

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

PROCESS PIPING PENETRATING

PRIMARY CONTAINMENT

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333

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TABLE 3.7-1

PROCESS PIPELINE PENETRATING PRIMARY CONTAINMENT

(Numbers in parentheses are keyed to numbers on following page; equal codes are listed codes are listed on following pages)

Line Isolated	Drywell Penetration	Valve Type (6)	Power to Open (5) (6)	Group	Location Ref. to Drywell	Power to Close (5) (6)	Isolation Signal	Closing Time (7)	Normal Status	Remarks and Exceptions
Main Steam Line	X-7A,B,C,D	AO Globe	Air and AC, DC	A	Inside	Air and spring	B,C,D,P,E	Note (1)	Open	
Main Steam Line	X-7A,B,C,D	AO Globe	Air and AC, DC	A	Outside	Air and spring	B,C,D,P,E	Note (1)	Open	
Main Steam Line Drain	X-8	MO Gate	AC	A	Inside	AC	B,C,D,P,E	15 sec	Closed	
Main Steam Line Drain	X-8	MO Gate	DC	A	Outside	DC	B,C,D,P,E	15 sec	Closed	
From Reactor Feedwater	X-9A, B	Check	-	A	Outside	Process	Rev. flow	NA	Open	
From Reactor Feedwater	X-9A, B	Check	-	A	Inside	Process	Rev. flow	NA	Open	
Reactor Water Sample	X-41	AO Globe	Air and AC	A	Inside	Spring	B,C	NA	Open	
Reactor Water Sample	X-41	AO Globe	Air and AC	A	Outside	Spring	B,C	NA	Open	
Control Rod Hydraulic Return	X-36	Check	-	A	Inside	Process	Rev. flow	NA)		
Control Rod Hydraulic Return	X-36	Check	-	A	Outside	Process	Rev. flow	NA)	Opens on Rod movement and closed at all other times, Note (4)	
Control Rod Drive Exhaust	X-38	SO Valves	Air and AC	A	Outside	Spring	Note (4)	NA)		
Control Rod Drive Exhaust	X-38	SO Valves	Air and AC	A	Outside	Spring	Note (4)	NA)		
Control Rod Drive Inlet	X-37	SO Valves	Air and AC	A	Outside	Spring	Note (4)	NA)		
Control Rod Drive Inlet	X-37	SO Valves	Air and AC	A	Outside	Spring	Note (4)	NA)		

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

PROCESS PIPING PENETRATING

PRIMARY CONTAINMENT

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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Section I - Description of the Change

This proposed Amendment to Appendix A of the James A. FitzPatrick Nuclear Power Plant Facility Operating License revises Table 3.7-1 (Process Pipeline Penetrating Primary Containment) on page 198 to correct an inconsistency revealed during normal operation and described in Reference b. Isolation signals for two Reactor Water Sample Valves (drywell penetration X-41) are changed to "B, C" from "B, C, D, P, E." (Refer to page 206 of Appendix A for a description of the isolation signal codes).

Section II - Purpose of the Change

During normal operation, it was determined that the Reactor Water Sample Isolation Valves associated with penetration X-41 did not isolate on all signals required by Table 3.7-1. The table was found to be in error. A design change (made during initial construction) had not been incorporated in the Technical Specifications. During construction of the plant, the isolation signals provided these valves were properly reviewed and changed. The FSAR was not updated to reflect this change, and therefore Table 3.7-1 was not updated. Immediate corrective action was to voluntarily keep these valves closed.

The Commission was informed of this via Reference b.

The purpose of this change is to rectify the inconsistency between the plant as it was constructed and Table 3.7-1 of the Technical Specifications.

Section III - Impact of the Change

This change will have no impact on the operation of the plant since it's only purpose is to correct an error in the Technical Specifications as they are currently written.

Section IV - Implementation of the Change

The change, as proposed, will not impact the ALARA or Fire Protection Program at FitzPatrick. The proposed change will not impact the environment.

Section V - Conclusion

The incorporation of these changes: a) will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility of an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR50.92.

Section VI - References

- (a) James A. FitzPatrick Nuclear Power Plant Final Safety Analysis, revised July 14, 1982.
- (b) November 1, 1982 letter, C.A. McNeill, Jr. (PASNY) to R. C. Haynes (NRC) regarding LER 82-045/03L-0.