

50-289

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METROPOLITAN EDISON COMPANY

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December 13, 1978
GQL 1971

Director of Nuclear Reactor Regulation
Attn: R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

This letter, and the enclosure, are in response to your letter of November 20, 1978, in which you requested Met-Ed to submit written responses to your questions concerning the inservice inspection and testing program at TMI-1.

Our responses reflect Met-Ed's position on the ISI Program as discussed during the meetings held October 18 and 19, 1978.

Met-Ed is preparing a revision to the inservice inspection and testing program as submitted on August 17 and September 30, 1977. This revision will include those changes deemed necessary as a result of the meeting of October 18 and 19, 1978, and may include, as deemed necessary, changes as a result of your forthcoming safety evaluation report.

Sincerely,

J. G. Herbein
J. G. Herbein
Vice President-Generation

JGH:DGM:cjg

Enclosure

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Response to NRC Questions

I. Class 1 Components

General

The NRC questions appear to be directed at obtaining technical justification for concentrating ISI inspections of Class 1 components on selected welds rather than on the random basis described in Section XI of the ASME Code. The technical justification for this approach was summarized in Paragraph V, Bases for Inspection, of Attachment A of Met-Ed's August 1977 submittal to the NRC; that paragraph reads as follows:

"V. Bases for Inspection

The inspection program detailed in Table A-1 below follows the Code, except that inspections are focused on those areas which engineering analysis indicates are subject to relatively more critical conditions of stress, fatigue, radiation, and/or thermal cycle. Inspections are also required of those areas which had recordable indications during the preservice baseline examination. It is considered that inspection of areas subjected to relatively more critical conditions or which have pre-existing indications will provide good assurance of identification of any potential problems before significant flaws develop in the Class 1 component pressure boundaries."

Fundamentally, the approach taken by Met-Ed in regard to the inservice inspection program has always been that the inservice inspection effort should be directed at those areas of the plant which are most likely to develop problems, and that areas for inspection should not be selected in a random basis. Met-Ed's reasons for taking this focused approach have been as follows:

By a more judicious selection of inspection locations, the effectiveness of the inspection is improved. For example, experience indicates that welds subjected to the highest fatigue and stress conditions are more likely to degrade than welds subjected to milder conditions in the same environment.

Likewise, experience indicates that defect growth often initiates at existing flaws. Accordingly, the focused approach concentrates the planned ISI inspections on the higher stressed and fatigued welds and on areas with known flaws.

The use of the focused approach is considered to provide at least the same degree of protection against undetected defect growth as the code approach, while requiring a reduced number of inspection. Met-Ed expects this to result in significantly reduced radiation exposure to personnel, which is considered to be highly desirable.

As mentioned above, the Met-Ed ISI program for Class 1 components has always been based on the focused approach. It was originally developed in 1968 and 1969, has been in the TMI-1 & 2 technical specifications since their original issue, and has been accepted by the NRC several times. Over the years the program has been updated to include relatively minor changes to reflect new information. The type of information used to update the program has included final values of calculated stress and fatigue usage factors, locations of recorded but acceptable indications in welds based on preservice inspection, and results of inspections at other plants. Met-Ed anticipates that further updating will be required in the future, as experience at TMI and other plants is obtained.

The most recent submittal to the NRC of the TMI-1 ISI program for Class 1 equipment was in Attachment A to the August 17, 1977, letter, and was prepared because of the change to 10CFR50, paragraph 50.55 a(g), which requires resubmittal of the program every 40 months. In December 1976, before preparing the submittal, Met-Ed had an informal meeting with cognizant NRC Engineering Branch personnel to discuss whether they were still receptive to the focused approach.

They indicated that they were receptive, because they consider the reduction in personnel radiation exposure associated with the focused approach to be very desirable, and probably absolutely necessary as plant radiation levels increase over the years. This favorable response confirmed Met-Ed's intentions to retain the focused approach, which Met-Ed considers to provide substantial advantages as compared to the ASME Code approach.

The focused approach type program in the August 17, 1977, submittal is essentially an update of the earlier program in the technical specifications, with changes to reflect new information as described above. It is based on a detailed review of the final calculated stresses and fatigue usage factors for TMI-1 components, to ensure that the highest stressed and fatigued welds are selected for inspection. This review is documented in MPR-397, Revision 1, "Technical Basis for TMI Unit No. 1 Inservice Inspection Program for Class 1 Components", dated April 1977. The August 1977 program also includes inspection of all indications recorded during the preservice inspections. Met-Ed considers the program described in Attachment A to the August 17, 1977, submittal to be a fully satisfactory program for assuring the continued integrity of the Class 1 pressure boundary.

Detailed Responses

Response to I.1:

As discussed under "General" above, in the focused approach inspections are concentrated on those welds which engineering evaluation shows are the relatively most likely to develop defects. This approach is applied to the reactor vessel and pressurizer shell welds as follows:

- a. The inspections of reactor vessel shell welds are concentrated on core belt welds, which are subject to embrittlement due to irradiation, and on flange-to-head, flange-to-vessel, and nozzle welds, which see the greatest stress ranges and fatigue usage factors. (See Figure 1 of MPR-397.) The remaining shell welds are subjected to significantly lower stress and fatigue conditions and thus are not selected for inspection.
- b. The inspections of welds in the pressurizer shell are concentrated on the weld intersections at the corners of the heater belt forging, since this is an area of significantly higher stresses and usage (see Figure 2 of MPR-397), and on nozzle to vessel welds, which also have significantly higher stresses and usage (especially the surge nozzle). The remaining welds are subjected to significantly lower stress and fatigue conditions, and thus are not selected for inspection.

Response to I.2:

The nozzles selected for inspection are those with the highest stresses and usage factors, as shown in Figures 1 and 2 of MPR-397.

Response to I.3:

The dissimilar metal weld on the core flood nozzle leading to tank A has the highest stress level of the two reactor vessel nozzles with dissimilar metal welds as shown in Figure 6 of MPR-397. The dissimilar metal welds at the four outlet nozzles of the reactor coolant pumps see higher stresses than the pump inlet welds, as shown in Figure 4 of MPR-397. Accordingly, the higher stressed core flood nozzle and the four reactor coolant pump outlet nozzle welds have been selected for inspection.

Responses to questions(a), (b), and (c) above are as follows:

- (a) The expected dose rate at the core flood nozzle safe ends is about 2 to 3 r/hr. The dose rates around the reactor coolant pump safe ends are about 0.5 r/hr.
- (b) The man-hours to perform a pump dissimilar metal weld examination is about 10. The man-hour to perform the inspection of a core flood nozzle have not been estimated in detail, but probably are in the neighborhood of 100 to 200, considering the need to remove and re-install the seal plate, sand plugs, insulation and remote inspection gear. This work will be at the reactor vessel flange level and below, in a radiation field of about 2 r/hr.

- (c) The total man-rem involved in completing a reactor coolant pump dissimilar metal weld inspection is estimated to be 5 man-rem and to complete a core flood nozzle safe end about 200 to 400 man-rem.

Responses to I.4:

Experience at other plants has shown that cladding inspections do not provide significant information. In some cases cladding cracks have been noted (e.g., in BWR vessel heads). However, the conclusion regarding these cracks has been that they are not deleterious. Because of this experience, Section XI of the ASME Code has been revised to delete cladding inspections. This deletion was not questioned by NRC in the NRC-ASME meeting of October 11, 1977, where all recent changes to the code were reviewed in detail, and where agreement was reached in regard to changes needed to make the code acceptable to the NRC.

Response to I.5:

Yes -- there are 24 peripheral control rod housings and the three selected for inspection are more than 10% of the peripheral housings.

Response to I.6:

There are no longitudinal welds. The length of the welds in the circumferential tubesheet to head welds which will be inspected because of ultrasonic reflectors found in the preservice inspection will meet the 5% requirement.

Response to I.7:

The nozzle-to-vessel welds in the steam generators are not scheduled for volumetric examination, since the stresses and usage factors for these nozzles are significantly less than those in the reactor vessel outlet nozzle welds and RC pipe surge nozzle weld (which are to be inspected) as shown in the table below:

<u>Weld</u>	<u>Stress Intensity (psi)</u>	<u>Usage Factor</u>
S. G. nozzle-to-vessel welds	20,000	0.01
RC pipe surge nozzle weld	22,000	0.10
Reactor vessel outlet nozzle weld	44,000	< 0.67, but much more than 0.1

Response to I.8:

The situation for these three categories of welds is as follows:

B4.5 - Circumferential and longitudinal pipe welds -- The number of welds to be inspected is less than the 25% called for by the Code. However, for each size range of pipe in each system, the highest stressed welds have been selected for inspection, which we consider to provide the optimum means to monitor for degradation.

B4.7 - Branch pipe connections 6 inches or less in nominal diameter -- The number of welds to be inspected is less than the 25% called for by the Code, but the number of inspections is more than required by the Code. There are 9 welds in this category, and the Code would require about 2 to be inspected each 10-year interval. However, rather than inspecting two welds, it was considered preferable to perform three inspections of the normal water injection nozzle, since it experiences cold water injection into a hot pipe, which has been found to be a severe condition.

B4.8 - Socket Welds - Except for one line, 25% of these welds will be inspected, since detailed stress analyses were not required or performed and thus do not permit the focused approach to be applied. The focused approach was applied to the auxiliary spray line and, for it, less than 25% of the welds were selected.

II. Class 2 Components

Response to II.1:

- a. The examinations intended for Item C1.1 will satisfy the requirements for the service life of the plant.
- b. Item C1.2 is incorrect as submitted. Four welds will be inspected during the service life of the unit. DH System-2 welds; Steam Generators-2 welds.

Response to II.2:

- a. Item C3.2 has been revised such that the bolting of one decay heat system flange will be inspected during the 10 year interval instead of during the service life of the unit.
- b. Item C4.2 has been revised such that the bolting of two decay heat system valves and one main steam system valve will be inspected during the 10 year interval instead of during the service life of the unit. Also, the bolting of one valve will be examined during this inspection period.

Response to II.3:

Item C3.4 has been revised to examine one pump support component during the inspection interval in lieu of during the service life of the unit.

Response to II.4:

Item C1.1 - This item is in compliance with the Code.

Item C1.2 - This item will be revised as stated in the response to NRC Question II.1, above.

Item C1.4 - This item will be revised to state that the pressure retaining bolting of 3 flanges will be inspected during the ten year interval. (MS system - two flanges and DH system - one flange) The bolting of one MS flange will be inspected this inspection period.

Item C2.1 - The number of welds to be examined during the service life should be changed from 180 to 176. The four air handling system welds should be included in Item C4.1.

Item C2.2 - There is an error in the number of welds to be inspected in the 40 year service life of the unit. This should be 90 rather than 170. Based on the 90 welds per service life, eight welds will be inspected in this inspection period.

Item C2.4 - Ten flanges will be inspected during the ten year interval rather than the service life of the unit. This change increases the number of flanges in the inspection period from one to three.

Item C2.5 - The number of integrally welded pipe supports should be increased from 28 to 95. This change is the result of the addition of 67 pipe to penetration welds inside containment. These welds, although not required by the Code to be inspected, are considered highly stressed and should be examined. In addition, the inspection frequency is changed from service life to interval. As a result, 32 welds will be examined during the period.

Item C2.6 - This item will be revised such that 93 pipe hangers will be examined during the ten year interval in lieu of during the service life.

III. Class 3 Components

Response to III:

The operating pressures of the buried piping systems listed on Table C-2 are as follows:

Nuclear Service River Water System: 20 - 40 psig

Decay Heat River Water System: 20 - 30 psig

Reactor Building Emergency Cooling System: 60 psig

Since these systems are low pressure high volume systems, the leaks that would result from pipe breakage would be of no great significance. If pipe breakage should occur, the piping alignment would be maintained by the soil around the pipe.

Decay Heat River Water System and the Reactor Building Emergency Cooling System are each redundant systems and loss of one underground line would only result in loss of system redundancy.

IV. Pump Testing Program

Response to IV.1:

IWP-4310 states, "The temperature of all centrifuged pump bearings outside the main flow path ... shall be measured..." Since the subject pump bearings are in the main flow path, there is no requirement to measure these pump bearing temperatures.

Response to IV.2:

The Code does not allow for this type of testing which would yield "ball park estimates" of flow. The results could not be applied to Table IWP-3100-2, Allowable Ranges of Test Quantities. Also, it is questionable if the results would be repeatable.

Response to IV.3:

Differential pressure will be determined by subtracting the calculated pump inlet static pressure from the pump running discharge pressure.

Response to IV.4:

These pumps have self contained oil reservoirs which do not have oil cooling pipe lines. Therefore, oil temperature can not be obtained, prior to cooling.

Response to IV.5:

All pumps are constant speed except EF-P1.

V. Valve Testing Program

Response to V.1:

BS-V30 A/B - Testing this valve in the manner suggested imposes undue risk on the plant in that water may be discharged from the Building Spray system spray nozzles.

BS-V21 A/B - Testing this valve in the manner suggested could result in the introduction of sodium thiosulfate into the Reactor Coolant System, contributing to corrosion and/or metallurgical problems.

BS-V52 A/B - The same situation exists with BS-V52 A/B as with BS-V21 A/B except that sodium hydroxide may be introduced into the system.

EF-V3 - The physical arrangement of the Emergency Feed Pump suction piping will permit testing as suggested, but since the piping is located in an inactive part of the system, the introduction of water would stir up sediment and corrosion products that may have accumulated.

Note: The maximum flow that can be obtained through a vent or drain connection is only sufficient to verify that the disc just leaves the seat. It is felt that the possibility of introducing dirty water or sodium compounds does not warrant the performance of a test that would yield insignificant results.

Response to V.2:

These check valves are in the fluid block system and are not containment isolation valves. Their function is to open to allow the system to pressurize the bonnets of containment isolation valves to a pressure greater than accident pressure such that any leakage through the containment isolation valve will be into containment.

Response to V.3:

SF-V23 is a containment isolation valve.

Response to V.4:

The function of WDL-V362 is to prevent backwashing of the Deborating Demineralizers. It has no safety function, therefore, and should be deleted from the submittal. Initially it was thought that in an accident, this valve would be called upon to close as the Boric Acid Pumps start to ensure boric acid flow to the Makeup Tank. However, during an accident, the Makeup Tank is isolated by closure of MU-V12 and high pressure injection is from the BWST.

Response to V.5:

The only pressure isolation valves important to safety are containment isolation valves and these are already defined by Tech. Spec. as Appendix J valves. Therefore, there is no requirement to list different types of Category A valves. The safety function of containment isolation valves is to close during accident conditions and this is already verified by Appendix J testing. The safety function of valves that are required to open during accident conditions is to open and, where possible, this function is tested.

Response to V.6:

RR-V10 A/B - Type of Test will be revised to Timed Stroke Test on a quarterly basis. MS-V⁴ A/B - Type of Test is to be changed to a Timed Stroke Test on a Cold Shutdown Frequency. MS-V6 - This is a regulating valve whose function is to control the Emergency Feed Pump Turbine speed. Its ability to do so is verified during the monthly pump functional test.

EF-V30 A/B and AH-V11 A/B - Para E.3. of the submittal applies. These tests are classified as Functional rather than Part Stroke because a Part Stroke Test requires a Full Timed Stroke Test at Shutdown Conditions.

Response to V.7:

The TMI-1 locked valve list as of 8-17-78 is attached. Many of these valves are not included within the scope of the TMI-1 inservice inspection program. A revised list will be included with the forthcoming program revision.

Revision to V.8:

Not all relief valves in safety related systems are listed, but all relief valves with a safety function are listed.

Response to V.9:

BS-V1 A/B and BS-V30 A/B are not defined by Tech. Spec. as containment isolation valves per Appendix J. Pressure isolation is not an issue since these valves are open during accident conditions and flow is into containment.

Response to V.10:

ASME Section XI IWV 2110 defines Category A valves as, "Valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their function". Valves DH-V4 A/B and DH-V22 A/B are not containment isolation valves per Tech. Spec. The safety function of these valves is open to supply borated water from the Borated Water Storage Tank and to recirculate borated water in the Reactor Building Sump for long term cooling after an accident.

Response to V.11:

DR-V21 A/B and DR-V22 A/B are shown on ISI Diagram C-300-014-GN1. Their safety related function is to open to supply bearing flushing and lube water to DR-P1 A/B. DR-V21 A/B is the primary water supply. DR-V22 A/B provides the secondary supply of water.

Response to V.12:

ASME Section XI IWV 2110 defines Category A valves as, "Valves for which seat leakage is limited to a specific maximum amount in closed position for fulfillment of their function". Valves CF-V4 A/B and CF-V5 A/B are not containment isolation valves as defined by Appendix J. Instead, these valves open to supply borated water from the Core Flooding Tanks to Reactor Vessel whenever the RCS pressure falls below the pressure in the tanks. Leakage past these two valves (which are arranged in series) need not be controlled in order for these valves to fulfill their function. Each Core Flooding Tank is protected by a pressure relief valve which exhausts inside containment.

In addition, Surveillance Procedure No. 1301-1 requires that the Core Flooding Tank level and pressure be monitored each shift, and checked for compliance with TMI-1 Technical Specification limitations.

We, therefore, believe that CF-V4 A/B and CF-V5 A/B do not meet the ASME Section XI definition of Category A valves and should not be classified or tested as Category A valves. We believe that the Tech Spec Surveillance Criteria is sufficient to ensure reliable operation of CF-V4 A/B and CF/V5 A/B.

CF-V1 A/B are open when the Reactor Coolant System pressure is above 700 psig, and during normal operation, these valves stay open. Therefore, CF-V1 A/B does not have a pressure isolation function.

Response to V.13:

The safety function of RR-V3A/B/C and RR-V4 A/D is to open on E.S. actuation signal to supply river water from the Reactor Building Emergency Cooling Pumps to the Reactor Building Emergency Cooling Coils. The Safety Function of RR-V9 A/B/C is to open to allow flow through the Reactor Building Emergency Cooling Coils. RR-V9 A/B/C will be listed and categorized.

NS-V11 is a normally open check valve and this valve is not relied upon for leaktightness, and therefore, it is not listed. However, the upstream gate valve, outside the Reactor Building (NS-V15), is checked for leaktightness.

Response to V.14:

This system is located on Metropolitan Edison Company Drawing C-300-014-GN1 and not C-300-007-GN1 as incorrectly stated in Table 3-1 of the submittal. Table E-1 will be corrected.

Response to V.15:

SW-V3 A/B - The function of this valve is to open upon pump start to allow screen wash flow. Valves SW-V27 A/B, SW-V11 A/B and SW-V13 A/B all open to provide a primary supply of bearing cooling water to the screen wash pumps. SW-V14 A/B provides a secondary supply of bearing flushing water to the screen wash pumps.

Response to V.16:

The safety function of MU-V16 A/D and MU-V107 A/D is to open to provide HPI (High Pressure Injection) on an ES actuation signal. HPI provides borated water from the Borated Storage Tank to the Reactor Core. DWG C-300-017 has been reviewed and we believe that the valves are listed where applicable and categorized correctly.

Response to V.17:

These valves are normally open check valves and are not relied upon for leaktightness and are, therefore, not leak checked. However, the upstream gate valves, outside of the Reactor Building, are checked for leaktightness.

Response to V.18:

EF-V3 - This valve was discussed in NRC Question V.1. EF-V4&5 - These valves are to be deleted from submittal as they are locked closed and included on locked valve list attached. EF-V11 A/B and 13 - The testing of these valves is to be deleted as there is no safe way of doing so without subjecting personnel to extremely high pressure water. EF-V12 A/B - Testing these valves imposes the EF nozzles to thermal cycling and should, therefore, be deleted. EF-V30 A/B - This valve was discussed in NRC Question V.6.

The remaining valves on Drawing C-300-009-GN1 have been reviewed and should not be listed.

Response to V.19:

RC-V2 is a normally open isolation valve for RC-RV2 that was installed for operator convenience. Under normal Reactor operation, RC-RV2 has no safety function since the Reactor Coolant System is protected by the Code Relief Valves.

When the Reactor Coolant System Temperature is below 275°F, RC-RV2 is switched to AUTO and will insure that NDTT limits are not exceeded. RC-RV2 will lift at 485 psig and will reseal at 435 psig when switched to the Auto Position.

TMI-1

LOCKED VALVE LIST

<u>VALVE</u>	<u>POSITION FOR NORMAL OPERATION</u>	<u>REMARKS</u>
<u>Condensate</u>		
CO-V32	Locked Open	
CO-V34	Locked Open	
CO-V112A	Locked Open	
CO-V112B	Locked Open	
CO-V112C	Locked Open	
CO-V176	Locked Open	
<u>Core Flood</u>		
CF-V26A	Locked Open	
CF-V26B	Locked Open	
CF-V30A	Locked Open	
CF-V30B	Locked Open	
CF-V-3A	Breaker Tagged Open	
CF-V-3B	Breaker Tagged Open	
<u>Decay Heat Removal</u>		
DH-V12A	Locked Closed	
DH-V12B	Locked Closed	
DH-V15A	Locked Open	
DH-V15B	Locked Open	
DH-V19A	Locked Open	May require throttling
DH-V19B	Locked Open	May require throttling
DH-V20A	Locked Closed	
DH-V20B	Locked Closed	
DH-V21	Locked Closed	
DH-V-38A	Locked Closed	
DH-V38B	Locked Closed	
DH-V52	Locked Closed	
DH-V56A	Locked Open	
DH-V56B	Locked Open	
DH-V62	Locked Open	
DH-V63	Locked Open	
DH-V64	Locked Closed	
DH-V68A	Locked Closed	
DH-V68B	Locked Closed	
DH-V69	Locked Open	
<u>Diesel Generator</u>		
EG-V-1006	Open	Locking Device or Collar Installed
EG-V-1007	Open	Locking Device or Collar Installed
EG-V-12A/A	Open	
EG-V-12A/B	Open	
EG-V-12B/A	Open	
EG-V-12B/B	Open	

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<u>VALVE</u>	<u>POSITION FOR NORMAL OPERATION</u>	<u>REMARKS</u>
<u>Emergency Feed</u>		
EF-V4	Locked Closed	
EF-V5	Locked Closed	
EF-V20A	Locked Open	
EF-V20B	Locked Open	
EF-V22	Locked Open	
<u>Extraction Steam</u>		
EX-V15A	Locked Open	
EX-V15B	Locked Open	
EX-V54	Locked Open	
<u>Feedwater</u>		
FW-V10A	Locked Open	
FW-V10B	Locked Open	
FW-V11A	Locked Open	
FW-V11B	Locked Open	
<u>Hydrogen Purge</u>		
HP-V1	Locked Closed	
HP-V6	Locked Closed	
HP-V7	Locked Closed	
<u>Instrument Air</u>		
IA-V6	Locked Closed	
IA-V20	Locked Closed	
<u>Leak Rate Test</u>		
LR-V1	Locked Closed	
LR-V2	Locked Closed	
LR-V3	Locked Closed	
LR-V4	Locked Closed	
LR-V5	Locked Closed	
LR-V6	Locked Closed	
LR-V49	Locked Closed	
<u>Main Steam</u>		
MS-V25A	Locked Closed	
MS-V25B	Locked Closed	
MS-V24A	Locked Closed	
MS-V24B	Locked Closed	
<u>Liquid Waste Disposal</u>		
WDL-V240	Locked Closed	
WDL-V241	Locked Closed	

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<u>VALVE</u>	<u>POSITION FOR NORMAL OPERATION</u>	<u>REMARKS</u>
WDL-V242	Locked Closed	
WDL-V378	Locked Open	
WDL-V379	Locked Closed	
WDL-V408	Locked Open	
WDL-V409	Locked Open	
WDL-V410	Locked Closed	
WDL-V411	Locked Closed	
WDL-V423	Locked Closed	

Makeup and Purification

MU-V64A	Locked Open	
MU-V64B	Locked Open	
MU-V64C	Locked Open	
MU-V68A	Locked Open	
MU-V68B	Locked Open	
MU-V69A	Locked Closed	
MU-V69B	Locked Closed	
MU-V72A	Locked Open	
MU-V72B	Locked Open	
MU-V72C	Locked Open	
MU-V75	Locked Closed	
MU-V74A	Locked Open	
MU-V74B	Locked Open	
MU-V74C	Locked Open	
MU-V76A	Locked Closed (Break-Away Lock)	
MU-V76B	Locked Open	
MU-V77A	Locked Open	
MU-V77B	Locked Open	
MU-V78	Locked Closed	
MU-V113	Locked Open	

Nitrogen

NI-V26	Locked Closed	
NI-V27	Locked Closed	

Penetration Pressurization*

PP-V-8	Locked Open	
PP-V-9	Locked Closed	
PP-V-17	Locked Open	
PP-V-18	Locked Closed	
PP-V-23	Locked Open	
PP-V-24	Locked Closed	
PP-V-32	Locked Open	
PP-V-39	Locked Closed	
PP-V-40	Locked Open	
PP-V-41	Locked Closed	
PP-V-47	Locked Open	
PP-V-50	Locked Open	
PP-V-51	Locked Closed	

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<u>VALVE</u>	<u>POSITION FOR NORMAL OPERATION</u>	<u>REMARKS</u>
PP-V-57	Locked Open	
PP-V-58	Locked Closed	
PP-V-63	Locked Open	
PP-V-64	Locked Open	
PP-V-65	Locked Open	
PP-V-66	Locked Closed	
PP-V-70	Locked Closed	
PP-V-76	Locked Closed	
PP-V-82	Locked Open	
PP-V-83	Locked Closed	
PP-V-90	Locked Open	
PP-V-91	Locked Closed	
PP-V-99	Locked Open	
PP-V-111	Locked Open	
PP-V-114	Locked Closed	
PP-V-139	Locked Open	
PP-V-141	Locked Open	
PP-V-142	Locked Closed	
PP-V-143	Locked Closed	
PP-V-168	Locked Closed	
PP-V-171	Locked Closed	
PP-V-174	Locked Open	
PP-V-177	Locked Open	
PP-V-178	Locked Open	
PP-V-179	Locked Open	

Reactor Building Spray

BS-V17A	Locked Open	
BS-V17B	Locked Open	
BS-V25A	Locked Closed	
BS-V25B	Locked Closed	
BS-V37A	Locked Open	
BS-V37B	Locked Open	
BS-V-37C	Locked Open	
BS-V-37D	Locked Open	
BS-V41A	Locked Open	
BS-V41B	Locked Open	
BS-V49A	Locked Open	
BS-V49B	Locked Open	
BS-V53A	Locked Open	
BS-V53B	Locked Open	
BS-V54A	Locked Open	
BS-V54B	Locked Open	
BS-V59	Locked Closed	
BS-V60A	Locked Closed	
BS-V60B	Locked Closed	

Reclaimed Water

CA-V171	Locked Closed	
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<u>VALVE</u>	<u>POSITION FOR NORMAL OPERATION</u>	<u>REMARKS</u>
<u>Service Air</u>		
SA-V2	Locked Closed	
SA-V3	Locked Closed	
<u>Spent Fuel</u>		
SF-V22	Locked Closed	
SF-V23	Locked Closed	
SF-V31	Locked Closed	
SF-V48	Locked Open	
SF-V66	Locked Open	
SF-V73	Locked Closed	
SF-V74	Locked Closed	
SF-V75	Locked Closed	
SF-V76	Locked Closed	
<u>Sump Pump & Drainage (Rx. and Aux. Building)</u>		
WDL-V540	Locked Open	
WDL-V541	Locked Open	
WDL-V542	Locked Closed	
WDL-V549	Locked Closed	
WDL-V539	Throttled to provide ~ 2 GPM Flow through RM-L8 with one sump pump running	
<u>Turbine Lube Oil</u>		
LO-V1	Locked Closed	
LO-V10A	Locked Closed	
LO-V10B	Locked Closed	
<u>Waste Gas</u>		
WDG-V30	Locked Closed	
WDG-V31	Locked Closed	
WDG-V32	Locked Closed	
WDG-V67	Locked Open	
WDG-V68	Locked Open	
WDG-V69	Locked Closed	
WDG-V70	Locked Closed	
WDG-V103	Locked Closed (RM-A7 Test Connection)	
<u>Nuclear River Water</u>		
NR-V30	Locked Open	
<u>Decay Heat Closed Cooling</u>		
DC-V20A	Open and Sealed	
DC-V20B	Open and Sealed	
DC-V21A	Open and Sealed	
DC-V21B	Open and Sealed	
DC-V23A	Open and Sealed	

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<u>VALVE</u>	<u>POSITION FOR NORMAL OPERATION</u>	<u>REMARKS</u>
DC-V23B	Open and Sealed	
DC-V24A	Open and Sealed	
DC-V24B	Open and Sealed	
DC-V31A	Open and Sealed	
DC-V31B	Open and Sealed	
DC-V32A	Open and Sealed	
DC-V32B	Open and Sealed	
DC-V33A	Open and Sealed	
DC-V33B	Open and Sealed	
DC-V34A	Open and Sealed	
DC-V34B	Open and Sealed	
DC-V35A	Open and Sealed	
DC-V35B	Open and Sealed	
DC-V36A	Open and Sealed	
DC-V36B	Open and Sealed	
DC-V37A	Open and Sealed	
DC-V37B	Open and Sealed	
DC-V38A	Open and Sealed	
DC-V38B	Open and Sealed	
DC-V39A	Open and Sealed	
DC-V39B	Open and Sealed	
DC-V40A	Open and Sealed	
DC-V40B	Open and Sealed	
DC-V42A	Open and Sealed	
DC-V42C	Open and Sealed	
DC-V43A	Open and Sealed	
DC-V43C	Open and Sealed	
DC-V44A	Open and Sealed	
DC-V44C	Open and Sealed	
DC-V58A	Open and Sealed	
DC-V58B	Open and Sealed	

Nuclear Service Closed Cooling

NS-V30A	NS & DH Pump	Open and Sealed
NS-V31A	Area Air Cooler	Open and Sealed
NS-V30B	Aux. Bldg.	Open and Sealed
NS-V31B	305' Elev.	Open and Sealed
NS-V69A		Open and Sealed
NS-V71A	Intermediate	Open and Sealed
NS-V69B	Bldg. 295'	Open and Sealed
NS-V71B	Elev. RB Fan	Open and Sealed
NS-V69C	Motor Cooling	Open and Sealed
NS-V71C		Open and Sealed
NS-V76	MU-P1B	Open and Sealed
NS-V77	Cooling	Open and Sealed
NS-V78	Aux. Bldg.	Open and Sealed
NS-V79	281' Elev.	Open and Sealed

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