CP&L

Carolina Power & Light Company

NOV 17 1992

SERIAL: NLS-92-266

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
FOLLOW-UP QUESTIONS ON SEISMIC QUALIFICATION ISSUES
(TAC NOS. M83211 AND M83212)

Gentlemen:

The purpose of this letter is to provide written responses to Nuclear Regulatory Commission (NRC) staff questions related to seismic qualification issues at the Brunswick Steam Electric Plant, Unit Nos. 1 and 2. These questions, which were documented in an NRC letter dated August 25, 1992, relate to Carolina Power & Light Company's (CP&L) July 16, 1992 submittal. The questions were discussed with CP&L during telephone conferences on August 6, 1992 and August 11, 1992. The Company's responses to the NRC staff questions are enclosed.

Please refer any questions regarding this submittal to Mr. D. B. Waters at (919) 546-2710.

Yours very truly,

D. C. McCarthy

D. C. McCarthy
Manager
Nuclear Licensing Section

WRM/wrm (nls92266.wpf)

Enclosures

cc: Mr. S. D. Ebneter
Mr. R. H. Lo
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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKET NOS. 50-325 & 50-324
OPERATING LICENSE NOS. DPR-71 & DPR-62
FOLLOW-UP QUESTIONS ON SEISMIC QUALIFICATION ISSUES
(TAC NOS. M83211 AND M83212)

Enclosure 1 of the NRC's August 25, 1992 letter provides NRC staff comments on the operability criteria of piping. The following staff questions/comments relate to Study report M-020, which is referenced in the response to NRC Question 1.F in CP&L's July 16, 1992 submittal.

NRC QUESTION 1:

The SRSS combination method for 2D responses is unacceptable to the staff. Provide clarification of the 1.38 factor used and why it was later found not to be required.

CP&L RESPONSE:

The Company's position on 2D SRSS analysis was provided to the NRC by letter dated May 29, 1979 (copy provided as Attachment 1). This letter also discusses the use of the 1.38 factor. During the subsequent reanalysis performed after 1985 by CP&L, the predominant analysis method was 3D SRSS using the Regulatory Guide 1.92 model and co-directional combination.

NRC QUESTION 2:

Confirm that the use of Code Case N-411 is in compliance with the requirements of Regulatory Guide 1.84, Revision 26.

CP&L RESPONSE:

The Company requested the use of Code Case N-411 by letter to the NRC dated May 22, 1985, (copy provided as Attachment 2). Subsequent approval was received from the NRC by letter dated August 28, 1985 (copy provided as Attachment 3).

NRC QUESTION 3:

If the table is for Class 1 piping, what are the criteria for Class 2/3 piping.

CP&L RESPONSE:

The table is for Seismic Class I, which was the UE&C designation for structures, systems, and components requiring seismic design. Seismic Class II is non-seismic. The Class 2/3 piping classification is an ASME designation; equivalent piping at the Brunswick Plant was designed as Seismic Class I.

NRC QUESTION 4:

Provide more specific discussion on the OBE/DBE conversion factor of 1.2 to 2.0.

CP&L RESPONSE:

The 1.2 conversion factor was developed for conversion of OBE loads and stresses. The development of this factor was presented to the NRC by letter dated May 29, 1979, (copy provided as Attachment 1). As documented in an NRC staff meeting summary dated June 12, 1979 (copy provided as Attachment 4), the NRC indicated acceptance of CP&L's method during a phone call on June 1, 1979. The design turnover program for piping criteria uses specific DBE response spectra curves or a conversion factor of 2.0.

NRC QUESTION 5:

Frequency cutoff at 20 Hz with no "missing mass" considered is clearly an unacceptable analysis procedure.

CP&L RESPONSE:

The frequency cutoff at 20 Hz with no "missing mass" was based on the licensing basis for BNP. Most of the UE&C calculations have been reanalyzed by CP&L. These calculations included a "missing mass" correction to 90 percent mass participation for long term calculations. Additionally, as stated in CP&L's letter to the NRC dated May 22, 1985 (copy provided as Attachment 1), a value of 33 Hz in conjunction with Code Case N-411 damping valve curves will be used in lieu of 20 Hz. The 33 Hz frequency cutoff was included in Amendment 4 to the Brunswick Plant Updated Final Safety Analysis Report, page 3.9.2-6.

Enclosure 2 of the NRC's August 25, 1992 letter provides the following NRC staff questions concerning CP&L's July 16, 1992 submittal.

NRC QUESTION 6:

In your response to NRC Question 1.H, you stated that "All masonry walls that were considered seismic in the original IE Bulletin 80-11 walkdown with structural angle restraints attached with expansion anchors that are located outside (underline added for emphasis) of the diesel generator building have been reviewed. The results of these inspections are summarized in the response of question I.B." However, your response to NRC Question I.B stated that you have inspected all the walls in the diesel generator building, the walls with IE Bulletin 80-11 modifications in both the control building and diesel generator building, and a group of seismic walls outside the diesel generator building (i.e., the control building and reactor building). The answer to question I.B is not clear as to whether all the walls with structural angle restraints attached with expansion anchors, classified as Category I or seismic, in the plant have been field inspected, design reviewed, and physically modified, if needed, to design standard. Please clarify your answer.

CP&L RESPONSE:

All Seismic Category I walls have been inspected for structural integrity and design reviewed for potential loads in addition to seismic. Repairs are being made as necessary.

NRC QUESTION 7:

In your response to NRC question I.B, you stated that 6 walls, which were restrained by anchor bolted angles by original design and construction outside the diesel generator building, were 100 percent inspected and in a few cases anchors were discovered to be 5/8-inch sleeve anchors in lieu of 3/4-inch anchors, but all met IE Bulletin 80-11. Since IE Bulletin 80-11 does not address or provide any requirements with respect to anchors, we are confused by your response of meeting IE Bulletin 80-11 as stated above. Expand or clarify your response.

CP&L RESPONSE:

Industry resolution for IE Bulletin 80-11 contains criteria for acceptance of masonry walls. The Company's response actually refers to the resolution documentation for IE Bulletin 80-11 which was provided by NRC letter dated January 30, 1985, "Masonry Wall Design, IE Bulletin 80-11" (copy provided as Attachment 5). This NRC letter issued the staff's Safety Evaluation and Technical Evaluation Report (TER). The statement referring to the six (6) walls is meant to indicate that the condition of the walls is acceptable under the referenced TER, although they were not constructed per the original design.

NRC QUESTION 8:

In your response to NRC question I.E, you stated that those anchors with frozen studs were load tested. Explain how the load test was performed and the level of load which was used, such as allowable design loads.

CP&L RESPONSE:

A manually operated calibrated torque wrench was used. Sufficient space was verified to exist between the plate and the anchor shell or shims were added during the test. The torque values specified in the procedure were greater than two times the manufacturers design load.

NRC QUESTION 9:

In your response to NRC question II.B, you stated that "An overall review of the IE Bulletin 80-11 program is underway for the Brunswick plant. The review will address existing masonry wall functions, including missile barrier, tornado barrier, ventilation barrier, or other functions for which it is not analyzed." We are confused by the words 'for which it is not analyzed." Expand or clarify your response.

CP&L RESPONSE:

The Company's review for masonry walls under IE Bulletin 80-11 included walls with functional requirements as indicated in item 1 of IE Bulletin 80-1* All missile barrier walls were reviewed under IE Bulletin 80-11 for the appropriate missile loa . The Brunswick Plant has no exterior masonry walls serving as tornado barriers; therefore, tornado loading was not previously considered to control the design. A more detailed building analysis for tornado pressurization indicates that instantaneous pressures on internal walls in the diesel building control the design rather than seismic considerations controlling the design in some cases. The Company's functional review also identified a new safety related function not considered in IE Bulletin 80-11. The new safety related function is a series of walls in the diesel building that serves as a barrier between the supply and exhaust air for the diesel bays. These walls are being seismically restrained commensurate with their safety related function.

NRC QUESTION 10:

On page 4 of the enclosure with file No. B09-13510c and Serial No. BSEP/82-1616, you stated that "Tests for proper installation were performed on 59 percent of the anchor bolts. .... All of the anchors with unremovable bolts or studs were successfully tested for preload, however, demonstrating the load capability of the anchors." What criterion was used for "proper installation?" Does 59 percent of the anchor bolts means 59 percent of the anchor bolts in the whole plant? What criterion was used for "preload?" Expand or clarify the words "however, demonstrating the load capability of the anchors."

CP&L RESPONSE:

This NRC questions contains a number of parts, each of which are addressed below:

Proper Installation

"Proper installation" was based on the following two tests:

1. Thread engagement between the bolt and the sleeve. The test acceptance criteria is shown in the enclosed Table 4 (copy provided as Attachment 6) from the test procedure.

2. Embedment depth, which for a self drilling anchor, was a function of plug depth (see the sketch provided as Attachment 7). The acceptance criteria is shown in the enclosed Table 3 (copy provided as Attachment 6) from the test procedure.

"59 Percent"

"59 percent" means 59 percent of the total number of available anchor bolts in this phase of IE Bulletin 79-02 testing for the Brunswick Plant, Unit 2.

The following historical information may aid in putting this statement in perspective:

The Brunswick Plant IE Bulletin 79-02 testing effort was completed somewhat independently for the two units due to different outage schedules. Also, some piping systems or portions of systems were missed by our architect-engineer when the scope of

IE Bulletins 79-02, 07, and 14 was originally determined. Our July 26, 1982 letter (copy provided as Attachment 8) was reporting completion of the anchor bolt testing for the Unit 2 portion of this additional scope. Hence, 59 percent means 59 percent of the Unit 2 anchor bolts on the missed systems or portions of systems.

Subsequently, in April 1992, when the masonry wall anchor bolt issue arose, CP&L management directed the IE Bulletin 79-02 effort be re-examined to ensure the problems with diesel generator building walls did not extend to pipe support anchor bolts. The July 26, 1982 letter needed clarification due to the number of "frozen" studs and nuts reported. Therefore, the April 1992 CP&L audit team retrieved and reviewed the individual anchor test data sheets to determine the reason for the frozen studs and nuts. The results of this audit have been discussed with the NRC in our letters dated April 15, 1992 (Serial No. NLS-92-118); May 29, 1992 (Serial No. NLS-92-148)

It appears the reason for the relatively large number of frozen studs and nuts found during this phase of the original IE Bulletin 79-02 inspections is due to the fact that this is when the corrosion prone areas were covered. Hence, corrosion, not fraudulent installation, accounted for the frozen studs and nuts.

Design Load Test

The "preload" or design load test involved the application, by hydraulic jack or nut torque, of a pull out load equal to or greater than the allowable design load of the anchor. The acceptance criteria was that the anchor "pull out" during the loading be no more than 1/16 of an inch.

Load Capability

"Demonstrating the load capability of the anchors" means the actual pull testing of the anchors, as described in the "Design Load Test" discussion above. This test demonstrates the anchor's capability to carry its design load.

ATTACHMENT 1

CP&L Letter Dated May 29, 1979

ATTACHMENT 2

CP&L Letter Dated May 22, 1985

ATTACHMENT 1

CP&L Letter Dated May 29, 1979

79-07

BC

CP&L

Carolina Power & Light Company

File: NG-3514(B) May 29, 1979 SERIAL: GD-79-1401

Office of Nuclear Reactor Regulation
ATTENTION: Mr. T. A. Ippolito, Chief
Operating Reactors Branch No. 3
United States Nuclear Regulatory Commission
Washington, D. C. 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 and 50-324
LICENSE NOS. DPR-71 AND DPR-62
SEISMIC ANALYSIS OF SAFETY-RELATED PIPING

Dear Mr. Ippolito:

At our meeting on May 21, 1979, Carolina Power and Light Company committed to provide the NRC Staff additional information concerning our response to IE Bulletin 79-07 on seismic pipe stress analysis. On May 23 and 24, 1979, the Staff identified to us, by telephone, and to a representative of United Engineers and Constructors, our architect engineer for the Brunswick Steam Electric Plant, several additional items that should be addressed in our response. The remainder of this letter and attachments respond to those requests.

1. The analysis of the loads for the pipe supports for the first ten (10) lines reanalyzed for pipe stresses has shown that there were ten cases where the load exceeded allowable. Table 1-1 summarizes the data on the 98 pipe supports on these ten (10) lines. Table 1-2 presents the details of the ten (10) supports that were overstressed.

   While evaluating these ten pipe supports, it was determined that the supports had been underdesigned initially. In no case was the overstressed condition a result of the new load from the seismic reanalysis. As shown on Table 1-2, the new load actually decreased in five cases, increased less than 2.5% in four cases, and increased 19% in only one case (which was already over capacity by 14.5%). These ten supports were analyzed to determine if their structural integrity would be maintained under the identified loads. Four of these supports were found to maintain stresses less than yield and thus would maintain structural integrity.

   When it was determined that structural integrity would be compromised for the six supports under the calculated loads, Carolina Power & Light Company decided to shut down both units and make necessary modifications to these supports to reduce stresses to less than allowable. These modifications have been initiated and the new capacity is shown on Table 1-2.

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During this evaluation, it was noted that the overloaded pipe supports failed in two ways: either concrete anchors or in torsion. An investigation was begun to look at all pipe supports on safety related systems to determine if similar overloaded conditions may exist under the original load. The resul s of this investigation will be available on June 1, and all necessary modifications will be made prior to returning the units to operation.

2. On May 24, 1979, the Staff informed us by telephone that the seismic stress analysis should be based on absolute sum if a two-dimensional seismic analysis was used, and that the square root of the sum of the squares (SRSS) was acceptable if a three-dimensional seismic analysis was made. The Staff further stated that a stress from a two-dimensional analysis calculated using SRSS and multiplied by a factor of 1.38 would be acceptable. At the time that BSEP was licensed, two-dimensional SRSS seismic analysis was acceptable criteria, and it is not apparent to us that the back-fit of a two-dimensional absolute sum seismic analysis has undergone the necessary requirements of 10CFR50.109. Although CP&L does not accept the Staff's position, we have prepared a revision to Table 2 of our letter of May 21 demonstrating the effect of multiplying the two-dimensional analysis results by the 1.38 factor. We have also taken credit for conservatism that exists in the relationship between the OBE and the DBE. The results of this exercise show that only one line of the first thirty-nine reanalyzed lines exceeds total allowable stress by 2%. For this line, the total stress is still less than $0.9S_y$.

For the unreanalyzed lines shown in Attachment 3 of our May 21 letter (GD-79-1342), we have used the 1.38 factor to establish criteria for priority of lines to be reanalyzed. We do not plan to base our conclusions of acceptability on the use of the 1.38 factor, since it is not the appropriate criteria for BSEP. In determining the criteria for priority of reanalysis of the remaining lines, SRSS stresses were estimated on the basis of a factor of 1.5 increase, and this resultant was then multiplied by 1.38. Credit for the conservatism of the OBE/DBE relationship was taken into account prior to applying the 1.5 increase. When this was applied to the 411 lines that have not been reanalyzed, 39 of the 411 exceeded allowable stress, and are tabulated in Attachment 2 to this letter. Our reanalysis priorities have been changed to include these 39 lines in those to be reanalyzed the week of May 28, and the results of this reanalysis should be available on June 1, 1979. We still anticipate completing the total reanalysis in accordance with our previously stated completion date of July 21, 1979.

3. As a result of an I & E inspection at the Brunswick Steam Electric Plant to verify that the as-built dimensions were the same as the as-designed (as-analyzed) system, four deviations were noted. These are discussed in Attachment 3.

   As stated in the meeting on May 21, 1979, and confirmed in our letter of May 22, 1979, Carolina Power & Light Company will verify as-built dimensions for all safety related systems at BSEP. This verification is currently in progress for those lines outside containment. The lines inside containment will be verified at the next scheduled outage. Due to the time constraints on reanalysis, the reanalysis is being conducted concurrently with the as-built verification. If any discrepancies are identified between the as-built/as-analyzed configurations, an evaluation by a stress analyst will be made to determine if the line should be reanalyzed. This evaluation will be based on evaluating the magnitude of the computed stresses for the area in question, and the impact (increase or decrease) on the stresses expected for such deviation. If it is determined that the line needs to be reanalyzed to determine the new stress level, we will promptly reanalyze the line.

4. During our recent meetings, the relationship of IE Bulletins 79-02 and 79-07 has been discussed. Some of the pipe supports analyzed in the first ten lines are anchored using concrete expansion anchors discussed in Bulletin 79-02. In the 79-07 support reanalysis, these base plates were and will continue to be analyzed using IE Bulletin 79-02 as a guide. The capacity established for the concrete anchors is 20% of the manufacturer's rated capacity. Using this criteria, two supports on the first ten lines had to be redesigned and now have sufficient capacity. As stated in item 1 above, the remaining supports using concrete expansion anchors are being investigated to determine their adequacy and will be reported on June 1. A final report on all of our analyses and testing related to the concrete expansion anchors and IE Bulletin 79-02 will be submitted in compliance with the bulletin schedule.

5. The Staff requested information on the location of the postulated pipe rupture for a LOCA relative to the point of highest stress. The BSEP piping design did not use the mechanistic approach of locating the pipe break at the point of highest stress. The postulated break for doubled-ended guillotine or longitudinal split was analyzed for the pipe break to occur at any point on the pipe, inside or outside containment.

6. We have been informed that during a meeting between NRC, another licensee and United Engineers & Constructors (UE&C), some questions were raised by the NRC staff about the subject

of valve operability. In the event the staff may have any questions concerning this topic as it may apply to BSEP, we will be prepared to address this issue.

7. Carolina Power & Light Company's criteria for determining if an overstressed condition is reportable is set forth below:

   a. Lines Yet To Be Reanalyzed

      The stress using new seismic data and revised analytical criteria are estimated for the lines that are yet to be reanalyzed. As stated previously, those with high estimated stresses are being analyzed first in the reanalysis program. We will not use estimated stress as a basis for determining overstressed conditions which are reportable.

   b. Reanalyzed Lines

      Those lines which have been reanalyzed and which show an apparent overstress condition will be evaluated in detail to determine if it is a reportable item. First, the known conservatisms will be removed from the analysis. The line will be analyzed to determine if the stress at any single modal point exceeds FSAR criteria of $0.9S_y$ or $1.8\ S_h$, whichever is the higher. If the pipe remains overstressed, this will then be considered a reportable item and the NRC will be informed within 24 hours.

   c. Reanalyzed Pipe Supports

      When the reanalyzed pipe data is available, the pipe supports will be reanalyzed for the revised load. If the load exceeds the apparent support capacity, the specific support will be analyzed in detail to determine if the stated capacity is the actual capacity without exceeding $0.9\ S_y$. If the load still exceeds the capacity, a determination is made if the support will maintain structural integrity even if the allowable is exceeded. If structural integrity is maintained, this is not considered reportable. If structural integrity is not maintained, the support is taken out of the computer piping configuration, and the line is reanalyzed. The results of this reanalysis are evaluated to determine if other supports and the pipe can take the additional load without exceeding their structural integrity. If the system maintains integrity, the item is not reportable. If the system does not maintain structural integrity, the item will be reported within 24 hours.

In summary, CP&L has evaluated the data from the lines reanalyzed to date, and the estimates for revised stresses for lines yet to be reanalyzed, and it is our conclusion that the continued operation of the Brunswick Steam Electric Plant, Units 1 & 2 is warranted without undue risk to the public health and safety, while the reanalyses of seismic design continues. The problem associated with those supports that were found to be overstressed is a result of initial underdesign of those supports, and is not related to the use of algebraic, square root sum of the square, or absolute summation of seismic stresses. The modifications of those supports which were originally under-designed will be completed in early June, and at that time, both units will be returned to power. As stated in our letter of May 15, 1979, and in item 7 of this letter, 24-hour reporting criteria have been established if any piping or supports are determined to be overstressed during the reanalyses. If you have any questions concerning this information, please do not hesitate to contact our staff.

Yours very truly,

E. E. Utley

E. E. Utley
Executive Vice President
Power Supply

DLB/sg

bcc: Messrs. D. L. Bensinger
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File: BC/A-4
File: B-X-0274

ATTACHMENT 1
PIPE SUPPORT ANALYSIS

An evaluation was performed on the pipe supports of the first ten lines that were reanalyzed in the seismic pipe stress reanalysis program. There are 98 pipe supports made up of snubbers, vendor catalog pipe supports, and fabricated supports. The recalculated loads compared to the original load and support structural capacity are tabulated on Table 1-1.

As can be seen on Table 1-1, the load did not increase appreciably due to the seismic stress reanalyses and recalculation of loads. The load decreased for 30% of the supports and increased less than 25% for 60% of the supports. The load increased greater than 25% for only nine supports, but the new loads were less than 75% of capacity for these supports.

However, ten supports were found where the load exceeded the applicable allowable. Further investigation revealed that these ten supports were underdesigned initially. For these ten supports, the new loads were less than the old loads in five cases, increased less than 2.5% in four cases, and in only one case, the increase was 19%.

These ten supports were analyzed in detail to determine if they would maintain their structural integrity under the specified loads even if they exceeded allowable. This is summarized on Table 1-2. In four cases, including the one where the new load was 19% higher than the old load, the supports retained their structural integrity. Six supports would fail.

The six supports that would fail under the specified load (old or new) were redesigned to have their stresses less than allowable. The new design loads for these pipe supports are shown on Table 1-2.

It has been concluded from the analysis of 98 pipe supports that the seismic stress reanalysis does not contribute to overloaded pipe supports. However, it has been recognized that there is a potential for certain supports to be overloaded due to an error in the initial design. These errors have been found to be with concrete expansion anchors and with torsion of the beam support. An investigation has begun to examine the pipe supports of the other safety-related piping for similar problems. The results will be reported at a later date.

<u>TABLE 1-1</u>

<u>SUMMARY OF PIPE SUPPORT LOADS</u>

LINE	POINT	LIMITING PART	CAPACITY C	UPSET LOAD OLD LOAD (OL)	UPSET LOAD NEW LOAD (L)	RATIO L/OL	RATIO L/C	REMARKS
17	65S	Strut. Cat. P.S.	13800	7037	7326	1.04	.53	
	95S	Cat. P.S.	15700	9129	9311	1.01	.59	
	110S	Snubber	3920	1134	1430	1.26	.36	
	137S	Cat. P.S.	4960	4192	4286	1.02	.86	
	172S	S.S. Sup't.	1836	606	595	0.98	.32	
	175S	Cat. P.S.	11630	6329	6527	1.03	.56	
	220S	X Cat. P.S. Strut	3920	968	1176	1.21	.30	
		Y Cat. Sup't.	6230	5544	5588	1.007	.90	
		Z Cat. P.S.	3920	1483	1634	1.10	.42	
	255S	Cat. P.S.	8000	4088	4160	1.01	.52	
	402S	Snubber	3920	411	576	1.40	.15	
24	136	XZ Snubber	3920	1339	2690	2.00	.68	
		Y Snubber	3920	1352	1400	1.03	.36	
	154	X Snubber	3920	+1576	+1243	0.09	.36	
		Z Snubber	3920	+1576	+1243	0.09	.36	
15B	61S	Z Snubber	3920	956	1078#	1.12	.27	
	61S	Y Snubber	3920	1381	1912	1.38	.49	
	105S	Snubber	3920	1544	2133	1.[illegible]	.54	
	18S	Weld	15300	11850	11466	0.96	.75	
	60S	Snubber	22177	11837	11578	0.97	.52	
	71S	Snubber	13666	7993	8392	1.04	.61	
	107S	Snubber	13800	3187	3196	1.002	.23	
	73S	Snubber	37600	25612	23214	0.90	.62	
	110S	X Snubber	29090	5239	4700	0.89	.16	
	110S	Z Snubber	20736	14316	13468	0.94	.65	

TABLE 1-1 (Cont'd)

LINE	POINT	LIMITING PART	CAPACITY C	UPSET LOAD OLD LOAD (OL)	UPSET LOAD NEW LOAD (L)	RATIO L/OL	RATIO L/C	REMARKS
237	420	Conc. Anchors	678	4399	4014	0.91	(5.92)	See Table 1-2
	420	Y-RSSA-20 Strut	20000	11124	13311	0.11	.67	
	440	X-EX.W8x17 (Torsion)	790	2784	2490	0.89	(3.15)	
	440	Z-Strut RSSA-10	10000	5641	5792	1.02	.58	
	466	Snubber	3920	1884	1945	1.03	.50	
	472	Weld-Post Ex.Stl	5374	4001	4551	1.13	.85	
	484	Snubber	13800	3546	3570	1.006	.26	
	503	Snubber	13800	3339	3539	1.05	.27	
	525	Weld-Post to Snub	6000	4613	4884	1.05	.81	
122	2092	Snubber	3920	2117	2101	0.99	.54	
	2094	Snubber	23600	3818	3827	1.002	.16	
	2220	Snubber	13800	9008	9048	1.004	.66	
	2143	Snubber	13800	5674	5670	0.99	.41	
	2156	X -W6 x 15.5 (Torsion)	660	2726	1775	0.65	(2.69)	See Table 1-2
	2156	Z-Snubber	23600	1521	1504	0.98	.06	
	2230	Snubber	13800	4804	4182	0.87	.30	
	2240	Snubber	13800	6179	3685	0.59	.27	
	2174	W6 X 15.5(Torsion)	660	6714	6874	1.02	(10.42)	See Table 1-2
	2062	Snubber	13800	3044	3046	1.00	.22	
121	3084	Conc. Anchors	840	2970	2991	1.007	(3.56)	See Table 1-2
	3083	Conc. Anchors	12960	3866	3207	0.82	.25	
	3067	Snubber	13800	6190	6244	1.008	.46	
	3066	Snubber	13800	8152	8010	0.98	.59	
	305S (55)	Clamp	11500	10798	9143	0.84	.79	

TABLE 1-1 (Cont'd)

LINE	POINT	LIMITING PART	CAPACITY C	UPSET LOAD OLD LOAD (OL)	UPSET LOAD NEW LOAD (L)	RATIO L/OL	RATIO L/C	REMARKS
	3048	Snubber	3920	2843	3214	1.13	.83	L/C = .95 for Emergency Conditic
		Snubber	13800	3302	3405	1.03	.24	
	3200	Snubber	3920	712	1026	1.44	.26	
6	13S	Snubber	37600	31000	18000	0.58	.48	
	133S	Snubber	23600	14499	13400	0.92	.57	
	[illegible]32/ [illegible]	Fab. Sup't	7460	3341	5463	1.63	.73	
	[illegible]	Snubber	13800	10189	9377	0.93	.68	
	[illegible]0S	Snubber	13800	4628	5752	1.24	.42	
	806S	Snubber	13800	6099	6863	1.12	.50	
	103S	Snubber	13800	5106	8402	1.64	.61	
	719S	Struct. Supt.	8800	11005	8600	0.78	.98	
	718S	Snubber	13800	7383	5753	0.77	.42	
	24S	Snubber	13800	7651	6137	0.80	.44	
	725S	Snubber	3920	1200	1249	1.04	.32	
	710S	Snubber	3920	2114	2423	1.14	.62	
	724S	Y Struct. Supt. Z Snubber	1000 3920	3406 1601	4112 1823	1.20 1.13	.41 .47	
	901S	Snubber	3920	2154	2566	1.19	.63	
	900S	Snubber	3920	1818	2160	1.18	.55	
	722S	Struct. Supt.	12200	2826	3442	1.21	.28	
	108S	Cat. Pipe Supt.	[illegible]0	7144	7693	1.07	.86	
510	133	Struct. Steel Supt.	3563	1948	1961	1.006	.55	
	125	Weld	3712	3361	3505	1.04	.94	
	132	Struc. Steel Supt. Channel	1350	1546	1844	1.19	(1.36)	See Table 1-2

TABLE 1-1 (Cont'd)

LINE	POINT	LIMITING PART	CAPACITY C	UPSET LOAD OLD LOAD (OL)	UPSET LOAD NEW LOAD (L)	RATIO L/OL	RATIO L/C	REMARKS
	195	Struc. Steel Supt.	27600	7049	7126	1.01	.26	
	148	Struc. Steel Supt.	6040	2316	2276	0.98	.38	
	120	Struc. Steel Supt.	9645	5510	5498	0.99	.57	
	111	Bolts	1445	77	99	1.28	.02	
	137	Struc. Steel Supt.	3897	2265	2264	0.99	.59	
	16	Bolts	5038	5601	5456	0.97	(1.08)	See Table 102
	230	Struc. Steel Supt.	3770	3415	3473	1.01	.93	
	224	Struc. Steel Supt.	12992	11140	11311	1.01	.87	
	225	Struc. Supt.	4960	3207	3211	1.001	.67	
	40	Struc. Steel Supt.	5250	5271	5255	0.99	1.0	
	116	Struc. Steel Supt.	3920	522	524	1.003	.13	
	113	Struc. Steel Supt.	4960	3743	3744	1.00	.76	
	30	Snubber	13000	8758	8881	1.01	.68	
	270	Struc. Steel	11800	18800	18900	1.005	(1.6)	
	136, 135, 803S	Struc. Steel (Pipe Section)	3170	6556	6570	1.002	(2.07)	
125	1072S	Struc. Steel	4192	4122	4093	0.99	.98	
	1065S	Conc. Anchors	12960	6302	7882	1.25	.61	
	1051S	Snubber	11500	3477	3476	0.99	.30	
	1120S	Snubber	13800	1452	1437	0.98	.10	
	1140S	Clamp	11500	10085	10488	1.03	.91	

TABLE 1-1 (Cont'd)

LINE	POINT	LIMITING PART	CAPACITY C	UPSET LOAD OLD LOAD (OL)	UPSET LOAD NEW LOAD (L)	RATIO L/OL	RATIO L/C	REMARKS
	1150S	X Beam (Torsion)	790	1470	1301	0.88	(1.65)	
		Z Snubber	3920	807	998	1.23	.25	
	1029S	Snubber	13800	9111	9630	1.05	.70	

TABLE 1-2

SUMMARY OF PIPES WITH L/C >1.0

LINE	POINT	LIMITING PART	CAPACITY (C)	CAPACITY TO YIELD (Cy)	RATIO (NL/C or Cy)	OLD LOAD (OL)	NEW LOAD (NL)	RATIO OL/NL	REDESIGN LOAD ***
237	420	Conc. Anchors	678	*	>1.0	4399	4014	1.09	Y 13430 X 4192
	440	8W x 17 I-Bm Torsion	790	*	>1.0	2784	2490	1.11	6762
122	2156	6W x 15.5 I-Bm Torsion	660	*	>1.0	2726	1775	1.53	2769
	2174	6W x 15.5 I-Bm Torsion	660	*	>1.0	6714	6874	0.97	11342
121	3084	Conc. Anchors	840	*	>1.0	2970	2991	0.99	4582
510	132	SS (Channel)	1350	2194	.84	1546	1844	0.83	**
	16	Bolts	5038	8184	.67	5601	5486	1.02	**
	270	Stru. Steel	11800	19656	.96	18800	18900	0.99	**
	136, 135 803S	Stru. Steel Pipe Section	3170	6510	≈1.0	6556	6570	0.99	**
125	1150S	I-Bm Torsion	790	*	>1.0	1470	1301	1.12	2243

* Cy does not apply

** Not redesigned for short term fix

*** Redesign Load consists of new calculated emergency load x 1.38 + transient to be less than AISC allowables (0.67 Sy).

ATTACHMENT 2
SEISMIC PIPE STRESS ANALYSIS

CRITERIA

As stated previously, the original seismic analysis for pipe stress used algebraic summation within each mode. A reanalysis effort was undertaken for all safety-related lines using the UE&C - ADLPIPE-2 Computer Code which employs the square root - sum-of-the-squares (SRSS) load combination within each mode.

The results of the reanalyses, given to the NRC Staff in our responses to IE Bulletin 79-07, in letters dated April 24, May 15, and May 21, 1979, used the SRSS method. On M 24, 1979, the NRC Staff notified CP&L that the use of SRSS with three-dimensional seismic analysis was acceptable, but for a two-dimensional seismic analysis the absolute sum method should be employed within each mode. The analysis for Brunswick uses a two-dimensional seismic input approach. At the time BSEP was licensed, the two-dimensional SRSS analysis was the acceptable criteria. Therefore, the acceptability of stress levels should not be based on absolute sum. However, to use the most conservative case for comparison purposes only, the stresses calculated using UE&C - ADLPIPE-2 were multiplied by 1.38 (a number acceptable to the NRC Staff) to obtain stresses for the Operating Basis Earthquake (OBE).

As discussed in Attachment 7 of our letter to the NRC GD-79-1342, dated May 21, 1979, the previous seismic analysis used a most conversative approach of relating stresses for an OBE to that for a Design Basis Earthquake (DBE), known today as a Safe Shutdown Earthquake (SSE). The stresses computed in the OBE were multiplied by 2 and used as the stresses for a DBE. As discussed on May 21, 1979 with the NRC Staff, our reevaluation of the OBE and DBE Amplified Response Spectra (ARS) indicates that the relationship between the two ARS in the frequency range that affects pipe stress is less than 1.2, and frequently less than 1.0. However, a value of 1.2 has been selected for use to convert OBE stresses to DBE stresses.

For the thirty-nine lines already reanalyzed, the conversative stresses for comparison purposes for the DBE and total are shown on Table 2-1. The DBE stresses in this table are calculated as follows: σ DBE = σ OBE x 1.38 x 1.2, where σ OBE is obtained using the UE&C - ADLPIPE-2.

For the lines yet to be reanalyzed, the stress for a DBE was estimated as explained in Attachment 7 to our May 21, 1979 letter using a factor of 1.5 to account for the highest expected increase in stress due to the reanalysis for SRSS (within each mode) in lieu of algebraic sum (within each mode) and which is based on the data from the reanalyzed lines. For those lines identified on Attachment 3 to our May 21, 1979 letter, the stress for a DBE were estimated as follows:

$$\sigma_{\substack{\text{DBE}\\\text{Est.}}} = \sigma_{\substack{\text{OBE}\\\text{Orig.}}} \times 1.38 \times 1.2 \times 1.5$$

where $\sigma_{\substack{\text{OBE}\\\text{Orig.}}}$ was computed in the original analysis. Those lines whose estimated stresses exceeded allowable are tabulated on Table 2-2.

EVALUATION

As can be seen from Table 2-1, one line (RHR-60, Residual Heat Removal) exceeds the allowable (1.8 S_h) by 1.7 percent. However, this stress is less than the stress equal to 0.9 Sy (32,400). The BSEP FSAR allows the use of 0.9 Sy or 1.8 S_h, whichever is greater, as the allowable stress during emergency condition (DBE). Therefore, the stresses are acceptable for all lines reanalyzed.

Table 2-2 shows that 39 of 411 lines yet to be reanalyzed exceed allowable (1.8 S_h). These stress values are not necessarily based on coincident point maximums, but rather the summation of maximum stresses for each individual loading. It should be restated that these stresses are estimated and that they were derived using a conservative factor of 1.5 to cover the maximum increase expected for the reanlaysis (old algebraic to new SRSS combination within each mode). As discussed in Attachment 7 and shown on Attachment 8 of our May 21, 1979 letter, in over 58% of the lines already reanalyzed, the new seismic stress was less than the original seismic stress. In over 87% of the cases, the new stresses were less than 1.25 of the original stresses.

As discussed previously in our letter, Carolina Power & Light Company commits to placing these lines in the highest reanalysis priority category, regardless of the priority category previously established on a function and size basis.

It should also be pointed out that of the 39 lines estimated to be overstressed, 27 are 2" or less in diameter.

ATTACHMENT 2
PIPE STRESS REEVALUATION
SUMMARY

SYSTEM NAME	ISO NO.	LINE SIZE (NPS)	EMERGENCY CONDITION (PSI) ORIGINAL TOTAL	ORIGINAL SEISMIC	TOTAL 5/21/79	* SEISMIC 5/21/79	** TOTAL 5/25/79	SEISMIC 5/25/79	ALLOWABLE	TOTAL STRESS ALLOWABLE
Main Steam	MS-15B	24	10724	3942	10640	3858	9976	3194	27000	37
Safety/Relief Valve	SRVL-121	10, 6	23012	12280	21910	11180	19987	9257	27000	74
Safety/Relief Valve	SRVL-122	10, 6	19685	15800	24439	13352	22143	11056	27000	82
Safety/Relief Valve	SRVL-237	10, 6	20432	12004	24588	16160	21809	13381	27000	81
Safety/Relief Valve	SRVL-125	10, 6	24270	13347	24316	20270	20830	16784	27000	77
Feedwater	FW-16	18, 12	18007	12420	20028	13296	17741	11009	27000	66
Residual Heat Removal	RHR-6	20	19406	13582	12644	6820	11471	5647	27000	42
Core Spray	CS-24	10	16952	10076	14366	7490	13078	6202	27000	48
High Press Cool Injct	HPCIS-17	14	12200	6446	12502	6748	11341	5587	27000	42
High Press Cool Injct	HPCIS-510	14, 12, 10	12004	7994	12092	8082	10702	6692	27000	40
High Press Cool Injct	HPCIS-10	14, 12, 10	9733	3886	11152	5530	10201	4579	27000	38
Residual Heat Removal	RHR-1	24, 20	24094	18584	17972	14366	15501	11895	27000	57
Residual Heat Removal	RHR-2	20, 16, 12	13309	7654	11471	5948	10448	4925	27000	39
Residual Heat Removal	RHR-5	24	9848	3896	9514	2960	9005	2451	27000	33
Residual Heat Removal	RHR-25	4, 6	18558	12904	18530	12876	16315	10661	27000	60
Nuclear Steam Supply	NSS-14	24, 10	14745	8446	16335	10036	14609	8310	27000	54
Safety/Relief Valve	SRVL-124	6, 10	25536	15928	25984	16376	23167	13559	27000	[illegible]

ATTACHMENT 2 (CONT'D)

SYSTEM NAME	ISO NO.	LINE SIZE (NPS)	EMERGENCY CONDITION (PSI) ORIGINAL TOTAL	ORIGINAL SEISMIC	TOTAL 5/21/79	* SEISMIC 5/21/79	** TOTAL 5/25/79	SEISMIC 5/25/79	ALLOWABLE	TOTAL STRESS / ALLOWABLE
Safety/Relief Valve	SRVL-126	6, 10	23361	18000	22197	17422	19200	14425	27000	71
Residual Heat Removal	RHR-52	14, 12	23271	17936	19539	14204	17096	11761	27000	63
Reactor Core Isolat. Cool	RCIC-21	3	7603	3588	7601	3586	6982	2967	27000	26
Resid Heat Rem Drain Line	RHR-173-B	1½	3808	2186	3838	2216	3457	1835	27000	13
Residual Heat Removal	RHR-28	20, 16, 12	15298	8814	13626	7142	12398	5914	27000	46
Nuclear Steam System	NSS-15	24	10974	4420	10458	3904	9787	3233	27000	36
Nuclear Steam System	NSS-120 (15C)	10, 6	19443	8902	17899	8298	16472	6871	27000	61
High Press Cool Injct	HPCIS-4	3, 6, 10, 12	23609	20876	25481	22748	21568	18835	27000	80
Nuclear Steam System	NSS-123 (15C)	6, 10	21027	16098	18577	16782	15691	13896	27000	58
Nuclear Steam System	NSS-187 (15C)	10, 6	21856	11337	23596	16424	20771	13599	27000	77
Residual Heat Removal	RHR-42	12, 14	18116	12976	17480	12340	15358	10218	27000	57
Residual Heat Removal	RHR-3	14, 16, 20, 24	25317	18328	23379	15590	20698	12909	27000	77
Residual Heat Removal	RHR-13	4, 8, 14	12532	10018	12620	10106	10882	8368	27000	40
Residual Heat Removal	RHR-59	4, 6, 10	12664	9970	13344	10650	11512	8818	27000	43
Residual Heat Removal	RHR-60	4, 6, 10	34618	33658	32971	32012	27465	26506	27000	102
Residual Heat Removal	RHR-168	1	23580	22658	26910	26198	22404	21692	27000	83
Residual Heat Removal	RHR-61	4, 6, 3/4	21117	17038	19393	15314	16759	12680	27000	62

ATTACHMENT 2 (CONT'D)

SYSTEM NAME	ISO NO.	LINE SIZE (NPS)	EMERGENCY CONDITION (PSI) ORIGINAL TOTAL	ORIGINAL SEISMIC	TOTAL 5/21/79	* SEISMIC 5/21/79	** TOTAL 5/25/79	SEISMIC 5/25/79	ALLOWABLE	TOTAL STRESS ALLOWABLE
High Pressure Coolant Injunction	HPCI-11	16, 14, 6	21386	19790	17214	15618	14528	12932	27000	54
Reactor Core Injunction Cooling	RCIC-196	1, 3/4	22706	19504	22504	19302	19184	15982	27000	71
Residual Heat Removal	RHR-41	3, 4	26802	20728	26824	20750	23255	17181	27000	86
Residual Heat Removal	RHR-199	4, 1, 1½, 3/4	23410	20242	23398	20230	19918	16750	27000	74
Reactor Core Isol. Cooling	RCIC-194	2, 1½, 1	24519	24118	24559	24156	20404	20001	27000	76

*Seismic stresses shown are obtained by multiplying the OBE Seismic Stresses by 2.

**Total stress (5/25/79) are based on:

$$\left(\frac{DBE}{2} \times 1.2\right) 1.38 + (\underset{5/21/79}{\text{Total Stress}} - \underset{5/21/79}{\text{Seismic}})$$

ATTACHMENT 2 - TABLE 2-2

PROB. NO.	SYSTEM	ISO/ SHEET NO.	LOCATION INS. OR OUTSIDE CONT.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI) TOTAL STRESS 5/21/79	SEISMIC (DBE) 5/21/79	TOTAL 5/25/79	SEISMIC 5/25/79	ALLOWABLE (1.8 S_h)	TOTAL STRESS/ ALLOWABLE
2	Primary Steam Condensate Drain Inside Dry Well (East) and (West)	128	In	2	24225	20022	29070	24867	27000	108
32	High Pressure Coolant Inj. (Main Pump to Barometer Cond.)	152	Out	1½	23242	22760	28750	28268	27000	106
34	High Pressure Coolant Inj. (Misc. Vents & Drains Booster Pump)	154	Out	3/4	23191	20780	28220	25809	27000	105
35	High Pressure Coolant Inj. (Turbine Exh.)	155	Out	2	23245	20990	28325	26070	27000	105
			Out	2	23346	21800	28622	27076	27000	106
			Out	½	23762	23476	29443	29157	27000	109
38	High Pressure Coolant Inj. (Misc. Vents, Test & Drains Lines)	158	Out	1	27538	25060	33602	31124	27000	124
			Out	3/4	24715	22760	30223	28268	27000	112
			Out	3/4	25923	22010	31249	27336	27000	116
46	Core Spray System (C.S. Min. Flow By-Pass Pump 2A)	39	Out	3	29154	26788	35636	33270	27000	132
			Out	3	25456	21860	30746	27150	27000	114
48	Core Spray System (RHR Conn. from C.S. Pump 2A)	105	Out	2	25235	22438	30655	27858	27000	114
	Core Spray System (RHR Conn. from C.S. Pump 2B)	105	Out	2	25235	22438	30655	27858	27000	114
54	Service Water Salt Water Supply to RHR, Service Water Pumps (South)	82	Out	20	29536	25810	35782	32056	27000	133

ATTACHMENT ? - TABLE 2-2 (Cont'd)

PROB. NO.	SYSTEM	ISO/ SHEET NO.	LOCATION INS. OR OUTSIDE CONT.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI) TOTAL STRESS 5/21/79	SEISMIC (DBE) 5/21/79	TOTAL 5/25/79	SEISMIC 5/25/79	ALLOWABLE (1.8 S_h)	TOTAL STRESS/ ALLOWABLE
55	Service Water System 6" Return Line from Pump Room Cooler 2A	106	Out	4	26139	25100	32213	31174	27000	119
56	Service Water System 6" Supply Header (South)	107	Out Out	6 4	24965 29907	17948 28212	29308 36734	22291 35039	27000 27000	109 136
57	Service Water System 6" Supply Header (North)	108	Out	2	24300	21352	29467	26519	27000	109
70	Reactor Water Clean-up R.W.C.U. Pump Suction	22	In	6	24303	17134	28449	21280	25940	110
80	Cont. Atmospheric Control Valve By-Pass Piping	211	Out	2	26825	24396	32729	30300	27000	121
81	Cont. Atmospheric Control Vent Purge Line From Drywell	212	Out	4	23948	21750	29212	27014	27000	108
83	Containment Venting	230	Out	½	25268	14896	28873	18501	27000	107
89	Instrument Air System Supply Line (North) West "L"	179	Out	2	24598	23348	30248	28998	27000	112
93	Instrument Air System Supply Header (North)	184	In	2	30632	29382	37742	36492	27000	140
95	Instrument Air System Pipe to Accum	189	In	1½	25719	24512	31651	30444	27000	117
96	Instrument Air System Supply Lines to Filters D-0005 and D-0006	190	Out	3/4	30507	29382	37617	36492	27000	139

ATTACHMENT 2 - TABLE 2-2 (Cont'd)

PROB. NO.	SYSTEM	ISO/ SHEET NO.	LOCATION INS. OR OUTSIDE CONT.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI) TOTAL STRESS 5/21/79	SEISMIC (DBE) 5/21/79	TOTAL 5/25/79	SEISMIC 5/25/79	ALLOWABLE (1.8 S_h)	TOTAL STRESS/ ALLOWABLE
97	Instrument Air System Outlet from RCVR at "18R" Supply West Col. "T"	181	Out	2	28127	26922	34642	33437	27000	128
98	Instrument Air Sys. Inner Air Supply Header Outer Air Supply Header	192	In	2	28344	27094	34900	33650	27000	129
99	Instrument Air System Recirc Pump 2B	201	In	3/4	25822	18666	30339	23183	25920	117
101	Instrument Piping, Piping at Temp. Equalizing D003B	206	In	3/4	24990	22410	30413	27833	27900	109
102	Instrument Piping Lines 2E21-701 & 702	207	In	3/4	21655	19622	26403	24370	26028	101
108	Nitrogen & Off Gas Services Bldg. Instr. Air Interruptable	309	Out	3/4	27170	24018	32982	29830	27000	122
110	RHR	545	Out	4	23765	18128	28151	22514	27000	104
113	RHR Drain to RW	548		4	23806	21690	29055	26939	27000	108
116	RHR Pumps 1A & 1B	605		2	24752	20878	29804	25930	27000	110
117	Service Water Sys.	606		6	26321	23760	32071	29510	27000	119
125	Instrument Air	690		3/4	26655	25529	32833	31707	27000	121
129	Cont. Atmos. Control Sys. Sup. Lines	709 710		8	15579	11038	29288	13709	27000	108
132	Service Water	716		1½	2314[illegible]	18610	27646	23113	27000	102

ATTACHMENT 3
AS-BUILT DRAWINGS

As a result of the NRC-I&E walk-through of approximately 67 pipe supports on safety related lines, four discrepancies were identified:

1. Isometric 17 High Pressure Coolant Injection main pump discharge line above el. 18'-9" data point 45 does not agree with piping drawing. Actual location of support is 9'-2" from valve F006 in lieu of 7.0' as shown on the analysis isometric.

   Comment: The analyzed location has been reviewed by stress analysist and confirmed that the actual placement of the support will have little or no effect on the results of analysis for the following reasons:

   1. The total maximum stress of the line is less than 50% of the code allowable stress. see attachment 2
   2. The placement of the support within approximately two pipe diameters of its analyzed position on this 14" Sch. 120 pipe will not adversely effect the analysis.

2. Isometric 20 Core Spray data point 101 is located approximately 1'-3" closer to valve F015A than shown on the isometric.

   Comment: Review by stress analysist confirms that since the data point is a snubber placing it closer to the valve is better than the original placement. In addition the new placement will have no adverse effect on the stress analysis since the new placement is in the same plane as analyzed.

3. Isometric 12 Reactor Core Isolation Cooling pump suction lines data point 272 vertical snubber is located on the opposite side of an elbow than is shown on the analysis isometric.

   Comment: Review by stress analysist confirm that placement has no effect on the stress analysis. The analysis program treats the elbow as a point in the model, therefore transfer from one side of an elbow to the other has no effect on the analysis results as long as the snubber acts in the required direction. Field check of the installation has verified that the snubber is acting in the correct (vertical) direction.

4. Isometric 18 Core Spray Pump suction line 2B, data point 236, is eleven inches closer to pump than shown on the analysis isometric.

   Comment: Review by stress analysist confirms that the location of the support within one pipe diameter will not adversely effect the stress analysis. In addition the support is a sliding dead weight support which has no effect for seismic support.

It should be noted that in the above cases it has been determined by a stress analysist that there is no adverse impact on the pipe stresses. However, Carolina Power and Light has committed to perform an as-built verification on all lines included in the reanalysis to increase the confidence that the as analyzed condition is consistent with the as-built condition. Thirty additional supports have been checked by field personnel and no additional problems have been found.

ATTACHMENT 2
CP&L Letter Dated May 22, 1985

BC/A-4

CP&L

Carolina Power & Light Company

MAY 2 2 1985

SERIAL: NLS-85-106

Director of Nuclear Reactor Regulation
Attention: Mr. D. B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
PIPING STRESS ANALYSES DAMPING VALUES

Dear Mr. Vassallo:

Pursuant to the Code of Federal Regulations, Title 10 Part 50.55a paragraph (a)(3), Carolina Power & Light Company (CP&L) hereby requests approval to utilize the damping curve developed by the Pressure Vessel Research Council (PVRC) in ASME Code Case N-411. Damping values extracted from the curve will be incorporated into seismic analyses for Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) events. The new damping values could be used for current piping modifications and future piping stress analyses as an option to the original damping values presented in the Final Safety Analysis Report (FSAR). The PVRC damping values will be used only for seismic response spectra analyses. They will not be applicable to time-history analyses.

If the PVRC damping values are approved for use at Brunswick, the following upgrades will also be incorporated when applying the new values:

1. A three-dimensional square root of the sum of the squares (SRSS) earthquake combination will be used in lieu of a two-dimensional SRSS combination.

2. Regulatory Guide 1.92 modal combinations accounting for closely-spaced modes will be used in lieu of a straight SRSS of all modes.

3. A rigid cutoff value of 33 Hz will be used in lieu of 20 Hz.

4. If, as a result of using the damping value curve presented in ASME Code Case N-411, piping supports are moved, modified, or eliminated, the expected increased piping displacements due to greater piping flexibility will be checked to assure that they can be accommodated and that there will be no adverse interaction with adjacent structures, components, or equipment.

The original FSAR criteria for piping analyses, including Regulatory Guide 1.61 values, and the proposed PVRC damping with the upgraded criteria presented above will be considered as valid options for pipe stress analyses and modification work at Brunswick. When performing an analysis, the damping values taken from the curve presented in Code Case N-411 are only to be used with the upgraded criteria, not with the original FSAR criteria. That is, no analysis will combine damping values and criteria which are not consistent.

8505290481 6pp.

Marked up copies of the affected FSAR tables and text (to be incorporated into the next FSAR revision) as well as the PVRC figure to be incorporated are attached.

Carolina Power & Light Company has reviewed this request in .ccordance with 10CFR170.12, and a check for $150 in payment of the required fee is enclosed.

Should you have any questions regarding this request, please contact Mr. Sherwood R. Zimmerman at (919) 836-6242.

Yours very truly,

A. B. Cutter - Vice President
Nuclear Engineering & Licensing

ABC/RWS/mf (1339RWS)
Attachment
cc: Mr. L. W. Garner (NRC-BNP)
Dr. J. Nelson Grace (NRC-RII)
Mr. M. Grotenhuis (NRC)

TABLE 3.7.1-1

DAMPING FACTORS

ITEM	PERCENT OF CRITICAL DAMPING	
	OBE	DBE
Reinforced Concrete:		
(a) Primary Containment Structure	4	7
(b) Reactor Building and other Class I Structures	4	7
Steel Structures and Assemblies:		
(Reactor Building & other Class I structures)		
(a) Bolted or Riveted	5	10
(b) Welded	2	5
Vital Piping	0.5*	2*
Equipment	1	2
Soil - Structure Interaction Damping	4	7

* For final reconciliation of pipe stress analysis or piping system backfits, damping values as defined in ASME Code Case N-411 (Figure 3.7.1.5) may be utilized for both OBE and DBE.

3.9.2.1.3 Piping Seismic Analysis

The piping systems were dynamically analyzed using the "lumped mass response spectrum method" of analysis. For each of the piping systems, a mathematical model consisting of lumped masses at discrete joints connected together by weightless elastic elements was constructed. Valves were also considered as lumped masses in the pipe, and valve operators eccentricity was considered (Reference 3.9.1-1). Stiffness matrix and mass matrix were generated and natural periods of vibration and corresponding mode shapes were determined. Input to the dynamic analyses were the applicable 0.5 percent damped acceleration response spectra.

Increased damping values may have been applied for final stress reconciliation or piping system backfits in accordance with ASME Code Case N-411 (Figure 3.7.1.5). If so, the following criteria were also used:

a) A three-dimensional square root of the sum of the squares (SRSS) earthquake combination in lieu of a two-dimensional SRSS combination.

b) Regulatory Guide 1.92 modal combinations accounting for closely-spaced modes in lieu of a straight SRSS of all modes.

c) A rigid cut off value of 33 Hz in lieu of 20 Hz.

d) A pipe displacement check performed if existing pipe supports were moved, modified, or eliminated.

The increased flexibility of the curved segments of the piping systems was considered. The results for earthquakes acting in the X and Y (vertical) directions simultaneously and Z and Y directions simultaneously were computed separately. The maximum responses of each mode were calculated and combined by the absolute sum. The response thus obtained was combined with the results produced by other loading conditions to computer the resultant stresses.

TABLE 3.7.3-1
CRITICAL DAMPING FOR STRUCTURES, PIPING, AND EQUIPMENT

Item	PERCENT OF CRITICAL DAMPING OBE	DBE
1. Concrete Structures	4%	7%
2. Piping	1/2 %*	2% *
3. Equipment		
a. Pumps		
b. Motors	1%	2%
c. Switchgear		
d. Exchangers		
e. Tanks		
f. Batteries and Racks		
g. Cable ray Systems		
h. Diesel Generator Units		
i. Others		
4. Cranes	4%	7%

* For final reconciliation of pipe stress analysis or piping system backfits, damping values as defined in ASME Code Case N-411 (Figure 3.7.1.5) may be utilized for both OBE and DBE.

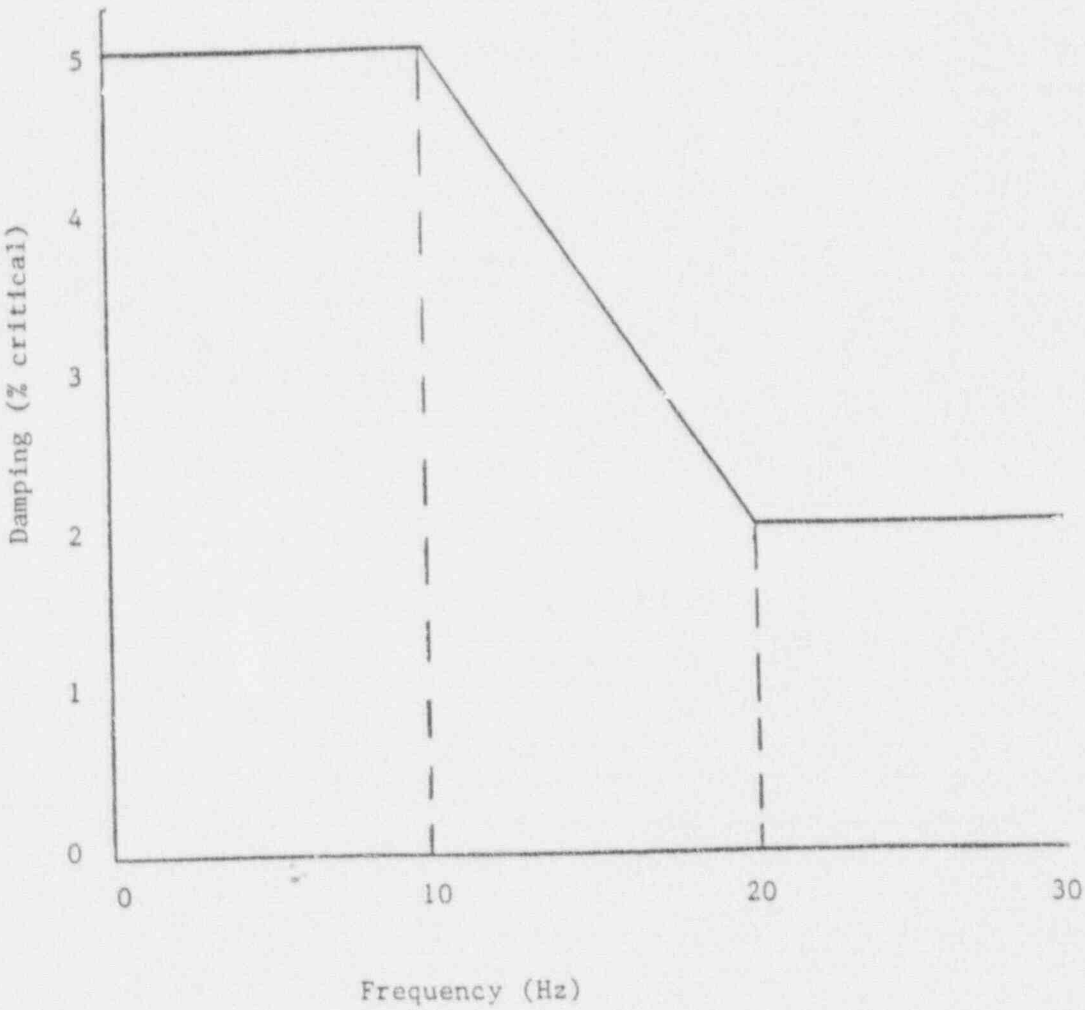

Figure 3.7.1.5

Damping Value for Seismic Analysis of Piping

(Applicable to both OBE & SSE, Independent of Pipe Diameter)

ATTACHMENT 3

NRC Letter Dated August 28, 1985

RECEIVED SEP 0 5 1985

9/6/85

NLS-85-568

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 28, 1985

Docket Nos. 50-325/324

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: USE OF ASME CODE CASE N-411

Re: Brunswick Steam Electric Plant, Units 1 and 2

By letter dated May 22, 1985, you requested approval to utilize the damping curve developed by the Pressure Vessel Research Council in ASME Code Case N-411. Code Case N-411 damping values would be used as an options to the original damping values presented in the Final Safety Analysis Report (FSAR). Your letter dated May 22, 1985 submitted the information and commitments. We have reviewed your request pursuant to 10 CFR Part 50.55a paragraph (a)(3) and find that, although Code Case N-411 has not been listed in Regulatory Guide 1.84 or 1.85, it is acceptable based on the information and commitments provided in your application. Our related Safety Evaluation is enclosed.

Sincerely,

for Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

~~8509060344~~ 1p.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO USE OF ASME CODE CASE N-411

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated May 22, 1985, the Carolina Power & Light Company (CP&L, the licensee), pursuant to the Code of Federal Regulations, Title 10 Part 50.55a paragraph (a)(3), requested approval to utilize the damping curve developed by the Pressure Vessel Research Council (PVRC) in ASME Code Case N-411. It further stated that damping values extracted from the curve will be incorporated into seismic analyses for Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) events; that the new damping values could be used for current piping modifications and future piping stress analyses as an option to the original damping values presented in the Final Safety Analysis Report (FSAR); that the PVRC damping values will be used only for seismic response spectra analyses; and that they will not be applicable to time-history analyses.

2.0 EVALUATION

We have completed our review of the CP&L request for authorization to use the damping values in ASME Code Case N-411 for application in the response spectrum seismic analysis for current modifications and future stress analyses of piping systems at the Brunswick Steam Electric Plant, Unit 1 and 2 as discussed in the licensee's letter NLS-85-106 dated May 22, 1985. The damping values in Code Case N-411 could be used as an option to the original damping values presented in the FSAR.

Code Case N-411, "Alternate Damping Values for Seismic Analysis of Piping Section III, Division 1, Class 1, 2 and 3 Construction" is a conditionally acceptable Code Case and is approved by the staff for specific plant applications pending revision of Regulatory Guide 1.61. Utilities wishing to use this Code Case shall submit in their request the following information or commitments:

(a) Commit to use the case for piping systems analyzed by response spectrum methods and not those using time-history analysis methods.

(b) Indicate if the case is to be used for new analyses or for reconciliation work and for support optimization.

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(c) Due to the increased flexibility of the system commit to check all system predicted maximum displacements for adequate clearance with adjacent structures, components and equipment, and that the mounted equipment, can withstand the increased motion.

(d) When the alternate damping criteria of this Code Case are used, they will be used in their entirety in a given analysis and shall not be a mixture of Regulatory Guide 1.61 criteria and the alternate criteria of this Code Case.

CP&L has complied with these commitments in the letter of May 22, 1985. Therefore, since the commitments with respect to the Code Case N-411 are documented in the referenced letter, the staff finds the licensee's request to use Code Case N-411 acceptable for use at the Brunswick Steam Electric Plant, Units 1 and 2 in the response spectrum seismic analysis of piping systems.

In an attachment to the letter of May 22, 1985, the licensee has included marked up copies of the affected FSAR Tables 3.7.1-1 and 3.7.3-1, page 3.9.2-6 and figure 3.7.1.5 which will be included in the next FSAR revision. The staff finds this material acceptable.

In addition, the licensee has stated that the following upgrades will also be incorporated when applying the damping values of Code Case N-411:

(1) A three-dimensional square root of the sum of the squares (SRSS) earthquake combination will be used in lieu of a two-dimensional SRSS combination.

(2) Regulatory Guide 1.92 modal combinations accounting for closely-spaced modes will be used in lieu of a straight SRSS of all modes.

(3) A rigid cutoff value of 33 Hz will be used in lieu of 20 Hz.

The use of these upgrades for use at the Brunswick Plant, Units 1 and 2 are consistent with design methodology being accepted by the staff for plants currently undergoing licensing review and is thus acceptable.

3.0 CONCLUSION

Based on our review and the above discussion, we find that the CP&L request to use the damping curve in ASME Code Case N-411 is approved as requested.

Principal Contributor: R. Kirkwood

Dated: August 28, 1985

ATTACHMENT 4

NRC Meeting Summary Dated June 12, 1979

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Recd 6/21/7
LBS
DLBensinger

JUNE 1 2 1979

Docket Nos. 50-325
and 50-324

MEMORANDUM FOR: Thomas A. Ippolito, Chief, ORB #3, DOR

FROM: Roby B. Bevan, Project Manager, ORB #3, DOR

SUBJECT: MEETING SUMMARY

A meeting was held with representatives of Carolina Power and Light Company (CP&L) and their contractors in Bethesda, Maryland on May 30, 1979. The purpose of the meeting was to provide the staff with the current status of the seismic pipe stress reanalysis being conducted for Brunswick Units 1 and 2 required for IE Bulletin 79-07, and to review the licensees program of identifying and modifying overloaded pipe supports and hangers found incident to the reanalysis.

A list of attendees at the meeting is enclosed. Also enclosed is a CP&L letter dated May 29, 1979, with attachments, addressing the items to be discussed at the meeting. On May 25, 1979, CP&L informed DOR staff by telephone that, in the course of their reanalysis for IE Bulletin 79-07, they had found six supports stressed beyond code allowable. On learning of this situation, they shut down both units and informed the DOR staff and I&E of their action. By telephone on May 28, 1979, CP&L requested this May 30, 1979 meeting.

At the meeting, CP&L explained that the six supports had not been adequately designed to account for the torsion stress on I-beams, a condition that might exist for some other supports in the plant. They had, therefore, initiated a program to investigate all pipe supports on all safety related systems. This program would be completed and all modifications made, before the two units would begin a return to power operation.

CP&L also described modifications in their reanalysis program, both for already-analyzed and for to-be-analyzed cases, to accommodate the staff position regarding the BSEP Final Safety Analysis Report commitment to use absolute summation techniques with the two dimensional model. The Staff agreed to review their proposal, and informed them by phone on June 1, 1979 that their method is an acceptable one.

CP&L has previously committed (letter dated May 22, 1979) to verify as-built dimensions for all safety related piping in and support systems. They reported that this program of "walking the lines" is in progress. During the current shutdown, all lines inside containment will be verified. The reanalysis of the lines is being done concurrent with the as-built verification. CP&L described procedures for handling deviation of as-built from as-analyzed.

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In response to a previous staff inquiry regarding the location of the postulated LOCA pipe rupture relative to the highest stress point, CP&L informed the staff that the piping design did not locate the break at the highest stress point, but instead analyzed for a double ended or a longitudinal break to occur at any point on the line, both inside and outside containment.

In response to a previous staff request, CP&L discussed their program of management controls and reporting criteria to assure appropriate licensee action when problems are identified in the continuing program of pipe and support reanalysis.

CP&L expressed their intention to return to operation in a few days. They therefore requested (and we agreed) to meet with us again on June 4, 1979 to review their status. Specifically, they would at that time provide the current status of their pipe and supports analyses, identifying modifications yet to be completed (if any), and verifying the as-built condition of all lines procedures for handling deviations found in their ongoing as-built verification program.

Roby Bevan
Roby B. Bevan, Project Manager
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
As stated

ATTACHMENT 5

NRC Letter Dated January 30, 1985

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Doc 3
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RECEIVED FEB 0 4 1985
NLU-85-67

January 30, 1985

Docket Nos. 50-325/324

Mr. E. E. Utley
Executive Vice President
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: MASONRY WALL DESIGN, IE BULLETIN 80-11

Re: Brunswick Steam Electric Plant, Units 1 and 2

On May 8, 1980 we issued IE Bulletin 80-11. We have reviewed your responses as listed on page 18 of the Technical Evaluation Report (TER), attached to the enclosed Safety Evaluation (SE). In addition, we have reviewed your response dated December 21, 1984. Based on our review, we find that the proposed modifications are in compliance with the staff acceptance criteria. We find the schedule for completion of the fix designs proposed in your letter to be acceptable. We are aware that the schedule for the actual modification work will be worked out in your 5-year program for the Brunswick plant.

This concludes our review of your response to IE Bulletin 80-11.

Sincerely,

D. B. Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO MASONRY WALL DESIGN, IE BULLETIN 80-11

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325/324

Introduction and Background

IE Bulletin 80-11 regarding Masonry Wall Design was issued on May 8, 1980. The Carolina Power & Light Company (CP&L) responded with letters dated July 7, November 5 and 25, and December 9, 1980. In response to requests for additional information, dated August 2, 1982 and February 21, 1984, responses dated July 29, 1983 and April 27, 1984 were submitted. A final response dated December 21, 1984 was received and reviewed after the Technical Evaluation Report (TER) Attachment 1 was completed and therefore is not included in the reference list on page 18 of that report.

The findings reported in this Safety Evaluation (SE) are based on the attached TER, prepared by Franklin Research Center (FRC) as a contractor to NRC, and the NRC review of the December 21, 1984 submittal. This TER contains the details of construction techniques used, technical information reviewed, acceptance criteria, and technical findings with respect to masonry wall construction at the Brunswick units. The staff has reviewed this TER, concurs with its technical findings and it is hereby incorporated into this SE. It is noted that on page 3 of the April 27, 1984 response (RAI 5b) that the NRC position paper on the energy balance technique is referenced. In view of the CP&L decision not to use this technique, this position paper was not needed.

The staff summary and evaluation of the major technical conclusions are included in this SE.

Evaluation and Conclusion

There were 87 masonry walls identified at the Brunswick units. The licensee qualified sixty of these safety-related walls by using the working stress approach which is in compliance with the staff acceptance criteria. The licensee planned to modify ten walls, which were originally qualified by nonlinear techniques, by providing steel pilasters, steel grating to restrain walls, and steel angles installed at their boundaries. These modifications would render these ten walls in compliance with the staff acceptance criteria.

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The December 21, 1984 submittal, received after the TER was completed, indicated the results of the evaluation of the additional 17 walls referred to in the TER. That submittal also indicated that a field inspection determined that two of the 87 walls were "Nonsafety Related" rather than "Safety Related." One of the latter walls was listed among the ten walls to be upgraded and one was among the last 17 to be evaluated. Five among the 17 walls were found to be within allowable limits and the remaining 11 will require additional reinforcement to bring them within allowable limits. That makes a total of 20 walls requiring modification.

In summary, there was a final total of 85 safety-related masonry walls identified at the Brunswick plant. Sixty-five were found to be within acceptable limits and 20 will require modification. The design fix for nine of the 20 walls is completed, the design fixes for the remaining 11 are scheduled to be completed in July 1985. The schedule for the actual modifications will be in the 5-year plan currently under review.

Based on the above findings, the staff concludes that, Items 2(b) and 3 of IE Bulletin 80-11 have been fully implemented at the Brunswick units and that there is reasonable assurance that the safety-related masonry walls at the Brunswick units will withstand the specified design load conditions without impairment of (a) wall integrity or (b) the performance of required safety functions.

Principal Contributor: N. Chokshi and M. Grotenhuis

Dated: January 30, 1985