



Carolina Power & Light Company

File: NG-3514(B)

May 21, 1979

SERIAL: GD-79-1342

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:

This letter supplements our letters of April 24, 1979 and May 16, 1979 on the Carolina Power and Light Company's response to IE Bulletin 79-07. It addresses the specific requests for additional data that were discussed in our meeting with the NRC on May 16, 1979. This letter contains the results of the seismic pipe stress reanalyses completed to date, a compilation of the original seismic stress data for the pipe yet to be reanalyzed, the schedule for completing the reanalysis, and the conclusions reached from the evaluation of the technical information available.

It was determined that algebraic summations were used in the seismic stress analyses portion of the computer code that was used in the original analysis. Realizing that a total reanalysis could not be completed in the time frame required to respond to the bulletin, ten (10) lines were selected for reanalysis in order to determine the effect of algebraic summation. The computer code currently in use by United Engineers & Constructors, a proprietary copy of which was given to you on May 16, 1979, does not use algebraic summations. The criteria for selection of the ten (10) lines was based on line function, size, and ability to project adequacy for other lines of similar function and size. This criteria is explained in more detail on Attachment 1. Subsequent to our May 16th meeting, additional lines have also been reanalyzed. As discussed in our meeting these lines were selected on the basis of completing the larger pressure boundary and other safety systems piping. In addition other lines having high stress values in the original stress analysis were reanalyzed. The results of these 39 lines are tabulated on Attachment 2. This table shows the maximum total stress and the maximum seismic stress component of each line for the original analysis and the reanalysis. In all cases, the calculated total stresses for the design basis earthquake are less than the allowable.

Another item discussed at our May 16th meeting was the loads on the pipe supports as a result of the reanalysis. These new loads are being evaluated and results for the first ten (10) lines will be filed the week of May 28.

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May 21, 1979

A total of 147 stress analyses problems have been identified for Unit 2. Most lines are identical between Unit 2 and Unit 1 so the stress analyses are applicable to both units. However, 24 cases where the lines are different were identified and will be reanalyzed separately for each unit. The total number of computer problems for Units 1 and 2 is 171 (with 39 already completed). Attachment 3 shows those lines that have not yet been reanalyzed with the total stress and seismic stress component for DBE from the original analysis. Also, the reanalysis priority category is shown and is based on the reanalysis priority criteria shown on Attachment 4. The schedule for completion of the reanalysis is given on Attachment 5.

There was some discussion on the confidence that the design analyzed for stress was the "as built" design. The quality assurance program used during construction, and in effect today, uses a "walk off" of each as installed system or modification. This inspection identifies discrepancies between system design and installation. These discrepancies are evaluated by the architect/engineer for acceptability. In addition, for piping and support differences, a stress analyst ascertains whether or not the system must be reanalyzed. If not, the isometric drawing and associated computer data deck remain unchanged. If the analyst concludes that the differences between "as-designed" and "as-built" would require reanalyses, the isometric is revised, the data deck is changed as necessary, and the stress analysis rerun. Whether the stress analysis was rerun or not, the design engineering drawing is changed to reflect the as-built configuration. This scenario is shown on the block diagram on Attachment 6.

After a thorough review of the original seismic piping design criteria, and the reanalysis of selected significant safety class piping, CP&L has concluded that Units 1 and 2 at Brunswick Steam Electric Plant can be safely operated without undue risk to the health and safety of the public. This conclusion is based on:

1. Evaluation of the conservative design margins provided in the original piping design, as discussed in Attachment 7.
2. The satisfactory results of the reanalysis of 39 selected major lines. The reanalysis shown in Attachment 2 represents approximately 60% of the lines 2½" and over in the pressure boundary and approximately 40% of the lines 2½" and over in the emergency core cooling systems.
3. An investigation of the as built versus as analyzed conditions.

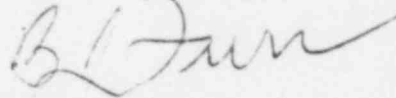
Mr. T. A. Ippolito  
SERIAL: GD-79-1342

-3-

May 21, 1979

4. The aseismic nature of the site as confirmed by an extensive on site and regional seismic program. See Attachment 10.
5. The accelerated schedule of pipe reanalysis. The final report will be filed on July 21, 1979. Interim reports of results of reanalysis will be filed at the completion of the reanalysis of each category.

Yours very truly,



B. J. Furr  
Manager,  
Generation

cc: Mr. James P. O'Reilly  
NRC Region II  
Office of Inspection & Enforcement

May 21, 1979

ATTACHMENT 1

INITIAL REANALYSIS SELECTION CRITERIA

<u>CRITERIA</u>	<u>BASIS</u>
Function	Major operating system with Primary Pressure Boundary (Mainsteam, Feedwater and SRV Line) or Safe Shutdown System (HPCI, RHR, Core Spray)
Size	Larger lines generally have higher stresses and hence less margin to the allowable
Similarity	Select the line which had the highest stresses from original analyses for those groups of lines with similar function (1 of 4 Mainsteam, 4 of 11 SRV, 1 of 2 Feedwater, 2 of 4 HPCI, 2 of 14 RHR and Core Spray)



ATTACHMENT 2  
PIPE STRESS REEVALUATION  
SUMMARY

SYSTEM NAME	ISO NO.	LINE SIZE (NPS)	EMERGENCY CONDITION (PSI)				
			ORIGINAL TOTAL	ORIGINAL SEISMIC	NEW TOTAL	NEW SEISMIC	ALLOWABLE
Main Steam	MS-15B	24	10724	3942	10640	3858	27000
Safety/Relief Valve	SRVL-121	10 & 6	23012	12280	21910	11180	27000
Safety/Relief Valve	SRVL-122	10 & 6	19685	15800	24439	13352	27000
Safety/Relief Valve	SRVL-237	10 & 6	20432	12004	24588	16160	27000
Safety/Relief Valve	SRVL-125	10 & 6	24270	13347	24316	20270	27000
Feedwater	FW-16	18 & 12	18007	12420	20028	13296	27000
Residual Heat Removal	RHR-6	20	19406	13582	12644	6820	27000
Emergency Spray	CS-24	10	16952	10076	14366	7490	27000
High Press Cool Injct	HPCIS-17	14	12200	6446	12502	6748	27000
High Press Cool Injct	HPCIS-510	14 & 12 & 10	12004	7994	12092	8082	27000
High Press Cool Injct	HPCIS-10	14 & 12 & 10	9733	3886	11152	5530	27000
Residual Heat Removal	RHR-1	24 & 20	24094	18584	17972	14366	27000
Residual Heat Removal	RHR-2	20 & 16 & 12	13309	7654	11471	5948	27000
Residual Heat Removal	RHR-5	24	9848	3896	9514	2960	27000
Residual Heat Removal	RHR-25	4 & 6	18558	12904	18530	12876	27000
Nuclear Steam Supply	NSS-14	24 & 10	14745	8446	16335	10036	27000
Safety/Relief Valve	SRVL-124	6 & 10	25536	15928	25984	16376	27000
Safety/Relief Valve	SRVL-126	6 & 10	23361	18000	22197	17422	27000
Residual Heat Removal	RHR-52	14 & 12	23271	17936	19539	14204	27000
Reactor Core Isolat. Cool	RCIC-21	3	7603	3588	7601	3586	27000
Resid Heat Rem Drain Line	RHR-173-B	1½	3808	2186	3838	2216	27000
Residual Heat Removal	RHR-28	20 & 16 & 12	15298	8814	13626	7142	27000
Nuclear Steam System	NSS-15 Line C	24	10974	4420	10458	3904	27000
Nuclear Steam System	NSS-120 (15C)	10 & 6	19443	8902	17899	8298	27000
High Press Cool Injct.	HPCIS-4	3 & 6 & 10 & 12	23609	20876	25481	22748	27000

ATTACHMENT 2 Cont'd

SYSTEM NAME	ISO NO.	LINE SIZE (NPS)	EMERGENCY CONDITION (PSI)				
			ORIGINAL TOTAL	ORIGINAL SEISMIC	NEW TOTAL	NEW SEISMIC	ALLOWABLE
Nuclear Steam System	NSS-123 (15C)	6 & 10	21027	16098	18577	16782	27000
Nuclear Steam System	NSS-187 (15C)	10 & 6	21856	11337	23596	16424	27000
Residual Heat Removal	RHR-42	12 & 14	18116	12976	17480	12340	27000
Residual Heat Removal	RHR-3	14 & 16 & 20 & 24	25317	18328	23379	15590	27000
Residual Heat Removal	RHR-13	4 & 8 & 14	12532	10018	12620	10106	27000
Residual Heat Removal	RHR-59	4 & 6 & 10	12664	9970	13344	10650	27000
* Residual Heat Removal	RHR-60	4 & 6 & 10	34618	33658	32971	32012	27000
Residual Heat Removal	RHR-168	1	23580	22658	26910	26198	27000
Residual Heat Removal	RHR-61	4 & 6 & 3/4	21117	17038	19393	15314	27000
High Pressure Coolant Injunction	HPCI-11	16 & 14 & 6	21386	19790	17214	15618	27000
Reactor Core Injunction Cooling	RCIC-196	1 & 3/4	22706	19504	22504	19302	27000
Residual Heat Removal	RHR-41	3 & 4	26802	20728	26824	20750	27000
Residual Heat Removal	RHR-199	4 & 1 & 1½ & 3/4	23410	20242	23398	20230	27000
Reactor Core Isol. Cooling	RCIC-194	2 & 1½ & 1	24519	24118	24559	24156	27000

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

\* Total stresses are within allowable using applicable factor to compute DBE stress from OBE stresses.

Job Order Number

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRIORITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>h</sub> )	SEISMIC/ALLOWABLE		
1	Nuclear Steam Supply (Vent & Inst. Line Above Bulkhead)	32	In	3/4	11282	5198	25900	20	1	
			In	4	3963	354	27000	1		
			In	2	10458	7672	27000	28		
			In	1	4316	1672	27000	6		
2	Primary Steam Condensate Drain Inside Dry Well (East) & West)	87, 128	In	3	12237	9380	27000	34	1	
			In	2	24225	20022	27000	74		
			In	1 1/2	8345	5838	27000	21		
			In	1	15171	12006	27000	44		
76	Control System - NSS (Inst. Sensing Line)	7	In	3/4	17945	12954	25900	50	2	
3	Residual Heat Removal Minimum Flow By-Pass RHR Pumps 2A & 2C	43	Out	4	6434	3732	27000	13		
			Out	4	3288	1762	27000	6		
			Out	3	21856	16332	27000	60		
4	Residual Heat Removal Torus Spray (South)	44	Out	6	9536	6784	27000	25		
5	Residual Heat Removal Drain Piping - RHR Pumps 2B, 2D	45	Out	4	14859	12378	27000	45		
			Out	4	18562	12462	27000	46		
6	Residual Heat Removal Drain Piping - RHR Pumps 2A, 2C	46	Out	4	21248	17250	27000	63		
7	Residual Heat Removal (Cross Conn. Drain Piping RHR Pumps 2A, B, C, D & RHR Drain Conn. to Rad-waste)	47, 48	Out	4	18944	15888	27000	58		
			Out	4	20325	17512	27000	64		
8	Residual Heat Removal Cross Conn. Drain Piping RHR Pumps 2A, B, C, & D South Co. 20R	65	Out	4	3335	928	27000	3		

FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 2 of 16

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# FINAL STRESS REPORT TABULATION

Sheet No. 3 of 16

Job Order Number

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRI- RITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>h</sub> )	SEISMIC/ ALLOWABLE		
15	Residual Heat Removal & Fuel Pool Colling & Filtering (RHR Suct. to Skimmer Tanks 2A, 2B)	64	Out Out	8 8	17652 10658	15652 7774	27000 27000	57 28	6	
16	Residual Heat Removal (Exchgr 2B to RCIC Suction)	31	Out Out Out Out	4 6 4 3	21014 576 1553 7331	17402 4396 13516 6696	27000 27000 27000 27000	64 16 50 24	2	
17	Reactor Core Isolation Cooling (Steam Supply to RCIC Turb.)	33	Out Out	3 3	8337 15615	5668 12356	27000 27000	20 45		
18	Reactor Core Isolation Cooling (RCIC Turbine Steam Exhaust)	34	Out	8	21108	16218	27000	50		
19	Reactor Core Isolation Cooling (RCIC Pump Disch-Bel. 10'6" 49, Above 10'6", 35)	35, 49	Out Out Out Out Out	4 4 2 2 1	12793 9520 18881 4829 3315	6402 6134 13096 514 586	27000 27000 27000 27000 27000	23 22 48 1 2		
20	Reactor Core Isolation Cooling (Test Conn. to HPCI)	50	Out	4	7221	2900	27000	10	3	
21	Reactor Core Isolation Cooling (RCIC Conn. to Feedwater)	63	Out Out	4 4	15098 13357	9488 6534	27000 26100	35 25		
22	Reactor Core Isolation Cooling RCIC Pump Disch. Line	66, 67	Out	4	12805	6932	27000	25		



# FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 4 of 16

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRIOR- ITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>h</sub> )	SEISMIC/ ALLOWABLE		
23	Reactor Core Isolation Cooling System (Misc. Lines at Turbine)	92B, 92C	Out	1	10999	5938	27000	21	5	
			Out	2	4350	2888	27000	10		
			Out	1/2	21144	18498	27000	68		
			Out	3	8015	5410	27000	20		
			Out	3/4	12427	10438	27000	38		
24	Reactor Core Isolation Cooling Drain From Turbine Inlet	161	Out	1	11796	8352	27000	30		
			Out	3/4	7972	5024	27000	18		
			Out	1/2	1855	1260	27000	4		
25	Reactor Core Isolation Cool. Barometric Condensate Pump Disc. to Suppression Chamber	164	Out	2	5559	4638	27000	17		
			Out	1 1/2	3303	3128	27000	11		
			Out	3/4	4471	3552	27000	13		
26	Reactor Core Isolation Cool. Turbine Drains & Relief Lines	195	Out	1	16705	15826	27000	58		
27	Reactor Core Isolation Cool. (RCIC Pump Suction Lines)	12	Out	6	19870	16526	27000	61	3	
			Out	6	4391	150	27000	0		
			Out	6	16311	12594	27000	46		
28	High Press. Coolant Inj. (Exhaust Steam Line)	9	Out	24	10395	4268	27000	15	2	
			Out	20	11629	7672	27000	28		
			Out	18	3707	1680	27000	6		
			Out	16	4467	614	27000	2		
29	High Pressure Coolant Inj. (Misc. Vents & Drains Main Pump)	151	Out	3/4	3438	1491	27000	6	5	
			Out	3/4	2570	770	27000	3		
			Out	3/4	4560	1056	27000	4		
			Out	3/4	2404	522	27000	2		
			Out	3/4	4455	2566	27000	10		
			Out	3/4	5920	3706	27000	14		
			Out	3/4	3753	1272	27000	5		
			Out	3/4	3398	1258	27000	5		
			Out	3/4	3127	567	27000	2		
			Out	3/4	2303	254	27000	1		





FINAL STRESS REPORT TABULAT

Job Order Number

Sheet No. 6 of 16

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Coat.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRI- ORITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>H</sub> )	SEISMIC/ ALLOWABLE%		
38	High Pressure Coolant Inj. (Misc. Vent, Test & Drains Lines)	158	Out	1	27538	25060	27000	93	5	Note 1
			Out	3/4	24715	22760	27000	84		
			Out	3/4	25923	22010	27000	81		
			Out	3/4	3896	928	27000	3		
39	High Pressure Coolant Inj. (Piping for Relief, Test & Valve Stem Drains)	159	Out	1	2132	234	27000	0		
			Out	1	16518	13414	27000	49		
40	Core Spray & Gravity Condensate System (Core Spray Pump Suction 2A)	18	Out	14	6901	4898	27000	18	2	
			Out	14	2073	14	27000	0		
			Out	12	19617	14674	27000	54		
41	Core Spray & Gravity Condensate System (Core Spray Pump Suction 2B)	18	Out	14	6901	4898	27000	18		
			Out	14	2073	14	27000	0		
			Out	12	19617	14674	27000	54		
42	Core Spray Pump Discharge (South)	19	Out	12	21811	16624	27000	61		
			Out	10	10290	5416	27000	20		
			Out	10	14747	9748	27000	36		
43	Core Spray Pump Discharge (North)	20	Out	10	11112	6438	27000	23		
			Out	10	7188	3120	27000	11		
			Out	12	14673	9400	27000	34		
44	Core Spray Pump Discharge (South)	22	Out	10	19272	13462	27000	49		
			Out	10	9832	3826	27000	14		
			Out	12	17062	11662	27000	43		
45	Core Spray Pump Discharge (North)	26	Out	10	15044	9218	27000	34	2	
			Out	10	9124	3316	27000	12		
			Out	12	22547	16786	27000	62		
46	Core Spray System (C.S. Mix. Flow By-Pass Pump 2A)	39	Out	3	29154	26788	27000	99	2	Note 1
			Out	3	25456	21860	27000	81		

FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No.7 of 16

[illegible]

FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 8 of 16

[illegible]

FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 9 of 16

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRIOR- ITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>H</sub> )	SEISMIC/ ALLOWABLE		
60	Service Water System Return From Motor Coolers R.H.R. S.W. Pumps 2A, B, C, D	142A  142B 142C	Out Out Out Out	1½ 3 1½ 1½	17673 1128 22372 11892	15660 802 20844 10312	31500 31500 31500 31500	49 1 66 32	4	
61	Service Water Supply and Return Headers At RHR Pump Room Coller 2A	162	Out Out Out Out Out Out Out Out	6 6 3 2½ 6 6 3 2½	1595 1373 1095 2285 2852 2127 1678 2217	37 180 158 648 312 296 160 458	27000 31500 31500 31500 27000 31500 31500 31500	0 0 0 2 1 0 0 1	4	
62	Service Water Supply and Return Headers At RHR Pump Room Coller "2B"	163	Out Out Out Out Out Out Out Out	6 6 3 2½ 6 6 3 2½	1567 1948 1825 2445 4390 3405 5305 8209	490 770 816 1370 3358 2392 4488 7330	27000 27000 27000 31500 27000 27000 27000 31500	1 2 3 4 12 8 16 23	4	
63	Service Water RHR Pumps 2B, 2D	214	Out Out Out Out Out Out Out Out	2 1½ 1 ¾ 2 1½ 1 ¾	9846 7102 5632 18217 4304 6256 18556 17955	8568 4740 3474 17338 3660 5630 17610 17568	32400 32400 32400 32400 32400 32400 32400 32400	26 15 11 54 11 17 54 54	6	
64	Service Water RHR Pumps 2A, 2C	215	Out Out Out Out Out Out Out Out	¾ 1 1½ 2 ¾ 1 1½ 2	19128 11897 15043 6718 5735 16717 9702 8440	15778 9012 11032 2708 4784 15180 7978 7132	33660 33660 33660 33660 33660 33660 33660 33660	46 26 32 8 14 45 23 21	6	

# FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 10 of 16

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRIOR- ITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>H</sub> )	SEISMIC/ ALLOWABLE		
65	Service Water Well Water Conn. To Vital Service Header	216	Out	1½	10090	8214	31500	26	6	
			Out	1½	2840	1182	31500	3		
			Out	2	8089	6460	31500	20		
66	Service Water C.S. Pump Room Cooling Units 2C, 2D Supply and Return	217	Out	2½	1098	660	31500	2	4	
			Out	2	5354	1772	31500	5		
			Out	2½	1873	1304	31500	4		
			Out	2	7731	2578	31500	8		
			Out	1	3996	3730	31500	11		
			Out	4	1389	320	31500	1		
			Out	2½	675	196	31500	0		
			Out	2	2554	1576	31500	5		
			Out	1	649	384	31500	1		
			Out	2½	1126	680	31500	2		
			Out	2	6584	1598	31500	5		
			Out	1	1668	1404	31500	4		
			Out	4	1358	306	31500	0		
67	Service Water RHR Service Water Pumps 2A, 2B, 2C, 2D	218	Out	2	1559	746	33800	2	6	
			Out	1½	18318	15052	33800	44		
68	Service Water Supply and Return Unit 1 & 2	300	Out	14	4791	1262	27000	5	4	
		301	Out	10	14310	10898	27000	40		
		302	Out	4	7539	4284	33228	16		
			Out	3	14175	10932	33228	33		
			Out	2½	10823	7610	33228	23		
			Out	2	15776	12572	33228	38		
			Out	1½	16182	13020	33228	39		
			Out	1½	10466	7318	33228	22		
69	Service Water (Cooling Water Return - South)	62	Out	20	4515	200	27000	0	4	
			Out	16	5879	3318	27000	12		
			Out	16	4412	1822	31500	5		
			Out	16	9875	3948	31500	12		
			Out	10	14230	7256	31500	23		
70	Reactor Water Clean-up R.W.C.U. Pump Suction	22	In	6	24303	17134	25940	66	1	
71	Reactor Water Clean-up (Reactor Drain & Conn. To Reactor Water Clean-up	132	In	2	9690	6548	27000	24	5	
			In	2	17800	13618	25920	52		



FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 11 of 16

[illegible]

FINAL STRESS REPORT TABULAT

Job Order Number

Sheet No. 12 of 16

[illegible]



FINAL STRESS REPORT TABULAT

Job Order Number

Sheet No. 13 of 16

[illegible]

FINAL STRESS REPORT TABULATION

Job Order Number

Sheet No. 14 of 16

[illegible]

# FINAL STRESS REPORT TABULATED

Unit 1

Sheet No. 15 of 16

Job Order Number

PROB NO.	SYSTEM	ISO/SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRIORITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>h</sub> )	SEISMIC/ALLOWABLE		
109	Core Spray Sys.	524	In	10	13226	6232	26000	23	1	
			In	10	13770	7362	27000	27		
			In	10	15824	8830	26000	33		
			In	10	12642	6234	27000	23		
110	RHR	545	Out	4	23765	18128	27000	65	6	
			Out	4	17334	15000	27000	54		
111	RCIC	535	Out	4	13755	7606	27000	28	3	
		549	Out	4	5160	794	27000	2		
			Out	2	8896	5100	27000	18		
			Out	2	6603	1290	27000	4		
			Out	1	4591	1324	27000	4		
			Out	1	6080	2662	27000	9		
112	RHR Pumps - 1A & 1C	546		4	20416	17624	27000	65	6	
	RHR Pumps - 2A, B, C & D	565		4	13146	11930	27000	44		
113	RHR 1A, B, C & D	547		4	12445	9072	27000	34	6	
	RHR Drain to RW	548		4	23806	21690	27000	80		
114	RCIC Conn. to FW	563		4	19799	14074	27000	52	3	
		568		4	17477	8958	28901	31		
				4	17137	10310	28901	36		
115	Condensate Drain N.S.S.	571		3	11374	9420	27000	35	6	
116	RHR Pumps 1A & 1B Suction Lines	605		2	24752	20878	27000	77	1	
117	Service Water Sys.	606		6	26321	23760	27000	88	1	
				4	7198	4738	27000	18		
118	Service Water Sys.	607		6	15932	13604	27000	50	5	
				4	19645	17400	27000	64		
				2	2611	116	31500	4		
		608		6	12432	10180	27000	38		
				4	6125	3872	27000	14		
				2	9931	8314	31500	31		
119	Service Water Sys.	609		6	18163	15348	27000	57	2	
				4	14771	12518	27000	46		
120	High Pressure Coolant Injection System	657		1	12239	10234	27000	38	5	
				3/4	5496	4619	27000	17		
				1/2	261	168	27000	1		
121	Service Water	662		6	1486	32	27000	1	4	
				6	2469	78	33120	3		
				3	880	116	33120	4		
				2 1/2	1457	310	33120	1		

PROB NO.	SYSTEM	ISO/ SHEET NO.	LOCATION Ins. or Outside Cont.	LINE SIZE	EMERGENCY CONDITION STRESS (PSI)				PRIOR- ITY CATEG	REMARKS
					TOTAL STRESS	SEISMIC (DBE)	ALLOWABLE (1.8 S <sub>H</sub> )	SEISMIC/ ALLOWABLE		
122	Instrument Air	674		2 1½ 3/4	6405 2053 2773	5065 756 1560	27000 27000 27000	19 3 6	6	
123	Instrument Air	675		2 3/4	10007 11262	8730 10112	27000 27000	32 37	6	
124	Instrument Air	677		2	21584	20244	27000	75	6	
125	Instrument Air	690		2 1½ 3/4	13912 3675 26655	12662 2468 25529	27000 27000 27000	47 9 95	6	
126	Instrument Air	679		2 3/4	11940 8660	3676 506	27000 27000	14 2	6	
127	Instrument Air	680		2	8153	4817	27000	18	6	
128	Instrument Air	682		2 1½	12316 1299	10975 2	27000 27000	41 0	6	
129	Cont. Atmos. Control Sys. Sup. Lines	709 710		24 20 18 8	2962 12551 2522 15579	464 9450 298 11038	27000 27000 27000 27000	2 35 1 41	6	
130	Instrument Air	681 691		2	11483	10142	27000	38	6	
131	Cont. Atmos. Control Sys. Relief Valve Piping	713		1 2 3/4 ½	20574 5408 5180 5168	14480 4000 4000 4000	27000 27000 27000 27000	54 15 15 15	6	
132	Service Water	716		1½ 1½	23143 1457	18610 276	27000 27000	69 1	6	
	Note 1: Total stress below allowable using "greater of 1.8 S <sub>H</sub> or .9 S <sub>y</sub> "									

May 21, 1979

ATTACHMENT 4

REANALYSIS PRIORITY CRITERIA

The following is the criteria that was used to establish the priority for the reanalysis for seismic stress of the safety related piping:

<u>CATEGORY</u>	<u>DESCRIPTION</u>
Complete 4/24/79	Initial reanalysis of 10 lines
Complete 5/21/79	Large pressure boundary lines in specified systems and lines with high stress from original analysis.
1.	Other pressure boundary lines 2½" and larger
2.	Other core standby cooling system lines 2½" and larger
3.	Other RCIC lines 2½" and larger
4.	Service Water lines 2½" and larger
5.	Pressure boundary, CSCS, RCIC lines 2" and smaller
6.	Other support systems (standby gas treatment, diesel generation fuel oil and cooling water, fuel pool cooling line to RHR, containment atmospheric control system, noninterruptible air).



## REANALYSIS SCHEDULE

UNIT No 142

5/12 5/19 5/26 6/2 6/9 6/16 6/23 6/30 7/7 7/14 7/21

## CATEGORY

STRESS  
ANALYSISSUPPORT  
ANALYSISCOMPLETED  
(39 LINES)

" 1 "

" 2 "

" 3 "

" 4 "

" 5 "

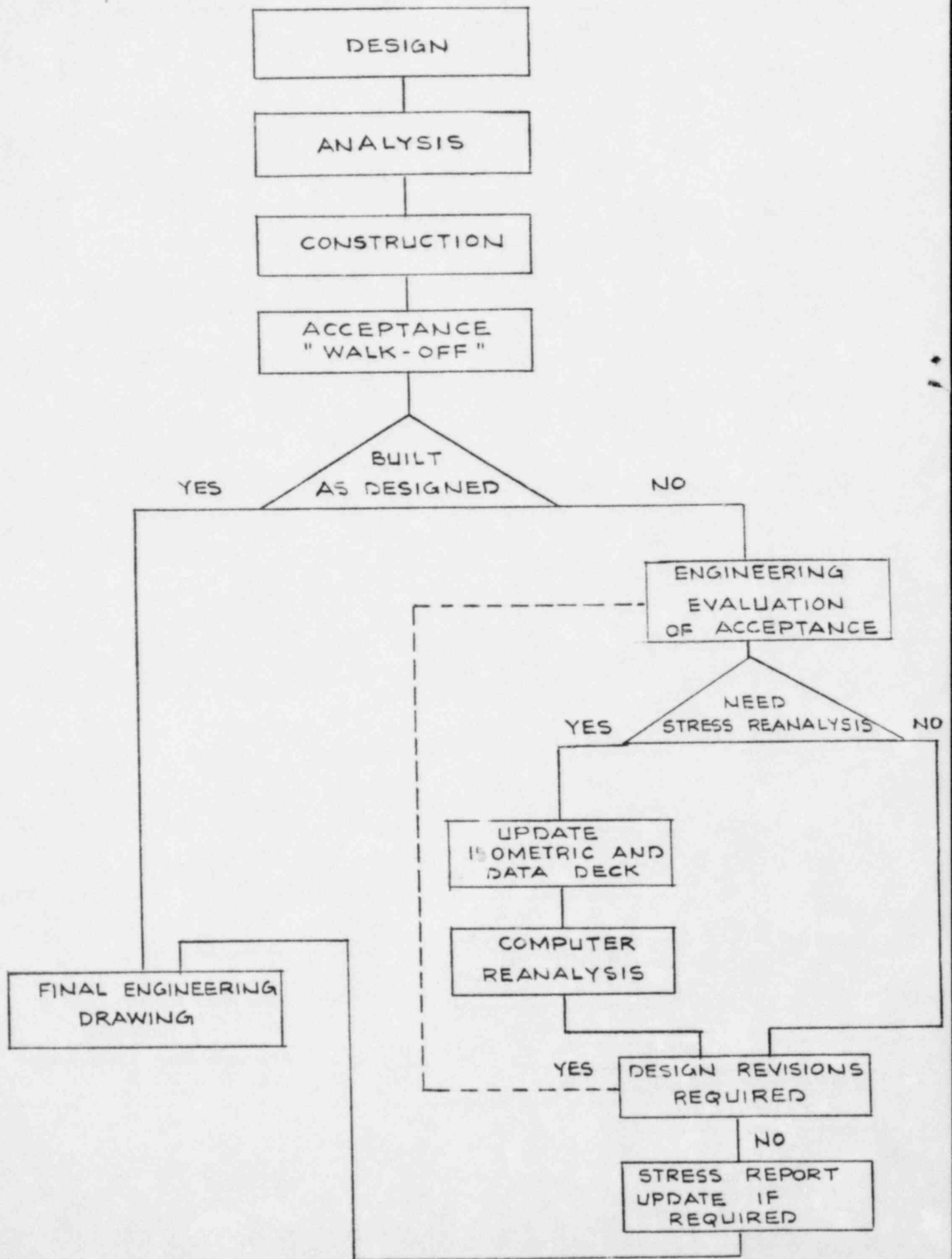
" 6 "

FINAL REPORT

PIPE STRESS ANALYSIS

SUPPORT ANALYSIS

ATTACHMENT 6





## ATTACHMENT 7

### REANALYSIS EVALUATION SUMMARY

Reanalysis of pipe stress has been completed for 39 lines. All the results are within applicable allowable limits. In Attachment 2 we have tabulated the results associated with the emergency condition to address in detail the ability of these lines to safely withstand the design basis seismic (DBE) event. A tabulation of original pipe stresses for the combination of all the applicable loading conditions (i.e. pressure, dead weight, seismic, transient, etc.) and the individual seismic loading conditions is provided to determine the variance between the original analysis and the reanalysis. Specifically the ratio between the piping seismic stress obtained from the reanalysis and the original analysis have been evaluated and the summary is presented in Attachments 8 and 9. In this evaluation, the results obtained from a reanalysis of ten lines in the Indian Point Nuclear Unit #3 has been considered. Inclusions of these lines in the evaluation is legitimate since the Brunswick Units and the Indian Point 3 Unit used a) identical computer programs in both the original analysis and the reanalysis, b) identical approach to combine seismic excitation components (i.e. one horizontal and one vertical component simultaneously) and c) comparable piping configuration complexities.

From this evaluation we believe that a value of 1.5 for the ratio  $R_{MC}$  represents a realistic upper bound.

To properly evaluate the ability to safely withstand the design basis seismic event for those lines which have not yet been reanalyzed, the above consideration on the ratio  $R_{MC}$  must be reviewed together with the conservatism used to evaluate the pipe seismic stresses for the DBE conditions. As shown in the Attachment 2 foot note, the DBE seismic stresses were obtained by multiplying the values obtained for the OBE condition by 2 (two). This approach was used to minimize the number of computer runs required to a) generate Amplified Response Spectra (ARS) and b) to evaluate pipe stresses. This approach was modified only in those rare cases where the pipe stresses exceeded the applicable limits.

The factor of two is very conservative and therefore the stresses shown in Attachment 3 for the lines yet to be reanalyzed for DBE stress are very conservative. Since pipe stresses evaluated with a computer code using a modal approach and ARS input (such as the code(s) used) are directly dependent on the values of the ARS, a more realistic approach must consider the actual difference between the ARS for the OBE and DBE. Figure 7-1, 7-2, 7-3 provide plots of portions of ARS for the frequency range generally affecting piping systems. These ARS are representative for those elevations of the reactor building which contain most of the piping systems under investigation.

From these figures it can be seen that the two curves do not significantly differ. This is due primarily to the difference in structural and equipment damping values allowed in the computation of the OBE and DBE ARS.

ATTACHMENT 7 Cont'd

A review of these figures indicate that piping with different fundamental frequencies may be affected differently and, in certain cases, DBE stress results may be even less than the OBE stress results. Since, in general, pipe stresses are affected mostly by the first fundamental frequency mode and a representative range for the first fundamental frequency is 4-10 cps, it was concluded that a conservative value of 1.2 for the ratio RARS between the DBE and OBE curves was more representative.

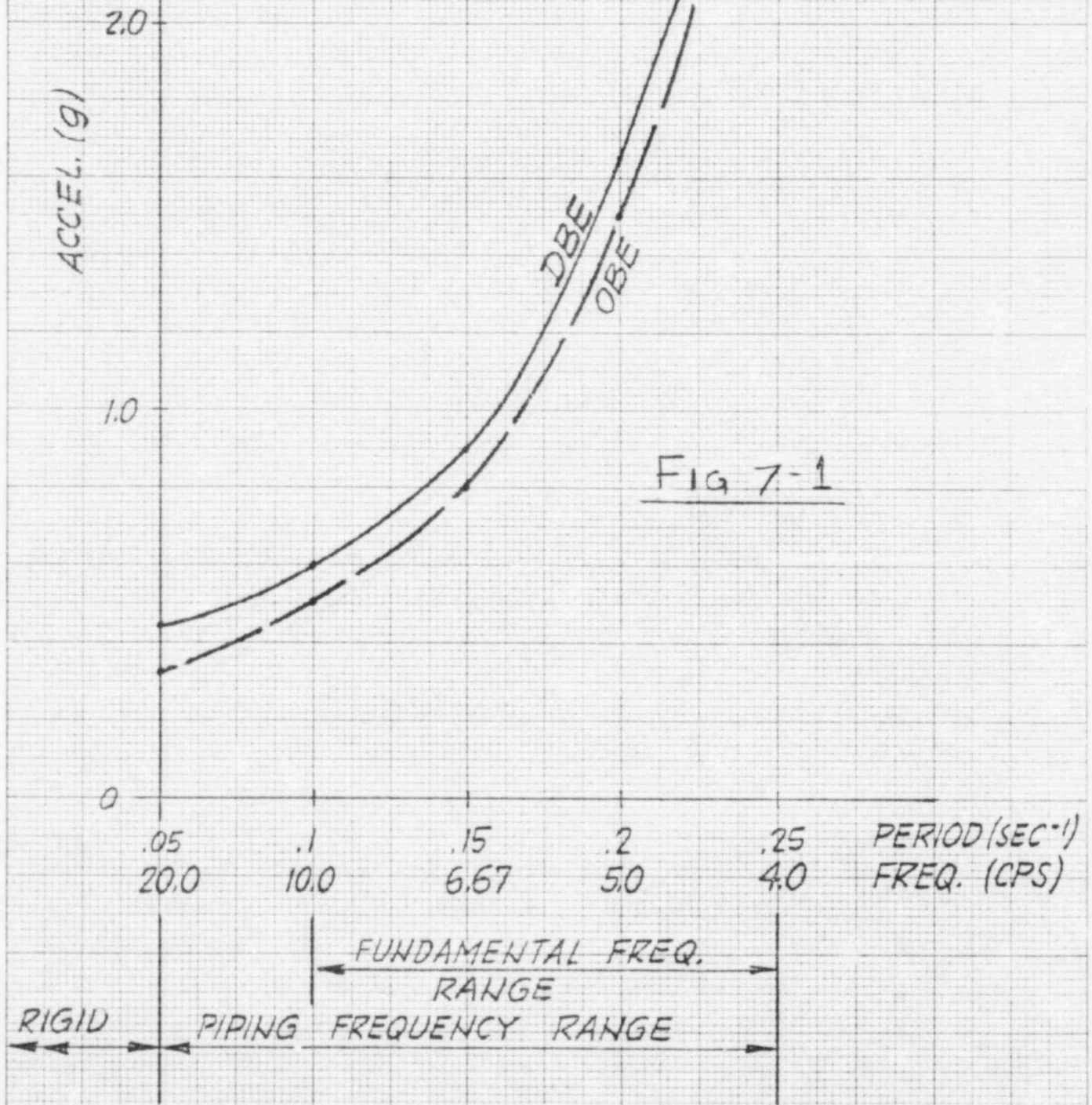
With the above considerations in mind, realistic (but still conservative) values for the DBE pipe stresses can be calculated by multiplying the values shown in Attachment 4 by the factor:

$$\alpha = \frac{R_{MC} \times R_{ARS}}{2}$$

The factor 2 above brings the DBE stress values shown in Attachment 3 back to the computer evaluated OBE results. The RARS factor accounts for the realistic difference between the DBE and OBE ARS and the RMC factor gives consideration to the effects introduced by using a square root of the sums of the squares approach rather than an algebraic summation for the earthquake component response combinations within each mode.

Since  $\alpha$  is approximately 1 and the values shown in Attachment 3 in effect will not change, we conclude that the original values for the total stresses are still representative and therefore the capability of the plant to safely withstand a seismic event in the interim is not impacted.

CAROLINA POWER & LIGHT CO.  
BRUNSWICK STEAM ELECTRIC PLANT  
UNIT 1 & 2  
AMPLIFIED RESPONSE SPECTRA  
OBE & DBE  
EL. 50'-0" (NODE 40)  
REF.: ADDENDUM 'B' TO DESIGN  
REPORT NO 4  
FIG. 12 & 26



ACCEL. (9)

1.0

DBE

DBF

FIG 7.2

0

.05

7

15

.2

.25

PERIOD (SEC<sup>-1</sup>)

20.0

10.0

6.67

50

4.0

FREQ. (CPS)

FUNDAMENTAL FREQ.  
RANGE

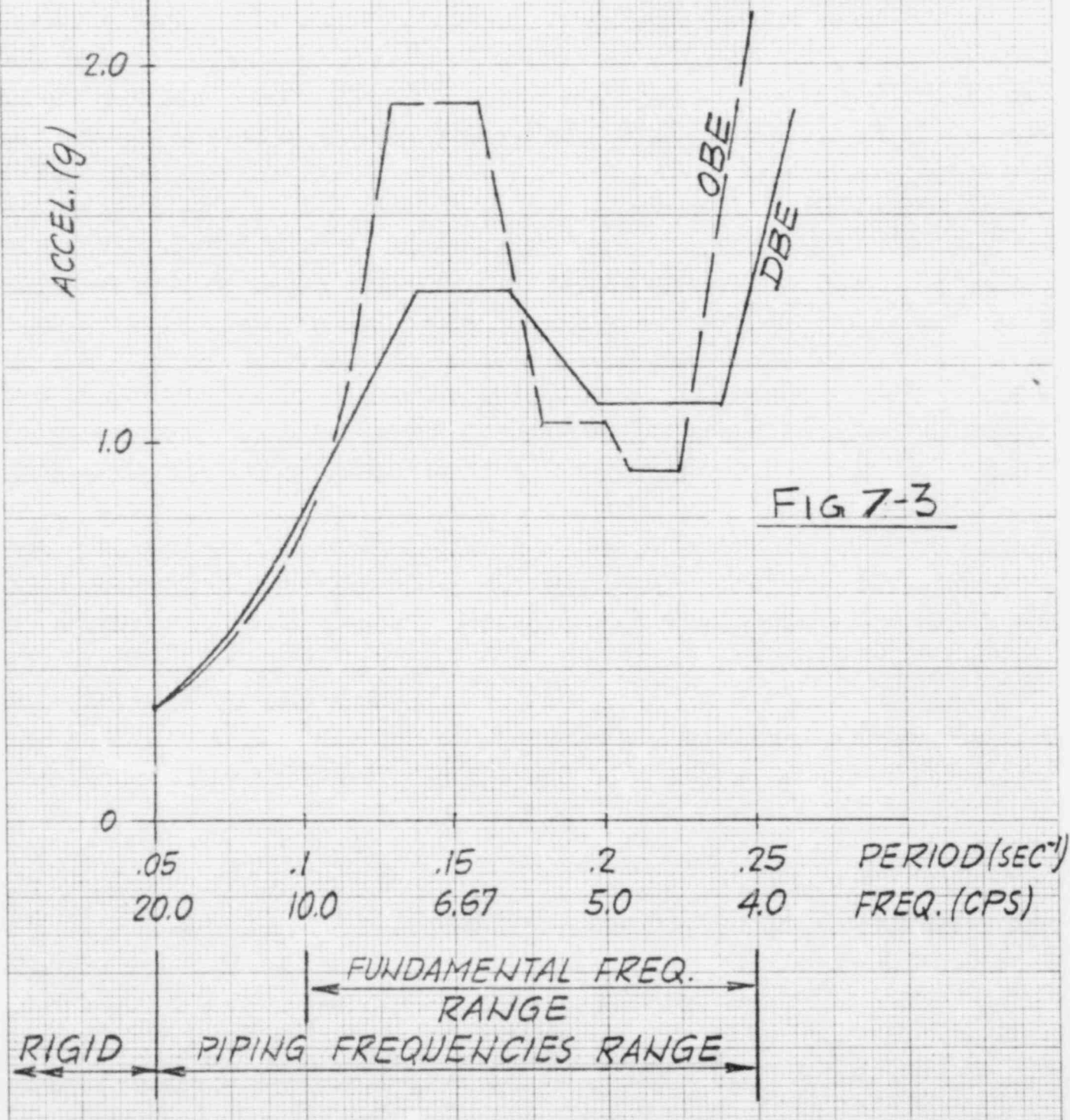
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## PIPING FREQUENCIES RANGE



ATT, 7

CAROLINA POWER & LIGHT CO.  
BRUNSWICK STEAM ELECTRIC PLANT  
UNIT 1 & 2  
AMPLIFIED RESPONSE SPECTRA  
OBE & DBE  
EL. (-) 17'-0" (NODE 42)  
REF: ADDENDUM 'B' TO DESIGN  
REPORT NO 4; FIG. 14 & 23



46 1242

14-2 20 X 20 TO THE 1/2 X 10 PIPES  
HORIZONTAL & VERTICAL

# ATTACHMENT 8

## RATIO $R_{MC}$ EVALUATION SUMMARY (BRUNSWICK 1 & 2 AND INDIAN PT. 3) COMBINED

BRUNSWICK 1 & 2 AND INDIAN PT. 3					
$R_{MC}$	0.50(MIN)	0.75	1.00	1.25	1.50 1.52(MAX)
NUMBER OF LINES ANALYZED (TOTAL 40)	4	19	12	4	1
PERCENT OF LINES			97.5%		2.5%
PERCENT OF LINES			87.5%	12.5%	
PERCENT OF LINES		58%		42%	

$$R_{MC} = \frac{\text{NEW SEISMIC STRESS}}{\text{ORIGINAL SEISMIC STRESS}}$$

# ATTACHMENT 9

## RATIO $R_{MC}$ EVALUATION SUMMARY (BRUNSWICK 1 & 2 AND INDIAN PT 3) INDIVIDUAL

### BRUNSWICK 1 & 2

$R_{MC}$	0.50(MIN)	0.75	1.00	1.25	1.50	1.52(MAX)
NUMBER OF LINES ANALYZED (TOTAL 30)	2	14	9	4	1	
PERCENT OF LINES	7%	47%	30%	13%	3%	
PERCENT OF LINES	54%		46%			

### INDIAN PT.3

$R_{MC}$	(MIN) 0.72	0.75	1.00	1.10 (MAX)
NUMBER OF LINES ANALYZED (TOTAL 10)	2	5	3	
PERCENT OF LINES	20%	50%	30%	
PERCENT OF LINES	70%		30%	

$$R_{MC} = \frac{\text{NEW SEISMIC STRESS}}{\text{ORIGINAL SEISMIC STRESS}}$$



## BSEP SEISMIC MONITORING PROGRAM

### ATTACHMENT 10

An intensive seismic monitoring program was conducted at the Brunswick site under NRC direction and review for approximately 2 years, ending in December, 1977. The BSEP seismic monitoring program was directed at predicting an earthquake by identifying precursory phenomena. After nearly two years of seismic analysis, applying state-of-the-art techniques, the NRC granted CP&L's request to terminate the seismic program, stating that "The data reported does not indicate dilatency or other earthquake precursory phenomena is occurring in the Wilmington-Southport area." (Letter: A. Schwenscer to J. A. Jones, December 28, 1977). The staff's safety evaluation further stated, "The lack of detection of such local earthquake activity by the network combined with the low level of recorded historical earthquake activity in this region suggest a low earthquake potential for the region."

The lack of precursory anomalies is significant, in that the duration of a precursor anomaly is related to earthquake magnitude. For example, an event with Richter magnitude 5 is preceded by an anomaly spanning about 4 months, whereas a major earthquake at about magnitude 7 would be preceded by an anomaly lasting about 14 years. Thus, even if a precursor were detected this week, the maximum earthquake magnitude that would be expected by August 21 would be less than magnitude 5. This is equivalent to a ground acceleration of about 0.04g's, The OBE is 0.08g, and the DBE for BSEP is 0.16g. The ongoing seismic monitoring program at our Shearon Harris site, (which commenced in September 1977) has a detection threshold such that it can detect seismic precursors in the Southport area.

No seismic events have been recorded by the BSEP or SHNPP Network, or the USGS network in South Carolina; and the SHNPP monitoring program continues to support the conclusion that the Southport area is unusually aseismic.