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U. S. Nuclear Regulatory Commission
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Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Response to Generic Letter 97-06: "Degradation of Steam Generator Internals"

Dear Sirs:

Enclosure 1 provides Arizona Public Service Company's (APS) response to Generic Letter 97-06: "Degradation of Steam Generator Internals".

Commitments Made in this Letter

This letter does not make any commitments by APS to the NRC.

Please contact Mr. Scott Bauer at (602) 393-5978 if you have any questions or would like additional information regarding this matter.

Sincerely,

James M Levine

JML/SAB/RMW/rlh

Enclosure 080014

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A001/1

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, J. M. Levine, represent that I am Senior Vice President - Nuclear, 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.

JM Levine
J. M. Levine

Sworn To Before Me This 30th Day Of March, 1998.

Christine A. LaBrash
Notary Public

My Commission Expires

August 29, 1999

ENCLOSURE 1

Response to Generic Letter 97-06:
"Degradation of Steam Generator Internals"

NRC Request 1:

Provide a discussion of any program in place to detect degradation of steam generator internals and a description of the inspection plans, including the inspection scope, frequency, methods, and equipment.

The discussion should include the following information:

(a) Whether inspection records at the facility have been reviewed for indications of tube support plate signal anomalies from eddy-current testing of the steam generator tubes that may be indicative of support plate damage or ligament cracking. If the addressee has performed such a review, include a discussion of the findings.

(b) Whether visual or video camera inspections on the secondary side of the steam generators have been performed at the facility to gain information on the condition of steam generator internals (e.g., support plates, tube bundle wrappers, or other components). If the addressee has performed such inspections, include a discussion of the findings.

(c) Whether degradation of steam generator internals has been detected at the facility, and how the degradation was assessed and dispositioned.

APS Response:

Introduction

PVNGS has addressed the requirements of Generic Letter (GL) 97-06 through an integrated generic industry and PVNGS-specific program. The program includes: (1) information addressing the applicability of steam generator internals degradation found at foreign and domestic pressurized water reactor (PWR) facilities, (2) a summary of visual, video and eddy current inspection results, where applicable, and (3) generic engineering and safety assessments regarding the design related applicability of identified steam generator secondary side internals degradation in CE-designed steam generators.

APS participated in both the Nuclear Energy Institute (NEI) Steam Generator Internals Task Force and the Combustion Engineering Owner's Group (CEOG) Steam Generator Task Force (SGTF) programs. This response highlights the results of the generic activities as they apply to PVNGS. Additionally, plant-specific activities undertaken at PVNGS are summarized. Based on the actions described below, APS concludes that no steam generator internals degradation has occurred to date, and the potential for internals degradation is low. This engineering position has been validated through inspection activities performed at the PVNGS units.

Background

In response to the issuance of a proposed GL on degradation of steam generator internals, NEI formed the Steam Generator Internals Task Force in January 1997. The purpose of this task force was to develop a coordinated industry-wide response to the proposed GL. Participation on this task force included the Electric Power Research Institute (EPRI), licensees (including APS), and representatives of the owners groups for each domestic steam generator design.

The Steam Generator Internals Task Force agreed that each owners group would initiate programs to assist its respective owners in assessing the susceptibility of tube damage or loss of decay heat removal (DHR) capability due to steam generator secondary-side degradation. An integral component in this assessment was an understanding of the applicability of the degradation found in several foreign units to domestic steam generators. EPRI responded to this need and with the cooperation of Electricite de France (EdF) developed Reference 2, GC-109558, *Steam Generator Internals Degradation: Modes of Degradation Detected in EDF Units*. Reference 2 provides evaluations of the causal factors involved in the modes of degradation experienced in the French units. Prior to the issuance of Reference 2, the CEOG also prepared report CE NPSD-1079-P, *Evaluation of NRC Information Notice 96-09, Including Supplement 1, Relative to Combustion Engineering Steam Generator Designs, April 1997* (Reference 3). The CEOG used both of these reports to gain insights into the applicability of the French experience to the CE steam generator designs and operating experience. Reference 2 was transmitted to the NRC via an NEI letter, dated December 19, 1997.

In assessing domestic plant susceptibility, the Steam Generator Internals Task Force considered all attributes considered to have an impact on degradation potential. These attributes included: steam generator internals design, fabrication and manufacturing techniques and plant operating history, including chemistry and related degradation, such as denting. Additionally, the owners groups compiled and assessed information on their respective visual, video and pertinent non-destructive examination (NDE) inspection experience to further enhance their evaluations regarding the susceptibility to internals degradation.

Additionally, the industry groups interacted with the NRC Staff. The Steam Generators Internals Task Force met with the NRC in May 1997, to gain a better understanding of the safety concerns discussed in the generic letter. Similarly, the CEOG SGTF met with the NRC in July 1997 (Reference 1), to discuss operability assessments performed in light of discovered tube support degradation in a CE-designed steam generator. As a result of these efforts, the CEOG developed safety and susceptibility assessments relative to the design and operating history of CE-designed steam generators. These assessments provide reasonable assurance that steam generator tube integrity and DHR capability is not compromised by internals degradation.

Finally, in an effort to coordinate the industry's response to Generic Letter 97-06, it is the intent of APS and the CEOG SGTF to provide the generic CEOG reports (References 5, 7 and 12) via NEI for NRC Staff information by April 30, 1998. The reports developed for the CEOG SGTF are discussed in the following section. PVNGS plant-specific efforts are discussed separately in the section entitled "PVNGS Activities".

<u>CEOG Activities to Address Generic Letter 97-06</u>

As a member of the CEOG, APS has participated in a formal evaluation of steam generator internals degradation experience at EdF facilities and domestic experience from CE-designed units, including PVNGS Units 1, 2 and 3. The CEOG has provided several additional means to convey steam generator internals degradation information to its members. APS, through its participation in the SGTF, regularly exchanges information with other CEOG members regarding steam generator inspections, evaluations, and issues. Additionally, APS through its chairmanship of the SGTF is actively involved in information exchanges with NEI and EPRI.

Initially, the EdF steam generator internals degradation experience was discussed and an evaluation conducted by ABB-CE under CEOG SGTF funding after the NRC Information Notice 96-09 was issued. The evaluation found that the specific causal factors identified by the authorities at EdF were not applicable to the CE design. Subsequently, in cooperation with the Steam Generator Internals Task Force, the CEOG SGTF funded an evaluation of the broader question of all types of steam generator internals degradation. The evaluation conducted by the CEOG dispositions all potential degradation mechanisms identified by review of steam generator secondary internals experience on the basis of design, manufacturing, and operational practice. CE-designed steam generators can be divided into three different groups based on tube support design: steam generators with carbon steel eggcrate supports with drilled plates at upper elevations; steam generators with carbon steel eggcrate supports only; and steam generators with ferritic stainless steel eggcrate supports only.

The CEOG program, in support of the NEI led industry effort, addresses the susceptibility of CE-designed units to identified degradation mechanisms and develops safety assessments of applicable degradation modes to ensure continued capability to maintain tube integrity and decay heat removal. In response to Items (1)(a) and (b) of GL 97-06, the CEOG assessments include a review and evaluation of inspection information. The steam generator internals of the various CEOG member utilities encompassing the three different steam generator tube supports designs identified above have been inspected. Table 1 provides a summary of the inspections that have been performed.

In response to Item (1)(c) of GL 97-06, the CEOG program also includes the development of three (3) reports regarding the assessment and disposition of potential degradation. The objectives and conclusions of the CEOG reports are summarized below.

I. CEOG Report, CE NPSD-1092, "Evaluation of Degraded Secondary Internals - Operability Assessment" (Reference 5).

OBJECTIVES:

1. Assess the applicability of EdF damage mechanisms to the CE-designed steam generators.

2. Assess the applicability of tube support erosion-corrosion experience at Maine Yankee and San Onofre Nuclear Generating Station (SONGS) Unit 3 to other CE-designed units.

3. Assess the impact of applicable damage mechanisms on tube integrity and decay heat removal capability.

CONCLUSIONS:

1. The primary damage mechanism related to the wrapper support failures in the French units are not applicable to CE-designed steam generators.

2. Tube support plate cracking is a residual effect of tube denting, but is not detrimental to the safe operation of the steam generator.

3. Adequate design margins against tube support failure have been demonstrated for the Flow-Accelerated Corrosion (FAC) damage observed in the SONGS Unit 3 eggcrate supports.

4. There are no reported tube wear indications directly related to tube support degradation.

5. Plants with degradation of tube supports, such as that observed at SONGS Unit 3, can continue to operate safely because adequate design margins against tube support failure exist and possible tube damage from support degradation can be detected in the normal ECT examinations.

6. Corrosion degradation of carbon steel eggcrate tube supports, as manifested by denting, such as detected in the lower tube supports of the Millstone Unit 2 original steam generators, has been determined to be acceptable by model boiler testing and analysis. Current chemistry practices can mitigate existing denting, and preclude further degradation due to denting.

7. Calvert Cliffs Unit 2, which may have experienced more tube support degradation than SONGS Unit 3, can continue to operate safely, because adequate design margins against tube support failure can be demonstrated and possible Flow Induced Vibration (FIV) tube wear is postulated to result in a leak-before-break (LBB) scenario.

8. CE steam generators designed with stainless steel tube supports, such as the steam generators at PVNGS, are significantly less susceptible to erosion-corrosion than the steam generators with carbon steel tube supports.

9. None of the degradation mechanisms reviewed pose a threat to the RCS pressure boundary integrity or the heat removal function of the steam generator.

CEOG Report, CE NPSD-1103, "Evaluation of Susceptibility of Internals Degradation in CE Designed Steam Generators" (Reference 7).

OBJECTIVES:

1. Review the history of steam generator internals degradation in CE-designed units.

2. Examine the susceptibility of CE-designed units to internals degradation mechanisms that have occurred in CE-designed and EdF steam generators.

CONCLUSIONS:

1. CE-designed steam generators have not encountered a significant amount of internals degradation. Of the degradation that has occurred, appropriate mitigating action has been taken to minimize the effect of this degradation.

2. The most common form of steam generator internals degradation has been from water hammer events or from erosion of components within the feedwater system. However, these degradation mechanisms are not considered to be safety significant.

3. The steam generator internals degradation mechanisms described in GL 97-06 are generally not applicable to CE-designed steam generators. The only degradation mechanism applicable to CE-designed steam generators that could have safety significance is FAC of peripheral eggcrate supports.

4. FAC of peripheral eggcrate support locations is primarily the result of secondary fluid flow redistribution caused by severe tube bundle fouling. Use of ammonia for pH control in heavily fouled steam generators increases the susceptibility to FAC.

5. Steam generators with ferritic stainless steel eggcrate supports, such as that used in the PVNGS units and Palisades replacement steam generators, have a chromium content of at least 10.5% which increases the resistance to FAC by at least an order of magnitude. Therefore, these supports are not considered susceptible to FAC.

6. Of the CE-designed steam generators with carbon steel tube supports, only those units with severe tube bundle fouling, as indicated by significant steam generator secondary pressure loss, may be susceptible to FAC of peripheral eggcrate supports.

7. No CEOG member plants have detected by NDE or visual inspections any FAC of drilled tube support plates to-date.

CEOG Report, CE NPSD-1104, "Evaluation of Degraded Secondary Internals - Bounding Analysis" (Reference 12).

OBJECTIVES:

1. Assess the impact of each steam generator internals degradation issue that is applicable to CE-designed steam generators.

2. Determine the bounding cases for degradation issues affecting nuclear safety.

PRELIMINARY GENERAL CONCLUSIONS:

At the time of this letter, Reference 12 has not been finalized. Preliminary general conclusions are provided here, but will not be final in accordance with CE QA requirements until April 1998. Therefore, this preliminary information should not be used in making any final assessments.

1. No CE designed plants are at risk for loss of tube integrity or decay heat removal function as a consequence of steam generator secondary side internals degradation, including FAC degradation for the limiting case.

<u>Applicability of CEOG findings to PVNGS</u>

The CEOG evaluation provides assurance that the only credible steam generator internals components damage mechanism of potential safety significance to CE-designed steam generators is FAC of eggcrate type tube supports. FAC of eggcrate supports has been detected in CE-designed units, as reported in Generic Letter 97-06. In general, FAC of tube supports is considered credible for units with carbon steel tube supports if tube bundle fouling results in the redistribution of flow such that FAC threshold velocities are exceeded. The CEOG evaluations indicate that FAC without substantial fouling is unlikely. Experience has indicated that the onset of substantial fouling is evidenced by a reduction in the normal plant operating steam pressure. Experience has also shown that plants can experience steam pressure reduction and not experience the onset of FAC in the tube supports.

In the event of a substantial reduction in steam pressure, the CEOG, supported by ABB-CE, analysis recommends that an inspection for the onset of FAC of tube supports be conducted at the next scheduled outage. FAC occurs preferentially toward the periphery of the hot leg side of the tube bundle at the upper tube supports. If warranted by a substantial steam pressure loss, remote visual inspection of the uppermost eggcrate supports will determine whether FAC has occurred. Reference 7 provides recommendations regarding the need to perform inspections.

With respect to applicability to PVNGS, the analyses and evaluations conducted by ABB-CE for the CEOG conclude that units with stainless steel eggcrate supports have a very low susceptibility to FAC of tube supports. The tube support design for the PVNGS Units are shown in Figure 1. The horizontal tube supports located along the vertical section of the tubes are all of the eggcrate design. The bend and horizontal regions of the tubes are supported by batwing and vertical lattice supports, respectively. All of the tube supports are manufactured from 409 ferritic stainless steel. To minimize sludge accumulation on the tubesheet, the PVNGS steam generators also include flow distribution plates (FDP) designated as 01H and 01C in Figure 1. The FDPs are 405 ferritic stainless steel plates with holes sized to provide a sweeping action across the tubesheet and reduce the particle dropout leading to sludge formation. Testing conducted by ABB-CE and EPRI (Reference 16) demonstrates that denting and associated plate damage is not credible for type 400 series stainless steel.

References 5 and 7 attribute tube fouling and magnetite deposition as a causal factor leading to conditions promoting FAC. Full bundle chemical cleaning has been conducted in all three units at PVNGS. Inspections performed by PVNGS confirm that tube deposits were removed. Since chemical cleaning, PVNGS has established targets for iron transport to the steam generators at less than or equal to 2.3 ppb. Actual valves have ranged from 2 – 5 ppb. Therefore the conditions leading to FAC in other CE-designed steam generators are not present at PVNGS.

Finally, the PVNGS steam generator design does not include carbon steel drilled tube support plates, and as such is not susceptible to failure mechanisms specifically associated with those supports due to the design differences.

Based on the information and conclusions developed by the CEOG, APS concludes that the steam generators at PVNGS are not susceptible to the degradation mechanisms identified at EdF and in other CE-designed steam generators. Despite this conclusion, additional inspections and evaluations have been performed at PVNGS and are described in the following section. All activities conducted by PVNGS confirm the results of the CEOG evaluation.

PVNGS Activities

As indicated in the CEOG evaluations, the steam generators at PVNGS are not considered susceptible to the degradation experienced at EdF and other CE-designed units based on the design features and operating experience of the PVNGS steam generators. Despite these conclusions, PVNGS has conducted inspections and evaluations of secondary internals as part of the PVNGS Steam Generator Degradation Management Program. All inspections and evaluations confirm the conclusions reached in the CEOG program. The following information summarizes the actions taken by PVNGS to address Item (1) of GL 97-06.

Chemical Cleaning

As indicated in Reference 2, tube support damage (ligament thinning) was identified by EdF at the Fessenheim 2 facility. The cause of damage was determined by EdF to be excessive flow velocities during recirculation of the iron step chemical cleaning solvent during the 1992 chemical cleaning event. Since PVNGS has conducted chemical cleaning in all three units at PVNGS, an evaluation of the conditions which led to the degradation at Fessenheim Unit 2 was conducted.

Several specific differences with process chemicals and application and steam generator design indicate that the issues identified at Fessenheim Unit 2 are not applicable to PVNGS. With respect to process chemicals, EdF used a proprietary process that was studied and rejected (Reference 3) during the early development of the EPRI/SGOG (Steam Generator Owners Group) process. The EPRI/SGOG process was applied at PVNGS Units 2 and 3 and a modified high temperature (BW Nuclear Technologies High Temperature Chemical Cleaning, BWNT HTCC) process was applied in Unit 1. The EdF process can be compared and contrasted with the applications at PVNGS as follows:

	EdF Process	EPRI/SGOG	BWNT HTCC
Chelant	Gluconic and Citric Acids	EDTA	EDTA
pH	3.3	6.0	8.0
Temperature	185°F	250°F	290°F
Corrosion Inhibitor	P6 by Multiserve	CCI-801	CCI-801
Atmosphere	Nitrogen	Nitrogen	Nitrogen

Note, specifically, the higher acidity of the EdF iron solvent (pH of 3.3 compared to the more neutral pH range of 6-8 associated with the PVNGS solvents). Additionally, EdF reused the iron solvent in the copper dissolution step. This was not done at PVNGS. The two-time application of the EdF solvent is considered to contribute to higher corrosion rates. In Reference 4, APS reported the results of corrosion testing performed for the PVNGS chemical cleaning. It should be noted with respect to tube support degradation, that the solvents applied at PVNGS had a negligible effect on 409 stainless steel samples representative of the steam generator tube supports.

The corrosion process at Fessenheim Unit 2 was further aggravated by the direct impingement of process chemicals on the upper tube support plates. EdF noted that in the most affected steam generator (steam generator 3), the solvent recirculation hoses had been misplaced to within 1.1 meters of the eighth support plate (See Figure 2). Conversely, the PVNGS solution was applied at the tubesheet level so as not to impinge directly on the tube supports. Post-cleaning video inspections revealed no indication of corrosion/erosion damage.

<u>Wrapper Support Failure</u>

EdF discovered wrapper support failure at Blayais Unit 3 in 1994. The failure was discovered as a result of an interference experienced during sludge lancing as the wrapper actually dropped 20mm. The failure was attributed to the inability of the EdF shroud design (Figure 3) to accommodate thermal expansion loads during expedited plant cooldown. The condition was aggravated by the presence of poor quality welds and potential fatigue induced crack propagation.

In References 3 and 5, ABB-CE assessed the applicability of the condition to CE-designed steam generators. The CE design has a shroud (or baffle) which not only separates the incoming feedwater and recirculating flow from the heat transfer tubing area, but also provides the main load path for the eggcrate supports. The shroud is a cylinder capable of handling the lateral loadings due to seismic forces on the tubes and tube supports.

The PVNGS steam generators are of economizer design. For this design, the baffle is segmented and separate for the economizer region. The baffle (Figure 4) is attached to the lower shell just above the tubesheet by eight lugs and is also attached to the two substantial lugs which support the secondary divider plate. The baffle is welded along the vertical length of the economizer. These ten lugs provide a point of fixation to the shell from which the baffle expands. The baffle is further supported at two higher elevations, each with sixteen lugs. These 32 lugs are cylindrical lugs with an external sliding collar. These lugs permit radial thermal expansion and limited differential thermal expansion, but restrain vertical and lateral movements due to seismic loading

on the tube bundle. A final area of support is provided at the top of the baffle by eight lugs welded to the baffle, which slide against the upper shell of the steam generator.

Based on a review of the EdF design and operational transients, ABB-CE and APS conclude that the conditions that led to failure at Blayais Unit 3 do not apply to PVNGS. The basis for this conclusion is that: (1) the CE design does not restrain thermal loads, and (2) excessive cooldown cannot not occur by design since the downcomer annulus can only be fed via a small downcomer feedwater nozzle. Additionally, welds used in the CE designs are backing bar type welds, which provide better welding access and better welding penetration to preclude any potential cracking of the wrapper or support blocks.

Although this condition is not considered applicable to PVNGS and would not require a dedicated inspection program, video inspections, as indicated in Table 1, were conducted in 1997 of the baffle support locations in Unit 3 and in 1996 in Unit 2. These inspections were conducted during steam generator modification activities reported to the NRC Staff in Reference 6. The inspections revealed no indication of degraded support lugs and welds. An example of the inspection findings is provided in Figure 5.

Based on this evaluation, and the results of video inspections conducted at PVNGS, APS concludes that the conditions experienced at Blayais Unit 3 are not applicable to PVNGS units. No further inspections are planned at this time.

Tube Support Degradation

In References 5 and 7, ABB-CE concluded that tube support corrosion and corrosion/erosion were the only internals damage mechanisms exhibited to date, both foreign and domestically, that are applicable to the CE steam generator design. Specifically, ABB-CE found that support plate cracking due to denting and FAC could occur in certain steam generator designs under applicable operating conditions. However, both ABB-CE and APS conclude that the design and operating history of the PVNGS steam generators would preclude either condition from occurring in the PVNGS Units. The basis for this conclusion is as follows:

With respect to the tube support plate cracking due to denting observed in older CE-designed steam generators, several specific design features and operating conditions eliminate this phenomenon as a potential damage mechanism at PVNGS. The PVNGS steam generators do not contain carbon steel tube support plates. All tube supports are of either eggcrate design or batwing and vertical strap. The tube supports are manufactured from Type 409 ferritic stainless steel. As stated previously, the PVNGS design also includes a flow distribution plate. The large flow area in both the FDP and tube 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 tube support plates and FDP are manufactured from Type 400 series ferritic stainless steel, they are not susceptible to the magnetite corrosion that has resulted in denting and lockup at plants with carbon steel tube supports. This expected observation with regard to denting has been substantiated via eddy current testing in all three units and during the steam generator tube pull activities that were conducted in PVNGS Unit 2. Additionally, industry studies have shown that boric acid addition is proven to further mitigate denting. All three units at PVNGS have been operating with boric acid addition since 1994. Finally, chemical cleaning conducted at PVNGS further reduced corrosion product build-up at tube/support intersections. Consequently, no further actions are required by APS to address this damage mechanism.

FAC-related damage of tube supports has been observed in EdF plants as well as two plants of CE steam generator design. ABB-CE has conducted an extensive evaluation of the causal factors that led to this support damage phenomenon. ABB-CE concluded that significant fouling resulting in elevated local velocities, combined with a susceptible material, resulted in a dislodging of the protective magnetite layer and initiated FAC at periphery eggcrate support locations. However, ABB-CE also concluded that the steam generator tube support materials and operating history at PVNGS result in significantly reduced susceptibility. As stated previously, the tube support material in the PVNGS steam generators is Type 409 stainless steel. This material has a minimum of a 10.5% chromium alloying element. NRC NUREG/CR-5007, *Prediction and Mitigation of Erosive-Corrosive Wear in Secondary Piping Systems of Nuclear Power Plants*, indicates that this level of chromium increases the resistance to FAC by several orders of magnitude over plain carbon steel.

In addition to low material susceptibility, APS has compared the tube fouling conditions at PVNGS with the conditions known to initiate FAC in other CE-designed steam generators. ABB-CE indicated in Reference 7 that FAC is a threshold phenomenon and degradation can occur in a relatively short period of time once the threshold is reached. As indicated, PVNGS conducted chemical cleaning in all three units in 1994-1995. The post-cleaning inspections revealed that the cleaning was extremely successful in the removal of magnetite deposits. Since 1995, PVNGS has set feedwater iron transport targets at less than or equal to 2.3 ppb. This value is five times lower than the feedwater iron transport levels of the CE plant with identified FAC damage. PVNGS actual feedwater iron valves have ranged from 2 – 5 ppb. Based on these conditions, PVNGS would not be expected to reach a threshold level for 30 - 50 effective full power years. Based on this evaluation, APS concluded that the PVNGS steam generators have negligible susceptibility to the FAC conditions that resulted in tube support damage at EdF and at domestic facilities with CE-designed steam generators.

This position has also been supported by a number of video inspections conducted by PVNGS either as result of steam generator tube pulls and steam generator modifications, or a specific secondary eggcrate inspection performed in Unit 2 in 1997 as a result of the findings at SONGS Unit 3. The PVNGS Unit 2 video inspection (Reference 10) was performed for the upper three eggcrate supports at two periphery locations on the hot leg side of steam generator 22. The batwing supports near these locations were also examined. None of these inspections found any evidence of erosion, thinning or fouling at the inspected locations. One small crack was observed at an eggcrate joint (Figure 6). The defect appears shallow, and was assessed by ABB-CE and APS to pose no threat to tube support integrity. The defect appears to be fabrication related and not operationally induced. Other bars in the general area showed no evidence of bending or cracking. Additional examples of findings from the 1997 inspection are provided in Figures 7-9. Based on this evaluation, the inspection findings, and the ABB-CE conclusions and recommendations contained in Reference 7, APS plans no further inspections at this time. APS will continue to monitor operational chemistry, and eddy current testing results, as well as trend industry information, to determine if future actions are required.

Other Secondary Side Internals Degradation Experience

In Table 2.1 of CEOG report CE NPSD-1103, *Evaluation of Degraded Secondary Internals Susceptibility Assessment* (Reference 7), ABB-CE summarizes all known degradation experience for CE-designed steam generators. The table identifies Feedwater System, Tube Bundle and Steam Drum Region issues and the steam generator models potentially affected. For the experiences listed, the System 80 design at PVNGS is not considered to be affected. This position is based on distinct design and material differences associated with the System 80 design. For example, the upper feedring at PVNGS has recently been replaced in all three units (1995-1996) as a result of steam generator modifications (Reference 6). The feedring material is Alloy 600, which is not considered susceptible to erosion/corrosion damage. The design is considered resistant to water hammer events as indicated by analysis and testing.

In 1992, Maine Yankee experienced severe erosion/corrosion damage of its steam generator moisture separators. The separators were of System 80 design. Although the damage at Maine Yankee was believed to be a result of significantly higher separator loading (245,000 lbs/hr versus 142,000 lbs/hr), PVNGS elected to conduct a steam generator entry in Unit 1, during a 1992 refueling outage, to inspect the moisture separators for similar damage (Reference 8). The inspection revealed no damage to the separators. Subsequent inspections conducted in 1995-1997 in all three units during the performance of steam generator modifications, confirmed that this phenomenon is not applicable to PVNGS.

With regard to secondary side foreign objects (loose parts), APS conducts eddy current testing and secondary side video inspection at every refueling outage at PVNGS for the presence of foreign objects which may interact and degrade steam generator tubing. The elements of the PVNGS steam generator foreign object program are contained in Reference 14. These elements include detection, evaluation, retrieval and mitigation methodologies. All tubes with detected loose part wear are removed from service regardless of size, if the offending object cannot be removed.

Future Actions

Based on the evaluations conducted by APS and by ABB-CE on behalf of the CEOG, no further specific actions are required as a result of this Generic Letter. APS, in accordance with the PVNGS Steam Generator Degradation Management Program, will continue to perform degradation assessments of industry and PVNGS-specific information, and implement any programmatic changes to assure safe and reliable operation of the PVNGS steam generators. The condition of secondary internals will be monitored, as capable, via comprehensive eddy current testing examinations and FOSAR (Foreign Object Search and Retrieval) activities that are performed at every refueling outage. APS through its continued participation with the CEOG SGTF, EPRI and NEI, will monitor industry trends and proactively incorporate information as necessary.

Summary

APS has reviewed the conditions and circumstances described by the NRC Staff in Generic Letter 97-06. APS has actively participated in generic industry groups to assess the potential for secondary side internals damage. APS, along with the CEOG and NEI, developed a program to detect and assess degradation of steam generator internals degradation. Despite assessment results that indicate that the PVNGS steam generators have low susceptibility to secondary internals degradation, APS undertook confirmatory actions to support the assessment conclusions.

Additionally, for steam generator activities after January 1, 1999, APS will adopt NEI 97-06, *Steam Generator Program Guidelines.* This guideline states in Section 3.9, "Maintenance of Steam Generator Secondary-Side Integrity":

Secondary-side steam generator components shall be monitored if their failure could prevent the steam generator from fulfilling its intended safety-related function. The monitoring shall include design reviews, an assessment of potential degradation mechanisms, industry experience for applicability, and inspections, as necessary, to insure degradation of these components does not threaten tube structural and leakage integrity or the ability of the plant to achieve and maintain safe shutdown.

As stated in APS letter to NRC, Response to Generic Letter 97-05: *Steam Generator Tube Inspection Techniques*, dated March 13, 1998 (Reference 17), APS has initiated this process, and the program guidelines will be fully implemented by January 1, 1999 as required by the NEI initiative.

<u>**NRC Request 2:**</u>

> **If the addressee currently has no program in place to detect degradation of steam generator internals, include a discussion and justification of the plans and schedule for establishing such a program, or why no program is needed.**

<u>**APS Response:**</u>

This request is not applicable to PVNGS.

Table 1

CEOG Member Plants SG Internals Inspection Data

Table 1
CEOG Member Plants SG Internals Inspection Data

UNIT	SG MODEL	LATEST INSPECTION DATE	INSPECTION TYPE	UPPER EGGCRATES		LOWER EGGCRATES		TUBE SUPPORT PLATE		SHROUD SUPPORTS	
				Inspected Yes/No	Degradation Note 1	Inspected Yes/No	Degradation (Note 1)	Inspected Yes/No	Degradation (Note 1)	Inspected Yes/No	Degradation Note 2
ANO-2	70 w/ TSP	May-97	Welch Allyn, NDE	Y	None	Y	None	Y	Note 3	N	-
CCNP1	67	May-96	Welch Allyn	Y	None	Y	None	N	-	N	-
FCS	Early	Fall 96	Remote Visual	N	-	N	-	Y	Note 4	Y	None
MY	Early	Apr-97	Fiber Optic	Y	Note 5	Y	Note 6	Y	Note 7	N	-
Palisades	RSG	Nov-96	Visual	Y	None	Y	None	NA	-	N	-
PSL2	67	Oct-95	Welch Allyn	Note 8	Note 9	N	-	N	-	N	-
PV2	Sys 80	Apr-96	Visual	N	-	N	-	NA	-	Y	None
		Oct-97	Visual	Y	None	N	-	NA	-	N	-
PV3	Sys 80	Mar-97	Visual	N	-	N	-	NA	-	Y	None
SO2	70	Dec-96	Remote Visual	Y (Note 10)	None	N	-	NA	-	Y	None
SO3	70	May-97	Remote Visual	Y	Y (Note 11)	Y	Y (Note 12)	NA	-	N	-
W3	70	Apr-94/Apr-97	Remote Visual	Y(Note 13)	None	Y(Note 14)	None	NA	-	N	-

Note 1: Examples of degradation: Cracking, EC Strip Thinning, Wear, TSP/Shroud Contact

Note 2: Examples of degradation: Shroud tearing/cracking, displacement, shroud/supp. separation

Note 3: TSP outer periphery cracks

Note 4: #8 TSP outer periphery: 1 post rim cut crack found in each SG.

Note 5: Highest EC localized area damage inactive

Note 6: Bottom-side in outer periphery rounded

Note 7: Outer periphery rows -- cracking

Note 8: Inspected upper 3 EC's periphery

Note 9: Minor fouling; no degradation observed

Note 10: Visual inspection of EC 6 -- 10

Note 11: Thinning, pitting, washboarding

Note 12: Thinning on EC 4 and 5

Note 13: Remote visual Apr-94 of SG#2 hot leg EC 10, 5 tubes in from periphery - no degradation.

Note 14: Visual inspection of EC 1 supports - no degradation.

ATTACHMENT 1

Figures

PVNGS Steam Generator Tube Supports

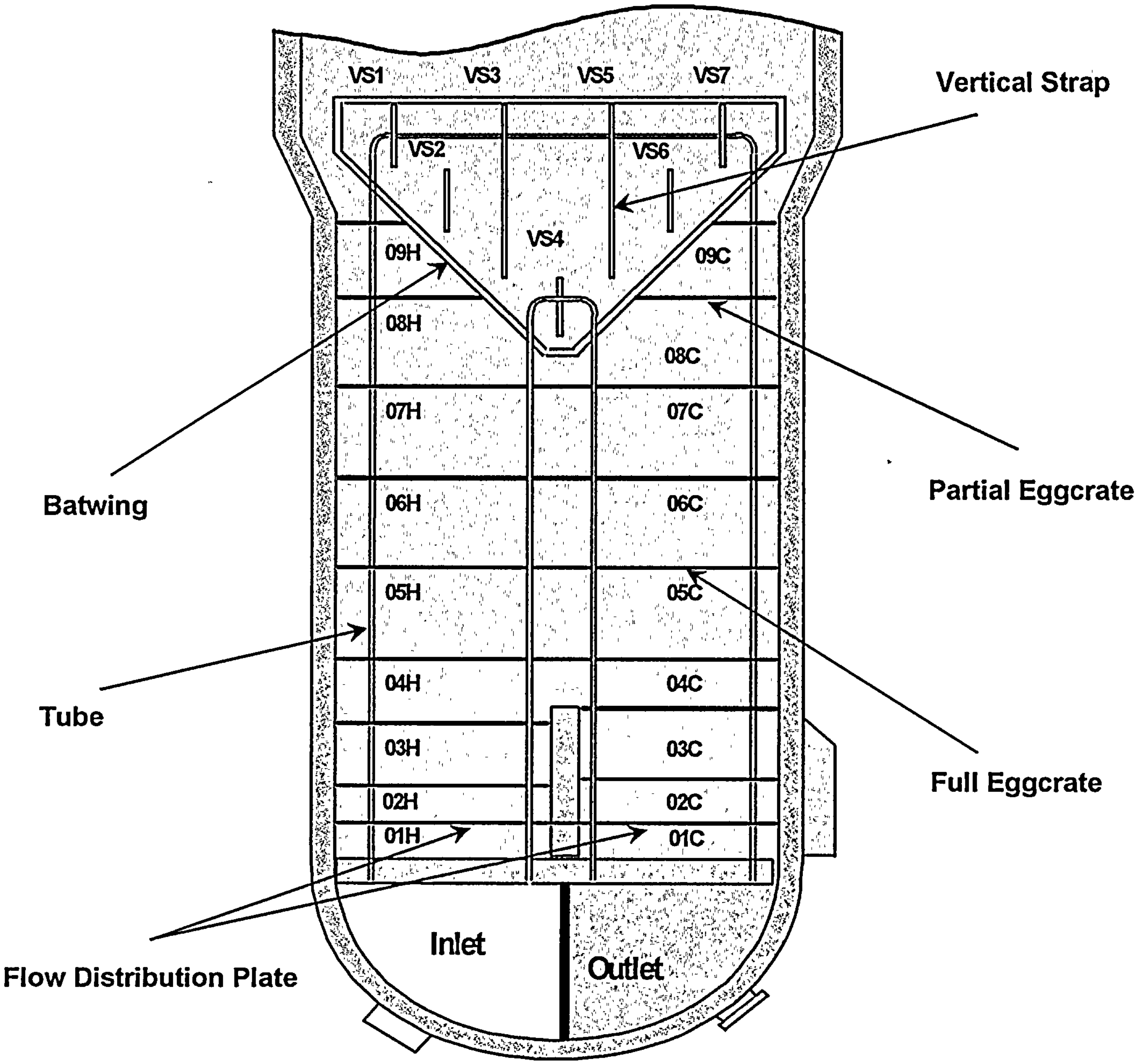


Figure 1

Chemical Cleaning at Fessenheim Unit 2

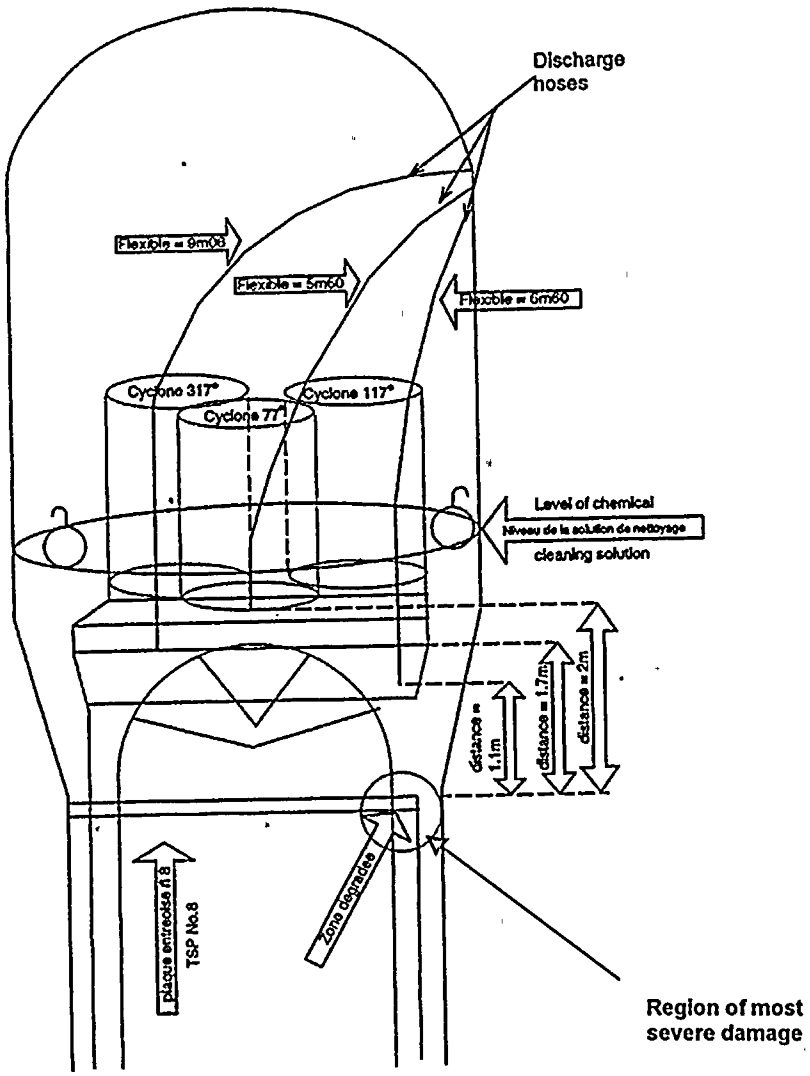


Figure 2

EdF Wrapper Support Design

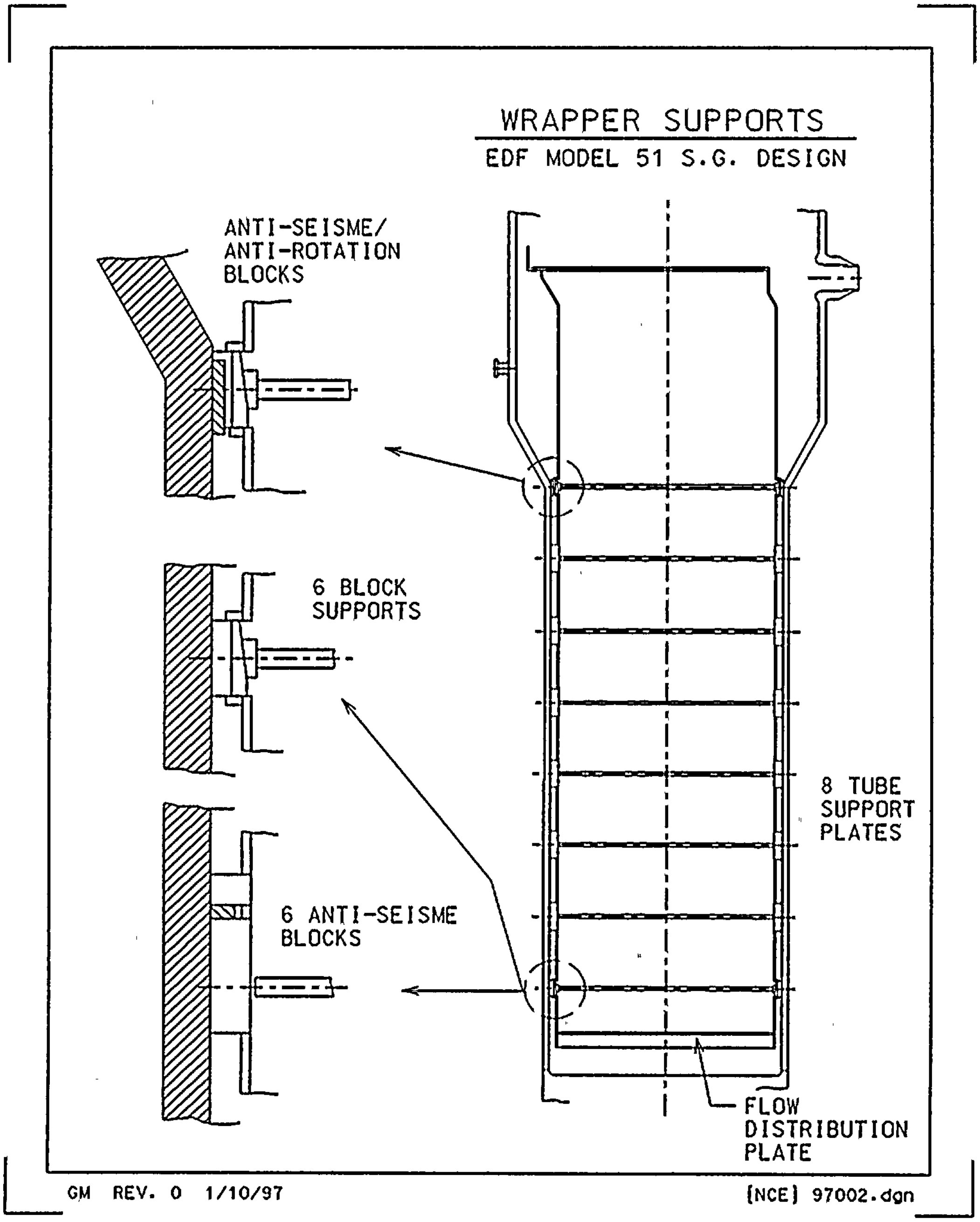


Figure 3

PVNGS Wrapper Support Design

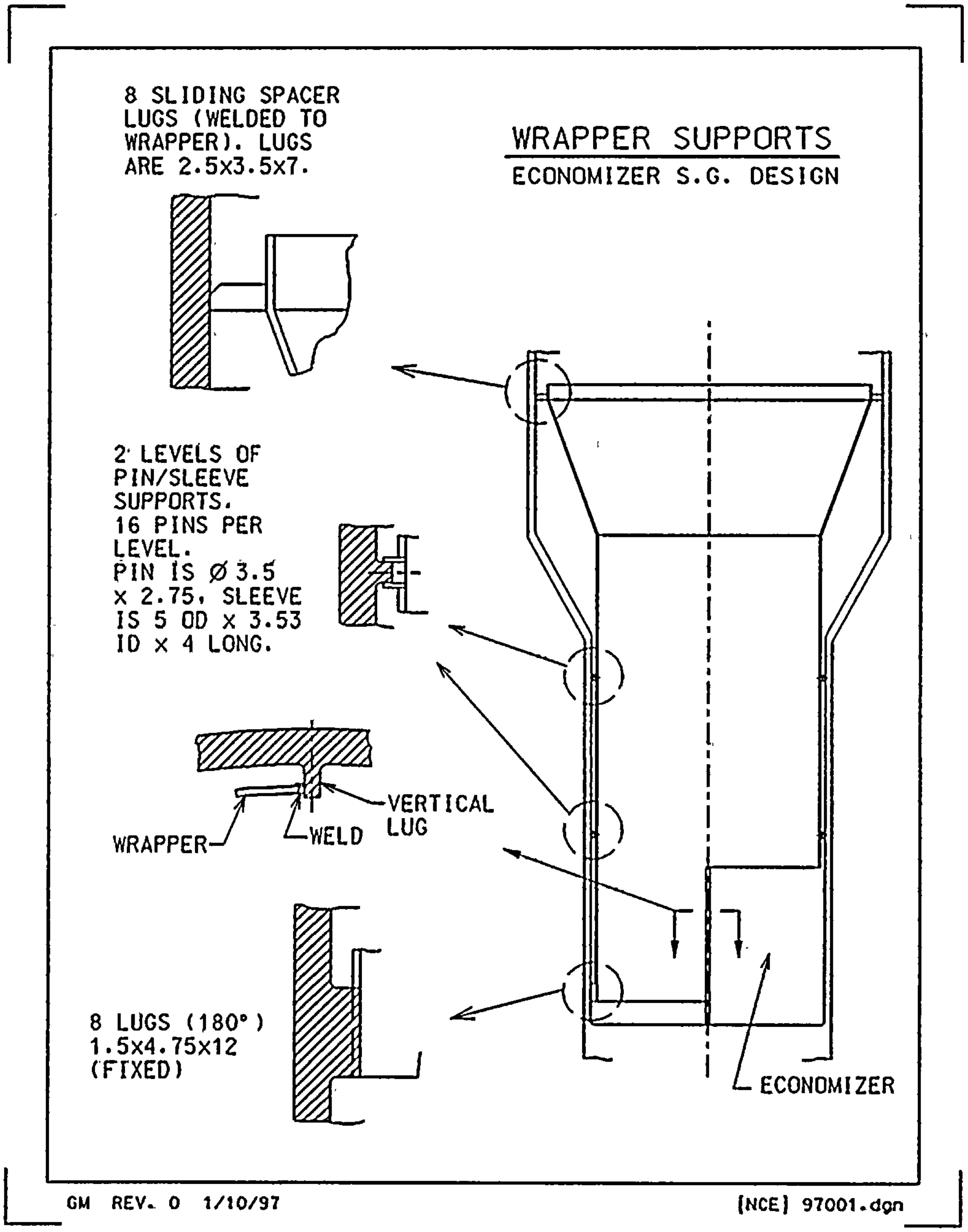


Figure 4

1997 Unit 3 Wrapper Support Inspection

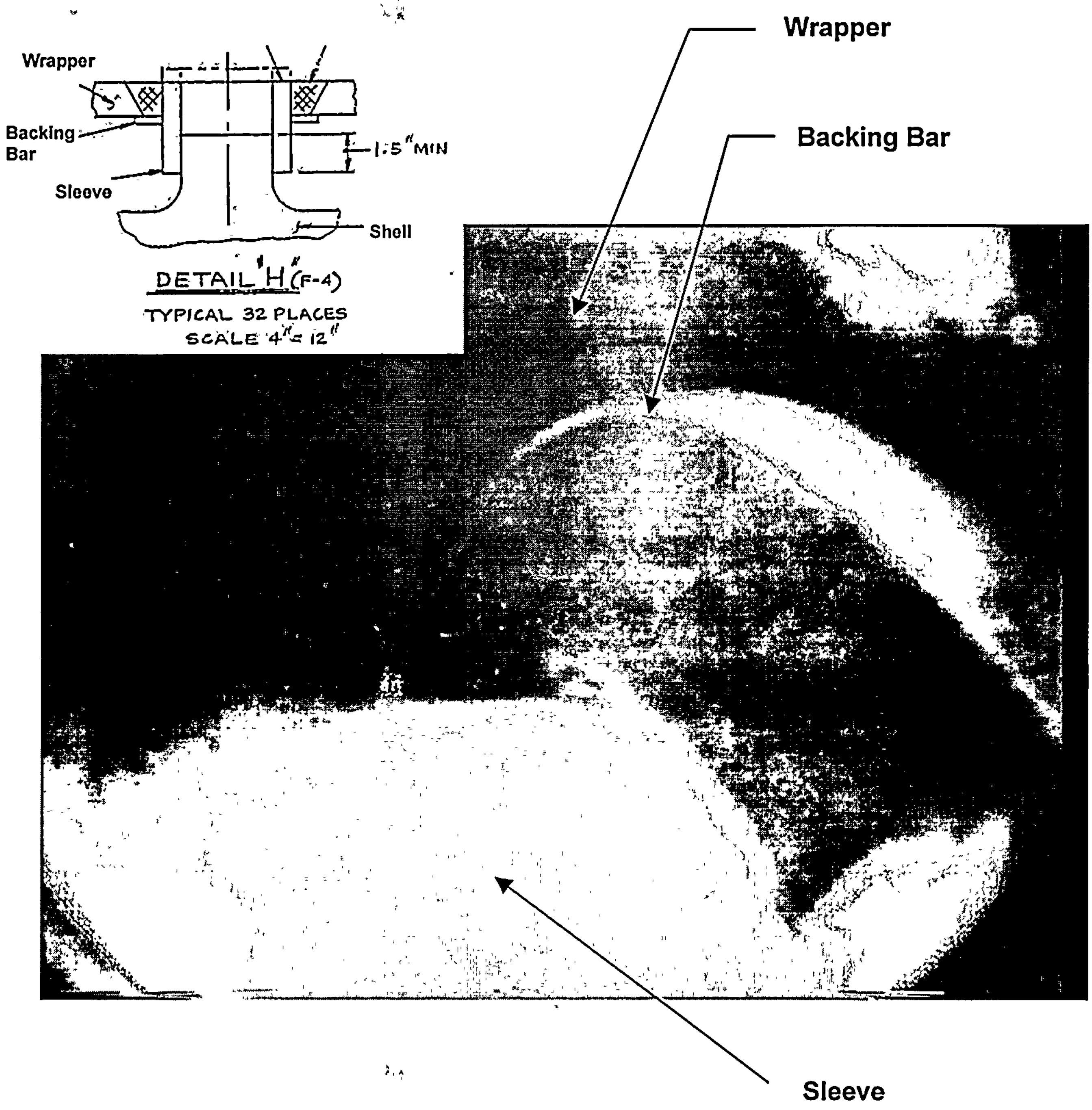
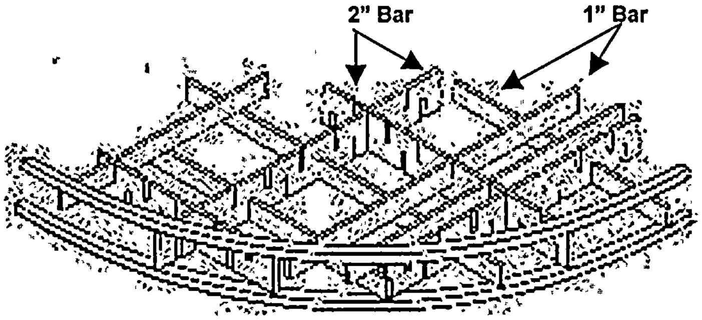
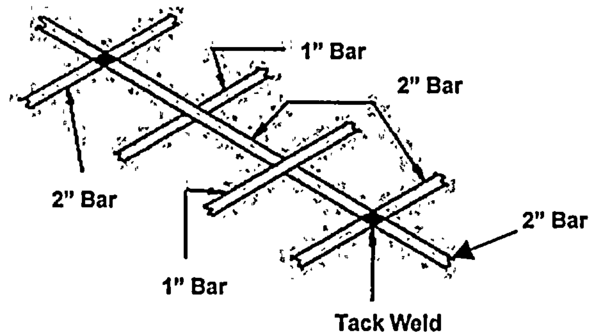


Figure 5



Eggcrate Assembly



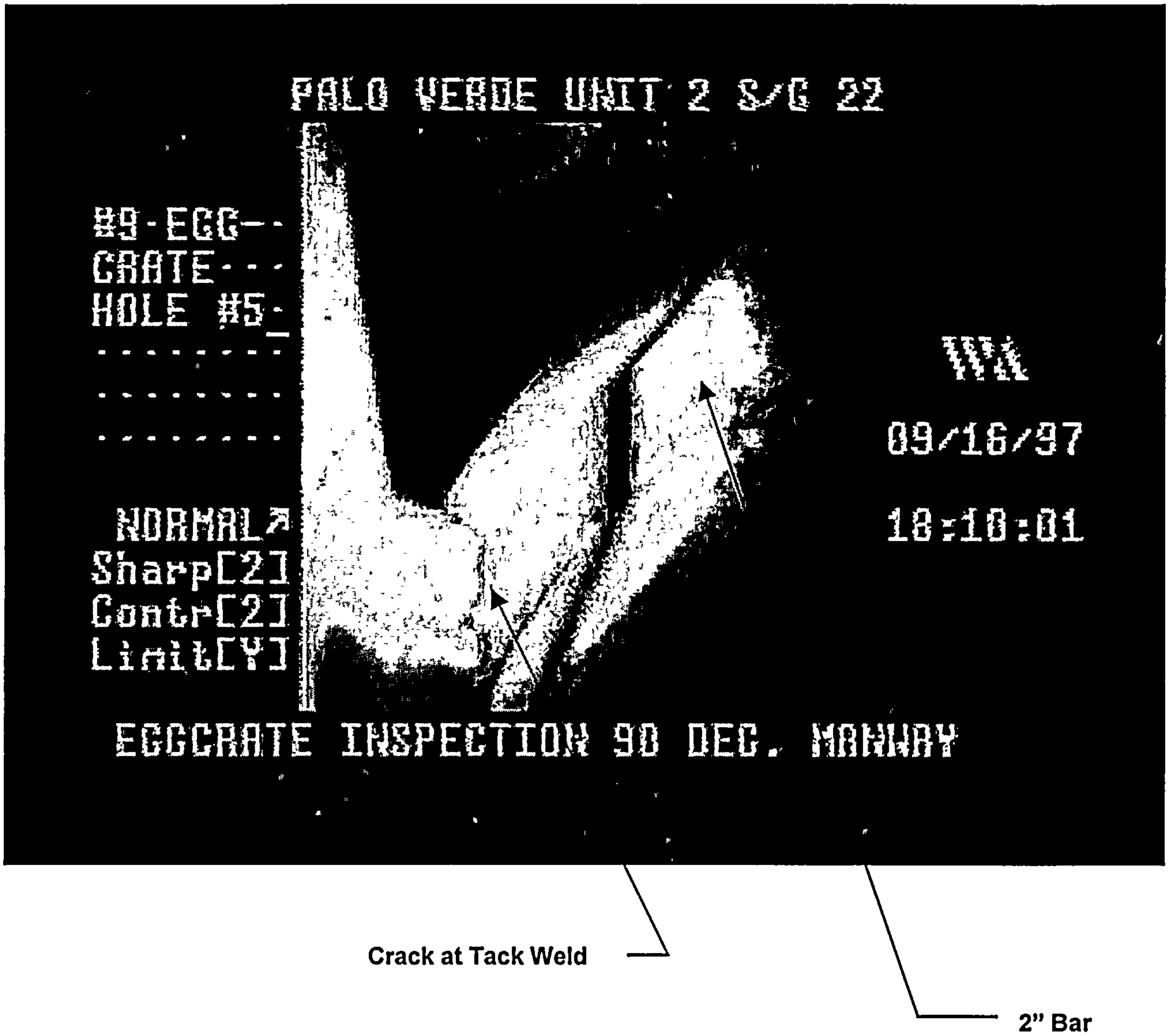


Figure 6

Unit 2 SG 22 # 9 (09H) EGGCRATE

Eggcrate

Tube

Figure 7

Unit 2 SG 22 # 8 (08H) EGGCRATE

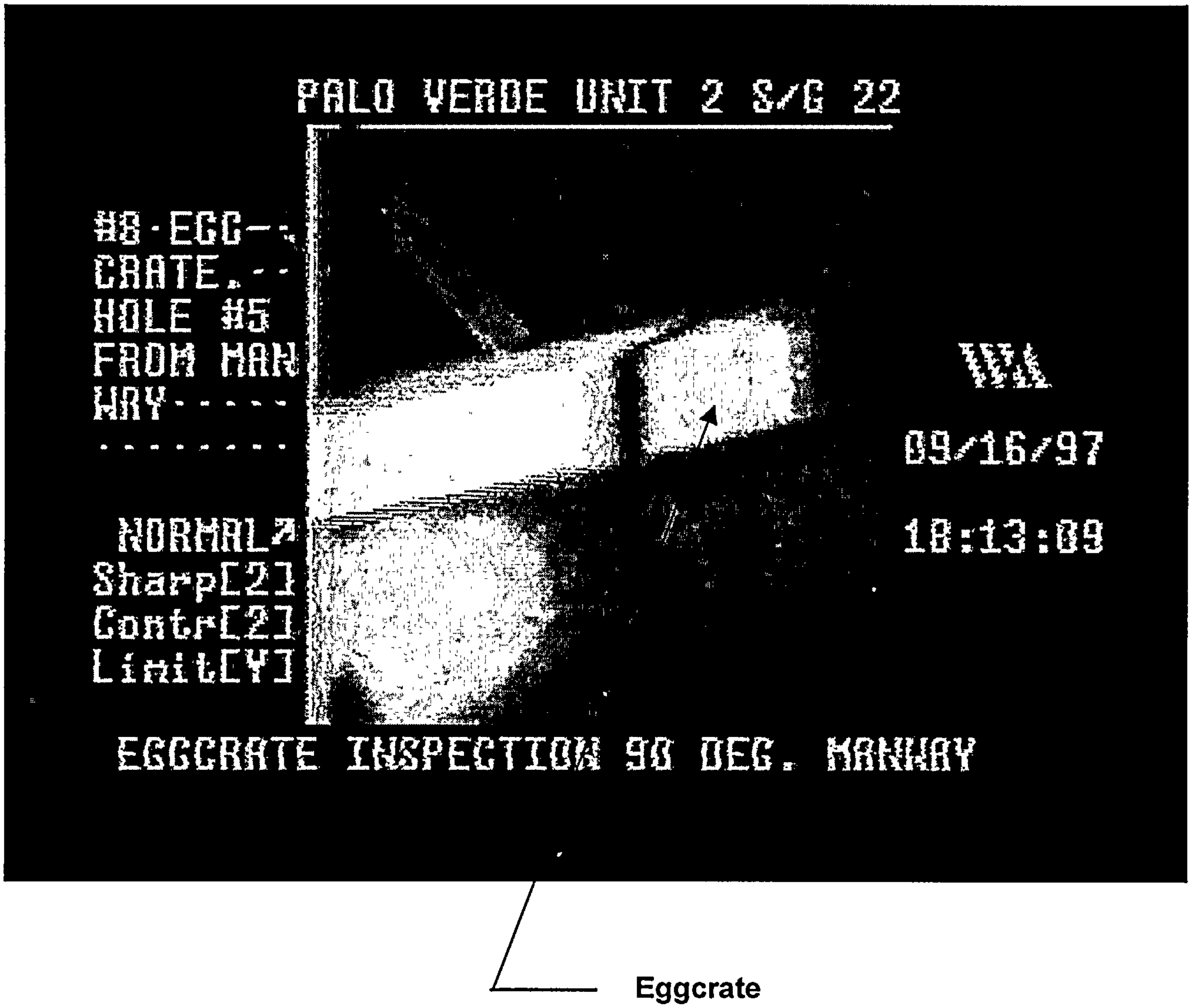


Figure 8

Unit 2 SG 22 # 7 (07H) EGGCRATE

Tack Weld

Eggcrate

Figure 9

ATTACHMENT 2

List of References

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17. APS Letter, *Response to Generic Letter 97-05: "Steam Generator Inspection Techniques"*, 102-04094-JML/SAB/RMW dated March 13, 1998.