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IPPOLITO, T. A. Operating Reactors Branch 3

SUBJECT: Submits status of seismic analysis of safety related piping.

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July 3, 1979

FILE: NG-3514(B) SERIAL: GD-79-1696

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:

Since our meeting and letter of June 4, 1979, our Architect-Engineer has continued the seismic reanalysis of safety-related piping. We have completed the pipe stress reanalysis for 157 of 199 isometrics, and have now completed those lines identified as Categories 3, 4 and 5 in our letter of May 21, 1979. The results for each of these categories are shown in the attached Tables 1, 2 and 3. As shown on Table 3, recalculated pipe stresses from two isometrics would exceed allowable stresses (1.8 S_H) stated in the FSAR, but both lines maintain structural integrity (stress is less than 2.4 S_H). In accordance with the commitment in our April 24, 1979 letter, we notified the NRC on June 22 that the current computer model for seismic analysis showed these lines exceeded allowable stress values. This condition is a result of changing the approach to analyze lines using valve operator eccentricity, as discussed at the June 4, 1979 meeting with the NRC staff. The previous approach used a lumped mass (valve and operator) located at the centroid of the valve. The revised approach separates the valve mass and the operator mass and locates them appropriately. The isometrics involved are numbers 8 and 657, as shown in Table 3. Support modifications for these lines are being designed, and will be installed no later than the next refueling outage.

In our previous correspondence, we also committed to verify the as-built piping support locations of safety-related systems compared to the as-designed (as-analyzed) configuration. This verification program has now been completed. Differences were documented and sent to United Engineers for their determination if a reanalysis would be required. U E & C is performing an engineering evaluation of these differences as they are received. Using the guidelines outlined in Attachment 2 of our June 4, 1979 letter, most lines do not require reanalysis. However, six lines have been reanalyzed to date. As shown in Table 4, all stresses are well within allowable limits.

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Pipe support analyses are also continuing. Pipe supports for the first 119 isometrics have been evaluated. Approximately 10% of the support loads have exceeded allowable values using the current computer model, but have not exceeded structural integrity limits. These supports have been identified as requiring a long term (next refueling outage) fix. On a subsequent isometric, two supports on BSEP-2 were identified on July 2 as requiring an immediate fix. Your Staff and NRC Region II were notified of these supports on July 2 by telephone. Modifications of these supports is in progress, and we will complete modifications prior to returning to operation from the current maintenance outage. During this outage, we installed a guide support on a 3/4" instrument line. It was determined during the as-built verification that this support was not in place. This was also reported to your Staff and Region II on July 2.

Our final report covering the remaining lines and pipe supports will identify supports and lines which will require long-term fixes, as well as summarizing all reanalyses. If you have any questions concerning this matter, please contact my staff.

Yours very truly,

E. E. Utley
Executive Vice President
Power Supply & Customer Services

MAC/DLB/jcb
Attachments

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

TABLE 1

CATEGORY 3

STRESS SUMMARY

EMERGENCY CONDITION (PSI)

SYSTEM	ISO. NO.	LINE SIZE	ORIGINAL TOTAL	ORIGINAL SEISMIC (1)	NEW TOTAL	NEW SEISMIC (2)	ALLOWABLE	RATIO NT/A
Reactor Core Isolation Cooling & Residual Heat Removal	34	8"	21108	16218	11823	6933	27000	0.44
Reactor Core Isolation Cooling & Residual Heat Removal	66, 67	4"	12805	6932	10403	4512	27000	0.38
Reactor Core Isolation Cooling & Residual Heat Removal	63	4"	15098	9488	12075	6465	27000	0.45
Reactor Core Isolation Cooling & Residual Heat Removal	50	4"	7221	2900	5425	1104	27000	0.20
Reactor Core Isolation Cooling & Residual Heat Removal	12	6"	19870	16526	11781	8064	27000	0.44
Reactor Core Isolation Cooling & Reactor Water Cleanup	563 & 568	4"	19799	14074	13944	5425	27000	0.51
Reactor Core Isolation Cooling	33	3"	15615	12356	9165	5902	27000	0.33
Reactor Core Isolation Cooling	535 & 549	4," 2," 1"	13755	7606	15035	11667	27000	0.55
Reactor Core Isolation Cooling	35 & 49	4," 2," 1"	18881	13096	12024	9087	27000	0.44

(1) DBE = OBE X 2

(2) DBE = OBE X 1.2

TABLE 2

CATEGORY 4

STRESS SUMMARY

EMERGENCY CONDITION (PSI)

SYSTEM	ISO. NO.	LINE SIZE	ORIGINAL TOTAL	ORIGINAL SEISMIC (1)	NEW TOTAL	NEW SEISMIC (2)	ALLOWABLE	RATIO NT/A
Service Water	86	8," 12," 16"	7875	5098	7747	4970	27000	0.28
Service Water	662	2-1/2," 3," 6"	2469	310	2353	146	27000	0.09
Service Water	85	16," 12," 8"	11781	7292	8287	3793	27000	0.30
Residual Heat Removal Reactor Building Service Water	30	20," 16"	8026	3650	6821	4076	27000	0.25
Service Water	109	4," 6"	18921	15280	15515	11874	27000	0.57
Service Water & Residual Heat Removal	29	20," 15"	10922	3214	13673	6065	27000	0.50
Service Water	110	3," 4," 10," 20"	19323	18532	7225	6434	27000	0.26
Service Water	162	2.5," 3," 6"	2853	648	2693	249	27000	0.09
Service Water	62	10," 16," 20"	14230	7256	13030	6038	27000	0.48
Service Water	163	2-1/2," 3," 6"	8209	7730	2957	2078	27000	0.10
Service Water	217	4," 2-1/2," 2," 1"	7731	3730	6990	1837	27000	0.26
Service Water	147	1 1/2," 3"	22372	20844	4152	2624	27000	0.15
Service Water	300 301 302	14," 10-1/4," 3" 2-1/2," 2," 1-1/2" 1-1/4"	16182	13020	18036	14874	33228*	0.54

(1) DBE = OBE X 2

(2) DBE = OBE X 1.2

* 1.8 S_h for Stainless Steel

TABLE 3

CATEGORY 5

STRESS SUMMARY

EMERGENCY CONDITION (PSI)

SYSTEM	ISO. NO.	LINE SIZE	ORIGINAL TOTAL	ORIGINAL SEISMIC (1)	NEW TOTAL	NEW SEISMIC (2)	ALLOWABLE	RATIO NT/A
Reactor Core Isolation Cooling	92	1/2", 3/4, 1, 2, 3"	21144	18497	5648	3002	27000	0.20
Reactor Core Isolation Cooling	161	1/2", 3/4", 1"	11796	8352	8908	5464	27000	0.32
Reactor Core Isolation Cooling	164	3/4", 1-1/4", 2"	5559	4638	6119	5200	27000	0.22
Reactor Core Isolation Cooling	195	1"	16705	15826	8337	7458	27000	0.31
Residual Heat Removal	170	1/2", 1"	19175	18077	11063	10855	27000	0.40
Residual Heat Removal	171	1/2", 3/4", 1"	5232	3335	3495	1194	27000	0.13
Residual Heat Removal	172	3/4", 1"	10005	7116	11772	8883.6	27000	0.43
Reactor Water Cleanup & RHR	132	2"	17800	13618	13398	9216	27000	0.49
High Pressure Coolant Inj.	151	3/4"	5920	3706	4490	2022	27000	0.16
High Pressure Coolant Inj.	156	3/4", 1"	11067	5900	8466	3528	27000	0.31
High Pressure Coolant Inj.	153	3/4", 1"	5103	580	4622	99	27000	0.17
High Pressure Coolant Inj.	159	1"	16518	13414	3729	625	27000	0.13

TABLE 3

CATEGORY 5 (continued)

STRESS SUMMARY

EMERGENCY CONDITION (PSI)

SYSTEM	ISO. NO.	LINE SIZE	ORIGINAL TOTAL	ORIGINAL SEISMIC (1)	NEW TOTAL	NEW SEISMIC (2)	ALLOWABLE	RATIO NT/A
High Pressure Coolant Inj.	657	1", 3/4", 1/2"	12239	10234	27815	25810	36000	0.77 (3)
Service Water	607	6", 4," 2"	19645	17400	9571	7326	27000	0.35
Service Water	608	6", 4," 2"	12432	10180	7699	5467	27000	0.28
Nuclear Steam	8	2," 1," 1/2"	18979	16874	35181	34010	36000	0.97 (3)
High Pressure Coolant Inj.	157	1," 3/4"	20511	16780	19347	15616	27000	0.71 (4)

(1) OBE X 2.0 = DBE

(2) OBE X 1.2 = DBE

(3) Overstressed due to motor operators. 36000 is 2.4 Sh allowable for structural integrity.

(4) New seismic analysis was performed based on actual design basis earthquake response spectrum.

[illegible]

TABLE 4

LINES REANALYZED AS OF JUNE 28, 1979 FOLLOWING AS-BUILT WALK THROUGH

STRESS SUMMARY

EMERGENCY CONDITION (PSI)

SYSTEM	ISO. NO.	LINE SIZE	ORIGINAL TOTAL	ORIGINAL SEISMIC (1)	NEW TOTAL	NEW SEISMIC (2)	ALLOWABLE	RATIO NT/A
Residual Heat Removal	25	4", 6"	18558	12904	15011	9357	27000	0.55
Feedwater	160	12", 18"	18007	12420	8825	3238	27000	0.32
Instrument Air	193	3/4", 2"	18022	16773	10559	9309	27000	0.39
Instrument Air	184	3/4", 2"	30632	29382	13629	12379	27000	0.50
Instrument Air	185	3/4", 2"	15820	14750	10899	9649	27000	0.40
Nuclear Steam Supply	87, 128	3", 2", 1-1/2", 1"	24225	20022	25997	20565	27000	0.96

(1) OBE X 2.0 = DBE

(2) OBE X 1.2 = DBE