UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

March 10, 2005

EA-05-008

South Carolina Electric & Gas Company
ATTN: Mr. Jeffrey B. Archie
Vice President, Nuclear Operations
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A GREEN FINDING (NRC INSPECTION REPORT 05000395/2005007, VIRGIL C. SUMMER NUCLEAR STATION)

Dear Mr. Archie:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC's) final significance determination for an issue at South Carolina Electric and Gas Company's (SCE&G) V. C. Summer Nuclear Station. The issue involved inadequacies in your corrective actions associated with a deficiency in the design of the emergency feedwater (EFW) system flow control valves.

The finding resulted from an assessment of Unresolved Item (URI) 0500395/2004009-01 of NRC Inspection Report 05000395/2004009, issued on December 22, 2004. The finding was reviewed further in NRC Inspection Report 05000395/2005006, dated January 14, 2005, and was assessed under the significance determination process as a preliminary White finding (i.e., an issue of low to moderate safety significance, which may require additional NRC inspection). The cover letter to the inspection report informed SCE&G of the NRC's preliminary conclusion, provided SCE&G an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary results for this finding.

At SCE&G's request, a regulatory conference was conducted with you and members of your staff on February 17, 2005, to discuss SCE&G's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference, and copies of the material presented by the NRC and your staff at the conference. In support of the conference, SCE&G also provided a written response dated February 9, 2005.

During the conference and as discussed in your February 9, 2005, response, SCE&G's presentation focused on the likelihood of the unavailability of the Condensate Storage Tank (CST) due to an F2 tornado, and the probability of a random CST failure. SCE&G's analysis concluded that an F2 tornado would not render the CST unavailable, and based on plant-specific information, a random tank failure event is not risk significant. As such, SCE&G concluded that the change in Core Damage Frequency (CDF) supports a Green finding. Other factors presented by SCE&G at the conference that contributed to its view that the finding is not risk significant included: its differing view that the reactor will not always trip upon a loss of the CST (as postulated by the NRC); operator training to only introduce service water (SW) into the steam generators as a last resort; very little tubercle material will be released by normal EFW flow rates; and any dislodged material will be pulverized in the EFW pumps. Consequently, the EFW flow control valves will not become plugged as assumed by the NRC.

In addition, by letter dated February 23, 2005, and in combination with the information provided in its February 9, 2005, letter and the material presented at the conference, SCE&G provided its conclusions that an adequate basis exists to allow for the NRC to reconsider the issuance of both the 10 CFR 50, Appendix B, Criteria III and XVI violations. In summary, SCE&G stated that the EFW flow control valve specification, selection, and procurement was originally prepared in accordance with prudent and generally accepted design practices, based on the known conditions of the process fluid, including SCE&G operating experience. Therefore, the subject design was developed in accordance and consistent with regulatory requirements. In addition, SCE&G concluded that corrective actions that have been considered and implemented have been appropriate and timely. SCE&G also stated that additional corrective actions currently being pursued to address industry and NRC staff concerns are considered to be enhancements and not the result of ineffective corrective actions.

After considering the information developed during the inspection, the information in SCE&G's written responses, and the information presented at the conference, the NRC has concluded that the final significance of the finding is appropriately characterized as Green, in the mitigating system cornerstone. In this case, the NRC acknowledges that its preliminary estimate of the change in CDF was White. The NRC considered a slight decrease in the random tank failure probability from that which was assumed in the NRC's preliminary estimate. In addition, SCE&G's view that the reactor would not always trip upon a loss of the CST appears to be plausible in this case. These factors and other less quantifiable factors discussed by SCE&G would result in a decrease in the change in CDF to a degree slightly less than the Green/White threshold. As such, the NRC has concluded that the significance of this finding should be characterized as Green.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified Green finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

Notwithstanding the information provided by SCE&G at the conference and in its written responses, the NRC determined that two violations occurred, involving the requirements of 10 CFR 50, Appendix B, Criterion III and Criterion XVI. The NRC concluded that, at the time of the inspection in November 2004, SCE&G had failed to adequately select and review for suitability of application of materials, parts, equipment and processes that are essential to the safety related functions of the EFW system. In this case, the safety related function of the EFW system, as described in the Updated Final Safety Analysis Report (UFSAR), Section 10.4.9, is for the plant to operate indefinitely, if required, without normal feedwater, and for the EFW system to take suction from the SW system for an indefinite period of time. The design of the EFW system is such that, under certain conditions, the EFW flow control valves could become plugged from tubercles or other debris when aligned to the safety related water supply. The second violation involved SCE&G's failure to correct a condition adverse to quality wherein the EFW flow control valves were not designed to handle relatively unclean SW and, consequently, could become plugged by tubercles or other debris from SW. The corrective action violation occurred during 1986 through 2004.

Because of the very low safety significance of the two violations and because the issues were entered into your corrective action program in CER 0-C-04-3416, the NRC is treating the violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this letter, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC, 20555-0001; and the NRC Resident Inspector at the V. C. Summer Nuclear Plant.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000395/2005007, and the above NCVs are identified as NCV 05000395/2005007-01: EFW Flow Control Valves Are Susceptible to Plugging by Tubercles or Other Debris from Service Water; and NCV 05000395/2005007-02: Inadequate Corrective Actions in Response to Potential EFW Control Valve Plugging. Accordingly, AV 05000395/2005006-01 and -02 are closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

Should you have any questions regarding this letter, please contact me at 404-562-4605.

Sincerely,

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Charles R. Ogle, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No.: 50-395
License No.: NPF-12

Enclosures:
1. List of Attendees
2. Material presented by SCE&G
3. Material presented by NRC

cc w/encls.:
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LIST OF REGULATORY CONFERENCE ATTENDEES

NUCLEAR REGULATORY COMMISSION:
L. Plisco, Deputy Regional Administrator, Region II (RII)
V. McCree, Director, Division of Reactor Projects (DRP), RII
C. Casto, Director, Division of Reactor Safety (DRS), RII
L. Wert, Deputy Director, DRP, RII
C. Ogle, Chief, Engineering Branch 1, DRS, RII
W. Rogers, Senior Reactor Analyst, DRS, RII
K. Landis, Chief, Branch 3, DRP, RII
C. Evans, Enforcement Officer and Regional Counsel, EICS, RII
R. Schin, Senior Reactor Inspector, DRS, RII
J. Moorman, Chief, Operator Licensing Branch, DRS, RII
D. Starkey, Senior Enforcement Specialist, Office of Enforcement

SOUTH CAROLINA ELECTRIC & GAS COMPANY:
J. Archie, Site Vice President
D. Gatlin, Plant Manager
M. Fowlkes, Engineering General Manager
G. Lippard, Operations Manager
B. Whorton, Senior Engineer
E. Rumfelt, PRA Engineer
R. Clary, Licensing Manager
T. Estes, PRA Team Leader
R. Guerra, Control Room Supervisor
T. Poindexter, Legal Representative

SCE&G®
A SCANA COMPANY

NRC Regulatory Conference
VC Summer Nuclear Station

February 17, 2005

Enclosure 2

VC Summer Personnel

Presentation

Jeff Archie – Site Vice President

Dan Gatlin – Plant Manager

Mike Fowlkes – Engineering General Manager

George Lippard – Operations Manager

Bob Whorton – Senior Engineer

Eric Rumfelt – PRA Engineer/Shift Engineer

Support

Ron Clary – Licensing Manager

Tyndall Estes – PRA Team Leader

Rick Guerra – Control Room Supervisor

Tom Poindexter – Morgan, Lewis & Bockius

Agenda

- Opening Remarks - Jeff Archie
- Discussion of Purpose - Dan Gatlin
- EFW Flow Evaluation - Mike Fowlkes
- Tornado Evaluation - Bob Whorton
- Operator Actions - George Lippard
- Risk Assessment - Eric Rumfelt
- Conclusions - Dan Gatlin
- Closing Remarks - Jeff Archie

Purpose

- Provide plant-specific information that better quantifies the actual significance of these postulated events
  - Additional insights concerning behavior of SW pipe corrosion
  - F2 tornado will not fail the CST
  - Actual operator actions reduce risk significantly
  - Initiating event frequency for loss of CST should be reduced
- Plant specific inputs reduce significance to "green"

Purpose

- We are not here to eliminate the need to address the issue.
  - Follow-up inspections and cleaning in RF15 (previously planned).
  - Evaluation of future inspections, improved maintenance, & potential design changes will occur.

VC Summer Engineering Assessment of Service Water Flow Through Emergency Feedwater Flow Control Valves

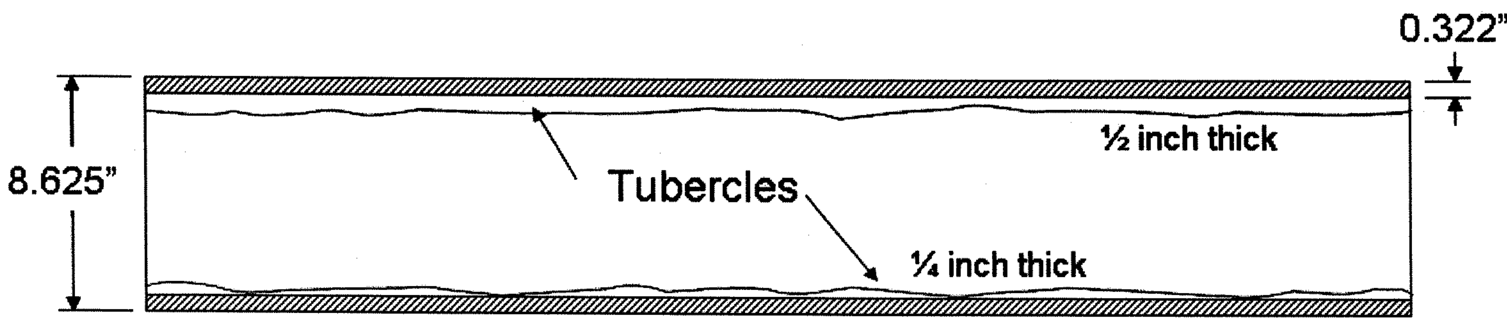
Mike Fowlkes

NRC Assumptions

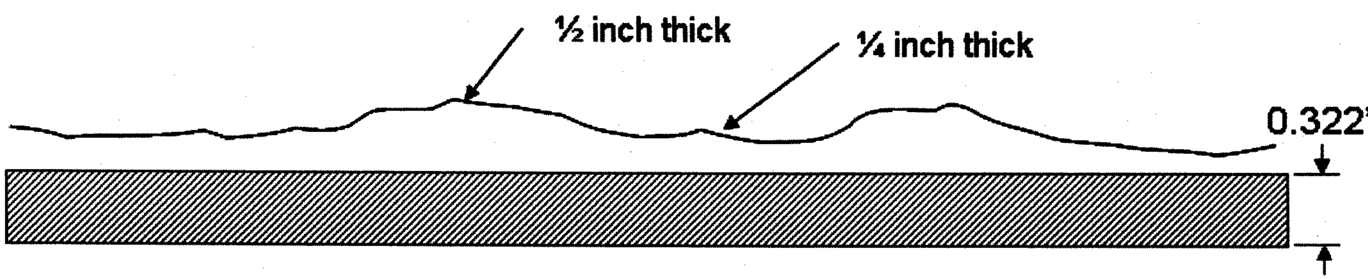
- Tubercles will be dislodged from SW pipe and will plug EFW flow control valves reducing flow below the required minimum
- Samples of material from SW screen are representative of the tubercles in the SW pipe and can not be pulverized when agitated
- A differential pressure of 100 psid across the valve results in 0.2 pounds force on a single plugged hole in the valve cage and is insufficient to force tubercle material through the hole

Limited Tubercle Buildup in SW Cross-Connect to EFW

- Tubercle layer is ¼ to ½ inch thick
- Diameters generally between ½ and ¾ inches.



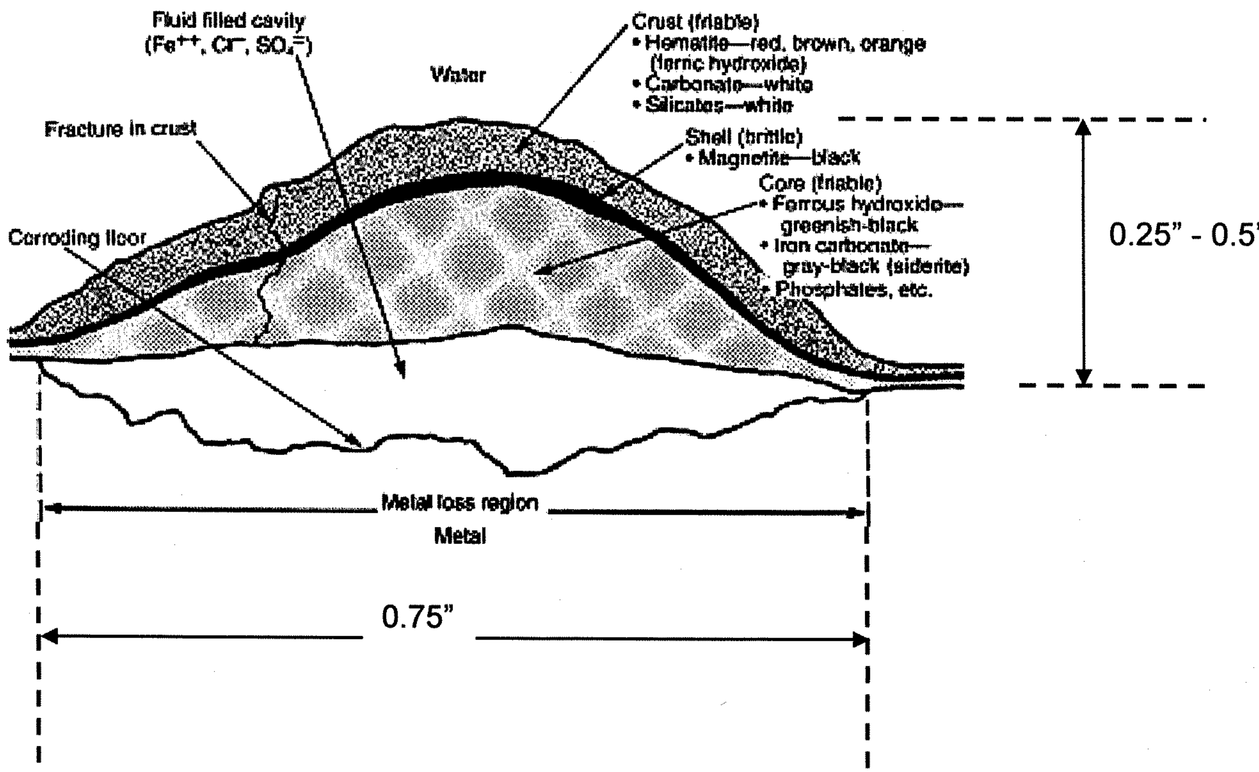
Drawing of 8 inch schedule 40 pipe with ½ inch thick tubercles on top and ¼ inch thick tubercles on the bottom of the pipe



Composition of Tubercle

Tubercles have a friable (easily crumbled or pulverized), hematite outer crust; a brittle magnetite inner shell surrounding friable core material.

This composition differs significantly from sample material removed from the SW screen structure.



V C Summer Experience with Tubercles

- Inspections demonstrate
  - Tip of a probe can break the tubercle surface and release a fine silty material from the outer layer
  - Hydro-lasing activities in RF-14 at 5000 psi would not completely remove tubercle material from the pipe surface
- At maximum expected EFW flow rates
  - Less than 0.25 psi on the surface of a tubercle
  - Few tubercles would be expected to be removed

EFW Pump Effects

- Several forces inside the pump would act to pulverize tubercle material
  - Turbulent Flow
  - Impact Forces
  - Pressure Variations
- Based on industry testing, the EFW pump will act to pulverize the friable material if released to the flow stream

Reaction of Material Inside the Flow Control Valve

- The friable material will be further broken down by turbulent, high velocity flow inside the flow control valve
- The force applied to the material in a plugged hole would be 20 times that assumed by the NRC calculation at the minimum Tech Spec flow
- Flow control valves can maintain design basis functionality with up to 68% of the flow area plugged

VC Summer Conclusions

- Tubercles adhere tenaciously to pipe walls, very little material will be released by normal EFW flow rates
- Any dislodged material is easily pulverized in pumps and passes through valves
- Samples from SW screen were not friable and are not typical of tubercles in the SW piping
- Flow control valves can maintain design basis flow with plugging up to 68%

VIRGIL C. SUMMER TORNADO EVALUATIONS
- CONDENSATE STORAGE TANK

Bob Whorton

NRC Assumption

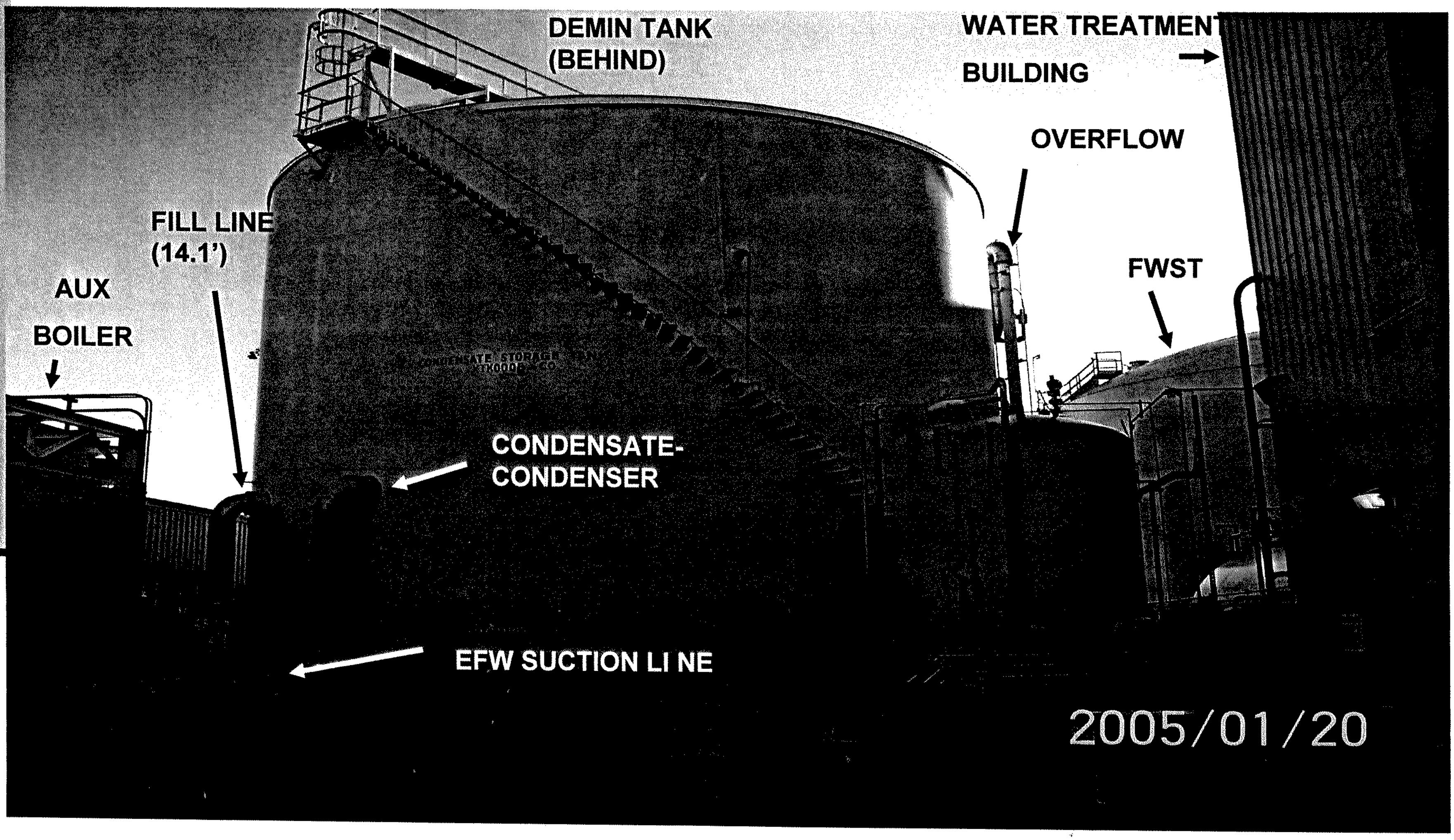
- The CST function is lost at the medium wind speed/intensity of an F2 tornado

VC Summer Evaluations

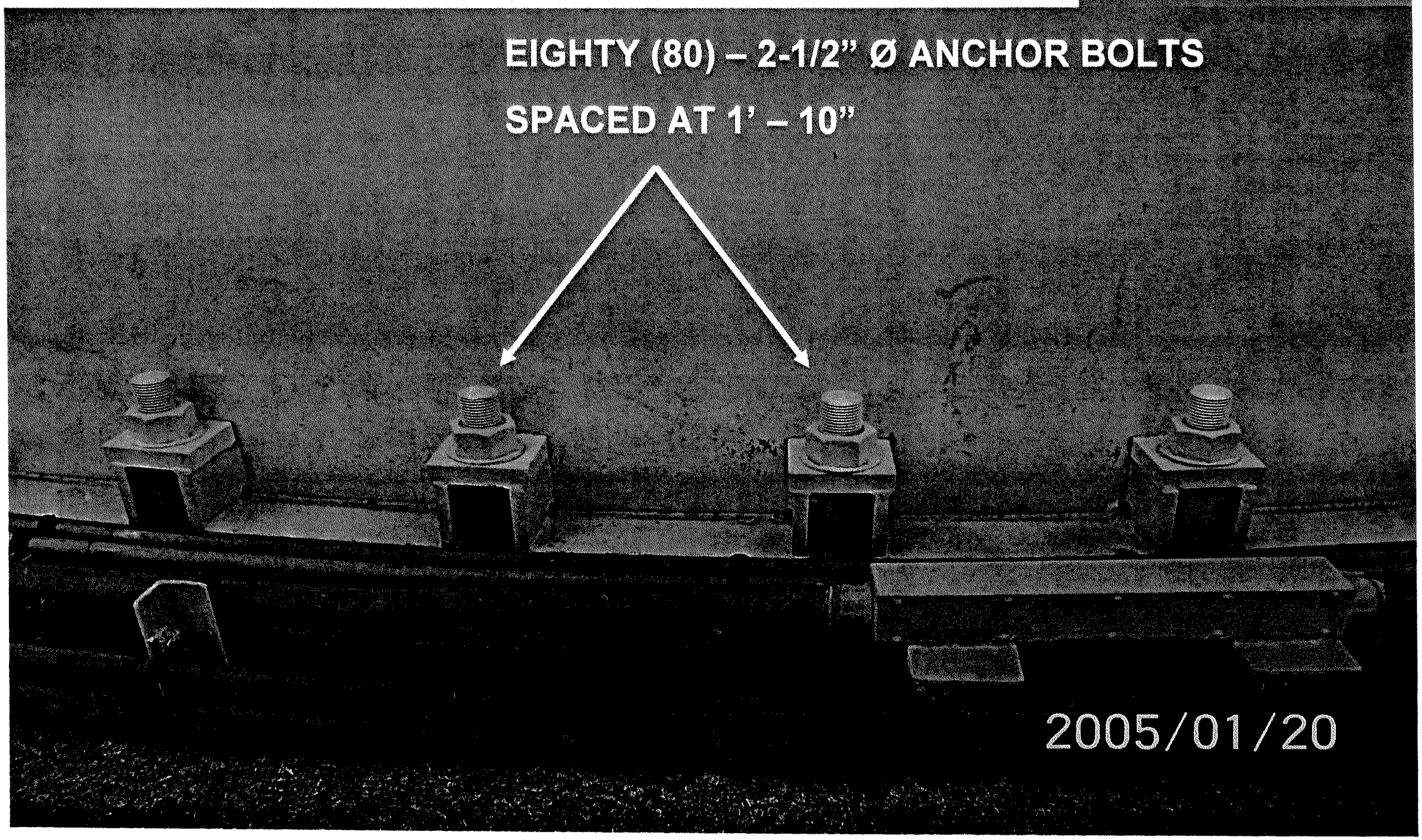
- <u>DETERMINISTIC EVALUATION</u>
  F2 Tornado Wind Loads and Missiles

- <u>PROBABILISTIC EVALUATION</u>
  Risk of Loss of Function of the CST
  (All F-Scale Tornados)

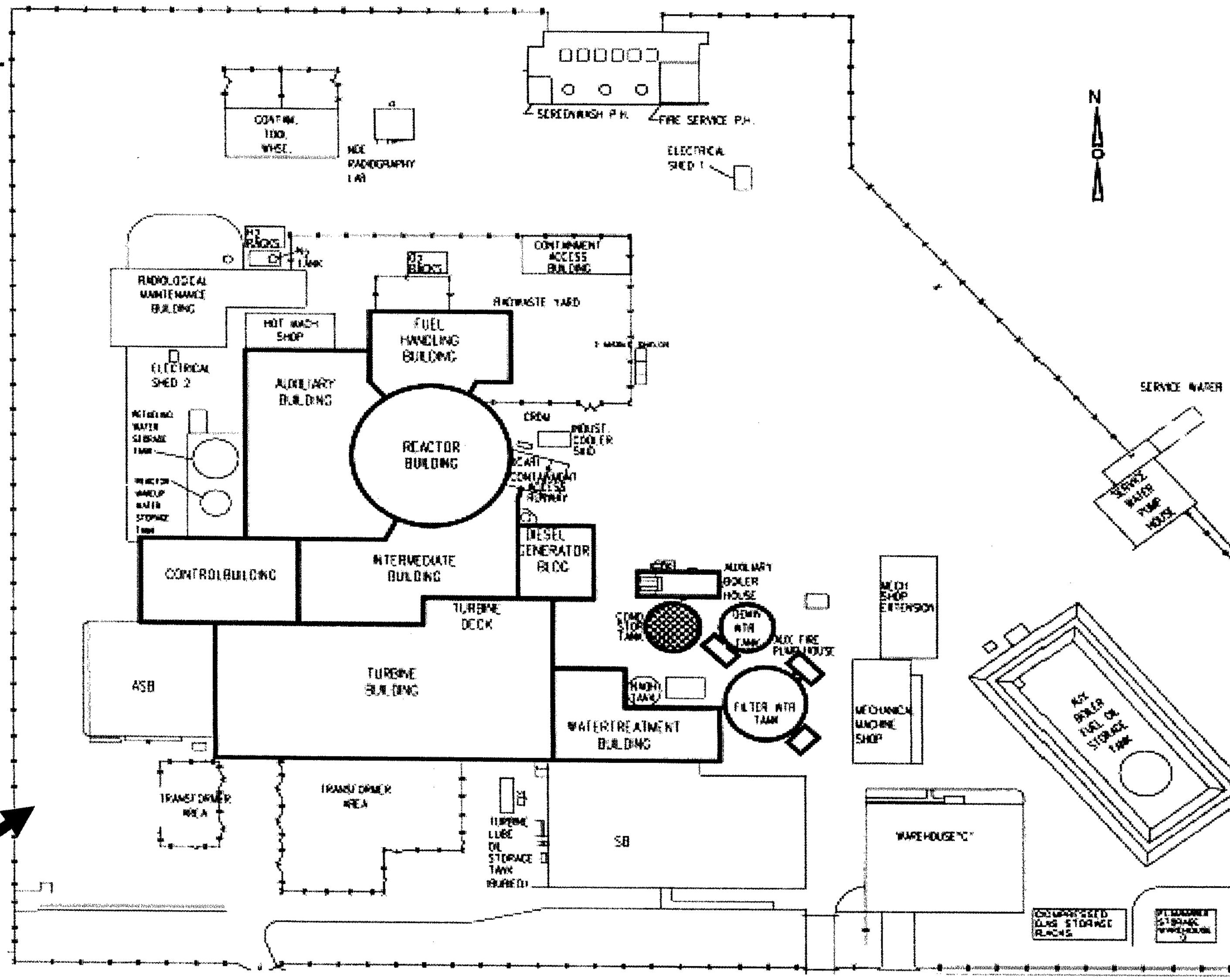
Condensate Storage Tank
(View - East from Turbine Building)



CST Anchor Bolts



CST Layout Drawing



PREVAILING WEATHER DIRECTION

CST Design Evaluation

Seismic Design – Tank Shell Stress	19.6 psi (Top) 41.1 psi (Bottom)
F2 (157 mph) Max. Wind Load	0.44 psi
F2 (Δ P) Load	0.52 psi

<u>Conclusion</u>: CST Seismic Design Significantly Exceeds F2 Tornado Wind Loads

F2 Tornado Missile Perforation Evaluation

	THICKNESS	MARGIN %
Missile Perforation	0.16”	---
25% Safety Factor	0.20”	25
CST Shell Plate	0.25” - 0.625”	25 to 212
EFW Pipe Wall	0.365”	82.5

Conclusion: F2 Tornado Missiles Will Not Result in Loss of Function of CST or EFW Suction

Deterministic Evaluation Conclusion

There is <u>No Loss of Function</u> of the CST or EFW Suction Line from the Wind & Missile Effects of an F2 Tornado

Probabilistic Evaluation

1. Used Seabrook PRA Methodology as Approved by NRC (NUREG-0896)
2. Normalized VCS Tornado Point Strike Probability (PSP) to Seabrook
3. Calculated Baseline MAIP for Specific VCS Targets (CST & EFW Suction Line)

Probabilistic Summary (con't)

4. Applied VCS Site Adjustments Based on Specific Plant and Component Details
5. Calculated Cumulative MAIP for CST and EFW Suction Line: **4.5×10^{-8}**

Conclusion: Potential Loss of Function from All F-Scale Tornados is Not Risk Significant

Operator Actions

Failure of the Condensate Storage Tank

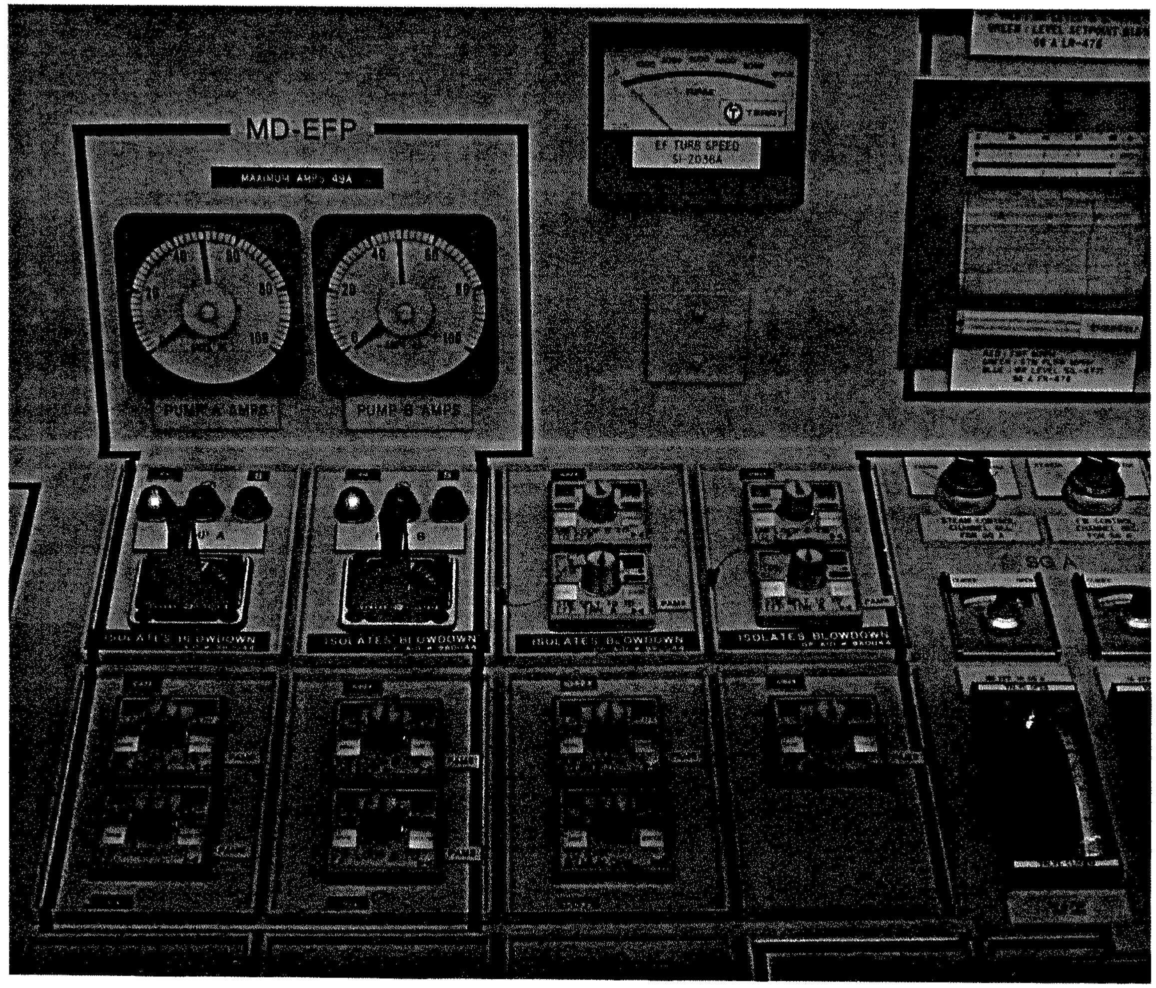
George Lippard

CST/Piping Failure Scenario

- **NRC Assumption**
  - Failure of CST as suction source for EFW system will cause a Rx trip during ensuing shutdown (Probability = 100%)
- **VC Summer Evaluation**
  - Plant Response
  - Mitigation Strategy
  - Operator Response

CST Failure: Plant Response

- SW valves open
- EFW remains in Standby



CST Failure: Mitigation Strategy

- Action interval for inoperable CST is 7 days
  - Emergency repairs would be initiated to return the CST to functional status
  - Clean water sources such as Filtered Water and Demineralized Water possible alternatives
  - Just-in-Time Training of Operator Crews

CST Failure: Operator Response during Shutdown

- No reactor trip would be required or expected upon shutdown. (Probability ≠ 100%)
  - EFW not placed in service until Reactor Power reduced to 1-3%, regardless of suction supply
  - Initial EFW flow creates maximum potential for plugging
  - Main Feedwater will **NOT** be secured until EFW flow is established and stable

CST Failure: Conclusions

- No transient due to initial tank failure
- Mitigation strategies possible with alternate suction sources
- **No reactor trip would be required or expected after a random CST failure.**

Risk Assessment

Eric Rumfelt

NRC Risk Significant Scenarios

- Tornado Event with ΔCDF of 1.4E-6/yr
- Failure of CST or piping segments with ΔCDF of 5.5E-6/yr

Tornado Event

- **NRC Assumption**: F2 tornado will cause Condensate Storage Tank loss of function
- **VCS Evaluation**:
  - No loss of function of the CST or EFW suction line from the wind & missile effects of an F2 tornado.
  - Potential loss of function from all F-Scale tornado wind loads and missiles is not risk significant (4.5E-8 probability).
- **VCS Conclusion**: **Loss of CST function due to tornados is not risk significant**

CST/Piping Failure Scenario

- **NRC Assumptions:**
  - A reactor trip will occur during shutdown following the event. (Probability = 100%)
  - CST failure rate of 1.0E-07/hr
  - EFW piping failure rates based on “Failure” values in source document

Reactor Trip During Shutdown

- **VCS Evaluation of CST Failure Event:**
  - No initial plant transient
  - Technical Specification AOT is seven days
  - CST Repair/Makeup Options
  - Controlled/Planned shutdown
- **VCS Conclusion:**
  - Reactor Trip is not a “given” if CST fails (Prob. ≠ 100%)
  - VCS assigns a 10% likelihood of reactor trip given a CST failure

CST Failure Rate

- **NRC Assumption:** CST failure rate of 1.0E-07/hr
- **VCS Evaluation**:
  - Value consistent with ALWR Database
  - Not specific to CSTs
  - ALWR Utilizes 3 inputs:
    - NPRD-3 (non-nuclear)
    - TMI (based on no failures)
    - Seabrook
  - Notes in ALWR (Data Base)
    - "Tank data quite limited"
    - Does not support tanks of different designs
    - NPRD-3 is non-nuclear and widely varying designs

CST Failure Rate

- **VCS Evaluation** (cont.):
  - Data Search:
    - One CST failure- Nine Mile Point (N/A for VCS)
      - Occurred before Commercial Operation
      - Construction Defect
      - Different Design (Fiberglass)
    - No CST failures were identified at nuclear plants during commercial operation (18.9 Million Hours)
    - Value based on Data Search:
      - (0.5 failures / 18.9M commercial hours) = 2.6E-08/hr

CST Failure Rate

- **VCS Evaluation** (cont.)
  - **VCS CST Experience**
    - >20 years of commercial operation
    - Lack of design and construction defects
    - Steady state operation
    - Coated Tank- Corrosion negligible
  - CST failure is not a standard initiating event in PRA models.
- **VCS Conclusion**: Based on review of CST operating experience, the Seabrook value (2.66E-08/hr) is appropriate for this use. This results in an annual CST failure frequency of (8760 hrs/yr * 2.66E-08 hrs) = 2.33E-04/yr.

EFW Piping Failure Rates

- **NRC Assumption**: "Failure" data from EPRI Risk Informed Piping Information
- **VCS Evaluation**:
  - "Failure" category includes leaks that are <50 gpm
  - "Rupture" category is for leaks >50 gpm
- **VCS Conclusion**: Use of the "Rupture" data is more appropriate
  - Total annual piping failure rate changes from:

    1.45E-05 failure/yr to <u>1.61E-06 failure/yr</u>

CST/Piping Failure Results

	NRC	VCS
CCDP (Conditional Core Damage Probability) – Rx Trip	**6.2E-03**	**6.2E-04**
CST Failure Frequency	8.76E-04/yr	2.33E-04/yr
Piping Failure Frequency	1.45E-05/yr	1.61E-06/yr
CST Initiating Event Frequency (CST Failure + Piping Failure)	**8.9E-04/yr**	**2.35E-04/yr**
ΔCDF	**5.5E-06/yr**	**1.46E-07/yr**

VCS Conclusion: This scenario results in a ΔCDF of 1.46E-07/yr, and is not risk significant

Overall Conclusions

Dan Gatlin

VC Summer Overall Conclusions

- We have demonstrated that our plant specific information significantly reduces the preliminary “bounding” NRC characterization
  - CST is robust and function is not lost
  - Actual operator actions per existing procedures significantly reduce the likelihood of a Rx trip
  - Initiating event frequency for loss of CST should be reduced

VC Summer Overall Conclusions

- This issue should be green, but regardless, we will resolve the SW corrosion issue

Questions

DRAFT APPARENT VIOLATIONS

1. 10 CFR 50, Appendix B, Criterion III, Design Control, requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment and processes that are essential to the safety related functions of structures, systems and components. Implicit in this requirement is that the measures result in the selection of materials, parts, equipment and processes which are suitable.

   The design basis as described in Updated Final Safety Analysis Report (UFSAR) Section 10.4.9, Emergency Feedwater System, states that the service water system provides a safety class backup source of emergency feedwater. The UFSAR further states: "The plant can operate indefinitely, if required, without normal feedwater. The emergency feedwater system (EFW) can take suction from the service water (SW) system for an indefinite period of time."

   Contrary to the above, the licensee’s measures did not ensure that the EFW system can take suction from the SW system for an indefinited period of time. Instead, the original purchase specification (SP-620-044461-000) for the EFW flow control valves identified the process fluid for the valves as ‘cold condensate’. Consequently, the EFW flow control valves were designed to handle clean 'cold condensate' and were not designed to handle comparatively unclean SW. As a result, if the EFW system were to take suction from the SW system, pieces of tubercles and other debris could plug the EFW flow control valves and cause a common mode failure of the EFW system. The design control aspect of this violation occurred prior to plant licensing in 1982 and existed through 2004.

Note: The apparent violations discussed at this regulatory conference are subject to further review and subject to change prior to any resulting enforcement action.

2. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that conditions adverse to quality are promptly identified and corrected.

   The licensee identified the potential for SW debris to plug the EFW flow control valves in 1986; conducted ISEG review of the issue in 1986 to 1992; reviewed related operating experience reports in 1988, 2003, and 2004; and photographed corrosion tubercles in the SW pipes to EFW in 2003.

   Contrary to the above, as of November 2004, the licensee failed to promptly correct a condition adverse to quality. The licensee failed to correct a condition wherein the EFW flow control valves were not designed to handle relatively unclean SW and, consequently, could become plugged by tubercles and other debris from SW.

Note: The apparent violations discussed at this regulatory conference are subject to further review and subject to change prior to any resulting enforcement action.