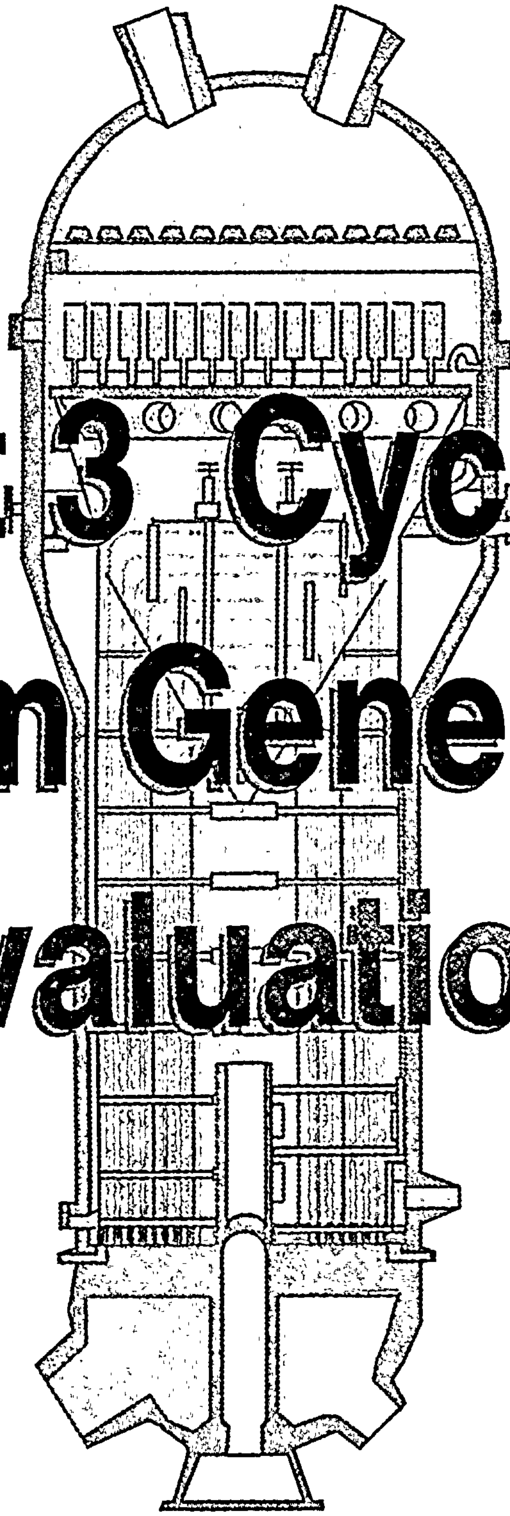
Palo Verde Nuclear Generating Station

Unit 3 Cycle 6 Steam Generator Evaluation



July 1996

PALO VERDE NUCLEAR GENERATING STATION

UNIT 3 CYCLE 6 STEAM GENERATOR EVALUATION

PREPARED BY: 6/26/96

Kevin M. Sweeney
Consulting Engineer

REVIEWED BY: 6/27/96

Richard F. Schaller
Manager, Steam Generator Projects

APPROVED BY: Jack A. Bailey

Jack A. Bailey
Vice-President, Nuclear Engineering

CONTRIBUTORS

APS Nuclear Engineering

Michael Melton

APS Steam Generator Projects

Richard Schaller
Douglas Hansen
Alice Nieto
Amy Wullbrandt

APS Inservice Inspection

Chris Brown

APS Licensing

Joe Provasoli

APTECH Engineering Services

Dr. Jim Begley
Brian Woodman
Tom Begley
Russ Cipolla
Steve Brown
Phil Besuner
Sat Mahajan

TABLE OF CONTENTS

LIST OF FIGURES

I. EXECUTIVE SUMMARY

This report describes the efforts conducted by Arizona Public Service (APS) to address tubing degradation in the Unit 3 steam generators at the Palo Verde Nuclear Generating Station (PVNGS). The assessment contained in this report supports continued Cycle 6 operation of Unit 3 until the next refueling outage. The refueling outage (U3R6) is scheduled for mid-March 1997, which represents 15.5 months of full power operation for Cycle 6.

The steam generator structural and leakage integrity analyses provided in this report are similar to the evaluations performed and presented to the USNRC Staff for Unit 1 in August of 1994, Unit 3 in May of 1995 and Unit 2 in August 1995. The analytical framework and results have been validated by recent steam generator inspections during U1R5, U3R5 and U2R6. The probabilistic run time analysis Unit 3 was performed by APTECH Engineering Services and reviewed by APS engineering. The processes of crack initiation, crack growth and eddy current inspection were modeled via a Monte Carlo simulation. End-of-Cycle (EOC) conditions were projected for Cycle 6.

The results support at least 15.5 months of full power operation in Unit 3. The conditional probability of tube burst for a postulated main steam line break is estimated as considerably less than 10^{-4} and therefore an acceptance criteria of 10^{-2} is satisfied. The probability of a tube defect exceeding the PVNGS structural design basis as set forth in Regulatory Guide 1.121 is less than 10^{-4} and this result has been further verified by a fully independent assessment. In over 10,000 Monte Carlo run time simulations, no instance of through-wall crack penetration was observed. Since no leakage at normal and postulated accident conditions is predicted, previous leakage integrity analyses submitted to the USNRC are considered bounding. Consequently, all of the analytical run time assessments performed for this report, strongly support full cycle operation in Unit 3.

This report also describes the elements of the PVNGS Degradation Management Program which provides a defense-in-depth approach to preventing a challenge to nuclear safety. Program actions include preventative measures, such as secondary chemistry improvements, primary temperature reductions, diagnostic equipment upgrades, comprehensive steam generator inspections and conservative plugging criteria. The PVNGS program is also prescriptive by specifying strict administrative controls on primary-to-secondary leakage and RCS activity levels, enhanced radiological monitoring, improvements in emergency operating procedures, 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 3 steam generators will be maintained, and that Unit 3 can be safely operated until the scheduled refueling at the end of Cycle 6.

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 3 steam generators. Other degradation mechanisms found in Unit 3, 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 specifically addressed 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 6 operation is justified in Unit 3 based on the end of Cycle 5 steam generator inspection results. The assessment contained in this report is structured to support continued operation in Unit 3 for at least 15.5 months following U3R5 until the next scheduled refueling outage. The U3R6 refueling outage is scheduled for mid-March 1997 which represents a 15.5 month operating run for Unit 3 Cycle 6.

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 3 was manufactured by Sandvik 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 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 3 commenced operation in October 1987, and at the end of Cycle 5, the steam generators had accumulated approximately 49,900 effective full power hours.

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.

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. Additional 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.

However, due to the complex synergy of these causal factors, APS has not determined the relative weight of each factor to the original tube failure. This issue has required that APS develop a comprehensive steam generator degradation management program in all three PVNGS Units. Due to design and operation similarities in the PVNGS Units, APS had to consider that this accelerated 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.

ECT crack indications exhibiting the same characteristics of bundle location, deposition and tube-to-tube crevice (bowing) were first observed in Unit 3 during the U3R4 inspection in March 1994. Upon completion of the U3R4 inspection program, a total of 17 ARC region defects were identified. The Unit 3 corrosion rates were considered to be bounded by the analyses performed for Unit 2 in Reference 1, and therefore, APS elected to inspect the Unit 3 steam generators after six months of operation (Reference 7). The results of this inspection are summarized in Reference 8. In May 1995, APS presented to

the USNRC Staff an evaluation justifying an 11 month operating run in Unit 3 (Reference 27), which when concluded, the unit was shutdown for refueling and the steam generators re-inspected.

During inspections conducted in U1R5 in April 1995, ARC region defects were identified in both the Unit 1 steam generators. The discovery was not unexpected, and now confirms the presence of this mechanism in all six PVNGS steam generators although only Unit 2 has exhibited a significant degradation pattern. No defects found in Unit 1 exceeded Regulatory Guide 1.121 structural limits after 15 months of operation. The inspection findings were consistent with the analyses presented to the USNRC in August 1994 (Reference 2), and no cycle length restrictions were required for Unit 1 Cycle 6 operation.

C. Safety Assessment

The safety significance associated with the operation of Unit 3 until the scheduled U3R6 refueling outage has been evaluated by APS. The results of the comprehensive inspection program conducted in U3R5 have been assessed statistically to determine the impact of leaving undetected ARC region defects through the remainder of Cycle 6 operation in Unit 3. The analyses concluded with high confidence that the conservative safety margins established in Regulatory Guide 1.121 are maintained. In assessing the safety significance of the current conditions in the PVNGS Unit 3 steam generators, APS has used, as guidance, the standards as defined in 10CFR 50.92 for determining whether a significant hazards consideration exists. The Code states in part that a significant hazard is not involved if the condition would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in the margin of safety. A discussion of these standards as they relate to Unit 3 operation until U3R6 follows:

<u>Standard 1</u> -- **Does the Unit 3 steam generator operation involve a significant increase in the probability or consequences of an accident previously evaluated?**

APS has determined via the analyses presented in this report, that operation of the Unit 3 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 (less than 0.0001) of generating a defect in excess of the structural margins specified in Regulatory Guide 1.121. The probability of leak or tube rupture at normal operating conditions has also been calculated to be less than 10^{-4} as no instance of through-wall crack penetration was observed after 10,000 Monte Carlo simulations. The conditional probability for tube rupture given a MSLB was calculated to be less than 10^{-4} which meets the criteria set forth in NUREG-0844 and the criteria listed in Generic Letter 95-05. 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 less than 10^{-4}. Therefore the analyses contained in Reference 10 with regard to leakage probability and core damage frequency are considered bounding. The Reference 10 PRA results 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 CDP, it is clear that continued operation of Unit 3 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. These features provide additional assurance that an orderly shutdown would be conducted prior to a through-wall leak propagating to a rupture.

APS has addressed the consequences of previously analyzed accidents with respect to operation with existing steam generator tubing conditions. 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.

Finally, a best estimate radiological assessement was performed by APS in Reference 9. The study concluded that in the unlikely event of a MSLB with consequential multiple tube ruptures, with the current administrative limits on reactor coolant system dose equivalent iodine, the resulting offsite doses would be less than 10CFR100 limits.

The defect management program at PVNGS ensures that the expected end-of-cycle steam generator condition will not significantly increase the probability or consequence of a previously analyzed accident.

Standard 2 -- Does the proposed operating interval for the Unit 3 steam generators create the possibility of a new or different kind of accident from any accident previously analyzed?

The analyses contained in this report demonstrate with high probability that steam

generator tubing structural integrity is maintained for the proposed operating interval. 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. The structural analyses conducted by APTECH indicated that no tube ruptures occurred in the Unit 3 Cycle 6 run time simulation after 10,000 trials, for a loading condition of 3ΔP. As a result the burst probability is less than 1E-4 for lesser loading conditions such as MSLB, stuck open safety valve (SOSV) or normal operating conditions. Since multiple tube ruptures are obviously rare events, the simulation code for the Unit 3 analysis was modified to define an extreme value distribution for burst pressure. Minimum burst pressure was defined as a Weibull distribution with a scale parameter of 7.37 ksi and a slope of 27.38. A limiting computation was performed by assuming the predicted Unit 3 EOC defect population were all at the limiting condition. The analysis indicated that the probability of <u>any</u> burst is less than 2E-6. Therefore, the probability of multiple tube ruptures either as an initiating event or the consequence of a MSLB is substantially less than the 1E-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 (1.2E-7) on the core damage probability (CDP) associated with operation of the Unit 2 steam generators until U2R6. The baseline core damage frequency at PVNGS is 4.74E-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.

Additionally, APS has conducted event specific training of operations personnel for tube rupture events, incorporated improvements in diagnostics via equipment and procedure upgrades (including the implementation of N-16 monitors), and incorporated upgrades to the Emergency Operating Procedures. These measures permit faster identification and isolation of the affected steam generator given an SGTR event, and decrease the likelihood of new or unanticipated concurrent events.

Therefore, APS concludes that the possibility of a new or different kind of accident than currently analyzed in UFSAR Chapters 6 and 15 is not created for the proposed operating run.

<u>Standard 3</u> **--Does the proposed run time involve a significant reduction in the margin of safety?**

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 concludes that these measures provide reasonable assurance that there are no reductions in the required safety margins, and that the PVNGS Unit 3 can be safely operated until the scheduled U3R6 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), two inspections in Unit 1 (U1R4 and U1R5) 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 tubes from Units 2 and 3, ultrasonic testing (UT) in Units 1 and 2 and in-situ pressure testing in Units 1 and 2. By integrating the information obtained from activities conducted in all three PVNGS units, APS has the tools to develop the elements of a Degradation Management Program for ARC region defects. 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 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 and economic 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 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 has implemented significant Steam Generator modifications in Unit 3 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 U3R5 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 APS engineering in defining inspection scope.

B. U3R5 Inspection Summary

A. Examination Scope

As in the previous inspections, the Unit 3R5 examination plan was developed in response to findings associated with previous eddy current examinations performed in Unit 3, as well as experience gained from recent inspections conducted in Units 1 and 2. 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.

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.

During U3R5, MRPC (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[1] (Plus Point) test program was performed on 100% of the hot leg tubesheet expansion transition locations. Additionally, a 10% 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 Unit 3 experience, MRPC testing in the rows 1 and 2 short radius u-bends was conducted from 07H-07C to determine if corrosion damage in this region was present. In summary, the base scope for the examination included:

- Examine 100% of Steam Generator 31 (SG 31) and Steam Generator 32 (SG 32) using bobbin coil techniques.

- Examine ~2500 tubes in each of SG 31 and SG 32 from 07H-2nd vertical support (VS) using MRPC techniques. The MRPC probe included the Plus Point coil for defect detection and characterization. These tubes were selected in the area of interest for ARC region axial indications.

- Examine ~185 tubes in each of SG 31 and SG 32 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 31 and SG 32 using MRPC (Plus Point).

1. The MRPC probe at PVNGS typically employs at least a 0.115" pancake coil and a Plus Point™ coil

- Examine ~110 tubes in each of SG 31 and SG 32 from 07H-07C using MRPC (Plus Point). This examination included all the rows 1 and 2 short radius U-Bend tubes.

- Examine historical >20% bobbin wear indications in SG 31 and SG 32 using MRPC (Plus Point). This inspection was performed to determine if corrosion was present at these locations.

- Examine ~1000 tubes in each of SG 31 and SG 32 of the cold leg tubesheet (TSC) expansion transition region using MRPC (Plus Point). This exam was performed to determine if cold leg corrosion was present.

The expansion criteria employed in Unit 3 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.

Table IV-1 - Unit 3R5 ECT Inspection Summary

SCOPE DESCRIPTION		SG 31	SG 32
Exam Description	Extents	Analyzed	Analyzed
FULL LENGTH BOBBIN	TEC-TEH	10,872	10,855
TUBESHEET (TSH) MRPC	TSH-TSH	10,872	10,872
U-BEND MRPC	07H-2nd VS	2549	2606
LOW ROW U-BEND MRPC	07C-07H	114	113
RANDOM ARC MRPC	07H-2nd VS	186	184
TUBESHEET (TSC) MRPC	TSC-TSC	1020	1013
MRPC OF PREVIOUS >20% WEAR	VARIOUS	226	73
EXPANSION 1 (SPECIAL INTEREST I-CODES/PID)	VARIOUS	107	89
EXPANSION 2 (SAI, MAI BOUNDING)	VARIOUS	77	97

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 U3R5 examination results is provided in Table IV-2. Additionally, a ARC region summary sheet is provided in Appendix B. The U3R5 inspection program resulted in the plugging of 17 tubes in SG 31 and 20 tubes in SG 32 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 MRPC voltage. The treatment of MRPC voltage and voltage growth in determining the allowable run time in Unit 3 is described in Section V and Appendix A.

Table IV-2 Indication Summary U3R5

INDICATION CATEGORY	STEAM GENERATOR 31	STEAM GENERATOR 32
Cold Leg Corner Eggcrate Wear		
0% to 19%	1	1
20% to 29%	0	0
30% to 39%	0	0
40% to 100%	0	0
Eggcrate Wear		
0% to 19%	772	522
20% to 29%	196	92
30% to 39%	30	5
40% to 100%	0	0
Flow Dist Plate Wear		
0% to 19%	1	0
20% to 29%	1	0
30% to 39%	0	0
40% to 100%	0	0
Batwing Wear		
0% to 19%	1491	1192
20% to 29%	341	147
30% to 39%	71	17
40% to 100%	2	0
Vertical Strap Wear		
0% to 19%	416	270
20% to 29%	101	36
30% to 39%	31	7
40% to 100%	1	0
Possible Loose Parts		
PLI	0	0
PLP	2	0

INDICATION CATEGORY	STEAM GENERATOR 31			STEAM GENERATOR 32		
Axial Indications	orig	exp1	exp2	orig	exp1	exp2
TSH	3	0	0	4	0	0
01H	0	0	0	0	1	0
Batwing/Vertical support	17	0	0	16	1	2
Circumferential Indications	1			9		
Single Volumetric Indications	31			51		

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 U3R5 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 3-coil MRPC probes and 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 31 and 32. 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 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 subsequently used in the U3R5, U2R5, U2R6 and U1R5 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 U3R5 employed a standard 0.115" diameter pancake coil, and a separate shoe contained the plus point coil. The benefits realized at PVNGS regarding 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. Recent industry results are expected to further validate the analytically predicted Plus Point POD. For the Unit 3 analyses, APS and APTECH elected to use an industry pancake coil POD curve which is regarded to be more sensitive than the 1993-1994 pancake coil PVNGS POD curve but not as sensitive as the Plus Point POD successfully used for Unit 2 Cycle 6 analyses (Reference 10).

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 APS and Rockridge 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. Upon conclusion of the U3R5 inspections, a select group of resolution analysts were retained to independently rereview the current inspection results and compare those results to the ECT data from previous inspections. 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 was generated and a final review of ARC region ECT calls for the Unit 3 data was conducted by the APS Level III analyst. Review of the ECT voltage data was also conducted by Steve Brown of APTECH. 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

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 microscopy 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 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 "deposit parameter" with a threshold value of $\rho V/1-X = 180$ lbm/sq ft•s. 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 in these sections that contaminant concentration and the corresponding initiation of cracks are related to a critical quality parameter. 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 has 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 supports the empirically determined threshold of 65% quality for the upper bundle region of 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.

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, two (2) major modifications were designed and implemented in Unit 3 during U3R5. The steam generator modifications are described as follows:

<u>Downcomer Feedwater on Hot Leg Side</u>
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.

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). Orifice plates installed in 62 of the 194 (per SG) steam separators were also removed to reduce recirculating loop resistance.

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 3 As-built Modification

The modification program in U3R5 was the first time 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. However, the shroud modification field experience in U3R5 found that shroud hole machining rates were slower than expected. Consequently, it was necessary to suspend the machining operation in SG 32 before all planned holes were completed, and hole machining was postponed altogether in SG 31. The final installation configuration for SG 32 is depicted in Figure IV-14.

Since the as-built condition did not conform to the original design, 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). From the assessment ABB-CE and APS concluded:

1. A core inlet temperature difference will not result

2. Tube vibration resulting from increased flow hole velocity is not expected to increase beyond the threshold of instability.

3. The conditions resulting from a partial steam generator modification are bounded by previously analyzed cases of no modifications and full modifications relative to its impact on the original design report (Reference 30).

4. 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.

5. 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 31	As-built SG 32
Maximum Quality	74%	56%	68%	58%
Recirculation Ratio	3.08	3.72	3.36	3.61
Tubes in Dryout	3000	1600	2800	2300
Tubes above Q=65%	1600	0	500	0
Flow Stability Margin	< 0.75	0.78	< 0.75	< 0.966

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 3, 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 6 to coincide with SG modification benefit and short term power needs.

Table IV-4 Plant Operating Parameters

	Design (VWO)	Current Operation	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

With respect to the operating run time analyses contained in this report, the key operating inputs of hot leg temperature and steam pressure (ΔP) are not affected. 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. The analysis to increase in RTP also reassessed allowable plugging limits. An increase in 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 of 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 and Generic Letter 95-05. The Unit 3 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 aspect of the analysis results. The number of observable defects, when adjusted for the 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.

3. In contrast to previous 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.

A complete description of the primary model analysis conducted by APTECH is provided in Appendix A. The following is a discussion of the fundamental components of the model including a summary and results.

1. Structural Assessment

The length and depth profiles of all cracks in tubes removed from Unit 2 have been analyzed and the critical regions of these profiles relative to burst pressures have been identified. As found in many industry examples, APS utilized MRPC detected crack length to assess structural margins. However, tube pull results consistently indicate that actual crack length, whether the defect is axial or circumferential, can be substantially longer than the MRPC detected length. This difference is not considered to be of issue since laboratory examination of PVNGS tubing removed from Unit 2 indicates that the shallow cracking at the ends of the crack profile is not a consequential factor in burst pressure. The Unit 2 tube pull data show that

the MRPC results provide an excellent indication of structurally significant length. Although there are individual instances where the MRPC crack length is slightly less than the directly corresponding structurally significant crack length, APTECH performed sampling of the MRPC crack length distribution. This approach provides a conservative estimate of the crack lengths in service. Since no known cracks are left in service, there is no need to adjust an individual crack length to cover the possibility that it may be undersized.

APS and APTECH continue to reconcile inspection changes with the structural design basis of the primary integrity model. A comparative assessment of the six 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. Since U2R5, APS has used the Plus Point coil nearly exclusively for crack length calls. The structural model based on PVNGS specific tube pulls demonstrated that pancake coil crack length is a good measure of the structural length of the flaw (See Figure 2.2 of Appendix A). The Plus Point coil indicated lengths are substantially longer than the RPC called lengths and therefore much larger than the structurally significant length. Use of Plus Point indicated length is clearly overconservative. In the Unit 3 model, a conservative Unit 2 pancake coil crack length distribution was utilized.

The field determination of structurally significant depth has been more difficult due to a number of plant specific differences at PVNGS versus more traditional industry experience. As reported in Reference 1, the PVNGS steam generator tubing was manufactured via a pilger process. The signal to noise ratio associated with the PVNGS tubing, combined with an IGA influenced morphology, and geometry variations associated with the tubing bends, has made the process of bobbin detectability and depth sizing of these defects virtually nonexistent. As a result, MRPC techniques have been utilized as the primary defect assessment technology.

APS, Rockridge and APTECH have considered several approaches for characterizing MRPC signal response. In principal, the phase angle of the impedance plane lissajous figure formed as the MRPC probe passes over the crack should provide a good correlation with crack depth. However, based on early field experience and tube pull results, significant difficulty had been encountered in attempting to characterize defect depth via phase angle measurements. APS and APTECH are continuing to develop the ability to dimensionally profile discontinuities by capturing the in-phase and quadrature signal components of the MRPC probe. Additional information concerning this effort was contained in Reference 10, and this work corresponds with recent industry programs for circumferential crack sizing.

Until this work is completed, an empirical correlation of pancake coil (MRPC) voltage with crack depth has been used to assess significant crack size. This correlation, as reported in References 2, 8 and 10, has proved to be reasonable in calculating crack depth, assessing crack propagation and determining structural integrity. However, a number of questions were raised by the USNRC Staff in review of early versions of the model regarding the physical basis of this correlation. The work conducted and presented in Reference 10 addressed these issues. The industry use of such a voltage correlation is being considered for circumferential cracks. It is important to note that the objective at PVNGS is not to accurately size all indications, since no detected cracks are left in service either as a function of measured depth or voltage. Instead, the MRPC voltage correlation was developed by APS as a means of assessing inspection data to arrive at a reasonable assessment of crack growth rates. As indicated in Section IV, APS has minimized the variability associated with field MRPC voltage values by conducting a thorough post inspection review of all the ECT crack indication data.

2. Growth Rates

Eddy current records of tubes found to contain MRPC indications during the U3R5 inspection were re-evaluated to determine if precursor signals were present at the time of the previous U3M5 inspection. The review indicated, that in some cases, precursor signals could be found. The results of the review are summarized in Appendix B. As stated previously, MRPC voltage is the ECT parameter used at PVNGS to monitor growth and severity of ARC region defects. Using this information, degradation growth rates can be estimated from past observations. Negative voltage growth rates are treated as zero growth. Approximately 15% of the voltage growth rates are negative or zero due to measurement uncertainties. If only positive values are considered, the average growth rate is 0.21 volts per EFPY. The maximum observed voltage growth rate is 0.52 volts per EFPY.

As in previous analyses, MRPC voltage changes have been analyzed by APS 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. The average factor applied to voltage growth is 16.2% through-wall per volt. A stochastic error term of 5.4% through-wall per volt was applied by APTECH to accommodate correlation uncertainty.

3. Probabilistic Model

As indicated previously, the probabilistic model used for Unit 3 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

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 length and tensile properties are assigned to a crack at initiation and are considered intrinsic properties of the crack. As stated previously, a comparative assessment of the six 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 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.

Upon detection, the length, depth, 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.

4. Summary

The APTECH 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 demonstrates 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 model predictions of low probability of RG 1.121 exceedance, through-wall leakers and numbers of detected defects. The inspections results indicate a continued trend towards corrosion rate reduction, and no challenges to plant safety have been identified.

5. Results

As stated in previous submittals, APS established a reasonable probability criteria for meeting Regulatory Guide 1.121 structural margins. This criteria was:

> *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.*

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 6. After 15.5 months of operation the probability of more than one RG 1.121 exceedance was estimated to be less than 10^{-4}. At this same point in time, the conditional probability of tube burst for a postulated main steam line break is estimated as considerably less than 10^{-4} and therefore an acceptance criteria of

10^{-2} is satisfied. With regard to leakage, in over 10,000 Monte Carlo run time simulations, no instance of through-wall crack penetration was observed. These results exceed the acceptance criteria established by APS in Reference 2, and demonstrate with high confidence that steam generator tube integrity in Unit 3 can be maintained until end of cycle.

6. Benchmarking

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 4.1 to 4.4 in Appendix A. The actual number of indications observed match up well with the most likely calculated number of indications for U3R4, U3M5, and U3R5 inspections. Approximately 150 new indications are predicted for the U3R6 inspection. The U3R6 predictions are probably very pessimistic reflecting the usage of a very conservative Weibull initiation model as used in the previous Unit 3 analysis (Reference 8). As opposed to the Unit 2 Cycle 6 assessment performed in Reference 10, the Unit 3 crack initiation function was not optimized for the new distribution of growth rates and a less sensitive POD function was used for Unit 3. Therefore, the predictions for the Unit 3 EOC (6) condition, with regard to size and number of defects, are considered to be conservative. The U2R6 inspection results confirm this conclusion, as the number and size of defects identified were much less than projected, despite the optimization process employed in Reference 10.

B. 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.

1. 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 3 Cycle 6. The key assumptions, inputs and model improvements developed for this analysis are summarized below.

2. BALIFE Model Assumptions and Input

A. The BALIFE Code, Version 5.051 has incorporated 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 last five Unit 2 inspections and the last Unit 3 inspections with reasonably accurate results for the subject ARC region cracks and other tube crack modes.

B. 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 3 BALIFE baseline analysis. The effects of various possible inspection improvements were then calculated by altering the best-estimate POD values.

C. Results from previous BALIFE predictions for Unit 2 and 3 show that ARC region ODSCC has a Weibull slope or β value of between 3 and 4.5. 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.

D. As the model is designed to track the damage classification via the inspection results several inspection data assumptions are used. In Appendix B the inspection results for U3R5 and look-up data form U3M5 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. 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 70% of the Plus Point calls were observed by 0.115 pancake. The following two assumptions were applied in the Unit 3 Cycle 6 assessment

- For U3M5, the indication counts for Plus Point and pancake still apply

- For U3R5, a tube with any pancake voltage indication greater than 0.2 volts would have been identified and plugged had the Plus Point coil not been used.

3. 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:

A. 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.

B. Damage Classification for Each Tube

- At any stage in the life of an unplugged tube, its worst crack belongs to one of five 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* Bob-

bin or both. This probability of a miss by *both* inspections is quantified below and defined as POM(RGE)=1-POD(RGE).

e. <u>Leak or burst.</u> The crack penetrates through the tube wall.

- As any tube ages, its worst crack moves from category one toward category five and plugging is the only way to halt this process.

- Any tube found to be damaged is immediately plugged.

C. <u>Inspection Simulation</u>

- From the POD curve (Figure V-1), 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.

- The last three assumptions are combined as a best-estimate baseline inspection model, and labeled as ***6-15-40.***

D. <u>Crack Growth</u>

- 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.

E. Unit 3 Forecast from Statistical Combination of SG 31 and SG 32 Forecasts

- The RGE analysis was first run for each steam generator. To make the calculations for RGEs in Unit 3, it is assumed that the number of RGEs in SGs 31 and 32 are independent Poisson-distributed variables. So steam generator "failure" rates were added to obtain best estimates for Unit 3. To calculate confidence bounds for Unit 3, 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.)

F. Future EFPY Buildup and Damage Exposure

- *No credit* was taken for primary temperature reduction or chemical cleaning. Damage is assumed to correlate with EFPY through the Bayesian Weibull model. EFPY is assumed to increase with future calendar time at past rates.

4. Results

As stated in previously, 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 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.*

Nine different cases were run by APTECH and the estimates are summarized in Table V-1. Based on the results, the probability of one (1) or fewer RGEs after 15.5 months of operation is 6E-4 or a 99.94% 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 99.97% chance of one or less RGE in Unit 3.

5. Comparison with Primary Model Results

The results of the independent evaluation compare well with the primary analyses for the operating run proposed in this evaluation. The results from both models are indicate with high confidence that the probability of 1 or more RG 1.121 exceedances to be no more than 6E-4. The objective of an independent check has been satisfied for the Unit 3 evaluation. Full cycle operation in Unit 3 during Cycle 6 is strongly supported.

Table V-1 - APTECH Independent Assessment Results

Analysis	% Chance of 0,1,or more than one RGE crack		
Type of BALIFE Analysis	Zero RGEs	Exactly one RGE	More than 1 RGE
Best Estimate of SG 31	99.21	0.79	0.00
90% High Bound of SG 31	98.66	1.33	0.01
10% Low Bound of SG 31	99.58	0.42	0.00
Best Estimate of SG 32	98.50	1.49	0.01
90% High Bound of SG 32	97.59	2.38	0.03
10% Low Bound of SG 32	99.26	0.74	0.00
Best Estimate of BOTH SGs of Unit 3	97.73	2.25	0.03
90% High Bound of Unit 3	96.67	3.27	0.06
10% Low Bound of Unit 3	98.56	1.43	0.01

C. Leakage Model

1. Introduction

A leakage model for PVNGS was 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 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 Section V, after 10,000 computer simulations no instances of through- wall cracks were observed for Unit 3 Cycle 6. 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 3 was deemed unnecessary as the lack of the development of a single through-wall indication in 10,000 run time simulations demonstrates that leakage is not an issue. At the very least, the Unit 2 Cycle 6 leakage analysis results are bounding. 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.

2. Acceptance Criteria

The MSLB analysis and associated assumptions are addressed in Section 15.1.5 of the PVNGS UFSAR. Based upon the review of these assumptions 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 a 95% probability that leakage from EOC tubing conditions with consequential MSLB will remain below the 6 gpm.

3. 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.

4. 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.

5. 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.

6. 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 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.

7. 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 Monte Carlo evaluation for 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.

8. Leakage Assessment

As indicated in Section V, after 10,000 computer simulations no instances of through-wall cracks were observed for Unit 3 Cycle 6. 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 3 was deemed unnecessary as the lack of the development of a single through-wall indication in 10,000 run time simulations demonstrates that leakage is not an issue. At the very least, the Unit 2 Cycle 6 leakage analysis results are bounding. 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 3 under worst case MSLB conditions will remain below the applicable 10CFR100 limits for off-site doses.

VI. OPERATIONAL RESPONSE

A. Description

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 detailed description of how PVNGS operational response is supported by integrating inspection, repair, leakage monitoring, and operator training has been described in References 2, 8 and 10 submitted to the USNRC Staff. No changes in the program have been made for Unit 3 Cycle 6.

VII. SUMMARY

The analyses and evaluations contained in this report demonstrate that the operating, inspection and repair program for the Unit 3 steam generators permit safe operation of Unit 3 for the remainder of Cycle 6. The progression of ARC region degradation in Unit3 is such, that further midcycle inspections 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 casual 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 show a favorable change in crack crevice chemistry tending towards neutral conditions. APS has exceeded EPRI action levels for sulfate by requiring 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.

- 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 6.

- 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 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 is considered to be bounding for Unit 3 Cycle 6.

These actions are all part of a defense in depth approach employed by APS, to provide reasonable assurance that PVNGS Unit 3 can be safely operated until the next scheduled refueling shutdown for further steam generator inspections. This approach incorporates additional best estimate analyses 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 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 3 Cycle 6, 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 3*, Calculation No. AES-C-1893-5, APTECH Report Number 93031893-1Q, Revision 2 dated June 1996

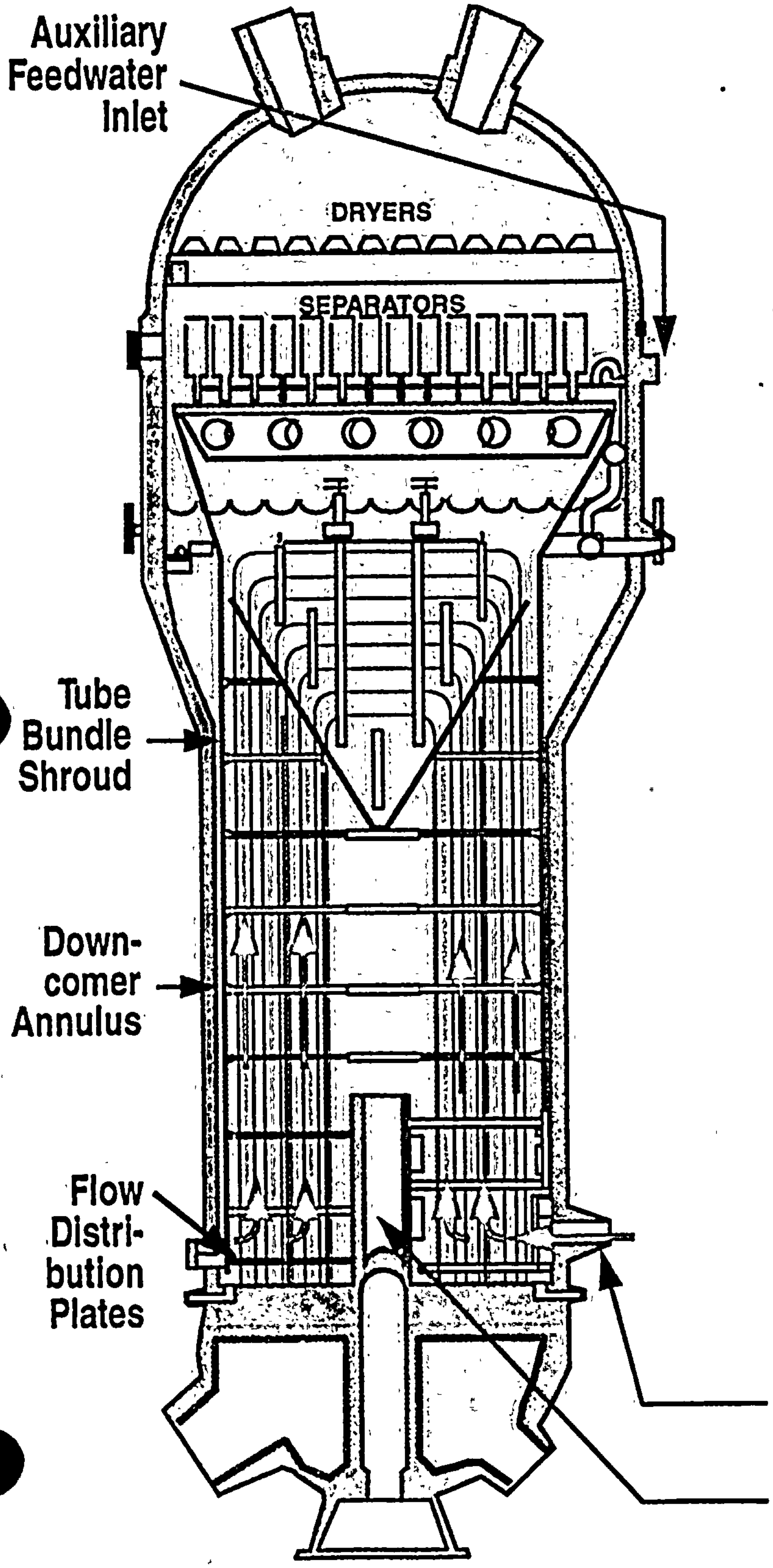
35. Draft Regulatory Guide x.xx, *Steam Generator Tube Integrity*

36. Letter, Brian Woodman (APTECH) to Kevin Sweeney (APS), *Multiple Tube Burst Probabilities for Palo Verde Unit 3*, dated June 20, 1996

IX. FIGURES

IX. FIGURES

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

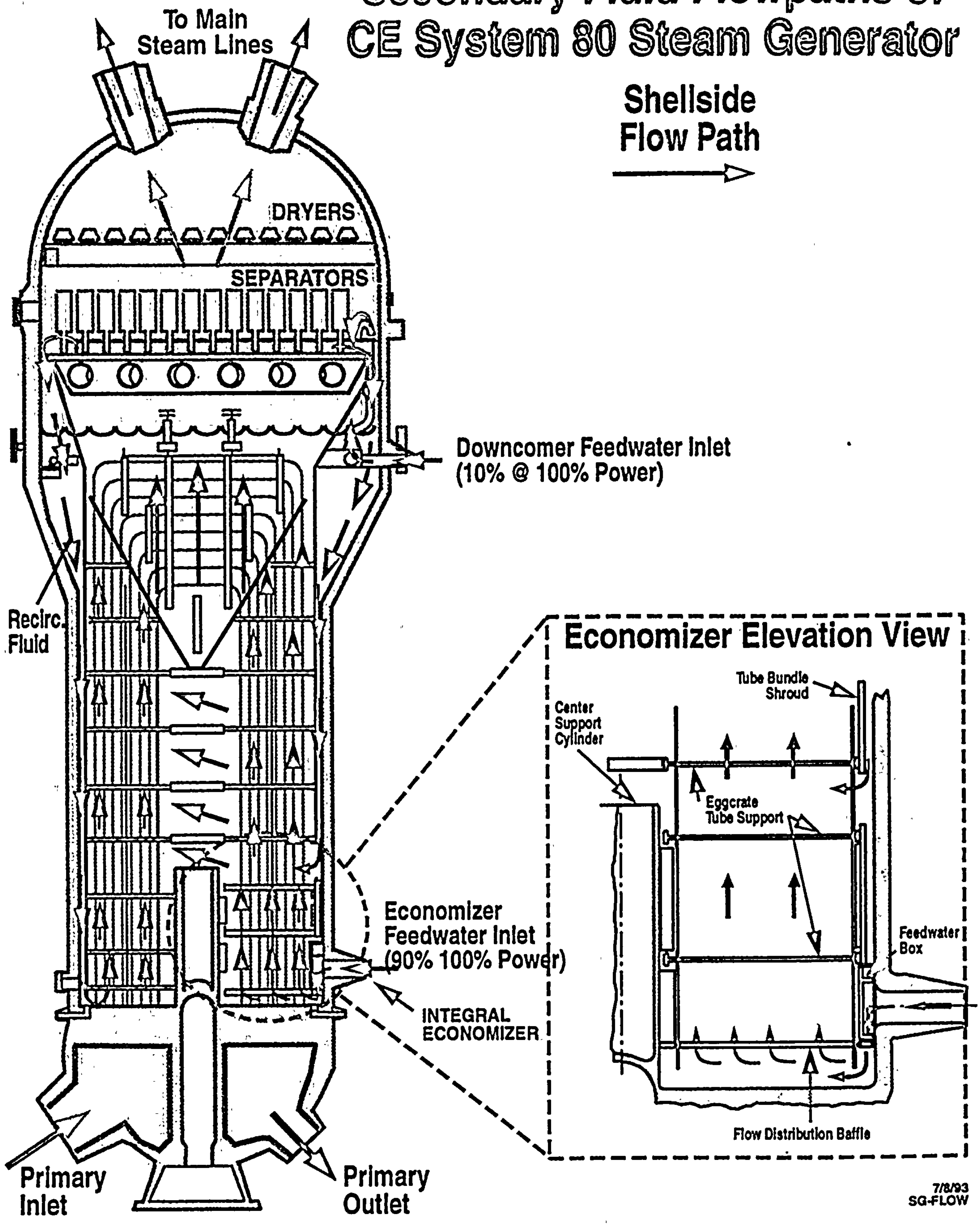


FIGURE II-2

Location of Unit 2 Axial Cracks

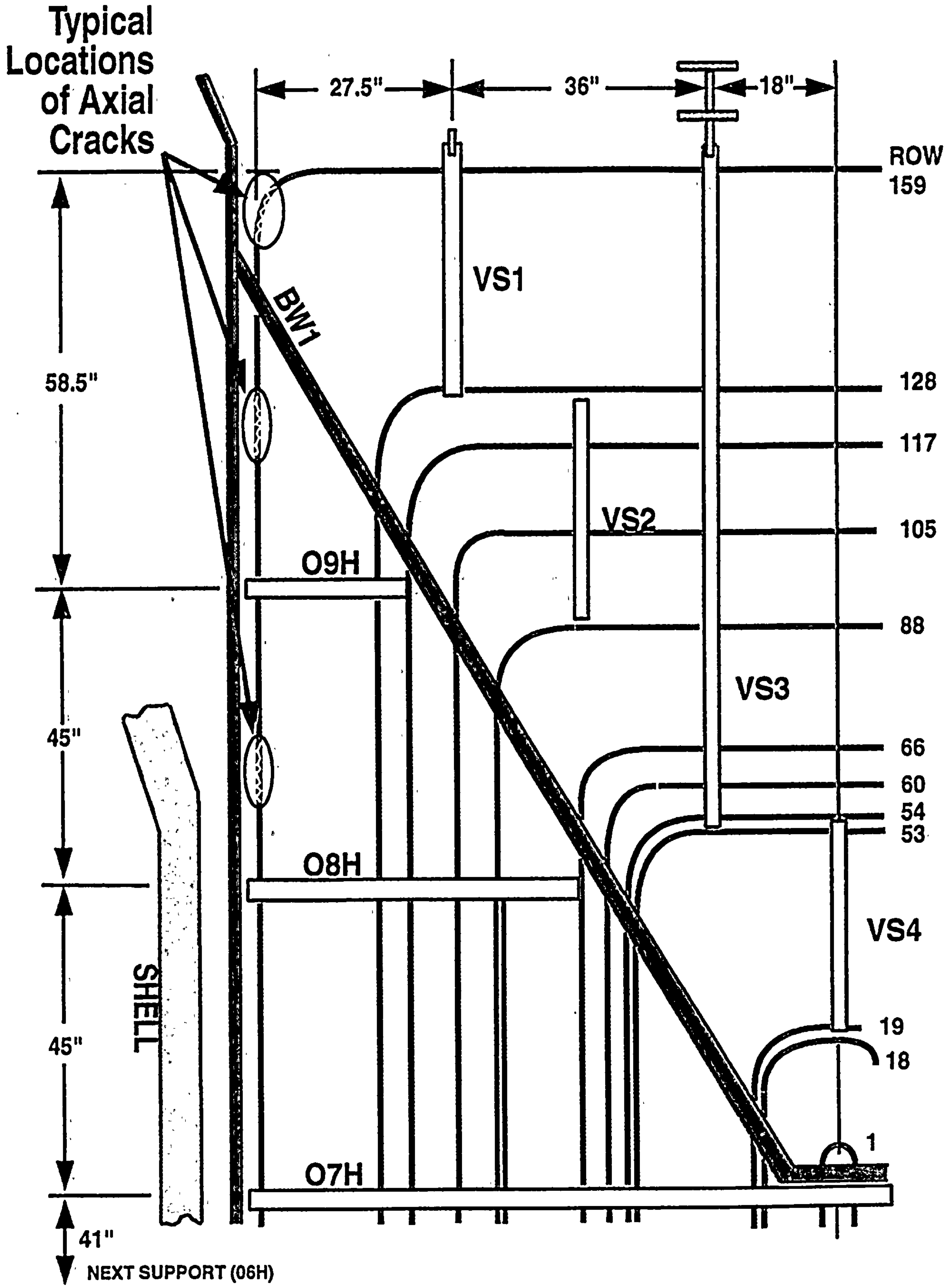


FIGURE IV-1

SQ_14JMA

10/95, ARIZONA PUBLIC SERVICE CO., PALO VERDE, UNIT 3

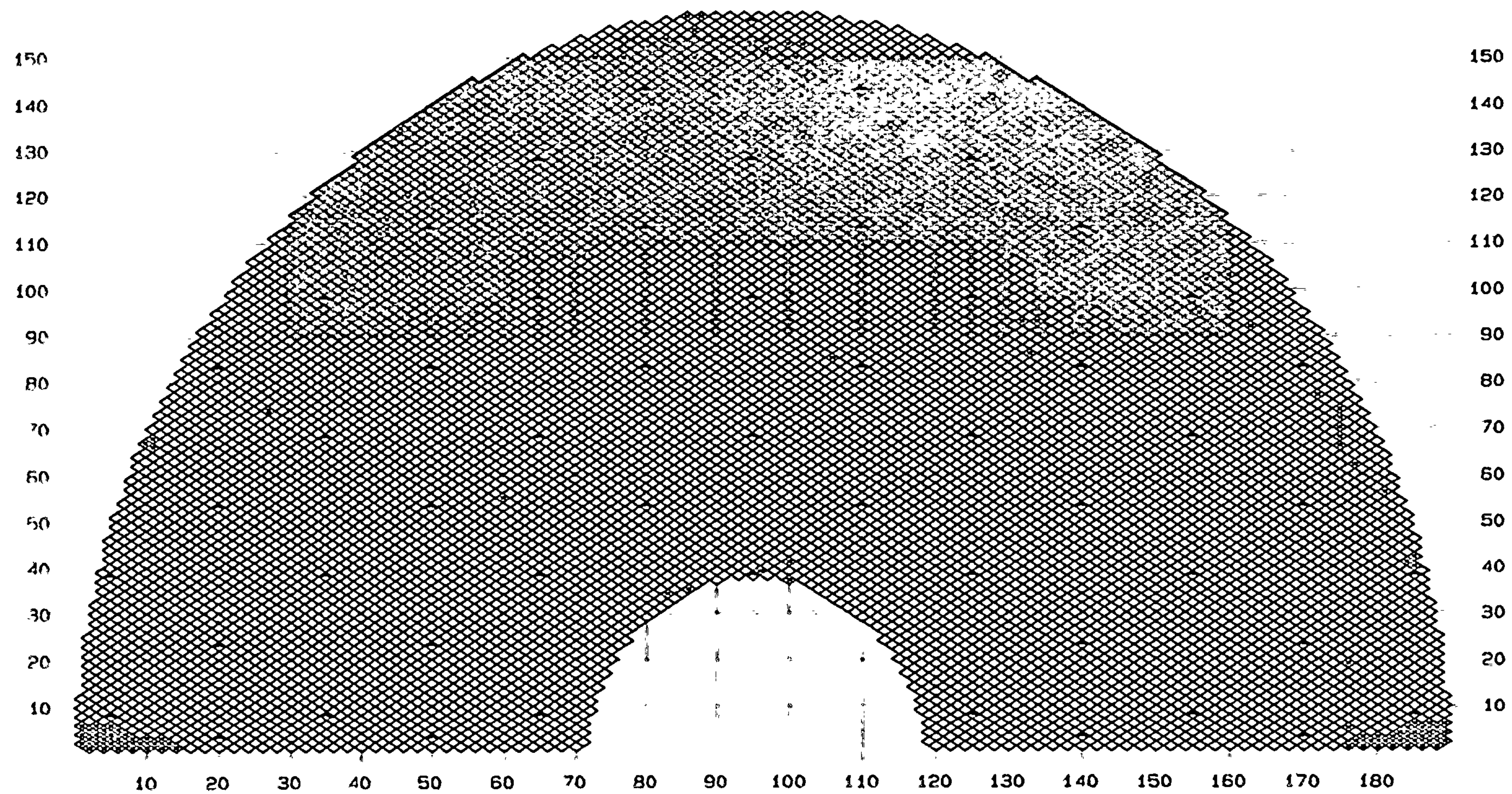
STEAM GENERATOR: 31
LOCATION: ALL

DATE: 06/21/96
TIME: 14:12:02

CRITERIA: ALL TUBES EVALUATED/EXAMINED

STAYS ▲

PLUGGED 140 ✕ EVALUATED [illegible]012 ◆ EXAMINED 0 ◆



ROCKRIDGE TECHNOLOGIES

CONAM NUCLEAR, INC. BW

FIGURE IV-2

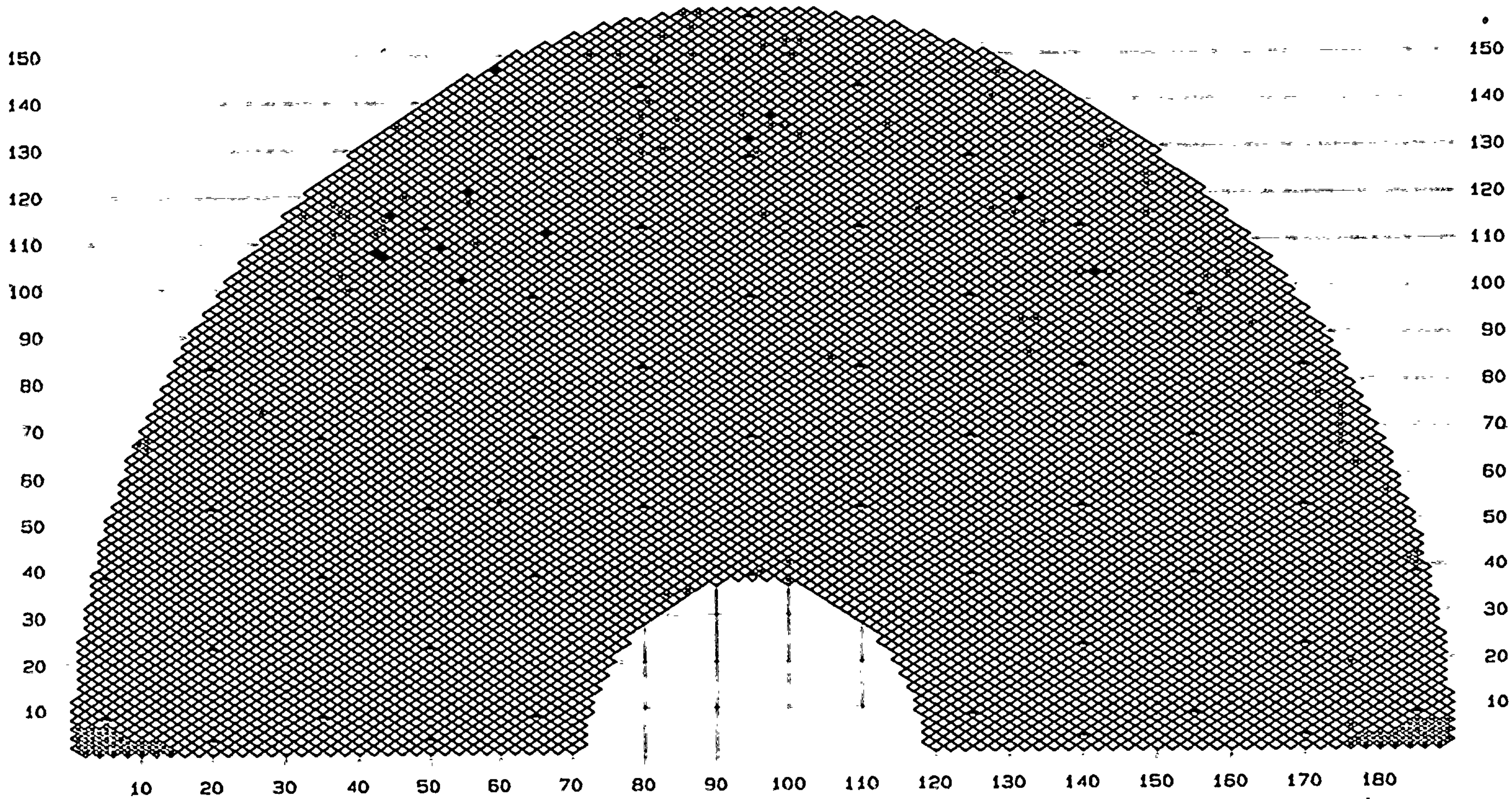
10/95, ARIZONA PUBLIC SERVICE CO., PALO VERDE, UNIT 3

STEAM GENERATOR: 31
MAI, SAI 07H TO VS3
Indication Location: 07H 0.00 to VS3 0.00 AND Percent: MAI, SAI

DATE: 06/21/96
TIME: 15:58:22

STAYS ▲

PLUGGED 140 X MAI 5 ◆ SAI 12 ◆



ROCKRIDGE TECHNOLOGIES

CONAM NUCLEAR, INC. BW

FIGURE IV-3

10/95, ARIZONA PUBLIC SERVICE CO., PALO VERDE, UNIT 3

STEAM GENERATOR: 32
LOCATION: ALL

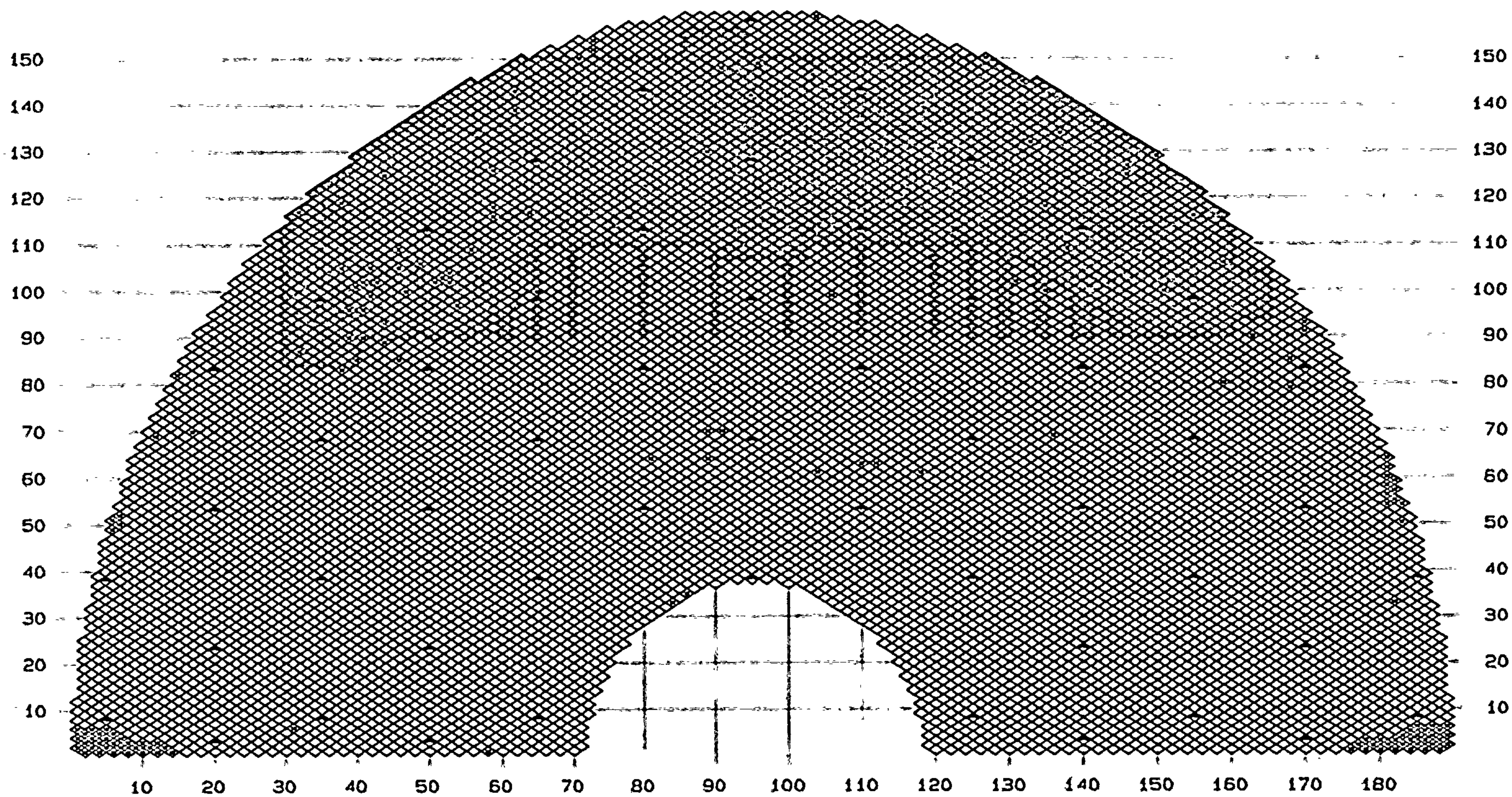
DATE: 06/21/96
TIME: 15:39:32

CRITERIA: ALL TUBES EVALUATED/EXAMINED

STAYS ▲

PLUGGED 157 X

EVALUATED 2885 ◆ EXAMINED 2 ◆



ROCKRIDGE TECHNOLOGIES

CONAM NUCLEAR, INC. BW

FIGURE IV-4

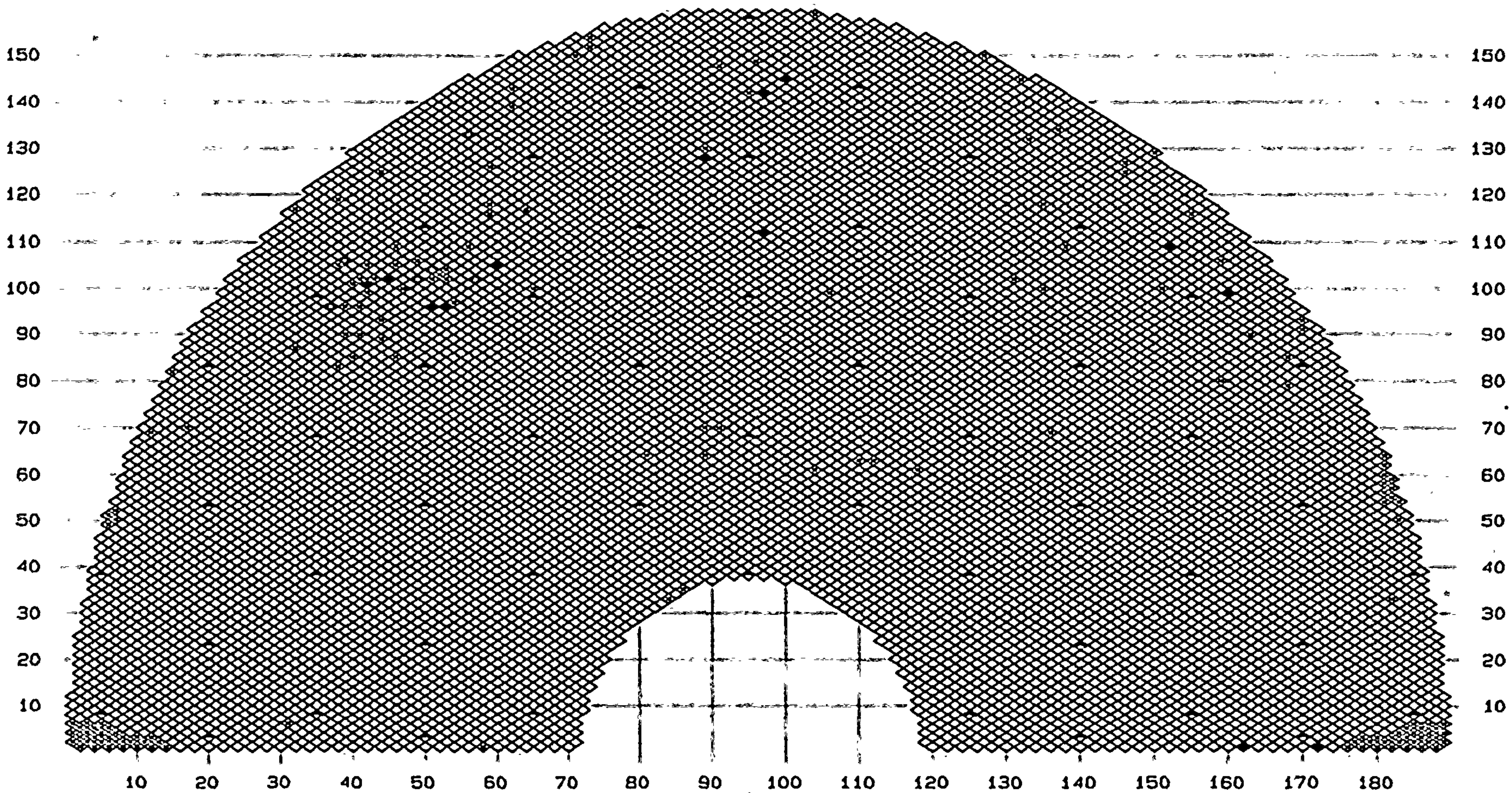
10/95, ARIZONA PUBLIC SERVICE CO., PALO VERDE, UNIT 3

STEAM GENERATOR: 32
OUTAGE DATA SET : CURRENT
Indication Location: 07H 0.00 to VS3 0.00 AND Percent: MAI, SAI

DATE: 06/21/96
TIME: 15: 43: 26

STAYS ▲

PLUGGED 157 X NAI 6 ◆ SAI 13 ◆



ROCKRIDGE TECHNOLOGIES

CONAM NUCLEAR, INC. BW

FIGURE IV-5

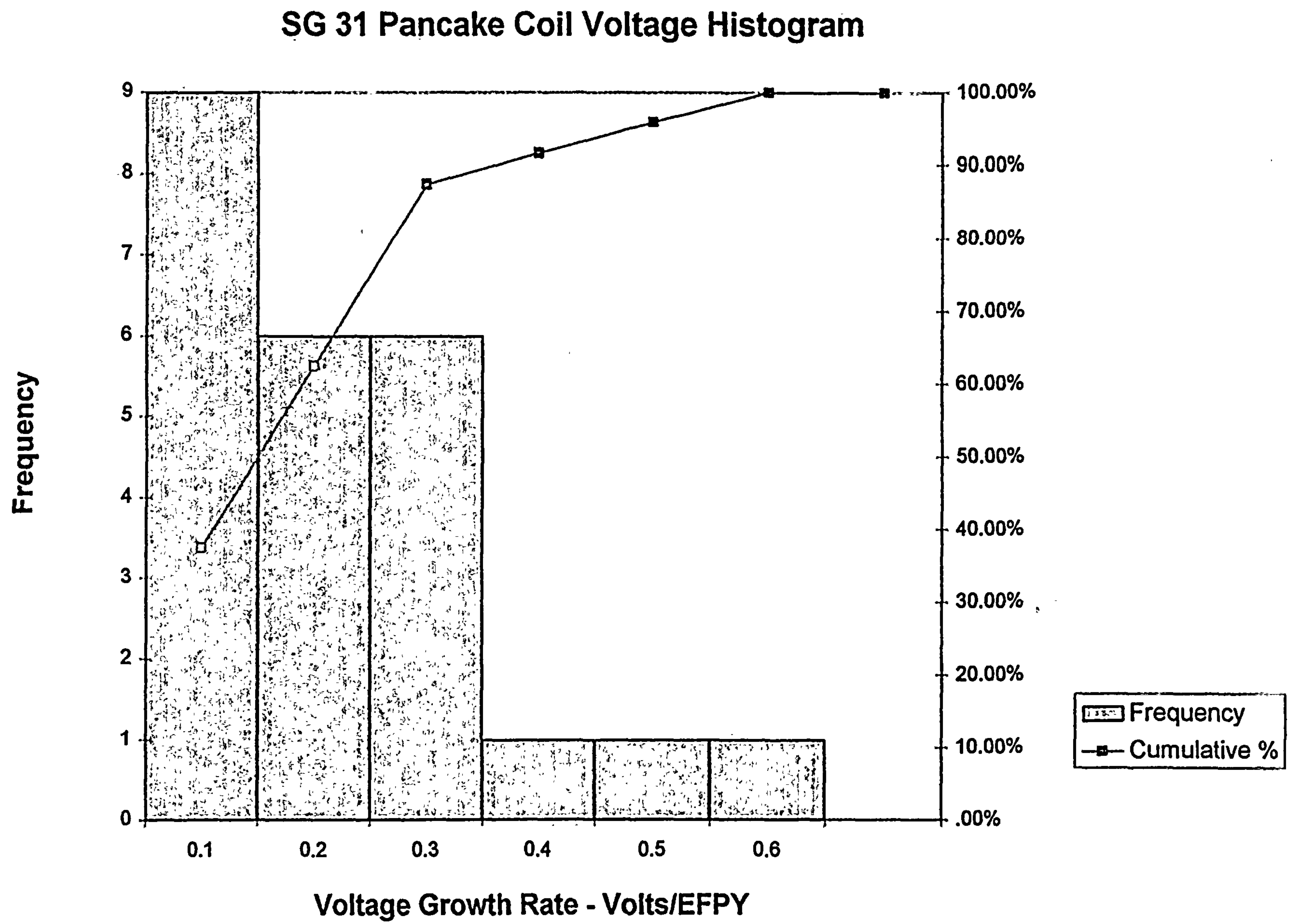

SG 31 Pancake Coil Voltage Histogram
Frequency
9
8
7
6
5
4
3
2
1
0
100.00%
90.00%
80.00%
70.00%
60.00%
50.00%
40.00%
30.00%
20.00%
10.00%
.00%
0.1
0.2
0.3
0.4
0.5
0.6
Voltage Growth Rate - Volts/EFPY
Frequency
Cumulative %

FIGURE IV-6

SG 32 Pancake Coil Voltage Histogram

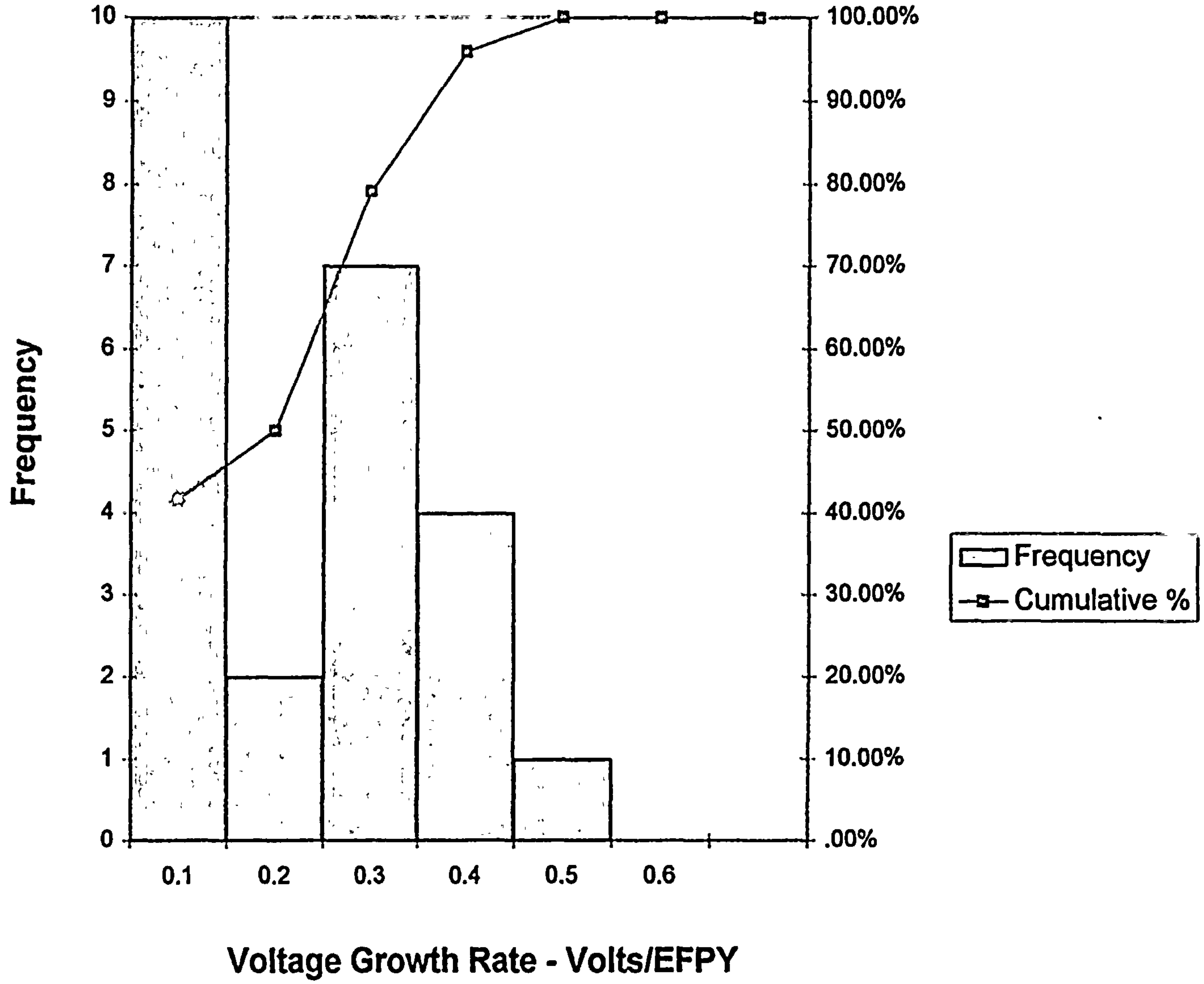
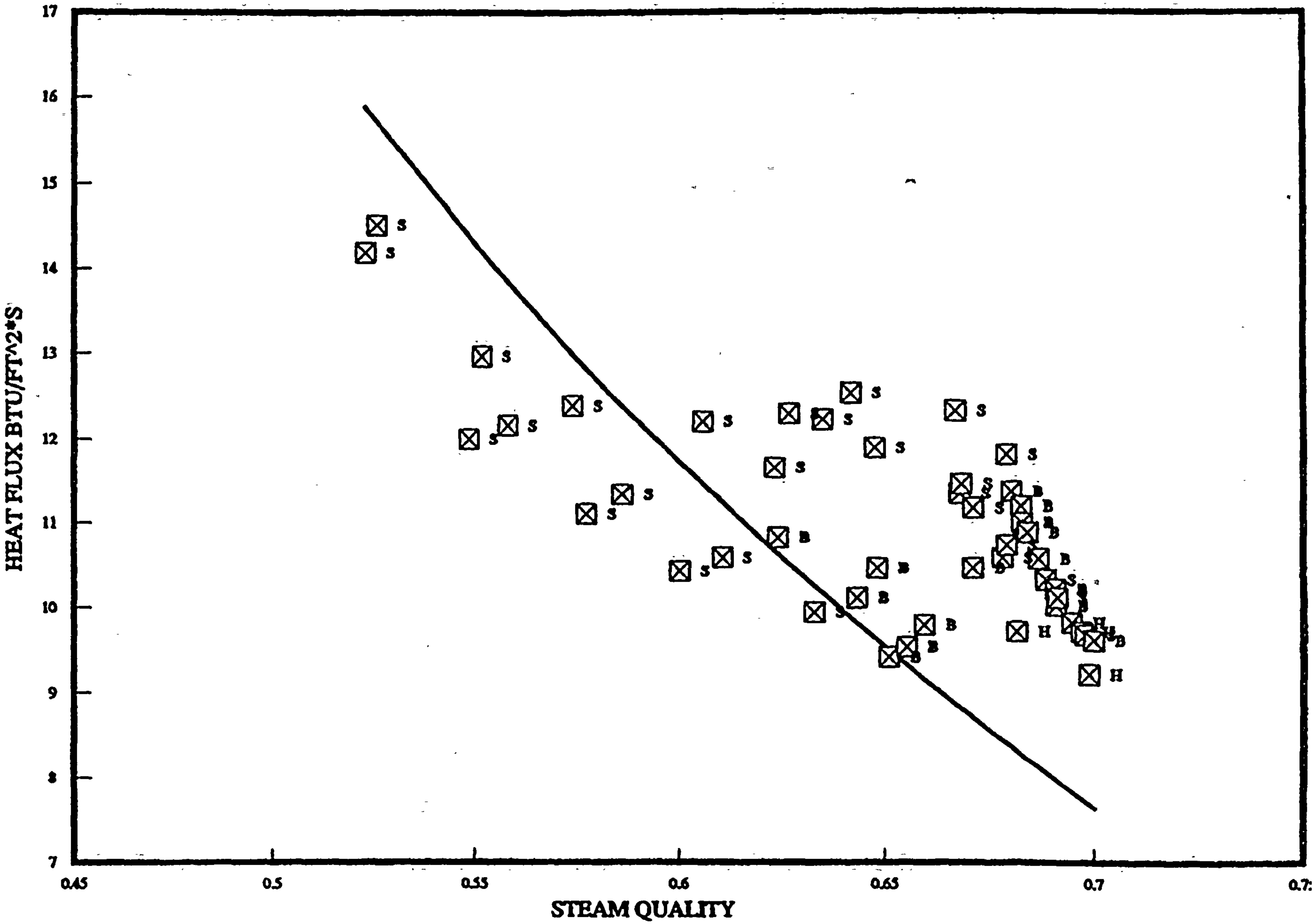


FIGURE IV-7

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

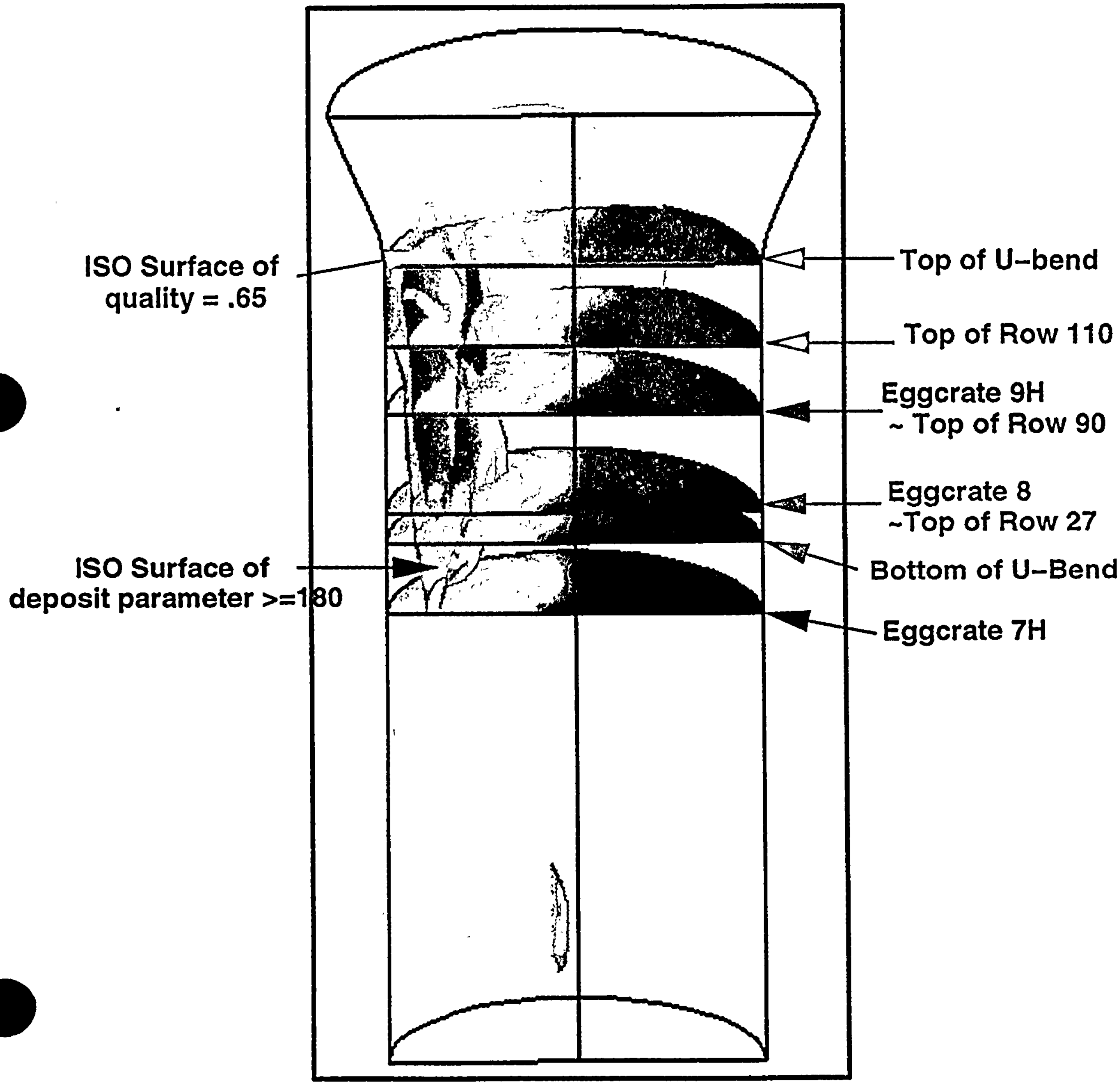


FIGURE IV-9

100% POWER

Tcold = 555 deg F

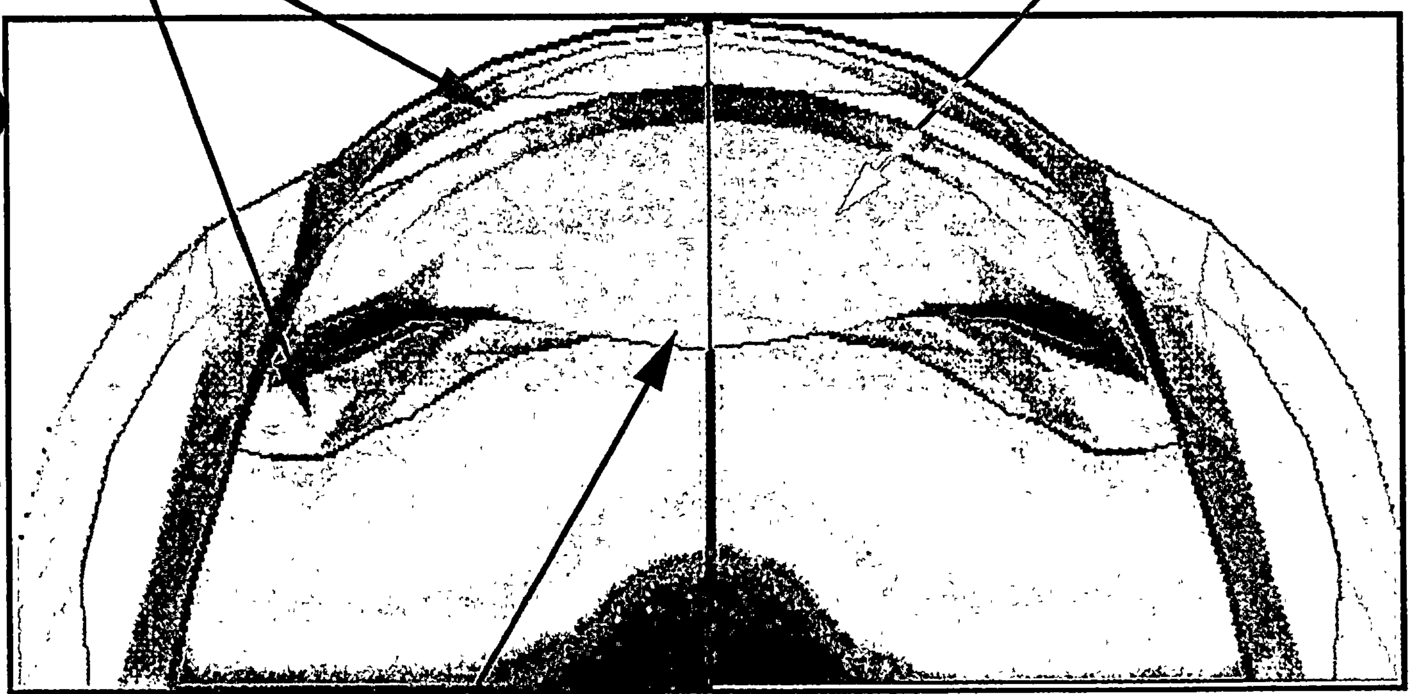
As Designed PVNGS Steam Generator

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

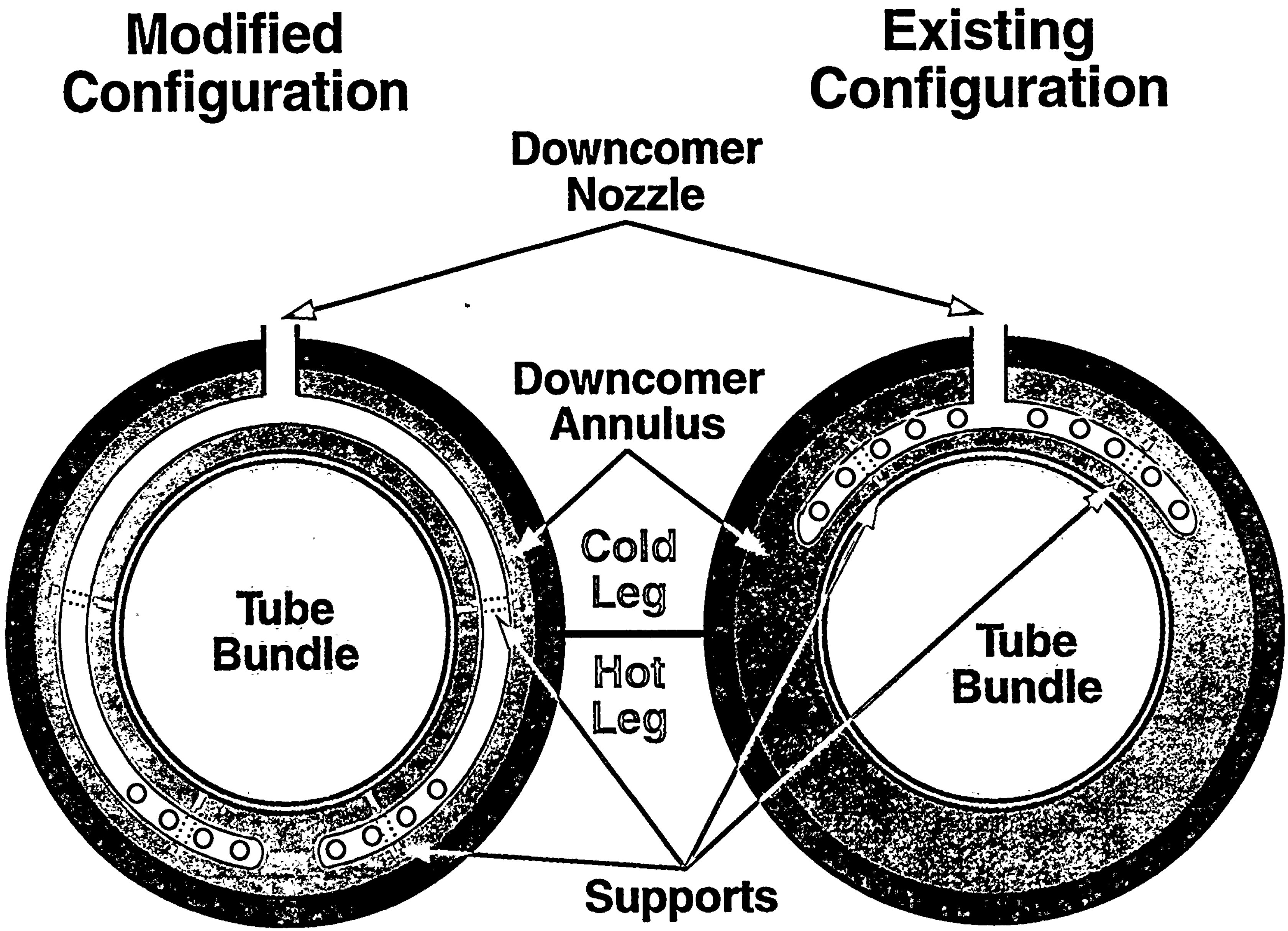


FIGURE IV-11

LEY2\FEEDRING

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

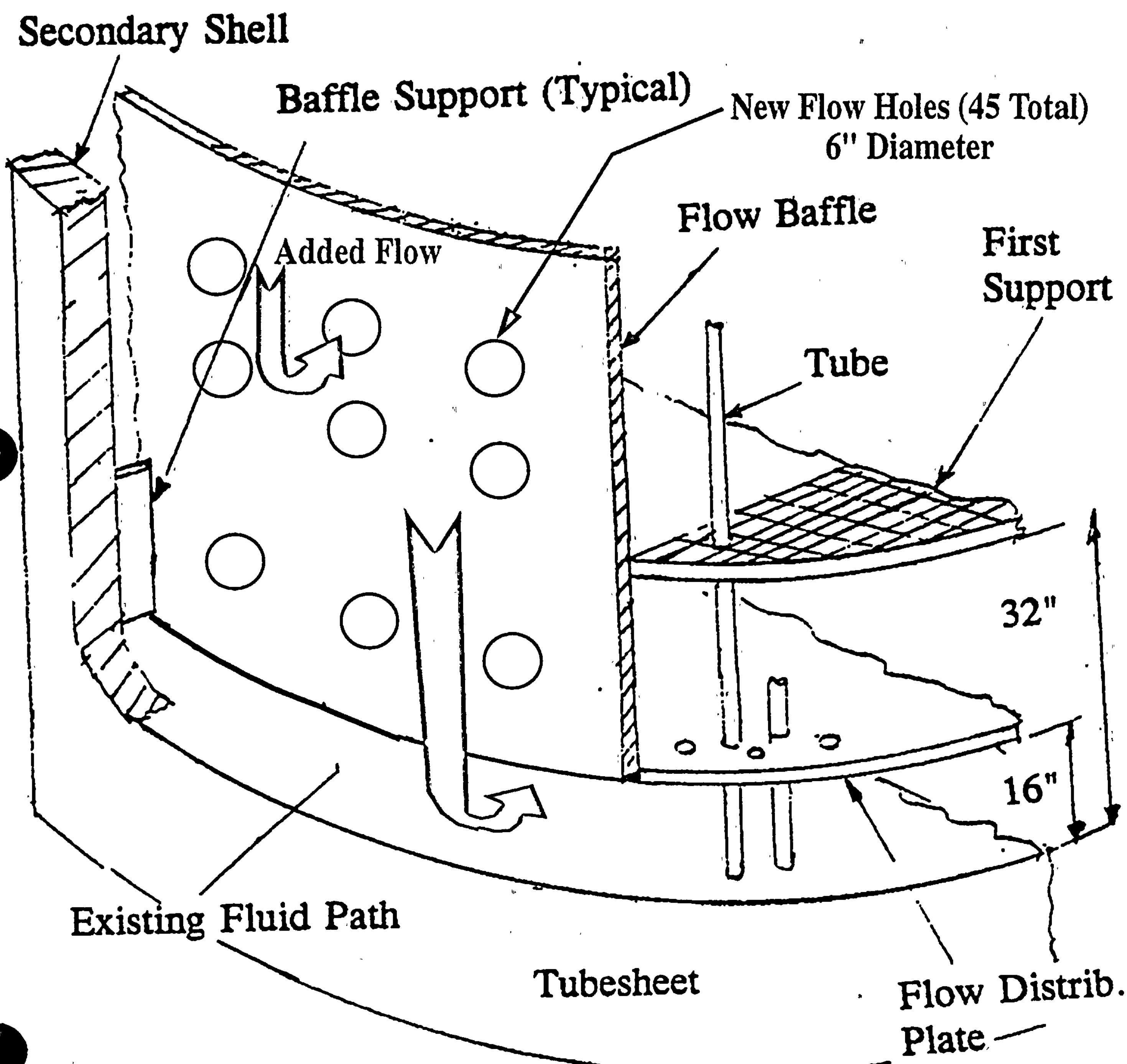
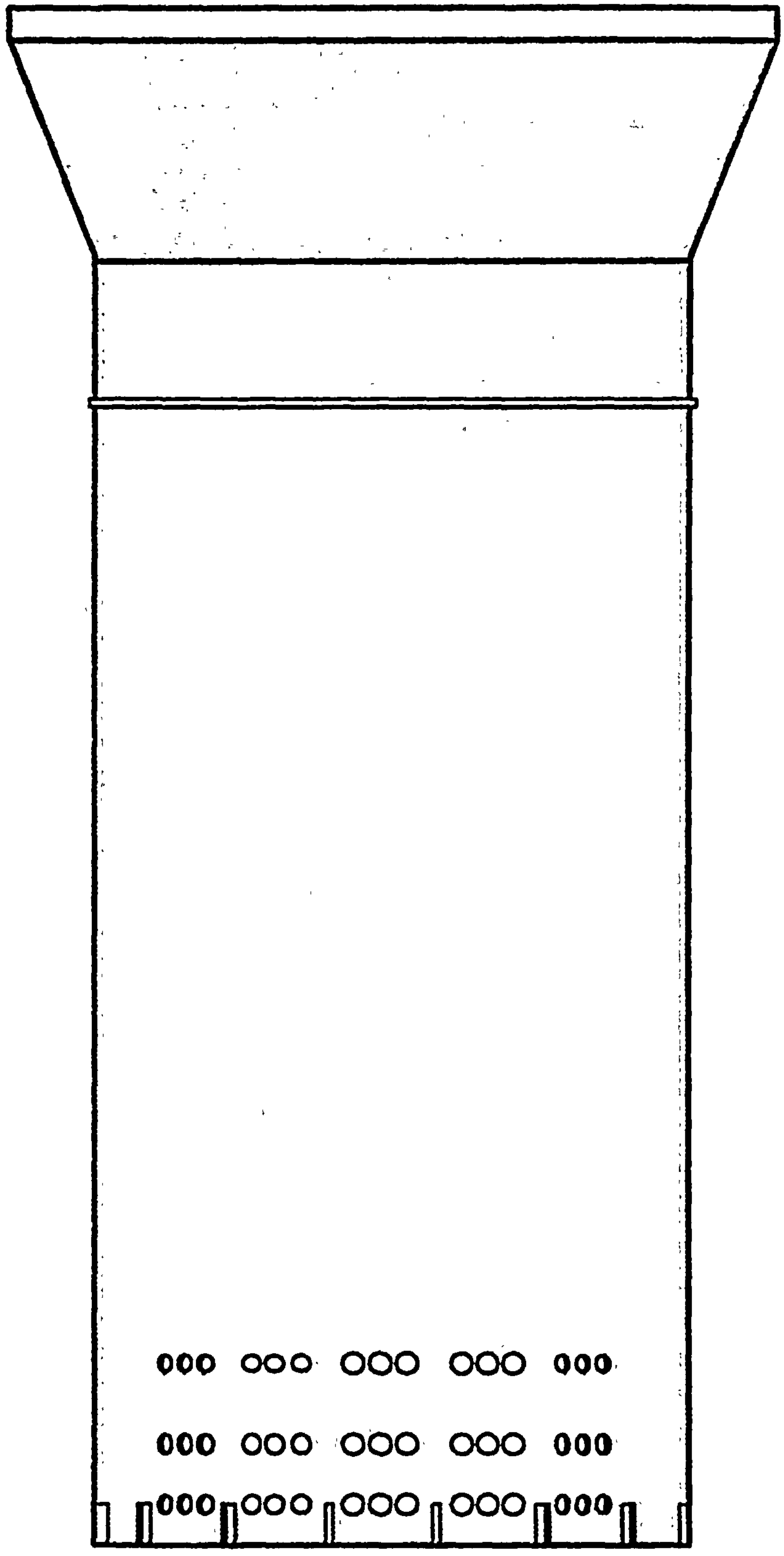


FIGURE IV-12

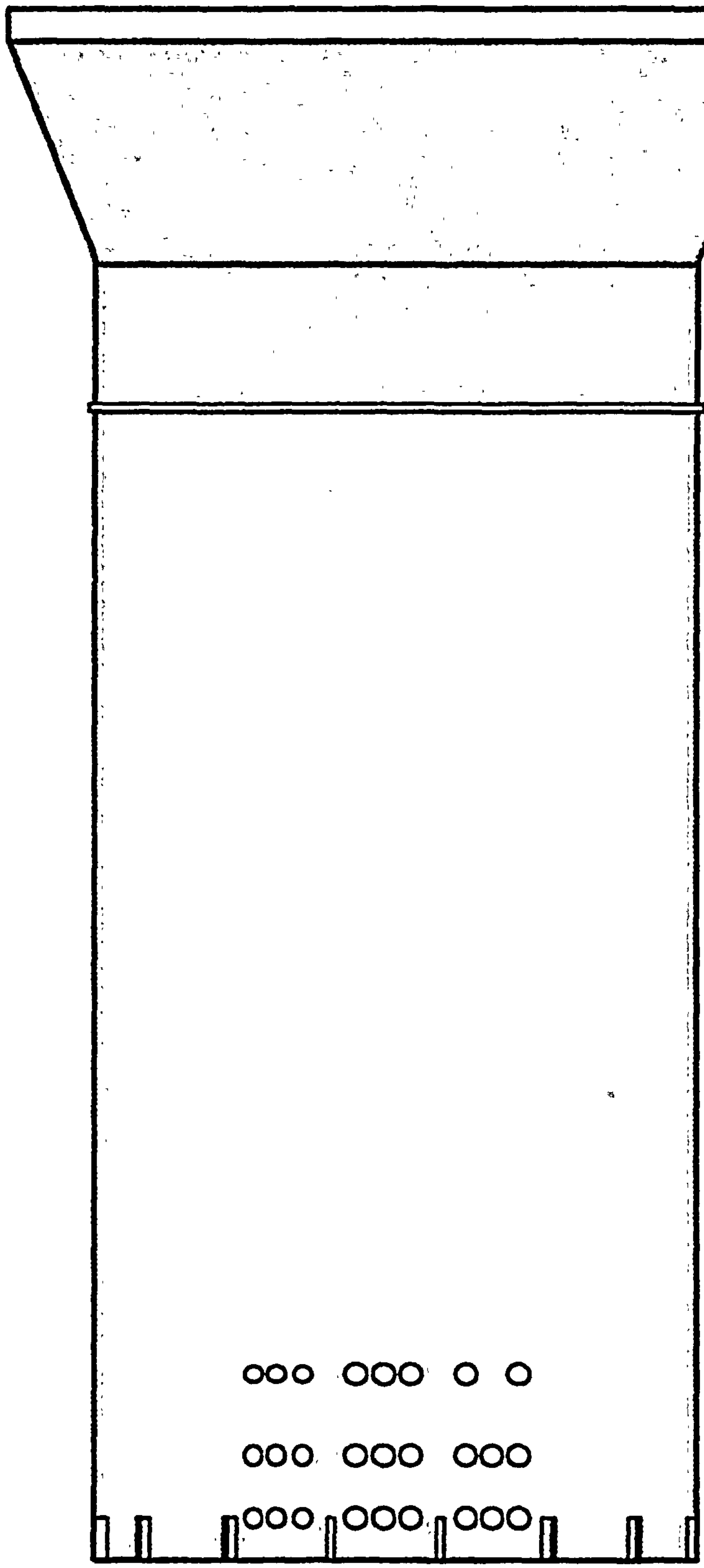
Palo Verde Steam Generator Shroud Modification Hole Configuration



7/20/95
/BAILEY3/SHROUD-1

FIGURE IV-13

Palo Verde Steam Generator 3-2
Shroud Modification Hole Configuration



11/15/95
/BAILEY3/SHROUD18

FIGURE IV-14

APTECH POD Analysis of Average Depth MRPC POD Data

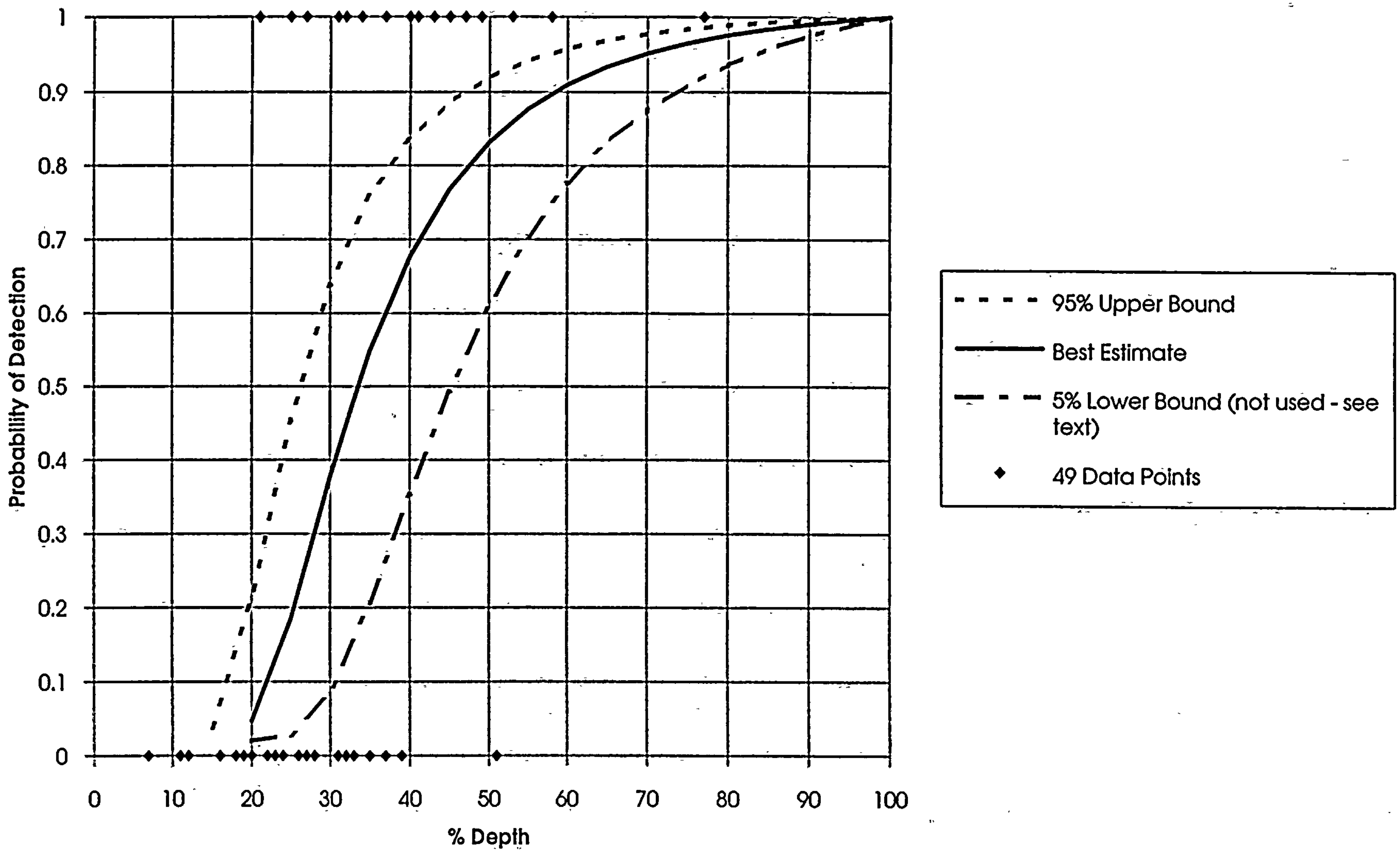


Figure V-1

APPENDIX A

Palo Verde Unit - 3
Run Time Analysis
Regarding the Impact
of Upper Bundle Corrosion
Degradation During Cycle 6

APTECH Engineering Services

AES 96022658-1-1
Revision 1
June, 1996

PALO VERDE UNIT 3 RUN TIME ANALYSIS REGARDING THE IMPACT OF UPPER BUNDLE CORROSION DEGRADATION DURING CYCLE 6

Prepared by

B. W. Woodman
J. L. Biffer
J. A. Begley

APTECH ENGINEERING SERVICES, INC.

Prepared for

Arizona Public Service
Palo Verde Nuclear Generating Station
P.O. Box 52034
Phoenix, AZ 85072-2034

Attention: Mr. Kevin Sweeney

TABLE OF CONTENTS

EXECUTIVE SUMMARY

An evaluation of the significance of upper bundle corrosion degradation on the structural integrity of Alloy 600 steam generator tubing at Palo Verde Nuclear Generation Station Unit 3 was performed. Axial, outer diameter stress corrosion cracking/intergranular attack (ODSCC/IGA) at upper bundle freespan locations was considered. Upper bundle freespan corrosion degradation has been observed in Unit 3 in the past three inspections. This degradation is proceeding at a relatively slow rate.

A probabilistic run time model was employed. The processes of crack initiation, crack growth and eddy current inspection were modeled in a Monte Carlo simulation. Predictions of the number of Plus Point probe eddy current indications were benchmarked against actual observations. End of cycle conditions were projected for cycle 6. Based on these projections, after 15.5 effective full power months (EFPM), the conditional probability of tube burst for a postulated main steam line break accident is estimated as considerably less than 10^{-4}. In terms of Regulatory Guide 1.121 structural margins, the probability of a structural limit exceedance after 15.5 EFPM is less than 10^{-4}. In over 10,000 Monte Carlo simulations, no instance of through-wall crack penetration was observed. Leakage at postulated accident conditions is not an issue. In terms of the significance of corrosion degradation at upper bundle freespan locations, full cycle operation is strongly supported.

Section 1
INTRODUCTION

In the last three eddy current inspections of steam generator tubing at Palo Verde Unit 3, indications of corrosion degradation have been observed at upper bundle freespan locations. As confirmed by pulled tubes from Unit 2, these eddy current indications are attributed to ongoing outer diameter stress corrosion cracking/intergranular attack (ODSCC/IGA) processes (1-5). To date, the cumulative number of tubes involved is approximately 76. No detected degradation has challenged Regulatory Guide 1.121 structural margin requirements. The progression rate of upper bundle corrosion degradation is relatively slow, both in terms of cumulative number of tubes involved and in terms of the growth rates of indication depths.

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).

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 are 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 6 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 REQUIREMENTS AND GENERAL APPROACH

Typically the limiting Regulatory Guide 1.121 structural requirement relative to the significance of corrosion degradation in steam generator tubing is maintaining an end of cycle burst pressure greater than three times the normal operating pressure differential across the tube wall. This is the circumstance for Palo Verde Unit 3. The required end of cycle (EOC) burst pressure is 3800 psi. Since the degradation mode of interest is axial ODSCC/IGA, the burst pressure is a function of the degradation length, average depth and tensile properties of the tubing. Figure 2.1 shows a plot of calculated versus measured burst pressures for pulled tubes from Unit 2 (1, 2, 3, 4, 5). Calculated burst pressures are based on the Framatome equation (6).

$$P = \frac{0.58(\sigma yp + \sigma uts)t}{R_i}\left[1 - \frac{L\frac{d}{t}}{L + 2t}\right]$$

where

P = burst pressure

σyp = yield strength

σuts = tensile strength

t = tube wall thickness

R_i = inner radius of tube

L = degradation length

d = average degradation depth

This equation gives conservative predictions of burst pressure when the average crack depth is based on observed crack depths over the total crack length. If the average crack depth is computed over the structurally significant crack length, then essentially no under conservative calculated burst pressures are observed and there is no associated large penalty of over conservatism. If the crack depth versus length profile is known, then the structurally significant crack length is determined by repeated sampling of the total crack profile to find the crack segment which dominates the burst pressure (1, 5).

Figure 2.2 shows that structurally significant crack lengths correlate with degradation length as indicated by a rotating pancake eddy current probe. The rotating pancake coil (RPC) crack length is either equal to or greater than the structurally significant crack length. If Plus Point probe indications are used for length sizing, then the indicated lengths are substantially larger than the structurally significant lengths. Use of Plus Point indicated lengths is clearly overconservative.

If crack depths are averaged over the structurally significant crack length, then the ratio of maximum crack depth to structurally significant crack depth is on the order of $4/\Pi$. That is, the structurally significant portion of a seemingly irregular crack profile can be idealized as a semi-ellipse. Figure 2.3 shows a plot of maximum crack depth versus average depth from pulled tube data of Unit 2. This figure illustrates that the appropriate average crack depth for burst pressure calculations can be obtained from a measured or estimated maximum crack depth. Figure 2.1 points to the success of this technique.

With the above information and knowledge of lower bound tubing mechanical properties, a deterministic Regulatory Guide 1.121 analysis can be summarized in simple graphical form. Figure 2.4 plots maximum crack depth

versus effective full power months. For illustrative purposes, the assumed beginning of cycle (BOC) maximum crack depth is 50% of the tube wall thickness. Two crack growth rates illustrate an average value and an upper bound value. The upper horizontal lines denote maximum depth Regulatory Guide 1.121 structural limits for various assumed crack lengths. Note that these limits are based on 95% lower bound tensile properties. The length limit for a through wall crack is 0.47 inches. A through-wall crack length of 0.76 inches is required for tube burst under a postulated steam line break pressure differential of 2400 psi.

The deterministic inputs to Figure 2.4 can be changed to selections from known distributions of tensile properties and crack growth rates. Crack lengths can be selected from past eddy current measured distributions. Finally, BOC crack depths can be input from a model which follows crack initiation, crack growth, and then periodic inspections with defined probabilities of detecting cracks. Undetected cracks then remain in service and form the BOC population. A model of this type has been constructed and used with excellent success in the past (5).

The Monte Carlo simulation model is described briefly in Section 4. In contrast to previous versions of this model, some concepts from NRC Generic Letter 95-05 (7) have been incorporated. Degradation growth rates are sampled directly from past observations without using a fitted analytic distribution. Also, negative growth rates are treated as zero growth. As noted in the Introduction, growth rates at a given location can vary from one cycle to the next. Hence, a crack which escapes detection in one cycle is assigned a new growth rate from the growth rate distribution. This feature is consistent with crevice conditions which may change from cycle to cycle even though the bulk solution chemistry is relatively constant. Cycle dependent growth rates for a given indication and the possibility of zero

growth rates combine to produce end of cycle distributions of degradation depths which provide an improved match with actual observations. Work in this area is continuing but the general result is a better representation of an occasional appearance of a very deep crack.

Model improvement is an ongoing process. One obvious area with regard to Unit 3 is simply more growth rate data. The limited Unit 3 specific growth rate data of the previous analysis required some conservative assumptions in the selection of an upper bound estimate. As better and more extensive information is utilized some changes in benchmarking calculations are to be expected. Other specifics of the present analysis are the use of a crack length distribution from pancake coil rather than Plus Point coil data and a probability of detection curve based on a summary of industry pancake coil data. As discussed in an earlier paragraph, pulled tube burst test results show that it is the length indicated by the pancake coil which is a conservative estimator of the structurally significant crack length. The better detection capability of the Plus Point at smaller degradation depths leads to overly conservative estimates of significant degradation length.

The calculated probability of detection curve for the Plus Point probe based on inspection transient data from Unit 2 (5) will apparently be validated by recent pulled tube results. While this data is being collected and analyzed, the present Unit 3 analysis utilizes a more conservative probability of detection curve. Use of the industry pancake coil data summary yields a probability of detection curve which is more sensitively than the curve based on Unit 2 pulled tube pancake data but not as sensitive as the calculated Plus Point curve. Since the Plus Point probe was used for inspection in the previous Unit 3 cycle, use of the intermediate sensitivity POD curve is an additional element of conservatism.

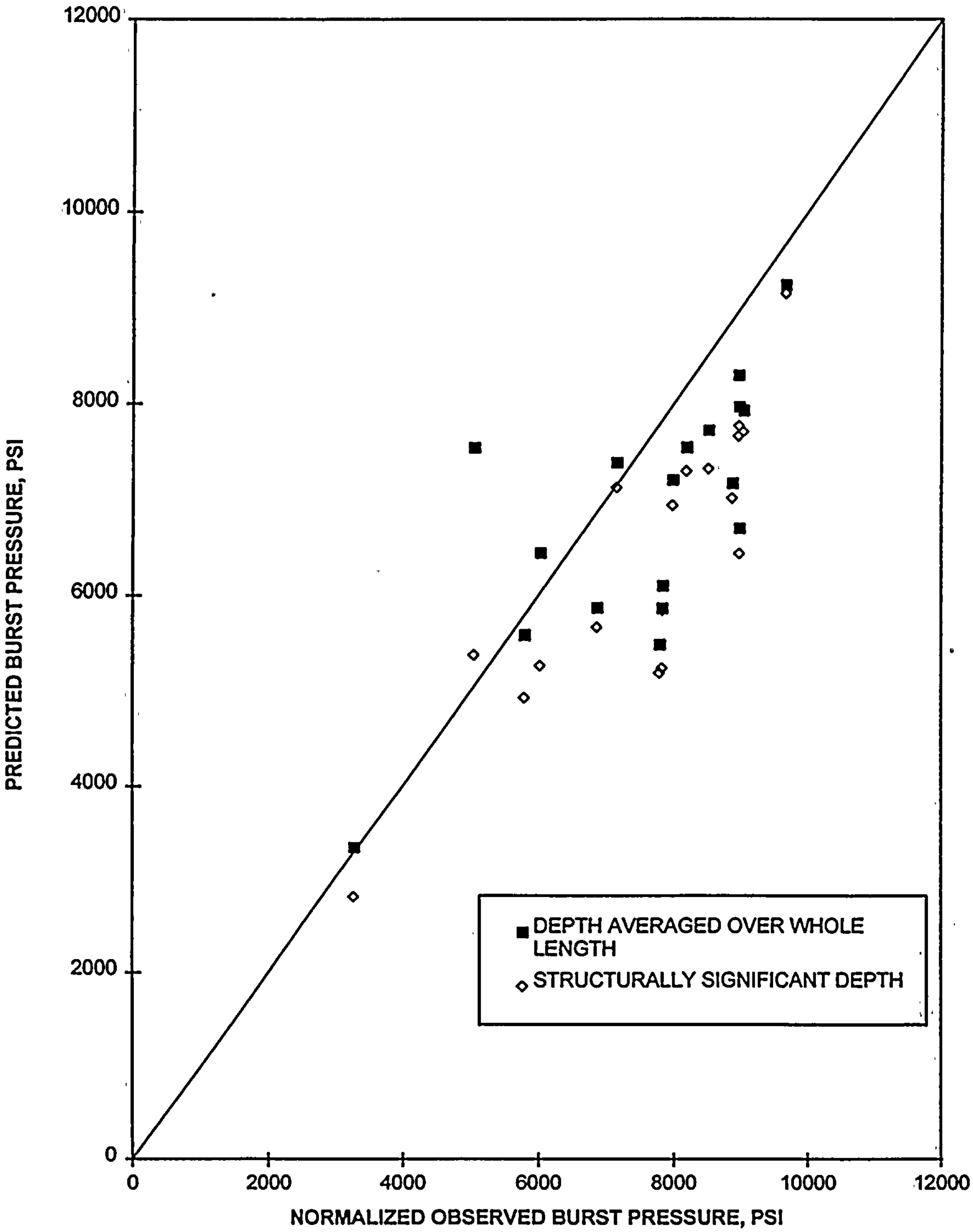


Figure 2.1 Predicted Burst Pressure Versus Normalized Observed Burst Pressure, PVNGS Unit 2 Pulled Tube Data.

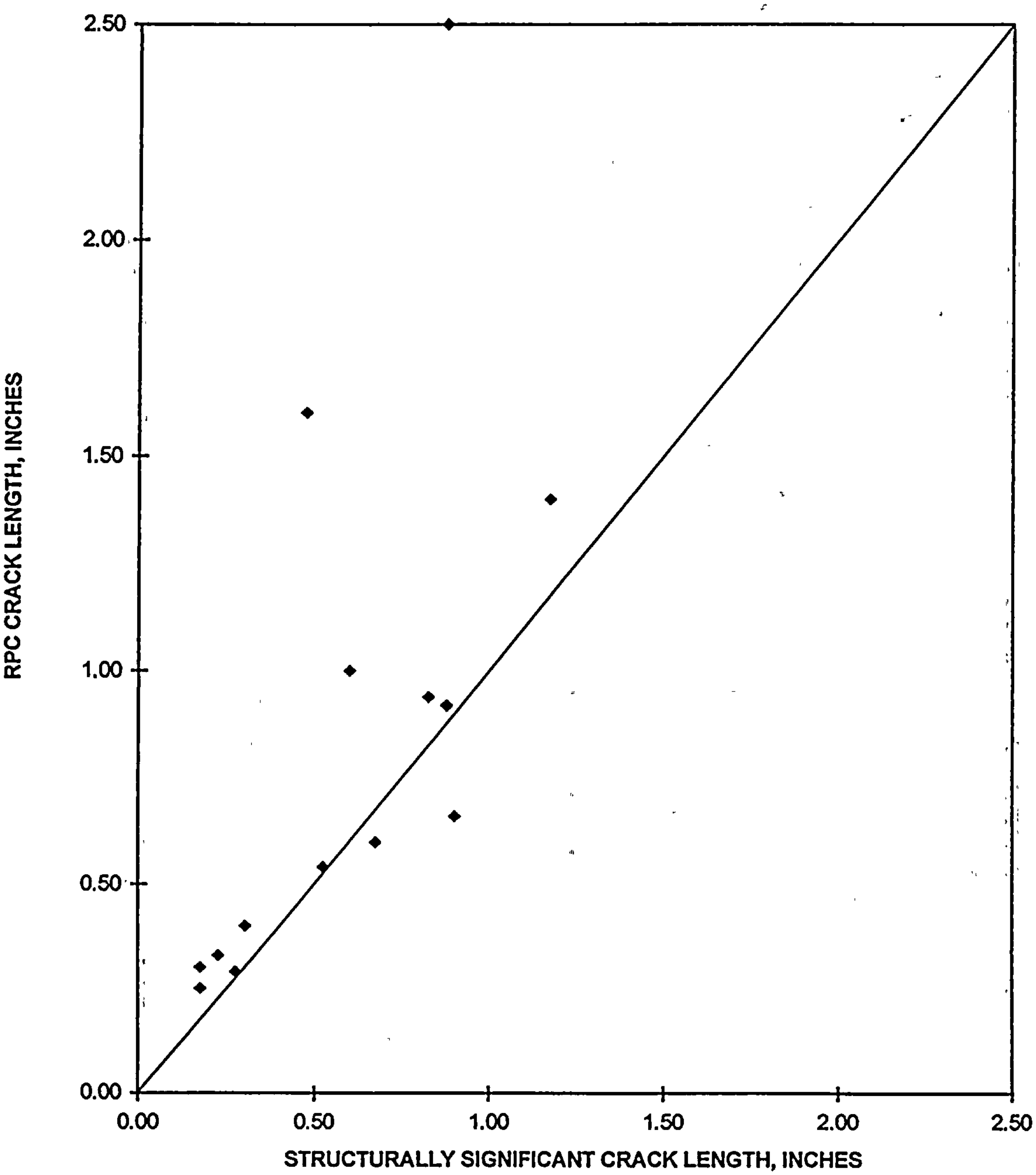


Figure 2.2 RPC Crack Length Versus Structurally Significant Crack Length, PVNGS Unit 2 Pulled Tube Data.

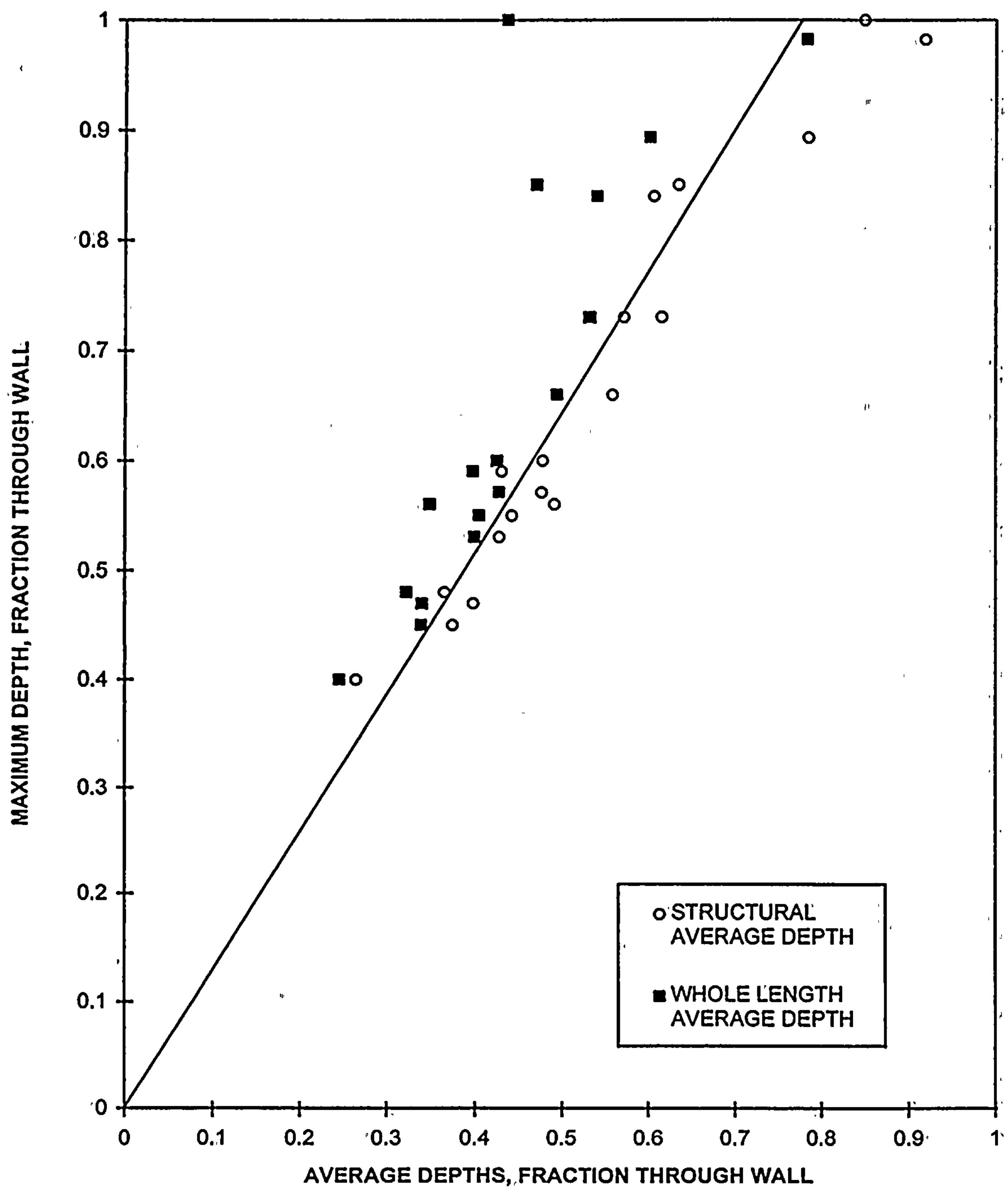


Figure 2.3 Maximum Crack Depth Versus Average Crack Depth, PVNGS Unit 2 Pulled Tube Data.

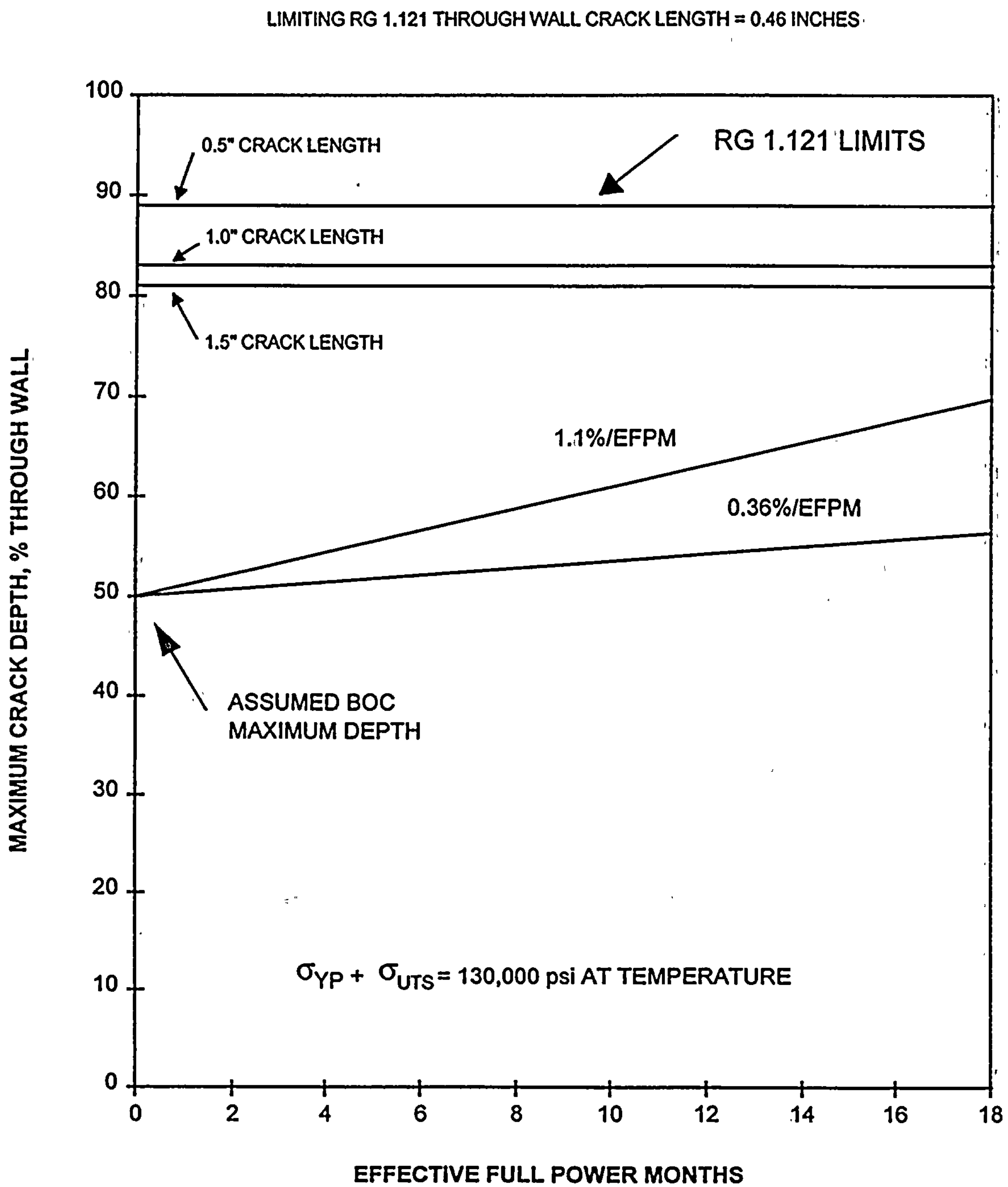


Figure 2.4 Schematic of Deterministic RG 1.121 Analysis.

Section 3

ANALYSIS INPUT PARAMETERS

The following paragraphs describe the input parameters to the probabilistic structural evaluation model. These parameters include tubing tensile properties, the distribution of degradation lengths, the probability of detection function used for the present analysis, and the distribution for crack growth rates.

3.1 TUBING MECHANICAL PROPERTIES

The actual yield strength and ultimate tensile strength values were available for Palo Verde Unit 3 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 3.1 shows the probability distribution function used for Unit 3 steam generator tubing in the simulation. This is identical to the normal distribution used in the previous Unit 3 analysis (8).

3.2 DEGRADATION LENGTH DISTRIBUTION

The distribution of cracks length judged to be the most appropriate for Palo Verde Unit 3 is based on an upper bundle RPC 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. The Weibull distribution parameters are a slope of 1.11 and a scale factor of 0.69 inches. These parameters were used in the

previous Unit 3 analysis (8). The axial length distribution used in the simulation is shown in Figure 3.2

3.3 PROBABILITY OF DETECTION

Industry-wide data has been used in the past to compute the probability of detection versus maximum relative axial crack depth for rotating pancake probes (Figure 3.3). As seen in the figure, pulled tube data from Palo Verde Unit 2 leads to a comparable but somewhat less sensitive probability of detection (POD) curve. Palo Verde Unit 3 inspections have utilized the plus point probe for the most recent two inspections. While very recent pulled tube data will apparently validate the calculated Plus Point POD current from the Unit 2 inspection transient data, use of the industry-wide POD function for the current Unit 3 analysis is selected as an element of conservatism.

3.4 DEGRADATION GROWTH RATES

The development of crack growth rate data is a dominant element in a run time analysis. A data set consisting of 66 paired observations of RPC voltage is available (9) to construct an appropriate crack growth rate distribution. Data from both Unit 3 steam generators for the past two inspections has been used. Individual voltage growth rate data points are simply the difference in voltage estimates from RPC data divided by the operating period of 0.83 EFPY.

Voltage growth rates versus estimated BOC voltages are plotted in Figure 3.4. Approximately 15% of the voltage growth rates are negative or zero due to measurement uncertainties. If only the positive values are considered, the average voltage growth rate is 0.21 VOLTS/EFPY. The maximum observed

voltage growth rate is 0.52 VOLTS/EFPY. A histogram of the positive voltage growth rate data is shown in Figure 3.4.

As in previous Unit 3 and Unit 2 analyses (5, 8), crack growth rates are inferred from voltage growth rates using a correlating function developed from pulled tube data. The average factor applied to the voltage growth is 16.2% through-wall/volt. A stochastic error term of 5.4 % through-wall/volt is applied to accommodate correlation uncertainty. The resulting crack growth rate distribution function is compared with that obtained for Unit 3 in the previous inspection (for a more limited data set) in Figure 3.5. As can be seen from the figure, the Unit 3 crack growth rates are relatively benign and can be expected to not effectively limit run time.

The positive voltage growth rates of Figure 3.4 were sampled directly in the probabilistic analysis. No distribution was fitted to the positive data points. The correlating factor (volts-to-depth) was modeled stochastically in the simulation to obtain fully randomized crack growth rates. That is, once a voltage growth value was selected for use, it was converted to a crack depth growth rate by sampling from a distribution of slopes of depth versus RPC voltage. The distribution of slopes was obtained by statistical analysis of the Unit 2 pulled tube data of crack depth versus RPC voltage. Multiplication of voltage growth times slope equals crack growth. Crack growth rates were allowed to vary from one cycle to the next. In the sampling process, negative growth rates were considered as zero. Thus in a given cycle, approximately 16% of the crack population was assigned zero growth.

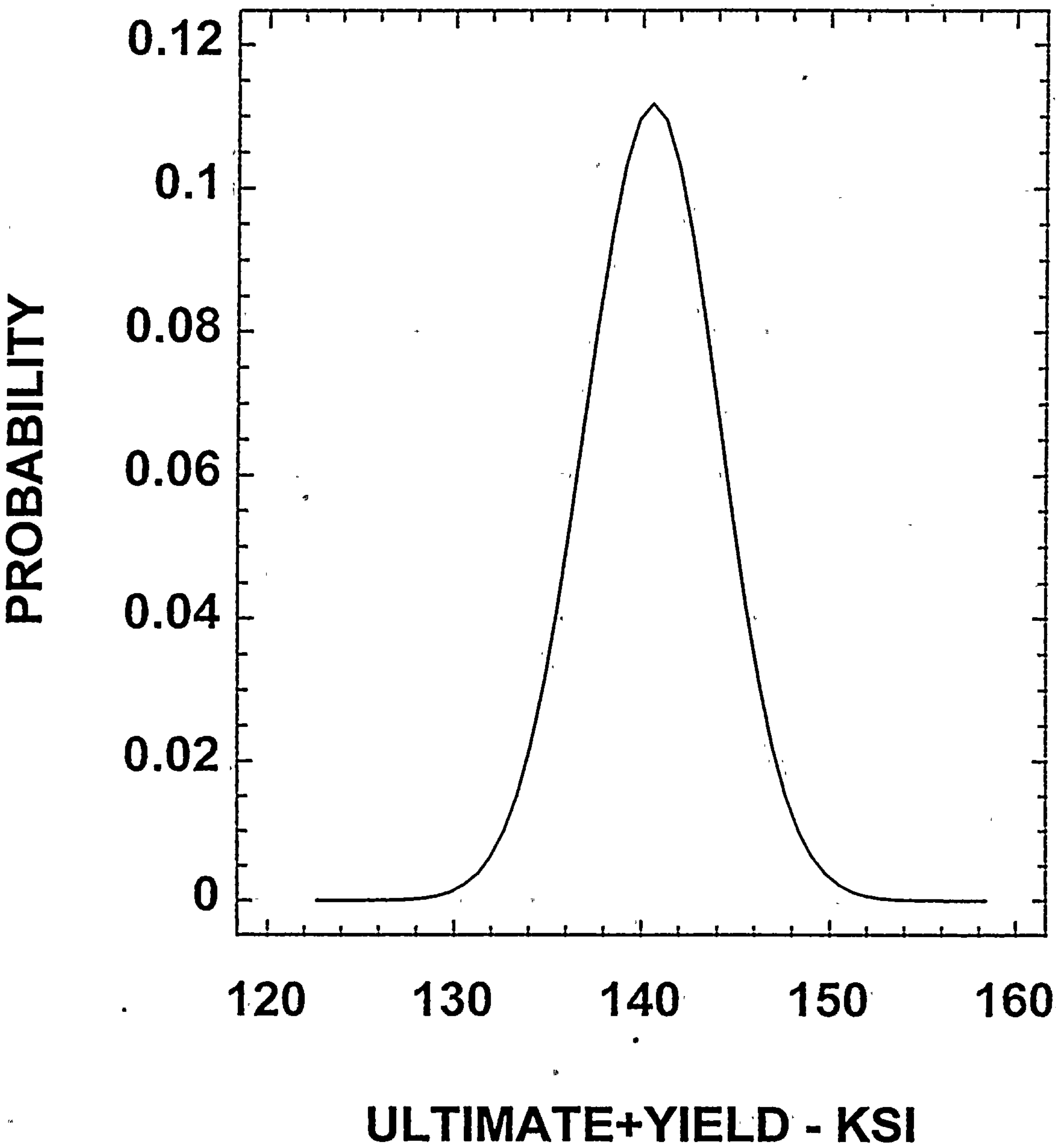


Figure 3.1 Palo Verde Unit 3 Tube Strength Distribution Function.

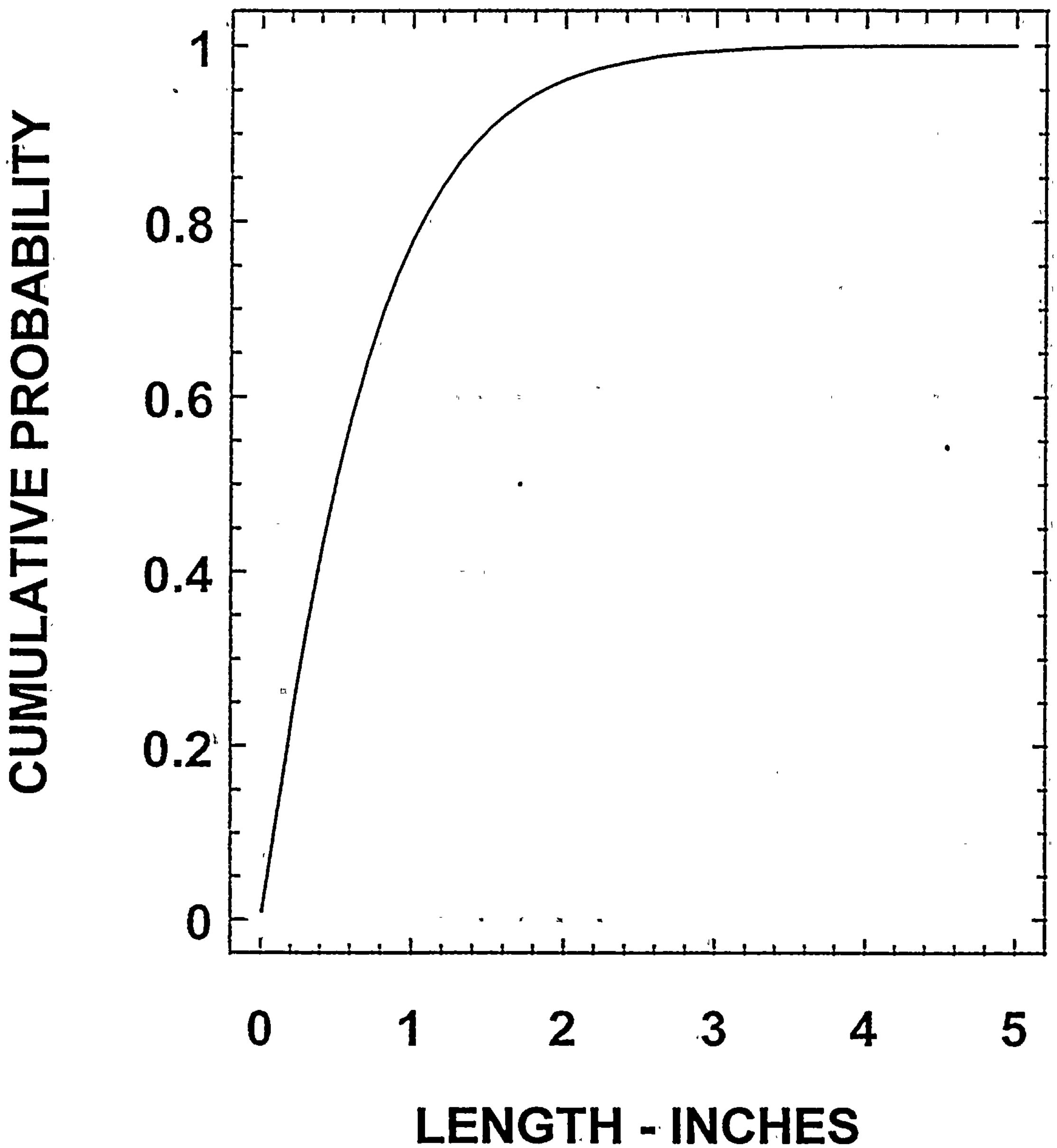


Figure 3.2 Palo Verde Unit 3 Distribution Function for EOC Crack Length.

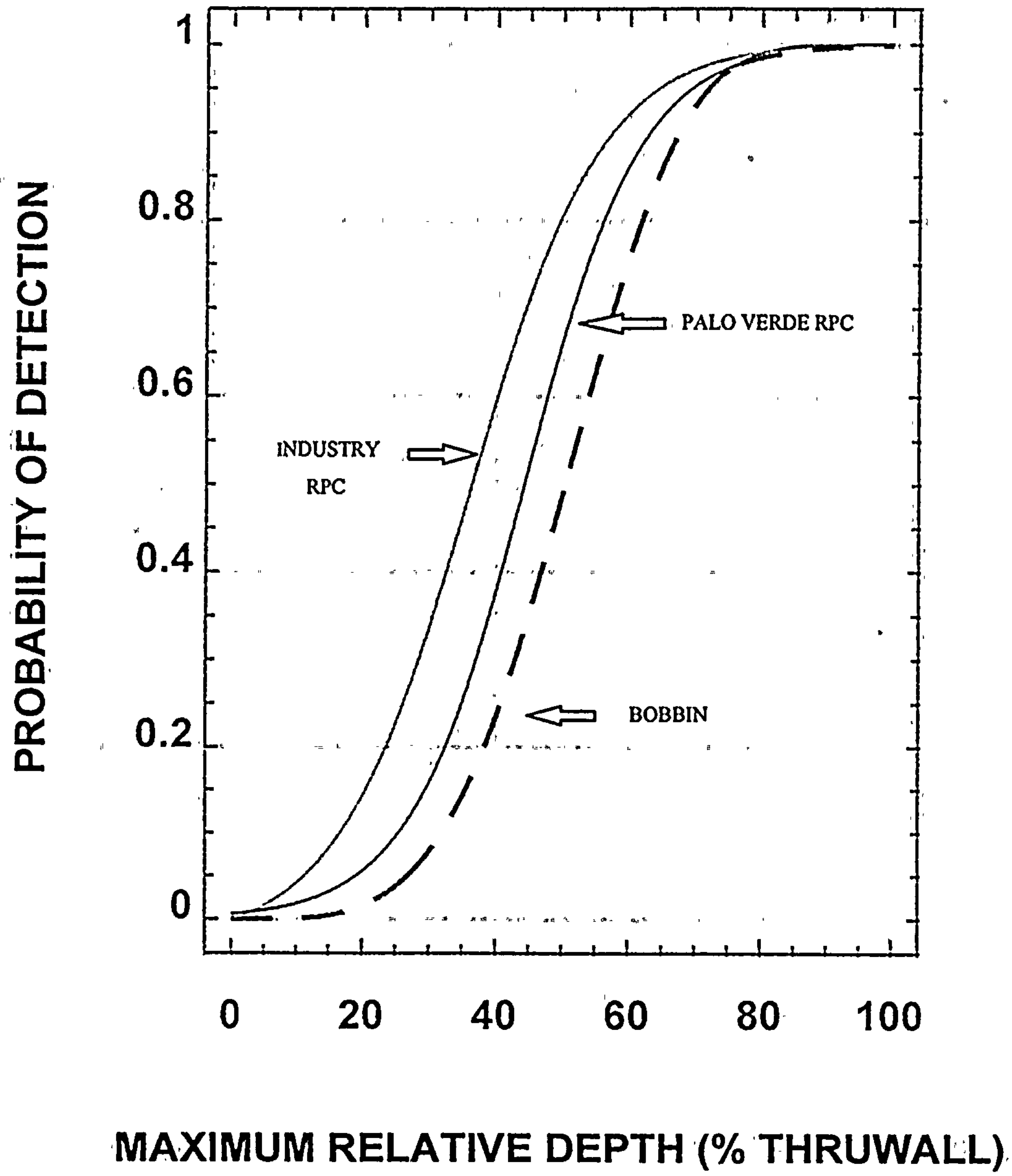


Figure 3.3 Comparison of Probabilities of Detection.

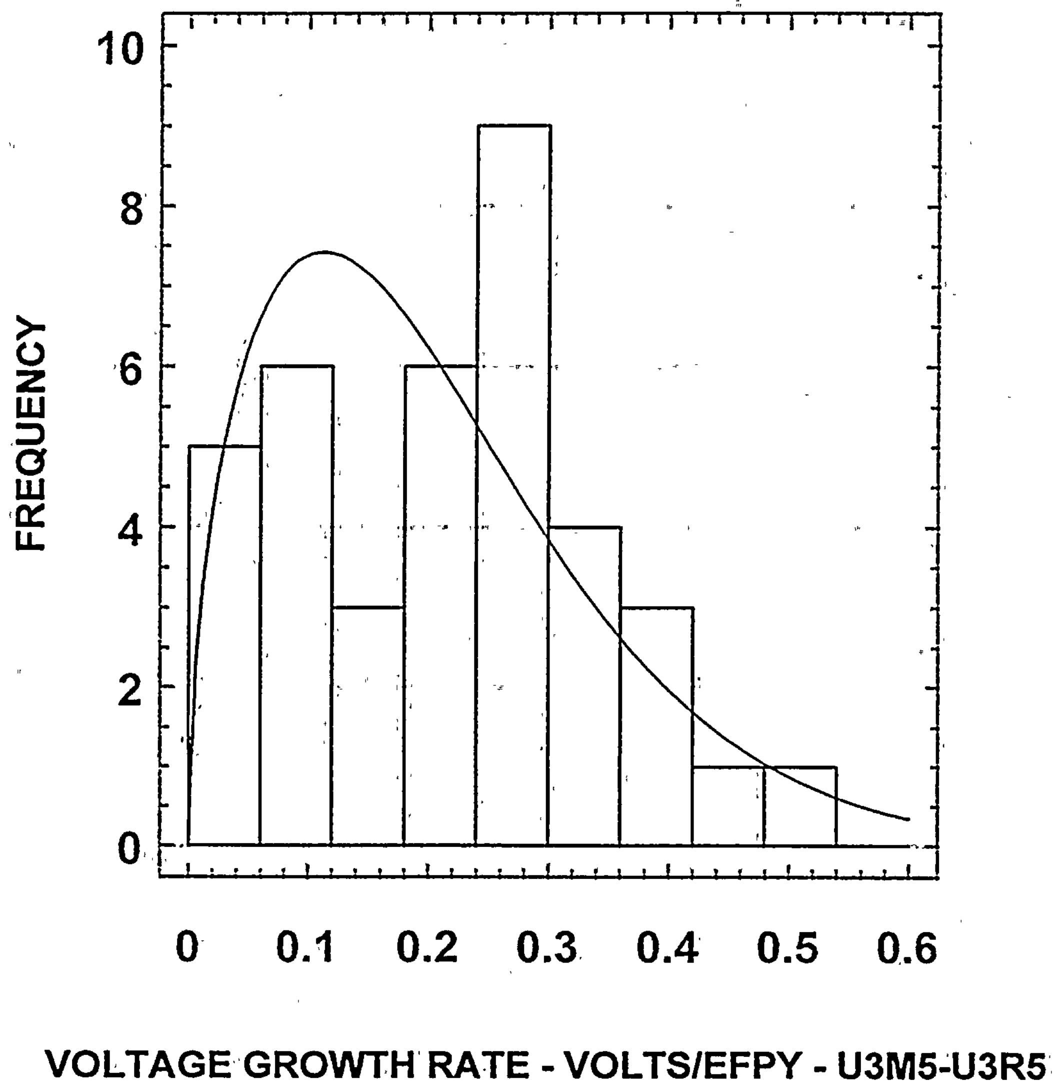


Figure 3.4 Palo Verde Unit 3 Voltage Growth Rates.

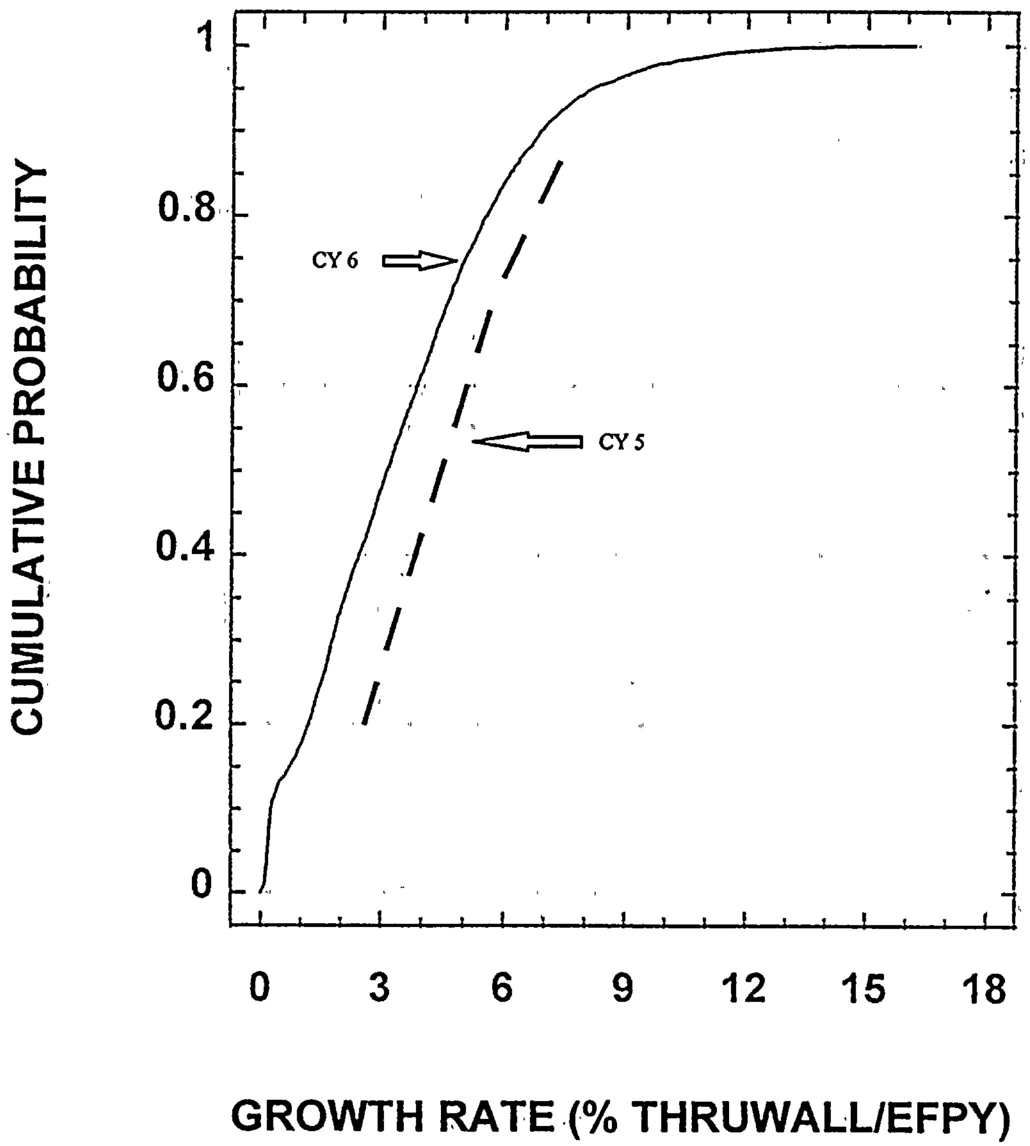


Figure 3.5 Comparison of Palo Verde Unit 3
Cycle 5 and Cycle 6 Growth Rates.

Section 4
PROBABILISTIC MODEL

The probabilistic run time model consists of a Monte Carlo simulation of the processes of crack initiation, crack growth, eddy current inspection, and removal or repair of degraded tubes which are detected. The basic simulation model has been in use for several years (5,8) and is being continually upgraded and refined.

Times to crack initiation 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 depends on the average crack growth rate.

After a crack is initiated, it is considered to grow at a constant rate in the depth direction until the next inspection. As noted in the previous section, the assigned dependent crack growth rate is random by selection from the observed voltage growth rates in Figure 3.4 in conjunction with the voltage/depth correlating function. If a crack survives detection in the simulated inspection, a new growth rate is selected for the next cycle. The new growth rate may zero, equal to or different from the old growth rate.

Crack length and tensile properties are assigned to a crack at initiation and are considered intrinsic properties of the crack. Analysis of the observed distributions of EOC structural significant crack lengths at PVNGS from mid cycle as well as end of cycle inspections show that these distributions are

reasonably constant. This statement is true for other plants and types of degradation. In general, run time has little effect on the length distribution. Naturally, the number of cracks at a given length does depend on the cycle run time.

The probability of detection (POD) function of Figure 3.3 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. Selection of the POD curve was discussed in Sections 2 and 3.

Upon detection, the length, depth, 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. Burst pressure calculations are discussed in Section 2. More extensive details are provided in a separate calculation note (10). Crack depths required for bursting at main steam line break (MSLB) or three times normal operation pressure (3ΔP) differential are large enough to virtually insure detection. 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.

One benchmark of the probabilistic model is a 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 4.1 to 4.4. The actual number of indications observed match up well with the most likely calculated number of indications for U3R4, U3M5, and U3R5 inspections. Approximately 150 new indications are predicted for the U3R6 inspection. The U3R6 predictions are probably very pessimistic reflecting the usage of a very conservative Weibull initiation model as used in the previous Unit 3 analysis (Figure 4.5). This initiation function has not been optimized for the new distribution of growth rates and a less sensitive POD function. As noted earlier, use of new and more extensive data and better representations of cycle to cycle growth will lead to some differences in benchmarking comparisons.

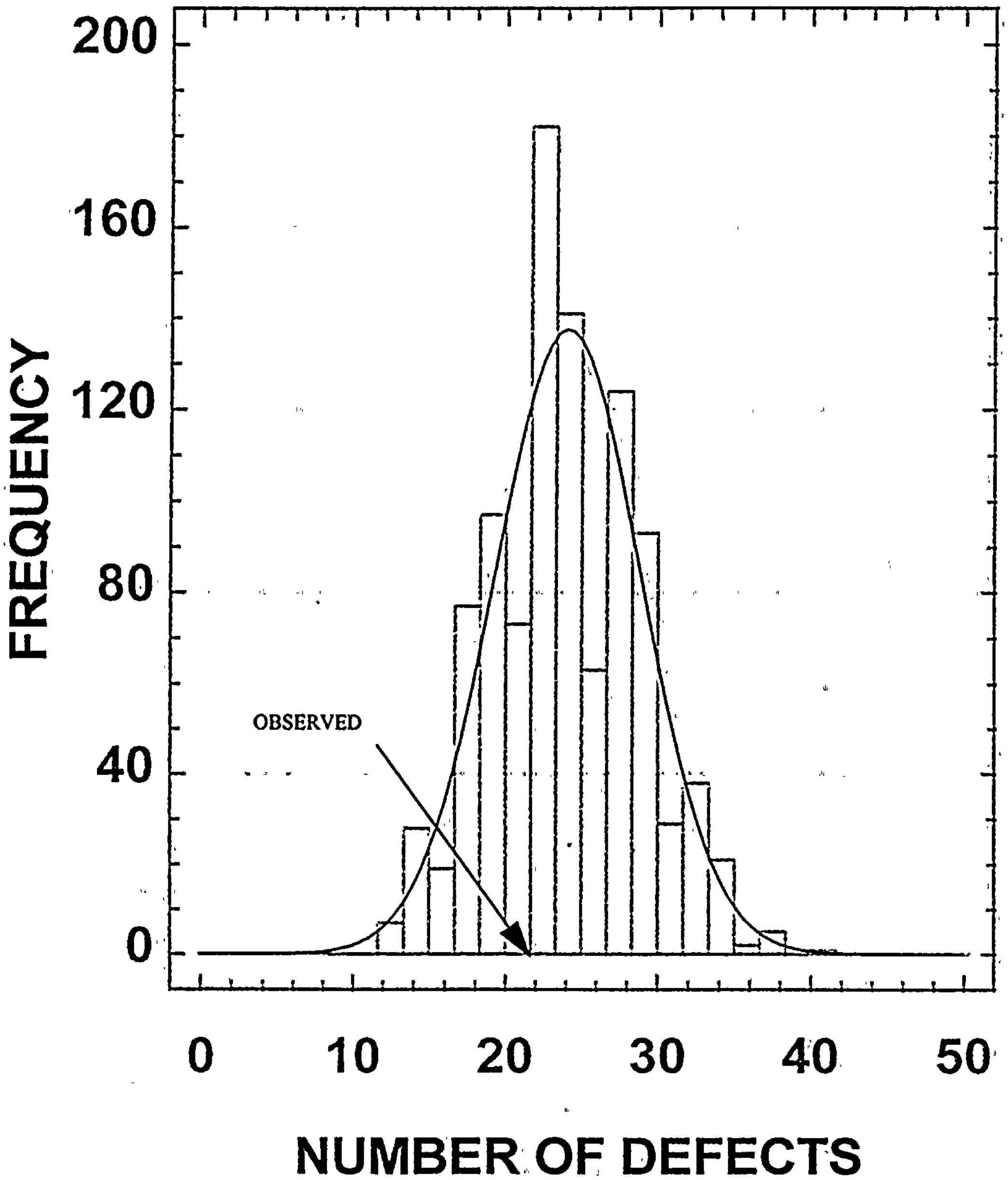


Figure 4.1 Predicted Number of Defects, U3R4.

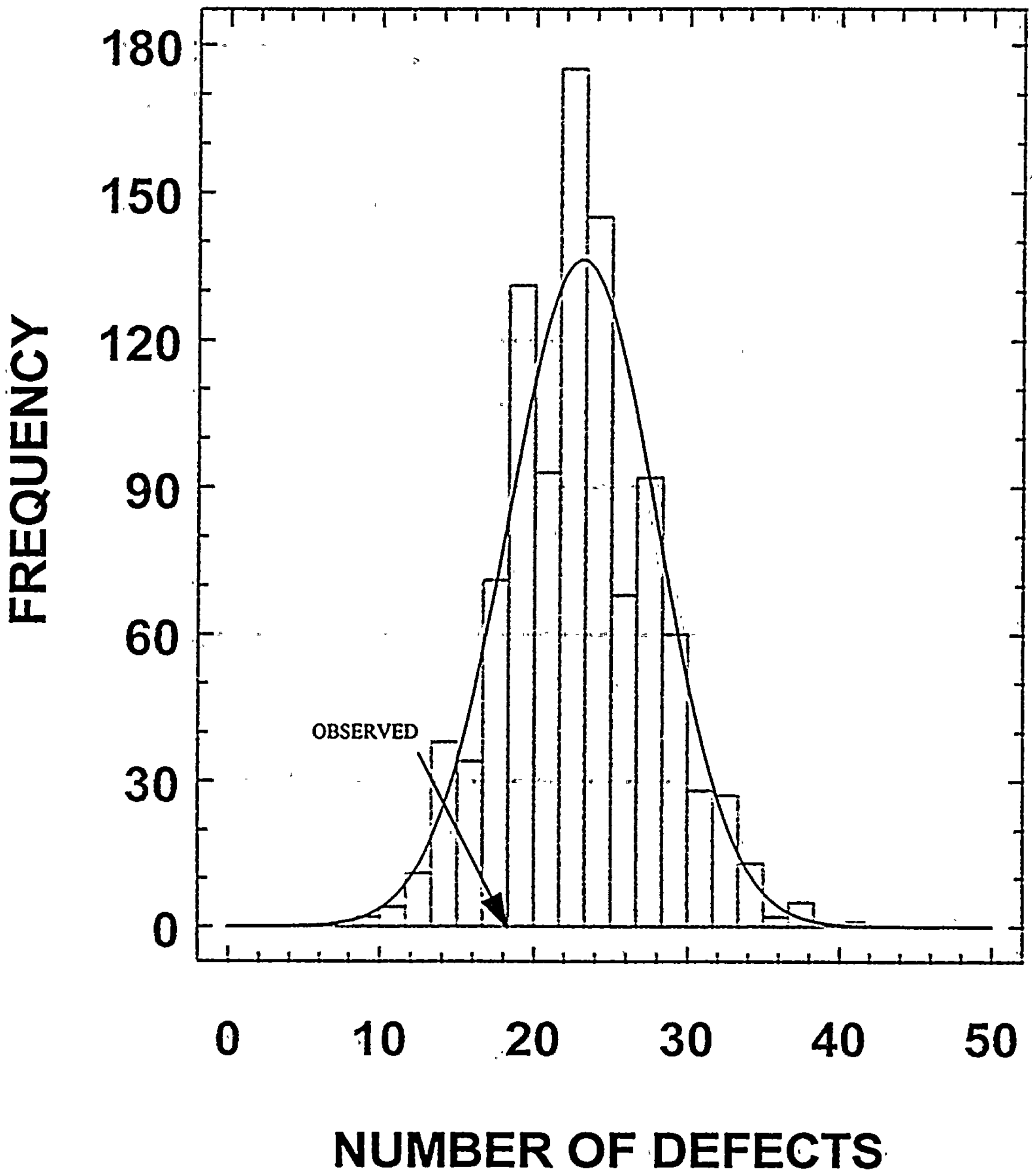


Figure 4.2 Number of Defects, U3M5.

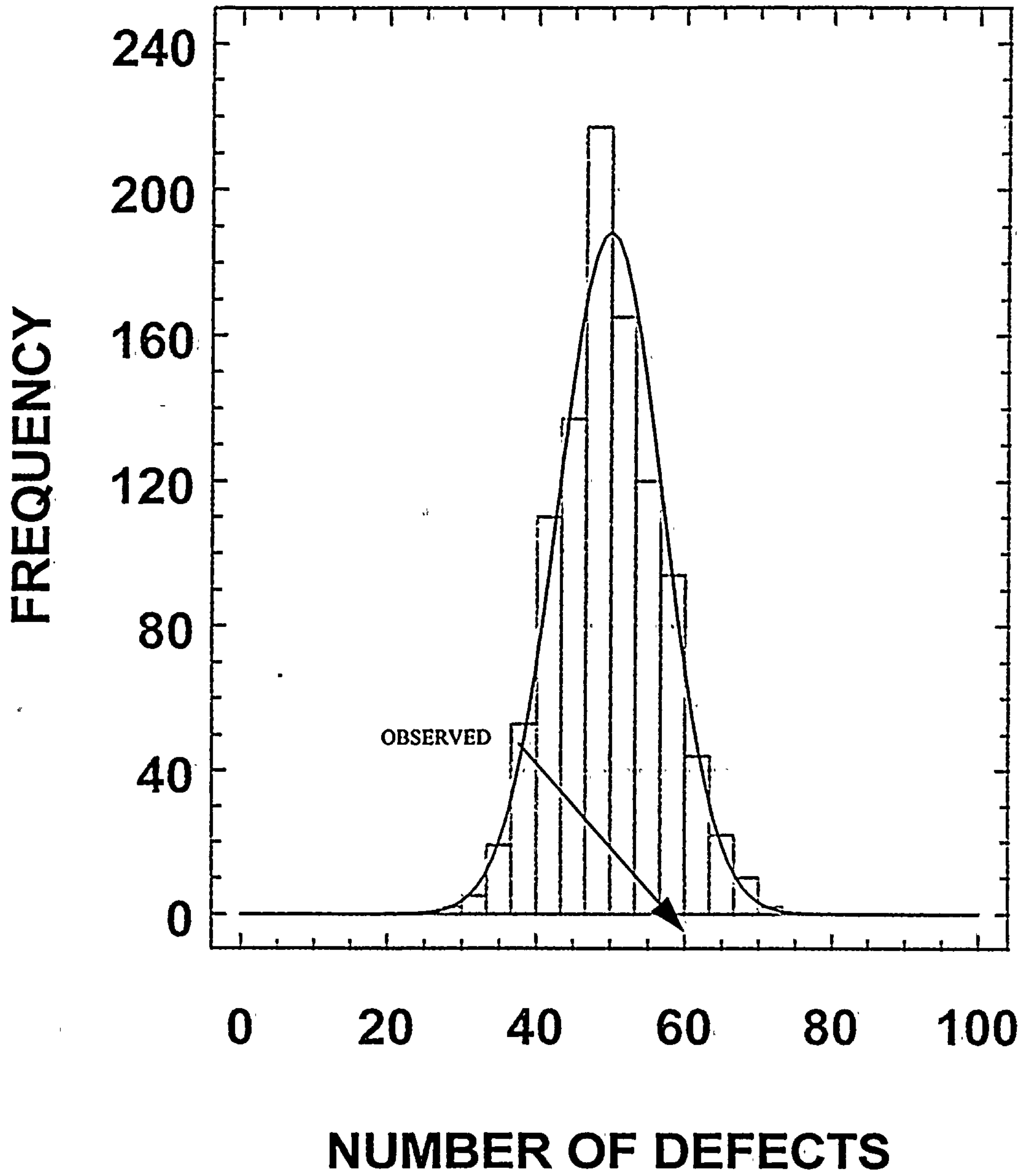


Figure 4.3 Number of Defects, U3R5.

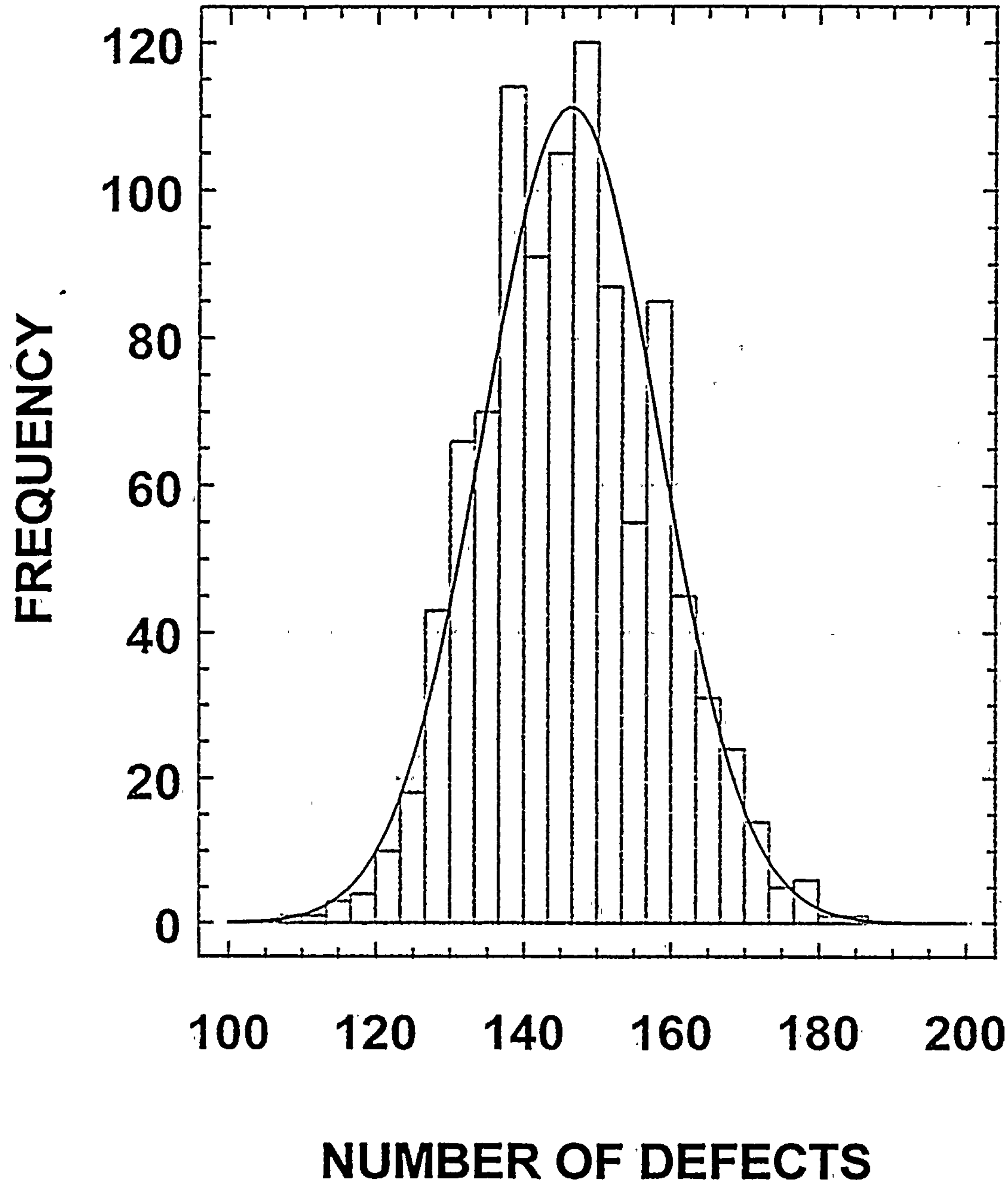


Figure 4.4 Number of Defects, U3R6.

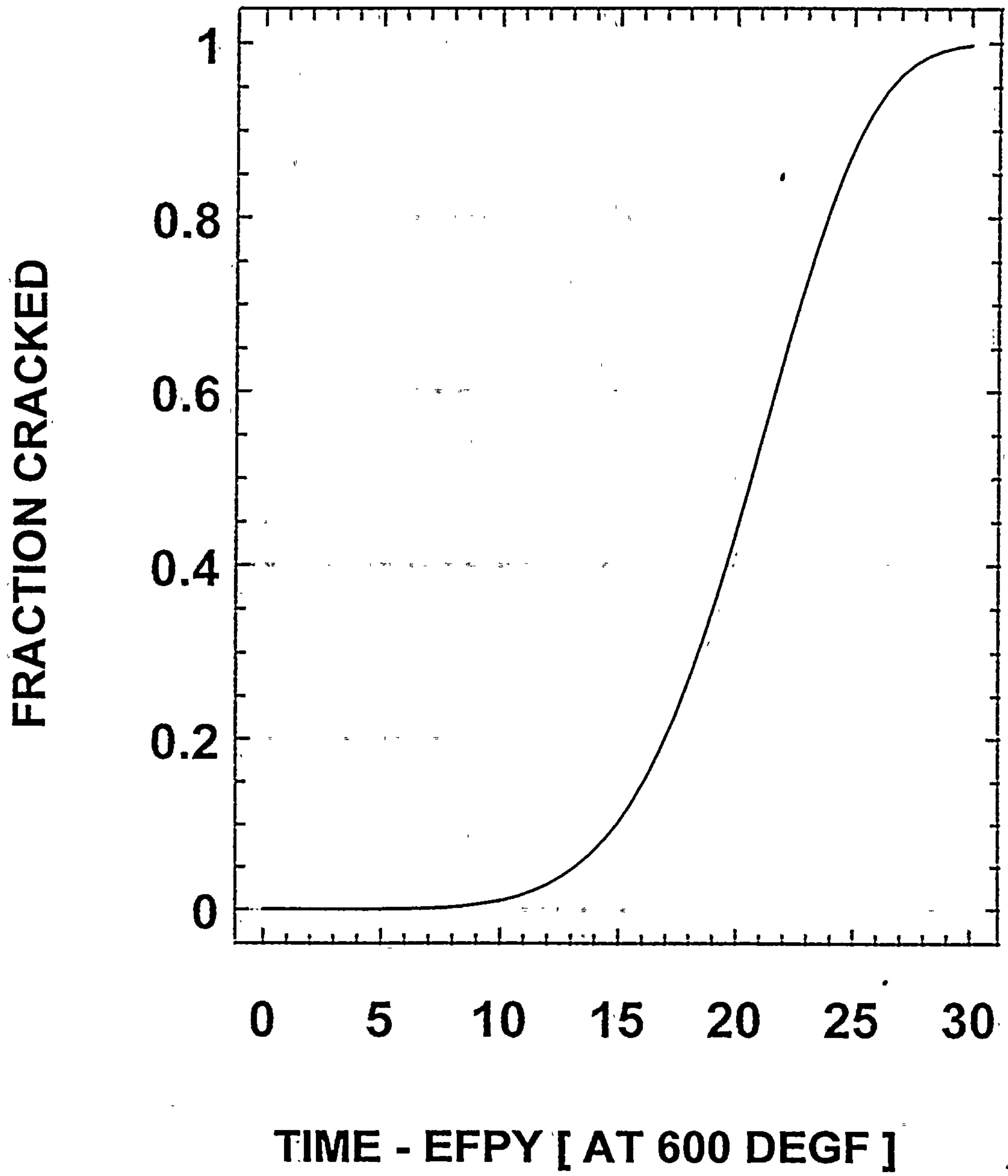


Figure 4.5 Palo Verde Unit 3 Crack Initiation Function.

Section 5

STRUCTURAL MARGIN EVALUATION

The parameters of interest in a probabilistic structural margin evaluation are the conditional probability of tube burst given a postulated main steam line break at end of cycle and the likelihood of multiple RG 1.121 structural limit exceedances. The expected leak rate under accident conditions is another important consideration. Both structural margin and leak rate evaluations are presented in the following paragraphs.

Relative to structural integrity, a reasonable figure of merit for conditional probability of an MSLB tube burst is provided by a historical value of 0.050. A value less than 0.01 provides a good benchmark of structural integrity. In terms of meeting Regulatory Guide 1.121 structural margins, a reasonable probabilistic criteria is a goal of at least 90% probability that one or fewer tubes will be expected to violate structural limits at end of cycle.

The Palo Verde Unit 3 analysis simulates operating cycle 6 as a 15.5 EFPM run with a hot-leg temperature of 611°F. The results of the Monte Carlo simulation model regarding the probability of a tube burst due to axial ODSCC/IGA at eggcrate support and upper bundle freespan locations given a postulated main steam line break showed no occurrences in 10,000 trials. The simulation under 3ΔP loading for the RG 1.121 study had the same outcome, thus the conditional probability of tube rupture in each case was less than 10^{-4}. This result is consistent with the cycle 5 analysis and the subsequent U3R5 inspection results, providing support for the 15.5 month anticipated operating period.

Since the situations of interest are obviously rare events, even for the limiting tube, the simulation code was modified to provide a special edit containing extreme value data for selected variables. The quantities of most interest were:

- Burst Pressure
- Average Crack Depth
- Maximum Crack Depth [Leakage Related]

The extreme value distribution for burst pressure is shown in Figure 5.1. The figure shows a histogram of the minima of the simulated burst pressures and an analytical fit derived from the data. The Type 3 (Weibull) extreme value distribution for minima was used for the analytical fit. As can be seen from the figure, the possibility of burst under SLB or $3\Delta P$ conditions is remote.

The extreme value distribution for average crack depth is shown in Figure 5.2. This figure compares the cumulative distribution functions (CDF's) of the simulated maxima with a theoretical asymptotic (Gumbel's Type 1 for maxima) distribution. As can be seen in the figure RG 1.121 challenging average crack depths are not achieved.

Figure 5.3 shows the extreme value distribution for the maxima of peak crack depths. This is important for leakage considerations since a through-wall crack is required for leakage to occur. As can be seen, the possibility of a through-wall crack in Palo Verde Unit 3 and subsequent leakage is remote.

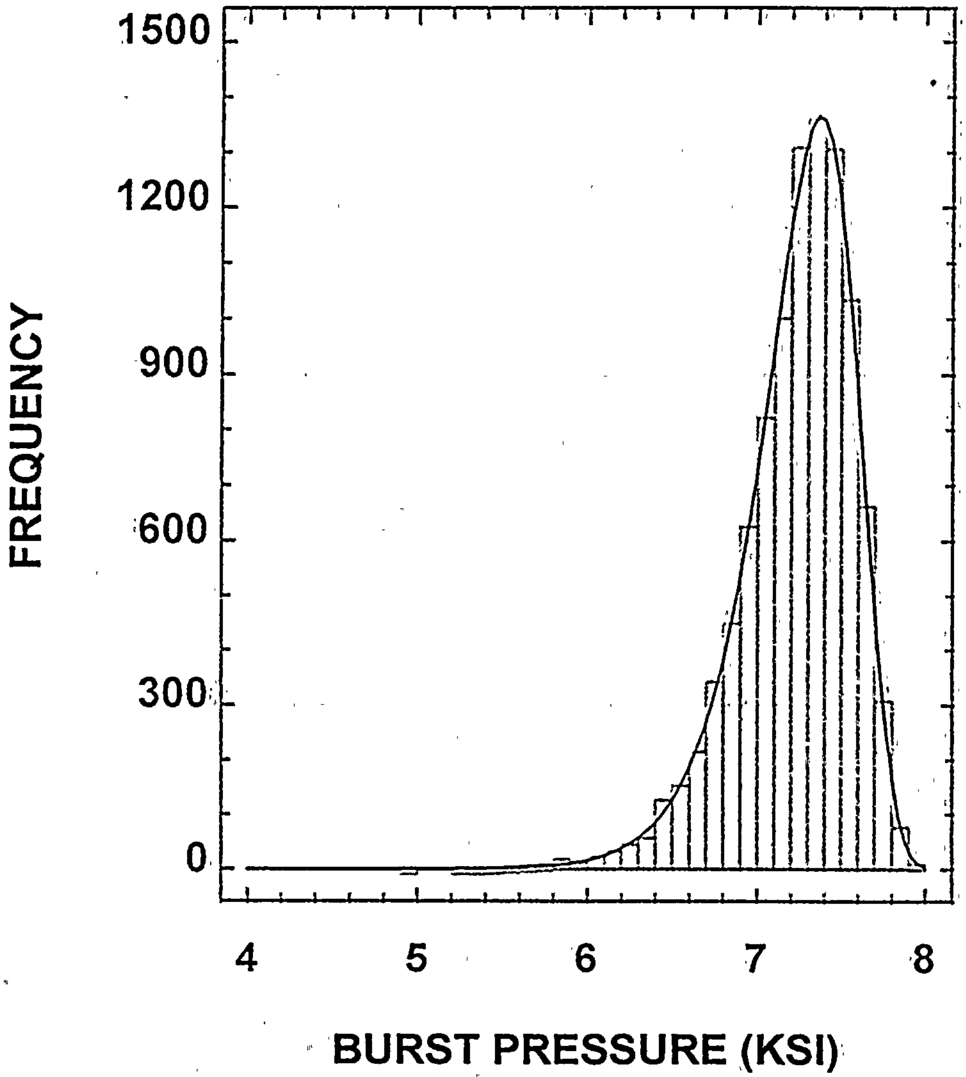


Figure 5.1 Extreme Value Probability Distribution Function for Burst Pressure.

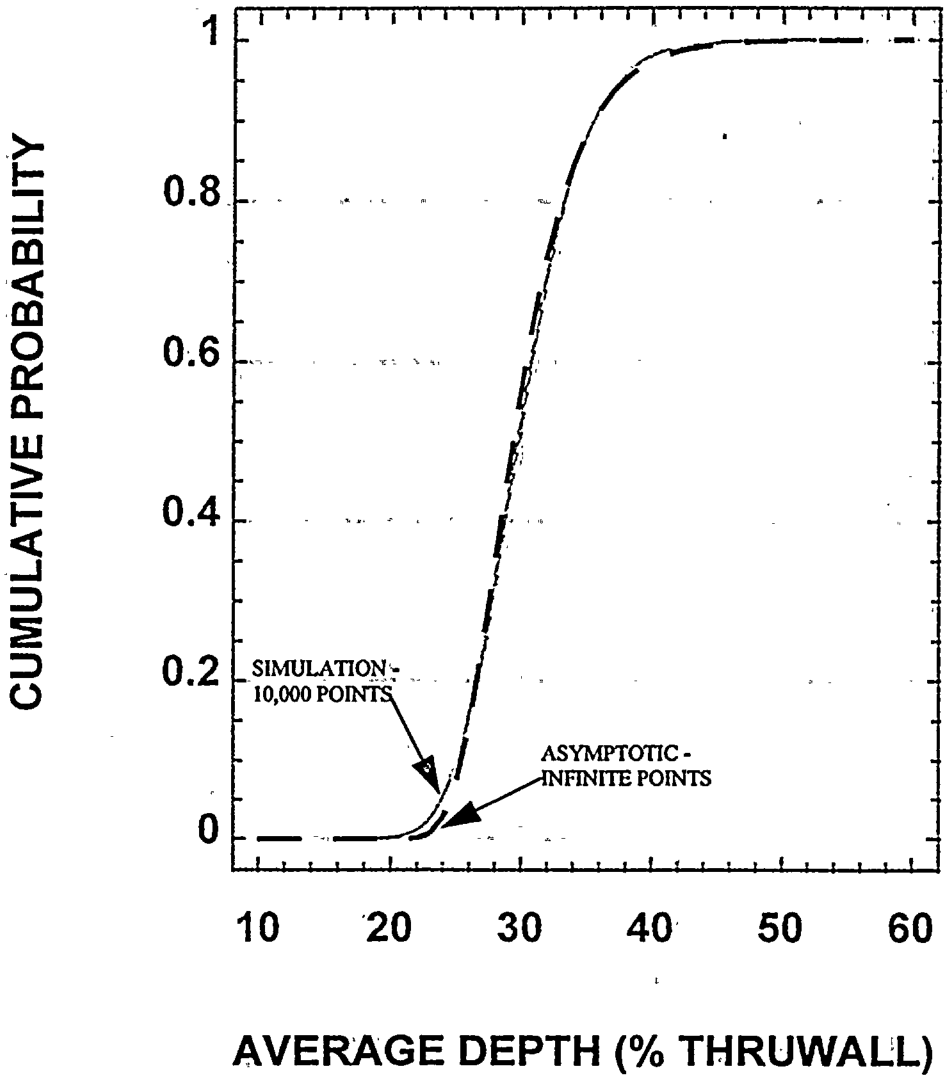


Figure 5.2 Extreme Value Distribution for Average Depth.

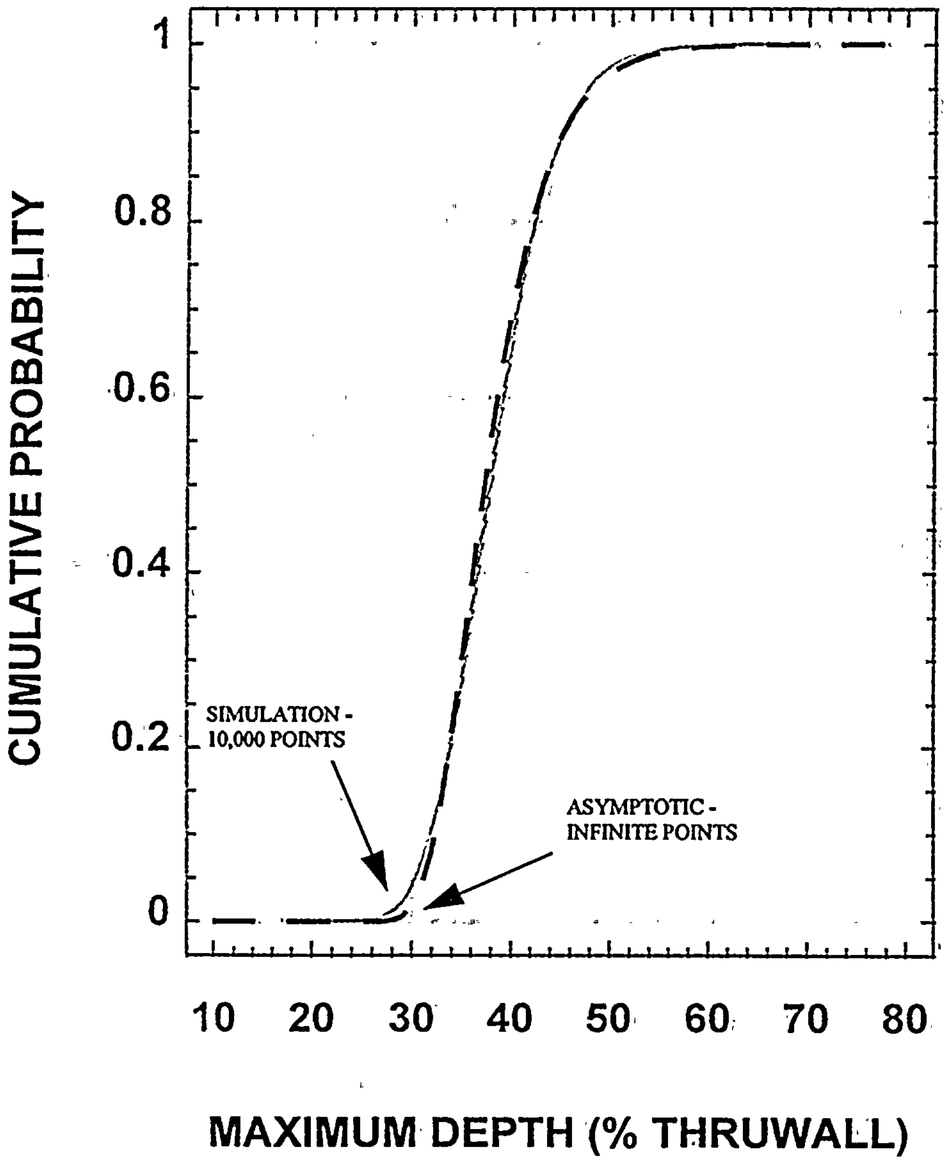


Figure 5.3 Extreme Value Distribution for Maximum Depth.

Section 6

SUMMARY AND CONCLUSIONS

An evaluation was performed of the significance of upper bundle corrosion degradation on the structural and leakage integrity of steam generator tubes at Palo Verde Unit 3. Eddy current indications in the upper freespan regions are interpreted as axial ODSCC/IGA, as confirmed by examination of pulled tubes from Unit 2. The progression of degradation in Unit 3 is considerably slower than in Unit 2. To date, approximately 76 tubes have exhibited upper bundle indications. Plus Point eddy current probes have used in the past two inspections.

In order to project end of cycle conditions relative to the number cracks present, and expected lengths and depths, a probabilistic run time model was employed. As in the past, the processes of crack initiation, crack growth and eddy current inspection were modeled in a Monte Carlo simulation. A cycle by cycle simulation was conducted. The model is benchmarked by comparisons to inspection results of previous cycles. Projections of end of cycle conditions have been very successful in the past for both Unit 2 and Unit 3.

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 are sampled directly from past observations without using a fitted analytic distribution. These refinements provide better projections and a better match with an occasional very deep crack observation.

Approximately 150 new indications are projected in the U3R6 inspection after a full cycle of operation. No instances of either through-wall cracks, Regulatory Guide 1.121 exceedances or tube burst under postulated steam line break conditions were observed in 10,000 simulations with a run time of 15.5 EFPM. The conditional probability of tube burst under postulated accident conditions is thus less than 10^{-4}. Required end of cycle structural margins are easily met and the lack of development of a single through wall in 10,000 run time simulations demonstrates that leakage is not an issue. Full cycle operation is strongly supported.

REFERENCES

1. Status of PVNGS Steam Generator Activities, Arizona Public Service Company Report, August, 1994.

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

3. 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.

4. 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.

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

6. 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, June, 1991, NP-6865-L, Vol. 1.

7. NRC Generic Letter 95-05: "Voltage-Based Repair Criteria for Westinghouse Steam Generator tubes Affected by Outside Diameter Stress Corrosion Cracking," United States Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, August 3, 1995.

8. Unit 3 NRC Submittal], May 1995.

9. Letter 102-03579, dated January 5, 1996, from W. L. Stewart, Executive Vice President, Nuclear, Arizona Public Service Company, to the U.S. Nuclear Regulatory Commission.

10. Cipolla, R. C., "Allowable Axial Flaw Sizes for Steam Generator Tubes in Palo Verde Unit 3", Calc. No. AES-C-1893-5, APTECH Report No. 93031893-1Q, Rev. 2, June, 1996.

APPENDIX B

Unit 3 Cycle 5
ECT Voltage Growth
Summary

ECT Summary Sheet - SG 3-1

SG	Number	Row	Line	Location	Length	Pancake Volts	Plus point Volts	ECT	Historical Length	Historical Pancake Volts	Historical PP Volts	DVDT
31	1	108	43	BW1 +2.33	1.8	0.23	0.56	SAI	NDD	0		0.28
	2	107	44	BW1 -0.26	0.9	0	0.64	SAI	0.6	0	0.52	0.00
	3	116	45	08H +38.84	3.5	0.27	0.54	SAI	1.4	0.19	0.41	0.10
	4	109	52	BW1 +0.73	2	0.4	0.47	SAI	1.2	0.2	0.36	0.24
	5	102	55	BW1 -0.57	2.2	0.31	0.67	SAI	1.9	0.22	0.67	0.11
	6	121	56	BW1 -0.76	1.2	0.4	0.65	SAI	0.6	0.17	0.55	0.28
	7	147	60	09H +7.97	0.7	0.25	0.31	SAI	None	None	None	0.30
	8	112	67	BW1 -0.90	1.8	0.35	0.85	SAI	0.8	0.2	0.73	0.18
	9	149	82	BW1+20.96	0.5	0	0.61	MAI	None	None	None	0.00
	10	149	82	BW1 +20.98	0.8	0	0.42	MAI	None	None	None	0.00
	11	149	92	BW1 +1.02	1.8	0.01	0.52	MAI	NDD	NDD	NDD	0.00
	12	149	92	BW1 +18.14	0.5	0.43	0.64	MAI	NDD	NDD	NDD	0.52
	13	149	92	BW1 +19.74	1	0.3	0.64	MAI	NDD	NDD	NDD	0.36
	14	132	95	09H +15.88	1.1	0.24	0.38	SAI	1	0.1	0.21	0.17
	15	137	98	BW1 +7.18	0.5	0	0.36	SAI	0.3	0	0.18	0.00
	16	135	100	BW1 -0.33	0.7	0	0.36	MAI	NDD	NDD	NDD	0.00
	17	135	100	BW1 +2.02	0.8	0.34	0.46	MAI	NDD	NDD	NDD	0.41
	18	119	132	08H +40.14	0.5	0	0.4	SAI	0.2	0	0.3	0.00
	19	103	140	08H +35.13	7.5	0.31	0.5	MAI	0.3	0.1	0.35	0.25
	20	103	140	BW1 +0.00	2.4	0.26	0.71	MAI	0.7	0.1	0.21	0.19
	21	103	142	08H +34.69	9.5	0.2	0.54	SAI	5	0.11	0.39	0.11
	22	103	144	08H +32.99	2.7	0.1	0.31	MAI	1.2	0	0.16	0.12
	23	103	144	08H +36.59	4.2	0.32	0.8	MAI	0.7	0.27	0.55	0.06
	24	103	144	BW1 -0.25	3	0.42	0.8	MAI	0.4	0.2	0.36	0.27

ECT Summary Sheet - SG 3-2

SG	Number	Row	Line	Location	Length	Pancake Volts	Plus point Volts	ECT	Historical Length	Historical Pancake Volts	Historical PP Volts	DVDT
32	1	96	37	VS2 -0.34	0.3	0.3	0.66	MAI	0.3	0.11	0.34	0.23
	2	96	37	VS2 +3.80	0.7	0.2	0.49	MAI	0.6	0.2	0.47	0.00
	3	106	39	BW1 +0.53	0.6	0	0.53	MAI	0.2	0	0.27	0.00
	4	106	39	VS2 +4.22	0.6	0.11	0.56	MAI	0.4	0.09	0.37	0.02
	5	101	42	VS2 +3.96	0.9	0.31	0.47	SAI	0.6	0.06	0.47	0.30
	6	102	45	VS2 +4.66	0.5	0.31	0.5	SAI	0.4	0.1	0.28	0.25
	7	102	49	BW1+21.54	0.5	0.16	0.27	MAI	0.3	0.09	0.18	0.08
	8	102	49	BW1+23.58	1.1	0.33	0.4	MAI	0.8	0.17	0.28	0.19
	9	96	51	BW1 -1.20	0.6	0	0.26	SAI	0.4	0	0.26	0.00
	10	104	51	08H +37.49	4.1	0.35	0.5	MAI	0.4	0.1	0.22	0.30
	11	104	51	BW1 +1.60	0.2	0.18	0.19	MAI	NDD	NDD	NDD	0.22
	12	96	53	BW1 +0.00	1.1	0.4	0.57	SAI	0.4	0.19	0.34	0.25
	13	133	56	VS1 +0.80	0.5	0.28	0.47	MAI	0.5	None	0.31	0.34
	14	133	56	VS1 +7.31	4.1	0.31	0.54	MAI	2.8	None	0.34	0.37
	15	102	57	BW1 -0.57	0.9	0.19	0.78	MAI	0.67	0	0.4	0.23
	16	102	57	BW1 +0.65	0.4	0.01	0.57	MAI	0.3	0	0.2	0.01
	17	105	60	08H +35.98	0.4	0.48	0.72	SAI	0.3	0.38	0.6	0.12
	18	128	89	BW1 +3.34	0.3	0.34	0.44	SAI	0.2	0.12	0.2	0.27
	19	112	97	08H +41.94	5	0.38	0.46	SAI	None	None	None	0.46
	20	142	97	BW1 +3.15	0.6	0.18	0.45	SAI	0.4	0.21	0.4	-0.04
	21	145	100	BW1 +2.34	1.6	0.01	0.23	MAI	NDD	NDD	NDD	0.01
	22	145	100	BW1 +19.65	1.8	0.2	0.23	MAI	NDD	NDD	NDD	0.24
	23	109	152	08H -0.17	0.9	0.08	0.57	SAI	NDD	NDD	NDD	0.09
	24	99	160	BW1 -1.23	0.7	0.01	0.48	SAI	0.5	0	0.47	0.01