Central File
50-280

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 5, 1979

Mr. James P. O'Reilly, Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Serial No. 725
PO/RMB:svm
Docket No.: 50-280
License No.: DPR-32

Dear Mr. O'Reilly:

This letter is in response to your letter of August 31, 1979. The concerns of your staff as stated in the referenced letter and in the Inspection Report No. 50-280/79-44 are answered in the attachment to this letter.

The concrete anchor bolt sampling program as addressed in our letter of July 24, 1979 has been further reviewed. The intent of the review was to clarify any questionable data and to modify the acceptance criteria to include your criteria for interim acceptability of plant operation as per the August 20, 1979 Supplement to Bulletin 79-02. This review has been completed and the results indicate that the anchor bolts do meet your requirements for interim operation. The accessible supports will be upgraded expeditiously as deficiencies are noted during plant operation and the non-accessible supports will be upgraded during the steam generator replacement outage as required by I.E. Bulletin 79-02.

The field "as-built" program per Bulletin 79-14 as addressed in our letter of August 31, 1979, has been completed for the non-accessible areas with no notable discrepancies found on the safe shutdown systems. All other field information is being evaluated and will be handled per the intent of the Bulletin.

We plan to fully comply with the intent and schedule of Bulletins 79-02 and 79-14 and to operate Unit No. 1 in compliance with the restrictions of the lifting of the Order to Show Cause dated August 22, 1979. The results prior to startup demonstrate that the Unit No. 1 will operate safely and will be capable of being safely shutdown following a seismic event.

As you are aware, the Virginia Electric and Power Company has a strong, continuing commitment to assure safe operation of all of our nuclear units. The activities associated with the Surry units this year has caused us to continuously assess the safe operation of Surry in relation to the Show Cause Order of March 13, 1979, the changing regulatory climate, and the disposition of several Inspection and Enforcement Bulletins, including IE Bulletin 79-02. Our deliberations and investigations have caused us to draw certain conclusions about the operation of Surry and its affect on the health and safety of the public and our operating personnel. Our conclusions are indicated below:

1. Power plant piping including Surry's, has an inherent margin of safety associated with it even considering major seismic activity. This margin

7910010780 OFFICIAL COPY

of safety, admittedly incalculable, does not negate its presence. Existing power plants have shown an almost remarkable resilience to seismic activity. (Appendix F, "Seismic Capability of Nuclear Piping," Vepco submittal Serial No. 453, June 5, 1979).

2. The Show Cause Order of March 13, 1979 has caused Vepco to reanalyze all piping originally analyzed dynamically. The effort constitutes all major piping inside the containment of greater than 6" diameter. Modifications have resulted from this reanalysis, but the modifications were the result mainly of as-built conditions. The as-built verification effort in IE Bulletin 79-14 currently underway at Surry will resolve all as-built differences. All modifications to piping systems in inaccessible areas will be completed prior to start-up and this provides us with a high degree of assurance of the operability of safe shutdown systems.

3. Base plate flexibility considerations, in accordance with IE Bulletin 79-02, have been and will be incorporated for those supports for which new loads, due to the Show Cause reanalysis, exceed the original design allowable loads and for any new supports required by the reanalysis effort (Vepco Serial No. 511, June 25, 1979, W. C. Spencer to Harold R. Denton). Therefore, base plate flexibility considerations have been a part of all present and future modifications associated with the Show Cause effort.

4. Surry Unit 1 cannot operate and must be kept in a shutdown condition for any seismic event which exceeds a ground acceleration level of 0.01 g (p. 3, Order, Docket No. 50-280, signed by Harold R. Denton, August 22, 1979). This value is a small fraction of the Category I piping system design ground acceleration value of 0.15 g. We believe this is a very conservative approach of interim operation.

5. The staff of the NRC Office of Nuclear Regulation has thoroughly evaluated the procedure for design of the piping at the Surry plant not subjected to computer seismic analysis, i.e., those systems now falling in the IE Bulletin 79-14 scope of work. They have compared the design procedure to the NRC's Standard Review Plan 3.7.2 and found our methodology to be acceptable (p. 9, Safety Evaluation, Surry Power Station, Unit No. 1, Docket No. 50-280, dated August 22, 1979). We believe the NRR staff shares our confidence in the design of these systems.

6. During any one month, there is a hazard of 4.5×10^{-4} of equaling or exceeding an earthquake with a peak acceleration of 0.04 g. Thus the chances are very slight that the plant will experience any significant shaking due to an earthquake during the period of interim operation prior to steam generator replacement. The chances of experiencing the DBE are extremely small (Section 7.8 of report transmitted to NRR, June 5, 1979, Vepco Serial No. 453).

7. The Surry Power Station has some anchor bolts that are not installed per the manufacturer's instructions. We do not agree that this condition unilaterally voids the anchor bolt design capacity. In fact,

our sampling program has demonstrated our stated confidence that the anchor bolts as installed can indeed carry the loads required in order for the supports to perform their intended function. Based on this fact, coupled with other arguments presented herein, it has been shown that the Surry piping systems can perform their intended function.

8. IE Bulletin 79-02 states case histories of poor anchor bolt installations at other units. Surry Power Station does not have the anchor bolt installation problems to an extent even remotely close to the units cited in the Bulletin.

9. Documentation problems exist, but this is not to be confused with lack of design. As this submittal demonstrates, acceptable and conservative design procedures were used originally.

10. Long term programs for IE Bulletins 79-14 and 79-02 have been initiated and are continuing. The results of these ongoing activities are available for your inspection at any time, and the alternative courses of action are not obviated by interim operation.

In the process of evaluating these factors, we can only draw the conclusion that interim operation of Surry Unit 1 will have no adverse affect on the health and safety of the public or our employees.

Your expeditious review is requested as the Unit No. 1 will be ready for startup on September 6, 1979.

Very truly yours,

W. L. Proffitt
Senior Vice President - Power

RMB/svm:2K3

cc: D. Burke

ATTACHMENT TO LETTER 725
ANSWERS TO NRC REGION II CONCERNS

NRC CONCERN

1. VEPCO could not demonstrate that stress analysis had been performed for certain systems for Surry 1. These systems could be categorized as piping systems less than 6 inches in diameter except those included in the NRC Show Cause Order. Systems for which documentation was not available included certain ones (both inside and outside containment) that would be necessary for the safe shutdown of the plant following the seismic event. VEPCO believes that this field run piping had been designed using hand calculations to determine stresses and that the supports were located using "typical" guidelines. Data presented or made available to NRC inspectors did not appear to provide necessary information to support a finding of technical adequacy.

Documencation was not available that would assure that the pipe support systems including expansion anchors for the piping systems described above had been designed to support the loads that would be imposed.

Based on the above, we believe that additional assurances by VEPCO are necessary and that VEPCO should provide justification for a sampling test program that would provide assurance that the unit can be safely operated and can be safely shut down in the event of a seismic event.

VEPCO RESPONSE

Information available at the meeting of August 28, 1979, was not complete and should be inspected with our Architect Engineer; however, the above concerns had previously been answered to the NRR staff during their overall inspection for the Order to Show Cause. To address the NRC concern expressed above, the piping systems referred to were designed using the criteria and design requirements in effect during the design and construction of Surry. The design of piping systems using "decal" loads is an acceptable method of design which has been reviewed and approved by the NRC. When Surry was constructed, provisions were not made to incorporate all field engineered changes into the construction drawings and, therefore, "as-built" discrepancies can be expected. The documentation and record-keeping requirements in effect today are more stringent than those in existance when Surry was built and therefore, supporting calculations and drawings for these discrepancies are not as accessible as for a plant recently constructed. Since the design requirements for this piping were the basis for field design modifications, VEPCO believes that the "as-built" discrepancies were engineered changes for which the documentation is not readily available and not changes which were made without any design basis. The use of "decal" loads as an acceptable method of design was reviewed by the NRR staff during their overall inspection for the Order to Show Cause. After the inspections by the NRR Staff at our Architect Engineer's offices in Boston, they were satisfied and addressed this question in their SER of August 22, 1979 as quoted below:

Verification of Analysis Methods

We have reviewed the acceptability of the analytical methods which are currently a basis for the facility piping design. The licensee has identified the following computer codes/analysis methods as applicable:

PSTRESS/SHOCK 0 (Initial 3 Versions of SHOCK 1)
Static Analysis Methods
NUPIPE

PSTRESS/SHOCK 0

This code was used for 12 safety related system problems and although it did not algebraically sum responses, the code was not equivalent to current practice. The licensee, therefore, reanalyzed these systems with the NUPIPE code.

Statis Analysis

Methods used for design of the piping at the Surry Plant not subjected to computer seismic analysis were based on simple beam formulations which, in essence, controlled seismic stress levels through use of pre-established seismic spans. These simple beam formulations were utilized to calculate maximum allowable spans based upon an assumed acceleration factor of 1.5 times the peak acceleration obtained from the response spectra. In calculating the maximum span lengths, it was conservatively assumed that a longitudinal pressure stress of 4,000 psi and a maximum deadweight stress of 1,500 psi were present in the pipe. This combined value of 5,500 psi was subtracted from the allowable stress (1.8 S_h for pressure and deadweight and seismic) to obtain a seismic allowable stress.

Calculating maximum spans by this procedure results in maximum allowable spans greater than the deadweight spans recommended in ANSI B31.1. Thus deadweight governs and provides a greater number of supports resulting in closely spaced restraints. To minimize effects of concentrated weights, restraints were placed as required at valves and other concentrated masses.

For Surry Unit 1, piping 6 inches in diameter and smaller was generally analyzed using the simplified static method, with the option of utilizing more rigorous methods available to the analyst.

Piping 2 inches and below was shown on the piping drawings diagrammatically (i.e., without detailed dimensions). The stress engineers located supports during the installation process working at the site with erection isometric sketches.

As described above, the stress analysis was performed by assuming many simple supported straight beams, the spans of which are governed by dead load spacing requirements of ANSI B31.1. The piping fundamental frequencies associated with these maximum allowable spans (9.7 to 13.6 cycles per second) are not in resonance with the building in which they are located (2 to 8 cycles per second). The method of equivalent static analysis outlined in this procedure has been compared with the NRC's Standard Review Plan 3.7.2 and is found to be acceptable.

End of NRC SER quote

VEPCO RESPONSE (CONT'D.)

Static Analysis

Methods used for design of the piping at the Surry Plant not subjected to computer seismic analysis were based on simple beam formulations which, in essence, controlled seismic stress levels through use of pre-established seismic spans. These simple beam formulations were utilized to calculate maximum allowable spans based upon an assumed acceleration factor of 1.5 times the peak acceleration obtained from the response spectra. In calculating the maximum span lengths, it was conservatively assumed that a longitudinal pressure stress of 4,000 psi and a maximum deadweight stress of 1,500 psi were present in the pipe. This combined value of 5,500 psi was subtracted from the allowable stress (1.8 S_h for pressure and deadweight and seismic) to obtain a seismic allowable stress.

Calculating maximum spans by this procedure results in maximum allowable spans greater than the deadweight spans recommended in ANSI B31.1. Thus deadweight governs and provides a greater number of supports resulting in closely spaced restraints. To minimize effects of concentrated weights, restraints were placed as required at valves and other concentrated masses.

For Surry Unit 1, piping 6 inches in diameter and smaller was generally analyzed using the simplified static method, with the option of utilizing more rigorous methods available to the analyst.

Piping 2 inches and below was shown on the piping drawings diagrammatically (i.e., without detailed dimensions). The stress engineers located supports during the installation process working at the site with erection isometric sketches.

As described above, the stress analysis was performed by assuming many simple supported straight beams, the spans of which are governed by dead load spacing requirements of ANSI B31.1. The piping fundamental frequencies associated with these maximum allowable spans (9.7 to 13.6 cycles per second) are not in resonance with the building in which they are located (2 to 8 cycles per second). The method of equivalent static analysis outlined in this procedure has been compared with the NRC's Standard Review Plan 3.7.2 and is found to be acceptable.

End of NRC SER quote

VEPCO RESPONSE (CONT'D.)

The following is a listing of the static analysis methods used for design of small piping.

SIMPLIFIED METHODS FOR PIPING ANALYSIS
SURRY POWER STATION

LICENSING REQUIREMENTS:

Piping systems 6 in. in diameter and smaller were stress analyzed, using acceleration loads from ground response spectra. Stops, guides, and snubbers, were located to preclude piping resonance with the supporting structure. Displacements of the piping were checked to ensure that there were no interferences with any other equipment or piping. The documents listed below specify the initial design requirements.

Date	Ref.	Method	Applicability
6-23-69	PS-4	Preliminary Issue Simplified analysis standard support loads.	Nominal pipe size 2" through 24"
7-25-69	PS-1	Kellogg method to span supports ± 50% of the peak structural resonant frequency.	Low temperature piping under 6" dia.
10-28-69	PS-4 Rev. 1	Simplified analysis standard support loads (total).	Nominal pipe size 2" through 24".
4-23-70	Aides	Computer tabulated seismic stress for given pipe, span, and acceleration plus nomographs developed from equations for seismic support span, pipe frequency, thermal stress and flexibility.	Nominal pipe size 6" and under.

Date	From/To	Content
7-17-67	Stone & Webster Procedure	Description of: Pipe stress analysis for earthquake and tornado resistant systems specifying: Systems designed for earthquake Type of analysis Technique Seismic forces Thermal and seismic stress Earthquake reaction Combination of stresses Allowable stresses Reaction forces and moments Tornado protection Structural response (seismic)

Date	From/To	Content
3-12-68	R. M. Rome to M. H. Pedell	"If all seismic pipe stress problems were to be analyzed dynamically, a simplified method of computer input is mandatory." Needs: 1. Accept input from stress program 2. Call dead loads 3. Ecompass limitation of 75 nodes 4. Auto. thermal, seismic and weight combination for comparison with the allowables.
12-22-70	R. P. Klause to O. Surgecoff	"Seismic analysis of field run piping" - 2" and smaller shown diagrammatically. - Boston stress engineer to analyze the lines at the job. - Engineer would walk lines and make freehand isometrics. - Hanging of diagrammatic lines is field responsibility. - Clip 3/8" sample lines every 4-5' to walls or cable trays.
1971	M. H. Pedell to R. H. L'Amoureux	"VEPCO Surry - Units 1 and 2 Stress analysis revised and seismic criteria" as per Surry FSAR seismic loads on all Class I piping must be increased. Discussion of impact of stress, support loading and schedule.
1971	S & W PSAR	Seismic adequacy review - Covers response spectra, reactor coolant system and Class I piping, equipment, and components. Section 4.1.1 "Method of Dynamic Analysis" Piping 6 in. and smaller were seismically supported to "preclude piping resonance with the supporting structure."
9-15-71	S & W	Seismic design review Equipment and piping Surry Power Station Same as above (of 6-1-71) revised 9-15-71 Ref. Sections remain the same Includes response spectra

Therefore, documentation is available to ensure that the pipe support systems, including expansion anchors, for the smaller than 6 inch lines of NRC concern had been designed to support the loads that would be imposed.

Additional assurances are available in that, per Bulletin 79-14, all 2 1/2 inch and larger piping will be or has been re-verified as to 'as-built' conditions. The Unit No. 1 piping has all been redrawn. The small bore inaccessible safe shutdown piping has been initially evaluated with no notable discrepancies found. All other safe shutdown inaccessible piping has been re-analyzed per the Show Cause Order and modifications made.

VEPCO RESPONSE (CONT'D.)

The small bore safe shutdown accessible piping has been initially "as-built" and initial walkdown Quality Control comments and discrepancies are being verified and will be evaluated on a priority basis and handled per the requirements of 79-14 as stated in our letter of August 31, 1979

Based on the above, we believe that sufficient assurances have been provided and that in fact the unit can be safely operated and can be safely shutdown in the event of a seismic event.

NRC CONCERN

2. VEPCO should demonstrate that anchors with sleeve top to expansion plug dimensions that indicate less than full sleeve embedment or less than full sleeve expansion will withstand a loading of at least two times the design load (1/4 manufacturer's ultimate). It is suggested that anchors with the worst dimensional anomalies be load tested but equivalent methods would be acceptable.

VEPCO RESPONSE

The Anchor Bolt Inspection and Test Program conducted required that anchors in the sample be proof loaded to demonstrate that the anchor will withstand at least its design capacity (1/4 of manufacturer's ultimate) in tension in accordance with the test requirements outlined in I.E. Bulletin 79-02. Tensile load was applied to the bolts by applying a torque to the bolt. The torque values used to apply the proof load were furnished by the manufacturer and were based on tests performed by others. These values are listed below:

Table 1

Anchor	Bolt ϕ (in.)	Proof Torque (Ft-Lb)
S12	1/2	30
S58	5/8	45
S34	3/4	70
S78	7/8	95

The above values were compared to those obtained from an analytical method to correlate installation torque to bolt tension. Since the proof load torque listed above was significantly higher than the calculated torque required to produce the required tension using this analytical method, the proof load torques were judged to be acceptable. The analytical method used is presented in the text "Mechanical Engineering Design" by J. E. Shigley, McGraw-Hill Book Company, 1963, and employed a coefficient of friction of 0.15 as a value for average bolts and nuts. Original design criteria for the anchor bolts installed at Surry Power Station was based on allowable bolt loads factored for 3000 psi concrete with a factor of safety of four (4) based on the manufacturer's published data. These allowable bolt loads were the tensile loads for which the corresponding torques were calculated. Table 2 shows these calculated torque values and the actual proof torque values provided by the manufacturer as indicated in Table 1 above. Table 2 also shows the factor of safety proven by applying these torques assuming the torque-tension relationship $T = K\ Fd$ given by Shigley as referenced above.

VEPCO RESPONSE (CONT'D.)

Table 2

Bolt Size	Design Load(1) Allowable (lb)	Calculated Proof (2) Torque (Ft-Lb)	Actual Proof Torque (Ft-Lb)	K (3) Value	Factor Of(4) Safety
3/8	1218	8	25	0.204	3.13
1/2	1826	15	30	0.201	2.00
5/8	2517	26	45	0.199	1.73
3/4	3487	42	70	0.194	1.66
7/8	3835	54	95	0.193	1.75

(1) based on factor of safety of four (4) and adjusted for 3000 psi concrete.

(2) based on analytical method referenced above

(3) Torque coefficient K given by Shigley, 1963 edition.

(4) Assumes approximate relationship T = K Fd

The proof load torque as discussed above demonstrates that the shell is sufficiently expanded and embedded to resist some margin above design allowable tension loads, and therefore, justifies concluding that it is also adequate for shear and combined shear/tension design load allowables.

The criteria used to evaluate the acceptability of shell movement during proof loading of the anchors was that shell outward movement was insufficient to permit the shell to come up into contact with the baseplate or that the shell did not rotate when the proof load torque was applied. Minor shell movement was accepted if the shell locked up and did not contact the baseplate when load tested.

On August 20, 1979, the NRC issued Supplement No. 1 to Revision 1 of I.E. Bulletin 79-02 to establish criteria for evaluation of interim acceptability of plant operation with less than the design factors of safety for piping supports due to as-built problems, under design, baseplate flexibility, or anchor bolt deficiencies. The supplement requires that for an anchor bolt, the factor of safety be equal to or greater than two (2). In order to demonstrate by test that a safety factor of two (2) exists in anchors with sleeve top to expansion plug dimensions that indicate less than full sleeve embedment or less than full sleeve expansion, several anchors with the worst dimensional anomalies and passed the proof load test were re-tested as suggested by the NRC.

The anchors were re-tested to at least two times the design load (1/4 manufacturer's ultimate in 3000 psi concrete) by applying an equivalent torque. The equivalent torque was again calculated using the approximate relationship given by Shigley as referenced above. Since this is an approximate relationship, a factor of 1.25 was applied to the calculated required torque to account for unknowns and to ensure that an equivalent tensile load was applied. The following relationship was thus

VEPCO RESPONSE (CONT'D.)

used to calculate the required test torque to demonstrate a safety factor of two (2).

$$T_{S.F.=2} = (2)\ (1.25)(K\ Fd)$$

Where $T_{S.F.=2}$ = Torque Required to Test Bolt to S.F. = 2
K = torque coefficient

F = design load allowable based on FS of 4 and 3000 psi concrete

d = diameter of bolt

The test torque values used are as listed below:

Bolt Size (in.)	Test Torque	Safety Factor Assuming Approximate Relationship T=1.25 K Fd
3/8	25	2
1/2	40	2
5/8	70	2
3/4	110	2
7/8	135	2

The bolts re-tested to demonstrate a safety factor of two (2) are listed in Table 3. The fourteen (14) bolts selected for re-test were representative of the anchors with the worst dimensional anomalies. The test results indicate that all but one (1) of the bolts re-tested passed the proof load re-test proving that a safety factor of greater than two (2) exists in these anchors. One anchor, when removed to check the dimensions, could not be re-inserted fully into the anchor when the anchor and sleeve began turning together. Therefore, it could not be shown that a safety factor of two (2) existed in this anchor prior to the re-test. This anchor will be repaired.

Based on the results of the re-test, we believe that reasonable assurance exists that a safety factor of at least two (2) exists in the bolts included in the representative sample which satisfactorily passed the initial proof load test.

TA[illegible] 3
TORQUE TEST RESULTS W[illegible]ING SAFETY FACTOR OF 2

IPING SYSTEM AND RAWING SERIES	SEQUENCE NUMBER	HANGER NUMBER	ANCHOR ID NO. AND SIZE (in.)	BACKOFF TORQUE (Ft.-lb)	DEPTH OF SET PLUG (in.)	SHELL PROJ'N (in.)	THREAD ENGAGE-MENT (in.)	PROOF LOAD TORQUE (Ft.-lb)	SHELL MOVE-MENT (in.)	DEFICIENCIES NOTED
CSRS 11448-FP-13A	1-9	H-92	4-3/4	70	2	-1/8	7/8	110	No	
odified Safety Injection 11448-MSK-2A	3-1	H-7	2-3/4	70	2-1/8	-1/8	3/4	110	No	
afety Injection 11448-ASK-2A	3-2	H-10	1-3/4	70	2	0	1	110	No	
RHR 11448-FP-12A	3-17	H-30	5-3/4	70	2-1/8	-1/8	1	110	No	
RHR 11448-FP-12A	3-27	H-45	5-3/4	70	1-3/4	-1/4	1-3/8	110	No	
CVCS 11448-FP-10C	5-53	H-206	2-3/4	<70	2	-1/8	1-1/4	110	No	
PSSRS 11448-FP-9A	6-12	H-111	3-3/4	70	2-1/2	-1/8	1	110	No	
PSSRS 1148-FP-9B	6-21	H-24	2-3/4	70	2-1/8	-1/8	1-3/8			Could not perform torque test - anchor & shell started turning.
CH 11448-FP-10R	6-39	H-170	1-3/4	70	2-1/4	0	1-1/8	110	No	
CH 11448-FP-10R	6-41	H-157	1-3/4	70	2-1/4	-1/8	1	110	No	
LHSI 1148-FP-11A	7-29	H-22B	1-3/4	70	2-1/4	0	1-1/8	110	No	
Component Cooling 11448-FP-16C	8-30	CC-A14	11-7/8	50	2-3/4	-1/8	1	135	No	

TABLE 3 (CONTINUED)

IPING SYSTEM AND RAWING SERIES	SEQUENCE NUMBER	HANGER NUMBER	ANCHOR ID NO. AND SIZE (in.)	BACKOFF TORQUE (Ft.-lb)	DEPTH OF SET PLUG (in.)	SHELL PROJ'N (in.)	THREAD ENGAGE-MENT (in.)	PROOF LOAD TORQUE (Ft.-lb)	SHELL MOVE-MENT (in.)	DEFICIENCIES NOTED
omponent Cooling 1448-FP-16M	9-5	R-134	3-1/2	30	1-1/4	-1/8	3/8	40	No	
omponent Cooling 1448-FP-16C	9-15	CC-RH-231	1-1/2	30	1-3/8	0	3/4	40	No	

NRC CONCERN

3. VEPCO should address, quantify, and analyze each case where significant deficiencies such as broken anchors, adjacent nuts missing and smaller than drawing required number and/or size of anchors installed were observed but excluded from the sample and thus from the summarized failure results in your letter of July 24, 1979.

VEPCO RESPONSE

The calculated failure rates presented in VEPCO letter Serial No. 146 were based on the deficiencies found during inspection of the 200 anchor bolts included in the sample program for which the installation attributes could be measured. The calculated failure rates can therefore be taken as representative of the total population since they were chosen by a random selection.

Deficiencies or discrepancies noted during the inspection but not included in the sample such as broken anchors, adjacent nuts missing, or smaller than required number and/or size anchors have been listed on the corrected data summary sheets contained in the Anchor Bolt Inspection and Test Program. These deficiencies/discrepancies are being analyzed or corrected in conjunction with deficiencies noted on bolts included in the representative sample. All deficiencies/discrepancies noted during conduct of the test program are addressed and analyzed in the following paragraphs.

Discrepancies found to exist between the original design drawings and the "as-built" condition are being evaluated to determine the adequacy of the support. These discrepancies consist of (1) different anchor size, (2) different number of bolts and different configuration, (3) welded baseplate instead of bolted, (4) missing supports, and (5) other "as-built" discrepancies as described below.

The evaluation consists of making "as-built" drawings and re-analyzing the support to determine the anchor bolt loads using the original analytical methods. The loading combinations being used are consistent with those which have been approved under the Surry Task Force interim start-up pipe support design criteria. Any non-conformance identified as a result of the re-analysis will be classed in accordance with the criteria given in Supplement No. 1 to I.E. Bulletin 79-02 to determine operability. The discrepancies being analyzed are quantified and analyzed below:

(1) Different Anchor Size

The proper anchor bolt size was not always installed as required on the support detail drawings. Those supports determined to have improper size anchors are being "as-built" to the existing field conditions and re-evaluated as described previously. There are 47 support baseplates containing different size anchors. The worst case found to exist involved the situation where anchor bolt of 1/2 inch diameter was installed in place of the required anchor bolt of 3/4 inch diameter. In an effort to qualify the difference in anchor size, the following relationship was used:

$$T_{(1/2)} = [(K)(F)(d)]\ (2)(1.25)$$

$$= [(0.201)(3487)(1/2)]\ (2)(1.25)(1/12)$$

$$= 73 \text{ Ft-Lb's}$$

VEPCO RESPONSE (CONT'D.)

where: T is the Test Torque used to qualify the 1/2 inch anchor

K is the Torque Coefficient for the existing anchor

F is the Design Load Allowable for the required anchor assuming 3000 psi concrete and a safety factor of 4.

d is the existing bolt diameter.

In the worst case referenced above, three 1/2 inch anchor bolts were torqued to 75 ft-lb's. This torque value justifies the ability of the smaller sized anchors to withstand a tensile load of equivalent to at least twice the design allowable load of the required anchors. The results of the testing are shown in Table 4 titled "Test Results for Anchors Smaller Than Design". The test was performed in accordance with procedure P-4 of Special Test 39 except for the increased test torque value. The test results indicated no deficiencies. The test results indicate that although lesser sized anchor bolts have been installed in place of the required anchors, they are capable of handling loads equal to twice the design alowable load for the required anchor size. In addition, preliminary results of "as-built" analysis of the 24 hangers analyzed, shows that a safety factor of at least two (2) is present for 4500 psi concrete.

(2) Different Number of Bolts/Configuration

Baseplates were also found and verified to have different numbers of bolts and different bolting configurations. Forty-six baseplates were found to exist in this condition. Each of the supports associated with these baseplates are being "as-built" to their existing field conditions and will also be re-evaluated as described previously. Preliminary results of the seventeen (17) hangers re-analyzed to date indicate that a safety factor of at least two (2) exists with 4500 psi concrete.

(3) Welded Baseplates

There were nine (9) baseplates found to be welded to structural members rather than bolted using the required concrete expansion anchors. The supports containing these baseplates are being "as-built" for re-evaluation. However, the initial determination is that this type of attachment meets the design requirements.

(4) Missing Supports

Several supports were listed as missing on the data sheets and as such were excluded from the sample program. All but one of these supports has been accounted for and identified during subsequent inspections. Therefore, these discrepancies have been resolved.

TABLE NO. 4
TEST RESULTS FOR ANCHORS SMALLER THAN DESIGN
(SAFETY FACTOR OF 2)

PIPING SYSTEM AND DRAWING SERIES	SEQUENCE NUMBER	HANGER NUMBER	ANCHOR ID NO. AND SIZE (in.)	BACKOFF TORQUE (Ft.-lb)	DEPTH OF SET PLUG (in.)	SHELL PROJ'N (in.)	THREAD ENGAGE-MENT (in.)	PROOF LOAD TORQUE (Ft.-lb)	SHELL MOVE-MENT (in.)	DEFICIENCIES NOTED
CVCS 11448-FP-10T	5-24	H-46	2-1/2	30	1-1/4	-1/8	3/4	75	No	1/2 inch, 3/4 inch required
CVCS 11448-FP-10T	5-55	H-220	1-1/2	30	1-7/16	-3/16	5/8	75	No	1/2 inch, 3/4 inch required
CVCS 11448-FP-10D	5-70	H-212A	4-1/2	30	1-3/8	-1/4	3/4	75	No	1/2 inch, 3/4 inch required

NRC CONCERN

4. VEPCO should address and justify the fact that the random sample did not include anchors from all safety related systems.

VEPCO RESPONSE

As described in the Anchor Bolt Inspection and Test Program, the baseplates inspected were selected randomly from those baseplates in safety related systems for which design data was available at the time. As a result, the random selection did not include anchors installed in all safety related piping systems. Although the inspection did not include bolts from each safety related system, the test program did include bolts installed in all piping systems 2 1/2" and larger that are required to operate for safe shutdown. Bolts were tested in each of those systems except the Auxiliary Feedwater System. These bolts were excluded from the sample since the baseplates were grouted and the studs could not be removed for inspection.

Those baseplates with anchor bolts tested are considered to be representative of the total population and the workmanship at the time of construction since there is no unique characteristic of these supports that would affect their randomness and no different construction practices were used for bolt installation from one system to another.

Since the bolts were selected randomly and included all systems required for safe shutdown with accessible anchor bolts, the program conducted is representative of all safety related piping systems and serves as justification for interim operation.

NRC CONCERN

5. VEPCO should summarize the results of the test program based on the reviewed and corrected data. The format should be similar to VEPCO's letter of July 24, 1979, and include the calculated failure rate(s).

VEPCO RESPONSE

All initial data sheets completed have been reviewed and those data sheets which had "questionable" data, such as large or small depths to the set plug or thread engagements that exceeded the space available or that exceeded the actual thread length measured in new anchors of that size have had the data verified and discrepancies have been resolved. The corrected data has been entered in the summary tables. Results of the test program based on the reviewed and corrected data is summarized below. The results given are based on the 200 bolts included in the sample for which the installation attributes could be measured. Deficiencies noted during the inspection but not included in the sample such as broken anchors, adjacent nuts missing, or smaller than drawing required number and/or size of anchors installed are addressed in Item No. 3 above. The calculated failure rates given below can therefore be taken as representative of the total population of anchors installed in safety related systems since they are based on the 200 bolts selected randomly.

The results were evaluated in accordance with the acceptance criteria contained in the Anchor Bolt Inspection and Test Program to determine if the anchor bolt installation was acceptable. The acceptance of the anchor was reduced to only four (4) attributes to determine if the bolt would perform its design function: anchor size, initial tightness, thread engagement, and proof load capability.

The inspection results indicated that 26 anchor bolts of the 200 included in the test sample or 13% had an installed anchor smaller than the anchor diameter specified on the original design drawing. Documentation could not be located to verify the adequacy of the "as-built" installation. For those baseplates where the same number of anchor bolts were used, but smaller anchor bolts were provided, the design factor of safety based on 3000 psi concrete is reduced from 4 to a minimum of 2 for tension and between 1.7 and 3.5 for shear depending on bolt size. The reduced factors of safety are higher when the actual concrete strengths are considered. The supports in which smaller anchor bolts exists are being "as-built" and re-analyzed by Stone and Webster Engineering Corporation as described in our response to Item 3 above. Any nonconformances identified will be classed in accordance with the acceptance criteria given in Supplement No. 1 to I.E. Bulletin 79-02 to determine operability.

Of the bolts inspected, 9 anchor bolts of the 200 included in the test sample or 4.5% had no initial tightness. Some initial tightness is required to restrain the baseplate to prevent displacement when the load is applied.

The inspection results indicated that 4 anchor bolts of the 200 included in the test sample or 2% did not have a thread engagement of at least 1/2 of the bolt diameter. Therefore, it can be concluded that a thread engagement problem does not exist at Surry.

Bolts in the sample were proof torqued to a value equal to a minimum of 1.66 times the allowable tension design load based on 1/4 of the ultimate pull out value for 3000 psi concrete. Of the 200 inspected, 13 or 6.5% did not hold when loaded to the proof load. Five (5) of the 13 bolts which did not meet this criteria rotated in the hole and thus would not hold the required torque. Even though the shell turned when the bolt was torqued, the bolts were recognized to have some capacity. Additional proof load testing conducted on anchors with the worst dimensional anomalies as described in Item 2 previously provided reasonable assurance that a safety factor of at least two exists.

In summary, the testing program conducted at Surry showed that 92.8% of the attributes measured to determine acceptability for interim operation were considered to be acceptable. The deficiencies observed during the sampling program indicate primarily that the safety factor above design load was less than required by I.E. Bulletin 79-02 for final design margins, but satisfies the interim operability requirements of Supplement No. 1 to I.E. Bulletin 79-02.