appendix D. An essentially Bayesian approach was used in stark contrast to the Monte Carlo simulation model described in the bulk of this report. The structural margin assessments of the two techniques are in remarkable agreement. The conditional probability of burst at postulated main steam line break conditions, via the Appendix D technique, is 2×10^{-4} at an upper 90% confidence value. This corresponds extremely well with the value of 3×10^{-4} at an upper 95% confidence level noted above. Two markedly different techniques have led to essentially the same low level conditional probability of tube burst.

6.3 BENCHMARKING OF THE CYCLE 7 MODEL

The Palo Verde Unit 2 model is benchmarked against observed inspection results in two ways. The first involves the number of defects observed in the prior Unit 2 inspection. This is a measure of the joint effects of the initiation, POD, and defect growth rate models. As can be seen from Figure 6.3 - 6.6, the benchmarking in this regard is in good agreement.

The second benchmark is distributional and compares the observed RPC voltage distribution from U2R6 with a distribution from the simulation. This comparison is shown in Figure 6.7. The simulated distribution is in good agreement with the observed. It should be noted that the simulation used in this benchmark used the observed Cycle 6 growth rates. The original projections using Cycle 5 growth rates predicted a much more adverse EOC voltage distribution. The second benchmark is a measure of the joint effects of the POD function and the growth rate distribution function.

6.4 LEAKAGE EVALUATION

No through wall defects were encountered in the structural integrity simulations (20,000 total trials). The upper 95% confidence estimate of the probability of any leak during Cycle 7 operation is 4×10^{-5}. An explicit computation of leakage was, therefore, not performed. Figure 6.8 shows the extreme value distribution for maximum defect depth which includes the effects of variable defect aspect ratio.

6.5 ADDITIONAL EVALUATIONS

An additional set of computations were performed using a more adverse growth rate distribution from Palo Verde Unit 3 EOC-6 (see Figure 5.10). The case was otherwise identical to that run for the 3ΔP (3800 psid) structural integrity evaluation described in Section 6.2 The new crack growth rate distribution was imposed for the simulation of operating cycles 6 and 7. The results were identical in terms of probability of tube burst and leakage to the initial computations in that no simulated bursts were encountered. The probabilities cited in Section 6.2 remain valid, both on a best estimate and 95% confidence level. Table 6.2 summarizes the probabilistic estimates. The predicted RPC voltage distribution for EOC-7 is given in Figure 6.9.

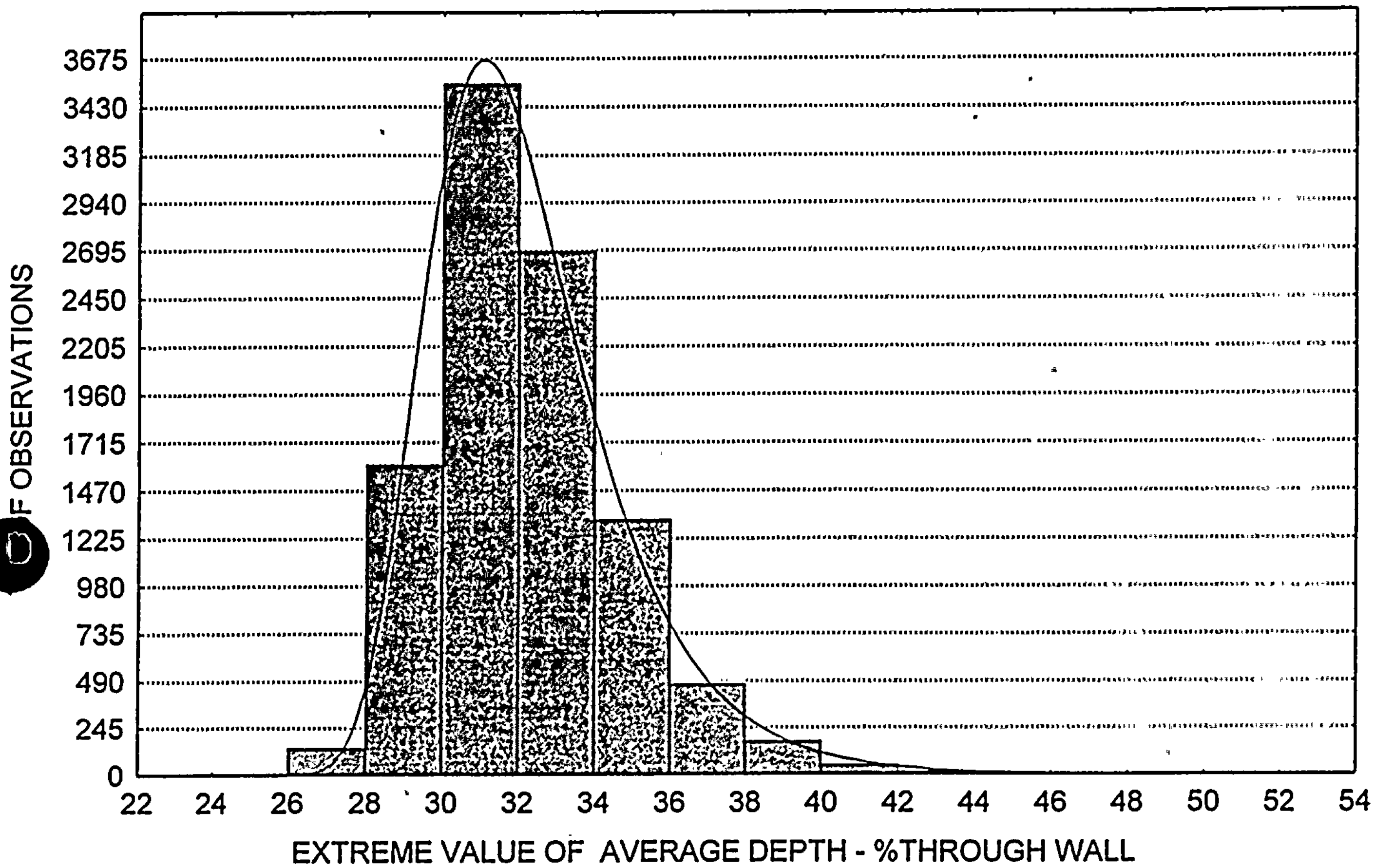


Figure 6.1 EXTREME VALUE DISTRIBUTION OF AVERAGE DEPTH (U2R7 PROJECTION)

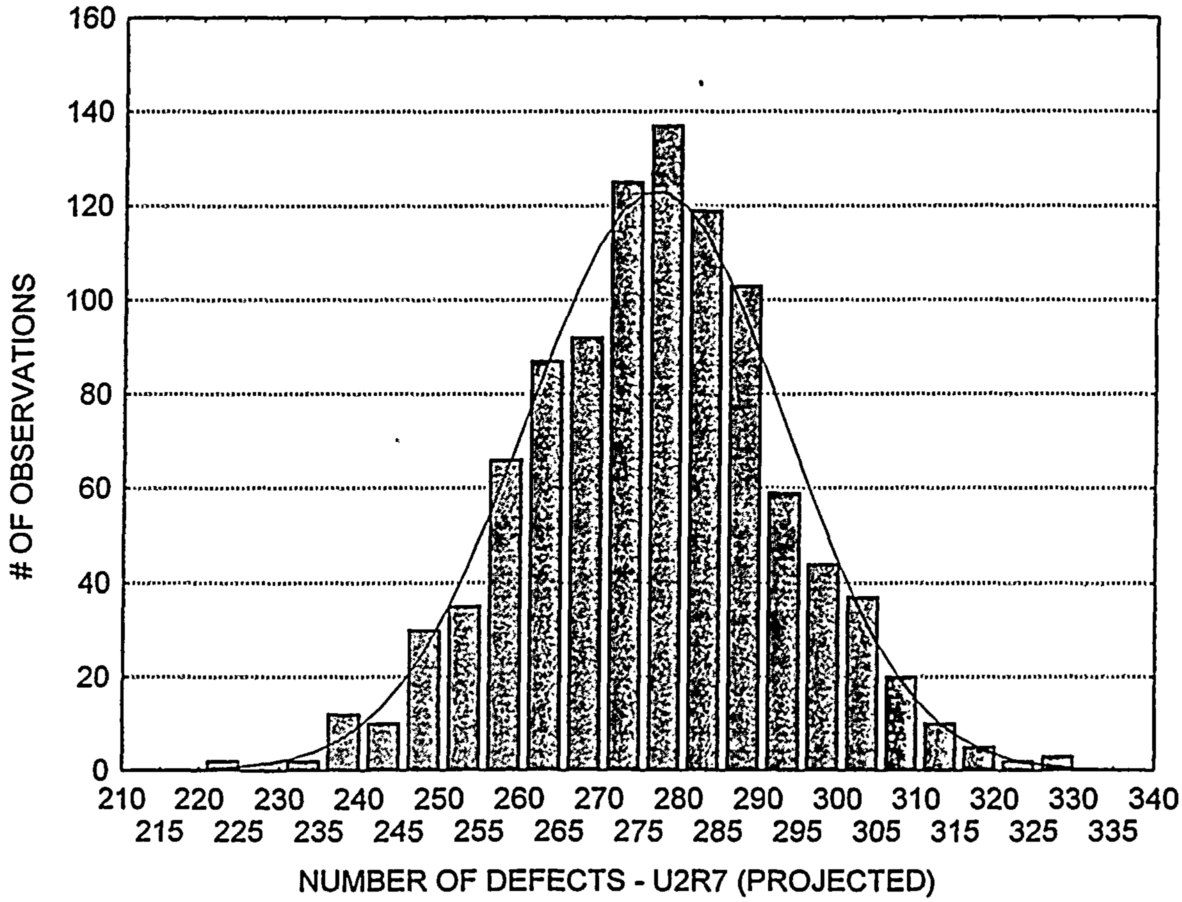


Figure 6.2 PROJECTED NUMBER OF DEFECTS (U2R7)

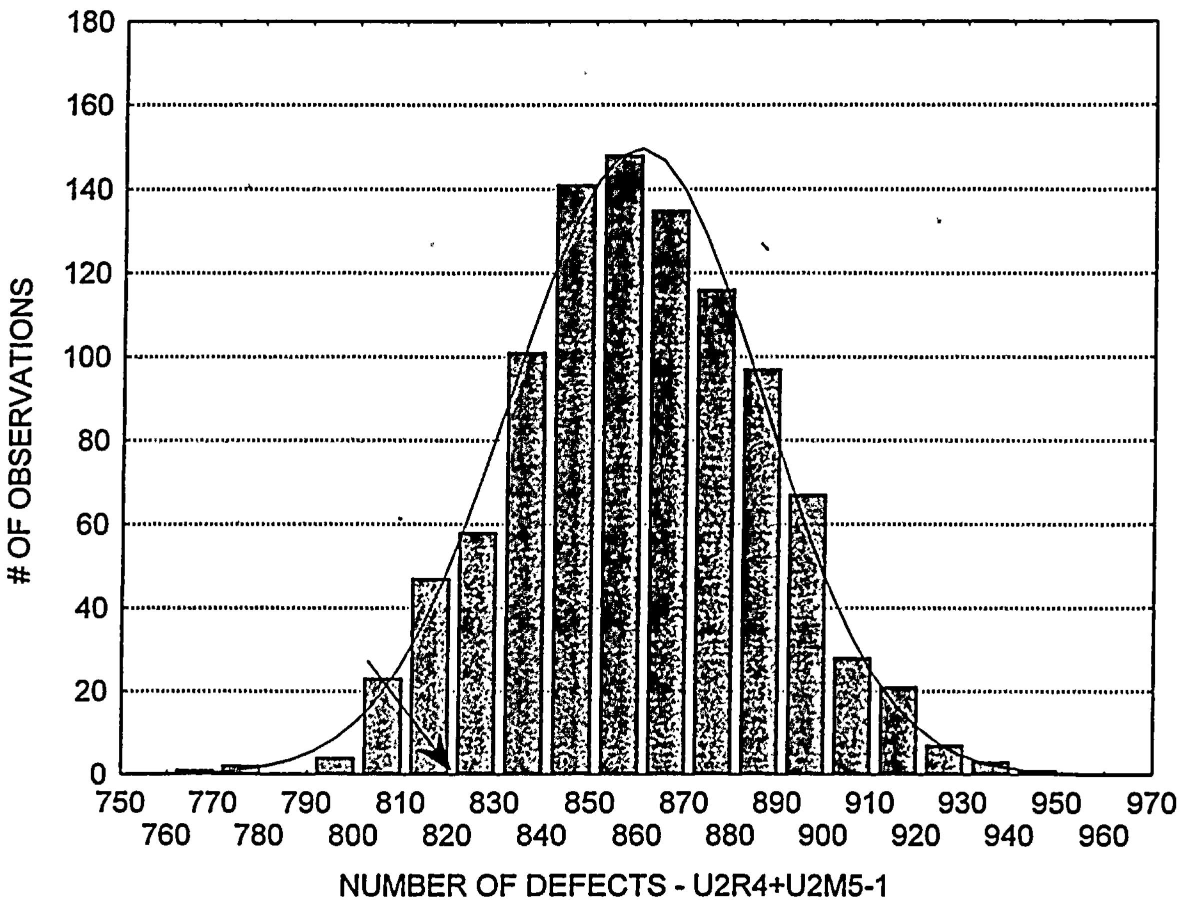


Figure 6.3 COMPARISON OF PREDICTED AND OBSERVED NUMBER OF DEFECTS (TOTAL U2R4 AND U2M5-1)

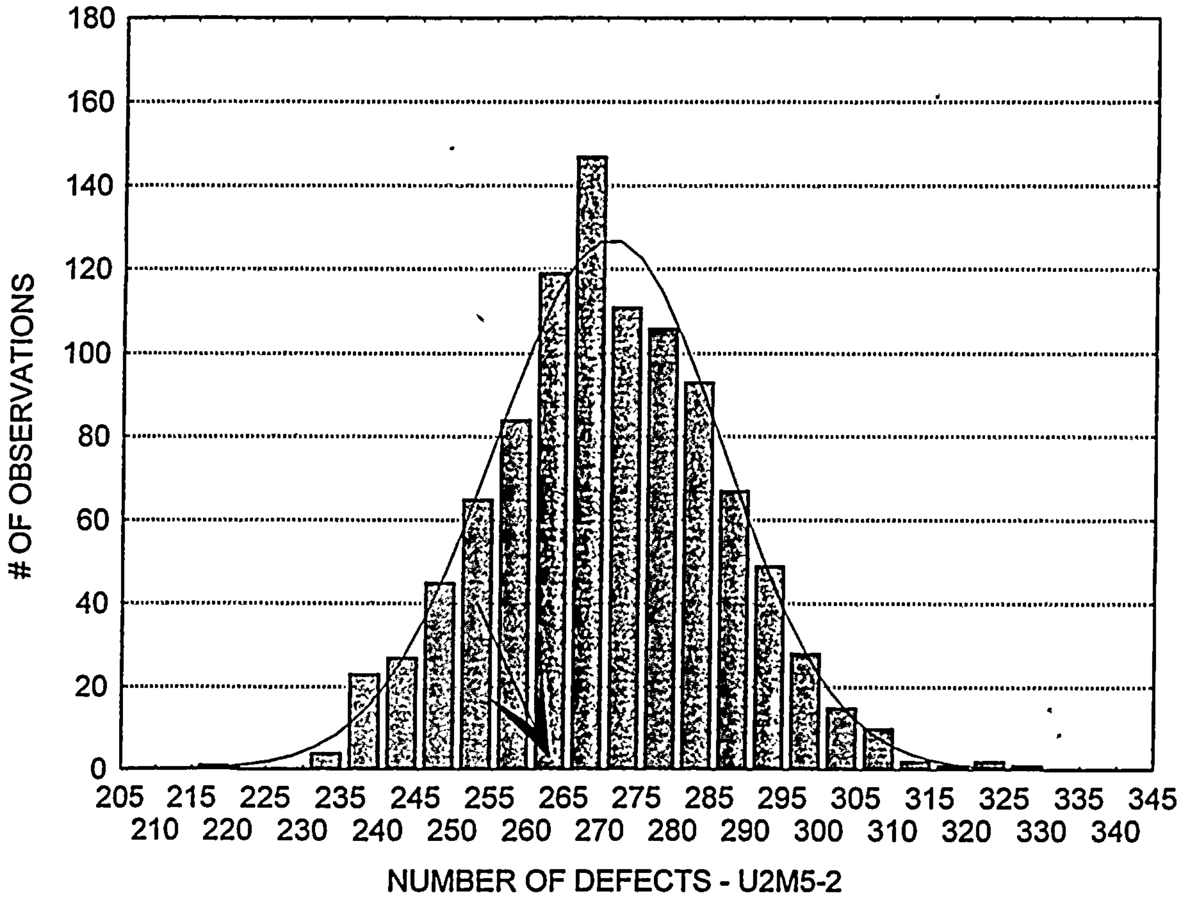


Figure 6.4 COMPARISON OF PREDICTED AND OBSERVED NUMBER OF DEFECTS (U2M5-2)

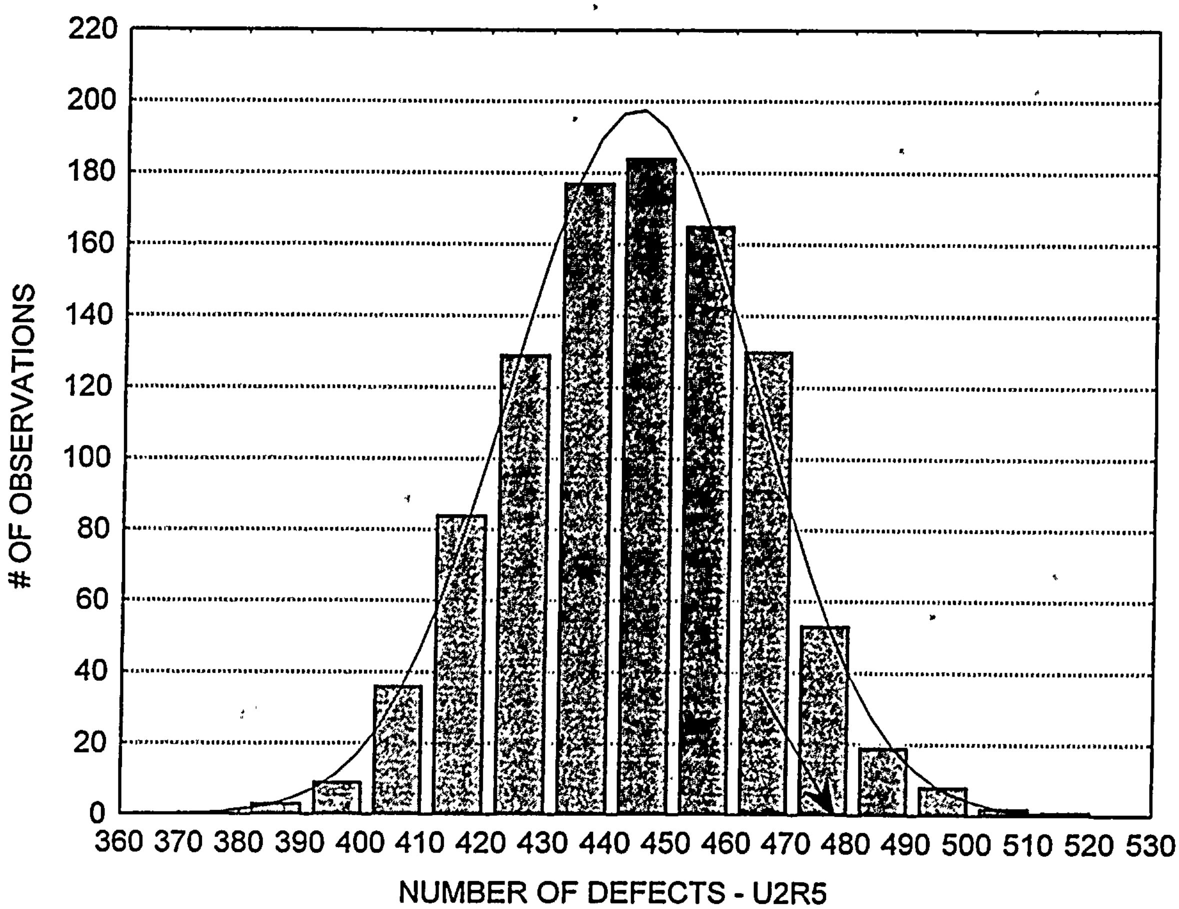


Figure 6.5 COMPARISON OF PREDICTED AND OBSERVED NUMBER OF DEFECTS (U2R5)

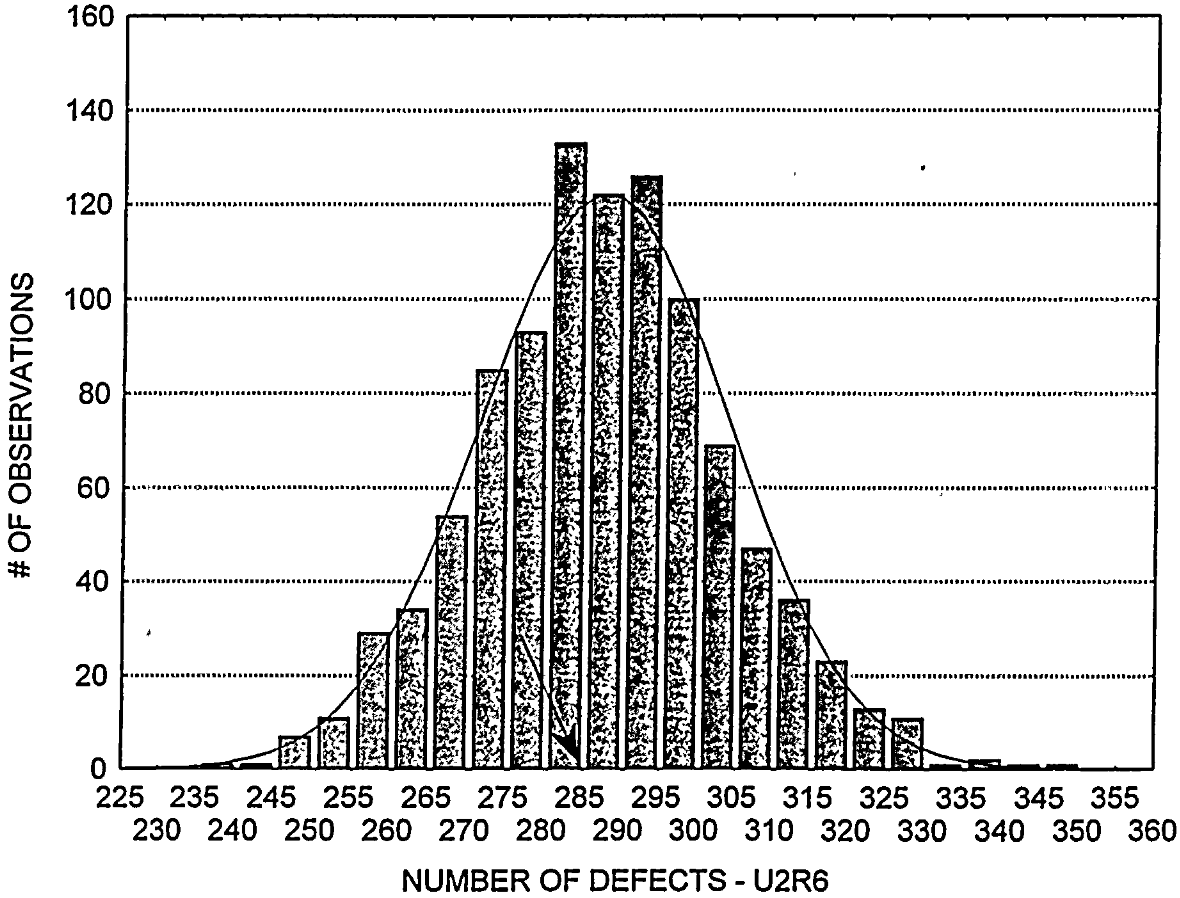


Figure 6.6 COMPARISON OF PREDICTED AND OBSERVED NUMBER OF DEFECTS (U2R6)

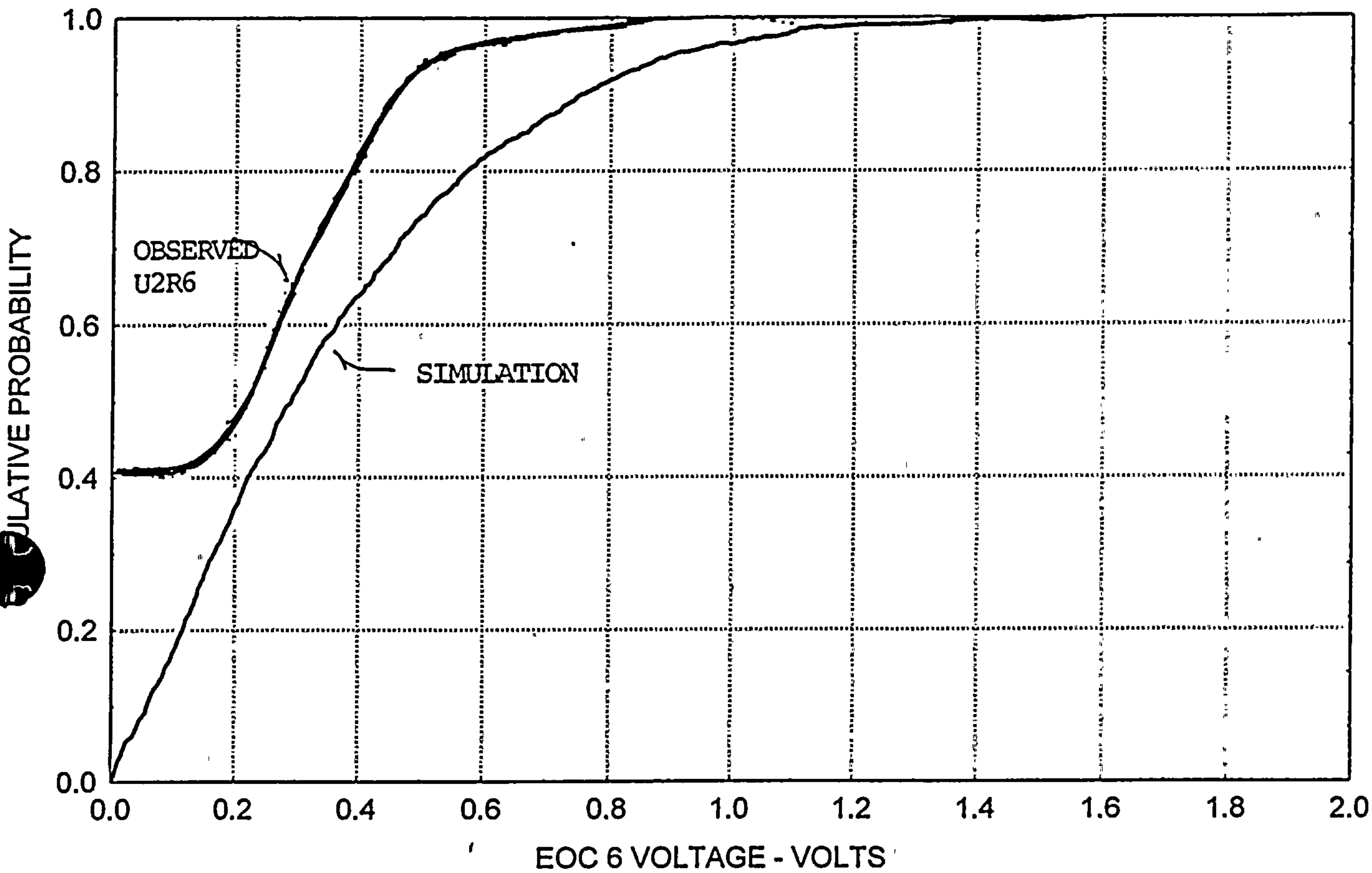


Figure 6.7 COMPARISON OF PREDICTED AND OBSERVED RPC VOLTAGE DISTRIBUTION (U2R6)

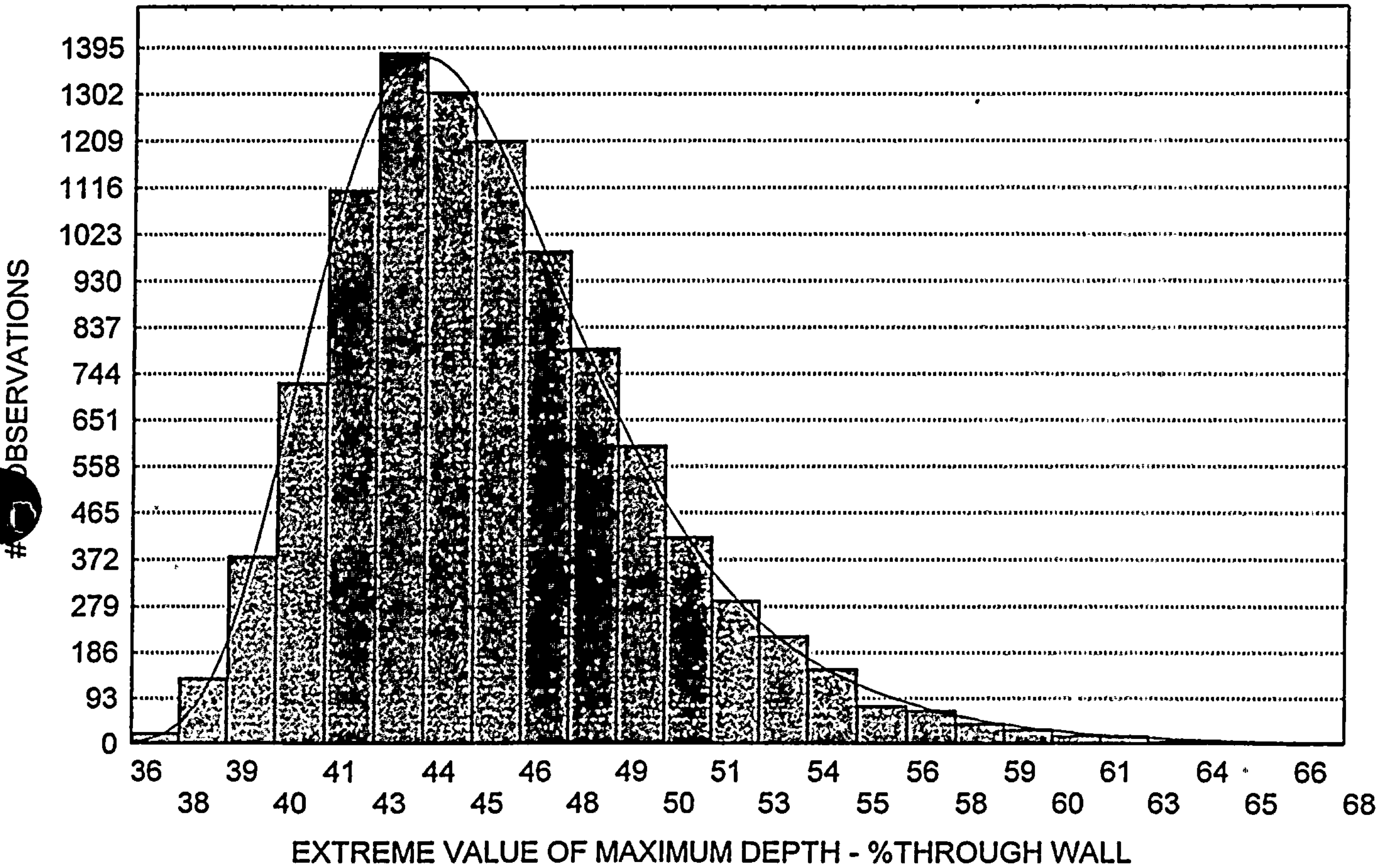


Figure 6.8 EXTREME VALUE DISTRIBUTION OF MAXIMUM DEPTH (U2R7 PROJECTION)

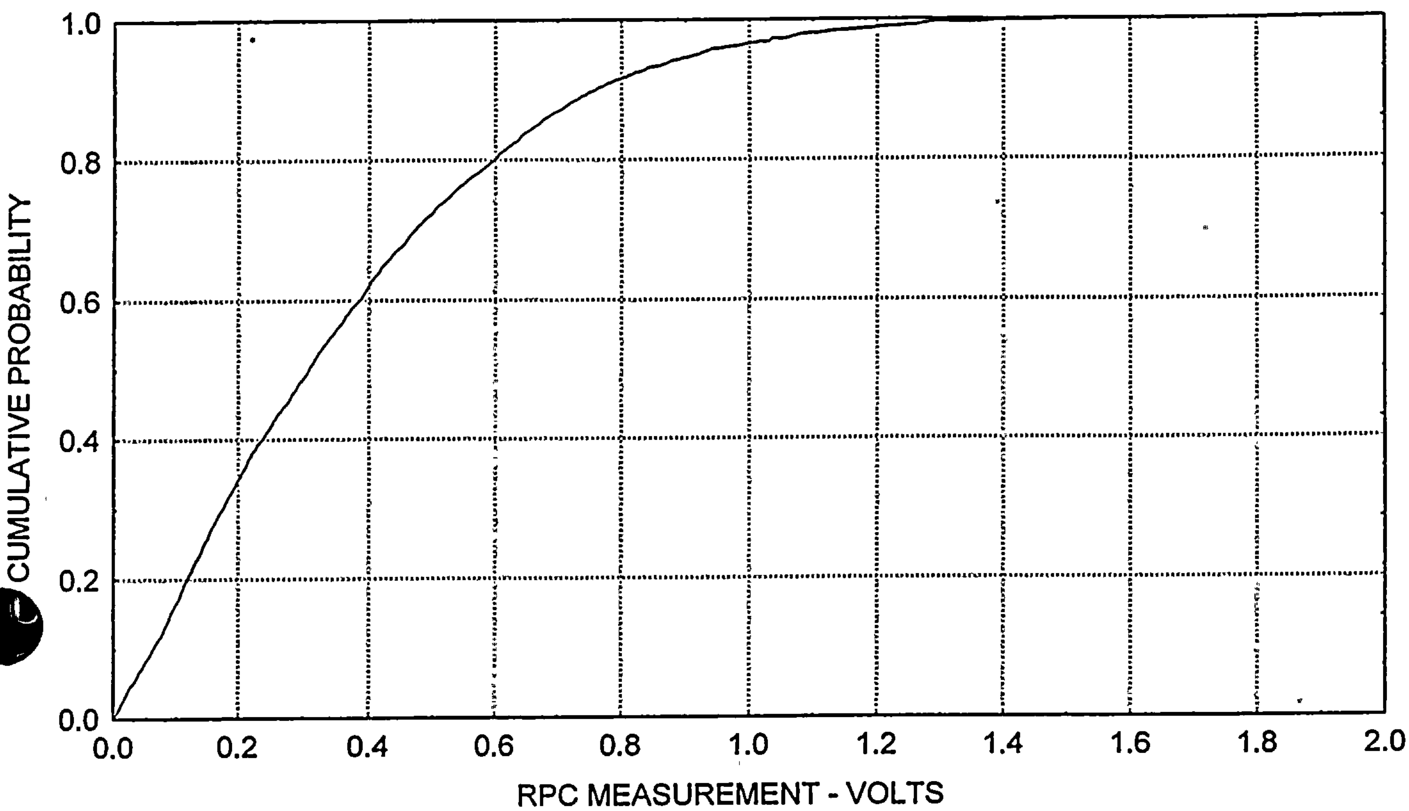


Figure 6.9 PREDICTION OF RPC VOLTAGE DISTRIBUTION PALO VERDE UNIT 2 EOC-7

TABLE 6.1

PALO VERDE CYCLE 7 SIMULATION: OPERATIONAL INPUT

INSPECTION/ OPERATING PERIOD	TIME [EFPY]	TEMPERATURE [°F]	POD FUNCTION	GROWTH RATE
R4 and M5-1	4.97	620[1]	RPC[2]	U2R5 Based
M5-2	5.46	613	RPC	U2R5 Based
R5	5.76	613	Plus Point	U2R5 Based
R6	6.76	613	Plus Point	U2R6 Specific
R7	8.14	613	Plus Point	U2R6 Specific

Notes:

1. 621°F through R4
   607°F R4-M5-1

2. Double inspection at 4.97 EFPY

TABLE 6.2

PALO VERDE CYCLE 7 SIMULATION: SUMMARY OF RESULTS

EVALUATION	UNIT 2 GROWTH			UNIT 3 GROWTH		
	# OBS[1]	Best Est.	U95	# OBS	Best Est.	U95
Structural Integrity						
SLB	0	$7x10^{-5}$	$3x10^{-4}$	0	$7x10^{-5}$	$3x10^{-4}$
3ΔP	0	$7x10^{-5}$	$3x10^{-4}$	0	$7x10^{-5}$	$3x10^{-4}$
Leakage	0	$3.5x10^{-5}$[2]	$4x10^{-4}$	0	$7x10^{-5}$	$3x10^{-4}$

Notes:
1. Number of occurrences in 10K simulations
2. 20K simulations

SECTION 7
SUMMARY AND CONCLUSION

The effect of upper bundle corrosion degradation on the expected performance of steam generator tubing in Palo Verde Unit 2 for the next cycle of operation was evaluated using a Monte Carlo simulation model. Unit 2 has experienced axial, upper bundle, corrosion degradation in the form of ODSCC/IGA for the past several cycles of operation. This degradation was conservatively modeled as the initiation and growth of single, planar, axial cracks. In the past, the simulation model has correctly or conservatively predicted end of cycle conditions in terns of the number and severity of cracks found in subsequent eddy current inspections.

Benchmarking of the model in terms of comparing predicted versus actually observed end of cycle conditions has been highly satisfactory. One powerful new benchmarking observation was provided by the inclusion of the Plus Point eddy current probe in the Cycle 5 inspection. The improved sensitivity of this probe led to an inspection transient in terms of a jump in the trend of observed indications versus time. Data from this inspection transient allowed an indirect computation of the probability of detection curve for the Plus Point probe. Recent pulled tube data from another plant allowed direct computation of the probability of detection curve for the Plus Point probe. The detection properties of the Plus Point probe inferred from the simulation model were in excellent agreement with those determined from pulled tube data. This is an excellent additional benchmark of the simulation model and is an excellent example of the power and utility of a well benchmarked simulation model.

The most recent eddy current inspection results, Cycle 6, are highly encouraging. The number and severity of eddy current indications was less adverse than expected. Analysis of the latest inspection results, with the aid of the simulation model, shows that the improved inspection results are due to decreased crack growth rates and the reduction of the severity of the beginning of cycle crack population. The application of remedial measures and the application of the highly sensitive eddy current probes has dramatically reduced the probability of axial corrosion degradation having any significant impact on the structural or leakage integrity of steam tubing in Cycle 7 for Palo Verde Unit 2.

After 16.5 months of operation, the projected end of cycle conditions lead to a calculated conditional probability of tube burst under postulated steam line break conditions of 3×10^{-4}. This is an upper 95% confidence estimate. The chance of developing any leaking crack during Cycle 7 is correspondingly remote at 4×10^{-5}. Similarly low probabilities are calculated for substantially longer periods of operation. Upper bundle corrosion degradation is not a limiting consideration for the Cycle 7 run time of Palo Verde Unit 2.

REFERENCES

1. Palo Verde Nuclear Generating Station Unit Two Steam Generator Tube Rupture Analysis Report, NRC Submittal by Arizona Public Service Co., Docket No. STN 50-529, July, 1993.

2. Hall, J.F., Fink, G.C., Magee, T.P., Molkenthin, J.P., and House, P.E., "Palo Verde-2 Steam Generator Tube Bend Region Examination and Metallurgical Evaluation", ABB Combustion Engineering Nuclear Operations, Report V-PENG-TR-005, January, 1995.

3. Sykes, L.P., Redmond, K.R., and Sherburne, P.A., "Examination of Steam Generator Tubes from Palo Verde Nuclear Generating Station", BWNT and Babcock & Wilcox Co., October, 1993.

4. Status of PVNG Steam Generator Activities, Arizona Public Service Co. Report, August, 1994.

5. Sweeney, K., "Palo Verde Nuclear Generating Station, Unit 2 Steam Generator Evaluation", Arizona Public Service Report, August, 1995.

6. Begley, J.A., Woodman, B.W., and Brown, S.D., "Palo Verde Unit-3 Steam Generator Tube Regulatory Analysis for Axial Cracking", Packer Engineering, Inc., Report B51909-R1-Rev. 1, July, 1994.

7. Woodman, B.W., Biffer, J.L., and Begley, J.A., "Palo Verde Unit 3 Run Time Analysis Regarding the Impact of Upper Bundle Corrosion Degradation During Cycle 6", Aptech Engineering Services, Inc., Report AES 96022658-1-4, Rev. 1, June, 1996.

8. Draft Regulatory Guide X.XX, Steam Generator Tube Integrity, 1996.

9. Begley, J.A., Woodman, B.W., Brose, W.R., and Brown, S.D., "An Analysis of ODSCC/IGA at Eggcrate Support Locations at Arkansas Nuclear One (ANO) Unit 1", Aptech Engineering Services, Inc., Report AES 95102556-1-1, Rev. 1, April, 1996.

10. Begley, J.A., Woodman, B.W., Brown, S.D., and Brose, W.R., "Analysis of ODSCC/IGA at Tubesheet and Tube Support Locations at St. Lucie Unit 1", Aptech Engineering Services, Inc., Report AES 96052749-1-1, October, 1996.

11. Cochet, B., "Assessment of the Integrity of Steam Generator Tubes - Burst Test Results - Validation of Rupture Criteria (FRAMATOME DATA)", Palo Alto, CA, Electric Power Research Institute, NP-6865-L, Vol. 1, June, 1991.-

12. Zahoor, A., Ductile Fracture Handbook, EPRI Report, NP-6301-D, January, 1991.

13. Begley, J.A., "Evaluation of Leak and Burst Characteristics of Roll Transitions Containing Primary Water Stress Corrosion Cracks", Westinghouse STC Report 92-8TE4-ODSCC-R1, SG-91-06-044, EPRI Research Project S406-7, February, 1991.

14. Preece, R.J., and McIntyre, P., "Burst Pressure and Leak Rate Data Analysis for Inconel 600 PWR Steam Generator Tubing with Through Wall Axial Cracks", CEGB Report RD/L/FTD/0063/M89, PWR/SGWG/P(89)542, March, 1989.

APPENDIX A

EFFECTS OF MEASUREMENT ERRORS AND PROBABILITY OF DETECTION ON ESTIMATED GROWTH RATES

A numerical study was conducted to demonstrate factors that affect the extraction of growth rate information from inspection results which contain significant measurement and detection uncertainties. Characteristic patterns that have been observed in field data are reproduced by specifying growth and initial depth distributions, simulating the addition of measurement error, and constructing the *perceived* growth versus *perceived* initial depth information. The significance of measurement error, the role of the probability of detection (POD), and effects due to changes in given growth and depth distribution parameters are discussed.

Perceived growth (ΔD) is defined as the difference between EOC and BOC inspection depth calls. The BOC depth call (D_I) represents a sum of the actual initial depth (D) and the initial measurement error (E_i). The EOC depth call is the sum of the actual initial depth (D), actual growth during the cycle (G), and final measurement error (E_f). For the purposes of the numerical study, the actual growth, depth, and error distributions are assigned, and 500 data points are drawn at random from each distribution. For each data set, perceived growth (ordinate) is plotted versus perceived initial depth (abscissa) as follows:

$$\text{ordinate: } \Delta D = (G + D + E_f) - (D + E_i) \quad \text{(A-1)}$$

$$\text{abscissa: } D_I = D + E_i \quad \text{(A-2)}$$

Unless otherwise noted, the growth distribution is assumed to be log-normal (percent through-wall growth is the natural logarithm of a normal distribution

with mean m_G = 1.6, standard deviation s_G = 0.6.) The BOC depth distribution is normal (mean depth m_G = 40%, s_D = 10%) and the measurement error distributions, independent of depth in this study, are normal with zero mean and 10% standard deviation. The perceived quantities (ΔD vs. D_i) and the actual quantities (G vs. D) using these nominal distribution parameters are plotted in Figures A-1 (a) and (b), respectively.

CORRELATION DUE TO MEASUREMENT ERROR

Because the initial error term, E_i, appears in the equations for both ordinate and abscissa, the resulting data plotted on these two axes is correlated by definition. This explains the apparent trend in the data that favors lower perceived growth rates at higher perceived depths. To further illustrate this point, the growth, depth, and final errors are constrained to be constant (D = 0.4, G = 0.05, E_f = 0.0,) leaving only variance due to the initial measurement error. Equations A-1 and A-2 are then rewritten:

$$\text{ordinate: } \Delta D = 0.05 - E_i \tag{A-3}$$

$$\text{abscissa: } D_i = 0.4 + E_i \tag{A-4}$$

Essentially, the above equations are parametric in E_i. Solving for and eliminating E_i produces the equation:

$$\Delta D = - D_i + 0.45, \tag{A-5}$$

which represents a line with slope of -1 upon which all points must lie for the given E_i distribution. Simulation results that demonstrate this effect are shown in Figure A-2. For full distributions of growth and depth, such as the nominal parameter case shown in Figure A-1, the data is scattered about a regression line with negative slope that is created by the correlation due to the initial error term.

POD EFFECTS

The probability of detecting a crack during an inspection is typically a function of the crack depth. The rotating pancake coil (RPC) and Plus Point probes are each capable of detecting very deep cracks a large percentage of the time, while shallow cracks are more difficult to detect. To incorporate probability of detection into this numerical study, it is assumed that BOC and EOC inspections are entirely independent; that is, a crack that is initially detected may go undetected during final inspection, and vice-versa. A random number is drawn from a uniform distribution between zero and one for each crack. This number is then compared to the value of the appropriate probability of detection function for the given crack depth and inspection technique; a random number less than the POD indicates detection.

The fact that cracks can go undetected during one inspection or the other creates limiting boundary lines in the perceived growth plots, as demonstrated in Figure A-3. Here, a constant POD = 0.6 (independent of depth) is used for illustration. The points clustered along the $D_i = 0$ vertical axis represent points that were missed during initial inspection, but detected by the final inspection. Note from Equation A-1 that the POD is applied independently to both parenthetical quantities that make up the perceived growth, ΔD. If the initial inspection misses the crack, the $D + E_i$ term is taken to be zero, leaving $G + D + E_f$ plotted on the vertical axis. This term can considerably overestimate the actual growth because it includes the initial depth term in addition to measurement error.

The second limiting boundary is defined by points that lie along a line starting at the origin with a slope of -1. These points represent cracks that were detected during initial inspection but not during final inspection. Again

considering Equation A-1, the G + D + E_f quantity is taken to be zero for undetected cracks during final inspection, leaving only -(D + E_i) on the ordinate, which is the opposite of the quantity plotted on the abscissa.

Figures A-4 (a) and (b) show perceived growth versus perceived BOC depth using probability of detection functions corresponding to the RPC and Plus Point techniques, respectively. These plots are compared to Figures A-5 and A-6, which display actual plant-specific data obtained using these techniques. Figures A-5 and A-6 are plotted in voltage units, which are, in the range of interest, proportional to depth to a first approximation. The qualitative similarities regarding the negatively sloped data trend (due to correlation via the initial measurement error) and limiting boundaries (due to probability of detection functions) are evident. Also evident in both the numerical and actual data is the fact that the Plus Point probe is more sensitive to smaller cracks, which implies that the corresponding probability of detection is higher for those cracks. Thus, percentage of data points lying along either of the limiting boundaries (caused by inspection misses) is smaller for the Plus Point technique.

INFLUENCE OF DISTRIBUTION PARAMETERS

A systematic parametric study was performed to consider the effect of growth and depth distribution characteristics other than those listed as nominal in the introduction. In general the results tend to confirm intuition, as depicted in Figure A-7. Varying the mean growth or mean initial depth shifts the location of the center of mass of the data points in the ΔD- D_i plane without much affect to the overall appearance (shape) of the data. Increasing the standard deviations of the distributions has the opposite effect – the mass of data points is stretched along one or the other axis, but the center of mass remains approximately stationary. However, for small

perturbations to the growth or depth variance, the relatively large scatter due to measurement error remains dominant and the effect may not be obvious.

PERCEIVED GROWTH DISTRIBUTIONS

A central issue is the approximation of the actual growth distribution from field data that includes measurement error. One technique that has been used in the past is to start with the perceived ΔD vs. D_i information and include only those indications that were detected during both initial and final inspections (that is, discard data that lies on either of the two limiting boundaries described above.) All perceived growths that are less than zero are assumed to be zero, and data with perceived initial depth less than 20% are discarded. This technique is applied to the numerical simulation results obtained above.

From Equation A-1, the perceived growth of indications that are detected during both inspections reduces to $\Delta D = G + E_f - E_i$. Figure A-8 shows the actual growth distribution, G, while Figure A-9 shows the sum of the growth and measurement error distributions, ΔD. Also plotted are the cumulative distribution functions for each case. The measurement error effectively stretches the distribution towards larger and smaller (negative) perceived growths. In Figure A-10, the extraction technique described above is applied, which results in a significant number of zero growths and a relatively wide distribution of larger growths. Finally, this reconstructed distribution is sampled at random four times to simulate four inspection cycles, and averaged. A new distribution is created based on 500 draws using this averaging method; the results are plotted in Figure A-11.

In addition to the fact that such resampling of growth rates does have physical merit, the mixing of a number of potential zero growths with larger

growths during the averaging procedure serves to reduce the variance of the approximating distribution so that it is more in line with the actual growth. In fact, in comparing the final growth approximation (Figure A-11) to the actual distribution (Figure A-8), it is noted that approximation is conservative, yet qualitatively similar. This has proven to be true for simulations involving various distribution parameters and limited population samples as well.

The preceding paragraphs deal directly with through wall depths, and illustrate the important role played by the probability of detection in the perceived change in the through-wall depth, or apparent growth. Where correlation data between depth and RPC coil voltage exists (e.g., Figure 5.6), a parallel procedure may be employed to demonstrate equivalent effects of measurement error and probability of detection upon observed changes in RPC voltage.

A regression analysis performed to relate actual through-wall depth (*D*) to voltage (*V*), assuming a linear relationship of the form $V = mD + b + E$. Here, *m* is the slope of the regression line, *b* is the intercept, and *E* is an error term that accounts for any number of potential sources of random deviation from the linear fit, such as variation in the regression parameters, measurement error, interpretation error, etc. Proceeding as before, a plot of the change in voltage over a cycle of operation versus the initial voltage reading includes the following terms:

ordinate: $\Delta V = [m(D+G) + b + E_f] - [mD + b + E_i]$

abscissa: $V_i = mD + b + E_i$

The initial depth (*D*), growth (*G*), and error terms (E_i and E_f) are considered to be distributed (stochastic) variables. Note that the form of these equations is identical to Equations A-1 and A-2, with the addition of a scale factor and offset due to the slope and intercept of the regression fit. Perhaps more

importantly, the *DV* quantity plotted on the vertical axis still includes two bracketed terms; the terms within the brackets represent voltage readings corresponding to the final and initial inspections. If either of the two inspections fails to detect the crack (due to the probability of detection function of the specific detection probe), the quantity within the corresponding bracket is taken to be zero.

Figure A-12 shows an example of the numerical simulation results using voltage as the parameter of interest. Data that fall on the vertical axis itself represent cracks that have gone undetected during initial inspection ($V_i = 0$), leaving the quantity $m(D+G) + b + E_f$ to be plotted vertically. Note that this quantity includes not only the true growth and error term, but initial depth as well, which inflates the apparent change in voltage. The correlation due to measurement error described earlier is also evident in Figure A-12, as seen in the downward trend in *DV* at higher initial voltages.

An alternative approach in translating between depth and voltage is to incorporate the estimation error into the slope of the regression fit. In other words, the depth-to-voltage transformation is achieved by a stochastic calibration factor with offset. This technique is employed in the main report, Section 5-4, based on physical arguments, to achieve a reasonable growth rate distribution. The quantities plotted in Figure A-13 are as follows:

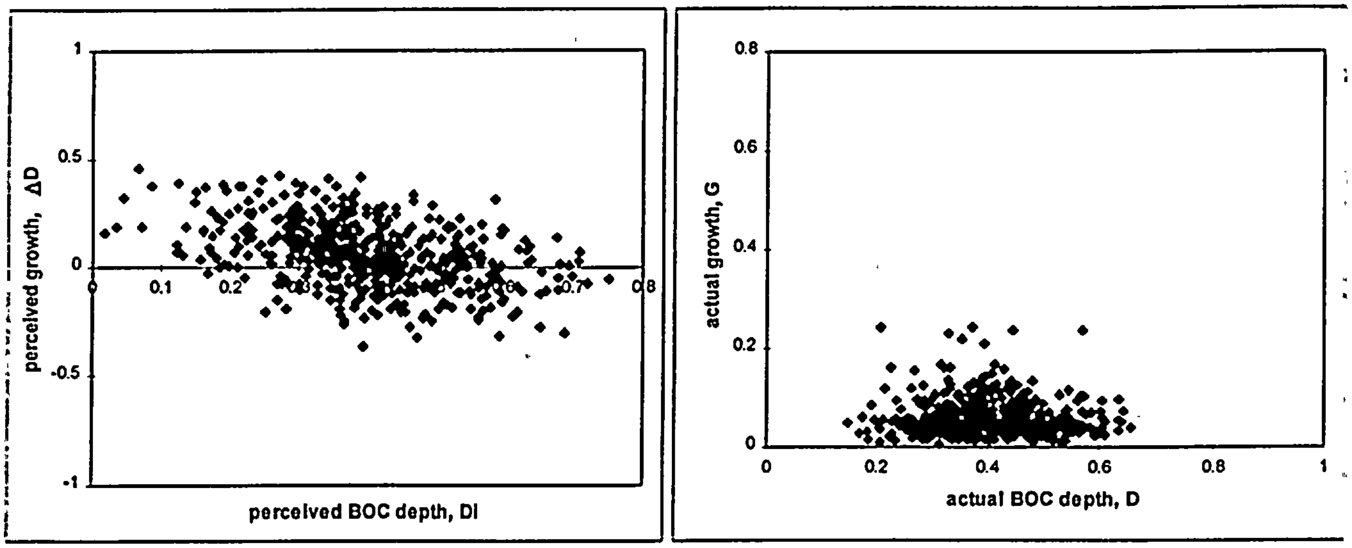
ordinate: $\Delta V = [\underline{m}(D+G)+b]-[\underline{m}D+b]$

abscissa: $V_i = \underline{m}D+b$

The slope, $\underline{m}$, is now a random variable with mean and standard deviation reflective of the mean slope and appropriate confidence interval for the slope of the original regression fit. An important point is that Figure A-13 retains the effects attributed to probability of detection, including an overestimation

of the growth rate indicated by the change in voltage along the vertical axis. Also noteworthy is that both techniques described above produce similar maximum *DV* readings for data that was successfully detected during both inspections.

The effects of detection and measurement errors upon perceived growth rates are similar whether expressed in terms of depths or voltages. The numerical studies of Appendix A support the use of only paired measurements. This approach was checked further by a comparison of Plus Point and pancake coil voltage changes. The better detection capability of the Plus Point resulted in only two unpaired voltage change data points in the examination of Unit 2, Cycle 5 and Cycle 6 inspection results. When voltage amplitudes between pancake and Plus Point probes are normalized, the maximum voltage changes in the Plus Point data match the maximum voltage changes of the paired pancake coil data. Hence, this serves as a confirmation that neglecting unpaired voltage data points, brought about principally by detection issues, is not unconservative.



(a) (b)

Figure A-1 GROWTH VS. INITIAL DEPTH FOR NOMINAL DISTRIBUTION PARAMETERS (a) APPARENT GROWTH AND DEPTH; (b) ACTUAL GROWTH AND DEPTH

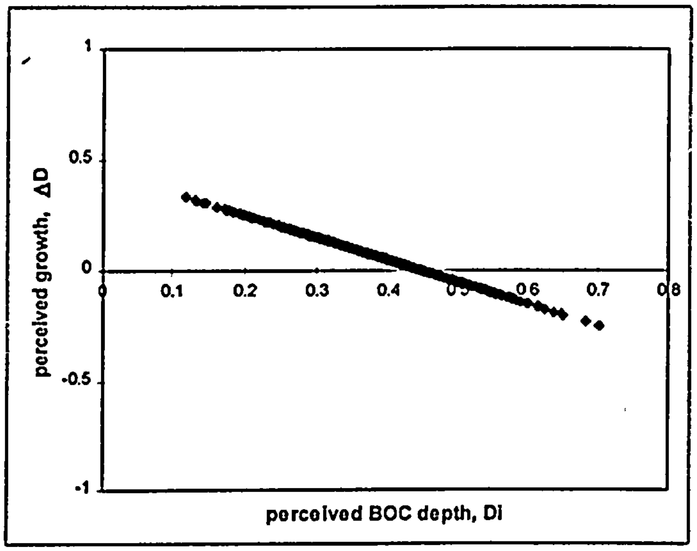


Figure A-2 DEMONSTRATION OF CORRELATION DUE TO MEASUREMENT ERROR

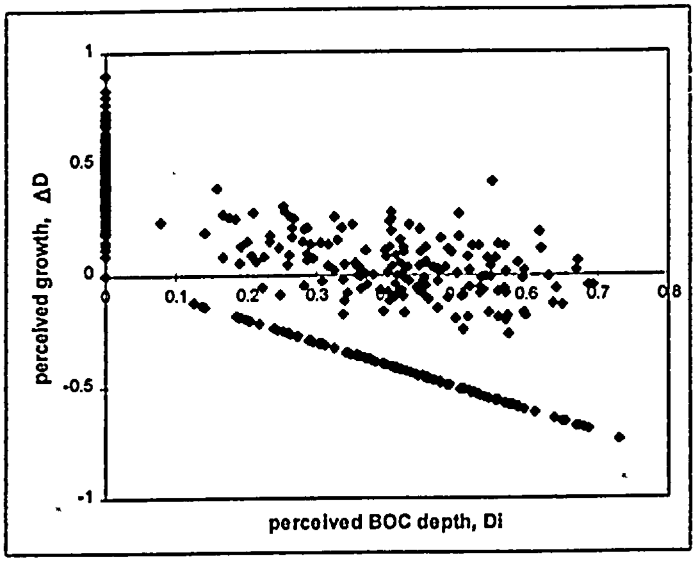


Figure A-3 DEMONSTRATION OF POD EFFECTS; POD = 0.6

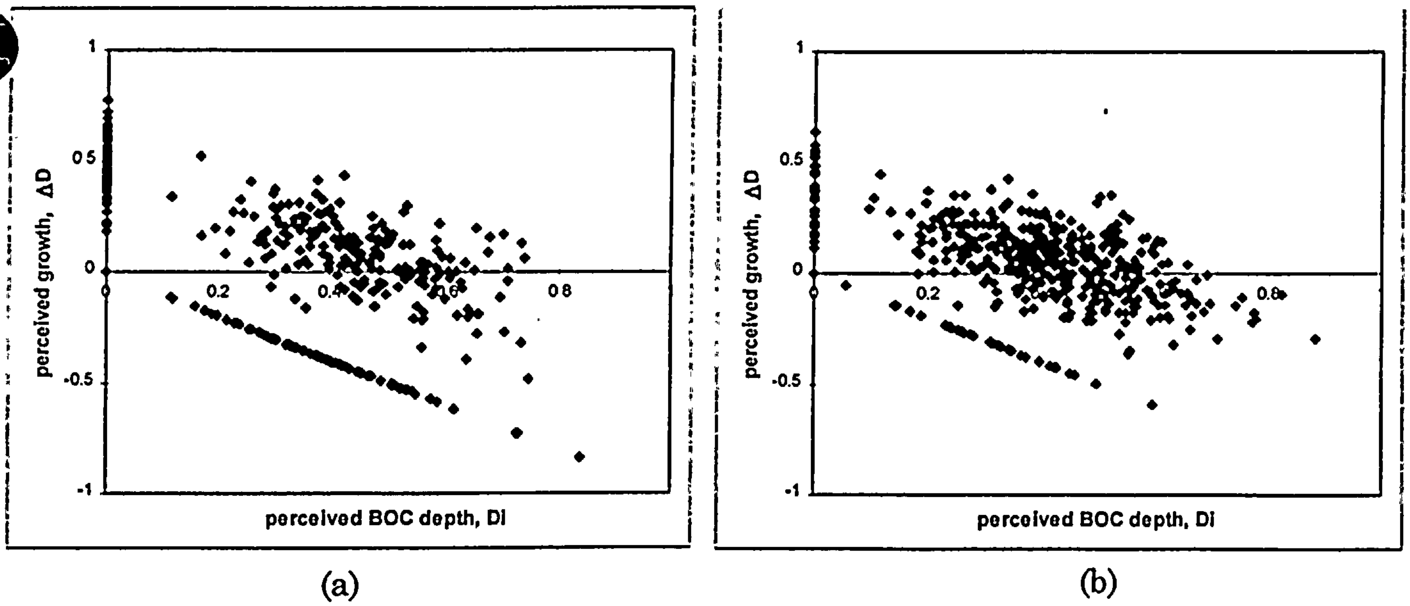


Figure A-4 EFFECT OF INSPECTION TECHNIQUE: (a) RPC; (b) PLUS POINT

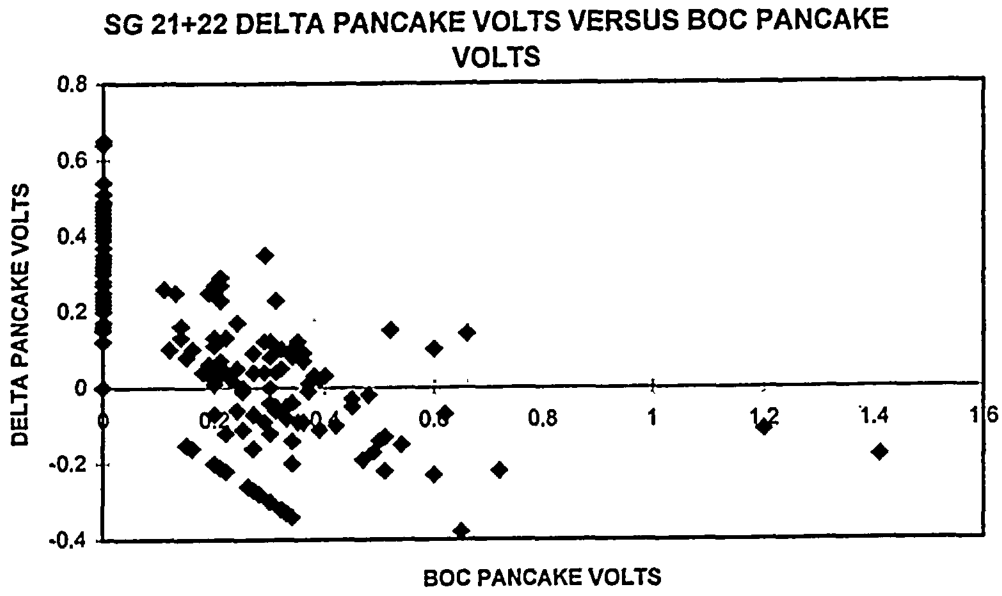


Figure A-5 ACTUAL RPC INSPECTION DATA

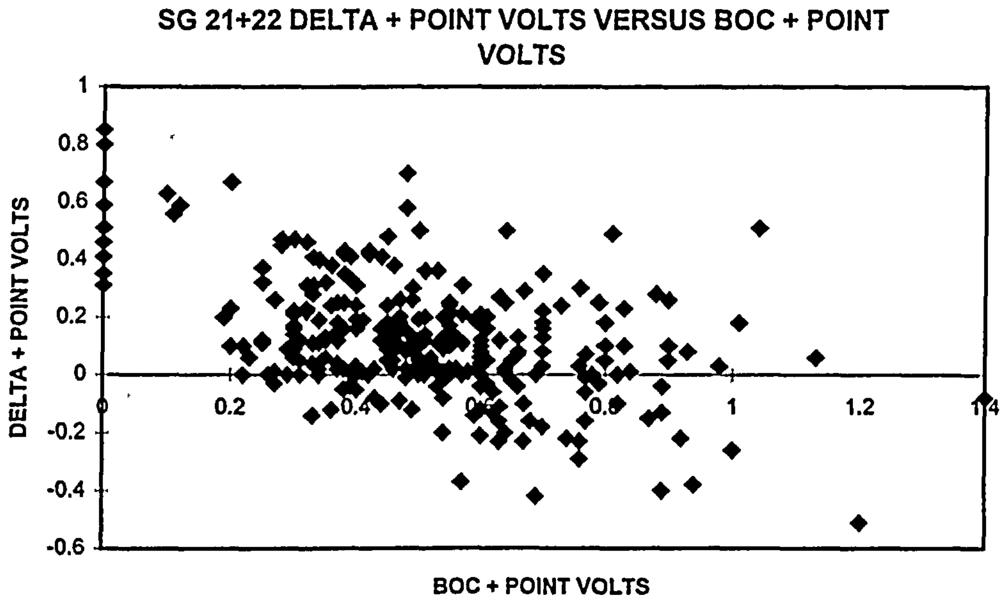


Figure A-6 ACTUAL PLUS POINT INSPECTION DATA

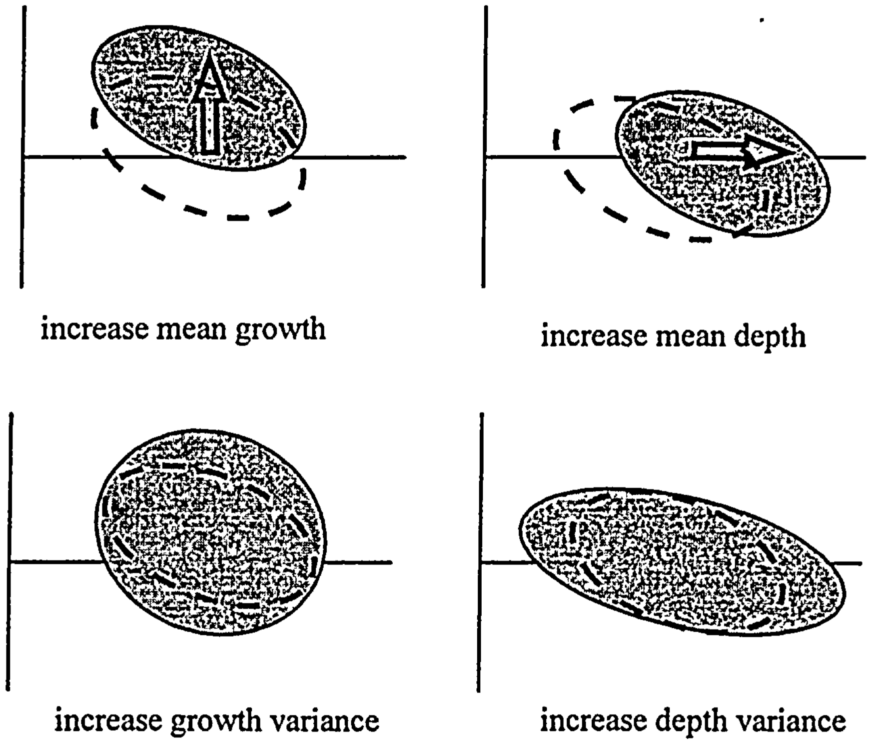


Figure A-7 ILLUSTRATION OF THE EFFECTS OF CHANGING GROWTH AND DEPTH DISTRIBUTION PARAMETERS

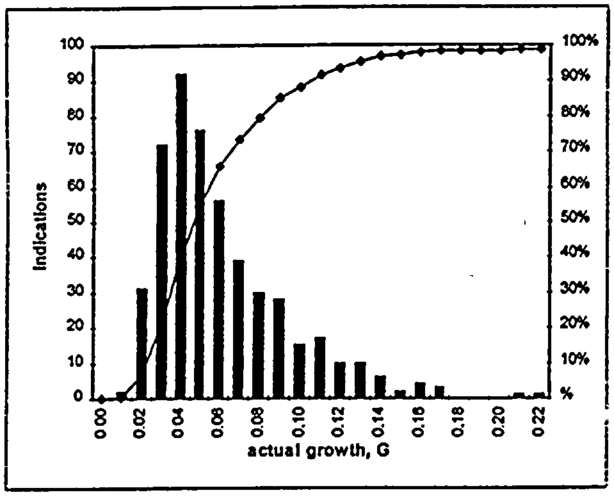


Figure A-8 ACTUAL GROWTH DISTRIBUTION

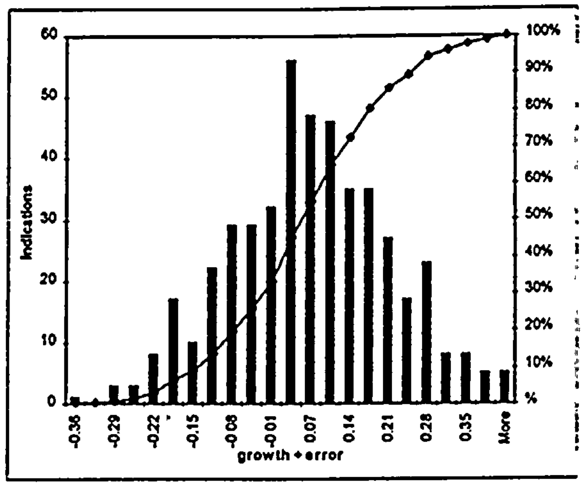


Figure A-9 GROWTH + ERROR DISTRIBUTION

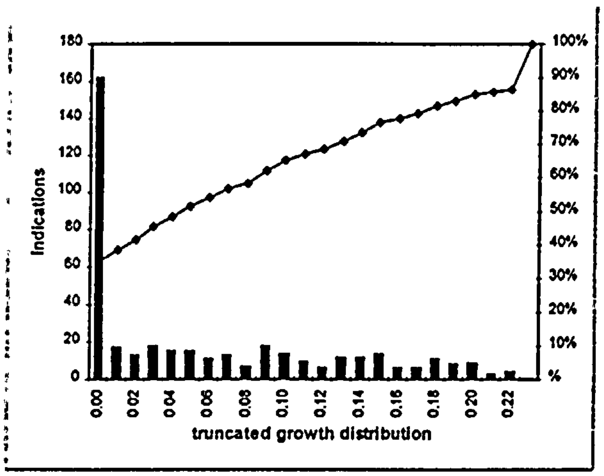


Figure A-10 TRUNCATED GROWTH DISTRIBUTION

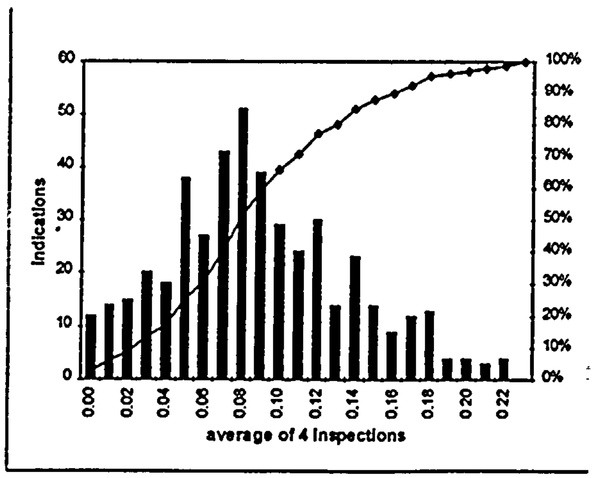


Figure A-11 AVERAGE GROWTH DISTRIBUTION AFTER 4 CYCLES

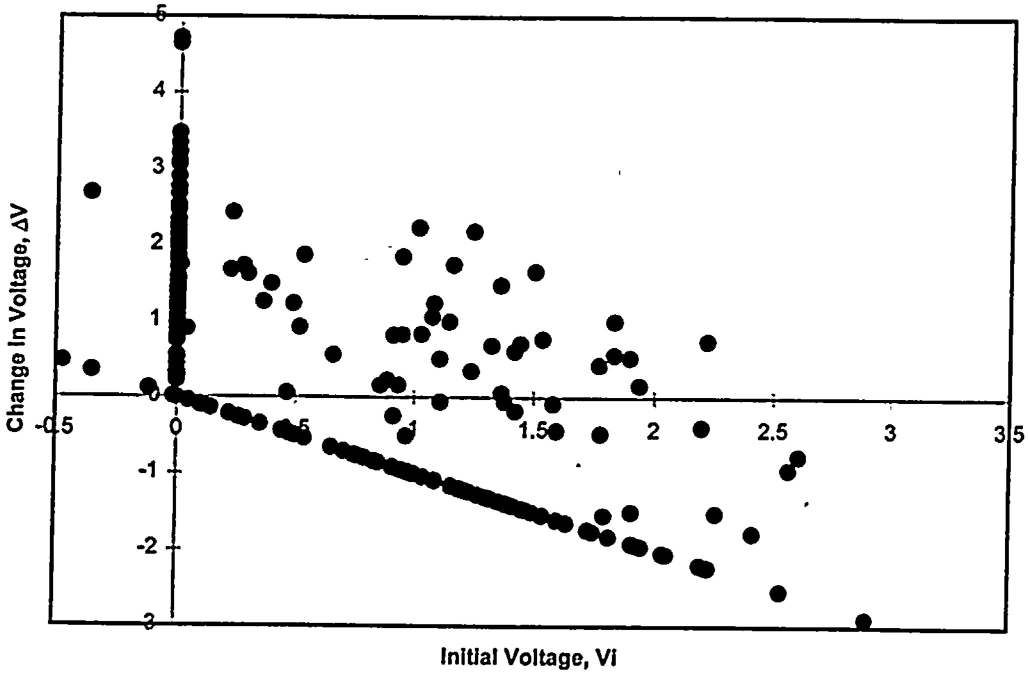


Figure A-12

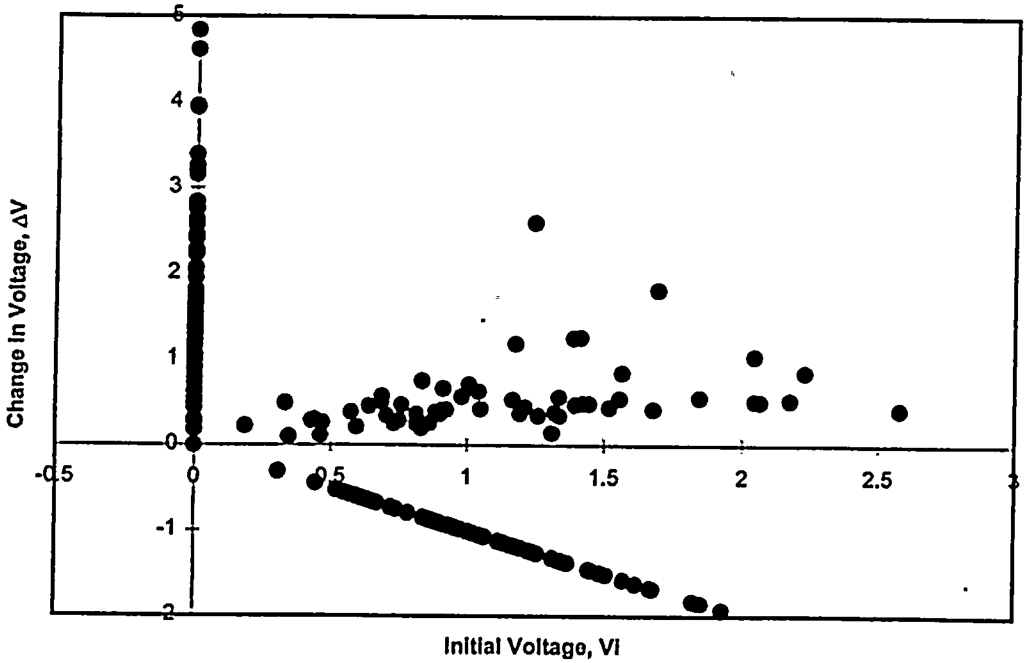


Figure A-13

APPENDIX B

CRACK MORPHOLOGY CONSIDERATIONS

Destructive examination of tubes pulled from PVNGS Unit 2 allowed construction of axial crack profiles. Using the Structural Minimum Method and the Framatome burst equation for axial partial through wall cracks, these crack profiles were evaluated to determine the section of the profile which determines the burst pressure. The length of the dominant section of a profile is termed the structurally significant length and the crack depth averaged over the structurally significant length is termed the structurally significant crack depth. Figure 2.5 shows a plot of maximum crack versus structurally significant average crack depth for tubes pulled from Unit 2.

The ratio of maximum crack depth to structurally significant depth, d_{max}/d_{ssa}, and the structurally significant length characterize the crack morphology when axial degradation is conservatively modeled as a single planar crack. The d_{max}/d_{ssa} ratio is termed the form factor. The relationship of the actual physical crack geometry to the form factor is not a straight forward geometric interpretation as used in the past. In general, the structurally significant crack depth of an idealized crack profile is not simply the average depth of the profile taken along the entire length of the profile. For example, Figure 2.1 illustrates the structurally significant length and depth of a triangular crack profile.

Planar crack profiles can be idealized as a family of curves of the form:

$$\left(\frac{x}{a}\right)^N + \left(\frac{y}{b}\right)^N = 1$$

Figure B-1 illustrates various profiles as N varies from 15 to 0.64. As N becomes very large, a rectangular profile is developed. For N equals 2, the figure is a semi-ellipse. A triangle is obtained by setting N equal to 1. If N is less than 1, a cusp-like figure is evident. The cusp-like figure is not as unrealistic as it first seems. The tails are not part of the structurally significant portion of the profile. The physical crack profile is better represented by truncating the tails past the structurally significant length. The physically unrealistic curvature of the cusp is eliminated by recalling that the smooth continuous profile is an idealization. A physical crack is composed of a number of coalesced crack segments. On a local scale, a realistic crack front curvature is maintained. However, on a macro scale, a central, very deep crack segment can easily create a cusp-like appearance. Furthermore, actual crack shapes are difficult to illustrate to scale. The lengths of interest are on the order of 1 inch while the maximum crack depth is the wall thickness of 0.042 inches. Figure B-2 illustrates a cusp-like profile drawn to scale with N equal to 0.88. Only the structurally significant portion of the profile is drawn. The contour is a reasonable idealization of actual measured crack profiles.

Analysis of pulled tube data from PVNGS Unit 2, yields the distribution of form factors shown in Figure B-3. When converted to physical crack shapes, the range of idealized physical morphologies is shown in Figure B-1. Recall that the tails of the triangular ad cusp-like figures are truncated at the structurally significant crack length. Figure B-4 is a plot of maximum crack depth versus crack depth averaged over the burst crack face length. In this case, a direct geometry conversion is more appropriate. The range of data is a good match to the idealized profiles in Figure B-1.

If N and the total profile length is held constant as the crack depth increases, the structurally significant crack length remains constant. When wall

penetration occurs, the structurally significant portion of the profile then changes. This change in structurally significant length after wall penetration must be considered in computing burst pressure. Appendix C describes the methodology used to evaluate the pressure bearing capacity of tubes with axial cracks having some through wall portion.

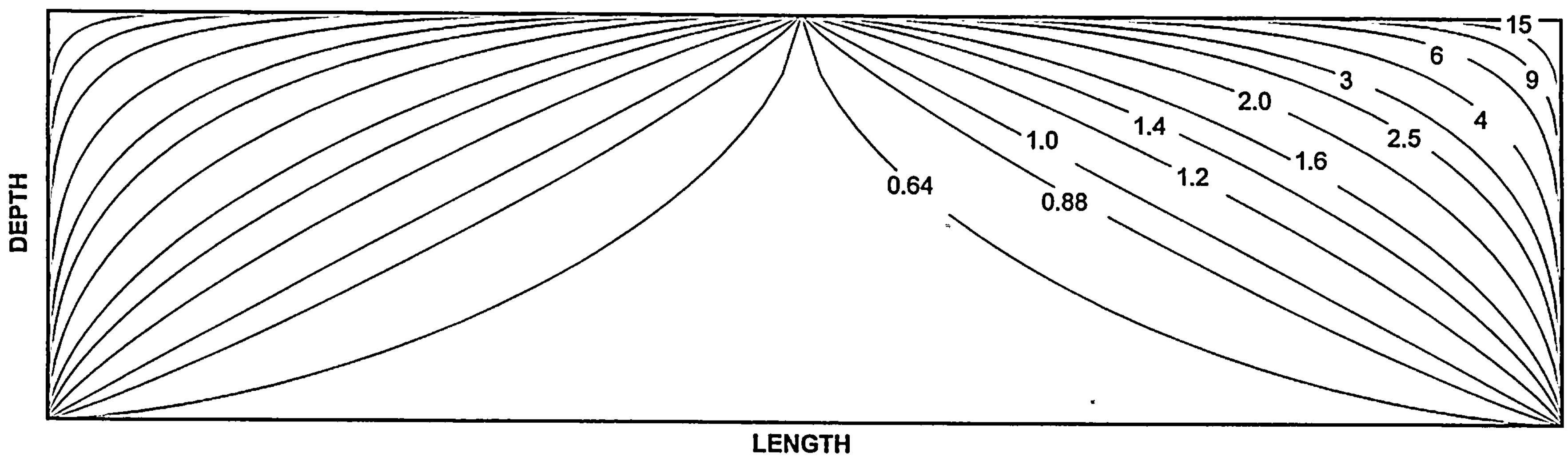


Figure B-1 SPECTRUM OF IDEALIZED, PHYSICAL, AXIAL CRACK PROFILES

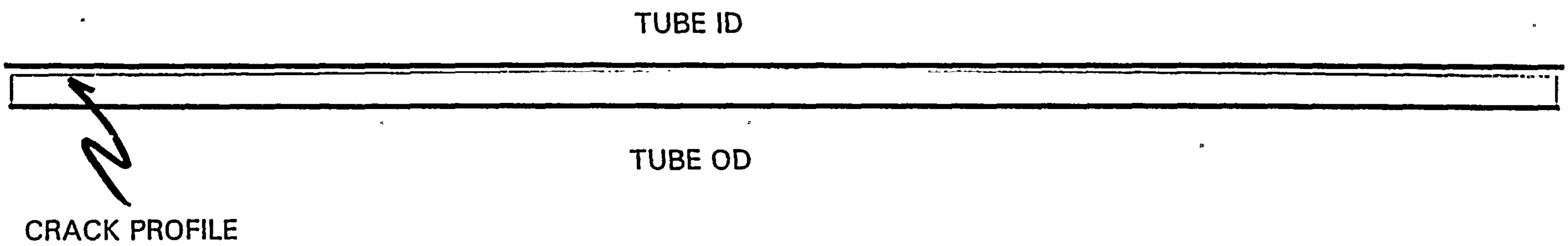


Figure B-2 CUSP-LIKE CRACK PROFILE DRAWN TO SCALE AND TRUNCATED AT THE STRUCTURALLY SIGNIFICANT CRACK LENGTH

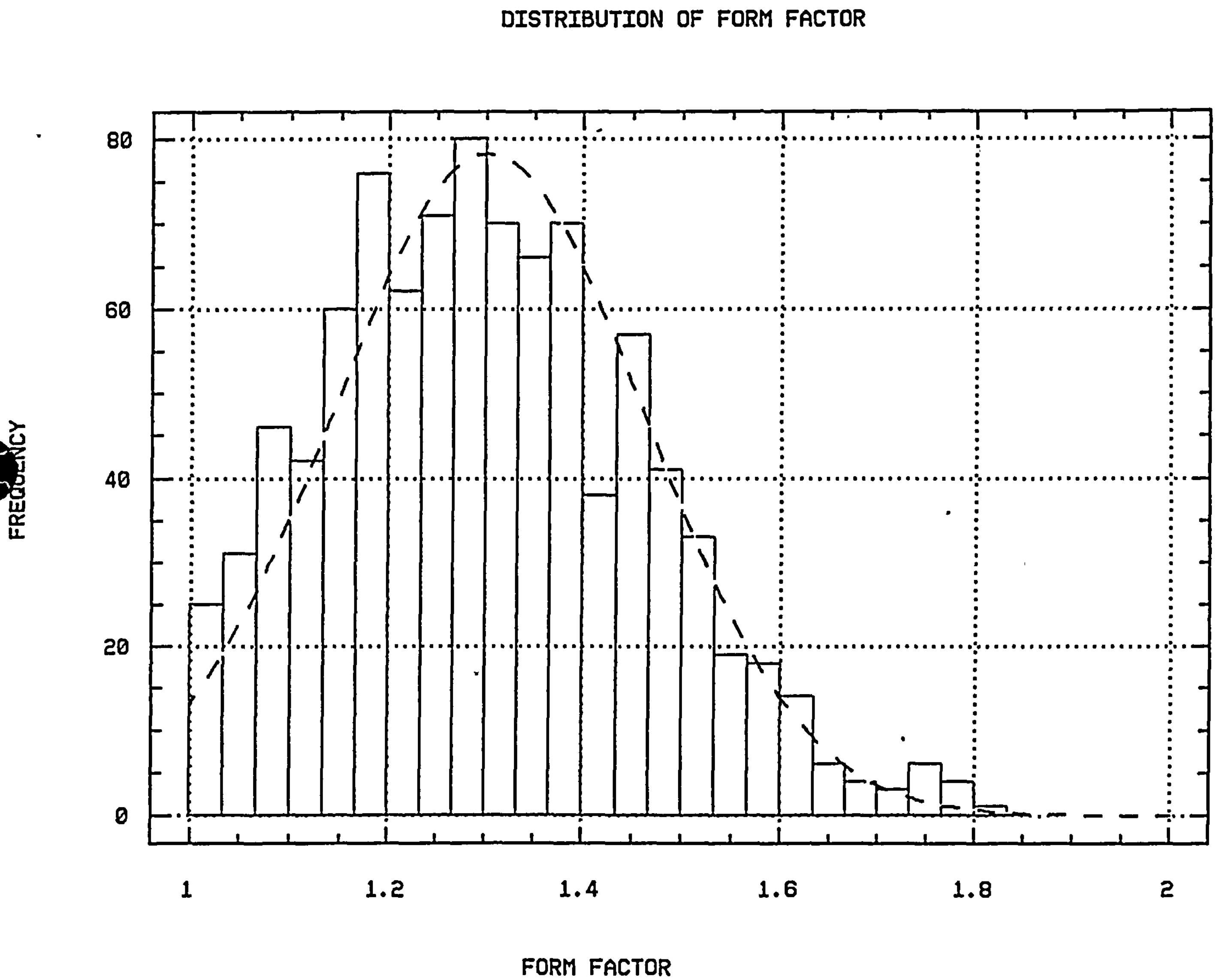


Figure B-3 DISTRIBUTION OF CRACK SHAPE FORM FACTORS, d_{max}/d_{ssa}, for PVNGS UNIT 2

MAXIMUM THROUGH WALL DEPTH VERSUS AVERAGE THROUGH WALL DEPTH

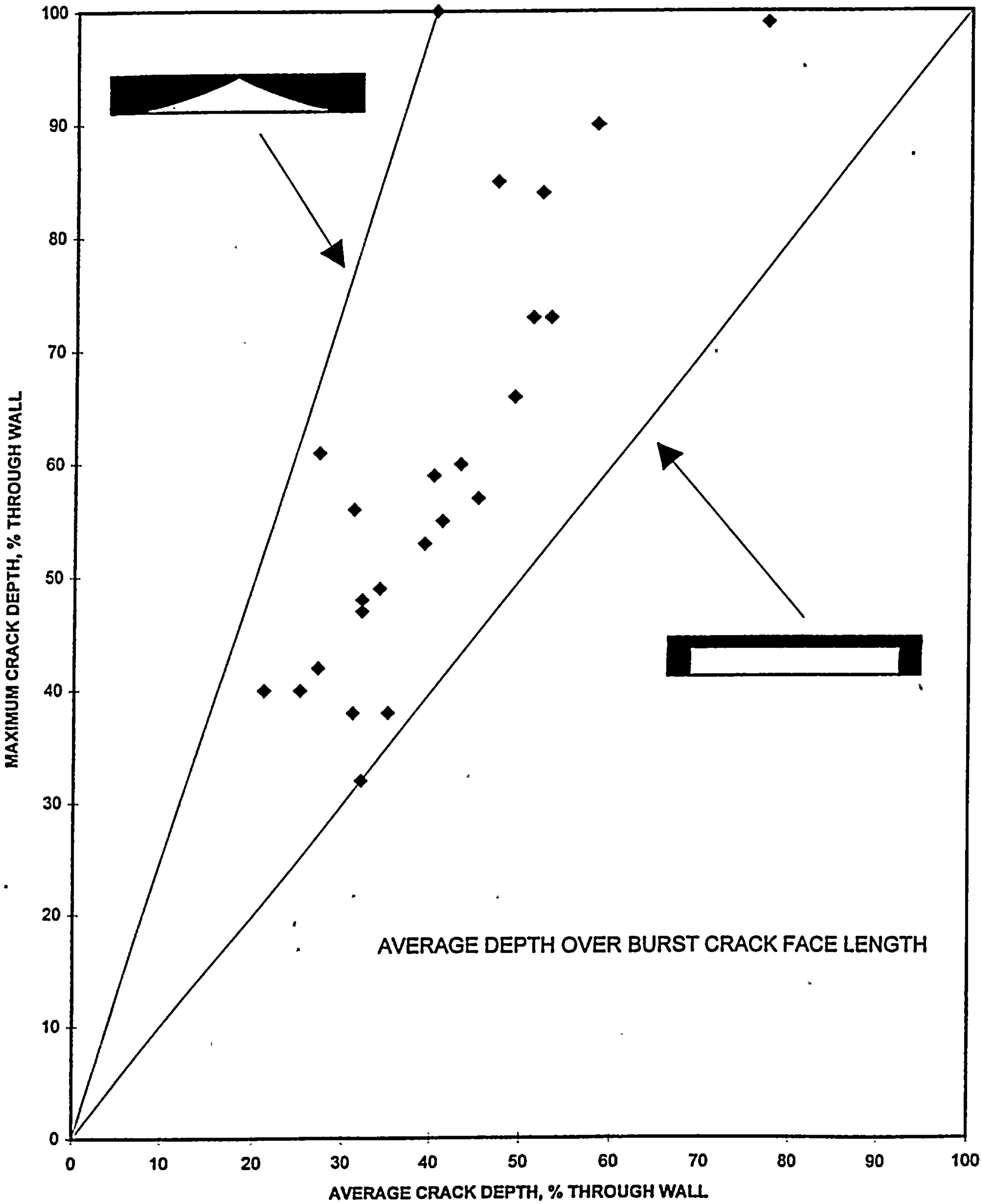


Figure B-4 MAXIMUM CRACK DEPTH VERSUS CRACK DEPTH AVERAGED OVER BURST FACE CRACK LENGTH FROM PVNGS UNIT 2 PULLED TUBES

APPENDIX C

BURST PROPERTIES OF CRACKED TUBES WITH THROUGH WALL SECTIONS

In general, plastic collapse considerations are sufficient to explain and predict the burst behavior of Alloy 600 steam generator tubing containing cracks. However, there are specific geometric regimes where the fracture properties of Alloy 600 must be considered. One geometry of practical interest is that of a very deep, rectangularly shaped crack. As illustrated in Figure C-1, there is a range of deep crack depths where the Framatome semi-empirical plastic collapse burst equation predicts burst for a partial through wall crack but at a pressure less than that required to propagate a through wall crack. The burst pressure of tubes with through wall cracks is very well defined by the most recent EPRI equation. The observed behavior in the regime where the partial through wall Framatome equation burst pressure is less than the through wall EPRI equation burst pressure is one of local instability. The crack "pops through" the wall as the ligament fractures but the resulting through wall crack length is not long enough to trigger a global burst instability.

In the above example, the fracture properties of the small ligament dictated a physical behavior, "pop through", but conditions leading to this behavior are predictable through reference to two plastic collapse approaches. Test results presented in Section 2 show that the Structural Minimum Method in conjunction with the Framatome equation can be used to predict local fracture instability, that is, ligament rupture, for arbitrarily shaped planar axial cracks in Alloy 600 tubing. For long (1.5 inches) deep cracks, particularly those with small tapered ligaments created by local wall penetration, the observed burst pressures are about one-half of those predicted by more conventional plastic collapse approaches. The Structural Minimum Method in

conjunction with the Framatome equation matches observed burst pressures. This match is a result of the fact that the Framatome equation is a markedly over conservative predictor of the burst pressure of through wall cracks. The end result is a much decreased emphasis on the strengthening effects of small ligaments beneath very deep cracks. This predicted decreased contribution of small ligaments happens to agree with the physical phenomena that small ligaments connect stretch and blunt very much before they fracture. That is, the fracture properties of the ligaments are sometimes the dominant consideration.

In the tests described in Section 2, local instability pressure, ligament tearing, coincided with the global instability pressure, full tube burst. Hence, the Structural Minimum Method with the Framatome equation predicted the observed burst pressure. Strictly speaking, this approach applies to the local instability of ligament tearing. Identifying local instability with the onset of a global instability is either correct or conservative. The onset of ligament tearing need not coincide with the onset of a full burst. A critical crack tip opening displacement criteria is nearing completion as a general burst criteria to deal with situations where ligament tearing is not the onset of a full burst. For the present, the onset of ligament tearing as predicted by the Structural Minimum Method and the Framatome equation is assumed to lead to a through wall crack with a total length equal to the structurally significant length of the original crack depth profile. The resultant through wall crack is assumed to burst if it exceeds the EPRI through wall crack burst equation.

There is an interesting subset of crack geometries where ligament tearing leads to a full burst. This subset is where crack growth due to corrosion degradation leads to a crack which penetrates the wall over some portion of the total crack length. Figure C-2 (a) illustrates growth of a cusp-like crack in the depth direction. The form factor and total length remain constant as

this crack grows in depth. This growth morphology leads to a constant structurally significant length, as shown, up to the point of wall penetration. After wall penetration, the form factor of the crack changes and the structurally significant length increases. See Figure C-2(b). Hence, when wall penetration occurs, the new crack shape must be reanalyzed to determine if the change in geometry will lead to burst. Burst may be possible before the critical through wall crack length is reached via continued corrosion degradation. The present simulation model considers bursting triggered by crack shape changes brought about by wall penetration.

LOCAL LIGAMENT BREAK THROUGH VS BURST REGIMES

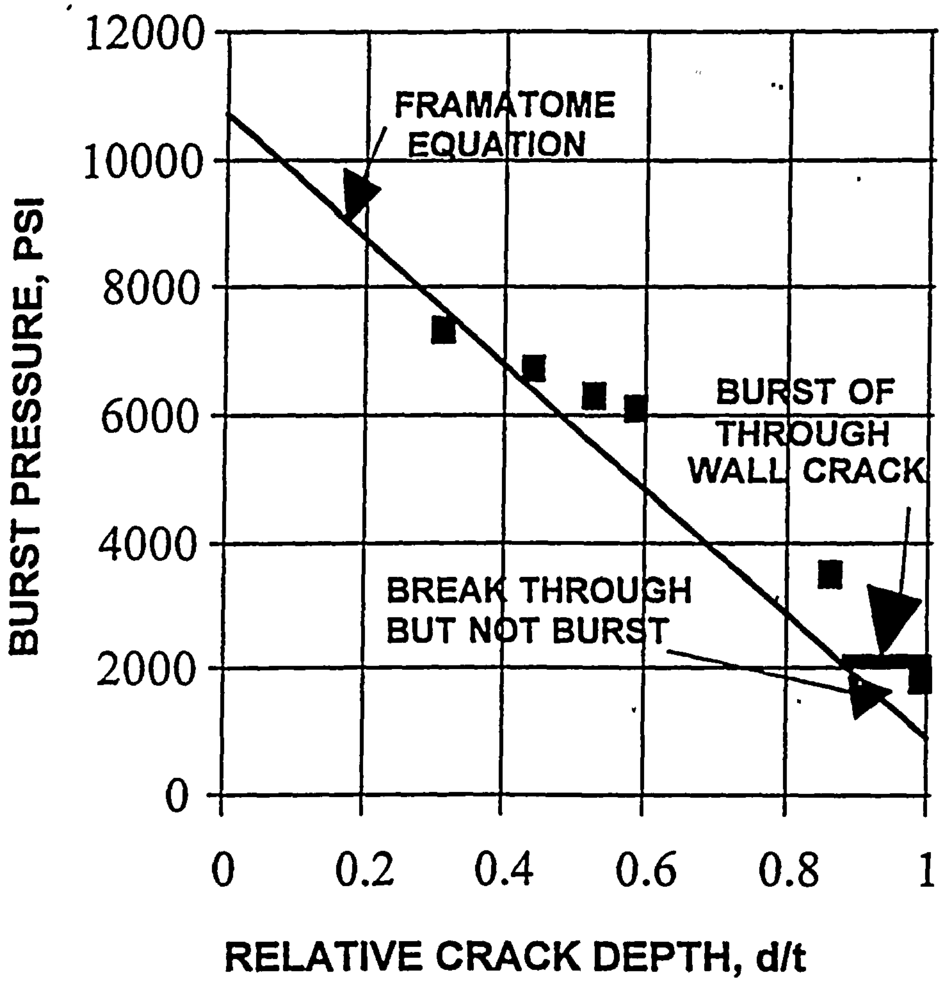


Figure C-1 ILLUSTRATION OF CRACK DEPTHS LEADING TO LOCAL INSTABILITY, "POP THROUGH", FOR A LONG RECTANGULARLY SHAPED CRACK

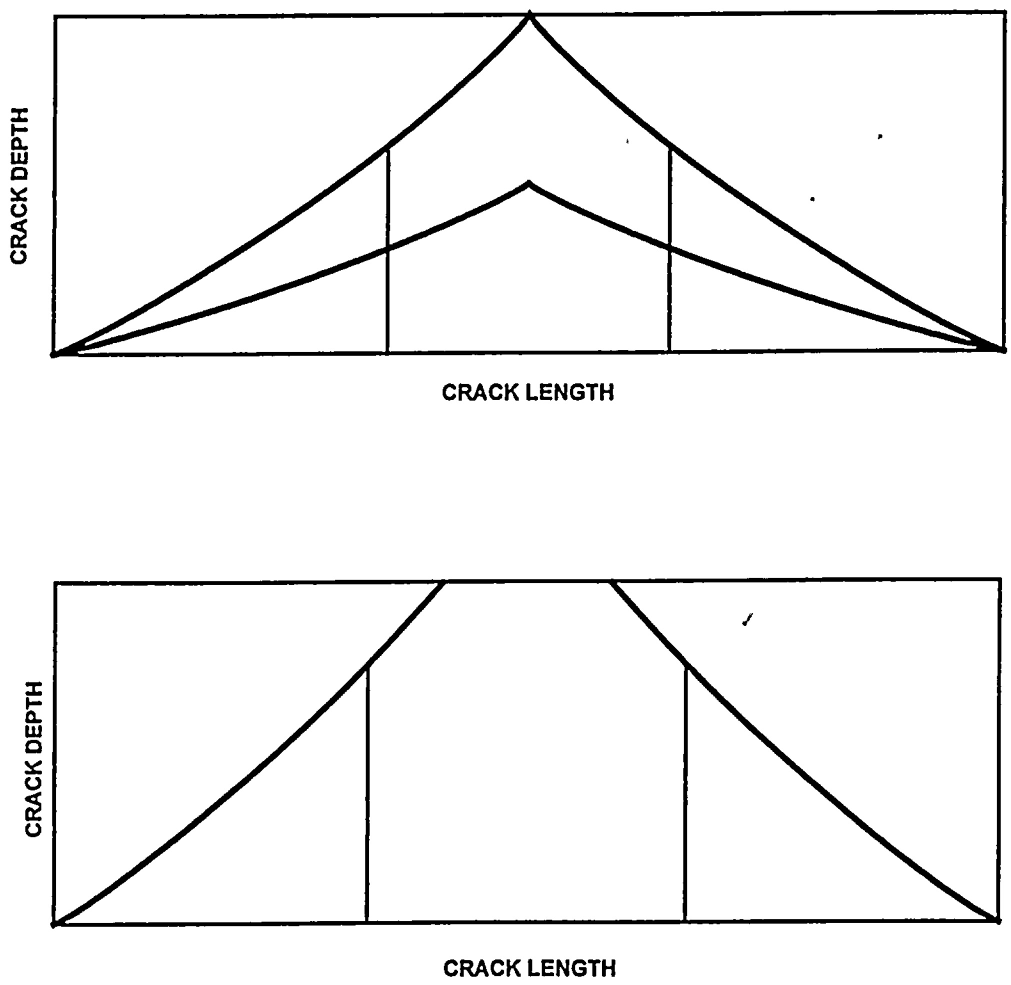


Figure C-2 CRACK GROWTH AND STRUCTURALLY SIGNIFICANT LENGTH CHANGES PRODUCED BY WALL PENETRATION

APPENDIX D

P. BESUNER

INDEPENDENT ASSESSMENT OF THE CHANCE OF CRACKS EXCEEDING THE LIMITS OF REGULATORY GUIDE 1.121 AND CAUSING TUBE BURST GIVEN A STEAM LINE BREAK

OBJECTIVE

The objective of this work was to update the "Regulatory Guideline 1.121 exceedance (RGE)" analysis in (1) for Palo Verde Unit 2 (PVNGS 2), steam generators (SGs) 21 and 22. This analysis was devoted to estimating an upper bound to the chance of freespan cracks large enough to violate USNRC Regulatory Guideline 1.121 at the next Unit 2 inspection. This next inspection, called U2R7 here, is currently scheduled in mid-September 1997, or 16.5 months after U2R6 which ended in early May 1996.

INSPECTION DATA

To update the Unit 2 RGE analysis, we used the latest inspection data from APS, as obtained from Sweeney (2) in a meeting of June 1996 at APTECH. These data cover Unit 2's sixth refueling cycle, U2R6.

METHOD

Methods and assumptions are very similar to those used in Refs. (1) and (3). The BALIFE code (4) was used here and has often been used to assess the probability of Regulatory Guide exceedances (RGEs) in the past. Based on the similarity of this analysis with previous work, only the new model improvements and assumptions developed for this report are summarized below.

BALIFE MODEL ASSUMPTIONS AND INPUT

We used the BALIFE Code, Version 5.051 as before with no changes. As known to APS (2), and summarized below, BALIFE (and all other forecasts) overpredicted the number of inspection indications found during U2R6, especially in SG 22. Of the many possible causes of this improving crack-initiation trend, two are considered most important:

- The worst part of the tube population is believed to be considerably smaller than the 2700 tubes per SG assumed here.
- The beneficial effects of chemical cleaning to inhibit new crack sites may be kicking in.

Neither of these effects are explicitly modelled in BALIFE. The new U2R6 data were simply added to the crack frequency database and BALIFE was run normally. BALIFE "compensates" for the smaller-than-expected U2R6 crack frequencies in the only way it can. It produces much lower Weibull shape parameters than before, especially for SG 22. Recall that, from APTECH's work on this and similar failure modes, industry data supports shape parameters ranging from 3.5 to 6.

- For SG 22, in (1) we previously calculated a shape parameter equal to 4.1 and this has now dropped dramatically to 2.65 to 2.8 This drop is because "only" 150 tubes were plugged in SG 22 during U2R6 for newly discovered freespan cracks, compared with about 350 predicted in (1). This brought the total SG 22 tubes plugged for these cracks to 953.
- For SG 21, in (1) we calculated a shape parameter equal to 4.5 and this has now dropped only to 4.0 This is because the SG 21 prediction in (1) was much closer to reality. 91 tubes were plugged in SG 21 during U2R6 for newly discovered freespan cracks, compared with about 120 predicted in (1). This brought the total SG 21 tubes plugged for these cracks to 280.

To explore these optimistic trends in the context of a possible "multiple-population model" more work was done. Trial and error runs with BALIFE (and some spreadsheet checks) revealed the following. If you assume that the at-risk population is only N=1100 tubes (rather than N=2700), the forecasts in (1) would have been accurate and " would be estimated as .4.5 for *both* SG 21 and SG 22. Obviously, if N=1100, it bodes well for SG 22. Less than 150 (1100-953) new tubes would *ever need plugging at SG 22 for new freespan crack initiations.*

INSPECTION MODEL ASSUMPTIONS AND INPUT

The BALIFE approach was supplemented to include an inspection/repair capability. The following new assumptions, as compared to (1), were used in the simulations.

Damage Classification for Each Tube

- At any stage in the life of an unplugged tube, its worst crack belongs to one of six damage categories (as added to five classifications in (1)):

  a. *Nonexistent and waiting to be initiated.* Conservatively, the chance of plugging such a tube is assumed to be zero (i.e., no false calls by the inspection).

  b. *Plus point-size.* These cracks are defined to be so small as to be possibly detected only by plus point probe. The probability of a miss is quantified below and defined as POM(pp)=1-POD(pp).

  c. *Bobbin-size.* These cracks are defined to be large enough to be possibly detected by *either* plus point *or* Bobbin or both. This probability of a miss by *both* inspections is quantified below and defined as POM(Bobbin)=1-POD(Bobbin).

  d. *RGE-size.* These cracks are defined to exceed the size defined in Regulatory Guide 1.121. For long cracks (2c>2") these cracks penetrate .64% through the thickness. They are large enough to

be possibly detected by *either* plus point *or* Bobbin or both. This probability of a miss by *both* inspections is quantified below and defined as POM(RGE) = 1-POD(RGE).

e. *Leak or burst given steam line break (SLB).* This is a new classification allowing us to compute the probability of such cracks in the next inspection interval. For 2" or longer cracks, we calculate for an SLB pressure of 2400 psi, these cracks penetrate .75% of the thickness. By definition, under SLB the crack will then grow through the tube wall.

f. *Leak or burst under normal under normal operation.* For 2" or longer cracks, we calculate for a normal pressure of 1267 psi, these cracks penetrate .88% of the thickness. By definition, the crack will then grow through the tube wall.

Note that the crack depths cited above for leak or burst are conservatively based on minimum tensile properties.

- As any tube ages, its worst crack moves from category "a" toward category "f" and plugging is the only way to halt this process.
- Any tube found to be damaged is immediately plugged.

Inspection Simulation

In (1) we used POD curves to set the following probabilities of missing (POM) various cracks.

- The inspection has a probability, POM(RGE) = 6% of missing a crack which exceeds Regulatory Guide 1.121 based on the POD curve.
- Each single inspection has a probability POM(Bobbin) = 15% assigned for missing a crack that is "normally" detectable by Bobbin inspection.
- Each single inspection has a probability POM(plus point) = 40% of missing damage that is normally detectable by plus point inspection. This 40% value is especially conservative in that it was originally used to quantify the chance of an *MRPC* probe of missing a "normally detectable indication."
- The last three assumptions are combined as a best-estimate *initial*

baseline inspection model, and labeled as "6-15-40."

As stated, the above initial POM values do *not* account for plus-point inspection probe improvements. So we built a new capability to allow POM values to be lowered during the last two inspections (U2R5 and U2R6). New POM values were estimated through trial-and-error runs to conservatively match the history of discovered crack frequencies of various sizes (b, c, and d as defined above). We obtained POM values of "2-4-15" for the latest inspection combinations of plus point and bobbin.

Crack Growth

The calculated average ratios for lives to various size cracks are taken to be equal to those in SG 22, which has the biggest freespan cracking problem and by far the most plentiful data. These calculations are updated and given below for convenience and detailed in Refs. (1) and earlier work on SG 22. The following life ratios were expressed as multiples of lifetime to a Plus point-sized crack and calculated for Weibull slope of 5:

Average Age to Plus Point-Size Crack / 1

Average Age to Pancake-Size Crack

= 2.3114 EFPY/2.1376 EFPY = 1.081

(See page D-3 of (1) for these EFPY values)

Average Age to Bobbin-Size Crack = 3.6669/2.1376 = 1.715

Average Age to RGE-Violator Crack = 4.8569/2.1376 = 2.272

Average Age to Leak or Burst Tube under 2400 psi = 6.3052/2.1376 = 2.950

Average Age to Leak or Burst Tube under 1267 psi = 6.8357/2.1376 = 3.198

For any other Weibull slope, these ratios can be computed through the equation

$$\text{Ratio for Slope b} = (\text{Ratio for Slope 5})^{(5/b)}$$

To show how these ratios are used, consider an actual example from this report for the upper best estimate for SG 22. There, *ignoring the effects of inspection and tube plugging*, we estimated b=2.65 and

$$\text{Average Age to Plus point Crack} = 8.71 \text{ EFPY.}$$

Then, the

$$\text{Average Age to an RGE Crack} = 8.71 * 2.272^{(5/2.65)}$$

$$= 40.9 \text{ EFPY (again given no tube plugging)}$$

- Within any inspection interval, the average number of tubes which move from a damage category (e.g., MRPC size) to the next higher damage category (Bobbin size) is directly proportional to the number of tubes within the lower damage category at the start of the interval.

Unit 2 Forecast from Statistical Combination of SG 21 and SG 22 Forecasts

This analysis was done exactly as in (1), using the same formulas for combining SG best estimates and confidence bounds to get Unit-based estimates and bounds.

Future EFPY Buildup and Damage Exposure as a Function of T_{hot}

- *No credit* was taken for primary temperature reduction or chemical cleaning. Damage is assumed to correlate with EFPY through the Bayesian Weibull model. For T_{hot}=611F, EFPY and effective age are both assumed to increase with future calendar time at past rates. For T_{hot}=615F, effective age is assumed to increase with future calendar time at 1.17 times past rates.

RESULTS

Fifty four different cases were analyzed and the estimates are summarized in Tables 1 through 3. Based on the results in Table 1, it is calculated for T_{hot}=615F that there is a 2.2% to 3.8% chance of developing an RGE in Unit 2 for a 16.5-month operating interval following U2R5. The confidence level associated with the 3.8% upper bound forecast is 90%. The best-estimate forecast shows a 2.9% chance of one or more RGE in Unit 2. SG 21 and 22 contribute about the same to this total, at 1.3% and 1.6% respectively. This is despite the larger number of past cracks suffered by SG 22. Two reasons that SG 21 contributes equally are

- the higher " values associated with its more quickly accelerating recent cracking frequency, and
- to a lesser extent, the fact that there are more unplugged tubes currently at risk in SG 21 than in SG 22.

Table 1 also shows the chance of tube burst *given an SLB*, computed from this conservative model. For T_{hot}=615F in Unit 2, if an SLB occurs, the chance of a tube burst from a freespan crack is 1.0 to 3.2 per ten thousand, with a best estimate of 1.7 chances in ten thousand.

Finally Table 1 shows the modest .20% to 50% increase of RGE and tube burst probabilities expected from a change of T_{hot} from 611F to 615F. Future SG 21 crack frequencies depend more strongly on T_{hot} than do SG 22 frequencies, again mainly due to the higher " values calculated for SG 21. Tube burst probabilities are a little more sensitive to T_{hot} then RGE ones, due mainly to the higher growth rates associated with larger crack sizes.

Similar calculations were made by APTECH for both steam generators in Unit 2 for inspection intervals of 14.5 months (Table 2) and 18.5 months (Table 3). The chances of RGE (and larger) cracks increase with inspection interval a little faster than linearly.

REFERENCES

1. Besuner, P. M., Engineering Reliability Forecasts for Tube Plugging at Palo Verde, Units 1, 2, and 3," AES 95072376-1, October 1995.

2. Sweeney, K., Personal Written and Verbal Communications with P. Besuner delivered in our meeting of June 1996.

3. PVNGS Unit 3 Steam Generator Evaluation, May 1995.

4. Besuner, P.M., BALIFE Version 5.051 (June 1995).

APPENDIX B

Unit 2 Cycle 6
ECT Voltage Growth
Summary

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
101	34	SAI	BW1 + 3.43	0.8	0.6	0.28	0.71	0.4	0.47	0.51	-0.19	0.2
107	38	SAI	BW1 + 7.24	1	0.8	0.44	0.85	1	0.21	0.44	0.23	0.41
109	38	MAI	VS2 + 0.55	0.6	0.3	0.23	0.62	0	0	0.66	0.23	-0.04
109	38	MAI	BW1 + 7.22	1.6	0.6	0.48	0.73	0.7	0.21	0.47	0.27	0.26
98	39	SAI	BW1 + 3.63	1.1	1.1	0.43	0.84	1.1	0.34	0.42	0.09	0.42
120	39	SAI	BW1 + 17.09	0.7	0	0	0.24	0.3	0	0.27	0	-0.03
107	40	SAI	BW1 + 7.06	0.9	0.9	0.25	0.96	0.8	0.23	0.67	0.02	0.29
120	41	MAI	BW1 + 13.83	0.3	0.1	0.24	0.58	0.2	0	0.52	0.24	0.06
120	41	MAI	BW1 + 16.45	0.5	0.1	0.21	0.78	0.7	0	0.59	0.21	0.19
117	42	SAI	BW1 + 10.20	1.7	0	0	0.62	1.3	0	0.49	0	0.13
128	43	SAI	BW1 + 19.72	1.2	0	0	0.78	1	0	0.78	0	0
130	43	SAI	VS1 + 0.70	0.5	0	0	0.57	0.4	0.3	0.57	-0.3	0
105	44	SAI	BW1 + 4.50	4	0.7	0.28	0.62	0.9	0.39	0.37	-0.11	0.25
119	44	SAI	BW1 + 9.32	2.3	0	0	0.67	1	0	0.54	0	0.13
127	46	SAI	BW1 + 12.43	1.7	0	0	0.69	1.2	0	1.2	0	-0.51
129	48	SAI	BW1 + 19.34	0.6	0	0	1.18	0.6	0	0.48	0	0.7
131	48	SAI	BW1 + 16.26	2.2	0	0	0.48	0.2	0	0.35	0	0.13
132	49	MAI	BW1 + 16.99	0.2	0	0	0.43	0.3	0	0.31	0	0.12
132	49	MAI	BW1 + 17.01	0.4	0	0	0.56	0.2	0	0.4	0	0.16
134	49	SAI	BW1 + 14.46	3.8	0	0	0.77	2.3	0	0.6	0	0.17
127	50	SAI	BW1 + 10.95	2.3	0	0	0.35	0	0	0	0	0.35
135	50	SAI	BW1 + 15.93	3.8	0	0	0.6	1.8	0	0.6	0	0
134	51	SAI	BW1 + 18.26	0.8	0.5	0.31	0.35	0.8	0.2	0.4	0.11	-0.05
127	52	SAI	BW1 + 9.79	1	0	0	0.34	1.6	0.32	0.44	-0.32	-0.1
128	53	MAI	09H + 0.18	0.4	0	0	0.28	0.3	0	0.27	0	0.01
128	53	MAI	BW1 + 11.62	1.2	0	0	0.22	1.2	0	0.22	0	0
101	54	SAI	BW1 + 19.50	0.8	0	0	0.37	0.3	0	0.25	0	0.12
99	56	SAI	BW1 + 4.25	0.9	0	0	0.71	0.7	0.3	0.4	-0.3	0.31
130	57	MAI	BW1 + 10.73	1	0.4	0.32	0.52	2.3	0	0.5	0.32	0.02
130	57	MAI	BW1 + 7.86	0.8	0.4	0.18	0.6	1	0.3	0.41	-0.12	0.19
129	58	SAI	BW1 + 7.11	1	0.2	0.11	0.2	0.8	0.27	0.57	-0.16	-0.37
125	66	SAI	VS1 - 0.75	0.5	0.3	0.67	1.05	0.3	0.52	0.7	0.15	0.35
137	70	SAI	BW1 + 5.45	1	0	0	0.56		0	0.48	0	0.08
149	70	SAI	BW1 + 1.56	1	0	0	0.59	0	0	0	0	0.59
147	74	SAI	BW1 + 2.93	1.9	0	0	0.69	1	0	0.69	0	0
148	75	SAI	BW1 + 3.06	1.7	0	0	0.57	0.9	0	0.55	0	0.02
150	77	SAI	BW1 + 13.74	2.6	1	0.26	0.4	2	0.22	0.4	0.04	0
148	77	SAI	BW1 + 16.75	2.5	1.5	0.38	0.55	2.2	0.3	0.37	0.08	0.18
136	77	SAI	BW1 + 5.92	1.5	0.4	0.64	0.85	1.4	0	0.8	0.64	0.05
148	79	SAI	09H + 26.99	0.2	0	0	0.89	0.2	0	0.53	0	0.36

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
138	81	SAI	BW1 + 4.55	0.5	0	0	0.55	0.5	0	0.53	0	0.02
149	84	SAI	BW1 + 19.58	4.5	2	0.26	0.69	3.8	0.22	0.69	0.04	0
140	87	SAI	BW1 + 9.50	3.5	2.3	0.55	1.06	2.4	0.62	0.83	-0.07	0.23
159	88	SAI	BW1 + 2.3	0.6	0.3	0.25	0.49	0.7	0.31	0.53	-0.06	-0.04
133	90	SAI	BW1 + 18.19	0.9	0	0	0.73	0.5	0	0.1	0	0.63
137	90	SAI	BW1 + 2.90	4.1	0.7	0.25	0.67	3	0	0.11	0.25	0.56
137	92	SAI	BW1 + 7.0	1.7	0.5	0.24	0.32	0.9	0	0.22	0.24	0.1
136	93	SAI	BW1 + 6.46	1.3	0	0	0.47	1.3	0	0.35	0	0.12
144	93	SAI	VS1 + 0.58	0.6	0	0	0.85	0.6	0	0.84	0	0.01
140	93	SAI	BW1 + 4.43	3.3	0	0	0.73	1.1	0	0.38	0	0.35
149	94	SAI	BW1 + 17.15	1.3	0	0	0.8	1.3	0.2	0.38	-0.2	0.42
150	95	SAI	VS1 + 1.15	0.3	0.3	0.7	1.55	0.2	0.6	1.04	0.1	0.51
133	96	MAI	BW1 + 3.12	0.3	0	0	0.33	0.4	0	0.38	0	-0.05
133	96	MAI	BW1 + 18.37	1.5	0	0	0.45	0.9	0.2	0.43	-0.2	0.02
134	97	SAI	BW1 + 3.75	0.8	0.6	0.8	1.04	0.4	0.66	0.79	0.14	0.25
137	98	SAI	BW1 + 6.33	2.9	0	0	0.79	2	0	0.55	0	0.24
143	98	SAI	BW1 + 14.03	2.4	0	0	0.71	1.3	0.28	0.77	-0.28	-0.06
149	100	SAI	BW1 + 16.54	1.6	0	0	0.41	1.5	0.22	0.4	-0.22	0.01
142	103	SAI	BW1 + 8.00	0.6	0	0	0.71	0.5	0.15	0.12	-0.15	0.59
146	105	SAI	09H + 32.09	3	0	0	0.69	2.5	0.32	0.56	-0.32	0.13
149	106	SAI	07H + 0.91	0.5	0.4	0.63	1.07				0.63	1.07
147	108	SAI	BW1 + 2.01	1.4	0.3	0.35	0.54	0.4	0.22	0.53	0.13	0.01
149	110	SAI	BW1 + 15.22	1	0	0	0.19	1	0.2	0.33	-0.2	-0.14
138	111	SAI	BW1 + 2.31	0.7	0.5	0.44	0.52	0.7	0.19	0.3	0.25	0.22
151	114	SAI	BW1 + 15.43	1.1	0	0	0.9	0.9	0.2	0.8	-0.2	0.1
146	115	SAI	BW1 + 11.25	4	0	0	0.55	4	0	0.55	0	0
135	116	SAI	BW1 + 4.00	1.1	0	0	0.41	0.8	0	0.3	0	0.11
145	116	SAI	BW1 + 18.18	0.2	0.2	0.37	0.47	0.2	0.11	0.3	0.26	0.17
149	116	SAI	BW1 + 16.72	2.1	0.7	0.25	0.56	1.8	0.25	0.5	0	0.06
132	117	MAI	BW1 + 3.41	0.4	0	0	0.38	0.4	0	0.29	0	0.09
132	117	MAI	BW1 + 5.16	1.7	0	0	0.46	0.6	0	0.3	0	0.16
106	119	SAI	BW1 + 2.71	1.5	1.4	0.77	1.1				0.77	1.1
132	119	SAI	BW1 + 5.40	2.3	0	0	0.4	2	0	0.39	0	0.01
106	121	SAI	BW1 + 3.00	0.7	0.5	0.23	0.89				0.23	0.89
132	121	SAI	BW1 + 3.56	7.4	0	0	0.88	4.9	0	0.7	0	0.18
131	124	SAI	BW1 + 7.44	1.5	0	0	0.29	1.4	0	0.29	0	0
146	125	SAI	BW1 + 16.90	3	0.2	0.24	0.31	1.9	0	0.31	0.24	0
143	126	SAI	BW1 + 16.88	1.6	1	0.16	0.66	1	0	0.6	0.16	0.06
140	127	SAI	BW1 + 15.79	1.4	0	0	0.64	0.9	0	0.6	0	0.04
146	127	SAI	BW1 + 18.15	0.3	0	0	0.4	0.2	0	0.4	0	0

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
145	128	SAI	BW1 +20.15	0.8	0	0	0.47	0.6	0	0.3	0	0.17
147	128	SAI	BW1 +17.99	0.8	0.5	0.49	0.55	0.3	0	0.5	0.49	0.05
134	129	SAI	BW1 +13.21	0.8	0.4	0.35	0.57	0.4	0	0.25	0.35	0.32
126	129	SAI	09H -0.26	0.5	0.2	0.33	1.14	0.2	0.29	0.64	0.04	0.5
132	129	SAI	09H -1.02	0.9	0.2	0.42	1	0.6	0.3	0.9	0.12	0.1
144	129	SAI	BW1 +19.12	1.1	0.4	0.3	0.62	0.7	0.14	0.5	0.16	0.12
134	131	SAI	BW1 +14.13	2.9	0	0.17	0.59	2.7	0	0.58	0.17	0.01
122	133	SAI	BW1 +6.71	1.4	0	0	0.36	1.2	0	0.34	0	0.02
127	134	SAI	09H -0.36	1.3	0.4	0.22	0.63	1.1	0.12	0.45	0.1	0.18
118	135	SAI	08H -0.64	0.5	0.3	0.64	0.92	0.4	0.29	0.7	0.35	0.22
138	137	SAI	BW1 +17.23	1.1	0	0	0.72	1	0	0.6	0	0.12
126	137	SAI	BW1 +8.59	0.3	0	0	0.62	0.3	0	0.25	0	0.37
100	139	SAI	BW1 +5.05	1.8	0.7	0.33	1	1.2	0.2	0.5	0.13	0.5
129	140	SAI	BW1 +18.56	0.7	1.1	0.54	0.8	0.5	0	0.39	0.54	0.41
134	141	SAI	BW1 +20.01	0.6	0	0	0.47	0.5	0	0.3	0	0.17
125	142	SAI	BW1 +18.81	1	0.6	0.35	0.6	0.6	0	0.36	0.35	0.24
129	142	SAI	BW1 +17.78	2.2	1.4	0.45	0.53	1.4	0	0.27	0.45	0.26
114	143	SAI	BW1 +0.71	0.5	0.3	0.22	0.7	0.4	0	0.6	0.22	0.1
130	143	SAI	BW1 +19.26	1.7	1.1	0.27	0.26	1.3	0	0.26	0.27	0
127	144	SAI	BW1 +21.06	1.3	1.1	0.21	0.81	0.8	0.2	0.6	0.01	0.21
119	144	SAI	BW1 +17.74	1.3	0	0	0.42	1.2	0.34	0.41	-0.34	0.01
131	144	SAI	BW1 +20.28	1.5	1	0.38	0.5	1	0.13	0.5	0.25	0
120	145	SAI	BW1 +15.77	1.9	1.4	0.42	0.54	1.6	0.32	0.45	0.1	0.09
128	145	SAI	BW1 +19.42	1.1	0.6	0.3	0.41	1	0.3	0.4	0	0.01
100	149	SAI	BW1 +4.35	1.4	1	0.35	0.93	1.1	0.31	0.45	0.04	0.48

SG 2-2 ECT Summary Data

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
106	31	SAI	08H + 34.52	1	0	0	0.37	0.4	0	0.49	0	-0.12
109	38	MAI	BW1 + 7.67	1.3	0.5	0	0.39	0.2	0	0.3	0	0.09
108	39	SAI	BW1 + 8.39	0.8	0.7	0.13	0.44	1.1	0.2	0.3	-0.07	0.14
99	42	SAI	BW1 + 3.53	0.5	0.5	0.42	0.98	0.2	0	0.8	0.42	0.18
109	42	MAI	BW1 + 8.51	0.2	na	na	0.34	0.4	0.22	0.34		0
109	42	MAI	08H + 41.61	0.1	na	na	0.41	0	0	0		0.41
119	42	MAI	08H + 0.63	0.5	0.5	0.43	1.19	0.5	0.4	1.01	0.03	0.18
119	42	MAI	BW1 + 0.99	0.9	0	0	0.4	0.3	0	0.38	0	0.02
119	42	MAI	BW1 + 3.69	0.8	0	0	0.45	0.6	0	0.34	0	0.11
119	42	MAI	BW1 + 6.35	3.4	3	0.47	0.9	0.8	0.2	0.63	0.27	0.27
127	42	SAI	BW1 + 14.84	2	0	0	0.51	0.7	0	0.51	0	0
106	43	SAI	BW1 + 20.70	5	3.5	0.45	0.74	4.3	0.36	0.66	0.09	0.08
126	43	SAI	BW1 + 20.23	3.6	0	0	0.36	1.1	0	0.25	0	0.11
130	43	SAI	BW1 + 17.46	1.5	0	0	0.57	1.6	0	0.67	0	-0.1
120	43	SAI	BW1 + 4.62	4.6	0	0	0.58	4.1	0	0.61	0	-0.03
115	44	SAI	BW1 + 7.69	1.7	1.1	0.42	0.86	1.1	0.45	0.7	-0.03	0.16
83	44	SAI	BW1 + 0.27	1.2	1.2	0.51	0.67	0	0	0	0.51	0.67
125	44	SAI	BW1 + 5.18	5.2	0	0	0.51	1.1	0	0.46	0	0.05
84	45	SAI	BW1 + 2.20	0.4	0	0	0.56	na			0	0.56
105	46	SAI	BW1 + 14.00	1	0.6	0.33	0.56	0.4	0	0.62	0.33	-0.06
129	46	SAI	BW1 + 19.19	1.2	0	0	0.36	1.2	0	0.3	0	0.06
89	48	SAI	BW1 + 3.30	0.9	0.5	1.22	1.6	na			1.22	1.6
104	49	SAI	08H + 33.31	4.4	0.6	0.23	0.41	4	0.15	0.37	0.08	0.04
130	49	SAI	08H - 0.37	0.6	0.3	0.37	0.66	0.4	0.6	0.61	-0.23	0.05
87	50	SAI	BW1 + 3.50	1.3	0.6	0.25	0.91	na			0.25	0.91
133	50	SAI	BW1 + 15.43	2	1.2	0.32	0.7	1.5	0	0.92	0.32	-0.22
121	50	SAI	BW1 + 7.71	1.5	1.3	0.47	0.52	2.9	0	0.7	0.47	-0.18
110	51	SAI	BW1 + 18.31	2.6	0	0	0.72	0.4	0	0.55	0	0.17
132	51	SAI	BW1 + 14.59	1.2	0.8	0.39	0.49	2.7	0.54	0.89	-0.15	-0.4
131	52	SAI	BW1 + 11.64	3.8	0.8	0.41	0.66	3.8	0.39	0.64	0.02	0.02
83	52	SAI	BW1 + 1.46	0.5	0	0	0.68	na			0	0.68
132	53	SAI	BW1 + 12.21	1.8	0.6	0.46	0.73	0.5	0	0.28	0.46	0.45
95	54	SAI	BW1 + 4.20	0.8	0.6	0.48	0.68	0.8	0	0.6	0.48	0.08
125	54	SAI	BW1 + 8.36	0.9	0	0	0.52	1.5	0.33	0.74	-0.33	-0.22
131	54	SAI	BW1 + 10.90	1	0	0	0.27	0.9	0.22	0.69	-0.22	-0.42
134	55	SAI	BW1 + 13.23	2	0.5	0.43	0.74	0.9	0	0.33	0.43	0.41
135	56	SAI	08H - 0.51	1	0.6	0.45	0.81	0.6	0	0.61	0.45	0.2
129	56	SAI	BW1 + 4.97	1.5	0.4	0.22	0.77	0.5	0.18	0.55	0.04	0.22
131	58	MAI	09H - 0.73	0.6	0.6	0.4	0.67	0.5	0	0.47	0.4	0.2
131	58	MAI	BW1 + 6.84	3.2	0	0	0.56	3.6	0	0.94	0	-0.38

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
135	58	MAI	09H - 0.51	0.6	0.4	0.2	0.64	0.5	0	0.33	0.2	0.31
135	58	MAI	BW1 + 10.52	3.3	0.9	0.36	0.97	3.6	0.37	0.73	-0.01	0.24
135	60	SAI	BW1 + 10.06	1	0.8	0.2	0.74	0.7	0	0.54	0.2	0.2
132	61	SAI	BW1 + 4.73	2.3	1.5	0.38	0.58	2.1	0.37	0.56	0.01	0.02
144	61	MAI	BW1 + 15.68	3.3	0	0	0.63	1.7	0	0.6	0	0.03
144	61	MAI	BW1 + 15.82	2.9	0	0	0.41	1.6	0	0.4	0	0.01
133	62	SAI	BW1 + 6.11	1.5	0.4	0.22	0.64	"4-94" .3	"4-94" .5	na		
135	64	SAI	BW1 + 4.50	4.3	1	0.49	0.87	3.9	0	0.51	0.49	0.36
117	66	SAI	09H + 0.85	1	0.6	1.09	1.32	0.9	1.2	1.4	-0.11	-0.08
139	66	SAI	BW1 + 8.48	0.6	0.4	0.27	0.54	0.6	0	0.38	0.27	0.16
132	69	SAI	BW1 + 5.79	1.7	0	0	0.56	1.3	0	0.6	0	-0.04
134	69	MAI	BW1 + 0.21	0.9	0.3	0.26	0.6	0.7	0.16	0.46	0.1	0.14
134	69	MAI	BW1 + 17.61	1.9	0.4	0.27	0.55	0.8	0	0.54	0.27	0.01
142	69	MAI	09H - 0.43	0.4	0.4	0.46	0.73	0.4	0.48	0.7	-0.02	0.03
142	69	MAI	09H + 0.70	0.4	0.4	1.23	1.19	0.4	1.41	1.13	-0.18	0.06
144	71	SAI	BW1 + 17.37	2	0.7	0.2	0.74	1.9	0	1	0.2	-0.26
146	71	MAI	BW1 + 19.60	2	0	0	0.57	1	0	0.48	0	0.09
146	71	MAI	BW1 + 19.70	1.7	0	0	0.39	0.6	0	0.19	0	0.2
135	72	MAI	BW1 + 15.43	0.7	0.6	0.23	0.48	0.8	0	0.6	0.23	-0.12
135	72	MAI	BW1 + 6.00	2.2	1.4	0.27	0.44	0.4	0	0.64	0.27	-0.2
144	73	SAI	BW1 + 19.29	2.4	0.6	0.41	1.3	2.3	0.24	0.81	0.17	0.49
142	73	SAI	BW1 + 6.91	2.2	0.4	0.21	0.51	1	0	0.3	0.21	0.21
132	73	SAI	BW1 + 18.34	2.2	1.1	0.18	0.69	1.1	0.24	0.45	-0.06	0.24
131	74	SAI	BW1 + 8.11	0.3	0	0	0.51	0	0	0	0	0.51
137	78	SAI	BW1 + 9.00	0.5	0	0	0.6	0.7	0.21	0.5	-0.21	0.1
131	78	SAI	BW1 + 9.93	0.6	0	0	0.46	0	0	0	0	0.46
144	81	SAI	BW1 + 17.47	1	0.5	0.17	0.78	1.1	0	0.7	0.17	0.08
134	81	SAI	BW1 + 16.45	0.9	0	0	0.54	0.9	0	0.45	0	0.09
145	82	SAI	BW1 + 6.33	1.6	0.4	0.29	0.57	1	0.51	0.47	-0.22	0.1
131	86	SAI	BW1 + 6.69	0.6	0	0	0.5	0.7	0	0.5	0	0
146	87	SAI	BW1 + 18.50	2	0	0	0.43	0.5	0	0.4	0	0.03
136	87	SAI	BW1 + 17.30	1.1	1	0.5	0.63	1.7	0.21	0.65	0.29	-0.02
124	87	SAI	BW1 + 7.25	0.5	0	0	0.69	1.7	0	0.5	0	0.19
145	88	SAI	BW1 + 19.09	0.7	0	0	0.56	0.4	0	0.44	0	0.12
140	89	SAI	BW1 + 13.31	0.4	0	0	0.55	0.2	0	0.32	0	0.23
143	90	SAI	BW1 + 18.31	0.8	0.7	0.45	0.64	0.8	0.2	0.4	0.25	0.24
148	93	SAI	BW1 + 14.20	2.3	1.5	0.31	0.5	1.9	0	0.48	0.31	0.02
148	95	SAI	BW1 + 14.94	2.8	1.7	0.41	0.95	2.1	0.38	0.9	0.03	0.05
123	96	SAI	BW1 + 3.17	0.6	0	0	0.24	0.7	0.3	0.36	-0.3	-0.12
140	97	MAI	BW1 + 10.63	1.1	0.3	0.15	0.35	0.3	0	0.39	0.15	-0.04

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
140	97	MAI	BW1 + 7.50	1.1	0	0	0.63	1.6	0	0.38	0	0.25
148	97	SAI	BW1 + 18.00	1.5	0.6	0.45	0.59	1.5	0	0.46	0.45	0.13
126	99	SAI	BW1 + 2.40	0.4	0.4	0.24	0.61	0.6	0.25	0.33	-0.01	0.28
141	100	MAI	BW1 + 9.02	1.8	0.4	0.23	0.63	0.5	0	0.54	0.23	0.09
141	100	MAI	BW1 + 3.24	0.5	0	0	0.36	0.4	0	0.31	0	0.05
141	100	MAI	BW1 + 14.23	0.6	0	0	0.57	0.3	0	0.52	0	0.05
141	100	MAI	BW1 + 12.92	1	0	0	0.65	0.4	0	0.55	0	0.1
150	101	SAI	BW1 + 17.84	1.5	0	0	0.44	1.3	0.2	0.33	-0.2	0.11
126	101	SAI	BW1 + 2.51	0.7	0	0.25	0.53	0.9	0.33	0.51	-0.08	0.02
141	102	MAI	BW1 + 18.92	1.2	0.6	0.17	0.64	1.1	0	0.53	0.17	0.11
141	102	MAI	BW1 + 5.86	1.9	0	0	0.34	2.3	0	0.54	0	-0.2
149	102	SAI	BW1 + 16.35	2.5	1.2	0.28	0.65	0.9	0.33	0.64	-0.05	0.01
152	103	MAI	BW1 + 17.85	0.8	0.3	0.2	0.49	1.4	0	0.37	0.2	0.12
152	103	MAI	BW1 + 19.50	4.7	0.7	0.28	0.85	1.8	0	0.42	0.28	0.43
148	105	SAI	BW1 + 15.40	2	0	0	0.85	2.3		0.89	0	-0.04
141	106	SAI	BW1 + 7.37	1.7	0.7	0.25	0.68	1.5	0.21	0.57	0.04	0.11
147	108	MAI	BW1 + 7.05	0.5	0	0	0.29	0.3	0	0.23	0	0.06
147	108	MAI	BW1 + 16.26	1.9	0.7	0.36	0.52	0.9	0.27	0.37	0.09	0.15
151	108	SAI	BW1 + 16.00	1.9	0	0	0.37	0.6	0	0.33	0	0.04
144	111	SAI	BW1 + 6.85	1	0.4	0.41	0.47		0	0.76	0.41	-0.29
143	112	SAI	BW1 + 4.63	3	0	0	0.47	0.8	0	0.63	0	-0.16
148	113	SAI	09H + 33.62	1	0.2	0.28	0.84	0.8	0.21	0.46	0.07	0.38
136	115	SAI	BW1 + 5.40	1.9	1	0.56	0.72	1.8		0.82	0.56	-0.1
147	116	SAI	BW1 + 15.90	1.8	0.7	0.41	0.54	1.4	0.29	0.32	0.12	0.22
149	116	SAI	09H + 21.93	6.7	5.5	0.37	0.73	6.7	0	0.66	0.37	0.07
133	118	SAI	BW1 + 4.75	0.9	0	0	0.75	1.2		0.49	0	0.26
135	118	SAI	BW1 + 5.30	0.9	0.5	0.49	0.74	0.6	0	0.34	0.49	0.4
147	118	SAI	BW1 + 12.00	0.8	0.4	0.34	0.87	0.8	0	0.2	0.34	0.67
132	119	SAI	BW1 + 3.86	1.2	0.9	0.21	0.72	1.3	0	0.39	0.21	0.33
133	120	SAI	BW1 + 6.33	1	0.9	0.34	0.61	1.1	0	0.77	0.34	-0.16
135	120	SAI	BW1 + 6.75	1.1	na	na	0.78	1.4	0	0.6		0.18
137	120	MAI	BW1 + 5.59	1.5	0	0	0.42	1.1	0	0.42	0	0
137	120	MAI	BW1 + 0.43	0.7	0	0	0.39	0.6	0.16	0.37	-0.16	0.02
153	120	SAI	BW1 + 15.39	0.2	0	0	0.59	0.2	0.21	0.4	-0.21	0.19
135	122	SAI	BW1 + 5.54	2.1	0	0.42	0.49	1.1	0.34	0.37	0.08	0.12
148	123	SAI	BW1 + 8.03	1	0	0	0.31	0	0	0	0	0.31
140	123	SAI	BW1 + 5.39	4	0	0	0.75	4.9	0	0.28	0	0.47
134	123	SAI	BW1 + 3.00	3	0	0	0.81	0.8	0.27	0.38	-0.27	0.43
135	124	SAI	BW1 + 5.98	4	1.2	0.31	0.79	0.5	0	0.49	0.31	0.3
147	124	SAI	BW1 + 18.47	2.6	1.1	0.33	0.77	1.4	0	0.3	0.33	0.47

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
142	125	SAI	BW1 + 14.93	0.5	0	0	0.85	0	0	0	0	0.85
144	125	MAI	BW1 + 9.24	1.7	0	0	0.3	1	0	0.2	0	0.1
144	125	MAI	BW1 + 14.89	1	0	0	0.43	0.5	0	0.2	0	0.23
135	126	SAI	BW1 + 3.32	3.6	1.2	0.4	0.7	2.9	0	0.65	0.4	0.05
147	126	SAI	BW1 + 17.51	3.2	1.3	0.26	0.35	5.6	0.35	0.43	-0.09	-0.08
142	127	SAI	BW1 + 9.19	3	1	0.29	0.44	3.2	0.24	0.67	0.05	-0.23
133	128	SAI	BW1 + 5.77	1.2	0.8	0.39	0.65	0.5	0	0.47	0.39	0.18
134	129	SAI	BW1 + 4.03	3.6	1.6	0.12	0.53	2.9	0	0.76	0.12	-0.23
132	129	SAI	09H + 0.17	0.5	0	0.14	0.82	0.9	0.34	0.82	-0.2	0
126	131	MAI	BW1 + 6.90	1.9	0.8	0.2	0.54	1.7	0.34	0.46	-0.14	0.08
126	131	MAI	BW1 + 2.12	0.7	0	0	0.48	0.8	0	0.46	0	0.02
144	131	SAI	BW1 + 13.90	2.5	0.9	0.44	0.79	2.3	0	0.66	0.44	0.13
125	132	SAI	BW1 + 6.26	0.8	0	0	0.53	0.6	0	0.34	0	0.19
124	133	MAI	BW1 + 1.92	1.1	0	0	0.47	1.1	0.16	0.48	-0.16	-0.01
124	133	MAI	BW1 + 5.42	2.1	0.5	0.4	0.8	2.1	0	0.55	0.4	0.25
126	133	SAI	BW1 + 0.26	2	0.4	0.29	0.63	1.8	0.24	0.51	0.05	0.12
126	133	SAI	09H +.77	0.4	0.2	0.12	0.76	0.3	0	0.55	0.12	0.21
139	134	SAI	BW1 + 14.42	1.2	0.7	0.46	0.76	0.8	0	0.89	0.46	-0.13
124	135	SAI	BW1 + 6.32	1.3	0.8	0.4	0.79	1	0.45	0.76	-0.05	0.03
130	135	MAI	09H + 15.82	3	3	0.32	1.06	3.6	0	0.48	0.32	0.58
130	135	MAI	BW1 + 9.45	1.4	1.2	0.4	0.76	1.1		0.77	0.4	-0.01
130	135	MAI	BW1 + 2.35	1.4	0.7	0.25	0.65	1	0.19	0.51	0.06	0.14
109	136	MAI	BW1 + 0.93	0.2	0	0.33	0.88	0.5	0	0.57	0.33	0.31
109	136	MAI	BW1 + 1.56	0.6	0.4	0.25	0.78	0.2	0	0.32	0.25	0.46
121	136	SAI	BW1 + 4.80	3.8	0.8	0.32	0.93	3.9	0.49	0.83	-0.17	0.1
123	136	SAI	BW1 + 4.80	4.7	0.7	0.27	1.01	4.2	0.65	0.98	-0.38	0.03
124	137	SAI	BW1 + 7.63	4.7	0.9	0.5	0.89	5.1	0.72	0.64	-0.22	0.25
118	137	SAI	BW1 + 5.89	1	0	0	0.5	0.4	0	0.49	0	0.01
119	138	SAI	BW1 + 6.24	1.8	0	0.2	0.46	1.5	0.29	0.54	-0.09	-0.08
126	139	SAI	BW1 + 2.46	1.3	na	na	0.63	1.9	0.52	0.48		0.15
124	139	SAI	BW1 + 6.19	3.8	3.2	0.26	0.83	4.3	0.3	0.7	-0.04	0.13
118	139	SAI	BW1 + 6.11	0.8	1.1	0.22	0.49	1	0.2	0.62	0.02	-0.13
122	139	SAI	BW1 + 5.42	2.4	1.5	0.2	1.06	1.5	0.27	0.76	-0.07	0.3
101	140	SAI	BW1 + 3.85	0.6	0.2	0.3	0.41	0.3	0	0.35	0.3	0.06
119	140	SAI	BW1 + 5.42	1.2	0.4	0.27	0.67	2.7	0.14	0.35	0.13	0.32
121	140	SAI	BW1 + 1.74	0.5	0.4	0.34	0.6	1	0	0.44	0.34	0.16
130	141	MAI	BW1 - 0.25	1.1	0	0	0.78	1.4	0.26	0.57	-0.26	0.21
130	141	MAI	BW1 + 13.59	12	1.9	0.28	0.54	8.2	0	0.38	0.28	0.16
119	142	SAI	BW1 + 5.50	1	0	0.32	0.52	0.9	0.42	0.68	-0.1	-0.16
129	142	MAI	BW1 + 15.54	0.6	0	0.1	0.6	1	0.22	0.36	-0.12	0.24

Row	Column	Call	Location	Max "L"	.115 "L"	.115 Volts	PP Volts	Hist Max "L"	Hist .115 V	Hist PP V	DVDT Pancake	DVDT Plus Point
129	142	MAI	BW1 + 6.27	1.9	0	0.43	0.45	0.8	0.36	0.43	0.07	0.02
132	143	SAI	BW1 + 17.79	1	0	0.31	0.39	0.5	0.27	0.6	0.04	-0.21
125	144	MAI	BW1 + 11.52	1	0.6	0.28	0.5	0.7	0.33	0.45	-0.05	0.05
125	144	MAI	BW1 + 7.67	0.9	0.7	0.44	1.01	0.6	0.34	0.93	0.1	0.08
127	144	SAI	BW1 + 1.26	0.3	0	0.14	0.52	0.3	0.25	0.54	-0.11	-0.02
131	144	SAI	BW1 + 16.68	1.8	0	0	0.45	1.6	0.2	0.59	-0.2	-0.14
135	144	SAI	BW1 + 18.03	1	0	0	0.38	0.8	0	0.47	0	-0.09
129	144	SAI	BW1 + 15.66	1.8	0.7	0.47	1.16	1.2	0.35	0.88	0.12	0.28
130	145	SAI	BW1 + 16.93	1.8	0.7	0.36	0.84	0.6	0.5	0.77	-0.14	0.07
126	145	SAI	BW1 + 11.10	1.2	0.6	0.3	0.76	1.2	0.34	0.79	-0.04	-0.03
131	146	SAI	BW1 + 16.08	2.5	0.4	0.26	0.77	2.1	0.31	0.61	-0.05	0.16
123	146	SAI	BW1 + 6.42	0.8	0.5	0.27	0.52	1.2	0.36	0.63	-0.09	-0.11
130	147	SAI	BW1 + 16.87	1.5	0.6	0.37	0.72	1.7	0.32	0.87	0.05	-0.15
132	147	SAI	BW1 + 17.30	0.6	na	na	0.8	0	0	0		0.8
106	149	MAI	BW1 + 0.20	0.7	0.3	0.35	0.63	0.4	0	0.32	0.35	0.31
106	149	MAI	BW1 + 1.99	1.3	0.8	0.54	1.16	0.7	0.31	0.9	0.23	0.26
106	149	MAI	BW1 + 5.17	0.6	0.4	0.38	0.4	0.6	0.51	0.63	-0.13	-0.23
103	150	SAI	BW1 + 5.39	0.4	0	0	0.42	0.6	0	0.42	0	0
122	151	SAI	VS1 + 0.00	0.6	0.4	0.35	0.56	0.5	0		0.35	0.56
120	151	SAI	BW1 + 14.55	1.8	0.8	0.23	0.74	1.1	0	0.36	0.23	0.38
101	154	SAI	BW1 + 2.69	0.7	0.6	0.65	0.75	0.7	0	0.63	0.65	0.12