# ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9011200300 DOC.DATE: 90/11/14 NOTARIZED: YES DOCKET #
FACIL:STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
AUTH.NAME AUTHOR AFFILIATION
CONWAY,W.F. Arizona Public Service Co. (formerly Arizona Nuclear Power
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Suppl application for amend to License NPF-41,revising TS Section 4.4.4.3 to extend interval for SGEC exam.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16+1
TITLE: OR Submittal: General Distribution

NOTES:STANDARDIZED PLANT 05000528

| | RECIPIENT ID CODE/NAME | COPIES LTTR | COPIES ENCL | RECIPIENT ID CODE/NAME | COPIES LTTR | COPIES ENCL |
|---|---|---|---|---|---|---|
| | PD5 LA | 1 | 1 | PD5 PD | 1 | 1 |
| | PETERSON,S. | 2 | 2 | TRAMMELL,C. | 2 | 2 |
| INTERNAL: | ACRS | 6 | 6 | NRR/DET/ECMB 9H | 1 | 1 |
| | NRR/DET/ESGB | 1 | 1 | NRR/DOEA/OTSB11 | 1 | 1 |
| | NRR/DST 8E2 | 1 | 1 | NRR/DST/SELB 8D | 1 | 1 |
| | NRR/DST/SICB 7E | 1 | 1 | NRR/DST/SRXB 8E | 1 | 1 |
| | NUDOCS-ABSTRACT | 1 | 1 | OC/LFMB | 1 | 0 |
| | OGC/HDS1 | 1 | 0 | REG FILE 01 | 1 | 1 |
| | RES/DSIR/EIB | 1 | 1 | | | |
| EXTERNAL: | NRC PDR | 1 | 1 | NSIC | 1 | 1 |
| NOTES: | | 1 | 1 | | | |

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 27 ENCL 25

AA2

**Arizona Public Service Company**
P.O. BOX 53999 • PHOENIX, ARIZONA 85072-3999

WILLIAM F. CONWAY
EXECUTIVE VICE PRESIDENT
NUCLEAR

161-03589-WFC/JRP
November 14, 1990

Docket No. STN 50-528

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Station P1-37
Washington, D. C. 20555

Reference: Letter from W. F. Conway, APS to USNRC (161-03526) dated October 6, 1990

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Revised Technical Specification Amendment
Extension of Interval for SG Eddy Current Examination
File: 90-F-005-419.05; 90-056-026

This letter supersedes the referenced letter and is being provided to request an amendment to the PVNGS Unit 1 Technical Specifications Section 4.4.4.3, Inspection Frequencies (inservice inspection of steam generator tubes). The proposed change would extend the performance interval for the Unit 1 steam generator eddy current examination from the existing intervals of not less than twelve nor more than twenty-four calendar months after the previous inspection to the end of the present fuel cycle (Cycle 3), or not more than 32 calendar months.

The performance interval for the Unit 1 steam generator eddy current examination should be extended to the end of the present fuel cycle to coincide with the next refueling outage presently scheduled to begin February 1, 1992. The bases and justification for the proposed change is presented herewith as an attachment.

Also enclosed with this amendment request are:

A. Description of the Amendment Request
B. Purpose of the Technical Specification
C. Need for the Technical Specification Amendment
D. Basis for No Significant Hazards Consideration
E. Safety Analysis of the Proposed Amendment Request
F. Environmental Impact Consideration Determination
G. Marked-up Technical Specification Change Page(s)



A001
1/1

Pursuant to 10 CFR 50.91(b)(1), and by copy of this letter and attachment, we have notified the Arizona Radiation Regulatory Agency of this request for a Technical Specification Amendment.

Should you have any questions, please call M. E. Powell of my staff at (602) 340-4981.

Sincerely,

James M. Levine for WFC

WFC/JRP/jle

Attachment

cc: C. M. Trammell
J. B. Martin
D. H. Coe
C. F. Tedford

161-03589-WFC/JRP
November 14, 1990

STATE OF ARIZONA )
) ss.
COUNTY OF MARICOPA )

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

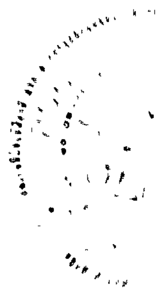
James M. Levine
J. M. Levine

Sworn To Before Me This 14 Day Of November, 1990.

Dora E. Meador
Notary Public

My Commission Expires

My Commission Expires April 6, 1991



# ATTACHMENT

## A. DESCRIPTION OF THE AMENDMENT REQUEST

The proposed Technical Specification amendment would revise Section 4.4.4.3, Inspection Frequencies (inservice inspection of steam generator tubes) to extend the performance interval for the Unit 1 steam generator eddy current examination. The inspection frequency currently applicable to the Unit 1 steam generators is as follows, "Subsequent inservice inspections shall be performed at intervals of not less than twelve nor more than twenty-four calendar months after the previous inspections." The proposed change would extend the performance interval for the Unit 1 steam generator eddy current examination to the end of the present fuel cycle (Cycle 3), or to not more than 32 calendar months. The performance interval for the Unit 1 steam generator eddy current examination should be extended to 32 calendar months to coincide with the next scheduled refueling outage. The 32 calendar months is based on a time frame of July 1989 to February 1992.

## B. PURPOSE OF THE TECHNICAL SPECIFICATION

The surveillance requirements for inspections of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

## C. NEED FOR TECHNICAL SPECIFICATION AMENDMENT

The Technical Specification surveillance requirements require that subsequent inservice inspections be performed at intervals of not less than twelve nor more than twenty-four calendar months after the previous inspection. The most recent inservice inspection of the Unit 1 steam generator tubes was completed in July 1989 during the second refueling outage.

The Unit 1 steam generator tubes have been eddy current examined on four previous occasions. The information presented lists the time frame of each examination and the number or percentage of steam generator tubes examined.

| | July - August 1981 (preservice) | January, 1987 (outage) | October, 1987 (1st refueling) | July, 1989 (2nd refueling) |
|---|---|---|---|---|
| SG 11 | 100% | 3516 (32%) | 2375 (22%) | 100% |
| SG 12 | 100% | 3496 (32%) | 3750 (34%) | 100% |

The types of problems noted in the Unit 1 steam generators can be summarized as follows:

- lower eggcrate wear
- minor dents and dings
- minor denting around flow distribution baffle
- possible loose parts indications
- minor batwing and vertical strip wear
- minor sludge buildup

The observed conditions are primarily associated with mechanical wear as opposed to chemistry or corrosion problems. The mechanical wear is due to vibration associated with normal plant operations. Based on the inservice inspection results the following tube plugging has resulted:

| | January, 1987 | October, 1987 | July, 1989 |
|---|---|---|---|
| SG 11 | 20 | 11 | 12 |
| SG 12 | 14 | 1 | 7 |

These tubes were plugged based on a criterion of 30% through wall indication. The overall results of the examination are encouraging in that after completion of the second cycle, the total number of degraded and/or defective tubes is minimal and no significant wear pattern has been observed.

The performance interval for the Unit 1 steam generator eddy current examination should be extended to the end of the present fuel cycle to coincide with the next refueling outage presently scheduled to begin February 1, 1992. Justification for the extension of the performance interval is as follows and is discussed in Sections C, D, and E.

- There are minimal degraded and/or defective tubes in the Unit 1 steam generators.

- No significant wear patterns have been observed (based on result from previous inservice inspections).

ATTACHMENT
(Continued)

- A 100% eddy current examination of the Unit 1 steam generator tubes was performed during the last refueling outage (July 1989).

- Adequate chemistry control was maintained in the Unit 1 steam generators during the shutdown and operational periods to date. (Based on SG manufacturers evaluation during MSIV rework.)

- Minimal mechanical wear occurred for the eleven month period following the most recent eddy current examination while the unit was shutdown.

Unit 1 entered mode 1 on June 30, 1990, ending its second refueling outage which began on March 5, 1989 when the unit entered Mode 3. If the proposed amendment request were not approved by the NRC the Unit 1 steam generators would be eddy current examined during the January 1991 surveillance test outage. Thus, the steam generators would only have been subjected to mechanical wear, which is the most significant contributor to tube degradation, for approximately seven months. As a result, the eddy current examination would not likely be meaningful in assessing tube wear patterns nor would there have been a sufficient period of operation to cause a concern for a significantly increased degradation of the tubes. Performance of the eddy current examination would require mid-loop operation in order to meet schedule restraints. And finally, there would be increased radiation exposure incurred while installing and removing nozzle dams.

## D. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment 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 a margin of safety. A discussion of these standards as they relate to the amendment request follows.

Standard 1 -- Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed Technical Specification amendment will not increase the probability or consequences of an accident previously evaluated.

The Unit 1 steam generator tubes have been eddy current examined on four previous occasions since 1981. The most recent examination was July 1989, with 100% of the tubes being examined. The results of the examination determined that the indications identified are primarily associated with mechanical wear as opposed to chemistry or corrosion problems. The mechanical wear is due to vibration associated with normal plant operation. The overall results of the examination are very good, in that after completion of the second fuel cycle, the total number of degraded and/or defective tubes was minimal (19 tubes being plugged) and no significant wear patterns were observed.

Since the July 1989 eddy current examination, Unit 1 was shut down until June 1990. During this time there was no mechanical wear that could contribute to tube degradation. Also during this shut down period, chemistry control was maintained in accordance with plant procedures which dictates strict adherence to prescribed lay-up practices and specifications.

The Unit 1 steam generators entered wet lay up conditions on April 15, 1989. Other than the nitrogen overpressure not being maintained within specifications because of the main steam isolation valve (MSIV) work undertaken and the wet layup recirculation line out of service for rework, the remaining chemistry control was well maintained with only a few instances when chemistry analyses indicated out of specification conditions. The pH was slightly low (SG #1) at 9.7 (spec. is 9.8 to 10.5) on May 5, 1989. Chloride, sulfate, and sodium were well below the 1.0 ppm specification limit during the entire period. Hydrazine ranged from 81 to 176 ppm with an average of 123 ppm (75-200 ppm is the range).

The two occurrences of possible concern were the lack of a nitrogen overpressure while the MSIV work was undertaken and the lack of sampling between September 25, 1989 and November 16, 1989. During wet-layup, steam generator sampling is accomplished via the wet-layup recirculation line. This line was out of service due to repair of a valve in the recirculation system. The first hydrazine analysis after this period showed no appreciable depletion in hydrazine concentration. Subsequent samples taken after restoration of the recirculation system showed a slight change from 155 to 123 ppm. Similar conditions existed for SG#2 during this time period with hydrazine still well within the band of 75 to 200 ppm as prescribed in procedure 74AC-9CY04. And again the sodium, chloride, and sulfate were well within the wet-layup specification. Both steam generators had hydrazine contents of at least 80 ppm, with chloride, sulfate, and sodium below the 1 ppm limit. Both steam generators had one day where the pH dropped below the specified 9.8.

The concern with the lack of nitrogen overpressure and wet layup chemistry would be the potential impact of corrosion on the steam generator. In order to evaluate this possibility, an evaluation was undertaken by the steam generator manufacturer.

ATTACHMENT
(Continued)

The materials of construction for the steam generators are grouped in the following categories:

- Alloy 600 is used for the heat transfer steam generator tubes.

- Ferritic stainless steel (type 405 or 409) used for eggcrates, batwings and flow distribution plate.

- Low alloy steels or carbon steel is used for tubesheet stay, shells, baffles, dryers and separators.

In summary, the lack of nitrogen overpressure surveillance when considered in conjunction with the remaining wet-layup chemistry is not expected to have any adverse corrosion effects. Specifically, pitting should not occur in the Alloy 600 tubing. Although the water was exposed to oxygen, the pH was maintained between 9.8 and 10.5, (except for one day where the pH dropped to 9.7) above which should prevent $CuCl_2$ induced pitting of Alloy 600 tubing. Therefore, general corrosion is not a problem with Alloy 600 at these shutdown conditions.

General corrosion is also not a problem with ferritic stainless steels containing at least 11% Cr at these shutdown conditions.

General corrosion is a concern with low alloy and carbon steel surfaces exposed to the vapor phase during wet layup if sufficient nitrogen overpressure is not maintained. It is assumed that atmospheric air would eventually replace the nitrogen originally present. As such, the presence of oxygen in the vapor space permits the oxidation of the protective magnetite ($Fe_3O_4$) to the less protective hematite ($Fe_2O_3$ - rust). At startup, this will add to the amount of material that must be processed and removed. In the immersed section, general corrosion of the carbon and low ally steels will not occur due to the high pH and the presence of hydrazine.

The presence of oxygen in the vapor space of the steam generators during shutdown should not affect the integrity of the system. Its presence however, is expected to increase the general corrosion to the exposed carbon steel and low alloy steel surfaces and create more sludge.

The accident or event of concern regarding the steam generators would be a steam generator tube rupture (SGTR). The radiological releases calculated for a SGTR event with a loss of offsite power and a fully stuck open atmospheric dump valve (ADV) are well within the guidelines of 10 CFR Part 100. The RCS and secondary system pressures are well below 110% of the design pressure limits, thus assuring the integrity of these systems.

ATTACHMENT
(Continued)

Additionally, no violation of the fuel thermal limits occurs, since the minimum DNBR remains above the 1.24 value throughout the duration of the event.

Based on the July 1989 100% eddy current examination where no significant wear patterns or corrosion buildup was observed and the fact that chemistry was maintained during wet-layup, the proposed change will not increase the probability of an accident previously evaluated. The proposed change will not increase the consequences of an accident previously evaluated due to the fact that in the event of a steam generator tube rupture the calculated radiological releases are well within the guidelines of Part 100.

Standard 2 -- Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed Technical Specification amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The Chapter 15 analysis assumes that the plant is challenged by a SGTR that includes additional events and failures beyond those postulated by the NRC Standard Review Plan (SRP) 15.6.3. In addition to the conservative assumptions of the SRP (loss of offsite power, iodine spiking, etc.), this analysis postulates that the operators open an ADV on the affected steam generator and that it both runs to the full open position and sticks full open for the duration of the transient. The results of which are well within the guidelines of 10 CFR Part 100 for any radiological releases and the RCS and secondary system pressures are well below the design pressure limits. Therefore the proposed Technical Specification amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 -- Involve a significant reduction in a margin of safety.

The proposed Technical Specification amendment does not involve a significant reduction in a margin of safety because no changes are being made to the way the facility is being operated. Thus, no new failure modes are being introduced.

If a SGTR were to occur, diagnosis of the event is facilitated by radiation monitors, which initiate alarms and inform the operator of abnormal levels and that corrective operator action is required. Additional diagnostic information is provided by RCS pressure and pressurizer level response indicating a leak, and by level response in the affected steam generator.

The most limiting SGTR event is for a leak flow equivalent to a double-ended rupture of a U-tube at full power conditions. This event has been analyzed for Palo Verde (UFSAR Section 15.6.3) and concludes that the

resultant radiological releases are well within 10 CFR 100 guidelines and the RCS and secondary system pressures are well below 110% of the design pressure limits and no violation of the fuel thermal limits occurs. Therefore, the proposed Technical Specification amendment will not involve a significant reduction in a margin of safety.

E. SAFETY ANALYSIS OF THE PROPOSED AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The amendment will not make changes to the facility. The additional time between surveillance intervals will not increase the probability of failure of equipment. The actual operating time on the Unit 1 steam generators will be within the twelve to twenty-four month requirement for the surveillance interval.

The Unit 1 steam generator tubes were 100% eddy current examined during the last refueling outage (July, 1989). The overall results after completion of the second fuel cycle are encouraging in that the total number of degraded and/or defective tubes were minimal and no significant wear patterns have been observed. (This is based on the results of three inservice inspections during the first two fuel cycles.) Therefore, the proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

The proposed Technical Specification amendment will not create the possibility of an accident or malfunction of a different type than any evaluated previously in the UFSAR. No physical changes are being made to the facility nor are there any changes being made which affect the operation of the facility. No new failure modes are being introduced by the change. The UFSAR Chapter 15 analysis includes a steam generator tube rupture with concurrent loss of offsite power and a stuck open ADV. This analysis concludes that radiological releases are within federal guidelines and the RCS and secondary pressures are well below the design pressure limits. Therefore the proposed Technical Specification amendment will not create the possibility of an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for any Technical Specification. During the Unit 1 second refueling outage 100% eddy current examination was performed on both steam generators with no corrosion related defects found. There has been little or no mechanical wear on the steam generators since the last refueling outage due to the fact of an unusually lengthy outage. If eddy current examinations were conducted during the January surveillance outage the steam generator will have had only seven

months of operations.

During the most recent Unit 1 refueling outage adherence to wet lay-up conditions were within specifications except for the period of time that nitrogen overpressure was not maintained to allow for the atmospheric dump valve modifications. Because lack of nitrogen overpressure was a deviation from established layup specifications the steam generator manufacturer was asked to evaluate the long term corrosion affects on steam generator materials that could be attributed to any out of specification layup condition.

In order to evaluate the corrosion impact of the out-of-specification layup condition on the Unit 1 steam generators, the materials of concern had to be identified.

The criteria for selection was considered the following; pressure boundary material (principally tubing), and large carbon steel surfaces which could contribute to sludge pile formation, support structures, and surfaces which could become fouled due to excessive corrosion (eg., separators and dryers). The following materials have been identified.

| Component | Material | Class |
|---|---|---|
| Tubes | SB-163 | Alloy 600 |
| Tubesheet | SA-508 Cl 1 | Low Alloy Steel |
| Tubesheet stay | SA-608 Cl 2 or 3 | Low Alloy Steel |
| Upper shell | SA-533 Gr A Cl 1 | Low Alloy Steel |
| Intermediate shell | SA-533 Gr B Cl 1 | Low Alloy Steel |
| Lower shell | SA-533 Gr A Cl 1 | Low Alloy Steel |
| Baffle | SA-515 Gr 70 | Carbon Steel |
| Eggcrates/Batwings | A-176 Ty 409 | Ferritic SS |
| Flow distribution plate | SA-240 Ty 405 | Ferritic SS |
| Dryers | A-245 Gr C | Carbon Steel |
| Separators | A-570-C/A-622 | Carbon Steel |

In performing this evaluation, rather than concentrate on every material, they were grouped in to the following categories that behaved in the same manner.

- Alloy 600 used for the heat transfer steam generator tubes.
- Ferritic stainless steel (type 405 or 409) used for eggcrates, batwings and flow distribution plate.
- Low alloy or carbon steels, used for tubesheet stay, shells, baffles, dryers and separators.

ATTACHMENT
(Continued)

Effect on Materials

a) Alloy 600

Whyte and Burchell (Reference 1) exposed Alloy 600 to solutions of (0.3 fold) seawater (6000 ppm chloride) with varying amounts of oxygen, hydrazine and ammonium hydroxide. Test temperature was 104°F. They found pitting to occur with oxygen contents of 0.1 ppm and above, but no pitting occurred with 0.01 ppm oxygen. In other tests with hydrazine additions (150 ppm) at this temperature, hydrazine had no beneficial effect. Ammonia additions to raise the pH, however, were significant. Pitting was retarded when the pH was raised form 7.6 to 8.5. Further retardation occurred at pH of 10.5. In their other pitting tests, Whyte and Burchell found that the presence of copper ion in chloride containing solutions greatly accelerated the rate of pitting over that observed in sodium chloride solutions at the same chloride concentration.

Sykes et al (Reference 2) also performed a pitting study on Alloy 600. Their tests to simulate layup conditions were run at 77°C (171°F) with a multitude of corrosive mediums containing chlorides or sodium sulfite or both. They found that environments containing $CuCl_2$ and $NO_2S_2O_3$ almost always produced pitting. Their other findings were as follows:

1. Sludge tended to inhibit pitting.

2. Pitting appears to be equally severe in aerated and deaerated environments.

3. Corrosion deposit was present in pits produced in about half of the environments which produced pitting.

4. Pitting was more severe in concentrated solutions than in dilute solutions.

5. Pitting developed within a few days.

6. Potentiostatic curves made of Alloy 600 in $CuCl_2$ (10,000 ppm Cl) showed only active corrosion (i.e., no passive behavior occurred). Also, the Alloy 600 had a higher oxidizing potential than did copper at this temperature.

7. The addition of hydrazine reduced the severity of pitting.

8. Although not described in their beaker tests to simulate shutdown, their experiments on thermal cycling; shutdown, heatup and high temperature (288°C) with pH of 9 were unsuccessful in that they could not induce pitting. The authors had to go to an acid media to induce pitting.

Sykes and Agrawal (Reference 3) continued this work on pitting. They found in their work in acid media, that as little as 1 ppm $CuCl_2$ could cause pitting in Alloy 600. Further, although oxygen had no effect on aggressiveness in concentrated solutions ($\geq$100 ppm $CuCl_2$), it did have an effect on dilute solutions. Deeper pits were found to form in dilute aerated solutions (<100 ppm $CuCl_2$) than in dilute deaerated solutions. In addition, they found that chemical inhibitors had little effect on pit initiation and propagation when added to 50 and 100 ppm $CuCl_2$ at 160°C.

Thus, in summary, acid solutions of $CuCl_2$ will cause pitting of Alloy 600. However, Whyte and Burchell found raising the pH above 8 will retard pit formation. Sykes et al were unable to cause pits with alkaline solutions (pH >9) even with oxygen present. Hydrazine reduced the pit depth, but did not stop the pitting. Therefore, as long as the aerated water in Palo Verde Unit 1 at shutdown had a pH of 9.7 and contained hydrazine, pitting would either not occur or would be severely reduced.

General corrosion of Alloy 600 is not a factor in shutdown conditions.

b) Ferritic Stainless Steels

Mentink et al (Reference 4) state that the advantage of stainless steel is resistance to corrosion. Chromium is the main alloying element, and the steel should contain at least 11% (Type 409 SS). Chromium is a reactive element, but it and its alloys passivate and exhibit excellent resistance to many environments. Types 409 SS, because of their chromium content (at least 11%) should have only very slight general corrosion and their function would not be impaired if pits were to form.

c) Low Alloy or Carbon Steels

Low alloy or carbon steels can be grouped together as neither contains as much as 11% chromium, the alloying element that makes stainless steel, stainless. Upon exposure to high temperature, high pressure, deaerated water, the steel forms a protective layer of magnetite, $Fe_3O_4$. During any oxygenated conditions of layup, the immersed sections of carbon/low alloy steel will be protected both by the high pH and by the presence of hydrazine (Reference 4). In the vapor areas, where high pH and hydrazine cannot protect the carbon/low alloy steel, the magnetite will oxidize to form hematite, $Fe_2O_3$ (Reference 5).

This high oxidation state oxide ($Fe_2O_3$), however, is not as adherent as the lower form and will spall off during startup to increase the sludge level in the secondary water system. The amount that will form, however, is small in comparison to the normal operational sludge. Cleary (Reference 6) estimates, that for a two month shutdown, only 7% of the normal yearly sludge is formed. For a six month layup, Cleary estimates that 21% of the sludge forms. Thus, for a two month layup with air exposure, the amount of sludge generated would be significantly less than for normal running at steaming conditions. This added sludge generated will add to the amount of filtration that must be done during startup.

Based on the evaluation, the following conclusions were drawn for the steam generator layup conditions for Unit 1.

1) Maintaining the pH above 9.5 will preclude pitting of the Alloy 600 heat transfer tubes.

2) No accelerated general corrosion is anticipated for the ferritic stainless steel components.

3) The carbon steel and low alloy steel components, present in the vapor phase, are expected to have a portion of the protective $Fe_3O_4$ film oxidized to $Fe_2O_3$ which may flake off in startup.

4) The carbon steel and low alloy steel components, present in the liquid phase, will be protected by the high pH and hydrazine and thus, not adversely affected by the presence of air in the vapor phase during the wet layup period.

Based on conclusions from the steam generator manufacturer, no corrosion mechanism should have been initiated during this refueling outage which would require an eddy current examination prior to the next scheduled refueling. Therefore, the proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for any Technical Specification.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed Technical Specification amendment request does not involve an unreviewed environmental question because operation of PVNGS Unit 1 in accordance with this change would not:

(1) Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board, Supplements to the FES, Environmental Impact Appraisals, or in any decisions of the Atomic Safety and Licensing Board; or

(2) Result in a significant change in effluents or power levels; or

(3) Results in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION PAGE

(See attached page 3/4 4-13)

## REFERENCES


1. D. D. Whyte and R. C. Burchell, "Laboratory Program to Examine Effects of Layup Conditions of Pitting of Alloy 600," EPRI-NP-3012, April 1983.

2. J. F. Sykes, A. K. Agrawal and W. E. Berry, "Pitting Corrosion of Alloy 600 Steam Generator Tubing: Results of a Laboratory Scoping Study," EPRI NP-3905, February 1985.

3. J. F. Sykes and A. K. Agrawal, "Pitting in Steam Generator Tubing: Causes and Corrective Actions," EPRI-NP-5207, June 1987.

4. C-E letter NLM-74-178, P.E.C. Bryant to J. J. Kurpen and D. F. Streinze, April 9, 1974.

5. H. W. Mentink, K. A. Hagen, C. G. Lovett, "Plant Layup and Equipment Preservation Sourcebook," EPRI-NP-5106, March 1987.

6. W. F. Clearly, "Evaluation of Secondary-System Layup and Cleanup Practices and Processes, EPRI-NP-2977, April 1983.