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April 1, 1980
IPN-80-35

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Asymmetric Pressure Vessel Loading

Dear Sir:

In response to Mr. L. Olshan's February 19, 1980 verbal request, this letter transmits a summary of "the original evaluations of the reactor coolant system (RCS) of Indian Point 3," referred to in WCAP-9117, at page 1-1.

WCAP-9117 entitled "Analysis of Reactor Coolant System for Postulated Loss of Coolant Accident: Indian Point 3 Nuclear Power Plant," was submitted to the Commission on June 15, 1977 by Consolidated Edison.

In addition, as requested, the attached summary includes tables which provide the seismic loads for RCS piping and supports. Seismic loads for the Reactor Vessel and internals are also attached and/or referred to by citing appropriate FSAR sections.

Very truly yours,

Paul J. Early
Assistant Chief Engineer-Projects

Att.
cc: Mr. T. Rebelowski, Resident Inspector
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SUMMARY OF STRUCTURAL ANALYSIS FOR INDIAN POINT UNIT NO. 3

Prior to commercial operation of Indian Point Unit No. 3, Westinghouse performed a structural analysis of the reactor coolant loop and primary equipment support system. This analysis was performed to determine the design adequacy and structural integrity of the loop and support system for the conditions specified as the design basis for the plant. Allowable loads were determined from USAS B31.1 for piping and AISC for supports. The analyses performed included consideration of thermal, deadweight, pressure, OBE, DBE, and LOCA. Details on load combinations and stress limits are provided in Appendix A of the Indian Point Unit No. 3 FSAR, Table A.1-2 and accompanying curves.

The seismic analysis was performed using the response spectrum technique. Two analyses were performed with differing (x or z) horizontal input combined with the vertical (y) input. For this plant, two distinct loops (Loops 31 and 32) were modeled because the tangential discharge pumps make the cold legs geometrically dissimilar.

Twenty-one reactor coolant system pipe ruptures (LOCA's) were analyzed on a time history basis, including breaks at the steam generator inlet and outlet, reactor coolant pump inlet and outlet, loop closure weld, and intermediate and branch locations for both loop geometrics. Reactor pressure vessel nozzle breaks were not evaluated.

The Westinghouse computer codes SATAN and STHRUST were used to develop hydraulic forcing functions. The code WESTDYN was used in the piping evaluation and STRUDL and THESSE were used for component supports.

Reactor coolant loop piping and support stresses from these analyses are shown in Tables 1-3 for faulted conditions. Seismic loads are also included in these tables in response to a separate NRC request for this information.

The results of these analyses established the ability of the reactor coolant loop and support systems to withstand the specified design basis events and to meet the requirements specified in the FSAR with margin remaining.

Seismic loads for the internals were provided to the commission in WCAP 7332-L-AR and WCAP 7822 and in Section 14.3.3 of the Indian Point Unit No. 3 FSAR. Tables 14.3.3-1 through 14.3.3-3 contain the results of these internals analysis including the seismic loads. Seismic loads for the reactor vessel are given in Table 4.

TABLE 1
FAULTED CONDITION PIPING STRESSES
(DBE + NORMAL)

Maximum Combined Stresses Due to DBE, Deadweight, and Membrane Pressure in the Reactor Coolant Loop Piping:

Piping	DBE Stress (psi)[a] S_{DS}	Deadweight Stress (psi) S_D	Pressure Stress (psi) S_P	Combined Stress (psi) $S_{DS} + S_D + S_P$
	Loop 31	Loop 31	Loop 31	Loop 31
Hot Leg	4,350	350	6,635	11,335
Crossover Leg	5,450	150	6,671	12,271
Cold Leg	3,050	250	6,600	9,900
	Loop 32	Loop 32	Loop 32	Loop 32
	4,450	350	6,635	11,435
	5,550	150	6,671	12,371
	3,750	250	6,600	10,500

Conclusion: $\sigma_a = S_{DS} + S_D + S_P \leq 1.2S$

where: $1.2S = 1.2 \times 17{,}050 = 20{,}460$ psi for piping

$1.2S = 1.2 \times 14{,}950 = 17{,}940$ psi for elbows

(a) S_{DS} = Maximum DBE stress due to shock in directions X + Y or Y + Z.

TABLE 2
HIGH-STRESSED POINTS IN EACH LEG WITH BREAK
IN OTHER LEG OR LOOP

Loop 31	Piping Stresses (ksi)						Stress Ratios				
	SI	S_Y	S_AMax.	S_AMin.	S_S	S_H	$\frac{SI}{S_Y}$	$\frac{S_A Max.}{S_Y}$	$\frac{S_A Min.}{S_Y}$	$\frac{S_S}{S_Y}$	$\frac{S_A}{S_Y}$
Hot Leg	35.51	18.0[a]	6.13	2.77	0.645	13.05	1.97	0.34	0.15	0.04	0.73
Crossover Leg	25.86	18.0[a]	3.55	1.46	2.70	10.53	1.44	0.20	0.08	0.15	0.59
Cold Leg	30.45	18.0[a]	7.78	1.94	1.07	13.0	1.69	0.43	0.11	0.06	0.72
Loop 32											
Hot Leg	29.32	18.0[a]	6.11	3.20	0.46	13.05	1.63	0.34	0.18	0.03	0.73
Crossover Leg	29.10	18.0[a]	3.57	1.53	1.01	10.53	1.62	0.20	0.09	0.06	0.59
Cold Leg	38.55	18.0[a]	7.08	-0.38	1.00	13.0	2.14	0.39	-0.02	0.06	0.72

(a) Conservatively used for all piping and elbows.
Refer to FSAR, Appendix A for allowables.

TABLE 3

STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORT STRESSES FOR MOST HIGHLY LOADED MEMBERS

(Stresses are expressed as percent of allowable)

SUPPORT MEMBER	LOADING CONDITION	
	DBE	DWT + PR + LOCA
1. SG Support		
a) Pipe stub columns	13.2	(1)
b) Vertical frame columns	4.8	70.1
c) Stub columns support beam	15.3	99.8
2. RCP Support		
a) Pipe column	17.	96.5
b) Bracing members	5.	(2)
c) Tie Rod	6.9	(3)

NOTES:

General: Some yielding of supports is allowed provided overall system integrity is maintained. See Appendix A of FSAR.

1) One of the four stub columns yields slightly.

2) One of the diagonal WF bracing members exceeds allowable stress by 8% based on Fy = 36 ksi. Actual properties for A36 rolled shapes generally exceed tabulated minimum valves by over 10%.

3) A tie rod yields based on material minimum yield strength of 36 ksi. However, available tie rod material certifications indicate that the actual material yield strength is 55 ksi and greater which is above the calculated stress in this member.

TABLE 4
DBE LOADS FOR REACTOR PRESSURE VESSEL NOZZLES

	INLET NOZZLE	OUTLET NOZZLE
FORCES (kips)		
X	175	232
Y	23	53
Z	175	70
MOMENTS (in-kips)		
X	1621	569
Y	5262	6031
Z	1621	5352