William J. Cahill, Jr.
Vice President

CENTRAL FILES

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

April 26, 1979

Re: Indian Point Unit No. 2
Docket No. 50-247

U. S. Nuclear Regulatory Commission
ATTN: Mr. Boyce H. Grier, Director
Region I
Office of Inspection and Enforcement
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier:

Enclosed is our response to IE Bulletin 79-06A which was received in this office on April 16, 1979, as revised by Revision No. 1 which was received in this office on April 23, 1979.

Should you have any questions regarding our response and the actions we have taken or are planning to take, please contact this office.

Sincerely

William J. Cahill, Jr.
Vice President

cc: U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

Mr. T. Rebelowski, Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York, 10511

Mr. Leonard N. Olshan, Project Manager
Operating Reactors Branch No. 1
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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Responses to IE Bulletin 79-06A
And Revision No. 1 to IE Bulletin 79-06A

Consolidated Edison Company of New York, Inc.

April 26, 1979

Item 1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simulataneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

Response: The requested review has been completed. This review was conducted with the following:

a) plant personnel
b) key engineering and support personnel
c) Station Nuclear Safety Committee
d) Nuclear Facilities Safety Committee
e) Westinghouse at meetings on 4/5/79, 4/16/79, 4/17/79 and 4/18/79.
f) US NRC Representatives and Con Edison operational and management personnel at Indian Point site on 4/19/79.

Item 1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

Response: The instructions to operational personnel have been made.

In addition, Westinghouse has reviewed the Guidelines for Emergency Operating Procedures which are provided to utilities and has prepared recommended modifications for incorporation into plant specific procedures. These recommended modifications to the procedures provide criteria for terminating Safety Injection and are based upon more than one plant parameter indication. These Westinghouse recommended procedure modifications are enclosed as Attachment I and II. Con Edison is currently reviewing the Westinghouse recommendations for incorporation as appropriate into our procedures.

Item 1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

Response: The review has been completed and documented.

Item 2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

Response: The procedure review has been completed. Existing procedures are considered adequate in that they contain appropriate precautions against approaching saturation conditions during operation.

In addition, Westinghouse has conducted a review of recommended actions with the result as follows:

The primary indication of steam void formation in the primary coolant system is pressurizer pressure below the hot leg saturation pressure. However, during a loss of coolant event, steam void formation in the primary coolant system would be expected with two exceptions: 1) the loss of coolant is being caused by a stuck open pressurizer relief valve which closes or is isolated before the system depressurizes to hot leg saturation, or, ii) the reactor coolant system reaches an equilibrium pressure above hot leg saturation, when the safety injection flow equals the break flow. Thus, in these two specific cases, confirmation of no voids in the system will be apparent by the pressure in the pressurizer. In the remaining cases the engineered safeguards system has been designed to cope with voiding. Thus, it is not necessary to be able to recognize void formation in those cases.

The Westinghouse NSSS Reference Operation instructions, have been reviewed. Instructions and discussion relating to recognition of the possibility of forming voids in the primary coolant system points out that maximum charging flow and reactor coolant pump seal injection flow cannot maintain the pressurizer water level under certain LOCA conditions. Further, the instructions point out that the pressurizer will be completely drained following a double ended steam generator tube rupture.

Item 2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

b. Operator action required to prevent the formation of such voids.

Response: (See response to Item 2.a.)

The result of the Westinghouse review of this item follows:

These are discussed as appropriate under Emergency Operating Instructions. Immediate actions, prior to diagnosis of the specific accident classification, which tend to prevent formation of voids, include:

1. Verify that reactor trip and safety injection have occurred.
2. Verify that residual heat is being dissipated; that is, reactor coolant temperature is not increasing.
3. Verify that feedwater is being supplied to the steam generators.
4. Operator action should be taken to maintain a water level in the pressurizer by charging and emergency makeup control, dissipate residual heat through the steam generators and maintain an indicated water level in all steam generators not directly affected by the accident.

For some LOCA cases, no operator action will prevent the formation of steam voids in the primary coolant system. The emergency safeguards system was designed to recover and cool the core following various degrees of primary coolant system voiding, depending on the break size and location Under a steam generator tube leak, Emergency Operating Instructions describe that if the leak rate is low enough, charging flow or safety injection flow will maintain the system pressure above saturation; if not the system will start to void. The operator should identify and isolate the secondary side of the faulty steam generator as quickly as possible to prevent or minimuze voiding. The procedure for isolating the faulty steam generator is also given in the Emergency Operating Instructions.

Item 2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

Response: (See response to Item 2a.)

In the unlikely event that such voids are formed, normal degassing systems (chemical and volume control systems), would be used to remove the gases and dissolve the gas bubble. Gases could be remotely vented through the pressurizer.

Further reccmmendations from Westinghouse for dealing with cases are discussed in response to Item 12.

Item 3.
(Revision 1)

For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. The pressurizer level bistables may be returned to their normal operating positions during the pressurizer pressure channel functional surveillance tests. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

Response:

Instructions were given to operational personnel on April 7, 1979 to manually initiate safety injection when the pressurizer pressure indication alone reaches the low actuation setpoint.

The low pressurizer level setpoint bistables were tripped on April 14, 1979.

Revision No. 1 to IE Bulletin 79-06A advised that the pressurizer level bistables would be returned to their normal operating positions during the pressurizer pressure channel functional surveillance tests. Accordingly, during the conduct of any surveillance test involving the pressurizer pressure channels, we plan to place all three pressurizer level bistables in the untripped mode. Immediately upon completing the test, these bistables will be returned to the tripped mode. This philosophy is consistent with discussions held with Region I personnel on April 19, 1979.

Item 4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

Response: Existing procedures have been reviewed and are considered adequate. Furthermore, the design of Indian Point Unit No. 2 incorporates such automatic containment isolation upon automatic initiation of safety injection.

Item 5. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures whch require the stationing of an indvidual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

Response: The design of Indian Point Unit No. 2 incorporates automatc auxiliary feedwater system initiation when required.

Item 6. For your facilities, prepare and implement immediately procedures which:

a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open.

Response: Our procedures incorporate these provisions. The installed indications and/or alarms available to our operators are:

a) decreasing pressurizer pressure
b) valve position indication on flight panel (stem mounted limit switches)
c) PORV/safety valve discharge line temperatures
d) increasing Pressurizer Relief Tank level (on supervisory panel)
e) increasing Pressurizer Relief Tank pressure (on supervisory panel)
f) increasing Pressurizer Relief Tank temperature (on supervisory panel)
g) possible increasing containment pressure, temperature, humidity, and radioactivity and fan cooler leakage detection system if Pressurizer Relief Tank rupture disk blows.

Item 6. For your facilities, prepare and implement immediately procedures which:

b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the setpoint for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

Response: Our procedures incorporate the requested actions.

Item 7. Review the action directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

Response: The requested review has been completed. We believe that our operating procedures and training instructions are adequate to ensure that operators do not override the automatic actions of engineered safety features, unless it has been confirmed that such actions are unnecessary or unsafe plant conditions would result.

In addition, Westinghouse has reviewed the guidelines for Emergency Operating Procedures and has prepared recommended modifications to guide the operator in terminating safety injection. This will avoid potentially adverse plant conditions under certain cases if continued operation of the safety injection system occurs.

In addition, similar criteria and guidance have been prepared for termination of safety injection following secondary side breaks which lead to a primary system cooldown. In this case continued operation of safety injection could lead to conditions which potentially could exceed reactor vessel pressure criteria. Similarly, following a steam generator tube rupture, criteria and guidance have been prepared for termination of safety injection to reduce the quantity of primary coolant which passes to the secondary side of the steam generator.

In all cases mentioned above, the criteria and guidance for terminating Safety Injection are based on the plant being in a stable, controlled condition prior to safety injection termination. Should a subsequent system disturbance occur, specific criteria for reinitiation of safety injection are also provided.

These recommended procedure modifications are enclosed as Attachment I and II.

Item 7. Review the action directed by the operating procedures and training instructions to ensure that:

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

(2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactiviated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

Response: Based on our review of procedure required actions, we believe that the High Pressure Injection System will not be defeated unless the criteria outlined in response to Item 7.a. is satisfied.

As discussed in the response to Item 7a, Westinghouse has prepared recommended modifications for Emergency Operating Procedures to provide for termination of safety injection where continued operation could potentially lead to unsafe plant conditions.

The criteria developed by Westinghouse permit termination of high head safety injection under conditions which are at least 50° subcooled, although no specific subcooling criterion is included. Rather, Westinghouse has prepared its recommended procedures based upon parametric indications available to the operator and have chosen parameter values consistent with a highly subcooled state in the primary system. For example, following a small break LOCA, the recommended criteria for terminating high head safety injection are:

a. Wide range RCS pressure $>$ 2000 psi, and
b. Wide range RCS pressure increasing, and
c. Narrow range level indication in at least 1 steam generator, and
d. Pressurizer level $\geq$ 50%.
e. No SI Flow

These provide for a subcooled primary side, stable primary side conditions and a heat sink via the steam generator.

For those LOCA conditions where both the high head and low head safety injection systems would be operating and delivering water to the primary system, current Westinghouse recommended Emergency Operating Procedure guidelines call for continued operation of both systems. Westinghouse does not recommend any changes in this portion of the procedures.

Con Edison is currently reviewing the Westinghouse recommendations for incorporation as appropriate into our procedures.

Item 7. Review the action directed by the operating procedures and training instructions to ensure that:

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating for two loop plants and at least two RCPs shall remain operating for 3 or 4 loop plants as long as the pump(s) is providing forced flow.

Response: We are continuing to review this recommendation with Westinghouse.

Since Westinghouse has not fully evaluated all of the cases encompassed by the above NRC recommendation, it is their tentative recommendation that the plant emergency instructions upon LOCA and steam break accidents remain as they are and that all RC pumps be tripped. In the Westinghouse specific guidelines, they have provided additional clarification of the conditions under which the pumps should be manually tripped. These include verification that SIS pumps are operational and that the pressure is decreasing and is below a specified setpoint which is less than SI actuation. Addiitionally, we have provided notations to alert the operator of action that should be taken to trip the pumps because of certain containment isolation or ECCS sequencing actions (i.e., isolation of component cooling). These are highlighted and consistent with existing pump instructions. (Also see Attachment I and II)

We are prepared to further discuss with Westinghouse and the Regulatory Staff the concerns that form the basis of the above.

Item 7. Review the action directed by the operating procedures and training instructions to ensure that:

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

Response: The review has been completed and instructions provided to the operators are considered adequate.

In the recommended modifications to Emergency Operating Procedures prepared by Westinghouse and discussed above in 7a, b and c, none of the criteria for operator action are based solely on pressurizer level. The criteria for operator action are based on several parameter indications. The indications provided to the operators are:

Wide range RCS temperature and pressure
Steam pressure
Steam generator water level
Containment pressure
RWST water level
Condensate storage tank level
Pressurizer water level
Boric acid storage tank level

The Westinghouse recommended procedure modifications are based on combinations of these parameter indications. From these combinations, conditions and parameters not directly measured can be inferred.

Item 8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Response: The required procedural reviews have been completed as well as a review of all accessible safety-related valve positions.

Safety-Related valves located inside containment have continuous indication of position in the control room.

Item 9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists

Response: Existing procedures have been reviewed and were found to be adequate. To aid in your review of this item, enclosed as Attachment III are updated schematics of all containment pipe penetrations.

For the specific response to item 9.a., we believe the lines being referred to are:

(See Attachment III for line No. reference)
Line No. 1, 8, 11*, 12, 18, 19, 20, 21, 31, 49, 50, 52, 53

These lines are automatically isolated by SI causing containment Phase A isolation. Resetting of SI does not reset Phase A isolation, this must be done separately. High radiation causes isolation of lines 48/49, 50 and blocks reset if the high radiation signal still exists.

*Closed system - Phase B isolation

Item 9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be cause by the resetting of engineered safety features instrumentation. List all such systems and indicate:

b. Whether such systems are isolated by the containment isolation signal.

Response: (see response to Item 9.a.)

Item 9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

c. The basis on which continued operability of the above features is assured.

Response: The basis for continued operability of the features discussed in response to Item 9.a., is the following:

a) Technical Specifications-Limiting Conditions For Operation Section, 3.5.
b) Technical Specifications-Surveillance Requirements, Section 4.1.

Item 10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

Response: The review has been completed and our procedures have been found to be adequate. Minor modifications were made to one administrative control document to provide further assurance that these verifications are accomplished.

Item 10. review and modify as necessary your maintenance and test procedures to ensure that they require:

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

Response: The review has been completed. No modifications were required.

Item 10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Response: The requested review has been completed. Our procedures contain the notification requirement.

Item 11. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

Response: Station Administrative Order No. 124 currently provides a procedure for the reporting, and appropriate followup of anomalous conditions e.g., unusual incidents, emergencies, unexpected transients, equipment malfunctions, etc. Furthermore, our "Emergency Plan for Indian Point Unit Nos. 1 and 2", which was recently revised using the guidance contained in Regulatory Guide 1.101 (Revision 1), and was submitted to the NRC on March 27, 1979 for approval, requires immediate notification of the NRC, N.Y. State and local government agencies of events which could potentially affect the health and safety of the general public.

Notwithstanding the above prompt reporting procedures, we are revising Station Administrative Order No. 124 to require that the NRC's Region 1 Office of Inspection and Enforcement be notified within one hour of the time the reactor is not in a controlled or expected condition of operation. This revision will require that an open continuous communication channel be established and maintained with the NRC.

Item 12. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

Response: For H_2 remaining in the primary system following such an event, Westinghouse review results are as follows:

The engineered safeguards are designed and analyszed to meet the limits of 10 CFR 50.46 which require that the hydrogen generation from clad water reaction in a LOCA be limited to less than 1% of the clad metal, and nowhere exceed 17% of the clad thickness.

The modes for removing hydrogen from the reactor coolant system are:

1. Hydrogen can be stripped from the reactor coolant to the pressurizer vapor space by pressurizer spray operation if the reactor coolant pump is operating.
2. Hydrogen in the pressurizer vapor space can be vented by power operated relief valves to the pressurizer relief tank.
3. Hydrogen can be removed from the reactor coolant system by the letdown line and stripped in the volume control tank where it enters the waste gas system.
4. In the event of a LOCA, hydrogen would vent with the steam to the containment.

If for some reason a non-condensible gas bubble becomes situated somewhere in the primary coolant systems, there are many options for continued core cooling and removing the bubble.

With a gas bubble located in the upper Reactor Vessel head several methods of core cooling are unaffected. The steam generator can be used to remove decay heat using reactor coolant pump forced flow or natural circulation. The safety injection system can be used to cool the core while venting through the pressurizer power operated relief valve. Core cooling by any of these methods can proceed indefinitely if the primary coolant pressure is held constant. If a lower system pressure is desired, a controlled depressurization will allow the bubble to grow slowly until it uncovers the top of the hot legs.

This controlled depressurization can be performed in two ways.:

1. If the reactor coolant pumps can be operated, depressurization can be performed with a steam bubble in the pressurizer. Depressurization would be through the pressurizer power operated relief valve. Extra control is achieved with the pressurizer heaters and sprays if available. As the bubble grows to the top of the hot leg, small bubbles are carried through the system. Degassing is done with the spray line and/or the Chemical and Volume Control System. The steam generators will carry away decay heat.

2. If the reactor coolant pumps cannot be operated or their operation is undesirable, the pressurizer can be made water solid with the safety injection pumps running and the power operated relief valve open. Balanced depressurization is controlled by judicious use of the various valves, lines and pumps available in the safety injection system and by adjusting the pressurizer relief valve. As the bubble grows to the top of the hot leg, it slides across the hot leg and up into the steam generators. As depressurization continues the gas bubbles grow in the steam generators and upper head but the core remains covered and cooled by safety injection water. If there is enough gas, the pressurizer surge line would eventually be "uncovered". Some of the gas would flow into the pressurizer and out the power operated relief valve. This process would continue until the system is at the desired pressure. At that time the current cooling mode could be continued or the system could be placed in an RHR mode (special care is needed for operation).

   Note that a gas bubble cannot be located in the steam generator with the reactor coolant pumps running. If a gas bubble forms in the steam generator during natural circulation, the reactor coolant pumps could be turned back for degassing or safety injection flow could be initiated with the power operated relief valve open.

   Also note that the gas bubbles cannot uncover the core in the above depressurization schemes because it will always tend to float to the top of the system and it cannot compress water.

The requested review for the unlikely event of the need to control significant amounts of hydrogen gas within containment has been completed. Redundant recombiners are installed inside containment at Indian Point 2 for H_2 control.

Item 13. (Revision 1) Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items and identify design changes necessary in order to effect long term resolutions of these items.

Response: At this time, we are planning design changes, (which will require technical specifications changes prior to implementation), to eliminate the coincidence on level portion of the pressurizer pressure and level as one of the initiating signals for safety injection.

We currently are planning to submit this change request within 30 days.

ATTACHMENT I

WESTINGHOUSE

RECOMMENDED CHANGES TO PROCEDURES FOR TERMINATION OF SAFETY INJECTION IN EMERGENCY EVENTS

GENERAL INSTRUCTIONS:

1. It is recommended that the emergency procedures should be revised to delay the Reset SI action until just prior to the first operator action which will change the status of the ECCS equipment automatically actuated by the S signal. This SI reset action should assure that containment isolation is maintained; do not reset Containment Isolation.

2. Folllowing an "S" signal, if offsite power is available, the operator should determine that the diesels are immediately available and operable. If possible, the diesels should be kept running at idle or minimum load.

3. For all cases where the RCS pressure equilibrates above the LHSI pump shut-off head, the LHSI pump should be stopped and placed in the standby mode. (Do not allow the LHSI pumps to operate in a mini-flow mode for prolonged periods of time unless cooling water is supplied to LHSI/RHR heat exchanger).

4. In all procedures where the plant is controlled by normal makeup and letdown and RCS pressure cannot be maintained above the setpoint for SI actuation, then it is recommended to:

   A. Manually initiate Safety Injection,

   B. Go to EOP-0.

5. In those cases where plant recovery and depressurization steps are conducted using Emergency Operating Procedures, it is recommended that those EOP's be reviewed with respect to manual or automatic steps which were placed there as a result of RCS cold overpressurization (Appendix G) concerns. It is recommended that the cold overpressurization protection systems not be activated for those cases. The operation of these systems could lead to an inadvertant RCS depressurization process through operation of the Pressurizer PORV's.

6. It is recommended that in all Emergency Operating Procedures should be revised to include the following as the <u>first step</u>:

   Monitor wide range RCS temperature and pressure, steam pressure, steam generator water level, containment pressure, RWST level, condensate storage tank level, pressurizer level and boric acid storage tank level. Verify indicator operation. Recorders should be activated.

SPECIFIC RECOMMENDATIONS FOR PROCEDURE E-1

I. It is recommended that the LOCA Procedures be revised to insert the following steps immediately after Step I above.

(1) Close all pressurizer power operated relief valves and backup isolation valves.

(2) If:

(A) Wide range RCS pressure >2000 psig, and

(B) Wide range RCS pressure is increasing, and

(C) SG narrow range level indication on at least 1 SG, and

(D) Pressurizer level $\geq$50%

Then:

(A) Stop HHSI pumps*.

*NOTE: If a RCS wide range pressure decrease in excess of 200 psi occurs following termination of HHSI flow, manually reinitiate safety injection to maintain system pressure. Control system pressure to the nominal value which occurred when HHSI was initially terminated.

Go to General Emergency Procedure (EOP-O) to reevaluate the event.

(B) Place all non-operating SI pumps in standby mode and maintain operable SI flowpaths. (Do not lock valves).

(C) Reestablish normal makeup and letdown to maintain pressurizer level in the normal operating range and to maintain system pressure at values reached when HHSI is terminated.

(D) Reestablish operation of pressurizer heaters. When system pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer level control only.

SPECIFIC RECOMMENDATIONS FOR PROCEDURE E-2

I. It is recommended that in the Emergency Operating Procedure for recovery from a loss of secondary coolant, the step which guides the operator to stop safety injection and to reestablish normal charging and letdown flows be modified or replaced to include the following:

IF:

(1) Any one wide range RCS Temperature T_{HOT} $<350^{o}F$, and

(2) Wide range RCS Pressure is increasing, and

(3) Pressurizer level is $\geq 20\%$ span and rising (heaters covered).

or

IF:

(1) Wide Range RCS Pressure >2000 psi, and

(2) All wide range RCS Temperatures T_{HOT} $>350^{o}F$, and

(3) Wide Range RCS Pressure is stable or increasing, and

(4) SG narrow range level indication in at least 1 steam generator, and

(5) Pressurizer level $\geq 50\%$.

SPECIFIC RECOMMENDATIONS FOR PROCEDURE E-2

(Continued)

THEN:

(1) Stop HHSI pumps*.

(2) Place all non-operating SI pumps in standby mode, and maintain operable SI flowpaths. (Do not lock valves.)

(3) Reestablish normal makeup and letdown to maintain pressurizer level in the normal operating range and to maintain system pressure at values reached when HHSI is terminated.

(4) Reestablish operation of pressurizer heaters. When system pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer level control only.

*NOTE: If a wide range RCS pressure decrease in excess of 200 psi occurs following termination of HHSI flow, manually reinitiate safety injection to maintain system pressure. Control system pressure to the nominal value which occurred when HHSI was initially terminated.

Go to General Emergency Procedure (EOP-0) to reevaluate the event.

SPECIFIC RECOMMENDATIONS AND COMMENTS FOR E-3

I. If the level in the pressurizer is restored by operation of the SI system to the normal operating range of level, the operator should expect to see a rising RCS pressure.

II. In the emergency procedure for steam generator tube rupture, where the SI flow is being modulated to maintain normal indicated pressurizer water level, the operator should expect to see falling pressure and level. If this does not occur, PRT temperature and pressure and the containment radiation, pressure and humidity monitors should be checked to determine that fluid discharge is not occurring from the pressurizer.

III. In the procedure for recovery from a steam generator tube rupture accident, the step which guides the operator to establish normal charging and letdown flows, it is recommended that the procedure be modified as follows:

A. Verify that all pressurizer PORV's are closed, and

B. Confirm "OPEN" status and operability of pressurizer PORV isolation valves.

C. THEN:

(1) Stop HHSI pumps*.

*NOTE: If normal charging and letdown cannot maintain RCS in equilibrium with the secondary side with pressurizer level in the normal indicated range, manually reinitiate HHSI and maintain equilibrium.

SPECIFIC RECOMMENDATIONS AND COMMENTS FOR E-3

(Continued)

(2) Place all non-operating SI pumps in standby mode, and maintain operable SI flowpaths. (Do not lock valves.)

(3) Reestablish normal makeup and letdown to maintain pressurizer level in the normal operating range and to maintain system pressure at values reached when HHSI is terminated.

(4) Reestablish operation of pressurizer heaters. When system pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer level control only.

ATTACHMENT II

WESTINGHOUSE

SPECIFIC RECOMMENDATIONS FOR PROCEDURES E-1, E-2 & E-3

I. The step in existing procedures relating to reactor coolant pump trip should be revised to contain the following instructions:

(1) Stop all RC Pumps after the high head safety injection pump operation has been verified and the wide range RCS pressure is decreasing and below 1,550 psig.

NOTES:

A. If component cooling to the RC pumps is isolated on a containment pressure signal, all RC pumps are to be stopped within 5 minutes because of loss of motor bearing cooling.

B. For plants with positive displacement charging pumps, the operator should establish RCP seal injection as soon as emergency equipment has been loaded on the safeguards busses.

ATTACHMENT III

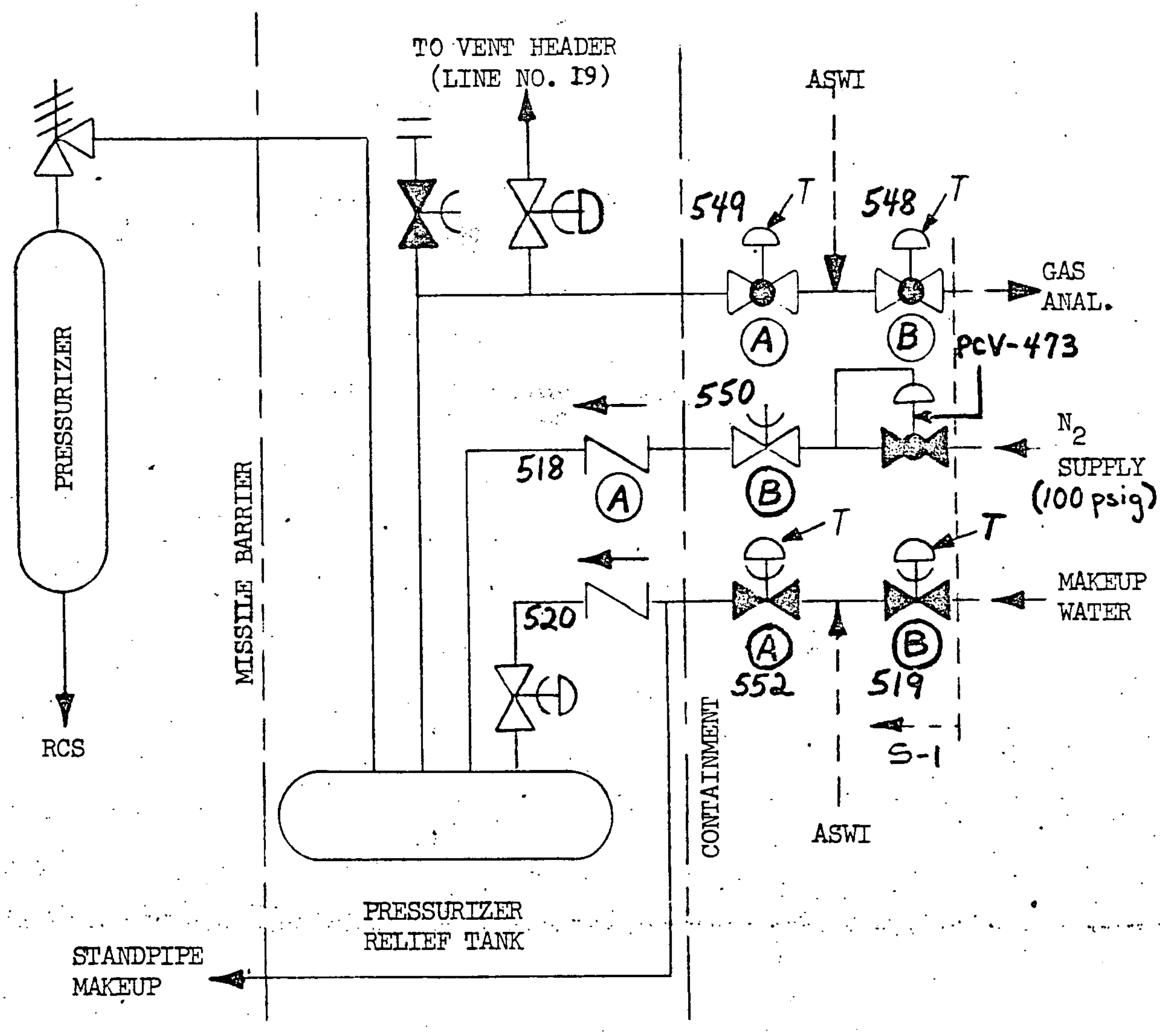
Indian Point Unit No. 2

Revised FSAR Figures Reflecting Results of Containment Isolation Valve Upgrade Study (4/79)

LINE NO. 1 PRESSURIZER RELIEF TANK TO GAS ANALYZER

LINE NO. 2 PRESSURIZER RELIEF TANK N_2 SUPPLY

LINE NO. 3 PRESSURIZER RELIEF TANK MAKEUP



ALTHOUGH THE PRESSURIZER RELIEF TANK IS MISSILE PROTECTED, THESE PENETRATING LINES CAN BECOME EXPOSED TO CONTAINMENT ATMOSPHERE IF THE PRESSURIZER DISCHARGE HEADER IS BREACHED DURING THE ACCIDENT.

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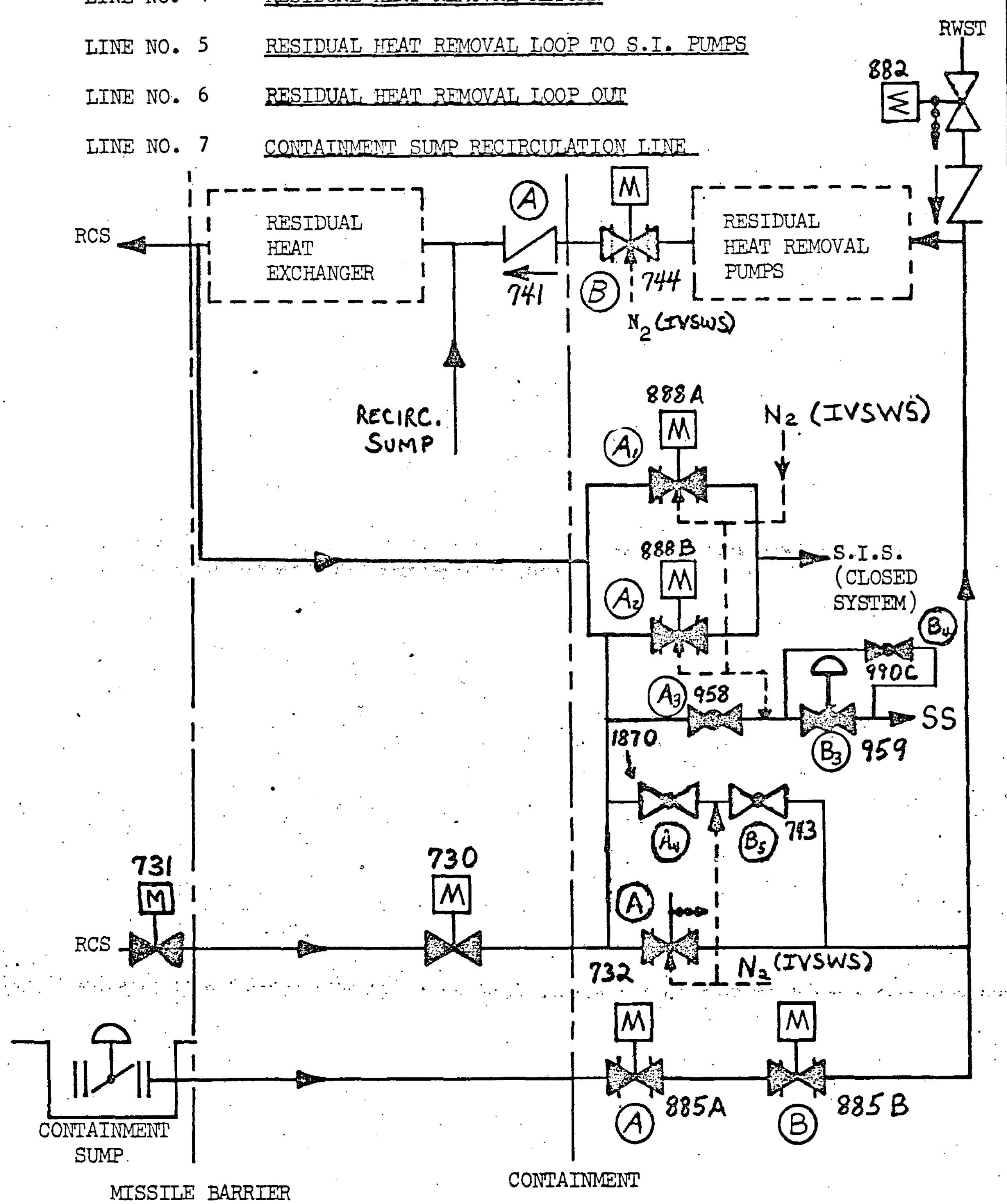
Figure 5.2-1

LINE NO. 4 RESIDUAL HEAT REMOVAL RETURN

LINE NO. 5 RESIDUAL HEAT REMOVAL LOOP TO S.I. PUMPS

LINE NO. 6 RESIDUAL HEAT REMOVAL LOOP OUT

LINE NO. 7 CONTAINMENT SUMP RECIRCULATION LINE



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N_2 - MANUAL N_2 PRESSURIZATION

SS - SAMPLING SYSTEM

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-2

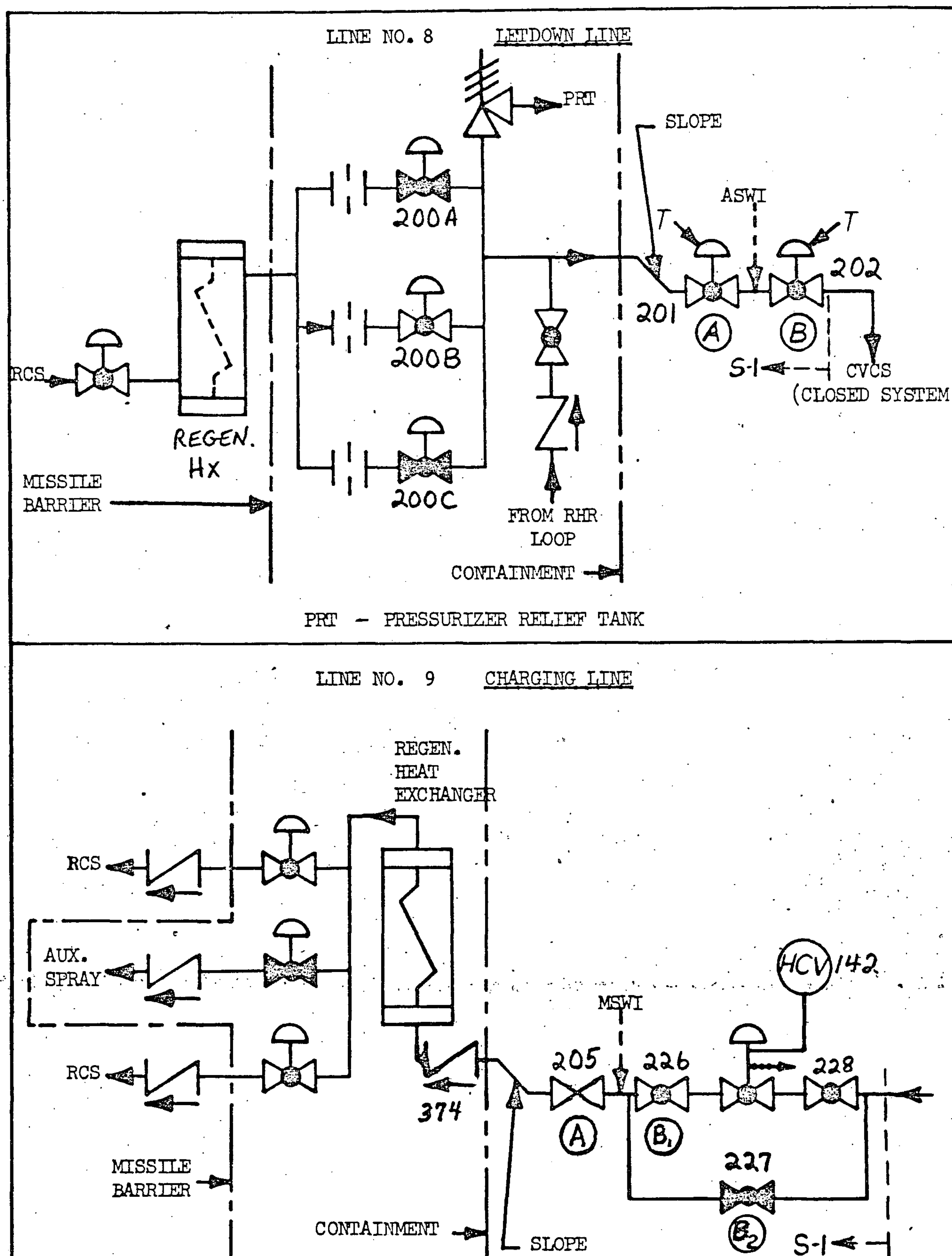

FIGURE 5.2-3

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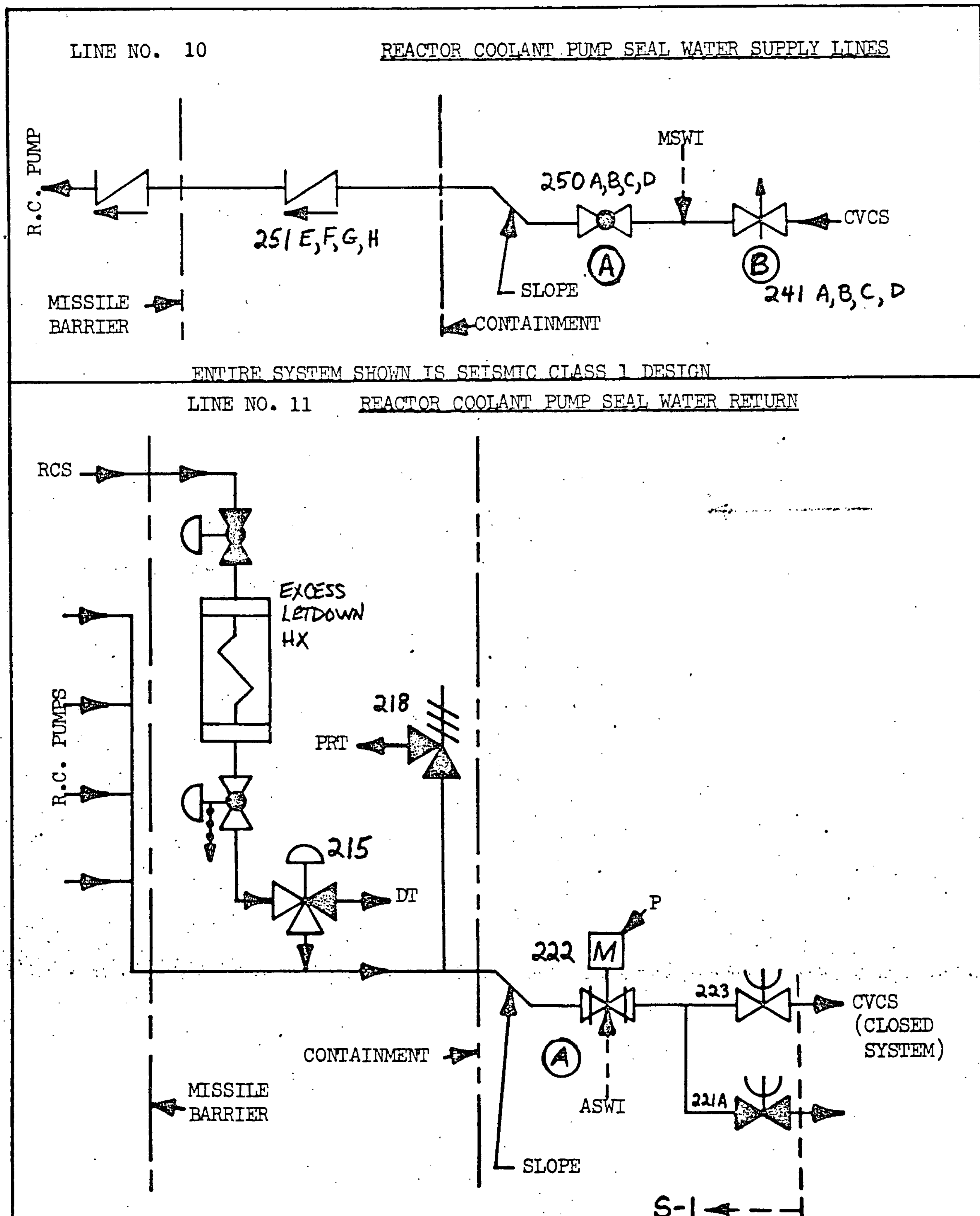

DT - REACTOR COOLANT DRAIN TANK

P - TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL PHASE B

PRT - PRESSURIZER RELIEF TANK

FIGURE 5.2-4

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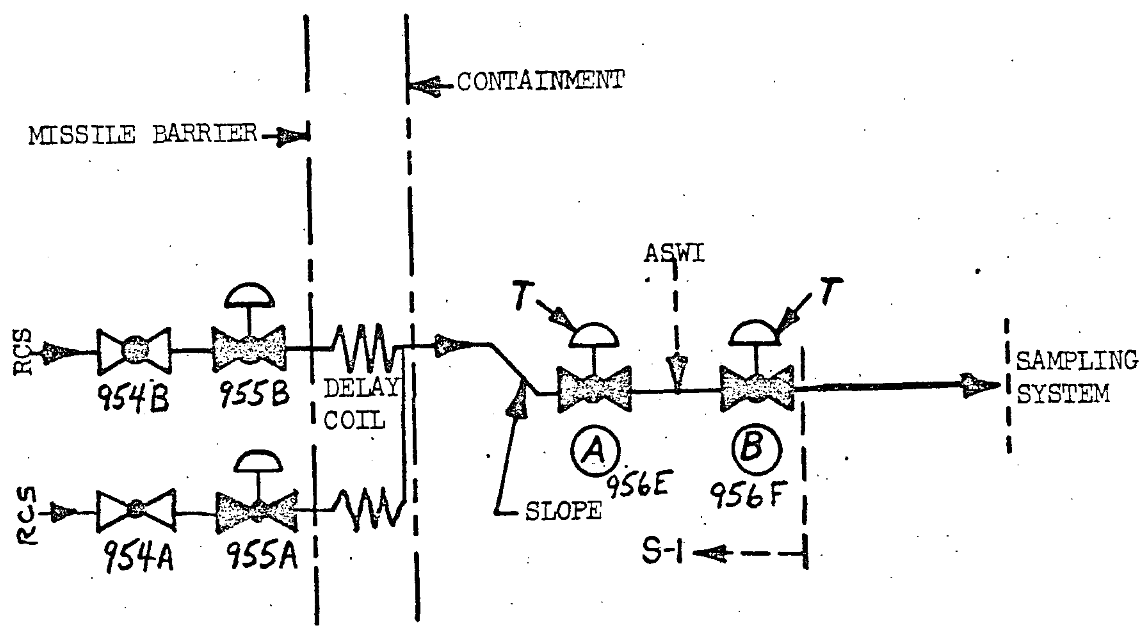
LINE NO. 12

REACTOR COOLANT SYSTEM SAMPLE LINES



LINE NO. 13

FUEL TRANSFER TUBE

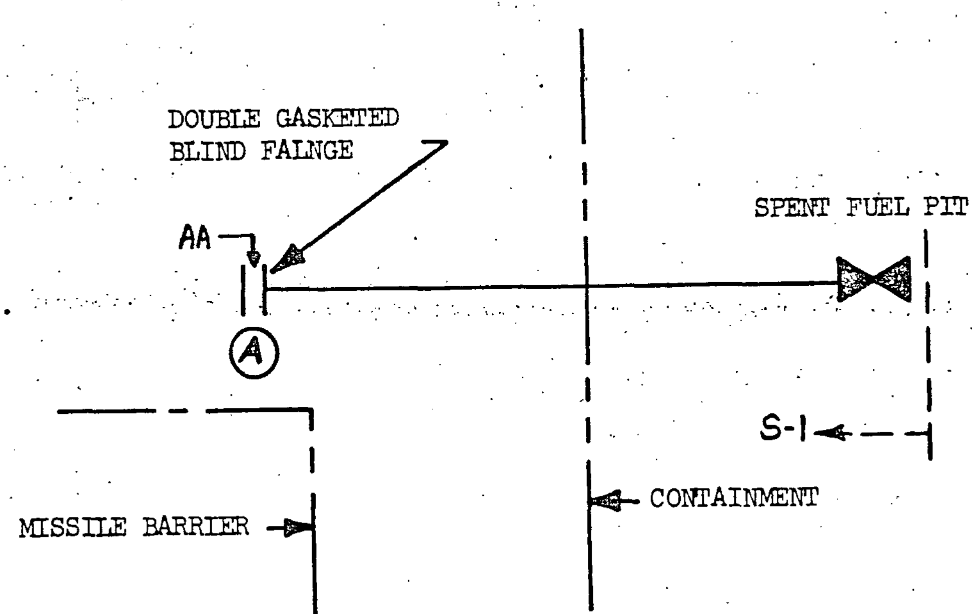


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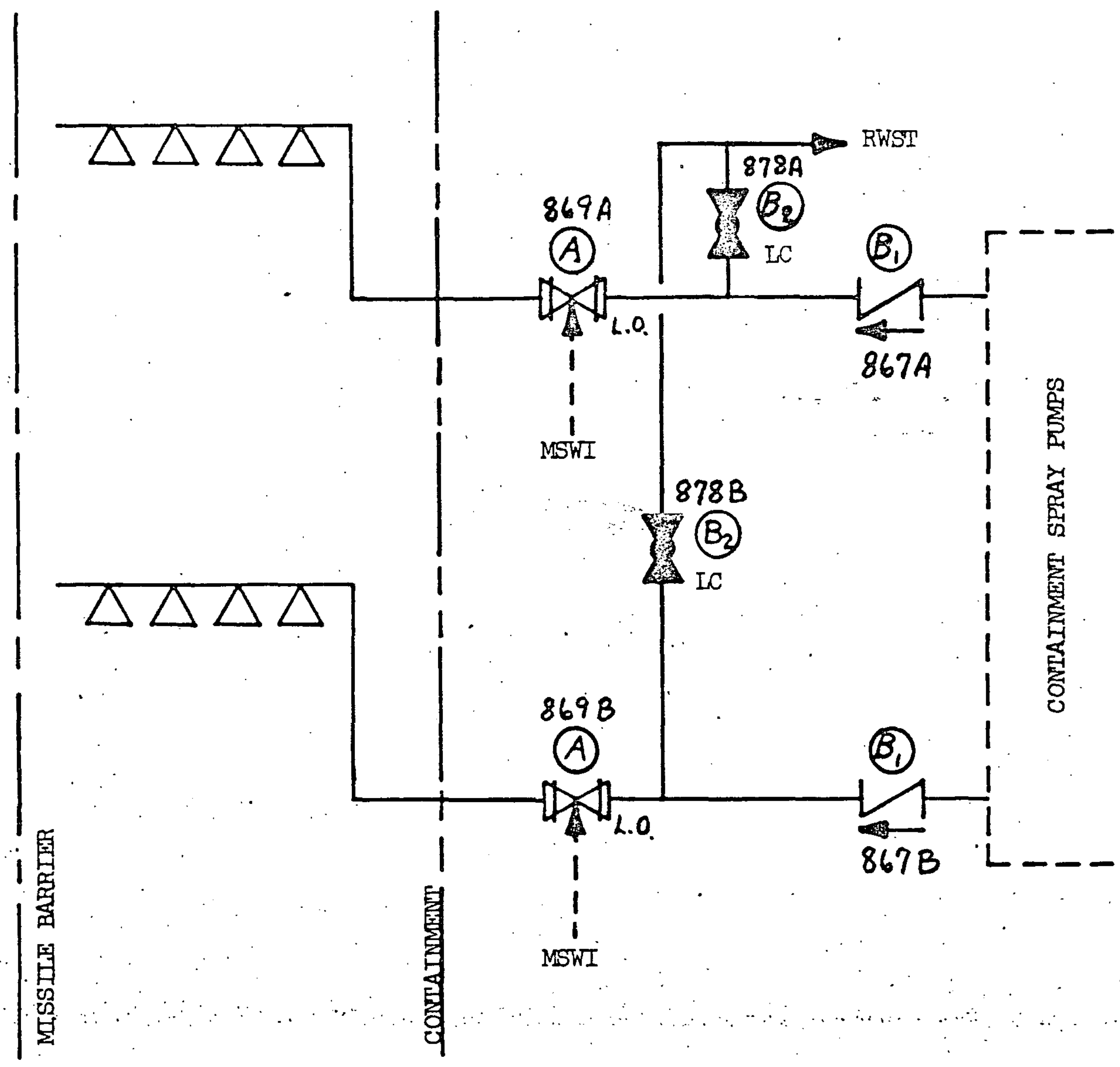
FIGURE 5.2-5

LINE NO. 14 CONTAINMENT SPRAY HEADERS



RWST - REFUELING WATER STORAGE TANK

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

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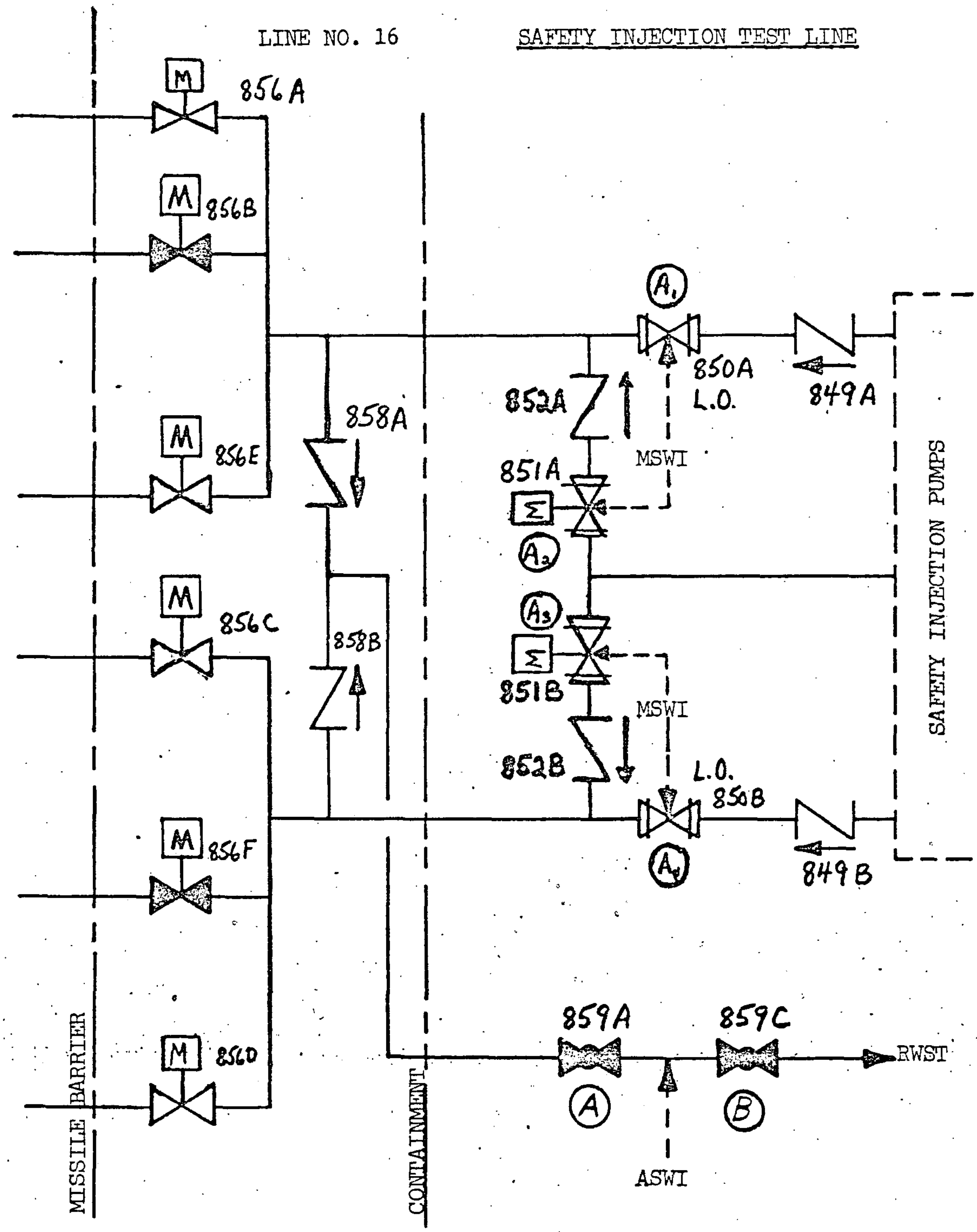
FIGURE 5.2-6

LINE NO. 15 SAFETY INJECTION HEADERS

LINE NO. 16 SAFETY INJECTION TEST LINE



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

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RWST - REFUELING WATER STORAGE TANK

FIGURE 5.2-7

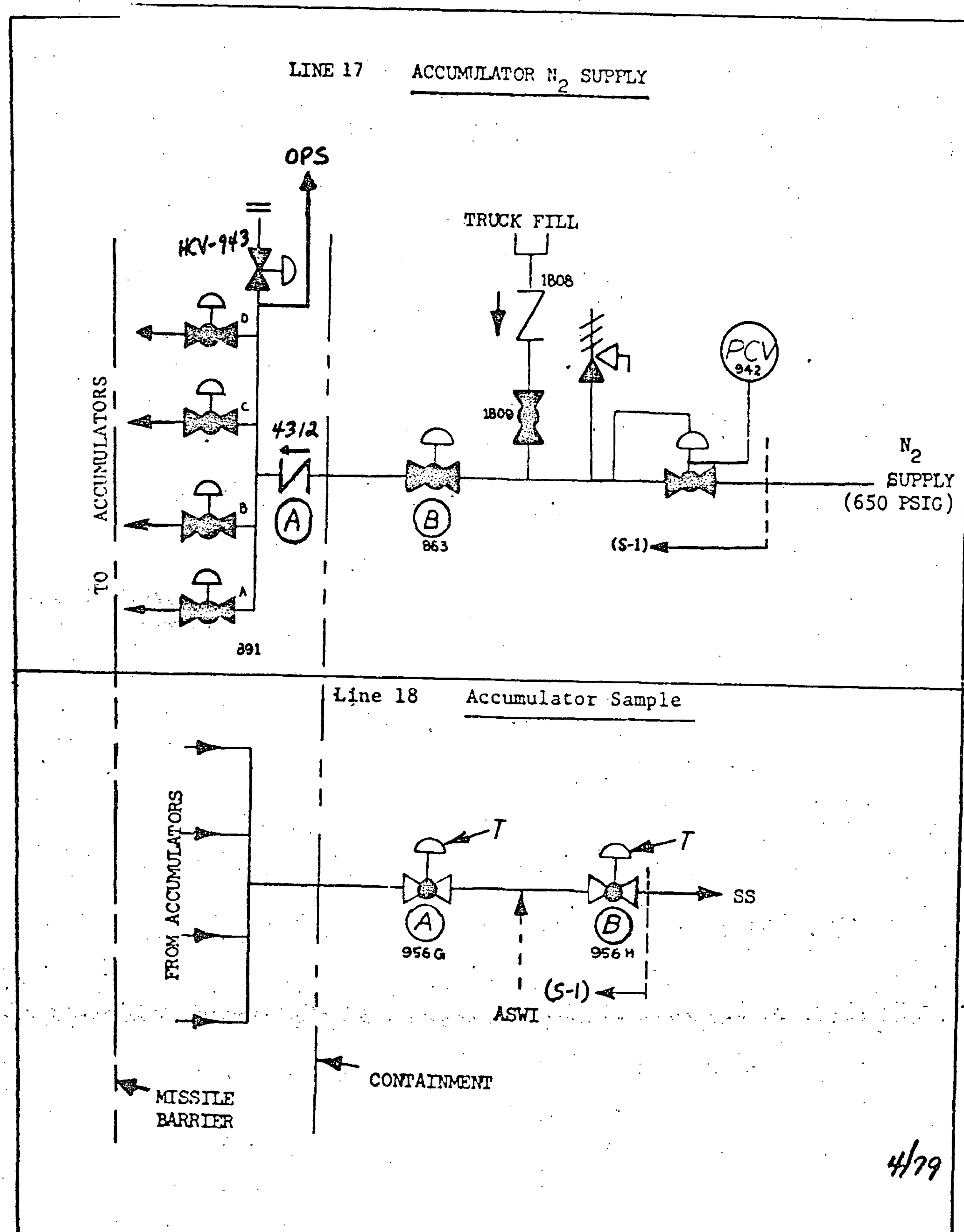
LINE 17
ACCUMULATOR N_2 SUPPLY
OPS
HCV-943
TRUCK FILL
1808
1809
PCV
942
D
C
B
A
TO ACCUMULATORS
43/12
A
B
863
N_2
SUPPLY
(650 PSIG)
(S-1)
891
Line 18
Accumulator Sample
FROM ACCUMULATORS
T
T
A
956G
B
956H
SS
(S-1)
ASWI
MISSILE
BARRIER
CONTAINMENT
4/79

FIGURE 5.2-8

LINE NO. 19 PRIMARY SYSTEM VENT HEADER AND N_2 SUPPLY LINE

LINE NO. 20 REACTOR COOLANT DRAIN TANK TO GAS ANALYZER

LINE NO. 21 RCDT PUMP DISCHARGE

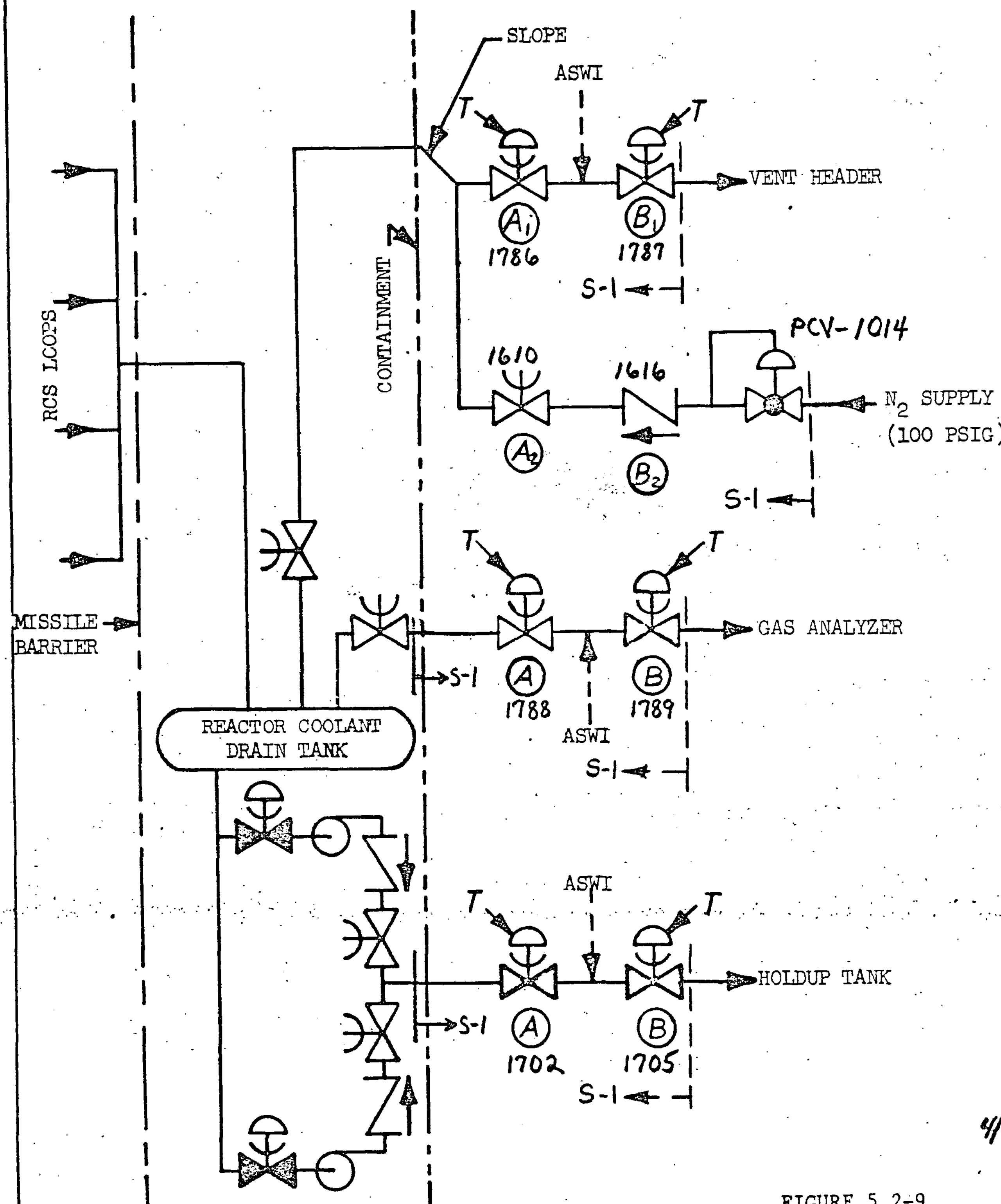

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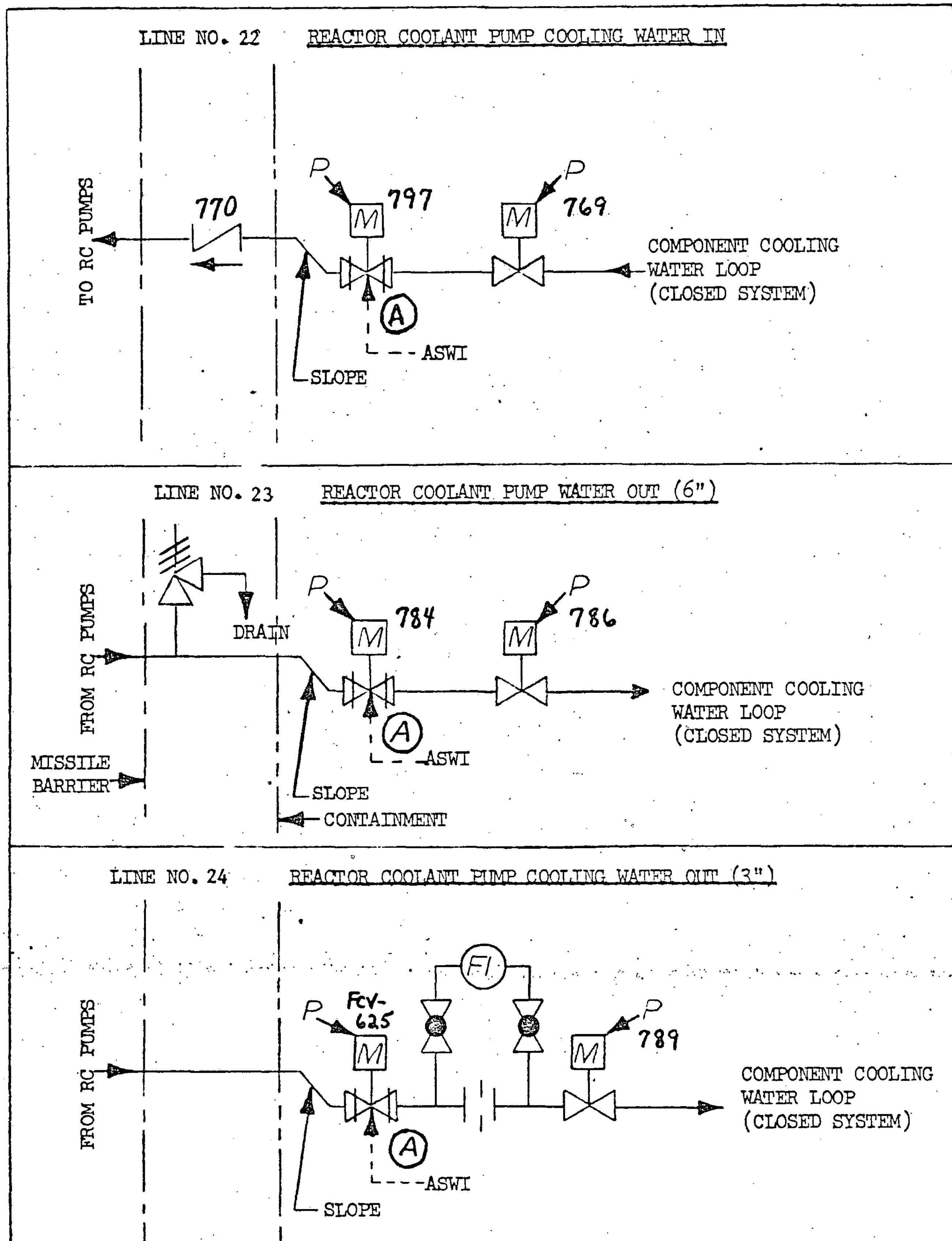
FIGURE 5.2-9



ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-10

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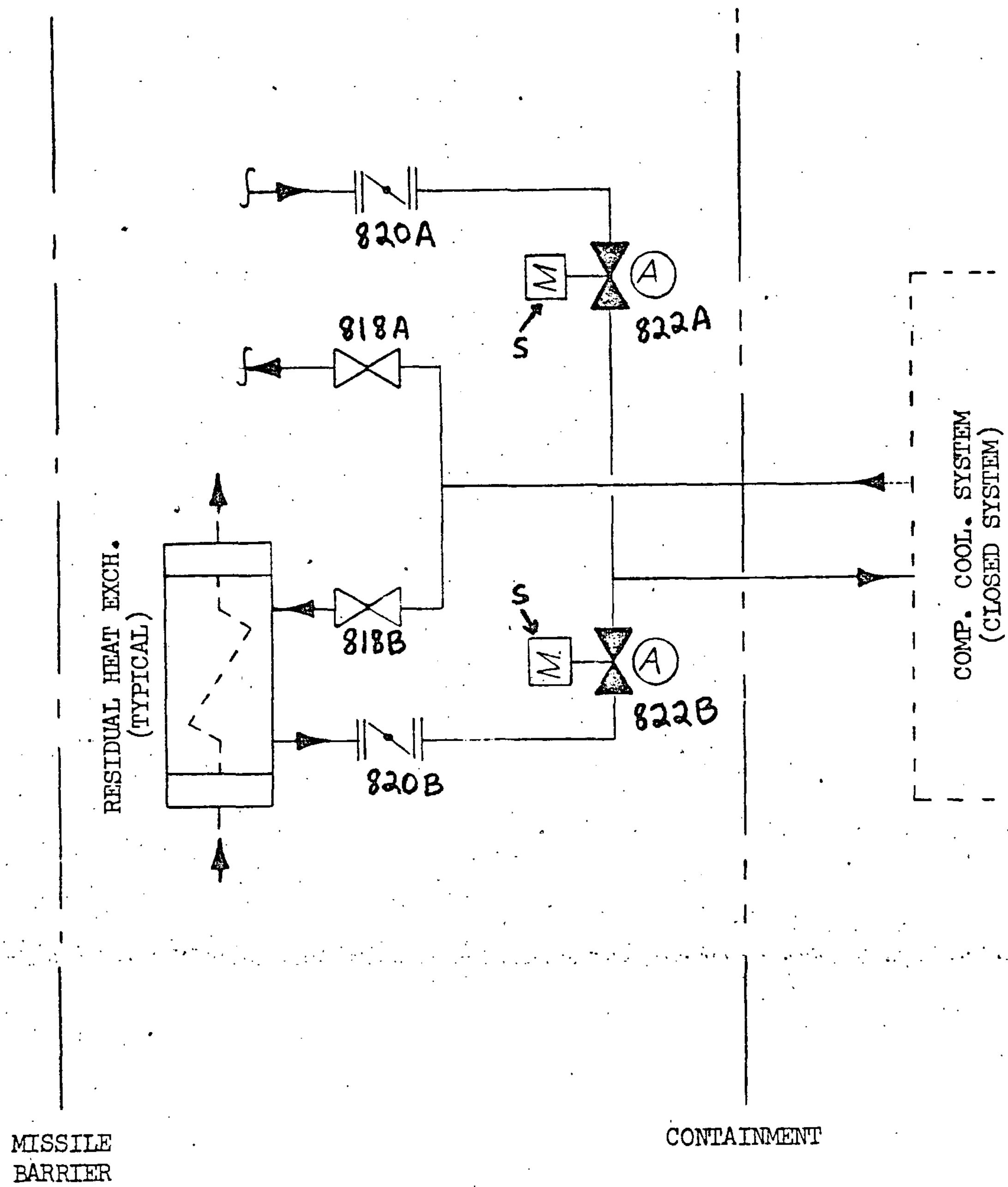
LINE NO. 25 RESIDUAL HEAT EXCHANGER COOLING WATER IN

LINE NO. 26 RESIDUAL HEAT EXCHANGER COOLING WATER RETURN

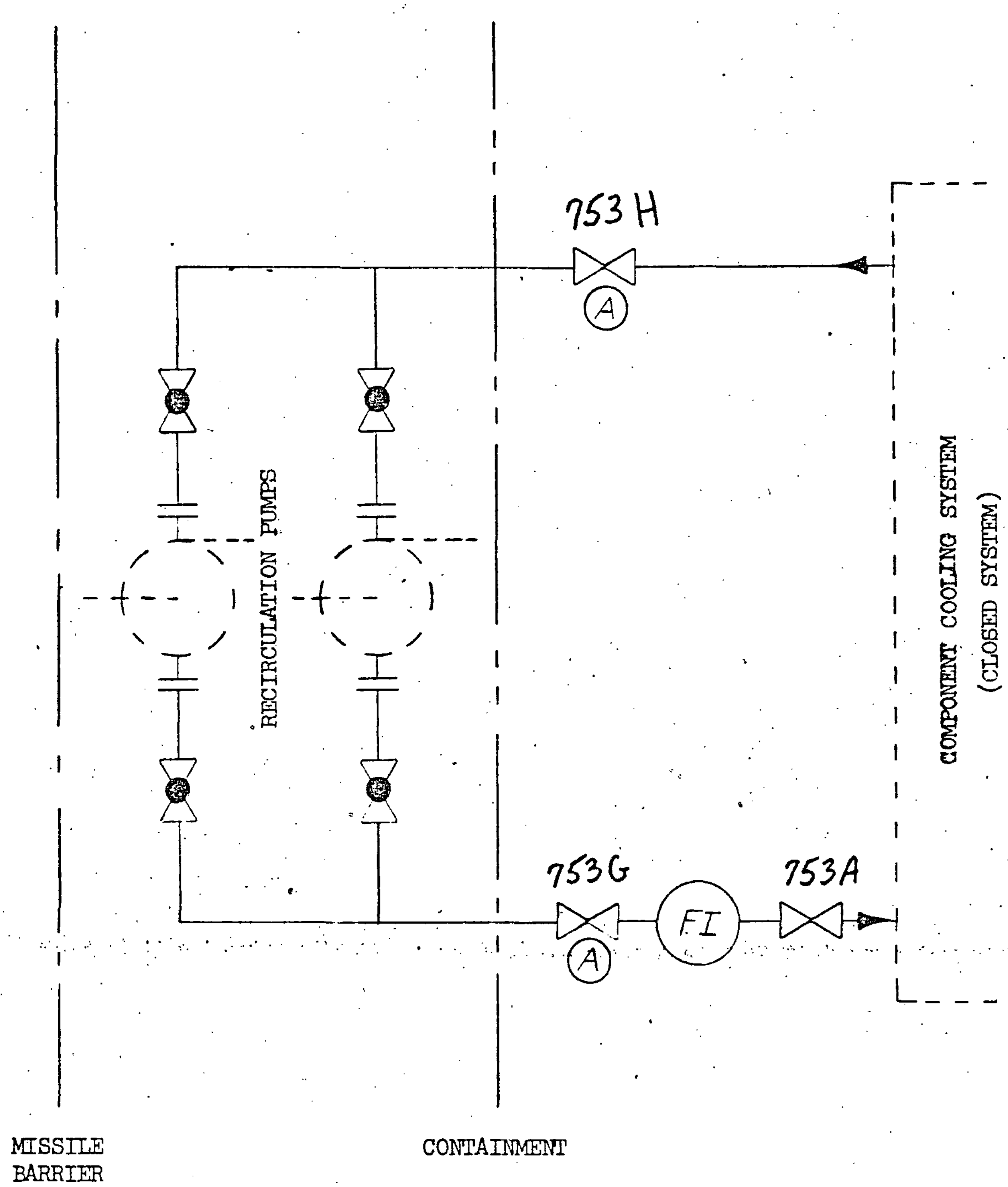


ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-11

LINE NO. 27 RECIRCULATION PUMP COOLING WATER SUPPLY

LINE NO. 28 RECIRCULATION PUMP COOLING WATER RETURN



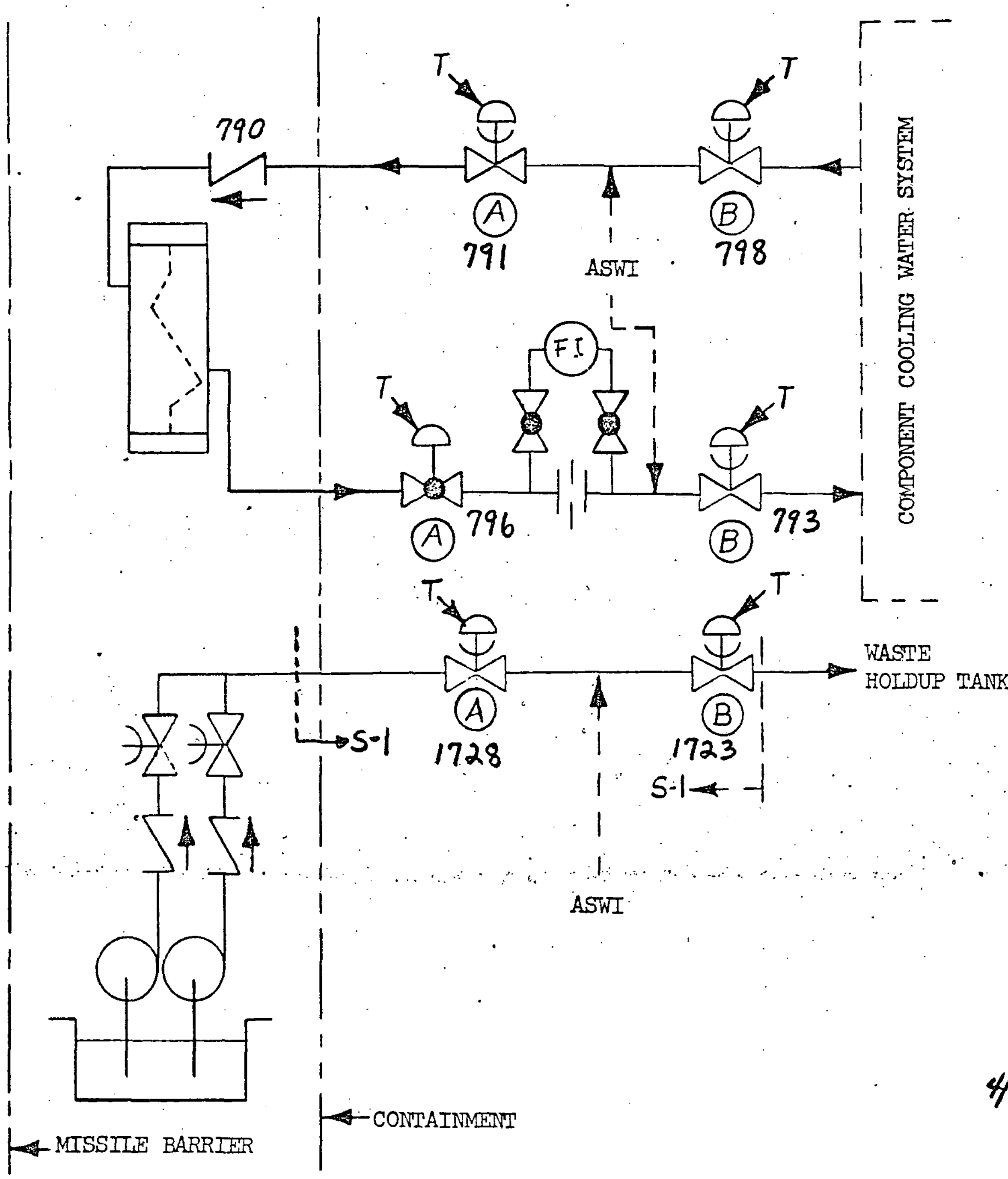
ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-12

LINE NO. 29 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN

LINE NO. 30 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT

LINE NO. 31 CONTAINMENT SUMP PUMP DISCHARGE



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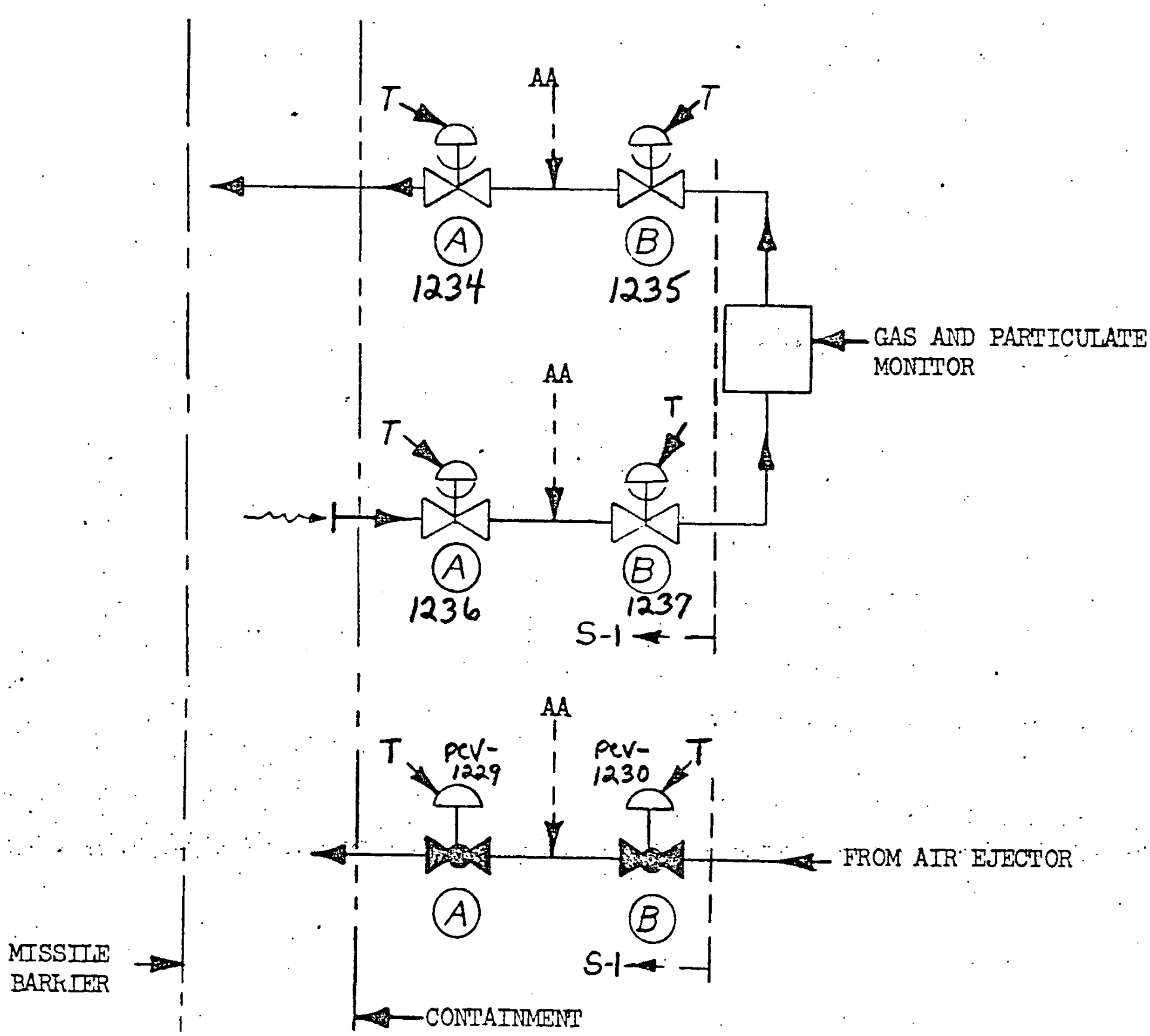
ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN FIGURE 5.2-13

LINE NO. 32 CONTAINMENT AIR SAMPLE IN

LINE NO. 33 CONTAINMENT AIR SAMPLE OUT

LINE NO. 34 AIR EJECTOR DISCHARGE TO CONTAINMENT



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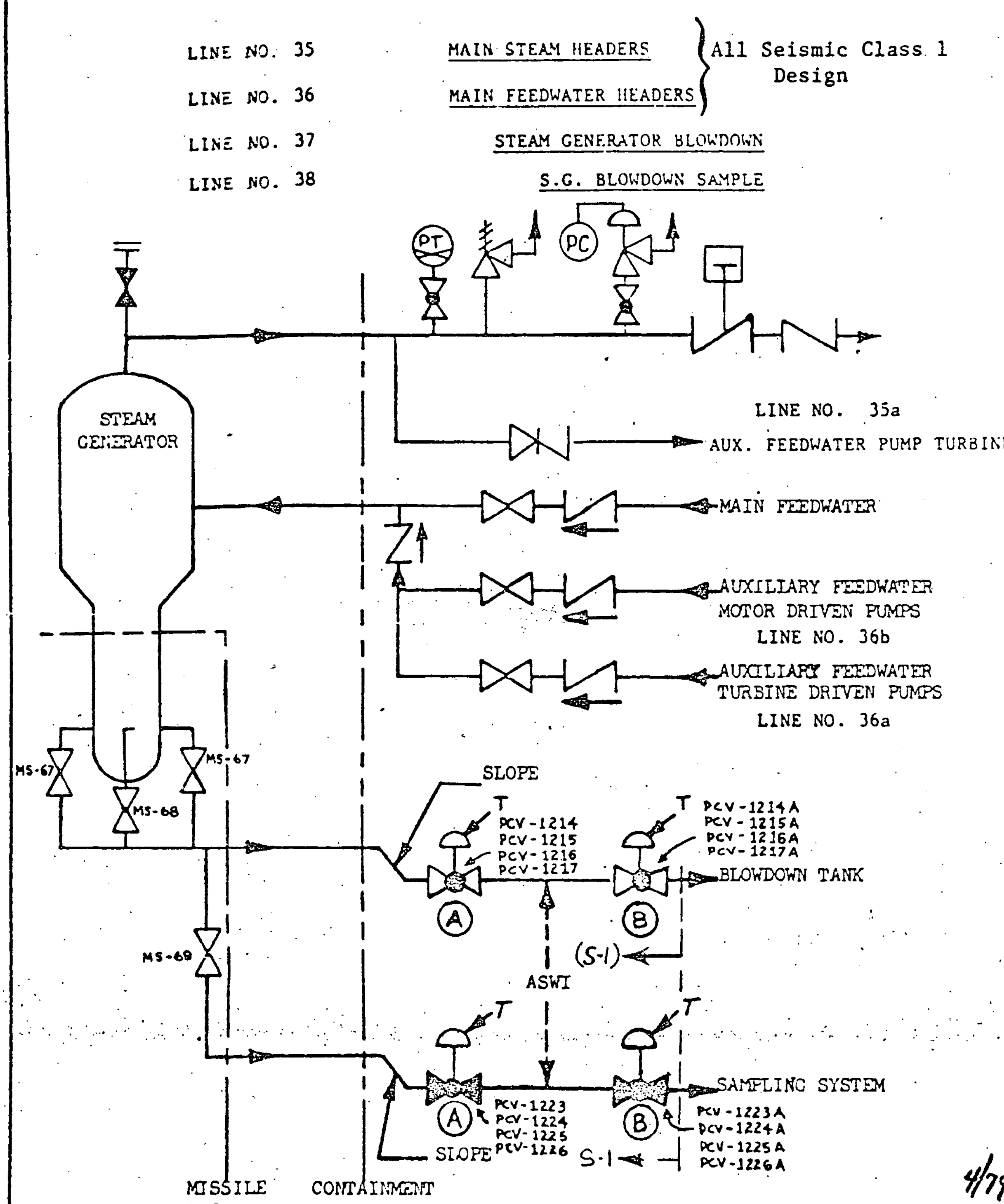
AA - AUTOMATIC PRESSURIZATION WITH AIR FROM PENETRATION PRESSURIZATION SYSTEM

FIGURE 5.2-14



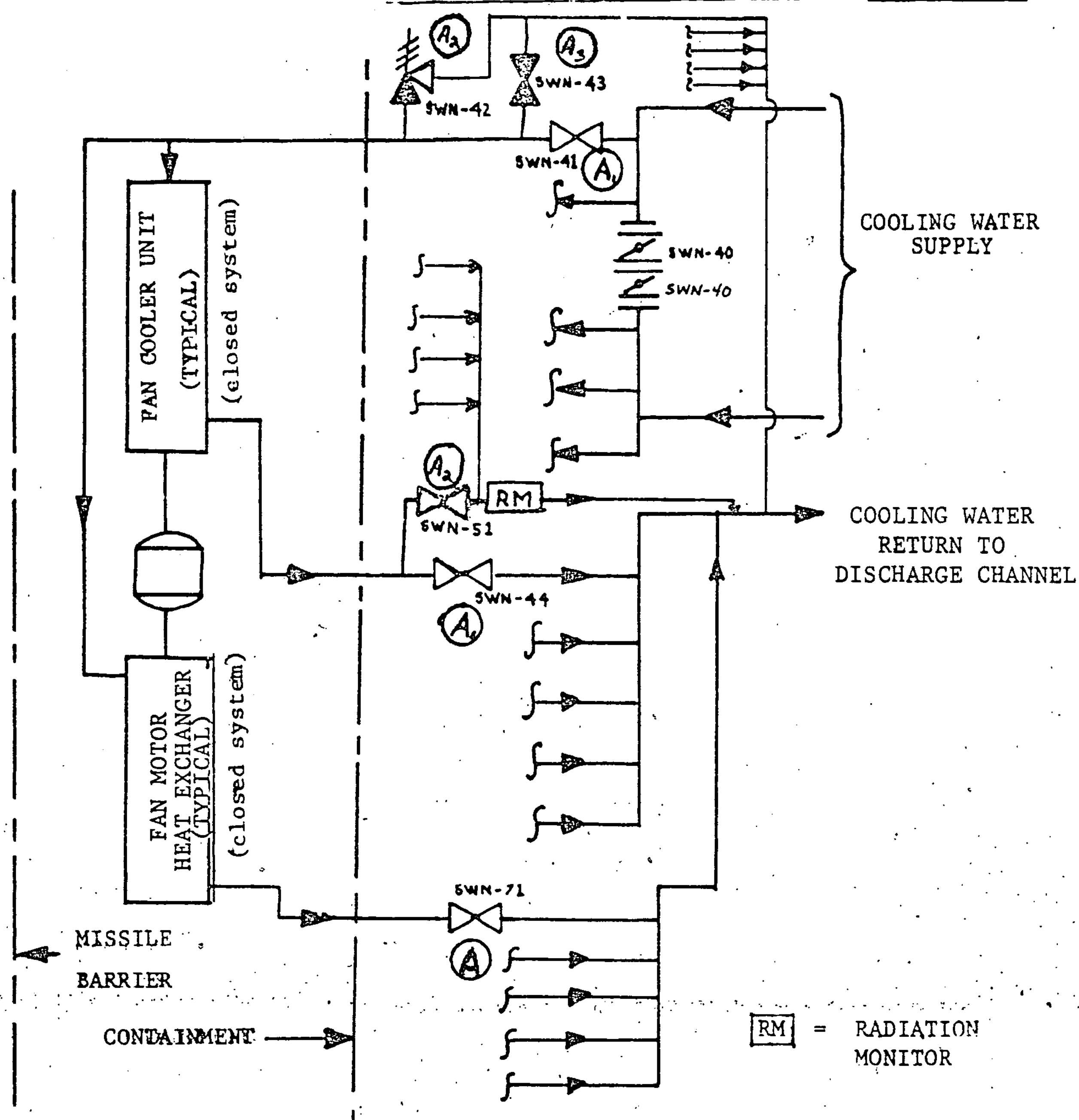
LINE NO. 35
MAIN STEAM HEADERS
LINE NO. 36
MAIN FEEDWATER HEADERS
All Seismic Class 1 Design
LINE NO. 37
STEAM GENERATOR BLOWDOWN
LINE NO. 38
S.G. BLOWDOWN SAMPLE
PT
PC
STEAM GENERATOR
LINE NO. 35a
AUX. FEEDWATER PUMP TURBINE
MAIN FEEDWATER
AUXILIARY FEEDWATER MOTOR DRIVEN PUMPS
LINE NO. 36b
AUXILIARY FEEDWATER TURBINE DRIVEN PUMPS
LINE NO. 36a
MS-67
MS-67
MS-68
SLOPE
PCV-1214
PCV-1215
PCV-1216
PCV-1217
PCV-1214A
PCV-1215A
PCV-1216A
PCV-1217A
BLOWDOWN TANK
A
B
(S-1)
MS-69
ASWI
SAMPLING SYSTEM
PCV-1223
PCV-1224
PCV-1225
PCV-1226
PCV-1223A
PCV-1224A
PCV-1225A
PCV-1226A
SLOPE
S-1
MISSILE BARRIER
CONTAINMENT
4/79

FIGURE 5.2-15

LINE NO. 39 10" VENTILATION SYSTEM COOLING WATER IN (5)

LINE NO. 40 10" VENTILATION SYSTEM COOLING WATER OUT (5)

LINE NO. 40a 2" VENTILATION SYSTEM MOTOR COOLING WATER OUT (5)



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

FIGURE 5.2-16

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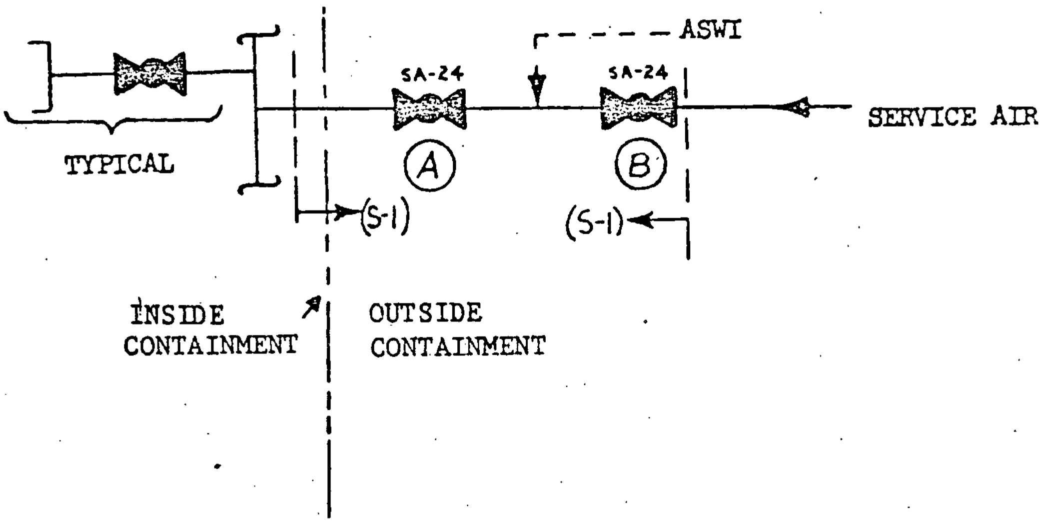
LINE NO. 41 SERVICE AIR

LINE NO. 42 [NOT ASSIGNED]



LINE NO. 43 WELD CHANNEL PRESSURIZATION AIR SUPPLY

LINE NO. 44 : DEAD WEIGHT CALIBRATOR

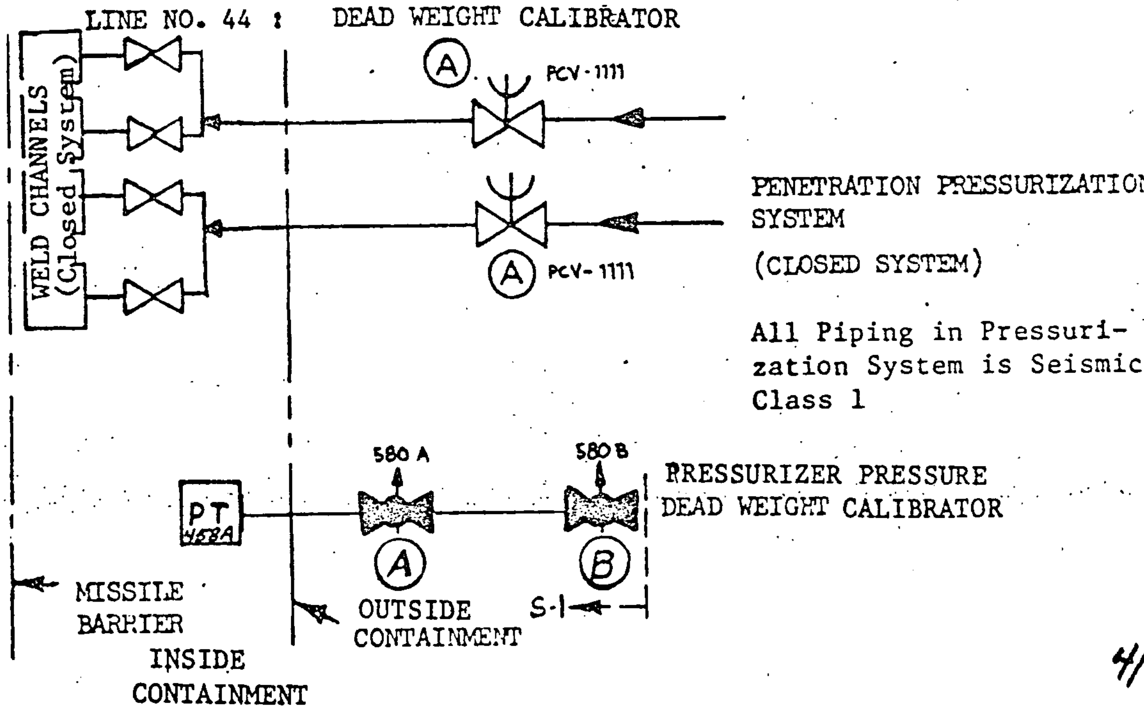


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FIGURE 5.2-17

LINE NO. 45 AUXILIARY STEAM SUPPLY

LINE NO. 46 AUXILIARY STEAM CONDENSATE RETURN

LINE NO. 47 CITY WATER

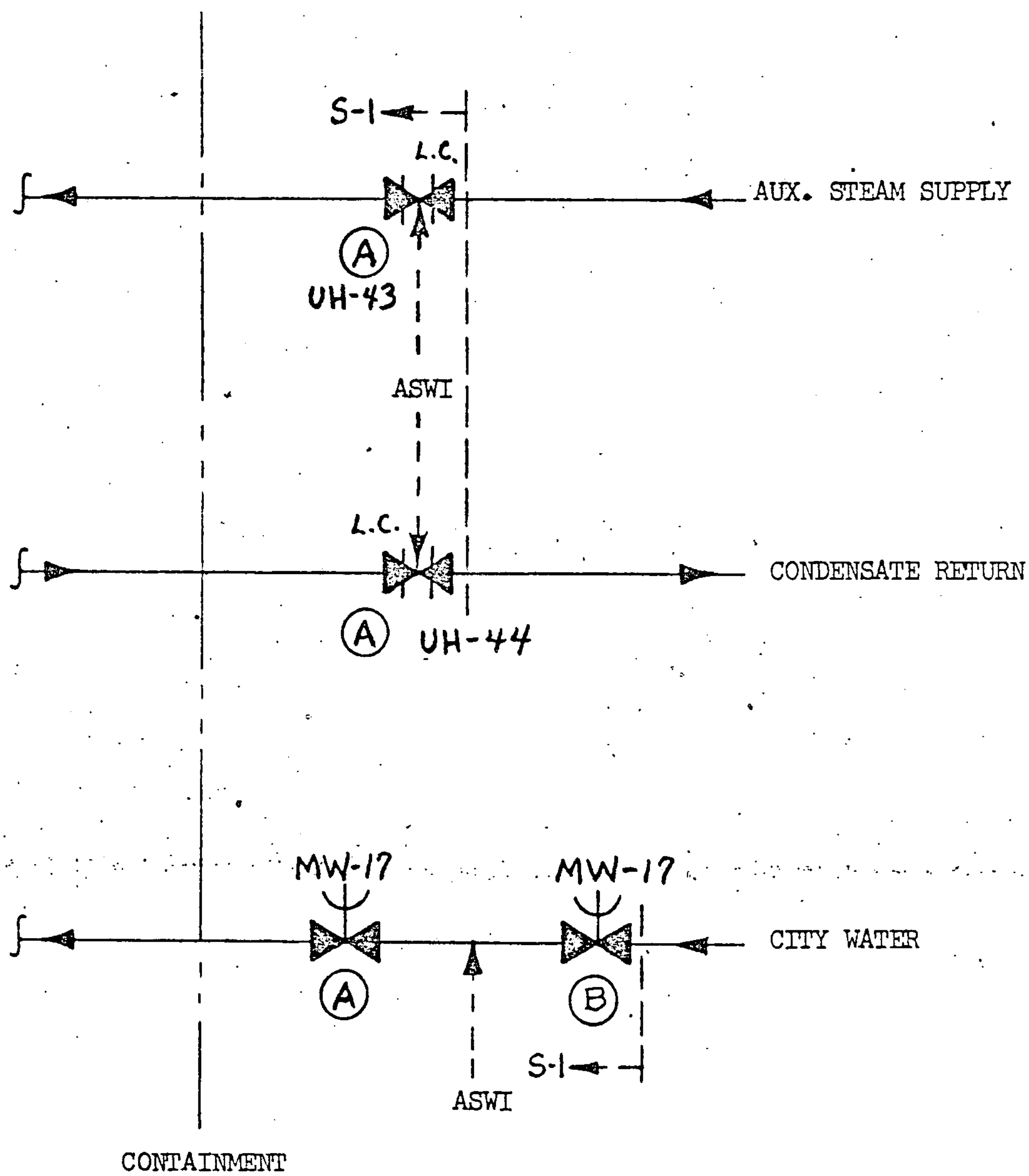

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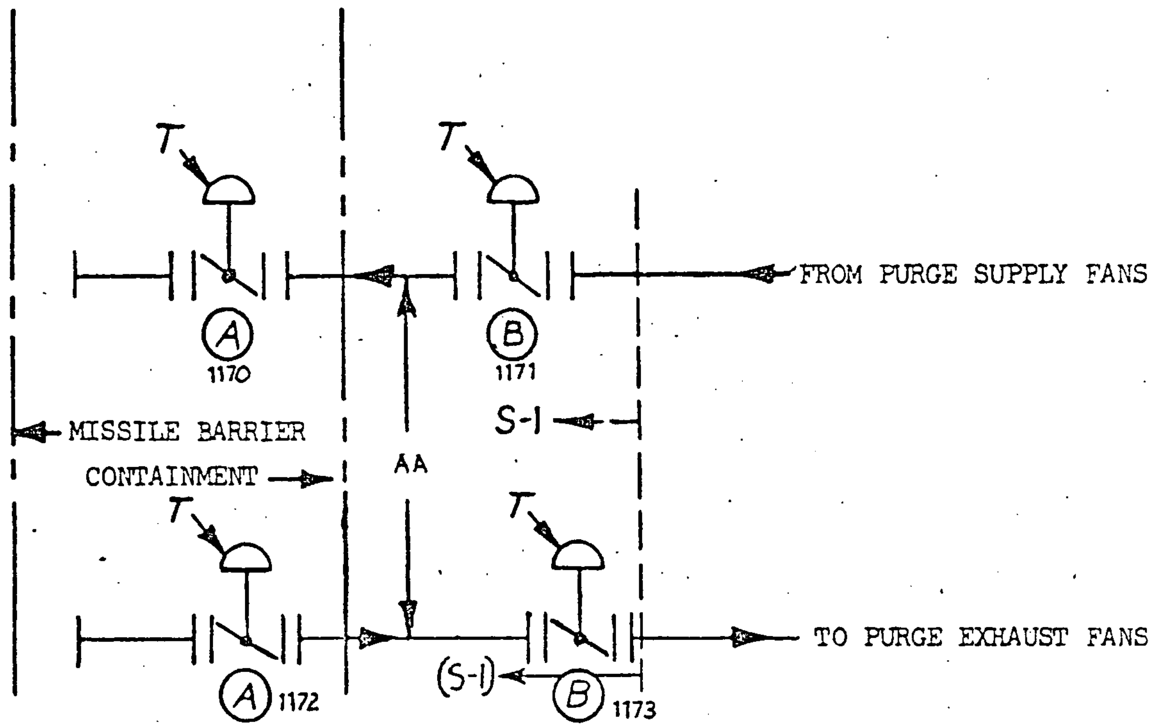
FIGURE 5.2-18

LINE NO. 48 PURGE SUPPLY DUCT

LINE NO. 49 PURGE EXHAUST DUCT



LINE NO. 50 CONTAINMENT PRESSURE RELIEF

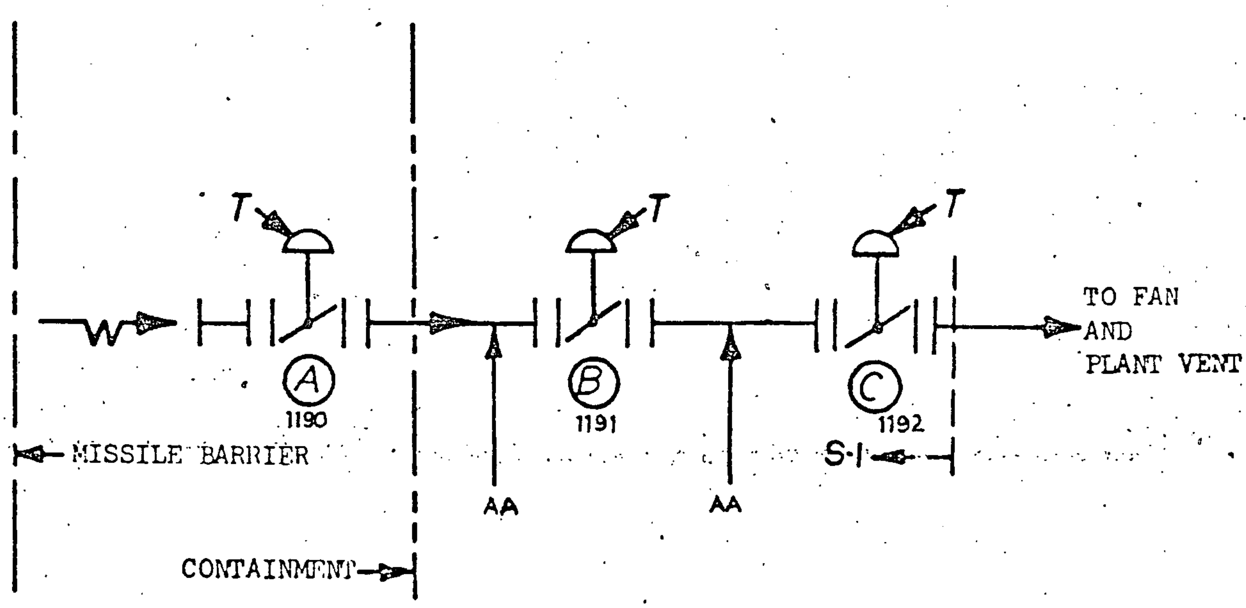


FIGURE 5.2-19

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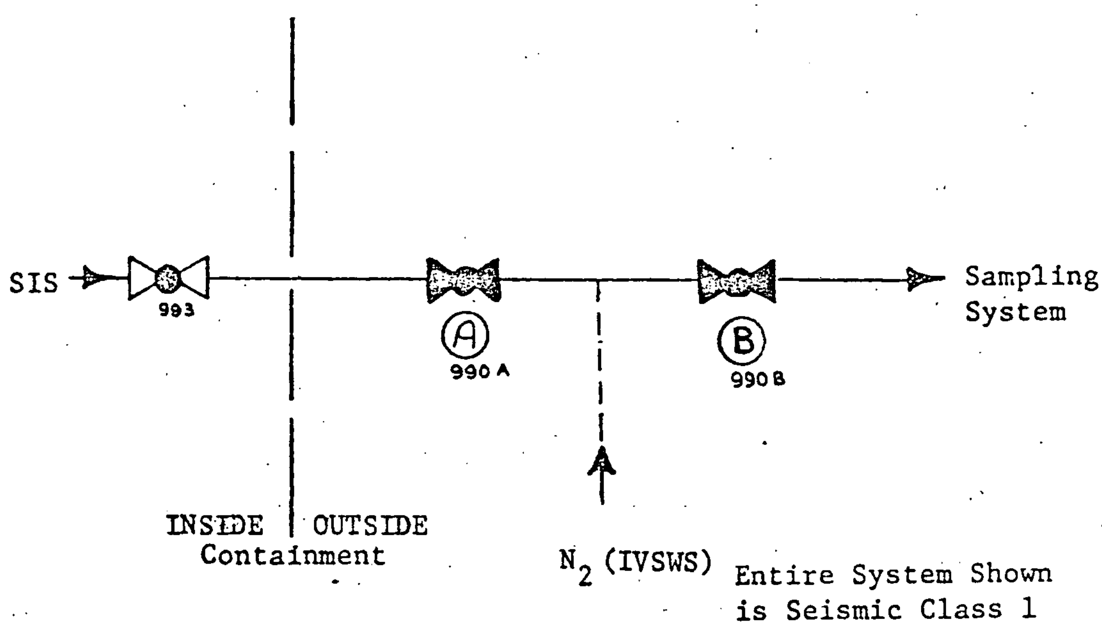
Line No. 51 RECIRCULATION PUMP DISCHARGE SAMPLE LINE



Line No. 52 PRESSURIZER STEAM SPACE SAMPLE
Line No. 53 PRESSURIZER LIQUID SPACE SAMPLE

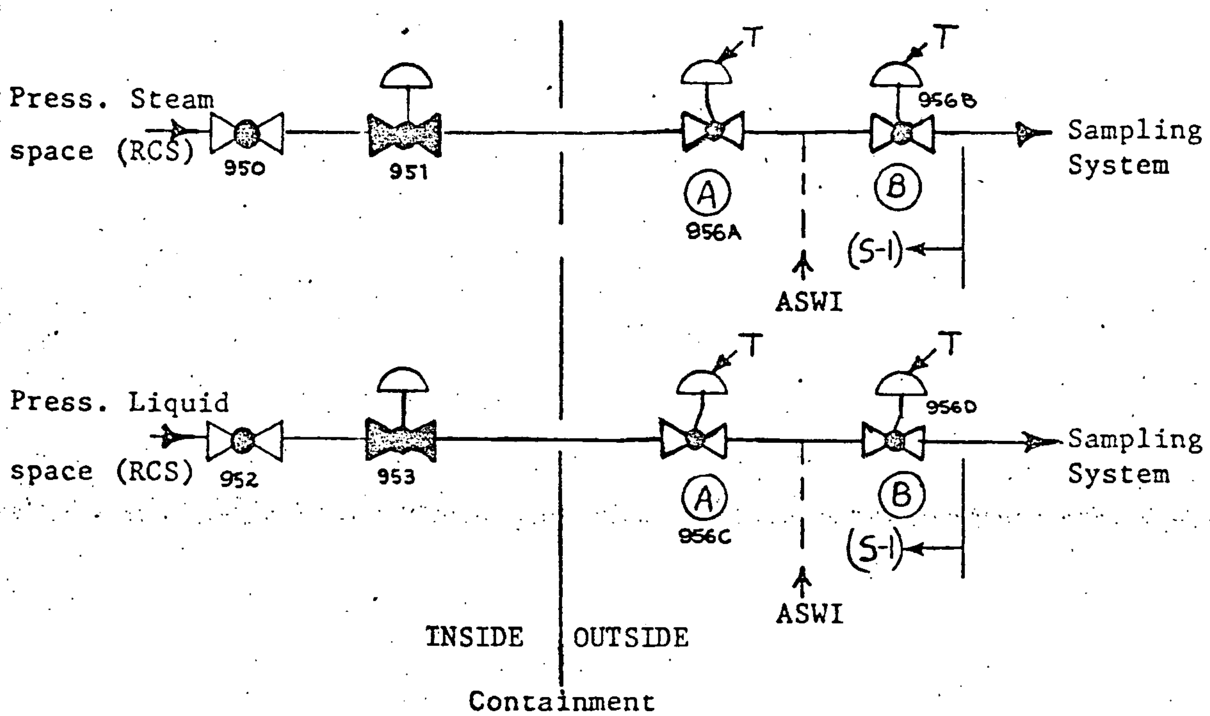


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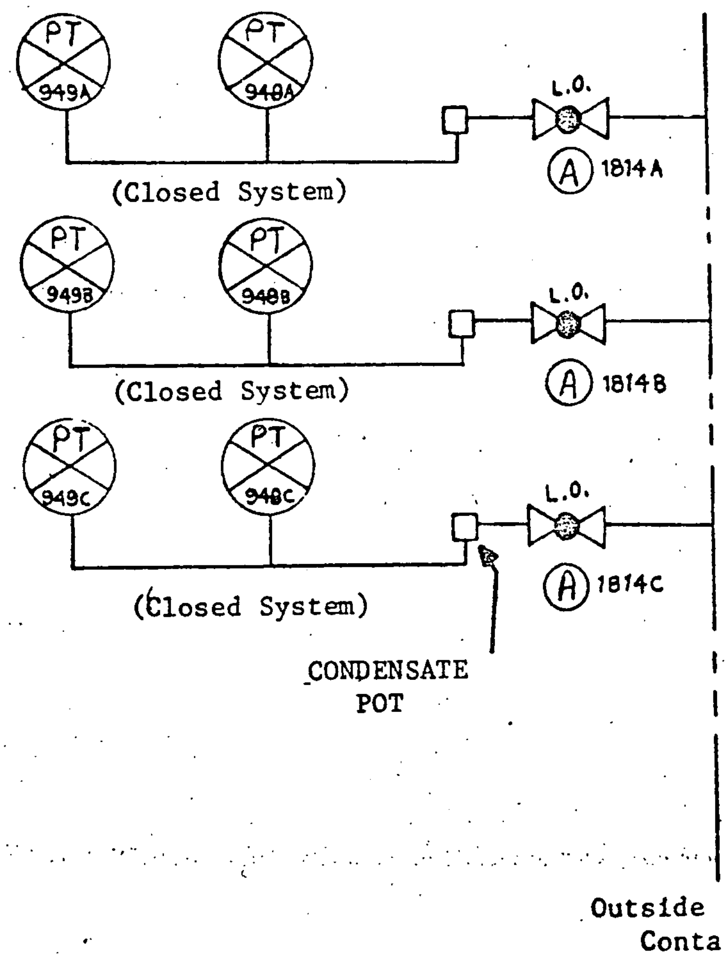
Figure 5.2-20

Line Nos. 54, 55, and 56.

CONTAINMENT PRESSURE INSTRUMENTATION LINES



Entire System Shown is
Seismic Class 1

Figure 5.2-21

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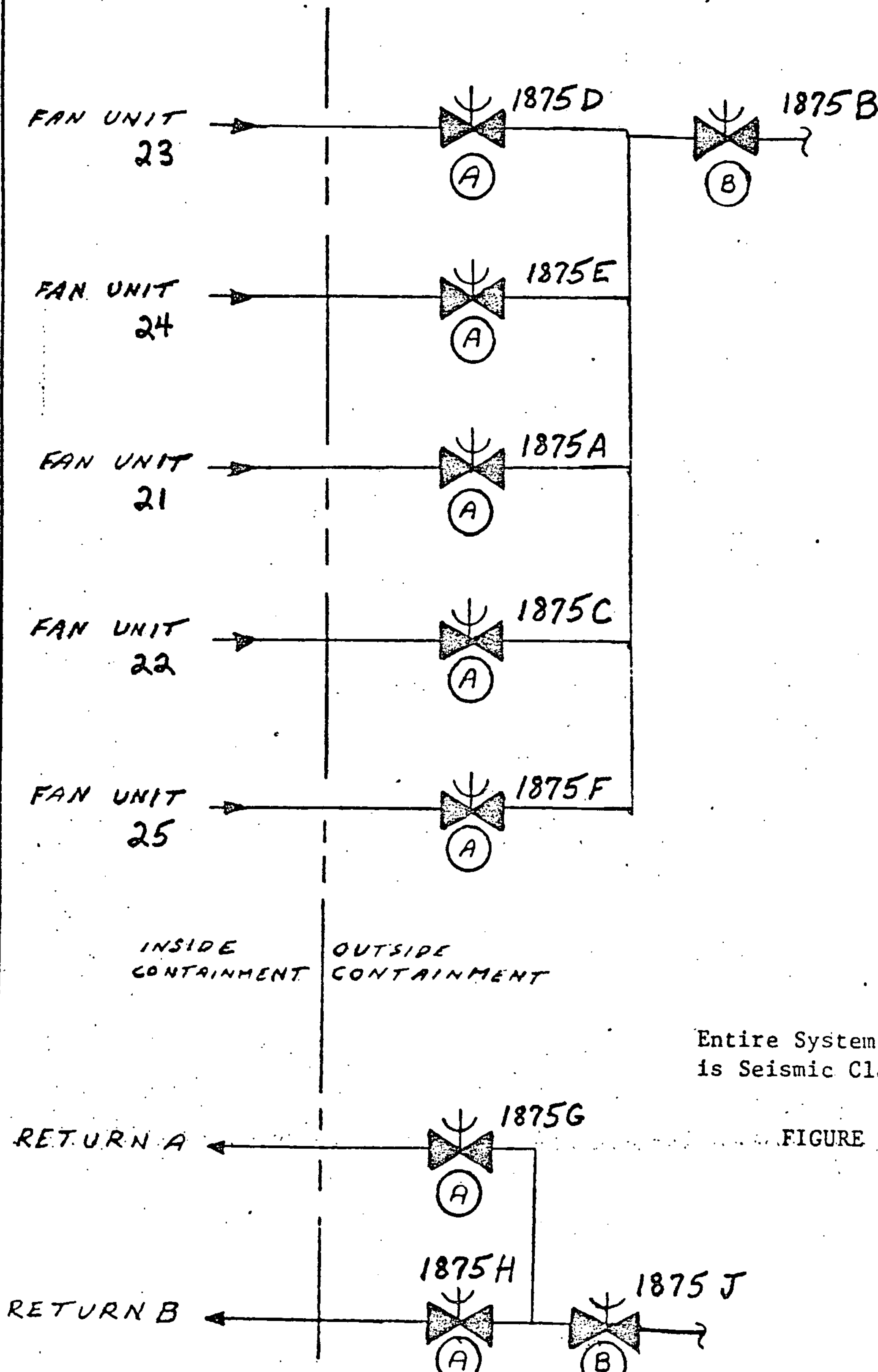
Line No. 57 Post Accident Containment Sampling Lines (Supply and Return)



Entire System Shown is Seismic Class 1

FIGURE 5.2-22

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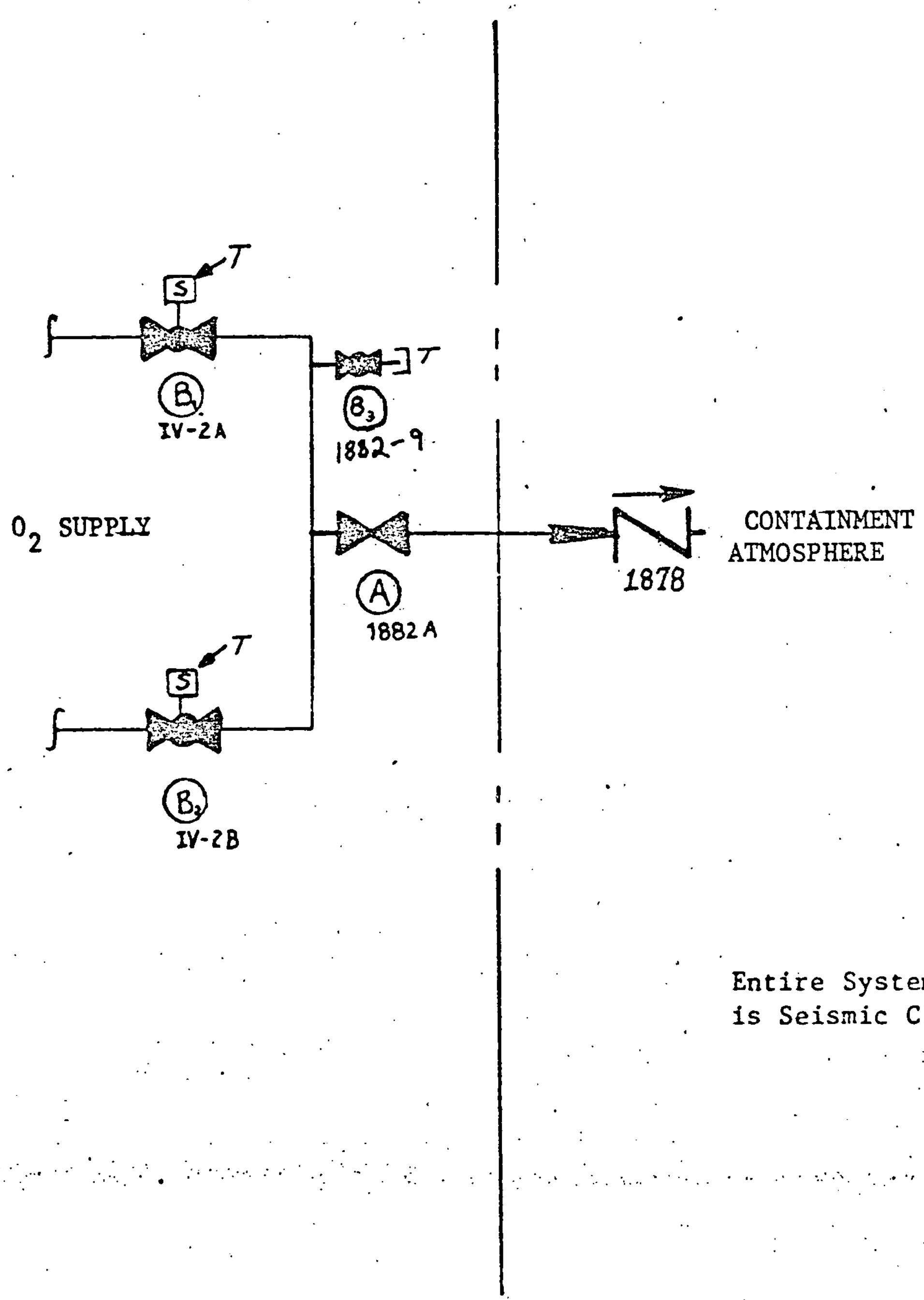
Line No. 58 Oxygen Supply To Containment



Entire System Shown
is Seismic Class 1

OUTSIDE CONTAINMENT

INSIDE CONTAINMENT

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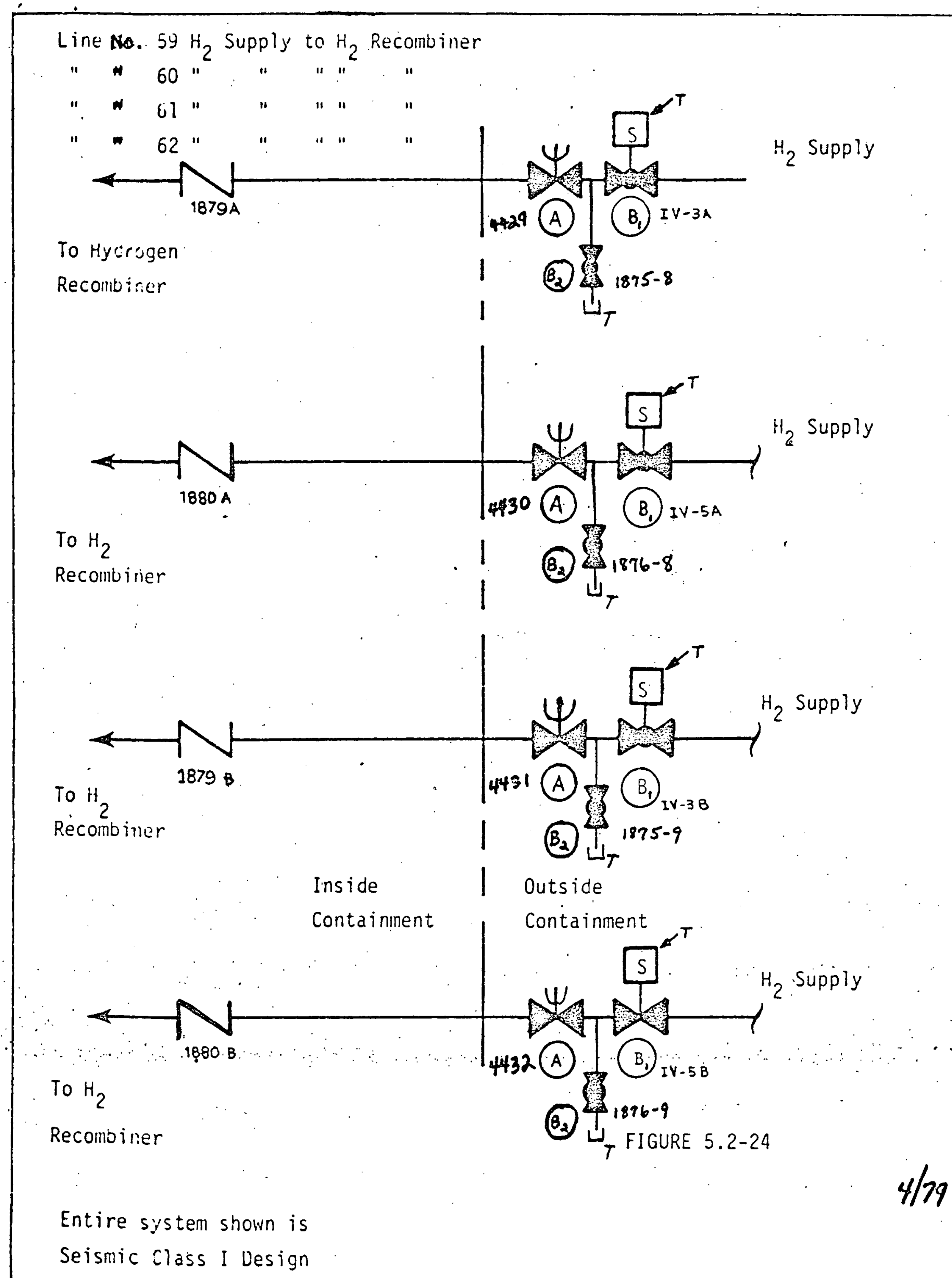
FIGURE 5.2-23


Line No. 59 H2 Supply to H2 Recombiner
" " 60 " " " " "
" " 61 " " " " "
" " 62 " " " " "
H2 Supply
1879A
4429
IV-3A
1875-8
To Hydrogen Recombiner
H2 Supply
1880 A
4430
IV-5A
1876-8
To H2 Recombiner
H2 Supply
1879 B
4431
IV-3B
1875-9
To H2 Recombiner
Inside Containment
Outside Containment
H2 Supply
1880 B
4432
IV-5B
1876-9
To H2 Recombiner
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Entire system shown is
Seismic Class I Design

FIGURE 5.2-24

POST ACCIDENT CONTAINMENT VENTING SYSTEM

Line No. 64 INSTRUMENT AIR/P.A. VENTING SUPPLY LINE

Line No. 65 P. A. VENTING EXHAUST LINE

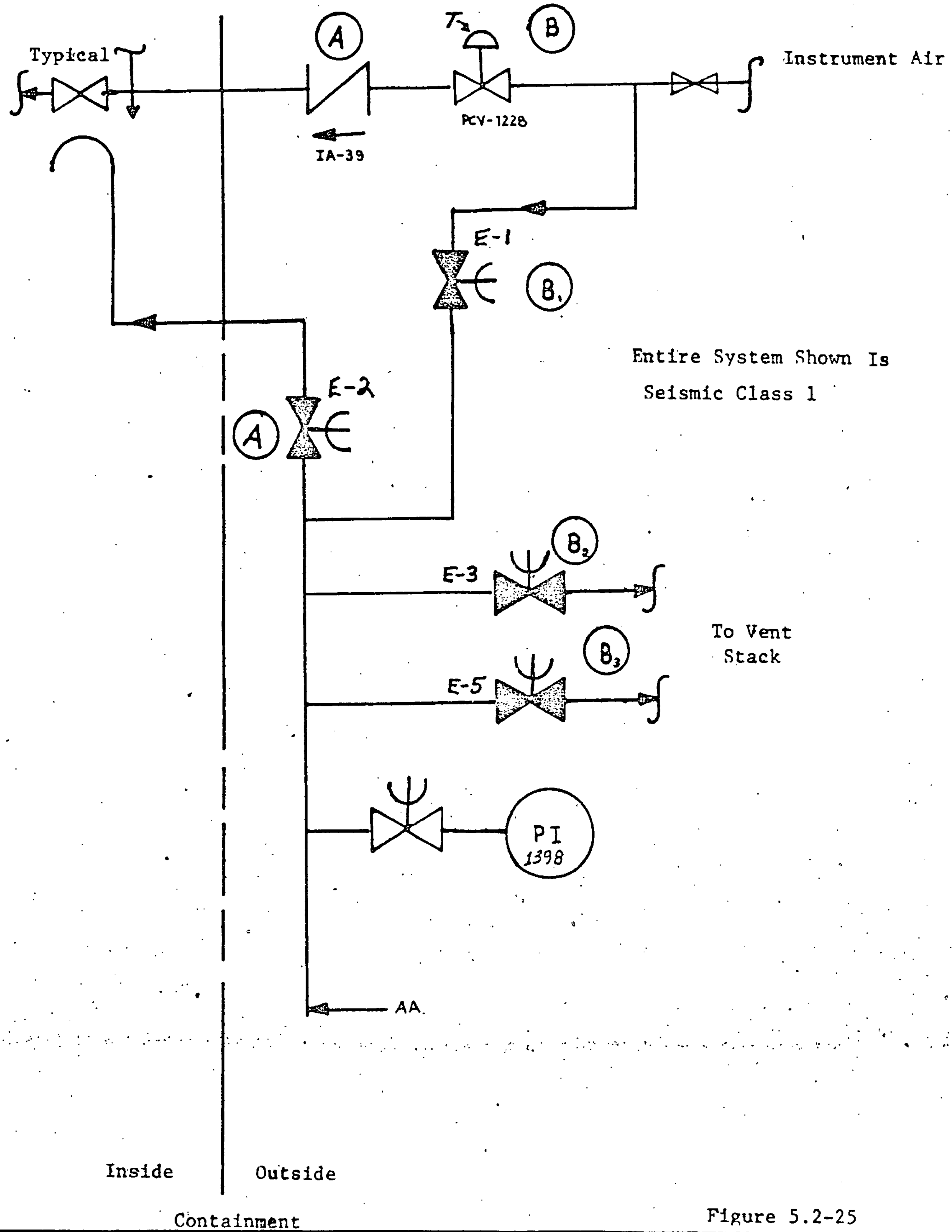

Figure 5.2-25

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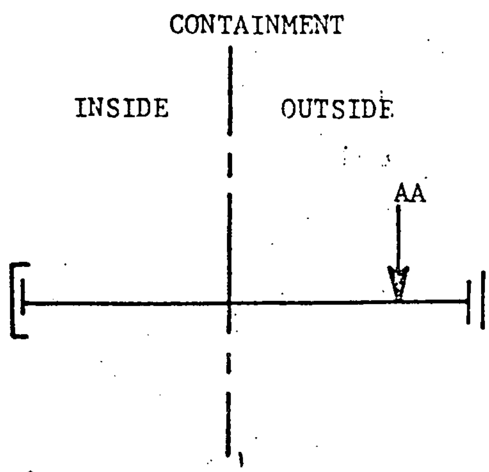
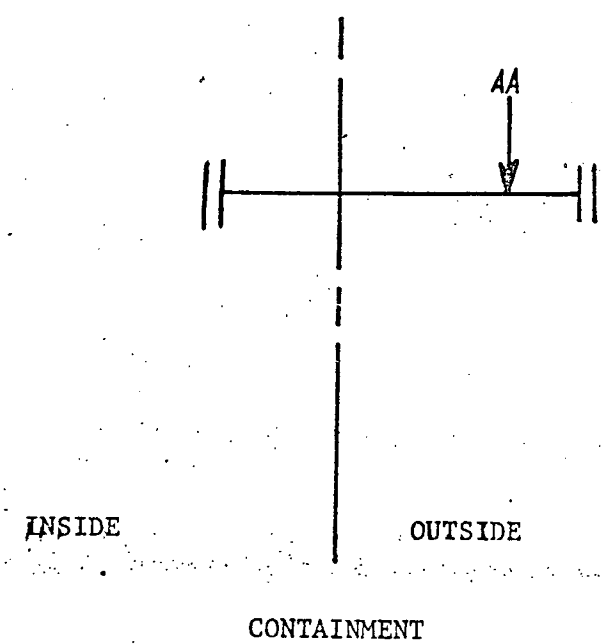
Line No. 66 CONTAINMENT LEAK TEST INSTRUMENT SENSOR LINE (4)



SEISMIC CLASS 1

Line No. 67 CONTAINMENT LEAK TEST AIR LINE (2)



SEISMIC CLASS 1

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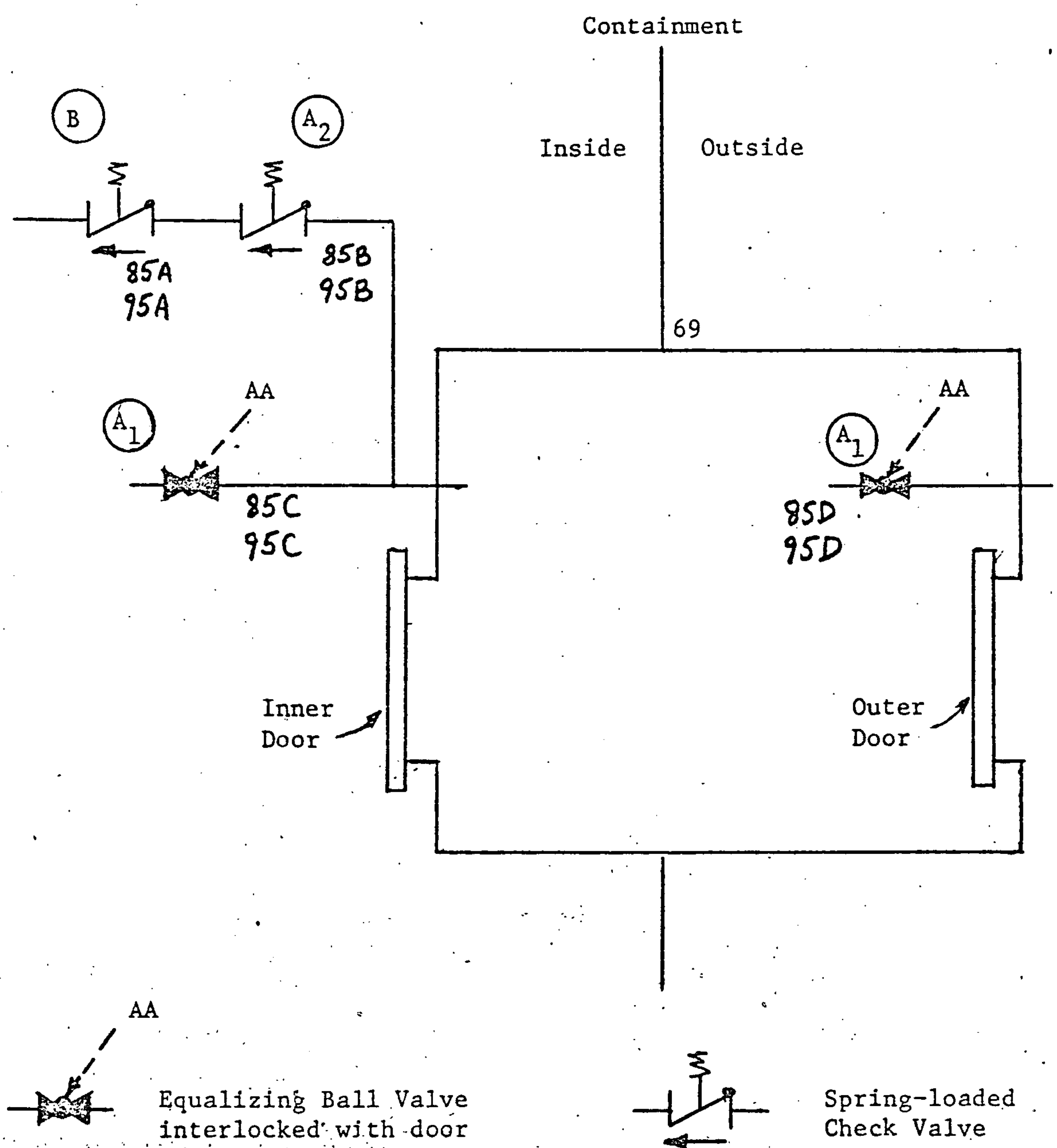
Figure 5.2-26

No. 69 PERSONNEL AIR LOCK (2)



Entire system shown is Seismic Class 1 Design.

Figure 5.2-27

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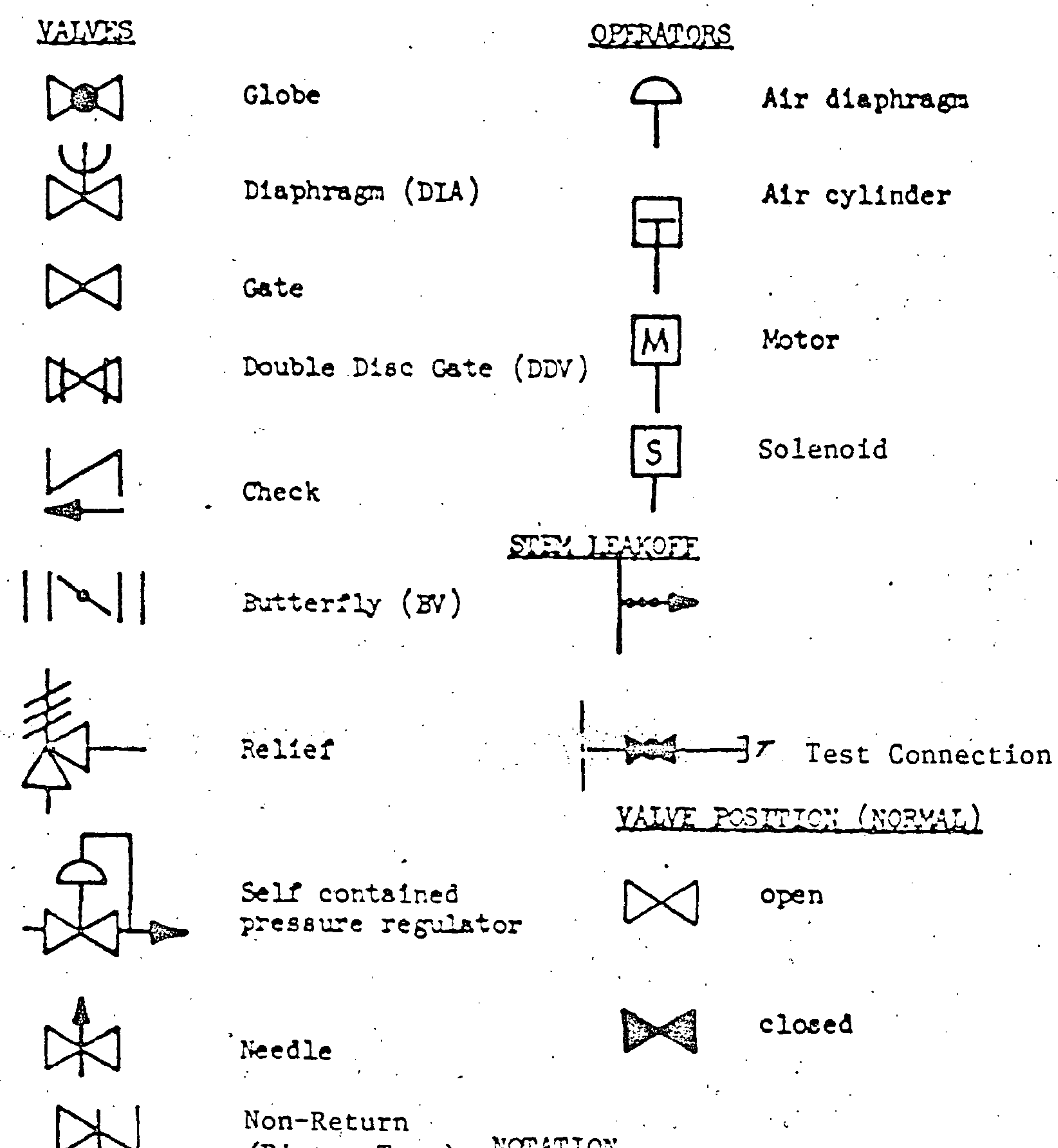
LEGEND



NOTATION

ASWI - AUTOMATIC SEAL WATER INJECTION
MSWI - MANUAL SEAL WATER INJECTION
AA - AUTOMATIC PRESSURIZATION WITH AIR
N_2 - MANUAL PRESSURIZATION WITH NITROGEN
LO - LOCKED OPEN
LC - LOCKED CLOSED
T - TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE A
P - TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE B
S-1 - SEISMIC CLASS I
S - OPEN S.I. Signal, Phase A

LEGEND FOR SYMBOLS, CONTAINMENT ISOLATION SYSTEM

FIGURE 5.2-28

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