SCE&G
A SCANA Company

South Carolina Electric & Gas Company
P.O. Box 88
Jenkinsville, SC 29065
(803) 345-4040

Ollie S. Bradham
Vice President
Nuclear Operations

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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Response to Generic Issues From
NRC Bulletins 79-02/79-14 Follow-Up
Inspection Report 50-395/89-200

Gentlemen:

This submittal responds to generic issues resulting from the Bulletins 79-02 and 79-14 follow-up review conducted at the Virgil C. Summer Nuclear Station (VCSNS) in November and December of 1989. These generic issues are documented in Appendix C of NRC Inspection Report 50-395/89-200, dated January 26, 1990. As noted in this report, the deficiencies did not raise any significant safety concerns regarding the adequacy of safety related structures, piping or as-built configuration. South Carolina Electric & Gas Company (SCE&G) presented its position on these issues to the NRC in a February 8, 1990, meeting which is documented in NRC minutes dated February 15, 1990. During that presentation, the following major points were discussed:

1. No significant safety issues were identified as a result of this inspection.

2. SCE&G met the intent of both IEB 79-02 and IEB 79-14.

3. SCE&G utilized a very rigorous design verification process during the effort to satisfy these bulletins, which included as-built verification, design input verification, a reiterative analytical verification process, and adequate technical management through G/C and other SCE&G contractors.

4. The licensing basis for the VCSNS documents the SCE&G compliance with the ASME code requirements for code allowables; documents the acceptable independent verification by SWEC of the EFW piping analysis as required by the NRC; documents the NRC Battelle Northwest Pacific Laboratories confirming analysis of the VCSNS piping analysis compliance with code allowables and a confirmation of the ability to use computer models; and

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documents the general acceptability of the SCE&G QA program for design control.

5. SCE&G (and our contractors) used what was considered accepted industry practices during the time frame when the IEB 79-02 and IEB 79-14 activities were performed. Industry practice was used due to the lack of regulatory prescribed modeling techniques and/or criteria. This allowed for different modeling techniques to be used by each contractor utilized by SCE&G, but did not jeopardize the acceptability of the final analysis even though analytical results for the same problem if performed by different contractors could vary by as much as 15-20 percent (i.e., one analysis technique could be 20% more conservative than another technique that is acceptable and conservative itself). Current industry practices may change some of these modeling techniques and criteria and could result in slightly different analytical results today with no significant impact to the original analysis.

6. Retrofitting the IEB 79-02 and IEB 79-14 analyses to current "state-of-the-art" practices is neither warranted nor justified.

7. Specific compliance deficiencies identified in this report will be corrected and evaluated as appropriate for generic impact.

In addition to the four generic issues from Appendix C of the Inspection Report, this submittal also addresses the open item on the PRYTEN computer program qualification and building internal seismic anchor movement. These open items are documented in Appendix A (open item 50-395/89-200-05 and 50-395/89-200-06, respectively) of the January 26, 1990, Inspection Report.

The four generic issues identified by the NRC as being non-conservative and/or not having supporting documentation reviewed in this report are:

1. Zero period acceleration (ZPA): Teledyne Engineering Services did not consider it while both Gilbert/Commonwealth and Impell did,

2. Seismic anchor movements (SAMs): directional responses of SAMs for adjacent structures were combined by the square root of the sum of the squares (SRSS) as opposed to the absolute sum, and the 1/8" absolute displacement criteria between adjacent buildings was non-conservative,

3. Containment growth was not considered, and

4. A decoupling ratio of 15% was used as opposed to 6%.

Additionally, SCE&G included its position on the issue of PRYTEN computer code verification and the evaluation of internal building SAMs because it was felt that they potentially had generic impact.

Attachments I through VI to this submittal provide a more detailed discussion of the six technical issues including the review process, evaluations performed, and conclusions reached. Attachment VII provides a list of references. On all six issues, the current project approach is shown to be technically acceptable with no procedural changes or rework necessary. SCE&G has revised its procedure to clarify the requirements to evaluate internal

building SAMs (Attachment VI, Appendix A). The detailed calculations referenced in this submittal are available for review at VCSNS.

For future analysis work, SCE&G will require the evaluation of ZPA for all new analyses problems. No other program changes are planned for these issues based on the attached conclusions. However, for future detailed reanalysis programs such as snubber reduction, SCE&G will evaluate its original practice and the current industry practice and standards and determine the appropriate criteria to be used on a case-by-case basis.

SCE&G anticipates that all other open items related to the 79-02/79-14 follow-up inspection will be completed by September 1, 1990. At that time, SCE&G will be prepared for a follow-up review by the NRC.

I declare that the statements and matters set forth herein are true and correct to the best of my knowledge, information and belief.

Should you have any questions, please call at your convenience.

Very truly yours,

[signature] for

O. S. Bradham

MAB/OSB:jm
Attachments

c: O. W. Dixon, Jr./T. C. Nichols, Jr.
E. C. Roberts
R. V. Tanner
S. D. Ebneter
J. J. Hayes, Jr.
General Managers
C. A. Price
R. B. Clary
K. E. Nodland
J. C. Snelson
NRC Resident Inspector
J. B. Knotts, Jr.
NSRC
NPCF
RTS (IE 89200)
File (815.01)

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ATTACHMENT I

1.0 ZERO PERIOD ACCELERATION (ZPA) - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX C, ITEM 1

The NRC IEB 79-14 follow-up review identified that Teledyne Engineering Services (TES) had not considered zero period acceleration (ZPA) in the analysis they performed while both Gilbert/Commonwealth (G/C) and Impell (EDS) had considered ZPA in all of their analyses. The NRC viewed this as an inconsistency in analysis technique.

1.1 ZPA EVALUATION

Virgil C. Summer Nuclear Station had no requirement for the inclusion of ZPA in any piping analysis problems. G/C's and IMPELL's modeling techniques automatically included ZPA, while TES's modeling technique did not. The analyses performed by G/C and IMPELL had additional conservatism beyond that required by our Design Basis. Bounding analyses have been performed on the TES analyses to demonstrate acceptable results with the effects of ZPA included as follows:

Of the total piping analysis scope, TES performed 33 analyses for which ZPA was not considered. Those analyses most susceptible to the effects of ZPA were determined based on the lowest mass participation (0 to 33 HZ) associated with the largest corresponding ZPA values for rigid response. This identified the most rigid piping/support systems which were the most susceptible to the effects of ZPA. This evaluation is tabulated in Attachment I, Appendix A. Of the 33 sub-systems analyzed by TES, EF-02 and RC-03C were selected as the two most susceptible to the effects of ZPA.

The two identified worst case sub-systems were analyzed to account for the effects of ZPA. The original project approach of screening for the larger of seismic inertia and ZPA was used. The analyses resulted in some support load and pipe stress increases. However, calculations performed have demonstrated sufficient margin exists to accommodate these increases.

Also, the original 79-14 analysis performed by TES has inherent conservatisms due to the method used by G/C in developing the response spectra curves. A damping value of 2% was used for Operating Basis Earthquake (OBE) and the resulting response spectra was scaled up for Safe Shutdown Earthquake (SSE). Regulatory Guide 1.61, dated October 1973, allows the use of 4% and 7% (OBE and SSE) respectively, for reinforced concrete structures. Using curves based on 2% structural damping over-predicts pipe stresses and support reactions.

1.2 CONCLUSIONS ON ZPA

Based on results from the two bounding analyses, plus the inherent conservatism in the response spectra, the inclusion of ZPA effects in the seismic analysis of the 33 sub-systems analyzed by TES would produce acceptable pipe stresses and support reactions. Details of the ZPA evaluation are documented in calculation TES 7255-1 (Ref.13) and G/C calculation NRC7914-07 (Ref. 21).

COMPARISON OF MODEL PARAMETERS FOR ZPA AFFECTS

SUB-SYSTEM NUMBER	NUMBER OF NODES	NUMBER OF SUPT'S	NUMBER OF MODES	FIRST MODE FREQ.	ACCUMULATIVE MASS PARTICIPATION RATIO @ 33 HZ. N-S (X)	VERT (Y)	E-W (Z)	MAXIMUM FLOOR ACCELERATION ZPA(X)	ZPA(Y)	ZPA(Z)
EF01	223	37	10	11.8	19.2	8.8	20.6	.470	.222	.466
EF02	153	27	8	14.1	7.8	34.4	4.0	.574	.235	.357
EF03	189	38	19	8.7	30.8	37.2	14.3	.574	.235	.357
EF04	105	24	16	12.1	25.0	56.1	26.6	.574	.235	.357
EF05	75	19	9	11.4	7.0	18.5	54.0	.574	.235	.357
EF20	89	17	23	2.0	66.4	77.5	54.2	.347	.195	.342
EF21	77	15	15	4.5	59.8	50.5	30.8	[illegible]	.195	.342
EF22	77	15	14	4.8	46.2	47.0	60.6	[illegible]	.195	.342
RC01	364	75	7	13.4	12.8	6.0	4.8	.347	.134	.342
RC03A	19	1	2	70.4	47.2	3.8	64.0	.304	.134	.298
RC03B	19	1	1	77.0	3.9	3.1	46.8	.304	.134	.298
RC03C	19	1	1	71.2	43.4	3.2	1.2	.304	.134	.298
RH01	87	17	8	7.5	18.5	27.6	36.2	.183	.157	.192
RH02	94	19	8	6.1	56.1	10.8	46.0	.183	.157	.192
RH03-1	364	107	56	5.0	35.6	46.5	47.7	.461	.222	.338
RH03-2	156	42	30	4.5	20.2	50.5	42.8	.461	.222	.338
RH09	64	11	12	5.3	72.2	64.0	71.0	.363	.109	.338
RH10	69	12	16	2.7	32.4	25.7	50.6	.363	.109	.338
RH15A*	269	58	63	3.4	54.6	57.2	60.8	.461	.222	.338
RH15B*	269	53	45	2.9	34.5	38.2	52.7	.461	.222	.338
RH15C*	315	85	69	5.8	69.2	42.9	67.2	.461	.222	.338

* Includes RH05, RH06, RH07 & RH08

COMPARISON OF MODEL PARAMETERS FOR ZPA AFFECTS

SYSTEM NUMBER	NUMBER OF NODES	NUMBER OF SUPT'S	NUMBER OF MODES	FIRST MODE FREQ.	ACCUMULATIVE MASS PARTICIPATION RATIO @ 33 HZ. N-S (X)	VERT (Y)	E-W (Z)	MAXIMUM FLOOR ACCELERATION ZPA(X)	ZPA(Y)	ZPA(Z)
SI01	65	16	3	17.3	35.3	19.9	20.9	.183	.157	.192
SI02	59	14	2	21.4	41.0	5.3	34.3	.183	.157	.192
SI03	60	10	5	9.0	23.9	17.4	59.5	.195	.157	.201
SI04	222	55	47	3.6	65.3	31.9	50.7	.461	.222	.274
SI08>*	221	41	33	4.7	47.6	37.0	34.1	.183	.157	.192
SI09<*	397	76	50	4.3	40.0	13.9	32.5	.221	.157	.219
SI10	183	36	33	10.0	41.4	46.8	42.5	.363	.230	.338
SI11	134	18	22	6.9	25.0	45.5	57.2	.221	.157	.219
SI12	254	33	40	3.0	47.6	45.1	41.6	.221	.157	.219
SI14**	362	68	43	4.4	40.0	30.9	28.9	.183	.157	.192
SI19	101	30	30	4.0	68.1	45.2	53.0	.221	.157	.219

>* Includes partial of SI14 & SI15
<* Includes SI20
** Includes balance of SI14 & SI15

ATTACHMENT II

2.0 SEISMIC ANCHOR MOVEMENT (SAM) ADJACENT STRUCTURE ISSUES - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX C, ITEMS 2a, AND 2b

The NRC IEB 79-14 follow-up review identified several items related to the project approach to addressing seismic anchor movements (SAMs). The open items are summarized here along with conclusions reached:

2.1 EVALUATION OF SAMs BETWEEN BUILDINGS - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX C, ITEM 2a

G/C Report No. 2439 (Ref. 1), states that absolute displacements between adjacent buildings of less than 1/8 inch can be neglected. The NRC inspection team recognized a general industry practice of excluding movements of less than 1/16 inch, but was unwilling to accept movements up to 1/8 inch without a quantitative technical basis.

During the inspection, SCE&G presented a paper to the NRC justifying 1/8 inch as an acceptable threshold for considering SAMs. This paper focused primarily on industry data supporting the fact that seismically qualified piping systems do not fail under inertia loadings and small seismic anchor movements. The NRC did not accept this position as sufficient justification, and requested a quantitative technical basis.

2.1.1 CONCLUSIONS OF SAMs BETWEEN BUILDINGS

The 1/8 inch threshold for considering building SAMs was shown to be acceptable based on the following:

(a) A fatigue evaluation accounting for the 1/8 inch SAM in combination with a 1/8 inch growth of Containment shows ample margin with respect to fatigue life. The evaluation is documented in Attachment II, Appendix A.

(b) A review of piping attached to essential equipment reveals that relevant SAMs were included in the analysis of the piping system and found to produce no unwanted reactions at the equipment. This review is documented in Attachment II, Appendix B.

(c) A review of rigid piping crossing building boundaries shows that pipe stresses and support reactions are acceptable for SAMs in the range of 1/16 inch to 1/8 inch. The review is documented in Attachment II, Appendix B.

2.2 JUSTIFICATION FOR COMBINING THE DIRECTIONAL RESPONSES OF ADJACENT STRUCTURE SAMs BY THE SRSS APPROACH - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX C, ITEM 2b

In Section 6.3.4-C of G/C Report No. 2439 (Ref. 1), the effects of adjacent structure SAM movements in the three global directions were combined by square root of the sum of the squares (SRSS) method rather than by absolute sum. During the inspection, SCE&G noted that modal frequencies and dominant modes differ for different structures and, therefore, the likelihood of simultaneous maximum opposite seismic displacements was very low, which justifies the SRSS combination method. The NRC inspection team felt that this justification was not adequate to address the issue.

The FSAR (Ref. 4) does not address requirements for combining SAMs between buildings. At the time of design, construction, and plant licensing, combining SAMs by the SRSS method was an industry accepted standard and, therefore, no documentation of approach was necessary.

NUREG 1061 (Ref. 12) addresses the methods of combining directional responses from peak group displacements occurring at the same moment. NUREG 1061 states these peak displacements, when combined by the absolute sum method, will combine the responses in a most unfavorable manner to calculate SAM components of the seismic response.

NUREG 1061 recommends the combination of SAM directional responses by the SRSS method.

The following provides additional technical justification for the SRSS approach for combining SAMs directional responses. Structural responses resulting from earthquake input are a function of earthquake characteristics. Due to the randomness of earthquake motions, random vibration theory is used to derive response combinations.

When two adjacent structures are subjected to seismic input, their relative motion can be calculated using the random vibration theory (Ref. 5). Assuming that the structural response is represented by a normal random process (or Gaussian random process), the variance of the relative motion is the sum of the variances of two adjacent structural motions. Variance is defined as the square of the standard deviation from the mean. Therefore, the relative motion between buildings can be represented in equation form as follows:

$$\sigma^2 = \sigma_1^2 + \sigma_2^2$$

where σ is the standard deviation of relative motion, and σ_1 and σ_2 are the standard deviations of two adjacent structural motions. Letting the maximum probable responses be proportional to the standard deviations, or

$$Q = C\sigma$$
$$Q_1 = C\sigma_1$$
$$Q_2 = C\sigma_2$$

we have

$$Q = (Q_1^2 + Q_2^2)^{1/2}$$

which means the maximum probable relative motion is the SRSS of the maximum probable responses of two adjacent structures. The above derivation can be the basis of the SRSS combination of the directional effects of SAMs between buildings.

A deterministic time history calculation (Ref. 14) was made to verify the above derivation. Two single degree of freedom oscillators were used to simulate the Reactor Building and Intermediate Building. A direct integration program was used to evaluate the responses of the two oscillators and differential displacements were compared on a time-by-time basis. The results of this analysis revealed a maximum relative displacement of 0.257 inch. An SRSS of the individual structural motions yielded a value of 0.260 inch. This supports the conclusion that the SRSS method of combining the directional response of SAMs between buildings yields acceptable results.

2.2.1 CONCLUSIONS ON COMBINING THE DIRECTIONAL RESPONSE OF ADJACENT STRUCTURE SAMs BY THE SRSS APPROACH

This discussion shows that combining the directional response of adjacent structure SAMs by the SRSS approach yields acceptable results when compared to results from a deterministic time history calculation.

FATIGUE EVALUATION OF 1/8" SAM THRESHOLD

ASME B & PV Code, Section III permits the evaluation of seismic anchor movements in accordance with NC-3652.3 - Thermal Expansion, as follows:

a. The effects of thermal expansion must meet the requirements of Equation (10).

$$\frac{iM_c}{Z} \le S_A \qquad (10)$$

where:

Z = section modulus of pipe

i = stress intensification factor

M_c = range of resultant moments due to thermal expansion. Also include moment effects of anchor displacement due to earthquake if anchor displacement effects were omitted from Eq. (9) (NC-3652.2).

S_A = allowable stress range for expansion stresses

Because of the cyclic nature of these stresses, they affect the fatigue life of the piping, and the Code provides for this by prescribing the evaluation of the stress range and by placing limits on both the number of cycles of stress and the stress range. This concept goes back to the very early days of development of the Code for Pressure Piping - ASA B31.1.

In the late 1940's and early 1950's, A.R.C. Markl, then Chief Research Engineer of Tube Turns Inc., conducted full-scale fatigue tests on pipe and components such as tees, elbows and mitered joints. Markl's tests resulted in the formulation and use of stress intensification factors which remain in Code use today. As a means of correlating fatigue data accumulated by test, Markl (Ref. 7) developed the following relationship between intensification factor, stress and number of stress cycles to failure:

$$iS = CN^{-0.2}$$

where:

i	=	stress intensification factor
S	=	nominal stress (one-half applied range of stresses)
N	=	number of stress reversals to failure
C	=	material constant (245,000 for carbon steel)

The term i S in Markl's equation is analogous to one-half the intensified stress range computed in Code Equation 10.
Therefore,

$$\frac{S_A}{2} = CN^{-0.2}$$

or

$$S_A = 2CN^{-0.2}$$

This relationship can be used to predict the fatigue life of piping under postulated conditions of stress range.

To make a limiting comparison, assume that in a given piping system, thermal expansion loads (moments) combined with 1/16 inch Seismic Anchor Movements (SAM) are evaluated using equation 10 and found to produce a stress level equal to S_A, the maximum combined allowable. Assume also that each load (thermal and SAM) contributes 50% of the total. Then if an additional 1/16 inch SAM is added (corresponding to 1/8 inch total SAM), we would expect the stress level to be 1.5 S_A. Finally, to completely envelope the conditions in question, a 1/8 inch displacement is added to account for containment thermal

growth, increasing stress level by another 1.0 S_A. This would bring the total equation 10 stress range to 2.5 S_A.

For A-106B Material

$$2.5\,S_A = 2.5\,(1.25\,S_c + .25\,S_h) = 2.5\,(1.25 \times 15{,}000 + .25 \times 15{,}000)$$
$$= 56{,}250 \text{ psi}$$

Using this value in Markl's equation and solving for N, we get

$$56{,}250 \text{ psi} = 2 \times 245{,}000\,N^{-0.2}$$

and N = 50,000 cycles to failure.

For V. C. Summer, the principal design basis cyclic conditions are as follows:

SAM - 20 seismic events @ 20 cycles = 400
Heatup/cooldown cycles (main steam) = 200
Containment thermal cycles = 40*

*Based upon a 12 month refueling cycle

Conservatively assuming <u>all</u> of the above occur 400 times, we calculate a safety factor of

$$\frac{50{,}000 \textit{ cycles to failure}}{400 \textit{ design basis, conservative}} = 125 \text{ Safety Factor}$$

Thus, there is a considerable margin to failure.

ESSENTIAL EQUIPMENT AND PIPING REVIEW FOR SAMs

The evaluation of the effect of a 1/8 inch threshold for inclusion of seismic anchor movements (SAMs) in piping stress analysis is addressed in three areas:

- The influence of SAM loads as they may impact essential equipment (equipment required for safe shutdown or mitigation of a design basis accident).
- The effect of SAM on piping systems crossing building boundaries.
- Rigid piping subject to SAM approaching the 1/8 inch threshold.

In the first case, the essential equipment list (Ref. 8) was used to identify equipment with connected piping that could be affected by piping imposed SAM loads, possibly impairing the item's safety-related function. Table 1 summarizes the components examined and the conclusions reached regarding SAM effects. Several conclusions were reached:

1. SAMs were included in the analysis of the piping system and found to produce no unwanted reactions at the equipment.
2. SAMs were of inconsequential magnitude because all connected piping was supported from the same building/structure.
3. SAM loads generated at building interfaces could not reach the equipment due to the isolating effect of multiple restraints between the building boundaries and the equipment.

In the second case, piping crossing building boundaries can experience significant increases in pipe stress and support reactions due to SAMs. A review of piping system analyses and piping drawings was conducted to identify lines susceptible to these effects. The review is summarized in Table 2. Where relative displacements exceed 1/8 inch, the movements were included in the piping analysis. Where piping runs between anchored equipment in the same structure, SAMs have little or no effect on the piping.

Finally, several lines were selected for analysis to confirm that SAMs up to 1/8 inch can be safely ignored, provided the support/restraint system is conceived to

accommodate the expected movements. In each case, the line selected for analysis had the following attributes:

1. Rigid piping (i.e., large diameter, heavy wall pipe).
2. Relatively high operating temperature, assuring that a representative position of the Section III, equation 10 stress allowable would be consumed by thermal expansion.
3. Routing of the piping exposes it to SAMs at or near the threshold of 1/8 inch.

The analyzed lines are described below:

- 32 inch Main Steam Line from Reactor Building Penetration No. 428 into the Auxiliary Building.
- 18 inch Feedwater Line from Reactor Building Penetration No. 206 to Intermediate Building Header.
- 12 inch RHR Line from Reactor Building Penetration No. 226 to RHR Pump B.

The results of the analyses (Ref. 17) are summarized in Table 3. The results indicate that 1/8 inch SAMs can generally be accommodated without increasing piping stress levels in excess of code prescribed limits. Where SAMs at the 1/8 inch threshold produce high support loads or pipe stresses, the inclusion of support stiffness and snubber deadband in the analysis significantly reduces the magnitude as demonstrated in the RHR evaluation.

<u>ATTACHMENT II</u>
APPENDIX B
TABLE 1
EVALUATION OF ESSENTIAL COMPONENTS FOR SAMs

EQUIPMENT TAG NO.	DESCRIPTION	LOCATION	LINE/FUNCTION	SAM EVALUATION
XAJ-15	Boric Acid Blender	Aux. Bldg. El. 436	Inlet Line Outlet Line	SAM ≈ 0 - All piping within same bldg.
XFL-8A,B	Seal Water Injection Filter	Aux. Bldg. El. 452	Inlet Line Outlet Line	SAM ≈ 0 - All piping within same bldg. SAM < 1/8" - Filter isolated by multiple restraints in same bldg.
XFL-14	Boric Acid Filter	Aux. Bldg. El. 463	Inlet Line Outlet Line	SAM ≈ 0 - All piping within same bldg.
XFL-39	Seal Water Return Filter	Aux. Bldg. El. 452	Inlet Line Outlet Line	SAM < 1/8" - Filter isolated by multiple restraints in same bldg. SAM ≈ 0 - All piping in same bldg.
XHE-11	Seal Water Heat Exchanger	Aux. Bldg. El. 412	Inlet Line Outlet Line	SAM < 1/8" - Heat exchanger isolated by multiple restraints in same bldg.
XHE-3	Regenerative Heat Exchanger	Reactor Bldg. El. 415	Pen. #409 to HXR HXR to Pen. #318 R.C. Loop to HXR HXR to R.C. Loop	SAM < 1/16" SAM < 1/16" R.C. Loop movements considered in analysis R.C. Loop movements considered in analysis
XHE-5A,B	RHR Heat Exchanger	Aux. Bldg. El. 412	From HXR to Pumps From HXRs to Pen. 322, 325, 227 Comp. Clg. Wtr. Lines	SAM ≈ 0 - All piping within same structure SAM < 1/8" - Heat exchanger isolated by multiple restraints in same bldg. SAM < 1/8" - Heat exchanger isolated by multiple restraints in same bldg.
XHE-2A,B	Component Cooling Heat Exchanger	Int. Bldg. Evev. 412	From HXR to Pump From Pumps to Int/Aux. Bldg.	SAM ≈ 0 - All piping within same structure analyzed
XTK-3	Component Cooling Surge Tank	Aux. Bldg. El. 463	From Tank to Pump Suction Lines	SAM < 1/8" - Tank isolated by multiple restraints in same bldg.
XTK-12A, B	Boric Acid Tanks	Aux. Bldg. El. 463	All	SAM ≈ 0 - All piping within same structure
XTK-20A, B	Fuel Oil Day Tank - Diesel Gen.	Diesel Gen. Bldg.	All	SAM < 1/8" - All piping in same structure

TABLE 1
EVALUATION OF ESSENTIAL COMPONENTS FOR SAMs

EQUIPMENT TAG NO.	DESCRIPTION	LOCATION	LINE/FUNCTION	SAM EVALUATION
XTK-8	Condensate Storage Tank	Outdoors	Suction Line to Emerg. F.W. Pump	Analyzed as buried piping
XTK-53A, B	Fuel Oil Storage Tank - Diesel Gen.	Outdoors	Suction to Fuel Oil Transfer Pumps	Analyzed as buried piping
XTK-25	Refueling Water Storage Tank	Aux.Bldg.	All	Tank isolated by multiple restraints in the same structure
XTK-60	Sodium Hydroxide Storage Tanks	Aux. Bldg.	All	Tank isolated by multiple restraints in the same structure
XTK-9A, B, C, D	Tank - Air Receiver - Diesel Gen.	D.G. Bldg.	All	SAM ≈ 0 - All piping within same structure
XTK-113, A, [illegible]	Expansion Tank - Chilled Water	Int. Bldg. El. 412	All	SAM ≈ 0 - All piping within same structure
XPP-[illegible]A, B, [illegible]	Service Water Pumps	S.W. Pump House	Pump Discharge to Int. Bldg.	Analyzed (buried piping)
XPP-45A,B	Service Water Booster Pumps	Int. Bldg. El. 412	Suction Lines Discharge Lines	Analyzed (buried piping) SAM < 1/16"
XPP-1A, B, C	Component Cooling Water Pumps	Int. Bldg. El. 412	Suction Lines from Heat Exchanger Discharge Lines	SAM = 0, All piping in same structure Analyzed
XPP-21A, B and XPP-8	Motor Driven Emerg. F.W. Pumps Turbine Driven Emerg. FW Pump Turbine	Int. Bldg. El. 412 Int. Bldg. El. 412 Int. Bldg. El. 412	Discharge Lines Suction Lines Steam	Analyzed Analyzed SAM < 1/16" - All piping in same bldg.
XPP-38A, B	Reator Bldg. Spray Pumps	Aux. Bldg. El. 374	Discharge Lines Suction Lines	Analyzed SAM ≈ 0 - All piping within same structure
XPP-31A, B	RHR Pumps	Aux. Bldg. El. 374	Suction Lines Discharge Lines to HXRs	SAM < 1/8" isolated by multiple restraints: 10 or more SAM ≈ 0 - All piping within same structure

ATTACHMENT II
APPENDIX B
TABLE 1
EVALUATION OF ESSENTIAL COMPONENTS FOR SAMs

EQUIPMENT TAG NO.	DESCRIPTION	LOCATION	LINE/FUNCTION	SAM EVALUATION
XPP-13A, B	Boric Acid Pumps	Aux. Bldg. El. 436	Suction Lines Discharge Lines	SAM ≈ 0 - All piping within same structure
XPP-43A, B, C	Charging/S.I. Pumps	Aux. Bldg. El. 388	Suction Lines Discharge Lines	SAM < 1/8" - Pumps isolated by multiple restraints in same structure SAM < 1/8" - Pumps isolated by multiple restraints in same structure
XPP-48A, B, C	Chilled Water Pumps	Aux. Bldg. El. 412	Suction Lines Discharge Lines	SAM < 1/8" - Pumps isolated by multiple restraints in same structure SAM < 1/8" - Lines connect to chillers in same structure
XPP-4A, B - 141A, B	Diesel Gen. Fuel Oil Transfer Pumps	D. Gen. Bldg. El. 427	Suction and Discharge Lines	SAM ≈ 0 - All piping in same structure - except buired suction lines which were analyzed
XPP-44A, B, C	Svc. Wtr. Screen Wash Pumps	S.W. Pump House	Suction and Discharge Lines	SAM ≈ 0 - All piping in same structure
XP-30A, B, C	Reactor Coolant Pumps	Reactor Bldg.	Various	SAM effects on major reactor coolant system components and piping were considered in the Class I analyses
XSG-2A, B, C	Steam Generator	Reactor Bldg.	Various	SAM effects on major reactor coolant system components and piping were considered in the Class I analyses
XTK-24	Pressurizer	Reactor Bldg.	Various	SAM effects on major reactor coolant system components and piping were considered in the Class I analyses
XTK-28A, B, C	S.I. Accumulator	Reactor Bldg.	Various	SAM effects on major reactor coolant system components and piping were considered in the Class I analyses
XAA-1A, B, 2A, B	R.B. Cooling Units	Reactor Bldg.	Cooling Water Supply and Return	SAM ≈ 0 - All piping supported from the same structure

TABLE 1
EVALUATION OF ESSENTIAL COMPONENTS FOR SAMs

EQUIPMENT TAG NO.	DESCRIPTION	LOCATION	LINE/FUNCTION	SAM EVALUATION
XAH-1A, B & 2	Cooling Units - S.I./Chg. Pump Rooms	Aux. Bldg. El. 388	Chilled Water Supply/Return	Cooling units isolated by multiple restraints in the same structure
XAH-4A, B	Cooling Units - RHR/Spray Pump Rooms	Aux. Bldg. El. 374	Chilled Water Supply/Return	Cooling units isolated by multiple restraints in the same structure
XAH-6, 8	ESF Swg. R.I. Cooling Units	Int. Bldg. El. 451	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure
XAH-11A, B	Cooling Units - Emerg. F.W. Pump Area	Int. Bldg. El. 412	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure
XAH-12A, B	Cooling Units - Control Room	Int. Bldg. El. 482	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure
XAH-13A, B	Cooling Units - Relay Room	Int. Bldg. El. 482	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure
XAH-19A, B	Cooling Units - Speed Switch Room	Int. Bldg. El. 436	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure
XAH-42A, B	Cooling Units - Battery Room	Int. Bldg. El. 412	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure
XAH-32, 33	Cooling Units - Aux. Bldg. Motor Control Centers	Aux. Bldg. El. 412 and 463	Chilled Water Supply/Return	Cooling Units isolated by multiple restraints in the same bldg.
XAH-9A, B	Cooling Units - Svc. Water Booster Pump Area	Int. Bldg. El. 426	Chilled Water Supply/Return	SAM < 1/8" - All piping within the same structure

ATTACHMENT II
APPENDIX B
Table 2
EVALUATION OF PIPING SYSTEMS FOR SAMs

PIPING SYSTEM	FROM	TO	SAM CONSIDERED?	COMMENT
Main Steam	Steam Gen.	Penetration	Yes	None
Main Steam	Penetration	Aux./Int. Bldg.	No., < 1/8"	Sample analysis demonstrates SAM ≦ 1/8" acceptable
Feedwater	Penetration	Interior Concrete	No, < 1/8"	None
	Interior Concrete	Stm. Gen.	Yes (W)	None
	Int. Bldg.	Penetration	No, < 1/8"	Sample analysis demonstrates SAM ≦ 1/8" acceptable
Emergency FW	Int. Bldg.	Penetrations	Yes	None
	Penetrations	Steam Gen.	Yes	None
	Condensate Str. Tk.	Pumps	Yes	None
Service Water	S.W. Booster Pumps	Aux. Bldg.	No, < 1/8"	None
	Penentrations	Aux. Bldg.	Yes	None
	Int. Bldg.	Aux. Bldg.	Yes	None
	Aux. Bldg.	Fuel Handling Bldg.	Yes	None
Reactor Coolant	Pressurizer	Interior Concrete	No, < 1/8"	None
	RTD Manifold	R.C. Loop	Yes	None
	R.C. Loop	Pressurizer	Yes	None
Component Cooling	Int. Bldg.	Aux. Bldg.	Yes	None
	Aux. Bldg.	Penetrations	Yes	None
	Penetrations	Interior Concrete	No, < 1/8"	None
	Interior Concrete	R.C. Pumps	Yes	None
RHR	R.C. Loop	Interior Concrete	Yes	None
	Int. Concrete	Penetrations	No, < 1/8"	None
	Penetrations	Aux. Bldg./Int. Bldg.	No, < 1/8"	Sample analysis demonstrates SAM ≦ 1/8" acceptable
Spent Fuel Cooling	Spent Fuel Bldg.	Aux. Bldg.	Yes	None

ATTACHMENT II
APPENDIX B
Table 2
EVALUATION OF PIPING SYSTEMS FOR SAMs

PIPING SYSTEM	FROM	TO	SAM CONSIDERED?	COMMENT
Reactor Bldg. Spray	Aux. Bldg.	Penetrations	Yes	None
	R. Bldg. Sump	Aux. Bldg.	No, < 1/8"	None
	RWST	Pumps	No	RWST and pumps in same structure
	Penetrations	Spray Rings	No	All supported from R. Bldg.
Chemical and Volume Control and Safety Injection	Penetrations	Interior concrete	No, < 1/8"	None
	Int. Concrete	R.C. Piping	Yes	None
	RWST	Charging Pumps	No	All supported from Aux. Bldg.
	Int. Bldg.	Aux. Bldg.	No, < 1/8"	None
	Int./Aux. Bldg.	Penetrations	No, < 1/8"	None
Steam Gen. Blowdown	Steam. Gen.	Interior Concrete	No, < 1/8"	None
	Int. Concrete	Penetrations	No, < 1/8"	None

ATTACHMENT II
APPENDIX B
TABLE 3
SUMMARY OF RESULTS

LINE	SAM	SUPPORT STIFFNESS INCLUDED	SNUBBER DEADBAND INCLUDED	MAX. SAM STRESS	MAX. EQUATION 10 STRESS
MS	$\Delta X = 0.064''$ $\Delta Y = 0.0474''$ $\Delta Z = 0.1228''$	No	0	1 KSI	16.6 KSI
FW	$\Delta X = 0.1087''$ $\Delta Y = 0.0476''$ $\Delta Z = 0.0655''$	No	0	2.7 KSI	3.6 KSI
RHR RB/IB	$\Delta X = 0.1181''$ $\Delta Y = 0.0276''$ $\Delta Z = 0.066''$	Yes	1/16"	2.8 KSI**	6.4 KSI
RHR AB/IB	$\Delta X = 0.1379''$ $\Delta Y = 0.0311''$ $\Delta Z = 0.1011''$	Yes	1/16"	0.6 KSI*	1.8 KSI

RB/IB = Reactor Bldg. to Intermediate Bldg.
AB/IB = Auxiliary Bldg. to Intermediate Bldg.
* Node 35
** Node 1 or 5

3.0 CONTAINMENT GROWTH ISSUES - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX C, ITEM 3

The NRC IEB 79-14 follow-up review identified that containment penetration movements were not considered in the piping analyses for the effects of post-accident pressurization or steady state temperature growth. During the inspection, SCE&G advised the NRC that post-accident pressurization is not an issue because only primary stress equations apply. Also, SCE&G responded that steady state thermal growth need not be considered because the maximum value is close to 1/8 inch which is considered negligible. The NRC accepted the response on pipe stresses associated with post-accident pressurization, but requested quantitative back-up for not considering pipe stresses associated with steady state thermal growth of containment. The NRC also requested a basis for not considering increased support reactions for both post-accident pressurization and steady state thermal growth of containment.

3.1 EVALUATION OF CONTAINMENT GROWTH ISSUES

Each of these issues are summarized below along with the conclusion reached:

3.1.1 Thermal Growth of Containment

Calculations were performed to predict the maximum thermal growth of the Containment Building under winter and summer start-up and shut-down conditions (Ref 18). The basis for the calculations is the thermal gradients presented in the FSAR (Ref. 4), Figure 3.8-23.

Two methods were used in performing the calculations. The first method involved a hand calculation approach using classical elastic equations developed by Hetenyi (Ref. 9). The second approach involves the ratioing of results from the original Kalnins (Ref. 10) shell analysis. Correlation of the two results was close, and the values to be used in the evaluation of

piping attached to containment are summarized in Attachment III, Appendix A.

It was also concluded and documented in the calculation that thermal growth of containment due to a LOCA is negligible. This is because of the short duration of the temperature spikes and the small difference in the mean temperature.

3.1.2 Growth of Containment Due to LOCA Pressure

The growth of containment due to LOCA pressure is based on results from the structural acceptance test. These results for both 30 psi and 65.6 psi are presented in Attachment III, Appendix B. Movements for a LOCA pressure were obtained by ratioing results from the Structural Acceptance Test (SAT) based on the maximum pressure for a LOCA (46 psi).

The maximum measured displacements for the SAT were 0.213 inch radial (@ Az 162°30', mid-cylinder) and 0.197 inch vertical at the ring girder (say 0.10 inch at mid-height). A review of the penetration layout drawings shows there are no penetrations within 40 feet of this location. The largest radial movement at a penetration is 0.18 inch. Using the 0.18 inch value for radial growth and 0.10 inch for vertical growth, the estimated radial and vertical displacements for a LOCA, are computed as follows:

$$\frac{46}{65.6}(0.18\ inch) = 0.126\ inch\ radial$$

$$\frac{46}{65.6}(0.10\ inch) = 0.07\ inch\ vertical$$

The larger of these displacements, which is in the radial direction, is approximately 1/8 inch.

3.1.3 Justification for Not Considering the Effect of LOCA Containment Growth

As stated above, containment growth due to LOCA temperature effects is negligible due to the short duration of the temperature spikes and the small difference in the mean temperatures. Therefore, there is no need to consider containment growth due to this aspect of a LOCA. Additionally, this also holds true for a main steam line break.

As discussed earlier, LOCA pressure can produce displacements in the range of 1/8 inch. The basis for not having to evaluate for these terminal end movements is that LOCA is considered a faulted plant condition and only primary stress equations apply per design specifications and the FSAR. Additional confidence that the secondary stress produced by LOCA pressure displacements will not create operability problems or cause loss of pressure boundary is the significant margin demonstrated by the Markl evaluation found in Attachment II, Appendix A. Because of conservation in our original LOCA analysts peak pressure, a main steam line break is also bounded by this examination.

3.1.4 Effects of Steady State Temperature Growth Coupled with 1/8 Inch SAM

The discussion in Attachment II, Appendix A, related to SAMs includes thermal growth of containment. As stated in this section, the cyclic growth of containment affects the fatigue life of the attached piping, and this can be addressed by using equations developed by Markl (Ref. 7). The results of an evaluation documented in Attachment II, Appendix A, show that ample margin exists for the combined movement due to a 1/8 inch SAM and steady state thermal growth of containment.

3.1.5 Evaluation of Support Reactions

As discussed earlier, the steady state thermal growth of containment will impose stresses in the attached piping. Justification for these piping stresses being acceptable has been addressed. Also, containment growth due to LOCA pressure will create increased pipe stresses which do not have to be considered because they are associated with a faulted condition. Finally, both the steady state thermal growth and LOCA pressure growth of containment influence support reactions. An evaluation of these increased support reactions is documented in this section.

Although the FSAR (Ref. 4) and project criteria do not specifically address these types of support loads which are considered secondary, the ASME Code of record does provide guidance. Section NF-3221.1 of the Winter 1973 Addenda of the 1971 Code provides the direction. Paragraph (2) states:

"The allowable stress for the combined mechanical loads and effects which result from constraint of free end displacements (defined in NB-3213.20), but not thermal or peak stresses, shall be lifted to three times the stress limits of NB-3221.1(a)(1). The use of this stress limit presupposes the use of linear elastic analysis. Constrained free end displacement and differential support motion effects need not be considered for the emergency condition."

The stress limit referred to in NB-3221.1(a)(1) is S_m. Going to a 3 S_m value is equivalent to designing to a value approximately equal to the ultimate stress, which for A36 steel is 58 ksi.

The code section referenced above provides clear direction that the support loading component due to containment growth from LOCA pressure does not have to be considered, since it is an emergency condition. Going to the 3 S_m value provides

ample margin for any increased support loads due to steady state thermal growth of containment. For bending, a value of 0.66 fy is used for an allowable stress (Ref. 11). This equates to 23.8 ksi for A36 steel. Assuming a support is fully stressed with other loading components, a margin of 34.2 ksi is available for the secondary loading components being addressed here. It is highly unlikely that increased support stresses due to these secondary loadings would exceed the remaining margin of 34.2 psi for A36 steel.

3.2 CONCLUSIONS ON CONTAINMENT GROWTH

The following conclusions are reached relative to containment growth issues.

(1) Post-accident pressurization need not be considered for pipe stress, since this is a plant faulted condition and only primary stress equations apply.

(2) Ample fatigue margin exists to sustain SAM movements combined with steady state thermal growth of containment on the attached piping. The fatigue evaluation is shown in Attachment II, Appendix A.

(3) The code of record does not require the evaluation of free end displacement effects on supports for an emergency condition, and therefore increased support reactions due to post-accident pressurization of containment are not required to be considered.

(4) The code of record requirement of specifying a 3 S_m limit for secondary stresses provides ample margin for increased support stresses due to steady state thermal growth of containment.

CONTAINMENT SHELL

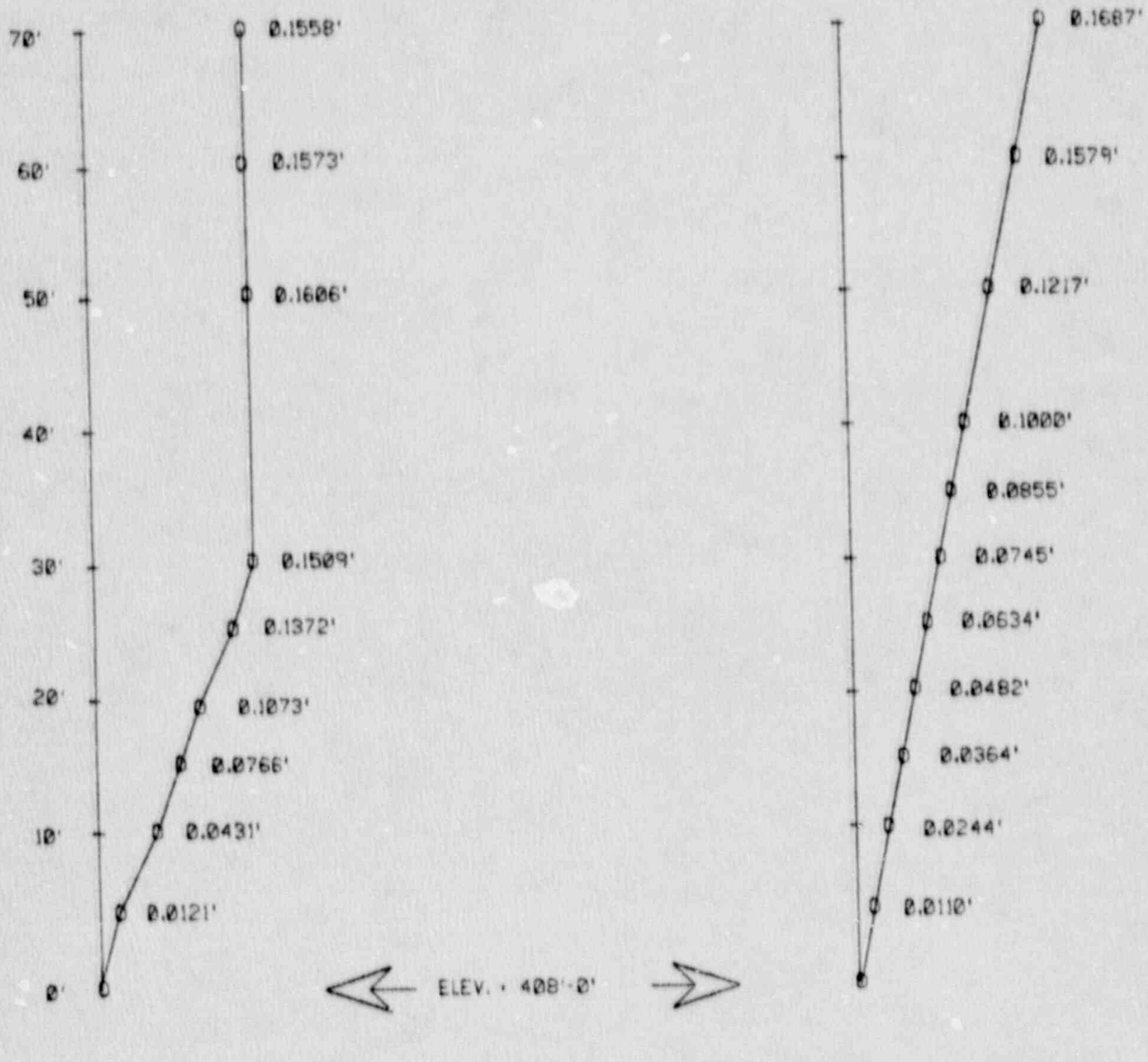
DISPLACEMENTS UNDER STEADY STATE THERMAL LOADS

(CONTROLLING CONDITION - WINTER START-UP & SHUT-DOWN)

SOURCE REFS: FSAR FIGURE 3.8-23

EXISTING CALCS 1.13.4 REV Ø & COMPUTER OUTPUT

CALCS 1.13.5.1 REV Ø



RADIAL DISPLACEMENTS
$\Delta T = 36°F$

VERTICAL DISPLACEMENTS
$\Delta T = 36°F$

CONTAINMENT SHELL
DISPLACEMENTS UNDER LOCA PRESSURE

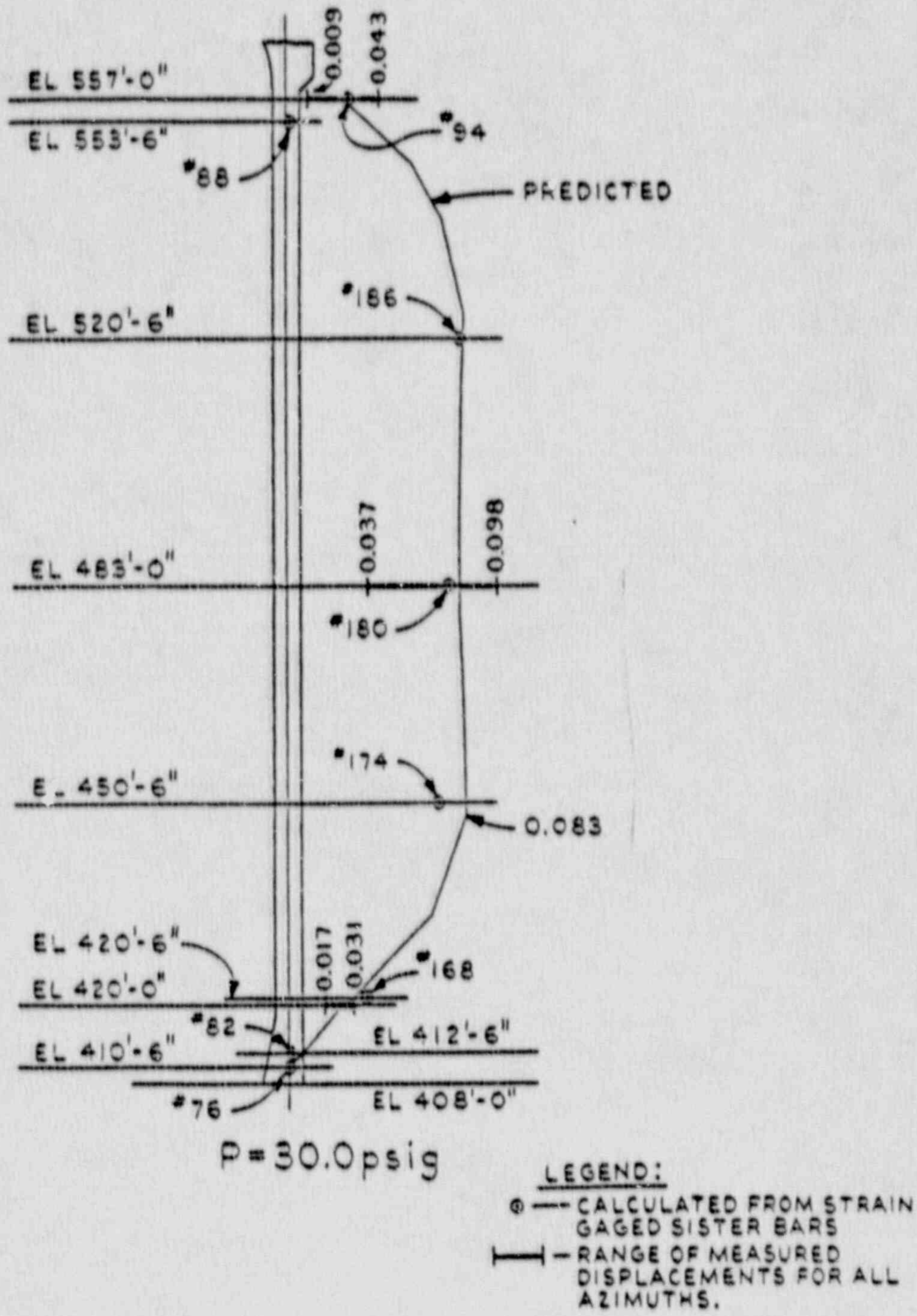


FIGURE 4
Displaced Wall Configuration
at 30 PSIG

CONTAINMENT SHELL
DISPLACEMENTS UNDER LOCA PRESSURE

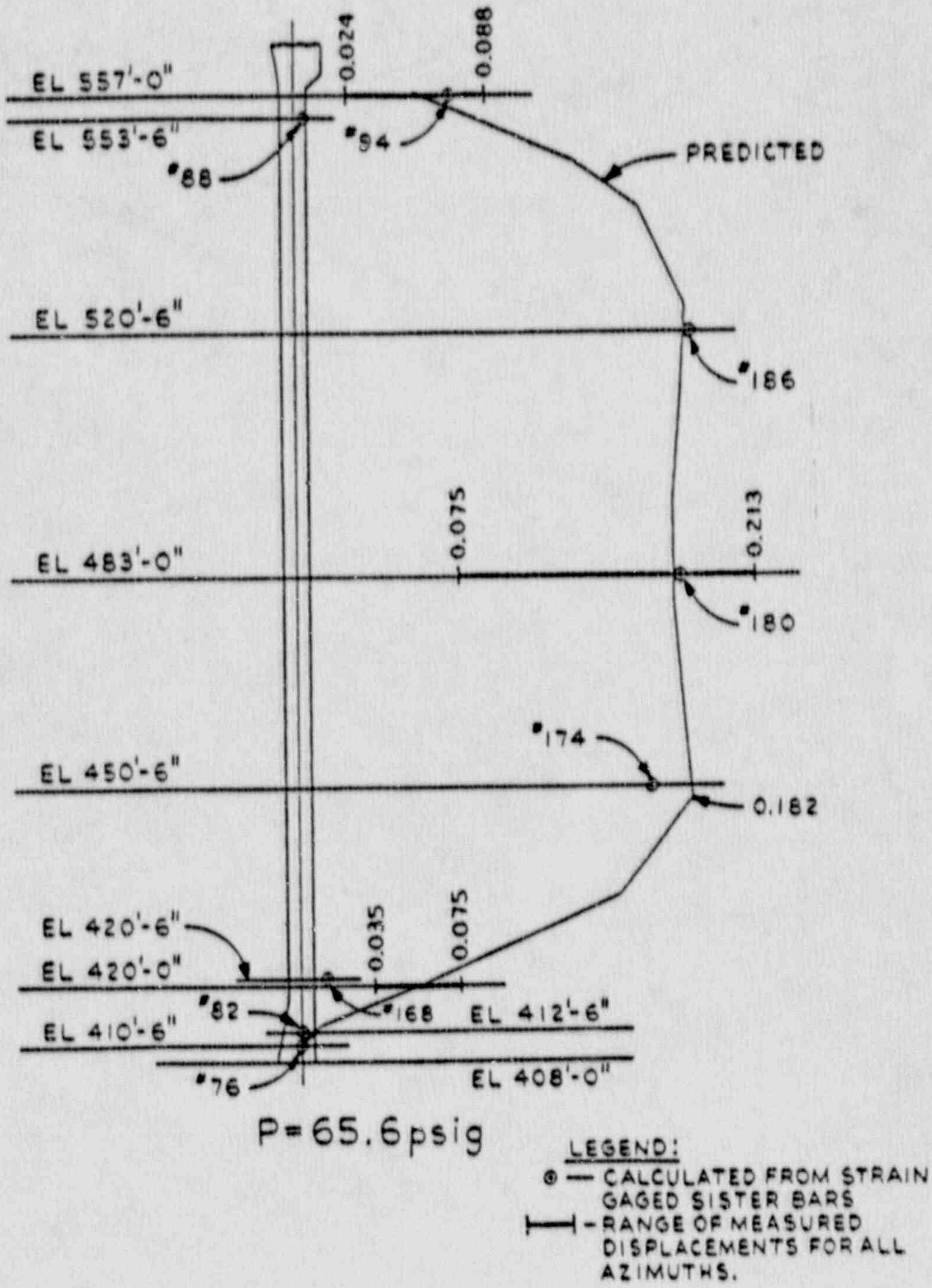


FIGURE 5
Displaced Wall Configuration
at 65.6 PSIG

4.0 DECOUPLING CRITERIA - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX C, ITEM 4

The NRC IEB 79-14 follow-up review questioned the acceptability of the decoupling ratio used on V. C. Summer Nuclear Station piping. In accordance with page 6-20 of G/C Report #2439 (Ref. 1), branch lines and instrument connections may be decoupled from the analysis model of the main pipe system provided the moment of inertia ratio of the two lines is equal to or less than 15 percent. The inspection team questioned this engineering assumption because they believe the industry practice is to use a decoupling ratio in the range of 6 percent. The NRC concern in using the larger decoupling ratio is that piping stresses and pipe support loads may be under-predicted.

4.1 DECOUPLING EVALUATION

The decoupling criteria for V. C. Summer Nuclear Station is addressed in the following two sections. Section 1 documents a review made of all analysis isometrics to determine the extent of decoupling. The number of decoupled situations are noted as well as the number in the range of 6 percent to 15 percent. Section 2 discusses two studies made of decoupled lines in the range of 6-15 percent. Conclusions are reached regarding the acceptability of the decoupling ratio and associated criteria stated in G/C Report #2439 (Ref. 1).

4.1.1 Survey of analysis isometrics

A review made of all analysis isometrics and D-306-series flow drawings (Ref. 3) revealed 912 locations where lines were decoupled. Of this number, 571 involve cases where the run pipe is safety related. The decoupled line includes any of the following: another rigorous line, a criteria line, a vent or a drain.

Of the total number of safety-related decoupled lines, two were found to have decoupling ratios in the range of 6% - 15% based on I_{Branch}/I_{Run} (I = moment of inertia). These are listed here:

Run Line	Branch Line	Run Line Size (inches)	Branch Line Size (inches)	I_{Branch}/I_{Run}
CC-07	CC-12	6	3	10.74%
CC-08	CC-09	8	4	9.97%

4.1.2 Analyses of Two Decoupled Lines

The analyses of the two decoupled situations noted above are documented in calculation NRC7914-04 (Ref 19). In both cases, the problems were evaluated on the PC based analysis program CAEPIPE by looking first at the lines in a coupled mode; i.e., the branch line modeled with a significant portion of the run line. The lines were then decoupled and run separately following the criteria given in Section 6.2.6 of G/C Report 2439 (Ref. 1).

After all runs were made, a comparison was made of pipe stress and support reactions for the combined and decoupled models. The following conclusions can be drawn based on the comparative analyses:

a. In both cases, run line stresses in the combined model are within 10% of those for the decoupled case.

b. In both cases, support reactions on the run line correlate closely between the coupled and decoupled cases.

c. On the branch lines, some seismic stress increases were noted. However, all stresses are within code allowables.

d. Likewise with support reactions, some increases were noted on branch line supports, primarily due to higher seismic components. However, all support reactions were found to be acceptable.

4.2 CONCLUSIONS ON DECOUPLING

The results show that pipe stresses and support reactions are acceptable for both the run lines and the branch lines. Therefore, the two identified decoupled lines with ratios greater than 6% are acceptable and supporting basis is established for the project decoupling ratio of 15%.

5.0 PRYTEN COMPUTER PROGRAM QUALIFICATION - RESPONSE TO NRC INSPECTION 50-395/89-200, APPENDIX A, OPEN ITEM 50-395/89-200-05

An open item from the NRC IEB 79-02/79-14 follow-up inspection questioned the acceptability of results from G/C's base plate evaluation program, PRYTEN. In particular, a concern was raised over the drastic increase in bolt forces as the bolt stiffness was increased from 10 kips/in to 3390 kips/in on a very thin sample plate. This could not be explained during the audit and was noted as a finding.

5.1 PRYTEN COMPUTER PROGRAM EVALUATION

Calculation NRC7902-01 (Ref. 20) was prepared to demonstrate the acceptability of PRYTEN results in the range of program usage. It was realized after the NRC inspection that the sample results viewed as being anomalies occurred for two conditions which would not be encountered under normal program usage. That is, the base plate stress exceeded the allowable stress by a factor of 10 (sample results in the range of 330-440 psi; allowable is 27 psi) and bolt forces exceeded the bolt preload. The referenced calculation evaluates the program within the range of normal program usage as defined by bolt and base plate allowables.

5.2 CONCLUSIONS ON PRYTEN COMPUTER PROGRAM

The calculation demonstrates that PRYTEN does provide close and conservative results when compared to results from a finite element program. No program or manual changes are necessary since using the program within the range of bolt and base plate allowables will ensure acceptable results.

ATTACHMENT VI

6.0 EVALUATION OF INTERNAL BUILDING SAMs - RESPONSE TO NRC INSPECTION REPORT 50-395/89-200, APPENDIX A, OPEN ITEM 6

FSAR Table 3.7-7a (Ref. 4) tabulates the horizontal and vertical displacements of the Service Water Pump House floors due to the OBE. These displacements are a mix of rigid-body rotation and intra-floor deformation. If these displacements were not due primarily to rigid-body rotation, then these displacements should be considered in the seismic qualification of the SW pumps and the SW discharge piping. The inspection team requested SCE&G either to provide evidence that the displacements tabulated in the FSAR table were due primarily to rigid-body rotation or to consider these SAMs in the seismic qualification of the SW pumps and discharge piping.

During the inspection, SCE&G responded that since the SWPH structure is located on soil, the primary mode of building lateral displacements is due to rocking, i.e., rigid body motion. There is very little relative movement between floors due to the rigidity of the structure relative to the supporting soil. An estimate of the relative movement between floors was presented to the NRC and these numbers were shown to be negligible (i.e., 0.0151 inch in the east-west direction between elevations 425 and 436). The NRC requested to see the building seismic analysis to confirm the rocking component. Since this data was not printed during the original seismic analysis, it could not be presented.

The question of why internal building SAMs do not have to be considered in analyzing V. C. Summer Nuclear Station piping is addressed in the following two sections. The Service Water Pump House structure is evaluated separately because of the large absolute building displacements shown in FSAR Table 3.7.7a. It is necessary to run a building seismic analysis to demonstrate that movements are due primarily to rigid building translation and rotation, and building deformations are small. All other structures are evaluated by subtracting absolute floor movements for successive floors and comparing these values to a 1/8 inch threshold. All relative floor movements in excess of 1/8 inch are considered in the piping analysis. The justification for this 1/8 inch threshold is addressed in Attachment II, Section 2.1, of this report.

6.1 Service Water Pump House

A dynamic analysis was performed of the Service Water Pump House (Ref. 15) structure to confirm that dynamic movements are primarily due to rigid body rotation of the structure as opposed to structural deformations. The soil-structure interaction was accounted for by using the FLUSH (Ref. 6) computer program. This program uses a combination of plate and structural elements to model the soil and building. To minimize work and yet be able to reach a general conclusion, only the east-west direction was run. This direction was chosen because in the original analysis it produced the largest lateral movements.

6.1.2 CONCLUSIONS ON INTERNAL BUILDING SAMs - Service Water Pump House

The results confirm the original response to the NRC that building movements are due primarily to rigid body rotation. The dynamic analysis shows that total relating structural deformation of the building between the bottom at Elev. 390'-0" to top at Elev. 459'-0" is 0.0155 inch. This is considered negligible with respect to considering SAM movements between building floors, the resulting relative movements between floors being considerably less than this value.

The dynamic analysis indicated a maximum structural displacements as follows:

Maximum Displacement at Top (Elev. 459') - 1.3008 in.
Maximum Displacement at Bottom (Elev. 390') - 0.7728 in.

The analysis also indicated that the rigid body rotation for all structure nodal points was the same:

Rigid Body Rotation $= 0.619 \times 10^{-3}$ radians

The rigid body translational measurement at the top of the building due to this rotation is:

828 In. (459' Elev. - 390' Elev.) x 0.619 x 10^{-3} radians = 0.5125 In.

And the deformation due to the structural deformation is:

Deformation = 1.3008 - 0 .7728 - 0.5125 = 0.0155 inches

This confirms the displacement due to structural deformation is negligible and that rigid body motion produces the prevalent displacement.

While the specific question on this issue relates to the Service Water Pump House, the NRC also requested an explanation of how internal building SAMs were addressed with other seismic Category I buildings.

6.2 All Other Structures

The approach taken on other structures to show that internal building SAMs need not be considered was to review the absolute building movements given in FSAR Table 3.7-7a (Ref. 4). By subtracting these OBE values on successively higher floors, it was possible to determine which exceed the 1/8 inch threshold. The 1/8 inch value has been justified as an acceptable threshold below which SAMs do not have to be considered.

The results of performing this review show that only the Fuel Handling Building has internal building SAMs exceeding 1/8 inch. A review was made of all piping within the Fuel Handling Building and it was determined that only two service water lines run between the floors with SAM movements exceeding 1/8 inch. The piping in both cases is attached to the upper floor with single rigid rods on the risers. This detail

accommodates any lateral movement between floors; therefore, the effects of the SAMs do not stress the piping.

6.2.2 CONCLUSIONS ON INTERNAL BUILDING SAMs - All Other Structures

The analysis performed (Ref. 16) demonstrate that internal building SAMs are not an issue. For the Service Water Pump House, the building was shown to move primarily by rigid body motion, with the structural deformations being negligible. Internal building SAMs are not an issue in other buildings because they are generally less than the 1/8 inch threshold. The Fuel Handling Building has internal building SAMs exceeding 1/8 inch; however, these were shown to be acceptable because of a support arrangement which permits lateral movement.

6.3 PROCEDURE REVIEW, INTERNAL BUILDING SAMs

During the NRC follow-up review it was identified that our procedures did not provide specific guidance with respect to internal building SAMs. SCE&G agreed to clarify the requirements for internal building SAMs in our existing procedures.

Attachment VI, Appendix A, is a copy of our revised procedure, G/C Report No. 2439 (Ref. 1), that addresses the requirements for internal building SAMs.

PROCEDURE REVISION ADDRESSING INTERNAL BUILDING SAMs

displacement is now equal to zero. In other words, the building does not bend.

In sketch (5), the rotation is considered zero and the absolute displacements (Δ_1 and Δ_2) are generated for bending the building or imposing a "relative" displacement between the adjacent floors. This is the displacement that must be considered. The structural output from their analytical model yields only absolute displacements resulting from both rigid body rotation and "relative" displacements. Separating the two becomes impractical. Subsequent discussions on this issue have led to the position that the majority of the displacement can be attributed to rigid body rotation; therefore, the relative displacement between floors can be neglected, if the difference is less than 1/8".

In sketches (6 through 8), the building response to a vertical acceleration is considered. Sketch (6) depicts the three floors with no vertical acceleration input. Sketches 7 and 8 reflect both the "in-phase" and "out-of-phase" response to the vertical input. Considering the increased flexibility of the floors and variation in design, it is considered quite possible to find "out-of-phase" response between adjacent floors for vertical accelerations. Considerations to this condition should therefore be made within a given building. A review of these displacements for the Virgil C. Summer project indicates maximum "out-of-phase" displacements less than pin tolerances for pipe supports, thereby never imposing a load into the connected piping system. This condition is therefore neglected.

In conclusion, seismic anchor movements were not analyzed for piping systems supported by a single structure, initially. In early 1990 calculations NRC 7914-05 and NRC 7914-06 evaluated all structures to determine if any seismic anchor movements exceeded 1/8" between adjacent floors within a structure. Based on this evaluation, this only

PROCEDURE REVISION ADDRESSING INTERNAL BUILDING SAMs

occurs in the fuel handling building. In all cases found, the as-built conditions were sufficiently flexible to meet Code stress requirements.

For all future analysis work, seismic anchor movements between floors shall be evaluated and if the relative displacements between adjacent floors exceed 1/8", they shall be included in the analysis.

For a given piping system that is supported or anchored in separate structures, the following consideration is given (see Figure 6.2-7, sketch 9).

For a given horizontal or vertical input, the respective floor responses within each building are considered "in-phase"; however, the possibility exists that motion of the buildings themselves may become "out-of-phase". Therefore, when considering SAM displacements, the maximum absolute displacement representative of location of the support in one structure will be evaluated against the maximum absolute displacement, representative of the location of the support, in the adjacent structures but in opposite directions.

In the consideration of SAM, the following approach should be taken for piping systems supported in adjacent structures.

For each of the three global directions, the most severe out-of-phase structural displacements should be considered. This is accomplished by displacing the constrained boundary joints within each structure in opposite directions. All supports included in the seismic analysis model shall be considered in the SAM model.

The results of these three static analyses are then combined by the "square-root-sum-of-squares" method (SRSS). These results are used in the evaluation of Equation 10 or 11 of the Code. The resulting support loads are superimposed onto the thermal expansion loads and evaluated accordingly.

REFERENCES

1. G/C Report #2439, "Piping Analysis Data," V. C. Summer Nuclear Station, South Carolina Electric & Gas Company. This document describes methods, conventions, and criteria used for piping stress analysis on the V. C. Summer Nuclear Station.

2. G/C Report #2440, "Project Pipe Support Design Data," V. C. Summer Nuclear Station, South Carolina Electric & Gas Company. This document provides guidelines for evaluating and designing pipe supports on the V. C. Summer Nuclear Station.

3. D-306-Series flow diagrams. These drawings define the analysis problem boundaries for rigorously analyzed safety related piping on the V. C. Summer Nuclear Station.

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