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Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

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U. S. Nuclear Regulatory Commission
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Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

90-Day Response to Request for Additional Information RE: Steam Generator Tube Integrity Assessment from the 2003 Refueling Outage (TAC NO. MC2747)

By letter dated April 13, 2004, as supplemented on June 28, 2004, Nuclear Management Company, LLC (NMC) submitted the steam generator tube integrity assessment report from the 2003 refueling outage. On September 2, 2004, the Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) concerning the steam generator report. NMC is providing the attached RAI response. Enclosure 1 contains the NMC responses to the RAI questions for the Palisades Nuclear Plant.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

for

Daniel J. Malone
Site Vice President, Palisades Nuclear Plant
Nuclear Management Company, LLC

Enclosure (1)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

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ENCLOSURE 1
90-DAY RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RE: STEAM GENERATOR TUBE INTEGRITY ASSESSMENT FROM THE 2003 REFUELING OUTAGE

Nuclear Regulatory Commission (NRC) Requested Information

By letter dated April 13, 2004 (ML041100667), supplemented by letter dated June 28, 2004 (ML041890415), Nuclear Management Company, LLC, the licensee for Palisades, submitted a report that summarizes the steam generator (SG) tube integrity assessment performed during the 2003 refueling outage. In reviewing these two letters and their attachments, the Nuclear Regulatory Commission (NRC) staff determined that the responses to the following questions are required to complete the evaluation.

NRC Requested Information

1. *Provide a general description of the replacement SGs, include in this general description the total number of tubes, tube diameter, tube wall thickness, tube pitch, tube support thickness and any other noteworthy design characteristics from a SG tube integrity standpoint (e.g., full length stress relieved of the row 1 through 10 tubes). In addition, provide sketches of the steam generators that depict the tube support naming conventions and tubesheet maps that depict the rows and columns of the tubes.*

Nuclear Management Company (NMC), LLC Response

1. The steam generators (SG)s that are currently in use at the Palisades Nuclear Plant are replacement Combustion Engineering (CE) SGs, Model 2530. The replacement SGs were installed in the fall of 1990. The tube material is mill annealed Alloy 600 with a 0.75-inch outside diameter and a 0.042-inch tube wall thickness. Each SG has 8219 tubes. The tubes were expanded through the full depth of the tubesheet using an explosive process. The tube bundle is supported by stainless steel eggcrate lattice type supports comprised of horizontal eggcrate supports, vertical straps and diagonal straps. The thickness for all supports and straps is 0.090 inches. The SG tube pitch is 1.0-inch. Tube rows 1-18 are U-bends and rows 19-165 are square bends. The SG tubing has not been thermally treated.

   Attachment 1 provides sketches of the steam generators that depict the tube support naming conventions and the tubesheet maps that depict the rows and columns of the tubes.

NRC Requested Information

2. *On page 7 of your in-service inspection report, you indicated that 100 percent of rows 1, 2 and 3 U-bends were inspected in SG E-50A. You also indicated that only 25 percent of rows 1 and 2 U-bends were inspected in SG E-50B and that no expansion of this sample was required. Given that you identified primary water stress corrosion cracking (PWSCC) in the U-bend region of a row 2 tube in SG E-50A and the potential for PWSCC to develop in the U-bend of other low row tubes, discuss the technical basis for not expanding the U-bend inspection to include 100 percent of the tubes in rows 1, 2 and 3 in SG E-50B. Discuss how this was accounted for in your operational assessment.*

NMC Response

2. In the 2003 refueling outage, SG E-50A U-bends indicated an active damage mechanism for PWSCC, and SG E-50B U-bends indicated only a potential damage mechanism for PWSCC. PWSCC has not been identified in the SG E-50B U-bends.

   Twenty-five percent of SG E-50B, rows 1 and 2 U-bends, were inspected with +Point™ during the 2003 refueling outage, with no degradation identified. Electrical Power Research Institute (EPRI) Pressurized Water Reactor (PWR) SG Examination Guidelines, Revision 5, dated September 1997, recommend a minimum of a 20% inspection sample for a potential damage mechanism. Inspection expansion into SG E-50B was not required per the EPRI PWR SG Examination Guidelines because no PWSCC indications were identified. Therefore, PWSCC remained a potential damage mechanism in SG E-50B.

   In the recently completed 2004 refueling outage, NMC inspected 100% of SG E-50A rows 1, 2 and 3 U-bends, and conservatively, 100% of SG E-50B rows 1 and 2 U-bends. The 2004 SG inspection results found no indications in the U-bends of either SG. These results now indicate only a potential damage mechanism for PWSCC exists in both SGs. This is preliminary data from the 2004 refueling outage. This data will be finalized and reported to the NRC, as required, within the next 12 months.

NRC Requested Information

3. *On page 8 of your in-service inspection report, you indicated that one tube (R135C100) was plugged due to a restriction to the passage of a +Point™ rotating pancake coil (RPC). Describe the nature and location of the restriction. Include in your response a discussion of the extent of the restriction and whether or not it was service induced. What was the largest size probe to be passed through the tube during this outage and previous outages? If the restriction was service induced, discuss the basis for not expanding the inspection to include 100 percent of the tubes.*

NMC Response

3. The restricted tube was reported in the ding/dent greater than 5-volt inspection program. It had a 24.61-volt ding/dent and was reported at vertical strap VS4. The restriction was noted in the smaller 0.580 inch +Point™ probe, but a tube end to tube end inspection was completed with the 0.610 inch bobbin probe. This tube had been successfully inspected, tube end to tube end, with the larger 0.610 inch bobbin probe in previous refueling outage SG inspections (the 1995, 1996, 1998, 1999, 2001, and 2003 inspections).

   The restricted designation can include a condition where the probe cannot pass. It can also include a condition where the geometry associated with the dent/ding permits probe passage, however, the data quality may be insufficient to permit analysis. The restriction was not service induced. This tube was preventatively plugged.

NRC Requested Information

4. *On page 9 of your in-service inspection report, you indicated that after tube plugging, there were 7 tubes non-expanded in SG E- 50A and 1 in SG E- 50B that remained in service. Then, on page 32, you stated that the non-expanded tube population is 6 tubes in SG E- 50A and 1 in SG E- 50B. On page 33 you indicated that only 7 tubes remained in service with no expansion in the tubesheet region. Clarify how many non-expanded tubes remain in service in each SG.*

NMC Response

4. Page 9 of the SG Tube Integrity Assessment 2003 Refueling Outage is correct. Pages 32 and 33 have typographical errors. There were seven non-expanded tubes in SG E-50A, and one non-expanded tube in SG E-50B, which were left in service for operational cycle 17. The seven non-expanded tubes in SG E-50A, and one non-expanded tube in SG E-50B, were hard rolled and preventatively repaired by tube plugging in the 2004 refueling outage. There are currently no non-expanded tubes left inservice.

NRC Requested Information

5. *On pages 15, 16, and 18 of your in-service inspection report, you indicated that 25 tubes in SG E-50A and 21 tubes in SG E-50B were plugged. You then stated that after 8 cycles of operation, 57 additional tubes in SG E-50A and 51 in SG E-50B have been plugged, which brings the total number of tubes plugged to 365 and 360, respectively. Please clarify how many tubes were plugged prior to placing the SGs in service and clarify how many tubes were plugged prior to the 2003 refueling outage.*

NMC Response

5. Prior to the installation of the SGs, Combustion Engineering advised Consumers Energy that the area around the center stay cylinder region was potentially

susceptible to fretting wear at the bat wing locations. As a result, 308 tubes in Steam Generator E-50A, and 309 tubes in Steam Generator E-50B, were plugged as a preventative measure.

Prior to the 2003 refueling outage, 32 additional tubes in SG E-50A had been plugged, for a total of 340 tubes plugged, and 30 additional tubes in SG E-50B had been plugged, for a total of 339 tubes plugged. Therefore, SG E-50A has 7879 active tubes with 4.14% of its tubes plugged and SG E-50B has 7880 active tubes with 4.12% of its tubes plugged.

NRC Requested Information

6. *With respect to the eggcrate wear indications, address the following: On page 26 of your in-service inspection report, you indicated that 100 tubes in SG E-50A and 68 tubes in SG E-50B were reported as having a distorted support indication (DSI) from bobbin, were +Point™ RPC tested and reported as VOL. On the same page, you stated that the remaining DSI reported by bobbin were at supports. These included 10 DSI in SG E-50A and 45 DSI in SG E-50B, which were +Point™ RPC tested and no degradation was found. Please clarify this discussion. Is it implying that 110 DSI's were reported in SG E-50A and of these 110, 100 were attributed to wear and 10 had no degradation? In addition, please clarify an apparent discrepancy in Table 16. Table 16 indicates that 114 tubes in SG E-50A had wear indications while the text indicates 100. Provide the technical basis for determining when wear at eggcrates should be RPC inspected. Specifically, address in your response whether there is any data that indicates any appreciable change in the bobbin coil signal will result from an axial or circumferential flaw that is present in a wear scar.*

NMC Response

6. In the 2003 SG Tube Integrity Report, 114 tubes were reported with distorted support indications (DSI) in SG E-50A. The 100 DSI reported in SG E-50A is a typographical error. There are 133 volumetric (VOL) calls from Bobbin testing at hot and cold leg eggcrates. Of these, 19 VOL are a second or third call on the same tube, either at the same elevation or a different elevation, but still valid calls. The final count of tubes with wear at hot and cold leg eggcrates was 114 VOL. This is the same reported total of DSI as in Table 16 of the 2003 SG Tube Integrity Report.

Palisades' +Point™ RPC tests all DSI from wear. The axial or circumferential indication determination will be from +Point™ RPC testing. All eggcrate support plate with new DSI indications or with DSI indications that were +Point™ RPC tested in previous outages, which exhibited any change or growth from bobbin history, were RPC tested during the 2003 refueling outage and confirmed that wear is the only damage mechanism occurring.

Palisades eggcrate structures are stainless steel, thus classical denting cannot occur at eggcrates at Palisades, as the eggcrate cannot experience corrosion attack with corrosion product buildup in the crevice. If classical denting cannot

occur, there is no mechanism that results in large axial tensile loads during operation. There is very little potential for circumferential outside diameter stress corrosion cracking (ODSCC) development at eggcrates at Palisades.

NRC Requested Information

7. *On page 39 of your in-service inspection report, you indicated that a sodium intrusion event has introduced sodium into the crevices. Discuss this event in detail or if already provided to NRC, please provide references to documents where this is discussed.*

NMC Response

7. This was discussed in the 12-month SG submittal to the NRC, dated April 19, 2002, "SG Tube Integrity Assessment 2001 Refueling Outage," Section 5.12, "Secondary Side Water Chemistry Sodium Intrusion."

NRC Requested Information

8. *On page 8 of your in-service inspection report, you indicated that stay rod locations are areas of high residual stress and susceptible to stress corrosion cracking. Discuss why this area is considered to be susceptible to high residual stresses and what degradation mechanisms are more prone to appear.*

NMC Response

8. This statement is a generic concern for CE SGs with carbon steel support plates that will dent as the carbon steel corrodes. A previous evaluation performed by CE concluded that, for dented eggcrate conditions, tubes around stay rod locations could experience elevated stress conditions, creating a potential for stress corrosion cracking development.

   The stainless steel eggcrate locations at Palisades are not dented, therefore, this potential does not exist for Palisades. Past inspections have conservatively included a +Point™ examination at the top of tubesheet for tubes surrounding stay rods if less than 100% of the hot leg tubes were inspected. Palisades has inspected 100% of the active hot leg tubes in the 2003 and 2004 refueling outage SG inspections. No indications have been identified at eggcrate locations associated with stay rods.

NRC Requested Information

9. *Regarding your disposition of the freespan differential (FSD) indications: Discuss what constitutes a "significant change" in the signals when compared to baseline and your technical basis for this criteria (e. g., Is it based on test repeatability?). Approximately 260 tube locations were inspected with an RPC probe because of a new FSD or because an existing FSD changed. Clarify whether the indications that required RPC testing were considered new or changed. If new indications, discuss the cause. If changed indications, discuss the reason for the change. Clarify the number of tubes with FSD indications that*

required RPC testing. Table 19 implies that only 1020 of the 1619 FSDs were dispositioned based on history review.

NMC Response

9. The Palisades criteria of "significant change" in the signals, when compared to baseline, were set conservatively to 15 degrees phase angle, or 0.5 volts, during the 2003 refueling outage.

   Palisades uses the 1990 baseline data for the freespan historical comparisons. Today's eddy current testers (TC6700) are driven at a higher voltage and have better signal to noise characteristics than the testers used in 1990 (MIZ-18). "New" FSDs reported today that are very small in voltage may have been present in 1990, but were not detected by the less sensitive equipment at that time. Also, phase angles of the signals will show differences between tests simply from the way the coil rides the tube and the use of different test variables (testers, cabling, probes) from year to year. The signals were +Point™ RPC tested for a change and some discrepancies were identified. This is due to the normal variance in signals between test runs from different equipment and different probes in different outages.

   In order to address current regulatory concerns, an extensive historic review was performed during the inspection. Freespan indications reported by bobbin were reviewed in baseline history. If the indication showed no significant change from 1990 baseline to the present, it was reported as FSD with an historic review comment to designate that it was reviewed. Those indications, which either could not be detected or showed a change from baseline history, were reported as a non-quantifiable indication (NQI), or an "I" code, and +Point™ RPC tested. A total of 69 indications in SG E-50A, and 194 indications in SG E-50B, were reported as an "I" code in the freespan. During the 2003 refueling outage, a +Point™ RPC identified no degradation in these 263 tubes. NQI calls were then changed to non-quantifiable signals (NQS) for future tracking. The result of the freespan bobbin indications resolved to non-repairable status by history review was shown in Table 19. In SG E-50A, 339 FSD's were identified over 208 tubes, and in SG E-50B, 1280 FSD's were identified over 812 tubes. Therefore, there were 1619 FSD indications dispostioned in 1020 tubes in the 2003 refueling outage.

NRC Requested Information

10. You indicated that no changes were noted in the bobbin coil examinations of dings. You further indicated that a bobbin coil review was performed for all free span dings greater than or equal to 2 volts. Clarify how many dings are present in your steam generators (e. g., " x" dings greater than 2 volts, "y" dings greater than 5 volts). Clarify that 95 percent of all dings (greater than 2 volts) are located at vertical straps. Clarify whether circumferential cracking has been observed at dings at Palisades or at other similarly designed and operated units. If it has been observed, discuss the basis for not expanding the scope of inspection to 100 percent. Based on the material provided, the staff inferred that for dings

whose voltages are between 2 and 5 volts, the 2003 bobbin coil data was compared to baseline data to determine if any change occurred. If this is the case, discuss the technical basis for the approach including whether the bobbin coil signal from a ding would change significantly if a flaw (axial or circumferential) were to initiate at this location. The staff notes that many plants with mill annealed Alloy 600 tubes inspect 100 percent of all dents/dings that have bobbin voltages greater than 5 volts.

NMC Response

10. An EPRI PWR SG Examination Guideline sampling and expansion program was used for dings/dents, ≥2 volts to ≤ 5 volts, in the bobbin screening program. For dings/dents >5 volts, the +Point™ probe was used in the 2003 refueling outage SG inspection. No ODSCC indications in dings/dents were identified in the 2003 refueling outage. The ding/dent eddy current inspection techniques for the bobbin screening and +Point™ RPC testing are qualified techniques. The qualified inspection probes and techniques that were used during the 2003 refueling outage were most recently described in NMC response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," submitted to the NRC by letter dated October 29, 2004.

In the 2004 refueling outage SG inspection, there was one distorted ding indication (DNI) screened from the bobbin inspection program in each SG. Both DNI were +Point™ tested and determined to be single axial ODSCC indications. Tube plugging repaired both single axial ODSCC indications. The single axial ODSCC indications associated with dings expanded the bobbin SG inspection program from 50% to 100%. The ding >5 volt program was also expanded from 25% to 100%.

The table below shows the SG ding/dent inspection total through the 2004 refueling outage. A ding is reported in freespan and dent is reported at structures.

Steam Generator	**Dings/Dents ≥2 Volts and ≤ 5 Volts**	**Dings/Dents ≥2 Volts and ≤ 5 Volts at VS4**	**Dings/Dents > 5 Volts**	**Dings/Dents > 5 Volts at VS4**
E-50A	749	327	603	438
E-50B	540	216	192	75

Approximately 95% of the dings occur at vertical straps between diagonal bar hot and diagonal bar cold at the center of the horizontal tube length in the upper bundle area with the majority at vertical strap VS4.

Circumferential ODSCC at dings has not been reported at Palisades. NMC has reviewed the EPRI SG database and did not find any reports, from similarly designed CE SGs, of circumferential ODSCC at dings.

NRC Requested Information

11. In your assessment of tube wear, discuss how you accounted for degradation in the tubes you did not inspect. In addition, clarify the discussion regarding the 67 percent through wall wear indication detected in 1999 (p. 26). Was the wear associated with this tube caused by some local support condition such that it necessitated inspection of surrounding tubes?

NMC Response

11. If a tube had observed wear from a previous inspection, it was inspected in the 2003 refueling outage. The tubes with greater than 20% wear have been inspected during each refueling outage. All tubes are tested with a bobbin within a 60 effective full power month (EFPM) rolling window as required by the EPRI PWR SG Examination Guidelines. Trended growth rates do not suggest that wear exceeding the structural limit will be observed at the next inspection of that tube. If a tube has not experienced wear, the growth rates, should indications develop, will be small.

    The 67% through-wall indication reported in the 1999 refueling outage SG inspection was associated with a tie bar that encircles the tube bundle in the vicinity of the diagonal bars. All surrounding tubes were inspected to ensure that the bar did not interact with other tubes. No additional indications have been identified.

NRC Requested Information

12. On page 28 of your in-service inspection report, you indicated that an axial U-bend outside diameter stress corrosion cracking (ODSCC) indication was screened. Was an axial ODSCC indication detected in the U-bend or is this a typographical error?

NMC Response

12. This is a typographical error. All indications to date found in the Palisades SG U-bends are PWSCC.

NRC Requested Information

13. Discuss whether any primary to secondary leakage has been observed during the current cycle.

NMC Response

13. The average leak rate was 0.0001 gpm during operational cycle 17, per Palisades steam generator chemistry monitoring program, for steam generator primary to secondary leak rate. This is considered just above background for detection purposes.

NRC Requested Information

14. *For all tubes with possible loose parts indications, discuss whether a visual inspection was performed at these locations. Were any loose parts left in the SG? If so, discuss whether a tube integrity evaluation was performed.*

NMC Response

14. A visual inspection was completed for periphery hot leg and cold leg top of tubesheet tubes using a manual video probe. Loose parts were removed using foreign object search and retrieval (FOSAR). In-bundle inspections were not performed on possible loose part (PLP) indications. An 100% +Point^{TM} probe top of hot leg tubesheet inspection was completed. All tubes with loose part wear were removed from service and repaired by tube plugging. All tubes with loose part wear had an in-situ and stabilizer evaluation completed. All tubes with loose part wear were evaluated for tube integrity and use of a stainless steel cable stabilizer prior to tube plugging. The 2003 Steam Generator Tube Integrity Assessment has identified all tubes with loose part wear that were stabilized and repaired.

    Steam Generator eddy current analysis in the top of tubesheet program has identified, and continues to monitor, PLP indications on the top of the tubesheet each refueling outage. The PLP indications are small ferromagnetic material, like sludge or magnetite, or demister wire (304 stainless steel), which were detected in +Point^{TM} RPC eddy current testing and are fixed in the permanent sludge pile. The magnetite is from iron transport and the loose metallic material in the secondary side is foreign material with the majority coming from the moisture separator reheater demister pads. The loose demister wire left in place has had an engineering evaluation completed as a result of a condition report. This evaluation has identified that the demister wire found in the secondary side of Palisades SGs will not have an adverse impact on SG tubes or overall plant operation (no tube leaks).

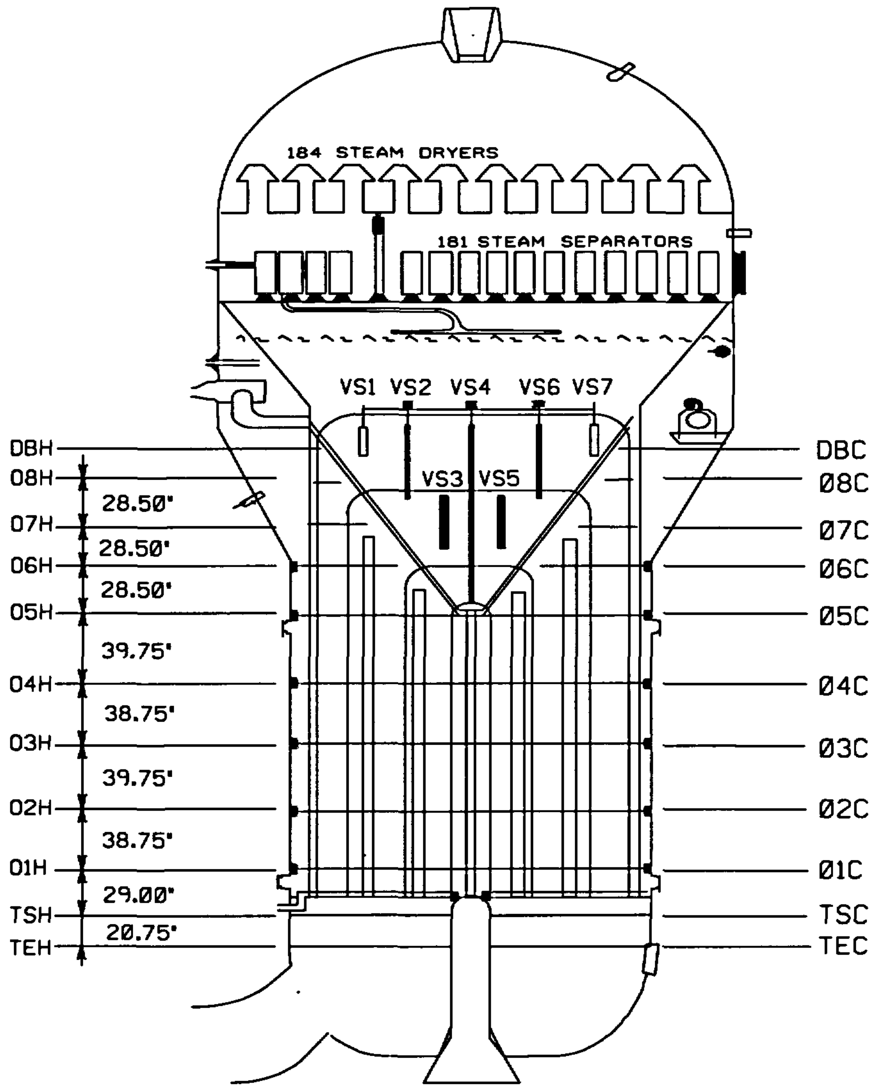

Figure 1: Steam Generators With Naming Conventions

Name	Description		Name	Description
TEH	Tube End - Hot Side		VS4	Fourth Vertical Strap
TSH	Top of Tubesheet - Hot Side		VS5	Fifth Vertical Strap
01H	First Eggcrate - Hot Side		VS6	Sixth Vertical Strap
02H	Second Eggcrate - Hot Side		VS7	Seventh Vertical Strap
03H	Third Eggcrate - Hot Side		DBC	Diagonal Strap - Cold Side
04H	Fourth Eggcrate - Hot Side		08C	Eighth Eggcrate - Cold Side
05H	Fifth Eggcrate - Hot Side		07C	Seventh Eggcrate - Cold Side
06H	Sixth Eggcrate - Hot Side		06C	Sixth Eggcrate - Cold Side
07H	Seventh Eggcrate - Hot Side		05C	Fifth Eggcrate - Cold Side
08H	Eighth Eggcrate - Hot Side		04C	Fourth Eggcrate - Cold Side
DBH	Diagonal Strap - Hot Side		03C	Third Eggcrate - Cold Side
VS1	First Vertical Strap		02C	Second Eggcrate - Cold Side
VS2	Second Vertical Strap		01C	First Eggcrate - Cold Side
VS3	Third Vertical Strap		TSC	Top of Tubesheet - Cold Side
			TEC	Tube End - Cold Side

Table 1: Tube Support Naming Conventions

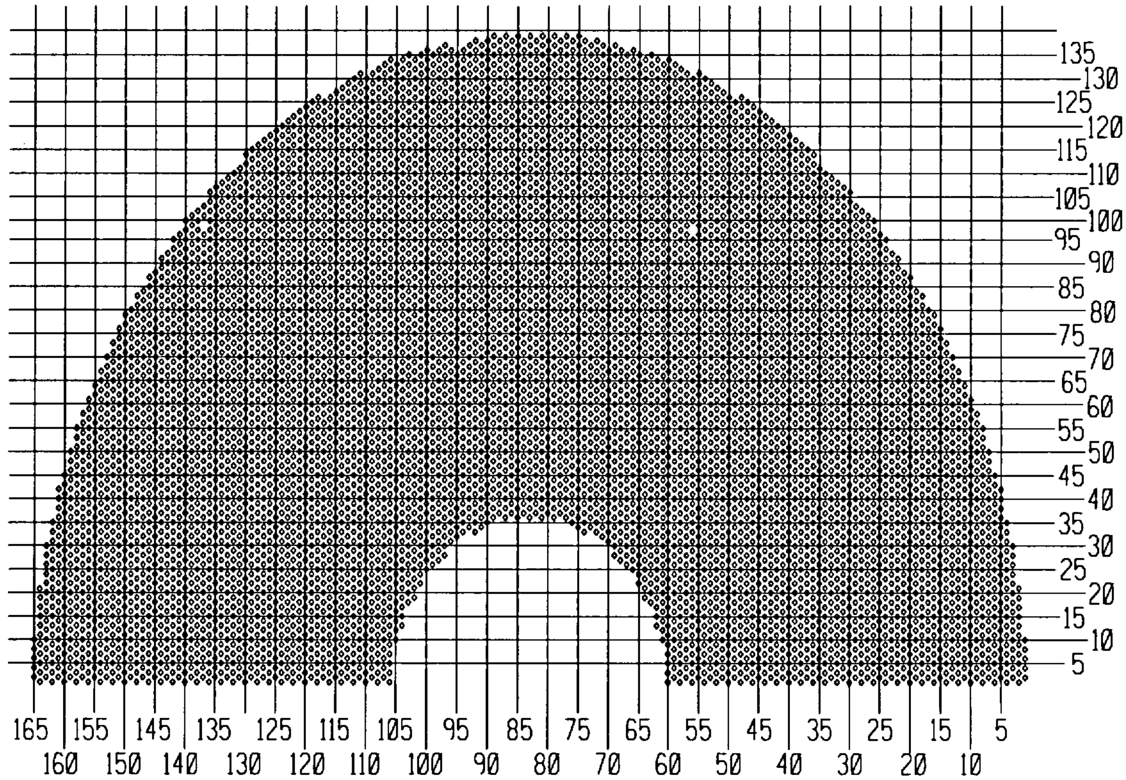

Figure 2: Steam Generator Tube Sheet Map