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AUTH.NAME AUTHOR AFFILIATION
STEWART,W.L. Arizona Public Service Co. (formerly Arizona Nuclear Power
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EXECUTIVE VICE PRESIDENT
NUCLEAR

102-03401-WLS/AKK/JRP
June 27, 1995

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, DC 20555-0001

Reference: NRC Generic Letter 95-03, Circumferential Cracking of Steam Generator Tubes

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Response To Generic Letter 95-03,
Circumferential Cracking of Steam Generator Tubes

This letter is being provided in response to NRC Generic Letter 95-03, Circumferential Cracking of Steam Generator Tubes, and is being submitted pursuant to 10 CFR 50.54(f). The enclosure to this letter describes the current status of efforts undertaken by Arizona Public Service Company (APS), to address the two major corrosion mechanisms present in the PVNGS steam generators: arc region outside diameter stress corrosion cracking (ODSCC) and transition zone circumferential, inside diameter and outside diameter cracking. In addition to the status, included is the requested safety assessment justifying continued operation that is based on the evaluations performed in accordance with requested actions (1) and (2) in the Generic Letter, a summary of the inspection plans developed in accordance with requested action (3) of the Generic Letter, and a schedule for the next planned outage.

The analyses and evaluations contained in the enclosure demonstrate that the operating, inspection, and repair programs for the PVNGS steam generators ensure the safe operation of the units. The ability to manage the corrosion mechanisms in the PVNGS steam generators is a primary safety and strategic objective. Comprehensive actions completed by APS to achieve these objectives are discussed in the enclosure. These actions are all part of a defense in depth approach employed by APS to provide reasonable assurance that PVNGS Units 1, 2, and 3 can be safely operated.

300077

A001

Should you have any questions, please call Scott A. Bauer at (602) 393-5978.

Sincerely,

James M Levine for WLS

WLS/AKK/JRP/rv
Enclosure

cc: L. J. Callan
K. E. Perkins
B. E. Holian
K. E. Johnston
I. Barnes

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, J. M. Levine, represent that I am Vice President Nuclear Production, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

J M Levine

J. M. Levine

Sworn To Before Me This 27 Day Of June, 1995.

Roxanna VanDillen

Notary Public

My Commission Expires

June 12, 1997

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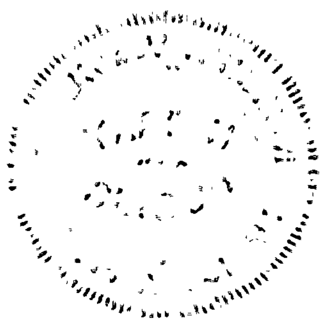

ENCLOSURE

RESPONSE TO GENERIC LETTER 95-03

CIRCUMFERENTIAL CRACKING OF

STEAM GENERATOR TUBES

I. Introduction

The purpose of this evaluation is to provide a response to USNRC Generic Letter (GL) 95-03, "Circumferential Cracking of Steam Generator Tubes" for the Palo Verde Nuclear Generating Station (PVNGS). The Generic Letter was issued in response to recent steam generator inspection findings at the Maine Yankee Atomic Power Station. The inspection found a number of circumferential cracks which were larger than expected after six (6) months of operation. The findings led to additional inspections with enhanced techniques. These techniques resulted in the identification of a significant number of circumferential cracks by eddy current testing (ECT). Penetrant testing confirmed the presence of circumferential cracks as indicated by the ECT signal response. Based on the results of these inspections, and other recent industry events, concerns have been raised regarding the ability to reliably detect and assess the presence of circumferential cracks in steam generator tubing.

Consequently, the USNRC Staff requested that licensees:

1. Evaluate recent operating experience with respect to the detection and sizing of circumferential indications to determine the applicability to their plant.

2. On the basis of the evaluation of Item 1 above, past inspection scope and results, susceptibility to circumferential cracking, threshold of detection, expected or inferred crack growth rates, and other relevant factors, develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections are performed.

3. Develop plans for the next steam generator tube inspections as they pertain to the detection of circumferential cracking. The inspection plans should address, but not be limited to, scope (including sample expansion criteria, if applicable), methods, equipment, and criteria (including personnel training and qualification).

In response to these concerns, and requested actions outlined in GL 95-03, Arizona Public Service (APS) has developed a technical and safety assessment which includes a summary of PVNGS plant specific experience as well as an assessment of industry experience recently summarized in a generic response to GL 95-03. This assessment also provides information regarding subsequent inspection and repair plans for all three PVNGS Units.

II. Operating Experience

Industry

In response to Generic Letter 95-03, the Nuclear Energy Institute (NEI) with support from utilities, NSSS vendors and EPRI, has compiled a detailed plant by plant compilation of data from steam generator inspections, in-situ pressure and leak testing, and pulled tube examination results (Reference 1). The information summarizes each plant's experience with circumferential cracking, and from this, APS can compare the actions taken to date at PVNGS to determine if these actions are considered appropriate given susceptibility and inferred defect growth rate.

[illegible]

Other industry data (References 2 and 3) have been consulted to determine if the presence of circumferential cracking at PVNGS is consistent with industry trends.

The steam generators at PVNGS were designed and fabricated by Combustion Engineering (CE), and are the only operating units of this design (System80). 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 was manufactured 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.

Since the tubing and tubesheet expansion specifications for CE steam generators are essentially the same, the occurrence of circumferential cracking in CE steam generators has been reviewed and assessed in the development of the steam generator inspection and repair program at PVNGS. Circumferential cracking has been reported in twelve of the fifteen operating units with CE designed steam generators. Included in this list, is the Maine Yankee facility which is the subject of GL 95-03. Circumferential cracking in CE plants has resulted in four (4) occurrences of tube leaks since 1983.

In currently operating CE steam generators, all reported circumferential cracking has occurred at the tube expansion transition on the hot leg of the steam generator. The reported defects have been either Primary Water Stress Corrosion Cracking (PWSCC) initiating at the inside diameter (ID) or Outside Diameter Stress Corrosion Cracking (ODSCC) initiating at the secondary side or outside diameter (OD) of the steam generator tubing. No indication of cold leg circumferential cracking has been reported for operating CE steam generators.

In retired CE steam generators, only ODSCC circumferential cracking was reported. At one retired unit, circumferential cracking was reported on both the hot and cold leg tubesheet transition locations. OD circumferential cracking was also reported at tube support plates and U-bends at another since retired CE steam generator.

There have been no reports of circumferential cracking at sleeve locations in CE units. However, the occurrence of these defects at Westinghouse units has been reviewed and assessed by APS. Occurrences of circumferential cracking in Hybrid Expanded Joints (HEJ) and Kinetically Welded sleeves have been reported at three (3) Westinghouse facilities. Currently, there are no sleeves of any design installed at PVNGS.

The severity, and inferred crack growth rates of circumferential cracking in CE steam generators can be assessed from not only ECT results but also the metallurgical examination and in-situ pressure testing which has been conducted by CE facilities. Fifteen tubes have been removed from CE steam generators for circumferential cracking. Additionally, 32 tubes at four (4) CE units have been in-situ pressure tested in accordance with the structural margins specified by Regulatory Guide 1.121. Confirmatory tests demonstrated that the tested tubes maintained the required structural margins (Reference 1).

PVNGS Experience

Circumferential cracking was first observed at PVNGS in Unit 1 in the course of ECT inspections conducted during U1R4 in the fall of 1993. At that time Unit 1 had operated a total of 4.57 effective full power years (EFPY) at a primary side operating temperature of 621°F. The presence of circumferential cracking was not unexpected according to industry predictions for PWSCC given the level of operating time at temperature (Reference 2). A detailed report outlining the actions taken by APS in response to the detection of circumferential cracks was submitted to USNRC in Reference 4. During the U1R4 outage APS conducted a 100% MRPC inspection program of the tubesheet transition region. Additionally, Ultrasonic Testing (UT) of 17 tubes with the largest ECT detected defects was conducted, and in-situ pressure testing of five (5) candidate tubes was performed using conservative selection criteria. All detected cracks were removed from service. Based on the inspection and repair results, APS concluded that all tubes maintained the required margins of safety as outlined in Regulatory Guide 1.121. At the request of the USNRC, APS developed several evaluations assessing the beginning of cycle distribution of undetected defects, crack growth rate and end of cycle structural margins. These evaluations (References 5 and 6) were submitted to the USNRC, and in Reference 7 the Staff concluded that Unit 1 could be safely operated for the full operating cycle during Cycle 5. For reasons other than circumferential cracking, PVNGS Units 2 and 3 have operated with reduced operating cycles since 1993. Circumferential cracks have been identified in these two Units to a lesser extent than in Unit 1. A summary of detected tubesheet transition circumferential cracks is summarized in Table 1.

Table 1: Circumferential Cracking at PVNGS

Inspection	Steam Generator Tubes Plugged		EFPY (years)
•	SG #1	SG #2	
U1R4	7	76	EFPY 4.57
U3M4	0	4	EFPY 4.08
U2M5-1	0	4	EFPY 4.91
U2R5	2	2	EFPY 5.32
U1R5	19	57	EFPY 5.77

Although industry experience indicates that circumferential cracking can occur elsewhere in the steam generator, such as sleeve locations, U-bends and tube support plate intersections, circumferential indications in other locations have not been detected at PVNGS.

Currently, there are no sleeves installed at PVNGS. The recent Technical Specification amendment submitted by APS (Reference 8) addressed recent experience with cracking of parent tubes in certain sleeve applications, and outlined the actions which would be taken by PVNGS to minimize the potential for parent tube cracking resulting from sleeve installation.

There has been no evidence of U-bend or tube support plate circumferential cracking at PVNGS. U-bend (Row 1 and 2) and square bend inspections with MRPC have been conducted at PVNGS in all three units with no indications of circumferential cracking reported. There has been no evidence of severe denting issues in the PVNGS steam generators. The horizontal supports in the System80 steam generators 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. The large flow area in this support design provides better irrigation and reduces the potential for steam blanketing, and are therefore less likely to be blocked by crud, boiler water deposits and corrosion products. Since the support material is type 409 ferritic stainless steel, it is not susceptible to magnetite corrosion which has resulted in denting and lockup at plants with carbon steel supports. This expected observation with regard to denting have been substantiated via eddy current in all three units and tube pull activities conducted in PVNGS Unit 2.

Some cold leg corrosion at the top of tubesheet has been observed in Unit 2 during recent inspections in March 1995. These indications appear to be mixed mode (both circumferential and axial ECT signal response). Although inspection results indicated that these indications could be identified via bobbin coil techniques, the Unit 2 ECT program was expanded to inspect 100% of the cold leg tubesheet transitions with MRPC. No evidence of cold leg circumferential cracking was observed. The corrosion observed appears to be limited to a specific region near the stay cylinder. A 20% cold leg MRPC sampling program was conducted in the following Unit 1 inspection, and no evidence of circumferential cracking or cold leg corrosion was observed. The cold leg region in Unit 1 was also inspected via a 100% bobbin coil program. Similar inspections are planned for Unit 3 in U3R5.

III. PVNGS Inspection and Repair Program

A comprehensive steam generator inspection program reduces the risk of leaving a significant defect(s) in service. APS has developed a steam generator inspection program which is responsive both to the issues identified within the industry as well as the unique issues associated with the PVNGS steam generators. As an additional measure of safety, before the PVNGS steam generators are returned to service, a thorough review by APS Engineering of all eddy current indications is conducted. PVNGS current administrative plugging criteria specifies that all tubes with detected cracks, axial or circumferential, regardless of size or depth, be removed from service.

ECT Inspection Program

The PVNGS steam generator ECT inspection program has continued to evolve in response to the use of state of the art equipment and technique development which incorporates lessons learned from PVNGS exams and tube pulls as well as industry experience. The objectives of the program are to acquire data in a timely fashion while maintaining and/or improving the ability to detect and characterize flaws. Table 2 summarizes the inspections conducted in response to circumferential cracking issues. Since 1993, APS has provided the USNRC information regarding ECT program objectives and results. In References 9 and 10, APS has submitted for USNRC review the inspection programs conducted in Units 2 and 3.

Table 2: ECT Inspection Programs 1993-1995

Inspection	Full Length Bobbin		MRPC Hot Leg TTS		MRPC Cold Leg TTS		Comments
	SG #1	SG #2	SG #1	SG #2	SG #1	SG #2	
U1R4 (10/93)	100%	100%	100%	100%	n/a	20%	Cold Leg MRPC performed in SG#2 due to number of defects found in SG#2
U3M4 (12/93)	37%	37%	20%	100%	n/a	N/A	MRPC Hot Leg expanded to 100% due to detection of circumferential indications
U2M5-1 (3/94)	40%	39%	21%	100%	n/a	n/a	MRPC Hot Leg expanded to 100% due to detection of circumferential indications
U3R4 (5/94)	100%	100%	20%	5%	n/a	n/a	SG#2 exam limited to 100% of sludge pile as 100% exam was performed just three months prior
U2M5-2 (10/94)	39%	34%	9%	5%	n/a	n/a	Exams limited to 100% of sludge pile as previous exam was performed just six months prior
U3M5 (12/94)	19%	18%	3%	6%	n/a	n/a	Exams limited to 100% of sludge pile
U2R5 (3/95)	100%	100%	100%	37%	100%	100%	MRPC Exams performed with Plus Point Probe
U1R5 (5/95)	100%	100%	100%	100%	20%	20%	MRPC Exams performed with Plus Point Probe

In response to the issues identified in the 1994 Maine Yankee inspections, APS submitted Reference 11 to the NRC Staff specifying key aspects of our MRPC inspection programs. Reference 11 also indicated that factors such as tubesheet denting or high copper levels which could interfere with ECT signal response were not present at PVNGS. Additionally, in References 12 and 13 the USNRC reported that the PVNGS ECT programs were responsive to the steam generator corrosion issues at PVNGS.

In the Fall of 1994 APS elected to use the Plus Point[1] MRPC Probe for MRPC inspections in upper tube bundle and the tubesheet region. Due to the critical nature and extent of MRPC inspections at PVNGS, the primary objective in any 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 U3M5 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 probe design utilized at PVNGS employs a standard 0.115" diameter pancake coil, and a separate shoe containing the plus point coil. During U3M5 this probe was primarily used for bend region inspections. The probe was used in U2R5 and U1R5 to conduct all hot and cold leg tubesheet exams.

The plus point probe was demonstrated to meet the stated objectives of faster production rates and improved detectability in U3M5, U2R5 and U1R5. In order to assess the capabilities of the plus point probe the majority of the pluggable indications (crack indications) were reinspected utilizing the standard three coil MRPC during U3M5 and U2R5, which has been benchmarked via tube pulls conducted by PVNGS. The ability of the probe to improve detectability was also demonstrated at Maine Yankee and during recent EPRI Appendix H qualification efforts.

Data Analysis

During 1995, Primary and Secondary analysis have been 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. Conam Nuclear Inc. provided the data acquisition and primary data analysis. Anatec International Inc. provided the secondary data analysis. In general, data analysis is performed in accordance with the EPRI PWR Steam Generator Examination Guidelines and the EPRI recommendations for assessment of circumferential indications as specified in Reference 14.

Each Level IIA individual from Conam Nuclear Inc. and Anatec International Inc. who perform data analysis at PVNGS is 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. The analysts are also trained and certified for plant specific data

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.

analysis with the Plus Point MRPC probe.

Ultrasonic Inspections

At PVNGS, the MRPC eddy current probe is used to size and characterize circumferential indications. APS has elected not to depth size with the MRPC, however signal response is utilized to characterize the circumferential extent of the defect(s). Since depth calls are not made, all cracks have been assumed to be 100% through-wall for the entire length. This technique allows APS Engineering to make a conservative assessment of defects which may not meet the structural margins required by RG 1.121. As reported to the USNRC in Reference 4, APS elected to inspect 17 flaws with UT techniques in an effort to compare the MRPC signal response and further assist in assessing the structural integrity of the steam generator tubing. In general the UT results were consistent with the MRPC results. The maximum UT depth call combined with the MRPC length was used by APS to establish possible RG 1.121 exceedances and determine candidates for in-situ pressure testing during U1R4. It should be noted that no circumferential defects found in Units 1, 2, and 3 subsequent to U1R4 exceeded the maximum MRPC length for a 100% through-wall defect. Therefore, no further UT testing has been conducted at PVNGS for tubesheet defects.

In-situ Pressure Testing

As reported in Reference 4, APS conducted in-situ pressure testing of five defects considered to approach or exceed the RG 1.121 structural limit as based on conservative MRPC and UT measurements. All five tubes were pressurized to 3900 psig based on a limiting differential pressure of 3540 psi adjusted for room temperature conditions, and held for approximately eight (8) minutes. All tubes were successfully pressurized to 3900 psig, indicating that the required safety margins against burst were maintained. The testing conducted at PVNGS as well as other CE facilities also demonstrates the conservatism of the MRPC detection and sizing techniques with regard to actual tube structural integrity.

Repair Criteria

As indicated previously, all detected cracks at PVNGS whether axial or circumferential are removed from service via plugging. All circumferential defects are also staked to provide protection to adjacent active tubes in the event that the cracked tube should subsequently sever. Additionally volumetric defects detected at the tubesheet are plugged, if a significant sizing change is noted from baseline inspections.

IV. Structural Assessments

As indicated previously, APS Engineering assesses the ECT signal response for circumferential defects to determine: 1) If additional testing is required and 2) the defects exceed the safety margins as required by Regulatory Guide 1.121. Depending on the ECT results, a deterministic or probabilistic assessment of beginning of cycle conditions, crack growth rates, and end of cycle conditions is made to determine if the steam generators can be safely operated for the next operating cycle.

Although APS demonstrated that none of the defects detected in U1R4 exceeded RG 1.121 structural margins, the USNRC requested that APS (Reference 15) perform additional analysis to demonstrate that the Unit could be operated for a full operating cycle. Using the inspection results from Unit 1, as well as other CE facilities (including Maine Yankee), APS demonstrated in References 5, 6 and17 that there was a high probability that Unit 1 could be operated a full operating cycle without a RG 1.121 exceedance. In Reference 7 the USNRC concurred with the overall integrated approach taken by APS. The recent results from ECT inspections conducted during U1R5 indicate that the analysis was indeed conservative, as no RG 1.121 violators were detected.

Units 2 and 3 have, for other steam generator issues, operated with reduced cycles since 1993. The circumferential defects found in these Units are considered bounded by the detailed analysis performed in References 5, 6 and16. For Unit 3, this position was restated in Reference 17, submitted to the USNRC on May 19,1995.

V. Mitigation Strategies

Since 1993, APS has implemented aggressive measures to minimize primary and secondary tube corrosion in the PVNGS steam generators. References 6 and17, submitted to the USNRC, provide extensive details regarding these actions. In summary, these actions include:

- Temperature reduction from 621°F to 611°F has 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 the steam generators, and has conducted extensive NDE and destructive examination in an effort to determine causal effects of corrosion damage. Although these tubes were removed to assist in assessing the root cause of the PVNGS upper bundle corrosion phenomenon, the results have led to substantial improvements in field ECT acquisition and interpretation. It should also be noted that destructive exams of the tubesheet transition region were performed on four tubes pulled in 1993, and no evidence of crack initiation was detected.

- APS has implemented the industry recommended primary and secondary chemistry controls to mitigate the initiation and propagation of SCC defects. The laboratory evidence from tubes removed from Unit 2 during U2M5-1 in 1994 show a potential favorable change in crack crevice chemistry tending towards a more neutral environment.

- The PVNGS ECT results indicate a relationship between sludge pile formation and the presence of circumferential cracks. Since 1993, APS has implemented several sludge and deposit removal strategies. Chemical cleaning has been conducted in all three PVNGS Units. Sludge lancing has become a routine activity since the installation of tubesheet level handholes in 1993. APS has also conducted a blowdown optimization program to maximize bulk water cleanup and minimize sludge pile formation via a optimized schedule of normal, and high rate blowdown.

- APS has converted from ammonia to ethanolamine (ETA) for secondary system pH control with the goal of reducing corrosion product transport to the steam generators. ETA has successfully reduced iron transport as evidenced by the comparisons prior to 1994 when ammonia was utilized. Feedwater iron levels have been reduced from levels >10 ppb to levels of 1-3 ppb since the conversion to ETA. This action helps to further reduce the size and depth of sludge and deposit accumulation in the PVNGS steam generators.

VI. Operational Response

Since crack growth models and ECT testing contain a degree of uncertainty, additional improvements in operational response have been taken by APS to assure safe plant operation. These include an improved leak rate monitoring program with administrative limits on primary to secondary leakage, training of operations personnel for tube rupture events, improvements in diagnostics via equipment and procedure upgrades and the implementation of N-16 monitors, and upgrades to the Emergency Operating Procedures to permit faster identification and isolation of the affected steam generator.

Leakage Monitoring

APS has incorporated an integrated leakage monitoring program, utilizing equipment and procedure upgrades, to permit plant operators 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 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 program is supported by aggressive outage activities which ensure that all detected SCC defects are removed from service. The program is described in detail in References 6 and 17 submitted to the USNRC Staff. The program includes:

- Equipment upgrades to the steam generator blowdown radiation monitors and condenser vacuum exhaust radiation monitor to provide greater sensitivity for detection and response to primary to secondary tube leakage.

- The installation of N-16 monitors to provide an additional diagnostic tool for primary to secondary leakage detection.

- An administrative leakage limit set at 50 gpd, well below the Technical Specification limit of 720 gpd. The administrative procedures also contain a leak rate hierarchy below 50 gpd and actions are implemented if leak rates experience a sudden change.

Operator Training
As also stated in Reference 17, extensive simulator training of operations personnel for tube rupture events and upgrades to the Emergency Operating Procedures implemented since 1993 permit faster identification and isolation of the affected steam generator. Improvements in operator response assure that in the unlikely event of a main steam line break with consequential multiple tube ruptures, the resulting offsite doses are maintained less than 10CFR100 limits.

VII. Safety Assessment - Unit 1

Since the discovery of circumferential indications in Unit 1 in 1993, the steam generators in this Unit have had the largest quantity of cracks. As indicated in Reference 6 submitted to the USNRC, it is believed that this condition is primarily a function of larger sludge piles in Unit 1. Based on ECT results in Units 2 and 3, APS concluded in References 6 and 17 that the structural and safety assessment for circumferential cracking in Unit 1 bounds the conditions observed in Units 2 and 3. Additionally, since the upper bundle or ARC region degradation in the PVNGS steam generators is considered the most limiting form of corrosion, the defense in depth philosophy adopted by PVNGS to manage this type of corrosion is applicable to operation of all the PVNGS Units

The safety significance associated with the operation of Unit 1 until its scheduled U1R5 refueling outage in April 1995 was evaluated by APS in References 5 and 6. The analyses concluded with high confidence that the conservative safety margins established in Regulatory Guide 1.121 would be maintained.

APS recognized that probabilistic models contain a degree of uncertainty associated with the upper tail of the crack growth rate distribution, therefore additional actions were taken by APS to assure safe plant operation. An improved leak rate monitoring program with administrative limits on primary to secondary leakage provided additional assurance that an orderly shutdown will be conducted prior to a through-wall leak propagating to a rupture. Training of operations personnel for tube rupture events, improvements in diagnostics via equipment and procedure upgrades and the implementation of N-16 monitors, and upgrades to the Emergency Operating Procedures as described also permit faster identification and isolation of the affected steam generator.

Finally, an analysis was performed by APS in Reference 18 to demonstrate that in the unlikely event of a main steam line break with consequential multiple tube ruptures, with the current

administrative limits on reactor coolant system dose equivalent iodine, the resulting offsite doses are less than 10CFR100 limits. APS believed that this "defense in-depth" approach, provided reasonable assurance that the PVNGS Unit 1 could be safely operated for a full operating cycle in Cycle 5. The USNRC concurred with this position in Reference 7, however the Staff requested that the U1R5 inspection results be carefully assessed to ensure that there continues to be an adequate basis to support full cycle operation in Unit 1.

The U1R5 inspection program as outlined in Table 2 has been completed. The 100% Plus Point MRPC program was consistent with the projected results from Reference 6, in that no defects exceeding RG 1.121 margins as conservatively measured by APS and described in this assessment were found. Consequently, it is the APS position that full cycle operation in Unit 1 is justified, as the U1R5 results are bounded by the assessment conducted in References 5, 6 and 17. Based on other commitments, APS expects to provide to the USNRC, a separate report regarding all the findings in the U1R5 inspections within six (6) months of operation following plant start-up for Cycle 6.

VIII. Safety Assessment - Unit 2

Since 1993, Unit 2 has been restricted to reduced cycle operation due to steam generator issues not related to circumferential cracking. The results from comprehensive MRPC exams conducted in multiple inspections during Cycle 5, revealed that the circumferential indications found in Unit 2 are bounded by the Unit 1 assessments by size, quantity and run time.

APS is committed to providing a report to the USNRC justifying a run time in excess of the currently restricted operating interval of six (6) months by August 31, 1995.

IX. Safety Assessment - Unit 3

Since 1993, Unit 3 has also been restricted to reduced cycle operation due to steam generator issues not related to circumferential cracking. In May 1995 APS submitted to the USNRC, justification for operation of Unit 3, including a safety assessment, until its next refueling outage in October 1995 (Reference 17). This represents 10 months of operation following a midcycle inspection in December 1994. During the U3R5 inspection the hot leg tubesheet region will be inspected 100% with the Plus Point MRPC.

X. Projected Inspection Programs

Due to uncertainties associated with allowable run times for PVNGS Units 2 and 3, the proposed tubesheet inspection programs for circumferential defects could change to reflect USNRC approved operating intervals. The minimum MRPC inspection program would be as follows:

- A 20% initial sample inspection with Plus Point MRPC will be performed. This sample would include, at a minimum, a 100% program of the kidney bean

sludge pile region. The expansion criteria would be a full 100% hot leg tubesheet MRPC exam, if a circumferential indication has been found and twelve months has passed since the last 100% inspection.

- APS intends to perform a cold leg MRPC program of 1000 tubes per SG during U3R5 in October of 1995. This program may be discontinued for U1R6 and U2R6 pending Unit 3 results.

- For all end of cycle outages, a 100% bobbin coil program is conducted.

- Acquisition methods, probe type, analyst training and monitoring are described previously, and in further detail in Reference 17 submitted to the USNRC.

A projected ECT program for the upcoming inspections in Units 1, 2, and 3 is provided in Table 3.

Table 3: Projected Eddy Current Program

Inspection	Full Length Bobbin		MRPC Hot Leg TTS		MRPC Cold Leg TTS		Comments
	SG #1	SG #2	SG #1	SG #2	SG #1	SG #2	
U3R5	100%	100%	100%	100%	10%	10%	MRPC Inspections to be performed with Plus Point Probe
U2R6	100%	100%	20-100%	100%	10%	10%	MRPC Inspections to be performed with Plus Point Probe
U1R6	100%	100%	100%	100%	-	-	MRPC Inspections to be performed with Plus Point Probe Cold Leg exams dependent on U2 and U3 results

XI. Conclusions

Based on the assessments contained in this response, and in PVNGS submittals referenced herein, it is APS's position that the inspection, repair, mitigation and response programs at PVNGS assure safe operation of the PVNGS steam generators with regard to circumferential cracking. The programs are considered prescriptive and preventative in managing corrosion and maintaining structural margins. Based on this position, no new compensatory actions are considered at this time.

XII. References

1. NEI Letter, *NRC Generic Letter 95-03: "Circumferential Cracking of Steam Generator Tubes"*, dated June 1, 1995

2. EPRI Report NP-7198-S, *Proceedings: 1990 EPRI Workshop on Circumferential Cracking of Steam Generator Tubes,* March 1991

3. EPRI Report TR-104030, *PWSCC Prediction Guidelines*, July 1994

4. *Unit 1 Steam Generator Report - October 1993,* submitted to the USNRC via letter 102-02569- WFC/JRP dated July 18,1993

5. *Steam Generator Tube Regulatory Guide 1.121 Analysis for Primary Side Circumferential Cracking,* submitted to USNRC by letter 102-3013-WLS/AKK/JRP dated June 20, 1994

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

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

8. *Technical Specification Amendment Request, Sleeving Process for Steam Generator Tube Repair,* submitted to USNRC by letter 102-03325-WLS/SAB/JRP dated April 18, 1995

9. *Steam Generator Inspection Plan - Unit 2,* submitted to USNRC by letter 102-02751-JML/ RAB/JRP dated December 6, 1993

10. *Steam Generator Inspection Plan - Unit 3*, submitted to USNRC by letter 102-02847-JML/ RAB/JRP dated March 2, 1994

11. *Eddy Current Program Review*, submitted to USNRC by letter 102-03124-WLS/AKK/JRP dated September 22, 1994

12. USNRC Letter, *NRC Inspection Report 94-15*, dated June 23, 1994

13. USNRC Letter, *Systematic Assessment of Licensee Performance Report,* dated January 12, 1995

14. EPRI Letter, *Points to Consider in Circumferential Crack Detection and Length Sizing,* dated February 23, 1995

15. USNRC Letter, *Summary of Meeting Held November 9, 1993 to Discuss Unit 1 Steam Generator Eddy Current Test Results,* dated December 29, 1993

16. ABB-CE Report V-PENG-TR-006, *Palo Verde - 1 Steam Generator Tube Regulatory Guide Analysis for Secondary Side Circumferential Cracking,* August 1994

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

18. *Steam Generator Tube Evaluation,* submitted to USNRC by letter 102-02797-WFC/TRB/RAB dated July 25, 1993