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MURRAY R. EDELMAN
VICE PRESIDENT
NUCLEAR

October 2, 1984
PY-CEI/NRR-0139 L

Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to NRC Question
No. 420.03 through 420.07-
Effects of High Energy Line Breaks
on Control Systems (Open Item No.14)

Dear Mr. Youngblood:

This letter and its attachments are provided in response to your request for additional information (dated January 10, 1984), regarding effects of high energy line breaks on control systems (Open Item No. 14) for the Perry Nuclear Power Plant (Units 1 and 2).

The information provided in the attachments will be incorporated into a future FSAR Amendment. If you have any questions, please feel free to call me.

Very truly yours,

A. Kaplan for M. Edelman

Murray R. Edelman
Vice President
Nuclear Group

MRE:njc

Attachments

cc: Jay Silberg, Esq.
John Stefano
Jack Grobe

B001
1/1

Q420.03 Provide an identification of the locations (elevations/areas) which contain high energy piping systems and in which components for the nonsafety related control systems are located. Relate these to the adverse conditions discussed in your letter dated March 14, 1983.

Response The following information identifies the locations (panel, building, elevation and coordinates) of non-safety control systems and high energy lines for adverse conditions #1-#5 discussed in our March 14, 1983 letter. We have referenced layout drawings contained in the FSAR for your convenience in identifying locations.

ADVERSE CONDITION #1

INSTRUMENT	PANEL	BUILDING	ELEVATION	COORDINATES
1B33N001A	1H22-P025	Containment	620	C1/12
1B33N001B	1H22-P041	Containment	620	C1/17
1C34N003B	1H22-P025	Containment	620	C1/12
1C34N003D	1H22-P041	Containment	620	C1/17

HIGH ENERGY LINE	BUILDING	ELEVATION	COORDINATES
C11-Control Rod Hydraulic System	Containment	630	C1/11
C41-Standby Liquid Control	Containment	602	C1/17

All of the above instruments and high energy lines are located between the drywell and the reactor building wall. Exact locations may be identified by referencing figures 3.6-4, 3.6-6, 3.6-12 and 3.6-14 of the FSAR.

ADVERSE CONDITION #2

INSTRUMENT	PANEL	BUILDING	ELEVATION	COORDINATES
1N27N156A	1H51P1147	Heater Bay	600	B/01
1N27N156B	1H51P098	Heater Bay	620	C/02
1N27N087A	1H51P098	Heater Bay	620	C/02
1N27N087B	1H51P098	Heater Bay	620	C/02
1N25N263A	Heater 6A	Heater Bay	600	D/03
1N25N263B	Heater 6B	Heater Bay	600	D/03
1N25N303A	Heater 5A	Heater Bay	600	D/02
1N25N303B	Heater 5B	Heater Bay	600	D/02
1N36N030A	1H51P1305	Heater Bay	620	B/01
1N36N030B	1H51P1330	Heater Bay	620	B/01
1N36N030C	1H51P1330	Heater Bay	620	B/01
1N26N153A	Heater 3A	Heater Bay	647	D/03
1N26N153B	Heater 3B	Heater Bay	647	D/04
1N21N339	Heater 4	Heater Bay	580	C/02

Heater Bay High Energy Lines	Elevation
N22 Main,reheat extraction & misc. drains	580, 600, 620, 647
N25 H.P. Htr. drain & vent	580, 600, 620
N27 Feedwater system	580, 600, 620, 647
N33 Steam seal system	600, 620, 647
N36 Extraction steam system	600 620, 647
N11 Main steam system	620, 647
P61 Auxiliary steam system	580, 600, 620, 647
N21 Condensate system	600, 620, 647

Refer to plant layout drawings, figures 1.2-4 and 1.2-5 of the FSAR, for the Heater Bay layout.

Note: A break in any of the above lines would not change the conclusions described in our March 14, 1983 letter (Q420.05/#2), which only addressed a feedwater line break in the Heater Bay.

ADVERSE CONDITION #3

INSTRUMENT	PANEL	BUILDING*	ELEVATION	COORDINATES
	1R25S002	TP	620	A/04
	1R22S003	TP	620	A/04
	1R23S003	TP	647	B/05
	1R23S004	TP	647	B/06
	1R23S006	TP	647	B/06
	1R24S034	TP	620	B/06
	1R42S017	TP	620	C/06
	1R42S021	TP	620	C/06
	1R42S022	TP	620	C/06
	1R42S023	TP	620	C/07
	1R14S008	TP	620	B/06
1C85N001A		TB	593	E/16
1C85N001B		TB	593	E/16
1C85N002A	1H51P187	TB	647	G/08
1C85N002B	1H51P187	TB	647	G/08
1C85N002C	1H51P773	TB	647	G/08

*TP - Turbine Power Complex
TB - Turbine Building

High Energy Lines - No high energy lines are located within the 620 and 642 levels of the Turbine Power Complex. No further review was necessary to determine high energy lines in the vicinity of the C85 components. Refer to plant layout drawings, figures 1.2-5 and 1.2-6 of the FSAR.

ADVERSE CONDITION #4

Refer to Adverse Condition # 1.

ADVERSE CONDITION #5

INSTRUMENT	PANEL	BUILDING*	ELEVATION	COORIDNATES
	1R11S004	TP	620	B/04
	1R11S005	TP	620	B/02
	1R14S008	TP	620	B/06
	1R22S002	TP	620	A/04
	1R22S003	TP	620	A/04
	1R22S004	TP	647	A/05
	1R22S005	TP	647	A/04
	1R23S004	TP	647	B/06
	1R23S006	TP	647	B/06
	1R22S016	TP	620	A/03
	1R23S017	TP	647	A/05
	1R23S003	TP	647	B/05
	1R24S034	TP	620	B/06
	1R42S017	TP	620	C/06
	1R42S021	TP	620	C/06
	1R42S023	TP	620	C/07
1C85N001A		TB	593	E/16
1C85N001B		TB	593	E/16
1C85N002A	1H51P187	TB	647	G/08
1C85N002B	1H51P187	TB	647	G/08
1C85N002C	1H51P187	TB	647	G/08

* TP - Turbine Power Complex
TB - Turbine Building

High Energy Lines- No high energy lines are located within the 620 and 642 levels of the Turbine Power Complex. No further review was necessary to determine high energy lines in the vicinity of the C85 components. Refer to plant layout drawings, figures 1.2-5 and 1.2-6 of The FSAR.

NOTE: An error was made in the feedwater pump section of Adverse Condition #5, "1R42S002" should be "1R22S002."

Q420.04 Provide a detailed analysis for the turbine trip without bypass event (FSAR Section 15.2.3) in conjunction with a high energy line break that causes a loss of feedwater heating (and subsequent increase in reactor power level). Without operator action, the staff is concerned that this event could lead to a turbine trip without bypass event from a higher power level than previously analyzed.

Response

A detailed review and walkdown of steam bypass and feedwater heating has determined that there is no single high energy line break (HELB) which could fail both systems. A high energy line break that would result in a loss of feedwater heating is an accident condition and is evaluated under 10 CFR 100 guidelines. Any HELB that could affect feedwater heating would be bounded by the Chapter 15 analysis for "Steam System Piping Break - Outside Containment" (FSAR Section 15.6.4). The radiological dose calculation for the event shows that its consequences are bounded by 10 CFR 100 limits (FSAR table 15.6-8).

The turbine trip without bypass transient has been analyzed under worst expected transient conditions (FSAR 15.2.3). This analysis is consistent with the requirements of SRP 15.2.2, including the requirements for initial reactor power level.

420.05 If used, provide the results of a zone analysis and a plant walkdown. If zone analysis was not used, describe the procedure by which the locations of non-safety related control system components affected by HELB's were determined.

Response

As stated in our March 14, 1983 letter (Question 420.05-approach), CEI identified all major components of systems which could significantly impact reactivity, reactor vessel level, reactor system pressure and decay heat removal. Once these non-safety related control system components were identified, locations were determined from plant layout drawings. Components were then assigned to their respective environmental zones. All equipment within a single zone would be assumed failed and the impact on system alignments was determined. The process was repeated for each environmental zone.

Potential system interactions were identified for further evaluation. A plant walkdown was then performed to determine the location of high energy lines with respect to potentially affected components.

The results of the plant walkdown plus engineering analysis then determined the consequences of each adverse condition. The results of the analysis are contained in the evaluation section, for each adverse condition, in our March 14, 1983 letter. Supplemental information is provided in our September 6, 1983 letter and also contained in this letter.

Q420.06 Verify that no credit was taken in the analysis for non-safety related equipment (e.g., feedwater trip on level 8) to mitigate the effects of the HELB and consequential control system failures.

Response According to 15.0.1, "only safety grade equipment can be assumed to mitigate an accident and safely shut down the reactor." Our response to 420.04 is bounded by FSAR Chapter 15 accident analyses, which do not take credit for non-safety related equipment.

Q420.07 Verify that the consequences of the worst case event combinations considered in your analysis are bounded by a small fraction (<10%) of 10 CFR 100 guideline doses.

Response Consequences of the scenario postulated in 420.04 are bounded by 10 CFR 100 limits. This is consistent with the SRP requirements for analysis of HELBs as defined in Branch Technical Position ASB 3-1 Section B.3.a.