123 Main Street
White Plains, New York 10601
914 681.6200

New York Power Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

August 14, 1984
JPN-84-54

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

Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Mark I Program

Reference: 1. NRC letter, D. B. Vassallo to J. P. Bayne, dated June 13, 1984.

2. NYPA letter, J.P. Bayne to D.B. Vassallo, dated June 7, 1984 (JPN-84-34).

Dear Sir:

The Authority met with members of your staff and NRC consultants on June 13, 1984 to discuss the Authority's Mark I containment program. As a result of this meeting, additional information regarding our program was submitted to Mr. C. Economos of Brookhaven National Laboratories.

Enclosed for your information and use are copies of the documentation provided to Mr. Economos. This documentation consists of FSAR pages 4.4-1 through 4.4-7, Table 4.4-1 and Figures 4.4-2 and 4.4-3 as well as five additional items discussed at the meeting. It is our understanding that this resolves all open items discussed at the meeting with the exception of justifying the 111°F pool temperature for case C3.2. As was discussed in a telephone conversation on July 31, 1984 with Mr. Byron Siegel of your staff, this question will be addressed in a separate letter.

A025
1/1

In response to Reference 1, for FitzPatrick, we did not implement the logic changes discussed nor use the revised analysis performed by General Electric. Rather, we have used the loads resulting from load case C3.3, which bound all other cases, in the SRV analyses (See Teledyne Technical Report TR-5321-2, attachment to Reference 2).

If you have any questions concerning this information, please contact Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

J. P. Bayne
Executive Vice President
Nuclear Generation

cc: Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, New York 13093

2064a/0020c

4.4 PRESSURE RELIEF SYSTEM

4.4.1 Power Generation Objective

The power generation objective of the Pressure Relief System is to limit any overpressure which occurs during abnormal operational transients.

4.4.2 Power Generation Design Bases

1. The safety/relief valves are designed to relieve overpressure during normal plant isolations and load rejections.

2. The safety/relief valves are designed to discharge to the primary containment suppression pool.

3. The safety/relief valves are designed to properly reclose following a plant isolation or load rejection so that normal operation can be resumed as soon as possible.

4.4.3 Safety Objective

The safety objective of the Pressure Relief System (11 safety/relief valves) is to prevent overpressurization of the Reactor Coolant System; this protects the Reactor Coolant Pressure Boundary from failure which could result in the uncontrolled release of fission products. In addition, the automatic depressurization feature of the Pressure Relief System acts in conjunction with the ECCS for reflooding the core following small breaks in the Reactor Coolant Pressure Boundary; this protects the reactor fuel cladding barrier from failure due to overheating.

4.4.4 Safety Design Bases

1. The Pressure Relief System prevents overpressurization of the Reactor Coolant System in order to prevent failure of the Reactor Coolant Pressure Boundary due to pressure.

2. The Pressure Relief System provides automatic depressurization for small breaks in the Reactor Coolant System so that the LPCI and the Core Spray System can operate to protect the fuel barrier.

3. The Pressure Relief System provides for manual depressurization at a remote auxiliary panel located outside the control room in the highly unlikely event the control room were to become uninhabitable.

4. The safety/relief valve discharge piping is designed to accommodate forces resulting from relief action and is supported for reactions due to flow at maximum relief discharge capacity so that system integrity is maintained.

5. The Pressure Relief System is designed for testing prior to Reactor Coolant System operation and for verification of the Pressure Relief System.

6. The Pressure Relief System is designed to withstand adverse combinations of loadings and forces resulting from operation during abnormal, accident or special event conditions.

4.4.5 Description

The Pressure Relief System consists of eleven safety/relief valves, all of which are located on the main steam lines, within the drywell, between the reactor vessel and the first main steam isolation valves (Figure 4.4-1). Table 4.4-1 shows the set pressures and capacities of the valves.

All the safety/relief valves can be either automatically actuated by excess steam pressure or the valves can also be opened manually through remote switches.

Seven of these valves are operated by a 2 position switch that can be placed in either a manual open or auto position. These seven valves are also automatically operated by relay logic circuits that are actuated by signals from the Automatic Depressurization Systems (ADS). The remaining four valves are operated by a 2 position switch that can be placed into either an open or closed position. The seven valves that are used during an automatic depressurization mode are discussed later in this section. Also, all eleven valves can be operated from a remote ADS auxiliary panel located outside the control room.

The main steam lines on which the safety/relief valves are mounted are designed, installed and tested in accordance with the applicable codes discussed in Section 16.5. The safety/relief valves are located on the four main steam lines. See Figure 4.4-1 for location of the valves and piping. The eleven safety/relief valves are designed, constructed and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1968 editions and addenda through Summer, 1970 and certified for flow in accordance with Article 9.

As shown on Figure 4.4-1, the eleven valves are identified as 71A, 71B, 71C, 71D, 71E, 71F, 71G, 71H, 71J, 71K and 71L.

The following description is typical for all eleven valves. Each pilot-operated safety/relief valve consists of two principal assemblies: a pilot stage assembly and the main stage assembly. These two assemblies are directly coupled to provide a unitized, dual function safety/relief valve.

The pilot stage assembly is the pressure-sensing and control element, and the main stage assembly is a system fluid-actuated follower valve which provides the pressure relief function. Self-actuation of the pilot assembly at set pressure vents the main piston chamber, permitting the system pressure to fully open the main assembly, which results in system depressurization at full rated flow.

Operation of the pilot assembly and main assembly is described in detail below. Refer to Figures 4.4-2 and 4.4-3 for schematic illustrations of the valve in closed and open positions.

The pilot assembly of the safety/relief valve consists of two relatively small, low flow pressure-sensing elements. The spring-loaded pilot disc senses the set pressure, and the pressure-loaded stabilizer disc senses the reseat pressure. Spring force (preload force) is applied to the pilot disc by means of the pilot rod. Thus, the adjustment of the spring preload force will determine the set pressure of the valve.

Operation of the pilot assembly is as follows:

During assembly, the pilot spring is adjusted to provide a preload force on the pilot disc which will establish the required set pressure of the valve. The spring preload force seals the pilot disc tightly to prevent leakage at normal operating pressures or lower system pressures.

In operation, as system pressure increases and reaches set pressure, the seating force acting on the pilot disc is reduced to zero causing the pilot disc to lift from its seat. Pilot disc lift results in the depressurization of the main piston chamber volume. Initial venting (depressurization) of the main piston chamber creates a differential pressure across the stabilizer disc in a direction causing the stabilizer disc to seat. System pressure acting upon the stabilizer disc via the internal porting maintains the pilot disc in the "lifted" position thereby maintaining main piston chamber venting until the required differential pressure across the main piston is achieved, at which point the main stage opens. When system pressure has decreased to the valve reseat pressure, the pressure-sensing stabilizer disc will unseat permitting the pilot disc to reseat; this in turn causes main piston chamber repressurization, which results in closing of the main stage.

The main assembly of the Target Rock safety/relief valve is basically a reverse (pressure) seated system fluid-actuated angle globe valve. Actuation of the main assembly permits discharge of fluid from the protected system at the valve's rated flow capacity and provides the system pressure relief function of the valve. The major components of the main stage are the valve body, disc/piston assembly and preload spring.

Operation of the main stage is as follows:

In its normally closed position, the main stage disc is tightly seated by the combined forces exerted by the preload spring and the system internal pressure acting over the area of the disc. Note that in the closed (no flow) position the static pressures will be equal in the valve inlet nozzle and in the chamber over the main stage piston. This pressure equalization is made possible by the internal passages provided; i.e., piston ring gap, vent hole, drain groove and stabilizer disc seat.

When system pressure increases to the valve set pressure, pilot stage operation will vent the chamber over the main stage piston to downstream of the valve via internal porting. This venting action creates a differential pressure across the main stage piston in a direction tending to unseat the valve. The main stage piston is sized such that the resultant opening force is greater than the combined spring preload and system pressure seating force.

Once the main stage disc starts to open, the seating force is rapidly reduced, allowing the main disc to open with its characteristic "pop open" action to the fully open position.

When system pressure has been reduced to design reseat pressure, the pilot disc reseats permitting repressurization of the main piston chamber. Flow of system fluid through the main stage piston ring gap and stabilizer seat then repressurizes the chamber over the piston. Main stage design is such that the repressurization of the piston chamber equalizes system pressure forces permitting the preload spring and flow forces to close the main stage. Once closed, the additional system fluid seating force, due to system pressure acting on the main stage disc, seats the main stage tightly.

Pneumatic operation is as follows:

A remotely controlled air operator is fitted to the pilot stage assembly to provide selective operation of the valve at system pressures ranging from 50 psig to valve set pressure. This is a diaphragm type pneumatic actuator which must be actuated to open the valve. It is actuated by means of a solenoid control valve which admits plant air to the air operator piston chamber and strokes the air operator stem, in turn stroking the pilot disc via the pilot rod. The main stage then opens as described in previous paragraphs. De-energizing the solenoid vents the air operator diaphragm chamber causing the air operator stem to return to its unstroked position. The pilot stage then reseats if system pressure is at the valve design reseat pressure or lower. Reseat of the pilot stage, which is as previously described, in turn causes reseat of the main stage as described therein.

The safety/relief valves are so installed that each valve discharge is piped through its own discharge line to a point below the minimum water level in the primary containment suppression pool, permitting the steam to condense in the pool. As a result of the Mark I Containment Program, a tee-quencher assembly has been installed at the discharge end of each relief valve line

to replace the ramshead. The tee-quencher hole pattern permits the discharge steam from the safety/relief valves to be distributed through one or more bays of the suppression pool. This division and distribution of SRV discharge has been shown in generic testing to reduce Torus shell pressure by factors of two or more when compared to ramshead pressures. Refer to figure 5.2-17 for details of the SRV tee-quencher. Vacuum relief valves are provided on each safety/relief discharge line to prevent drawing water up into the line due to steam condensation following termination of safety/relief valve operation. Without the vacuum relief valves, water in the line above the suppression pool water level would cause excessive pressure at the safety/relief discharge when the valve is again opened. The safety/relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid having to remove sections of this piping when the reactor head is removed for refueling. In addition, the safety/relief valves are more accessible for maintenance during a shutdown when they are located on the main steam lines.

Each of the eleven safety/relief valves is equipped with a nitrogen accumulator and check valve arrangement. These accumulators ensure that the valves can be held open following failure of the nitrogen supply to the accumulators, and they are sized to contain sufficient nitrogen for a minimum of five valve operations.

The automatic depressurization feature of the Pressure Relief System serves to back up the HPCI System under LOCA conditions. If the HPCI System does not operate and a discharge pressure signal exists at any one of four of the LPCI or either of two core spray pumps, the Reactor Coolant System is depressurized sufficiently to permit the LPCI and Core Spray System to operate to protect the fuel barrier. Depressurization occurs when some of the safety/relief valves are opened automatically to vent steam to the suppression pool. For small line breaks when the HPCI System fails, the Reactor Coolant System is depressurized in sufficient time to allow the Core Spray System or LPCI to cool the core and prevent any fuel cladding melting. For large breaks, the vessel depressurizes rapidly through the break without assistance. The signals that are associated with the automatic depressurization mode of seven of the safety/relief valves are described in the Technical Specifications. Overpressure protection is described in NEDA-24011-P (applicable revision) and results of the analysis are given in the supplemental reload licensing submittal. Further descriptions of the operation of the automatic depressurization feature are found in Section 6.4 and in Section 7.4.

Depressurization of the Reactor Coolant System can be effected manually in the event the main condenser is not available as a heat sink after reactor shutdown. The steam generated by core decay heat is discharged to the suppression pool. To control Reactor Coolant System pressure, the eleven safety/relief valves are operated by remote manual controls from the Control Room. Also a remotely located ADS control panel for manual control is provided in an area located away from the Control Room.

The number, set pressures and capacities of the safety/relief valves are provided in Table 4.4-1.

Criteria for the design and installation of safety/relief valves include the following:

a. Discharge tees are provided on safety/relief valves to equalize the discharge thrust force.

b. The discharge tees of the safety/relief valves are oriented to minimize detrimental effects resulting from steam impinging on other drywell equipment and structures.

c. Clearance of at least 6 inches is provided between valves and other equipment and structures.

d. Space greater than 2t + 2" (where t is minimum wall thickness) is provided between all welds on the header for inspection.

e. Clearance is provided between header and bottom of flange for bolt removal when valve is installed.

f. A flange rating of 1500 lbs. was provided for structural stability instead of a 900 lbs. rated flange required for pressure temperature rating.

g. An inlet pipe schedule of 160 was used for structural stability instead of Schedule 80 required for pressure-temperature rating.

h. The discharge piping on the safety/relief valve provides for equalization of thrust forces.

4.4.6 Safety Evaluation

The ASME Boiler and Pressure Vessel Code requires overpressure protection for each vessel designed to meet Code Section III. The code permits a peak allowable pressure of 110 percent of vessel design pressure (1375 psi gage for a 1250 psi gage vessel). The code specifications for safety/relief valves additionally require that the lowest safety/relief valve setpoint be at or below vessel design pressure (1250 psi gage) and the highest safety/relief valve setpoint be at or below 105 percent of vessel design pressure (1313 psi gage).

There are two major transients, the closure of all main steam line isolation valves and a turbine trip with a coincident closure of the turbine steam bypass system valves that represents the most severe abnormal operational transient resulting in a Reactor Coolant System pressure rise.

For JAF the transient produced by the closure of all main steam line isolation valves represents the most severe abnomal operational transient resulting in a Reactor Coolant System pressure rise when direct scrams are ignored. The required safety valve capacity is determined by analyzing the pressure rise from such a transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure of 1020 psig. The analysis hypothetically assumes the failure of the direct isolation valve position scram, the reactor is shut down by the back up, indirect, high neutron flux scram. Refer to the latest reload submittal for details of this analysis.

4.4.7 Inspection and Testing

The safety/relief valves were tested in accordance with Owner approved quality control procedures to detect defects and to prove operability prior to installation. The following final tests were witnessed on an audit basis by a representative of the Owner:

a. Hydrostatic test

b. Pneumatic leakage

c. Set pressure test

d. Response time test

The safety/relief valves were installed as received from the factory. The setpoints were adjusted, verified, and indicated on the valves by the vendor. Proper manual and automatic actuation of the safety/relief valves are verified periodically during plant life in accordance with requirements as set forth in the Technical Specifications.

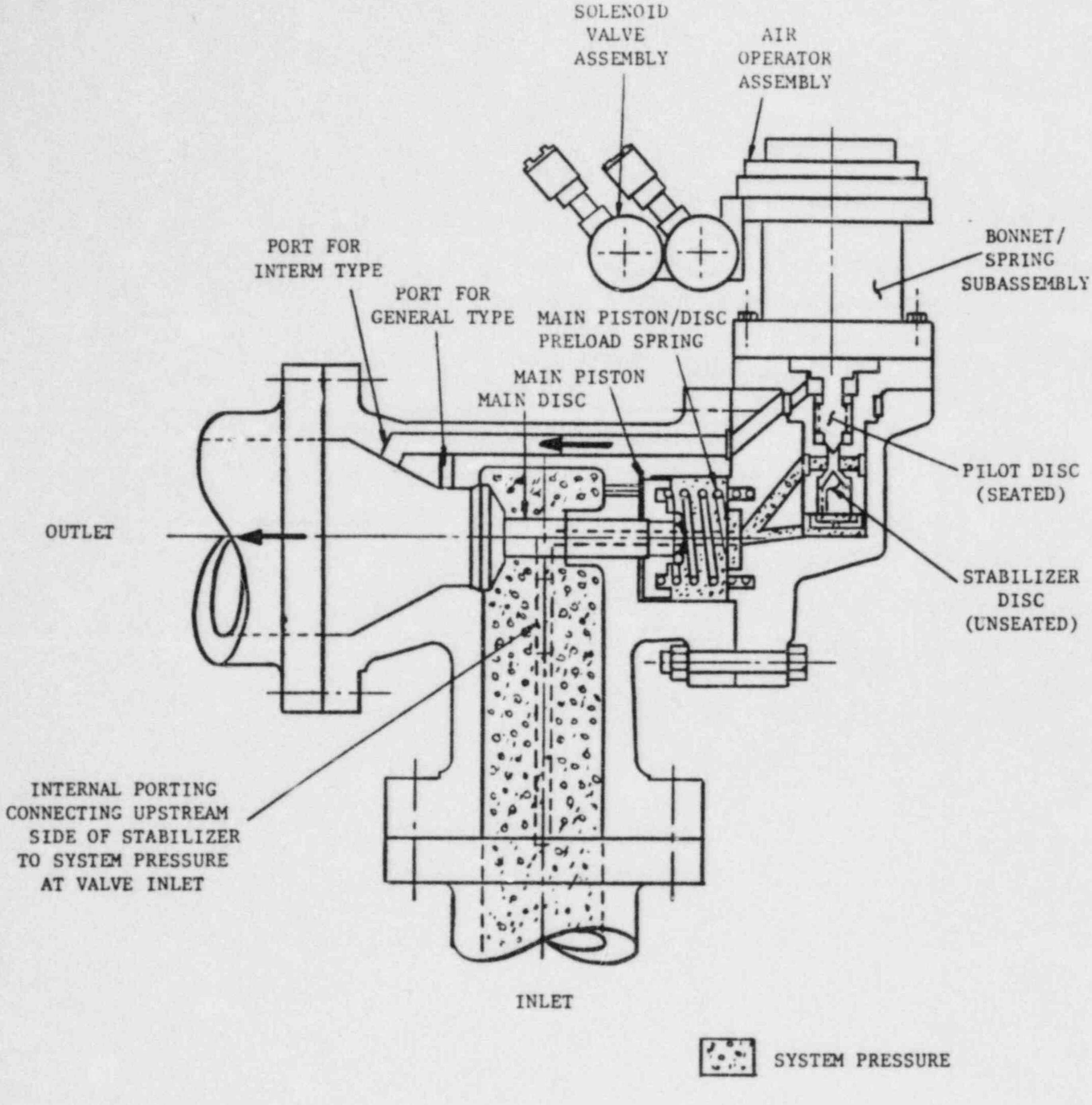
It is recognized that it is not feasible to test the safety/relief valve set-points while the valves are in place or during normal plant operation. The valves are mounted on 6 inch diam, 1500 lb primary service rating flanges so that they may be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The external surface and seating surface of all safety/relief valves are 100 percent visually inspected when the valves are removed for maintenance or bench checks.

The presently installed valves replace the old valves which were actuated by a bellows assembly, therefore testing procedures have been revised to incorporate this change.

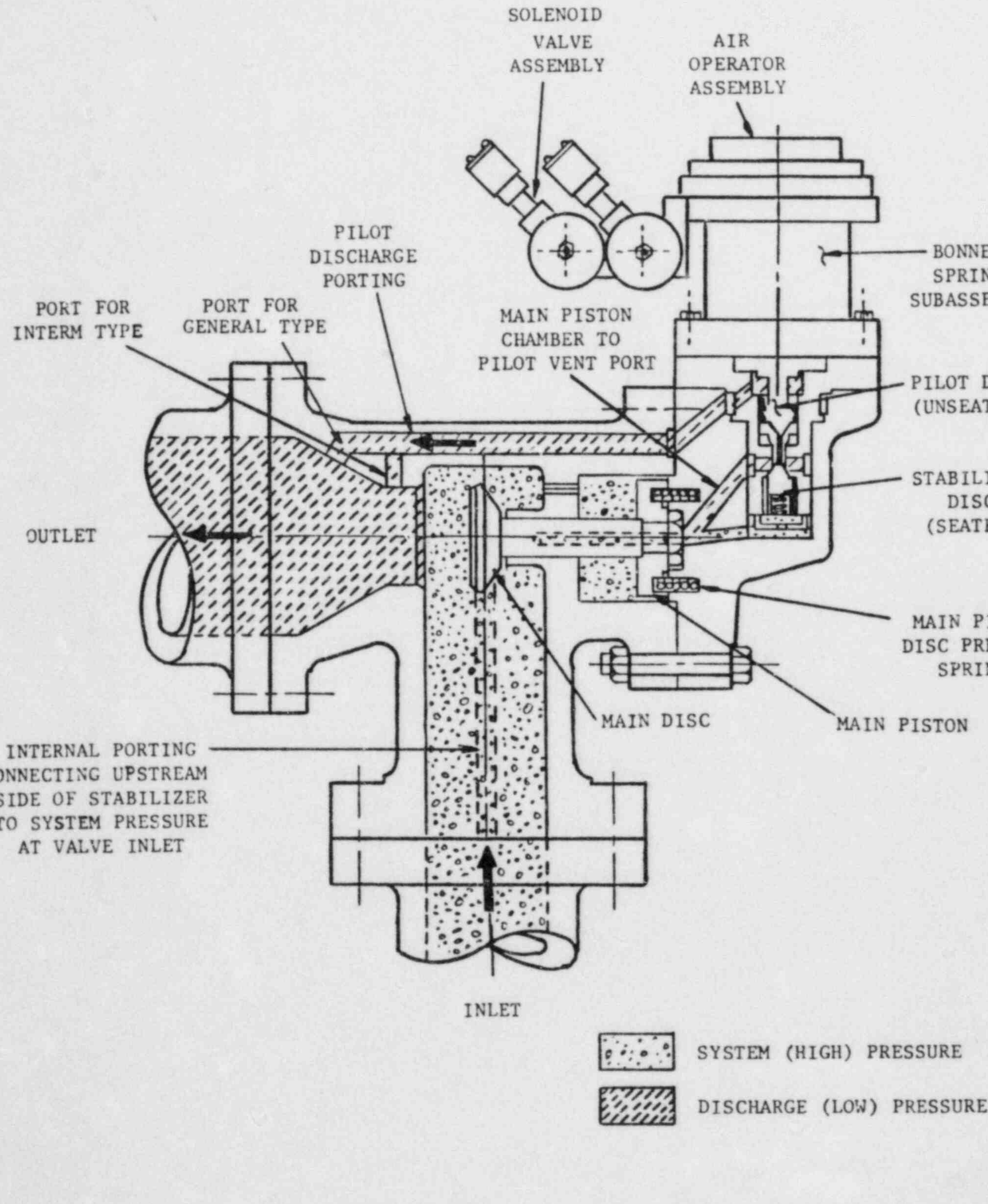
TABLE 4.4-1

REACTOR COOLANT SYSTEM SAFETY/RELIEF VALVES

Number of Valves	Set Pressure (psig)	Capacity at 103% Set Pressure (each) (lb/hr)
2	1090	818,000
2	1105	829,000
7	1140	855,000



JAMES A. FITZPATRICK	FSAR UPDATE
REACTOR COOLANT SYSTEM SAFETY/RELIEF VALVES - GENERIC TYPE -CLOSED POSITION	
REV. 0 JULY, 1982	FIGURE NO. 4.4-2



JAMES A. FITZPATRICK		FSAR UPDATE
REACTOR COOLANT SYSTEM SAFETY/RELIEF VALVES - GENERIC TYPE - OPEN POSITION		
REV. 0	JULY, 1982	FIGURE NO. 4.4-3

DRAFT

Response to Open Items from NRC/BNL/NYPA/TES Meeting

June 13, 1984

Item 1 Identify SRV lines which provide the ADS function

Response There are seven ADS lines at FitzPatrick. These are lines A, B, C, D, E, G, H. They are shown on the attached sketch.

Item 2 Provide the elevation of sensors for the temperature monitoring system

Response All sensors are located at the same elevation as the center of the quencher arms; i.e., five feet above the bottom of the pool.

Item 3 Provide a corrected figure showing SRV line entry angle into the pool (part of presentation to Item D-3)

Response The corrected figure is attached. The figure included in the handout had an error in location of one dimension.

This figure was included only to show entry angle and did not effect any other part of the presentation.

Item 4 a. Provide corrected pressures for the four SRV tests at FitzPatrick (part of presentation item D-2.2)

Response The correct test data is presented below:

Test No.	Maximum Pressure (psi)	Minimum Pressure (psi)	Frequencies (Hz)
1C	5.0	3.7	4.3, 6.3
2C	5.3	3.7	5.0, 6.3
3C	5.8	4.0	5.0, 7.1
4C	5.7	4.3	4.2, 6.3

Item 4 b. Provide the allowable multipliers which can be applied to SRV pressures without exceeding Code allowables for the containment structure

Response A full set of SRV multipliers for important containment elements was presented for the Vermont Yankee plant, at the PUAR meeting on February 16, 1984. We believe the relative order of allowable multipliers will be the same for VY and for

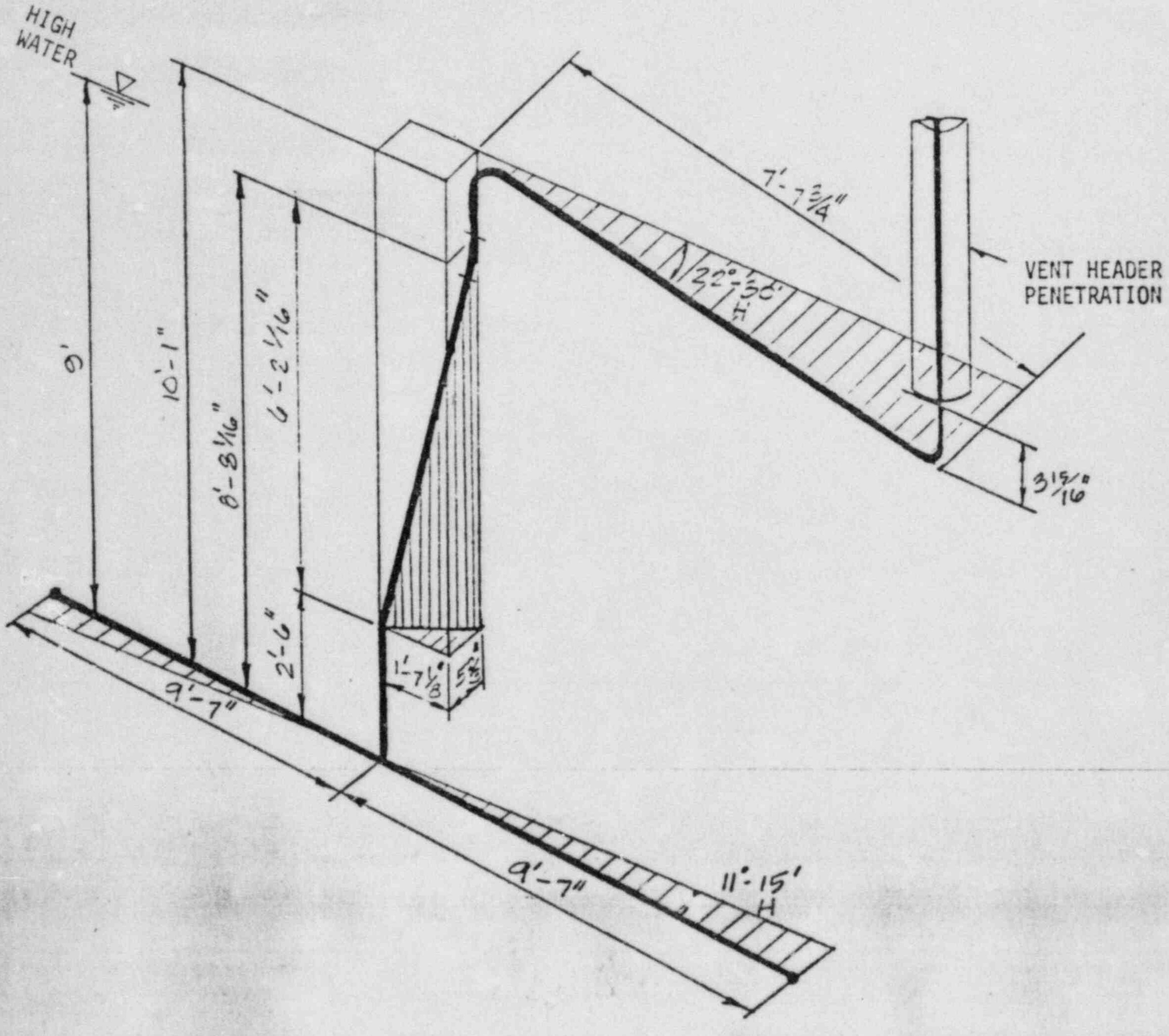
FitzPatrick; that is, structures with high multipliers on one plant will have high multipliers on the other. Based on this, we have directed our efforts to the component with the lowest allowable multiplier; the torus shell. Analysis for FitzPatrick shows that this multiplier is 5.66, compared to 3.35 for Vermont Yankee.

The next lowest number for VY was stress in the saddle (3.48). This is a local stress condition and does not reflect a safety margin for the saddle structure. The saddle clearly should not represent a real concern for an SRV overstress.

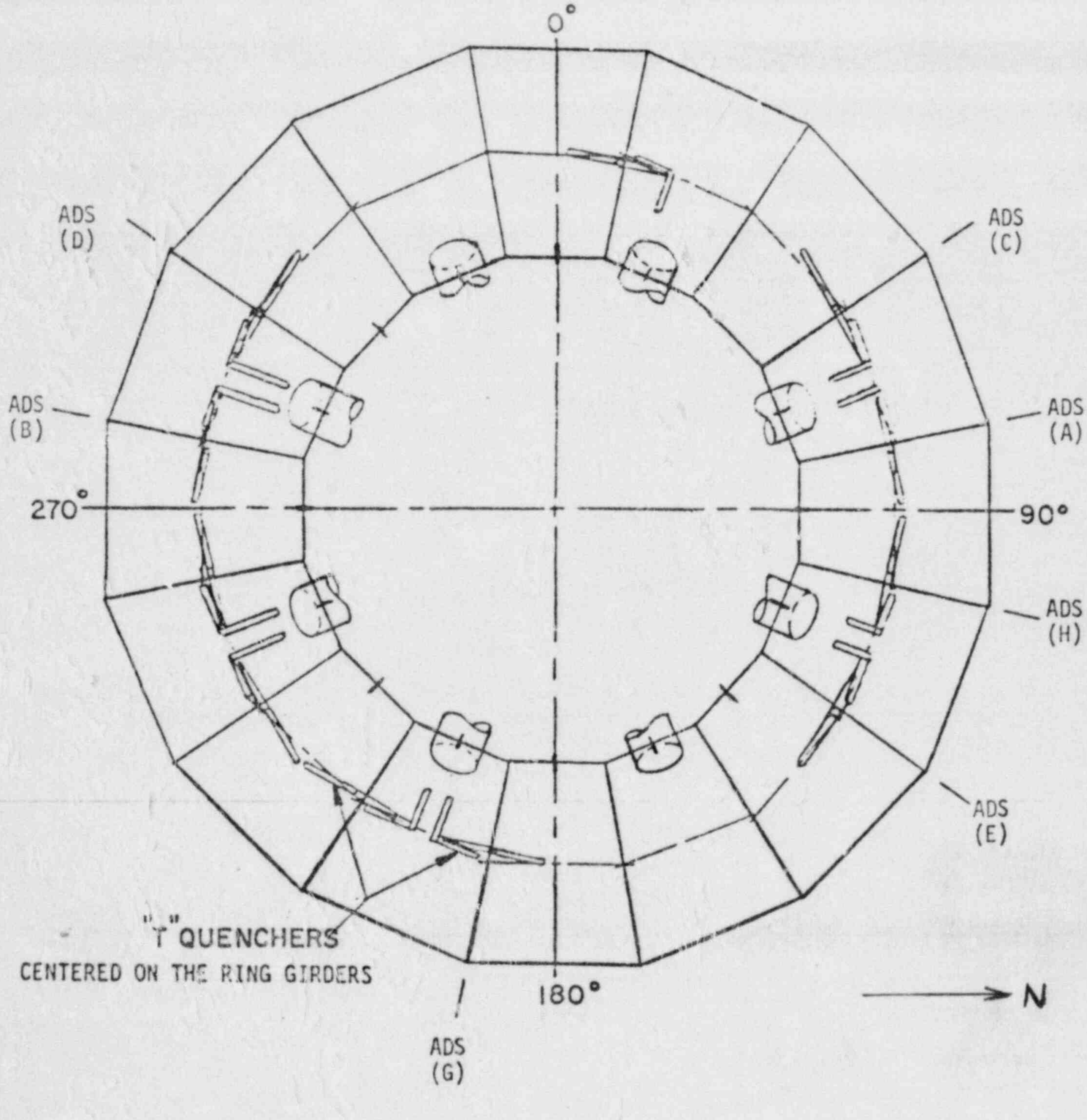
All other numbers previously calculated for VY were near or above a factor of 4.0 and were not recalculated for this review.

Item 5 Provide details of the safety relief valves used at FitzPatrick

Response This data was provided at the meeting on June 13, 1984. We believe this data is complete and provides an acceptable response to this item.



TYPICAL FITZPATRICK SRV LINE ROUTING IN POOL



T-QUENCHER LOCATIONS - FITZPATRICK