VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

January 10, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 015
PO/DLB:scj
Docket Nos. 50-280
50-281
50-338
50-339

Dear Mr. Denton:

Lessons Learned Short Term Requirements
Surry Power Station Units 1 and 2
North Anna Power Station Units 1 and 2

In our letters of October 24, 25 and November 26, 1979 we discussed our commitments to implement the short term lessons learned requirements on our Surry and North Anna units. The status of implementation of these requirements and certain information requested in your letter of September 13, 1979 are included in the attachments. Attachment A provides implementation status and information applicable to Surry Units 1 and 2. Attachment B provides implementation status and information applicable to North Anna Units 1 and 2. Attachment B also includes several items of additional information as requested by members of your staff in a meeting held on November 30, 1979 to discuss the status of implementation of the short term requirements on North Anna Unit 2. Attachments C through G include information which is applicable to both the Surry and North Anna Stations.

We will continue to update you on our progress in implementing the short term requirements and will be glad to answer any questions you may have.

Very truly yours,

W L Stallings for
C. M. Stallings
Vice President-Power Supply
and Production Operations

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cc: Mr. James P. O'Reilly

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Attachment A

Implementation of Lessons Learned
Short Term Requirements

Surry Power Station Units 1 and 2

INTRODUCTION

This attachment includes a summary of the status of each of the short term lessons learned requirements for Surry Units 1 and 2. Also included are summaries of certain tests, reviews, and additional information as requested in NRC letters of September 13, 1979 and October 30, 1979. Items are numbers as in NUREG 0578.

OPERATIONAL STATUS

Surry Unit 1 is currently at power.

Surry Unit 2 is currently shutdown for replacement of steam generators. Startup is expected in April, 1980.

LESSONS LEARNED SHORT TERM REQUIREMENTS

2.1.1.3.1. Pressurizer Heater Power Supply

Our previous response of October 24, 1979 identified two (2) actions necessary to establish full compliance with the requirements of this section. The status of these items is as follows:

1. Revisions to station emergency and abnormal procedures were required to instruct the operator in the use of pressurizer heaters in establishing and maintaining natural circulation. These procedure revisions are complete.

   Our October 24, 1978 response referenced a NSSS study which established that 125 kw of pressurizer heater capacity was needed to provide natural circulation in 3-loop plants. A copy of that study is included in Attachment C for your information.

2. A review of the qualification of the pressurizer heater motive and control power has been completed. The existing system meets all qualification requirements as clarified in the NRC letter of October 30, 1979.

2.1.1.3.2. Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

Our previous response of October 24, 1979 and November 26, 1979 identified two (2) actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. A modification to the high pressure air supply tanks is required to provide redundant motive power to the PORVs. This modification is complete on Unit 1 and will be completed on Unit 2 prior to startup from the current outage.

2. The qualification review of the motive and control power connections to the emergency buses for the PORVs and block valves has been completed. The existing system meets all requirements as clarified in the NRC letter of October 30, 1979.

2.1.2. Relief and Safety Valves Testing

Vepco, as a participant in the Westinghouse Owners Group, is working with other PWR owners and the Electric Power Research Institute (EPRI) in the development of a program for qualification of relief and safety valves under expected operating conditions involving solid-water and two-phase flow conditions.

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force submitted a description of the test program entitled "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems". We have reviewed the program and consider it to be fully responsive to the requirements presented in NUREG 0578. The EPRI program provides for completion of the essential portions of the test program by July, 1981. Vepco will be participating in the EPRI program to provide program review and to supply plant specific data as required.

2.1.3.a. Direct Indication of Valve Position

Our previous responses of October 24, 1979, and November 26, 1979 identified three (3) actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. In our initial response we committed to install acoustic monitors on the reactor coolant system safety valves. Details of the safety valve acoustic monitoring system were included in Attachment A of our November 26, 1979 response. Installation of these monitors on Unit 1 is complete. Installation on Unit 2 will be completed prior to startup from the current outage.

2. Pressurizer PORVs currently have direct indication derived from limit switches on each valve. Based on the clarification provided in the NRC letter of October 30, 1979, we committed to provide PORV alarms based on the limit switches and to replace the existing limit switches with environmentally qualified limit switches. Since that time we have decided to install acoustic monitors on each PORV. The acoustic monitors for the PORVs will be identical to those to be installed on the safety valves. Alarms will be provided in the control room. The acoustic monitors on the PORVs are installed on

Unit 1 and will be installed on Unit 2 prior to startup from the current outage.

3. The acoustic monitoring system used to monitor the PORVs and safety valves will be seismically and environmentally qualified by the vendor. Additional information regarding the qualification of these instruments will be forwarded as soon as it becomes available.

2.1.3.b. Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

Our previous responses of October 24, 1979 and November 26, 1979 identified three (3) actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. Changes to emergency procedures have been made to emphasize the need to ensure adequate coolant flow and to ensure that the reactor coolant temperature and pressure are maintained or immediately adjusted to achieve an appropriate margin to saturation. Emergency procedures have been revised to incorporate NSSS vendor generic guidelines for the identification of and recovery from inadequate core cooling conditions.

2. In our response of November 26, 1979, we committed to the installation of a Westinghouse core subcooling monitor. The subcooling meters will be installed on Unit 1 by January 31, 1980 and on Unit 2 prior to startup from the current outage.

3. In our response of November 26, 1979, we committed to install a reactor vessel water level monitor. The Westinghouse Owners Group, of which we are members, is continuing its review of the available or potential technologies to provide this function. Certain design and functional problems remain unresolved at this time. Accordingly we are not able to provide a conceptual design at this time. We will forward a design description of the reactor vessel water level meter as soon as it is available.

2.1.4. Diverse Containment Isolation

As explained in our previous responses of October 24, 1979 and November 26, 1979, Surry Units 1 and 2 already meet all the requirements of this section.

2.1.5.a. Dedicated H_2 Control Penetrations

As explained in our October 24, 1979 response, Surry Units 1 and 2 already meet all the requirements of this section.

2.1.5.b. Inerting BWR Containments

This section is not applicable to Surry Units 1 and 2.

2.1.5.c. Capability to Install Hydrogen Recombiners at Each Light Water Nuclear Power Plant

As explained in our previous responses of October 24, 1979 and November 26, 1979, Surry Units 1 and 2 already meet all the requirements of this section.

2.1.b.a. Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems)

All systems outside containment which are likely to contain highly radioactive fluids during or following an accident have been inspected and leak tested. Priority maintenance orders have been issued for those components from which leakage was identified. Leakage rates from the systems tests were as follows:

System	Number of Leaks Identified	Approximate Leak Rate
Boron Recovery	5	< .1 gallon per day
Resin Waste Disposal	0	0
Sampling	4	.25 gallon per day
Containment and Recirculation Spray	1	< .1 gallon per day
Chemical and Volume Control	24	< .1 gallon per day
Safety Injection	5	.3 gallon per day
Containment Vacuum	0	0

A long term leakage measurement and preventive maintenance program has been developed. Inspections of the systems included will be performed periodically and any corrective actions required will be handled utilizing priority maintenance orders. We currently have two ongoing programs to reduce the possibility of valve leakage. These programs are (1) periodic Grinnel valve diaphram replacement and (2) carbon steel stud replacement in systems containing boric acid. Station procedures used to return components to service following maintenance currently address shaft and other forms of leakage. A review of these and other procedures is in progress to ensure that leakage reduction measures have been adequately addressed.

The special test that was recently completed to estimate existing leakage rates provides the basis for developing a periodic integrated leakage test to be performed on a refueling frequency. Systems to be included are those which were included in the leakage tabulation listed above. Those systems that have been excluded from the leakage reduction program are, Liquid Waste, Containment Purge, Vent and Drain, Gas Stripper, Reactor Cavity Purification, Spent Fuel Pit Cooling and Purification, and those portions of the Chemical and Volume Control System which are not part of the Safety Injection function.

(2.1.6.b) Design Review of Plant Shielding And Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations

We have performed a preliminary plant radiation and shielding design review for Surry Units 1 and 2. The preliminary review is based on the North Anna radiation "zone maps" modified to accommodate significant differences in plant design. The minor differences will be factored into the final shielding review.

As a result of this review, we have found that additional shielding and plant modifications will be required.

The shielding review has been conducted in two parts: the Mitigation phase and the Recovery phase of an accident. The Mitigation phase, which is assumed to last for six months, considered the radiation levels as the result of operating the recirculation portion of the Safety Injection and Recirculation Spray Systems, and the Post-Accident Sampling System. In addition, the review also considered the radiation levels from the auxiliary building sump, and the drain lines from the discharge of the auxiliary building and safeguard building sump pumps to the low level liquid waste tank, as well as the containment.

The Recovery phase has been identified as the period six months after the accident when cleanup and plant recovery is undertaken. Based on the TMI-2 experience, the Recovery phase is considered to be a controlled evolution that will be planned and carried out to meet the specific recovery requirements of the particular accident.

Mitigation Phase

The integrated radiation dose calculated during this review is comprised of the 40 year normal dose and a 6 month mitigation phase dose. The safey equipment required to operate during the mitigation phase will be the same as that equipment tabulated for NRC I&E Bulletin 79-01. The source term developed to calculate the 40 year normal operating dose is based on the assumptions in the FSAR. The source terms assumed to calculate the 6 month mitigation phase dose are based on TID-14844 and Regulatory Guide 1.4 as follows:

	Source Term	Basis
Sump Water	0% Noble Gas	Sump Water is Degassed
	50% Halogens	TID-14844
	1% Solid fission products	TID-14844
Primary Coolant Sample	100% Noble Gas	RG 1.4/TID-14844
	50% Halogens	TID-14844
	1% Solid fission products	TID-14844
Containment Atmosphere	100% Noble Gas	RG 1.4/TID-14844
	25% Halogens	RG 1.4
	0% Solid fission products	RG 1.4

The exact course of any accident which has the scope and complexity of that experienced at Three Mile Island - 2 is unpredictable. It is impracticable to determine the dose rate and shielding requirements for every possible location and time duration associated with each possible failure, or incident, which requires personnel access. We have, therefore, developed radiation "zone maps" to be used as post-accident administrative guidelines. These radiation "zone maps" show estimated worst case gamma rates in various areas of the plant as a function of time. The "zone maps" will be used to help the plant operators to plan access and egress routes, help evaluate the relative benefits of delaying certain actions to allow for radioactive decay. It should be noted that the gamma dose rates shown on the "zone maps" are based on worst case source terms, but do not consider an airborne source term. Based on the severity of the situation, actual dose rates may be smaller. The "zone maps" therefore are used only as guidelines and actual radiation levels will be determined through actual surveys. In addition, the dose rates listed on the "zone maps" are based on the highest, or one of the highest, dose rates in that area. The dose rate at other locations within that area may be lower.

The final radiation "zone maps" will be reviewed in conjunction with the I&E Bulletin 79-01 equipment review which is still in progress.

Shielding for continuous occupancy areas, per NUREG 0578, must be designed to maximum dose rate of less than 15 mrem/hr due to radiation from piping and components. The continuous occupancy areas have been defined as the control room, security control center, technical support center, and operational support center.

We have done a preliminary evaluation of the areas requiring continuous occupancy for dose rates from piping systems as required by NUREG 0578 based on the modified North Anna radiation "zone maps". The doses are expected to be similar except the Surry hydrogen recombiner is located inside the containment rather than in the auxiliary building. In addition, we have assumed the North Anna dose rate to these areas from the containment dome. Table 1 is a preliminary estimate of the dose rate to each area from the piping and the containment dome. Based on Table 1, additional shielding does not appear to be required for the control room, operations support center, or technical support center. The other areas will require additional shielding, relocation, or procedure modifications as identified below to meet the requirements of NUREG-0578.

TABLE 1

		Estimate of Dose Rate from Piping (Mr/Hr)		Estimate of Dose Rate from Containment (Mr/Hr)	
Area	Accident (Hr)	Accident in Unit 1	Accident in Unit 2	Accident in Unit 1	Accident in Unit 2
Control Room	0+	3	3	110	110
	1	**	**	16	16
Technical Support Center	0+	3	**	110	110
	1	**	**	16	16
Counting Lab/ HP Area	0+	10^2	10^2	9.2×10^3	1.3×10^4
	1	–	–	1.5×10^3	2×10^3
	24	–	–	17	21
Security Control Center	0+	86	**	1.1×10^3	**
	1	37	**	160	**
	8	9	**	25	**
	24	2	**	2	**

NOTE: The dose rates are based on preliminary calculations but we do not expect that the values will change significantly. The calculations will be checked by July, 1980.

0+ indicates shortly after accident initiation
** neglibible

Recovery Phase

Our evaluation considered the possible operation of the liquid, gaseous, and solid waste systems, letdown and charging portion of the chemical volume and control system, boron recovery system, vent and drain system and the containment purge portion of the atmosphere cleanup system with regards to personnel and equipment irradiation.

Based on the results of our evaluation as presented below, the installed radioactive waste systems, boron recovery system, containment purge system, letdown and charging portion of the CVCS system will not be used for post-accident cleanup of highly radioactive fluids. It should be noted that the design basis for these systems and associated shielding did not include this use.

1. The majority of electrical equipment in these systems is not qualified to meet the integrated radiation doses to which they would be exposed in processing and concentrating the highly radioactive water or gas.

2. The activity levels (based on Regulatory Guide 1.4 and TID-14844) of the influent to the liquid waste or boron recovery system is approximately 2 x 10^3 uci/cc after 6 months of radioactive decay. The area radiation dose rate from the concentrated waste and storage tanks would severely limit access to parts of the auxiliary building and hinder operation of both units.

3. Since the radioactive waste systems are common to both Units 1 and 2, the use of these systems for cleanup of waste in the accident unit would preclude the normal use of the radioactive systems for the non-accident unit.

4. There is extensive piping for the recovery systems throughout the auxiliary building. The resulting dose rate from all these systems operating simultaneously would severely limit access for the required operation of both units. Shielding for the recovery system piping and components would be very difficult and in some cases may be impossible to install due to the arrangement of the piping and equipment.

Conclusions

As a result of the plant radiation and shielding review, we have identified the following shielding and plant modifications required to meet the personnel exposure limits and equipment irradiation qualification required by NUREG 0578 and subsequent clarifications:

1. Shielding of portions of the lines added as part of the new post-accident sampling system may be required.

2. Shielding for the Post-Accident Sampling Facility will be required.

3. The drain system for the auxiliary building sump and the safeguards building sump will be modified such that these sumps can be pumped to the affected unit's containment instead of to the high or low level waste tanks. This would eliminate a significant potential source of activity in the basement of the auxiliary building.

Conclusions (cont'd)

4. Shielding will be added in or around the auxiliary building sump to reduce accident level dose rates.

5. Sampling procedures are being modified and temporary shielding is being added to limit dose rates at the present sample facility .

6. Additional shielding, area relocation, or procedural modifications are being evaluated to limit radiation dose rates in the counting lab, and the security control center.

7. The letdown line will be modified to divert letdown to the containment sump under post-accident conditions.

8. System modifications to permit interfacing with external process systems designed and shielded after the accident is being evaluated. The external process sytem design would be based on the extent of the accident and would utilize the most current technology available at the time of the accident.

2.1.7.a. Auto Initiation of Auxiliary Feed

As explained in our previous response of October 24, 1979, Surry Units 1 and 2 already meet all the requirements of this section.

2.1.7.b. Auxiliary Feed Flow Indication

Our October 24, 1979 response identified one modification which would be necessary to place Surry Units 1 and 2 in full compliance with the requirements of this section. The modification involved relocation of the auxiliary feedwater flow indicators power supplies to alternate cabinets in order to meet the diversity requirements. This change is complete on Unit 1 and will be completed on Unit 2 prior to startup from the current outage.

(2.1.8.a) IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

A design and operational review of the reactor coolant and containment sampling systems has been performed for Surry Units 1 and 2. This review was conducted to determine the capability of promptly obtaining samples under post-accident conditions without incurring a radiation dose to any individual in excess of 3 Rem to the whole body or 18.75 Rem to the extremitites. The source terms for these fluid systems are provided in section 2.1.6.b.

This review has determined that there are major difficulties with the existing sampling facility in obtaining representative samples and with performing the required analysis on these highly radioactivity samples. Therefore, the sample system will be modified, in the long term, to provide the capability to obtain and analyze post-accident samples from the reactor coolant system, containment atmosphere and containment sump.

Short Term Sampling

Special procedures are being developed for monitoring of the sample area prior to entry and for taking samples under accident conditions. Lead shielding and a small sample bomb will be utilized to minimize exposure. These procedural and equipment changes to facilitate sampling in the short term will be completed by January 31, 1980.

Proposed Long Term Design Basis

The shielded sample area will be located in an area of relatively low background activity. It is proposed that all analysis and dilution will be performed within the shielded area. However, provisions will be made to remove samples for remote analysis. During post-accident operation the discharge would be collected and processed or routed back to the containment. The sampling facility airborne activity will be controlled by the use of a ventilation system which contains prefilter, charcoal and HEPA filters. A decontamination capability will be provided to reduce personnel exposure when access for sample acquisition, maintenance, or calibration of equipment is required.

The sample system piping and components, up to the second isolation valve, will conform to the QA Category and seismic requirements of the system to which each sample line is connected. The piping and components downstream of the second isolation valve will be designed to Quality Group D as defined in Regulatory Guide 1.26 and nonseismic Category I requirements. All sample lines will be 3/8 inch tubing to limit reactor coolant or containment air leakage due to a failure of the sample line. For all sample lines which may contain large inventories of activity, connections outside the containment or sample enclosure will be welded or capable of being leak tested.

Shielding of the new sample lines will be provided where required and will be consistent with the shielding survey conducted in accordance with section 2.1.6.b. The design basis for shielding the sampling area is to limit the dose rate to 500 mrem/hr to an operator when drawing a sample one hour after an accident. Radiation exposure to personnel in transient will not exceed GDC 19 limits. To ensure representative samples and to reduce radiation exposure to personnel and equipment, provisions will be made to purge and flush highly radioactive fluid back to the containment or to a specially designed waste handling system.

On-line equipment will be utilized to perform much of the analysis. In addition, a backup provision for obtaining and analyzing these samples manually, by means of manipulator arms and a viewing window, is being evaluated.

The selection of equipment is not final. Proposals for sampling techniques and equipment are still under review for equipment availability, reliability and qualification. However, the sample system will provide for the samples and analyses as outlined below:

Reactor Coolant Sample

Provisions will be made to draw within the first hour a pressurized reactor coolant sample from either the hot leg of one loop or the cold leg of a different loop. Provisions will exist to analyze this sample within one hour after sample acquisition and to quantify the following constuituents: radioisotopes, boron concentration and gross dissolved gas. Provisions will be made to draw a sample from the reactor containment sump and analyze it for radionuclides.

Containment Atmosphere Sample

Provisions will be made to draw, under negative or positive pressure, within one hour after an accident, a representative sample of the containment atmosphere. Provision will be made to analyze the sample for radionuclides. Hydrogen concentration in the containment atmosphere will be determined in accordance with the requirements of Section 2.1.9.

2.1.8.b. Increased Range of Radiation Monitors

Our responses of October 24, 1979 and November 26, 1979 identified two (2) areas where actions were necessary to meet all requirements of this section.

1) In our previous responses, we expressed our concerns that fully qualified commercially available monitors may not be obtainable for the extended ranges specified. We still believe this to be the case but we will continue to make every effort to meet the requirements for inplant monitoring capabilities by January 1, 1981.

2) In our November 26, 1979 response we committed to provide interim methods for the quantification of high level releases.

Shielded area radiation monitors that meet the requirements specified in the NRC letter of October 30, 1979 have been purchased to measure the radiation levels on main steam safety valves, the process vent and the ventilation vent. The monitors are NRC Industries Model TA-600 detectors (10^{-4} to 10^{+4} R/hr) with check source and Model TA-900 controllers to be installed in the control room to provide a continuous readout. Normal AC power with a backup power supply will be provided.

Procedures are being developed for estimating release rates. The procedures use predetermined calculational methods to convert the measured radiation level to radioactive effluent release rates.

Installation of monitors and finalization of procedures for Unit 1 will be completed by January 31, 1980 and for Unit 2 prior to startup from the current outage.

2.1.8.c Improved In-Plant Iodine Instrumentation Under Accident Conditions

Our previous responses of October 24, 1979 and November 26, 1979 identified two actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. An adequate stock of "silver zeolite" sampling cartridges has been purchased and is in stock at Surry Power Station. Included in Attachment D is a manuscript entitled "Retention of Noble Gases by Silver Zeolite Iodine Samples" which documents the adequacy of the silver zeolite cartridges under accident conditions.

2.1.8.c. (cont'd)

2. Procedures for the use of silver zeolite cartridges in iodine sampling have been developed and incorporated into station Emergency Plan Implementation Procedures.

2.1.9. Analysis of Design and Off-Normal Transients and Accidents

Our responses of October 24, 1979 and November 26, 1979 identified three (3) areas where actions were required in order to establish full compliance with the requirements of this section. The status of these items is as follows:

1. Station emergency procedures have been revised to include the new generic guidelines developed by Westinghouse. This includes small break LOCA emergency procedures and procedures for recognition of and recovery from inadequate core cooling conditions. Operator training on the procedure revisions has been completed.

2. In our response of November 26, 1979, we committed to the installation of a reactor vessel head vent. A conceptual design of the proposed reactor vessel head vent system is included in Figure A-1.

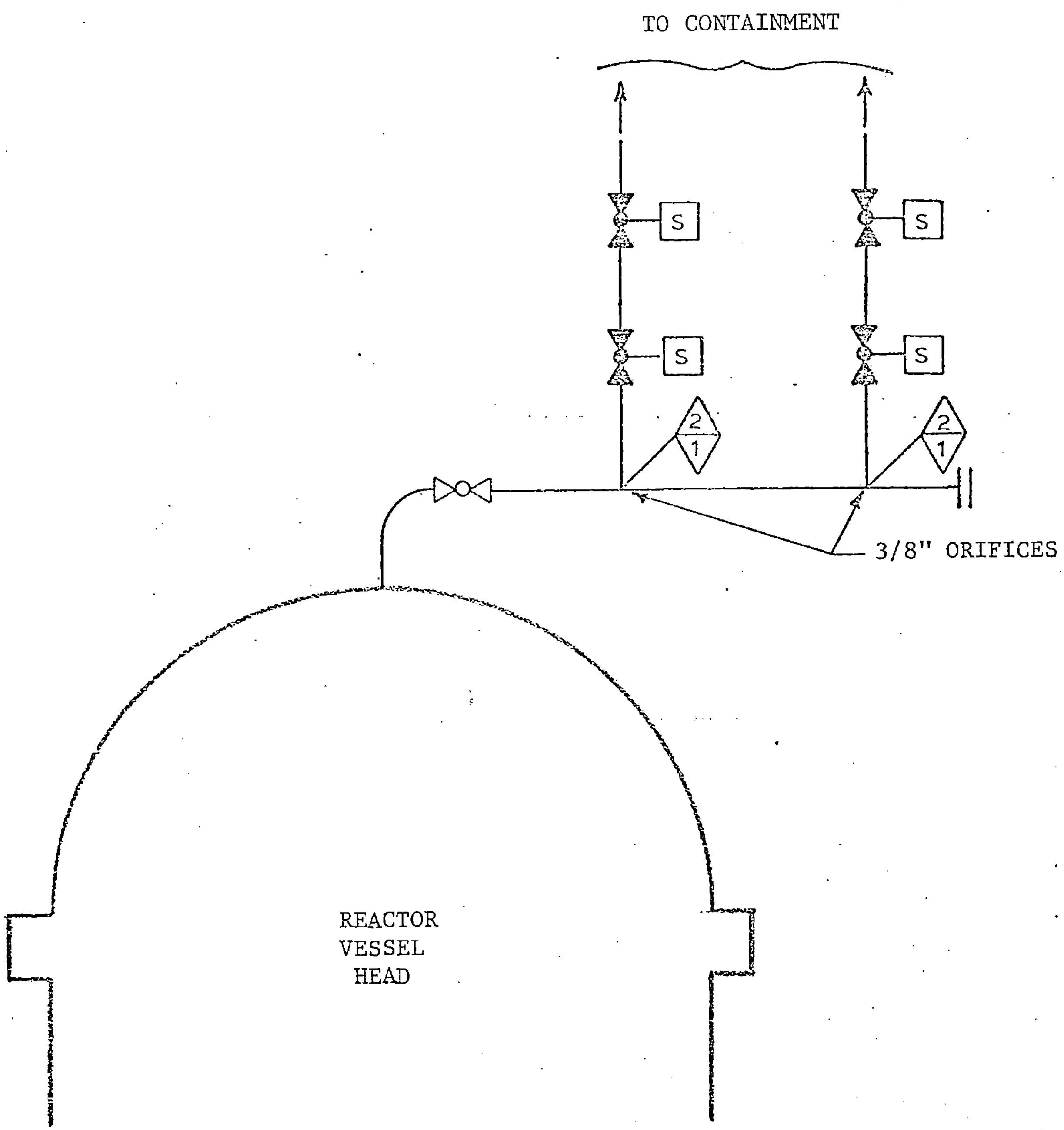
   The system is designed such that no single failure will prevent vessel gas venting or prevent venting isolation. The 3/8 inch orifice restricts the flow rate from a pipe break downstream of the orifice to within the makeup capacity of one charging pump.

   Procedures and techniques for operation of the vessel vent are still under evalation pending selection of a reactor vessel level instrument. Procedures for the use of the vessel vent system will be finalized following the final selection and design of the level instrument, and prior to operation following installation of the vent.

   We also will install a vent system on the pressurizer. A conceptual design of the proposed pressurizer vent system is included in Figure A-2. The system is designed such that no single failure will prevent pressurizer venting or prevent vent isolation.

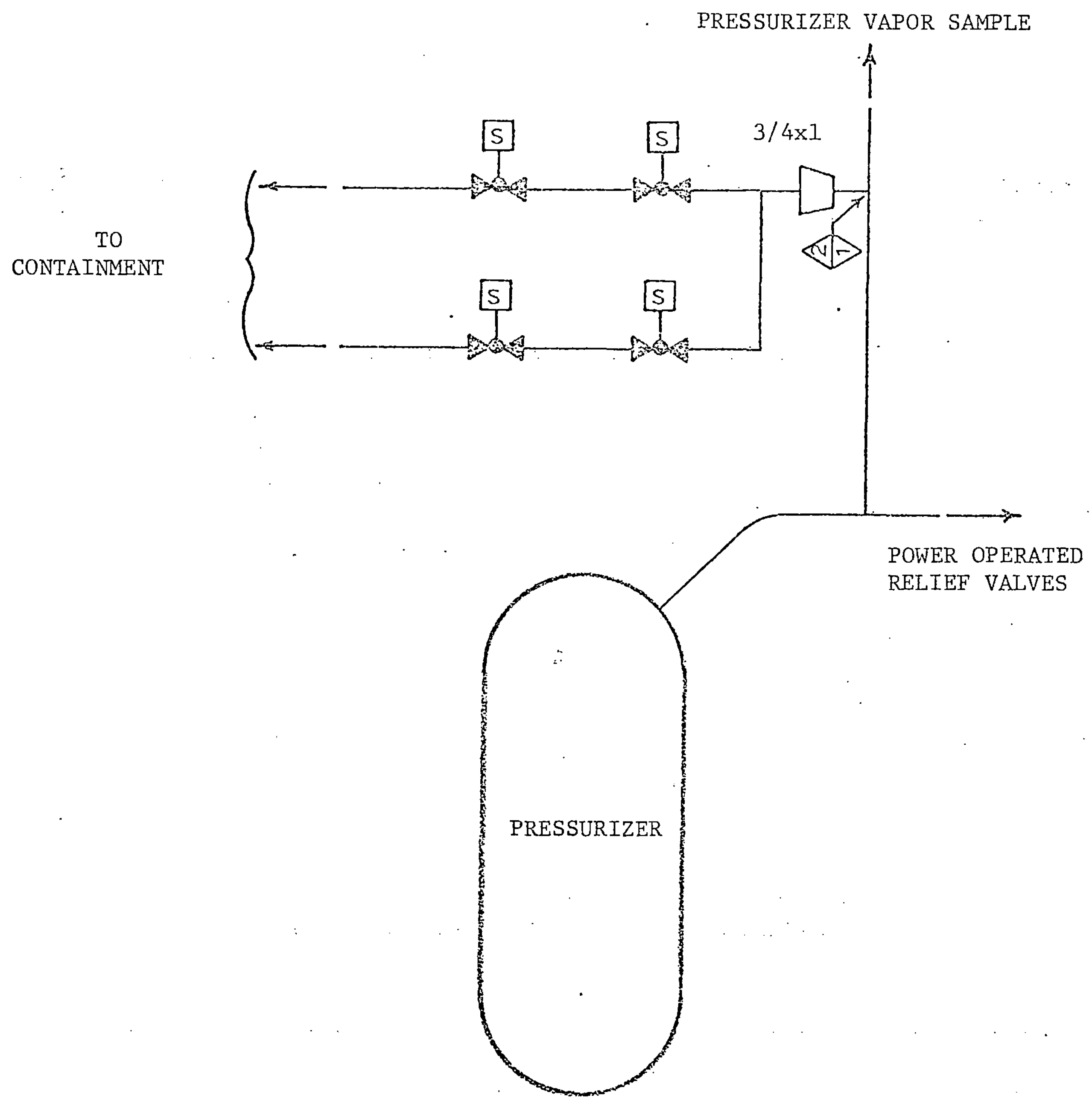
   Procedures for the use of the pressurizer vent system will rely on the existing pressurizer level instrumentation and will be prepared prior to operation following installation of the vent.

FIGURE A-1



FLOW DIAGRAM OF THE REACTOR VESSEL HEAD VENT SYSTEM

FIGURE A-2



FLOW DIAGRAM OF THE PRESSURIZER VENT SYSTEM

2.1.9. (cont'd)

3. In our November 26, 1979 letter we committed to install additional containment instrumentation including wide range water level, increased range pressure indication, and improved hydrogen indication. We also committed to upgrade the existing narrow range containment level instrumentation to meet the intent of Regulatory Guide 1.97. Work on these instruments is proceeding toward completion by the required date of January 1, 1981. We will keep you informed of our progress.

2.2.1.a. Shift Supervisors Responsibilities

Full compliance with the requirements of this section has been established by completion of the following actions.

1. A management directive from the Vice President-Production, Operations and Maintenance has been issued to emphasize the duties and responsibilities of the shift supervisor. A copy of the directive is included in Attachment E. This directive will be reissued on an annual basis.

2. The station administrative procedures which delineate the duties and responsibilities of shift supervisors and control room operators have been reviewed and revised to address the concerns expressed in NUREG 0578.

3. Future SRO training programs and retraining programs will include emphasis on the responsibility for safe operation and the management function the shift supervisor is to provide for assuring station safety. Details of this additional training are included in Attachment F.

4. The Director of Nuclear Operations has participated in the review and revision of administrative procedures with specific emphasis on the delegation of miscellaneous duties to personnel other than the Shift Supervisor. In addition, the Director of Nuclear Operations has issued a directive emphasizing that such duties must be delegated so that the Shift Supervisor will not be distracted from his primary responsibility of assuring safe station operation.

2.2.1.b. Shift Technical Advisor

Our responses of October 24, 1979 and November 26, 1979 included our commitments and specific methods for providing the two functions of the Shift Technical Advisor.

Since that time, we have made significant progress in the development of a training program for Shift Technical Advisors. Specific details of the training program are included in Attachment F.

2.2.2.a. Shift and Relief Turnover Procedures

Our response of October 24, 1979 identified several areas in which our existing shift turnover practices and procedures could be improved. Revisions to the turnover procedures to incorporate these improvements have been completed.

2.2.2.a. Control Room Access

In our response of October 24, 1979 we committed to make revisions to existing administrative procedure to reflect more stringent control room access requirements and to establish the authority of the Shift Supervisor to limit access. These revisions are now complete.

2.2.2.b. Onsite Technical Support Center

Following are responses to information items la through lg as requested in the NRC letter of October 30, 1979.

A. Our responses of October 24, 1979 provided the location and description of the onsite Technical Support Center (TSC).

B. Administrative procedures have been developed which delineate the staffing requirements, interfacing between engineering and management support groups, and interfacing between the TSC, the control room and other emergency response centers.

C. Dedicated communications have been provided between the TSC, the control room, the near site emergency operations center and the NRC. Communications consist of either separate telephones, sound powered phones or administratively controlled intraplant telephone.

D. Portable direct radiation and airborne radiation monitors have been provided for the TSC. Appropriate operating and calibration procedures have been developed. In addition, the requirements for and the alarm setpoints of these monitors have been incorporated into the Emergency Plan.

E. Administrative procedures have been developed indicating the software and hardware requirements for the TSC, including manuals, drawings, computer displays and printers and CCTV's.

F. The TSC is part of the Control Room environmental envelope. If, due to some unlikely event, evacuation of the TSC becomes necessary, the accident assessment function would be performed from the control room. Administrative procedures have been developed to cover this situation.

2.2.2.a. (cont'd)

G. In the longer term, it will be necessary to construct a new building in order to meet all requirements for the TSC. We are currently developing a design and schedule for completion. Additional information and an estimated schedule for completion are included in Attachment G.

2.2.2.c. Onsite Operational Support Center

In our response of October 24, 1979 we designated the location of the Onsite Operational Support Center and committed to revise procedures to address the role of the center under emergency situations. The procedure revisions have been completed.

Attachment B

Implementation of Lessons Learned
Short Term Requirements

North Anna Power Station Units 1 and 2

INTRODUCTION

This attachment includes a summary of the status of each of the short term lessons learned requirements for North Anna Units 1 and 2. Also included are summaries of certain test, reviews, and additional information as requested in NRC letters of September 13, 1979 and October 30, 1979.

A meeting with members of the NRC staff was held on November 30, 1979 to discuss the status of implementation of the short term lessons learned requirements on North Anna Unit 2. At that meeting several items of clarification or additional information were requested by the NRC staff. That information, which is applicable to both Units 1 and 2 is included herein.

Items are numbered as in NUREG 0578.

OPERATIONAL STATUS

North Anna Unit 1 is currently completing its first refueling outage with return to power scheduled for approximately January 14, 1979.

North Anna Unit 2 is awaiting the issuance of an operating license.

LESSONS LEARNED SHORT TERM REQUIREMENTS

2.1.1.3.1 Pressurizer Heater Power Supply

Our previous responses of October 24, 1979, October 25, 1979 and November 26, 1979 identified two areas in which actions were required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. Revisions to station emergency and abnormal procedures have been completed to instruct the operator in the use of pressurizer heaters in establishing and maintaining natural circulation.

   A review of emergency diesel loading has determined that procedure revisions were necessary to prevent overloading of the diesel when loading the pressurizer heaters under certain operating conditions. Specifically, during a blackout the pressurizer heaters would trip off. The operator would then be required to reset the UV relay to get the heaters operable on the emergency diesels. When the Unit is operating on natural circulation during this time if a LOCA where to occur which caused a CDA, the diesels could be overloaded since the pressurizer heaters would not trip off as they would during a LOCA simultaneously with a blackout.

   This condition would not occur if pressurizer level were to go <25% during the LOCA, because the pressurizer heaters would trip off from this condition.

To prevent overloading the diesel, the procedures for SI and blackout have been changed to verify the load is <2730 KW prior to resetting the UV relay for the bank of heaters the operator plans to use. The procedures were changed to require the operator to trip the pressurizer heaters, CRDM fans, and the Containment Air Recirc. fans should an S. I. signal occur during natural circulation after a blackout. This tripping will prevent overloading the diesel.

Our initial responses referenced an NSSS vendor study which established that 125 KW of pressurizer heater capacity was needed to support natural circulation in 3-loop plants. A copy of that study is included in Attachment C for your information.

2. A review of the qualification of the pressurizer heater motive and control power has been completed. The existing system meets all qualification requirements as clarified in the NRC letter of October 30, 1979.

2.1.1.3.2 <u>Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators</u>

Our previous responses of October 24, 1979, October 25, 1979, and November 26, 1979 identified two (2) actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. A modification to the nitrogen supply system was required to provide redundant motive power to the PORVs. These modifications are now complete on both units.

2. The qualification review of the motive and control power connections to the emergency buses for the PORVs and block valves has been completed. The existing system meets all requirements as clarified in the NRC letter of October 30, 1979.

In our response of November 26, 1979 we provided a brief description of the motive and control power supplies to the PORVs and block valves. The following information is provided to further clarify the design and operation of the motive and control systems for these valves.

1. The motive power for the PORVs is instrument air backed up by high pressure nitrogen. The instrument air compressor is powered from an emergency bus normally supplied by offsite power. High pressure nitrogen provides additional backup motive power in the event of a loss of instrument air.

The control power for the PORV's is provided by 125 VDC emergency battery buses. The control power to the solenoid valves for the respective PORVs is from separate trains. The PORVs fail closed on a loss of power to the solenoids.

During reactor power operation, each PORV is opened by energizing (manual or high pressure signal) two solenoid valves which admit instrument air or nitrogen to the PORV activator to open the valve. De-energizing the solenoid valves isolates the air supply and bleeds instrument air from the activator causing the PORV to close.

During normal shutdown water solid operation, each PORV is opened on a high pressure signal by energizing a third solenoid valve which admits nitrogen to the PORV activator. De-energizing the solenoid valve bleeds nitrogen from the activator and causes the PORV to close.

2. Each block valve is motor operated and requires power to open and close. The respective block valves are powered from separate 480 VAC emergency buses which are normally supplied from the offsite power supply. Failure of the offsite power supply will cause the motive and control power for the block valves to shift automatically to their respective emergency on site power supplies.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves is through safety grade circuit breakers.

4. Our responses of October 24, and 25, 1979 identified those indiciations of PORV positions which were already available to the operator. Improvements to procedures have been made to aid the operator in the proper interpretation of these indications. Precautions are being added to procedures to instruct the operator that PORV tailpipe temperature can only be used to indicate the initial opening of the valve and not to indicate that they have reseated.

   The reason for this caution is that the temperature greatly lags any actuation of the valve, either opening or closing, and there are other indications which the operator can utilize near the same location of the control board to verify the valve has reclosed. These indications include PRT level, temperature and pressure, the accoustic monitors, and the limit switches on the valves.

2.1.2. Relief and Safety Valves Testing

Vepco, as a participant in the Westinghouse Owners Group, is working with other PWR owners and the Electric Power Research Institute (EPRI) in the development of a program for qualification of relief and safety valves under expected operating conditions involving solid-water and two-phase flow conditions.

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force submitted a description of the test program entitled "Program Plan for the Performance Verification of PWR Safety/ Relief Valves and Systems". We have reviewed the program and consider it to be fully responsive to the requirements presented in NUREG 0578. The EPRI program provides for completion of the essential portions of the test program by July, 1981. Vepco will be participating in the EPRI program to provide program review and to supply plant specific data as required.

2.1.3.a. Direct Indication of Valve Position

Our previous responses of October 24, 1979, October 25, 1979 and November 26, 1979 identified three (3) actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. In our initial response we committed to install acoustic monitors on the reactor coolant system safety valves. Details of the safety valve acoustic monitoring system were included in Attachment A of our November 26, 1979 response. This system is operational on Unit 1 with individual valve indication available at the monitor control panel in the Control Room. However, additional status lights for the vertical board are not yet installed. Completion of the system including all indications will be by January 31, 1980. Installation on Unit 2 will be completed prior to startup.

2. Pressurizer PORVs currently have direct indication derived from limit switches on each valve. Based on the clarification provided in the NRC letter of October 30, 1979, we committed to provide PORV alarms based on the limit switches and to replace the existing limit switches with environmentally qualified limit switches. Since that time we have decided to install acoustic monitors on each PORV. The acoustic monitors for the PORVs are identical to those to be installed on the safety valves. Alarms will be provided in the control room. This installation is complete on Unit 1 and will be completed on Unit 2 prior to startup.

2.1.2 (cont'd.)

3. The acoustic monitoring system used to monitor the PORVs and safety valves will be seismically and environmentally qualified by the vendor. Additional information regarding the qualification of these instruments will be forwarded as soon as it becomes available.

2.1.3.b. Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

Our previous responses of October 24, 1979, October 25, 1979 and November 26, 1979 identified three (3) actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. Changes to emergency procedures have been made to emphasize the need to ensure adequate coolant flow and to ensure that the reactor coolant temperature and pressure are maintained or immediately adjusted to achieve an appropriate margin to saturation. Emergency procedures have been revised to incorporate NSSS vendor generic guidelines for the identification of and recovery from inadequate core cooling conditions.

2. In our response of November 26, 1979, we committed to the installation of a Westinghouse core subcooling monitor. The subcooling meters will be installed on Unit 1 by January 31, 1980 and on Unit 2 prior to startup.

3. In our response of November 26, 1979, we committed to install a reactor vessel water level monitor. The Westinghouse Owners Group, of which we are members, is continuing its review of the available or potential technologies to provide this function. Certain design and functional problems remain unresolved at this time. Accordingly we are not able to provide a conceptual design at this time. We will forward a design description of the reactor vessel water level meter as soon as it is available.

2.1.4. Diverse Containment Isolation

In our initial responses of October 24 and 25, 1979 we indicated that we were in full compliance with the requirements of this section. However, as reported in our Licensee Event Report Number 79-141, we have identified one containment isolation path which reopens under certain circumstances without operator action. The penetration involved is for the condenser air ejector discharge to the containment. The isolation valves, TV-SV-102-1 and 103 are both normally clsoed and are opened upon receipt of a high-high radioactivity level from the main condenser air ejectors.

Normally, non-condensables from the condenser are vented to the atmosphere but a high-high radioactivity level shifts the venting into the containment. A containment isolation signal will close these valves. Upon resetting the containment isolation signal, the valves will reopen if the condenser high-high radioactivity signal is still present. Since there is a very remote possibility of this situation developing (i.e. the probability of both a primary to secondary leak with a LOCA in the containment is very small), and since there is a check valve inside the containment which would still isolate the containment when required, no change to this sytem is necessary.

2.1.5.a. Dedicated H_2 Control Penetrations

As stated in our previous responses of October 24, 1979, October 25, 1979 and November 26, 1979 the existing hydrogen control penetrations meet the requirements of NUREG 0578 as clarified in the NRC letter of October 30, 1979. While the penetration design is satisfactory, some modifications to the hydrogen recombiner system will be required as discussed in our response to section 2.1.5.c.

2.1.5.b. Inerting BWR Containments

This item is not applicable to North Anna Units 1 and 2.

2.1.5.c. Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant

As explained in our previous submittals, North Anna has two (2) Post Accident Hydrogen Recombiners rated at 50 scfm each that are shared between Units 1 and 2.

A review of the operation of the recombiners under accident conditions has been completed in conjunction with the plant shielding review. This review identified the need for modifications to the recombiner system to reduce radiation exposure during system operation. These modifications will involve the addition of shielding and the conversion of certain manual valves to remotely operated valves. Additional information on these modifications is included in our response to Section 2.1.6.a. Since the recombiner system forms a portion of the containment boundary when the recombiner is in operation, the system design was reviewed in order to identify any modifications which would be desirable in order to reduce the potential for leakage. It was determined that the addition of redundant valves in certain locations would be advisable. The proposed modifications will be finalized by March 1, 1980 and will be implemented prior to January 1, 1981.

Procedures for recombiner operation have been revised to include precautions and operational practices which will minimize exposure. These procedures will be revised again to reflect the above listed modifications concurrent with completion of the modifications.

2.1.6.a. Integrity of Systems Outside Containment Likely to Contain Radioactive Materials

Mitigation Systems

The only mitigation systems outside of the containment are the Safety Injection and Outside Recirculation Spray Systems. Immediate testing of these systems in underway, using existing Periodic Tests performed by Operations. These procedures are temporarily modified to require a system walkdown after the pump has pressurized the system piping. All leaks are stopped or a maintenance request is prepared for later repair.

Non-Mitigation Systems

Many non-mitigation systems have been eliminated from consideration because they are either not designed for use under the radiological conditions anticipated after an accident, or their normal operating conditions preclude the need for leak testing (i.e., the process vent system operates continuously at a subatmospheric pressure, thus any leakage will be into the system and not out of the system).

Current plans call for immediate testing, on a priority basis, of the following systems;

- Boron Recovery (de-gas mode only)
- Containment Atmosphere Clean-up
- Liquid Waste Disposal (Sumps to high level waste drain tanks only)
- Gaseous Waste
- Sampling
- Containment Purge

The priority for testing will be based on the likelihood of contamination after an accident.

The containment atmosphere cleanup system has been tested with zero leakage.

Unit 1 leakage test results to date are as follows:

System	Leakage
Low Head Safety Injection	< .1 gpd
1-SI-P-1A	0
1-SI-P-1B	
Containment Atmospheric Cleanup System	0

Outside Recirculation Spray		-0-
	1-RS-P-2A	<.65 gpd
	1-RS-P-2B	
Boron Injection Tank		
	1-SI-TK-2	-0-

Leakage tests on other Unit 1 systems will be completed by January 31, 1980. Additional results will be provided upon completion. Unit 2 tests will be completed prior to startup.

Long Term Leakage Testing

Long term leakage of these systems will be performed at least once every 18 months. Long term testing will be based on a volumetric balance under a constant pressure, i.e., what leaks out will be replaced from a reservoir to maintain constant pressure. A flow instrument located between the reservoir and the system under test will indicate that leak rate. The leakage test procedures will include provisions for leak detection and quantification, corrective action and reporting. Test procedures for the mitigation systems are now complete. Test procedures for non-mitigation systems will be completed by January 31, 1980.

(2.1.6.b) Design Review of Plant Shielding And Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations

We have performed a plant radiation and shielding design review for North Anna Units 1 and 2. This review has evaluated the radiation qualification of vital equipment required to mitigate the effects of a LOCA. In addition, radiation "zone maps" have been produced to be used as administrative guidelines in the control of access and reduction of personnel exposure during the course of the accident. As a result of this review, we have found that additional shielding and plant modifications may be required.

The shielding review has been conducted in two parts: the Mitigation phase and the Recovery phase of an accident. The Mitigation phase, which is assumed to last for six months, considered the radiation levels as the result of operating the recirculation portion of the Safety Injection and Recirculation Spray Systems, along with the Post-Accident Sampling System and Post-Accident Hydrogen Recombiner System. In addition, the review also considered the radiation levels from the auxiliary building sump, the drain lines from the discharge of the auxiliary building and safeguard building sump pumps to the low level liquid waste tank, as well as the containment.

The Recovery phase has been identified as the period six months after the accident when cleanup and plant recovery is undertaken. Based on the TMI-2 experience, the Recovery phase is considered to be a controlled evolution that will be planned and carried out to meet the specific recovery requirements of the particular accident.

Mitigation Phase

The integrated radiation dose calculated during this review is comprised of the 40 year normal dose and a 6 month mitigation phase dose. The safey equipment required to operate during the mitigation phase will be the same as that equipment tabulated for NRC I&E Bulletin 79-01. The source term developed to calculate the 40 year normal operating dose is based on the assumptions in the FSAR, Table 11.1-3. The source terms assumed to calculate the 6 month mitigation phase dose are based on TID-14844 and Regulatory Guide 1.4 as follows:

	Source Term	Basis
Sump Water	0% Noble Gas	Sump Water is Degassed
	50% Halogens	TID-14844
	1% Solid fission products	TID-14844
Primary Coolant Sample	100% Noble Gas	RG 1.4/TID-14844
	50% Halogens	TID-14844
	1% Solid fission products	TID-14844
Containment Atmosphere	100% Noble Gas	RG 1.4/TID-14844
	25% Halogens	RG 1.4
	0% Solid fission products	RG 1.4

The exact course of any accident which has the scope and complexity of that experienced at Three Mile Island - 2 is unpredictable. It is impracticable to determine the dose rate and shielding requirements for every possible location and time duration associated with each possible failure, or incident, which requires personnel access. We have, therefore, developed radiation "zone maps" to be used as post-accident administrative guidelines. These radiation "zone maps" show estimated worst case gamma rates in various areas of the plant as a function of time. The "zone maps" will be used to help the plant operators to plan access and egress routes, help plan maintenance activities, determine stay times, and help evaluate the relative benefits of delaying certain actions to allow for radioactive decay. It should be noted that the gamma dose rates shown on the "zone maps" are based on worst case source terms, but do not consider an airborne source term. Based on the severity of the situation, actual dose rates may be smaller. The "zone maps" therefore are used only as guidelines and actual radiation levels will be determined through actual surveys. In addition, the dose rates listed on the "zone maps" are based on the highest, or one of the highest, dose rates in that area. The dose rate at other locations within that area may be lower.

A review of the radiation "zone maps" has indicated that the main steam valve house and quench spray pump house areas may encounter integrated radiation doses in excess of those used for the NRC I&E Bulletin 79-01 review. The calculation of integrated radiation does to the equipment in these areas due to the NUREG 0578 source terms will be completed by January 31, 1980. All other A/E supplied safety equipment on the above mentioned tabulation is qualified to the post-LOCA mitigation phase dose. The review of the I&E Bulletin 79-01 for NSSS vendor supplied equipment will be completed approximately six weeks after all the supporting test documentation is received. This information is not expected from Westinghouse before January 31, 1980.

Shielding for continuous occupancy areas, per NUREG 0578, must be designed to a maximum dose rate of less than 15 mrem/hr due to radiation from piping and components. The continuous occupancy areas have been defined as the control room, security control center, technical support center, and operational support center. (shop areas and health physics area).

We have evaluated the areas requiring continuous occupancy for dose rates from piping systems as required by NUREG 0578. In addition, we have calculated the dose rate to these areas from the containment dome. Table 1 is a summary of the dose rate to each area from the piping and the containment dome. Based on Table 1, additional shielding is not required for the control room. The other areas will require additional shielding, relocation, or procedure modifications as identified under the Conclusions section to meet the requirements of NUREG-0578.

TABLE 1

		Estimate of Dose Rate from Piping (Mr/Hr)		Estimate of Dose Rate from Containment (Mr/Hr)	
Area	Accident (Hr)	Accident in Unit 1	Accident in Unit 2	Accident in Unit 1	Accident in Unit 2
Control Room	0*+	3	3	110	110
	1	**	**	16	16
Technical Support Center	0*+	46	**	1×10^4	3.7×10^3
	1	36	**	1.6×10^3	590
	2	13	**	960	350
	24	2	**	18	7
Counting Lab/ HP Area	0*	4.4×10^6*	4.4×10^6*	9.2×10^3	1.3×10^4
	1	1.2×10^6*	1.2×10^6*	1.5×10^3	2×10^3
	24	1.6×10^5*	1.6×10^5*	17	21

Shop Area	0*	775	18	6.1×10^3	1.1×10^4
	1	460	16	800	1,700
	8	140	3	120	240
	24	61	**	9	19
Security Control Center	0*+	86	**	1.1×10^3	**
	1	37	**	160	**
	8	9	**	25	**
	24	2	**	2	**

* Major dose from operation of hydrogen recombiner
**Negligible
0+Indicates shortly after accident initiations

NOTE: The dose rates are based on preliminary calculations but we do not expect that the values will change significantly. The calculations will be checked by March 15, 1980.

Recovery Phase

Our evaluation considered the possible operation of the liquid, gaseous, and solid waste systems, letdown and charging portion of the chemical volume and control system, boron recovery system, vent and drain system and the containment purge portion of the atmosphere cleanup system with regards to personnel and equipment irradiation.

Based on the results of our evaluation as presented below, the installed radioactive systems, boron recovery system, containment purge system, letdown and charging portion of the CVCS system will not be used for post-accident cleanup of highly radioactive fluids. It should be noted that the design basis for these systems and associated shielding did not include this use.

1. The majority of electrical equipment in these systems are not qualified to meet the integrated radiation does to which they would be exposed in processing and concentrating the highly radioactive water or gas.

2. The activity levels based on Regulatory Guide 1.4 and TID-14844 of the influent to the liquid waste or boron recovery system is approximately 2×10^3 uci/cc after 6 months of radioactive decay. Thus, the concentrated effluent in the boron recovery or liquid waste evaporator would be so highly radioactive that shielding, processing and handling of the waste by conventional methods may not be possible. In addition, the distillate from the liquid waste or boron recovery evaporators would have a higher activity level, approximately 2 uci/cc, than that of reactor coolant during normal operation. The area radiation dose rate from the concentrated waste and storage tanks would severly limit access to parts of the auxiliary building and hinder operation of both units.

3. Since the radioactive waste systems are common to both Units 1 and 2, the use of these systems for cleanup of waste in the accident unit would preclude the normal use of the radioactive systems for the non-accident unit.

4. There is extensive piping for the recovery systems throughout the auxiliary building. The resulting dose rate from all these systems operating simultaneously would severly limit access for the required operation of both units. Shielding for the recovery system piping and components would be very difficult and in some cases may be impossible to install due to the arrangement of the piping and equipment.

Conclusions

As a result of the plant radiation and shielding review, we have identified additional shielding and plant modifications required to meet the personnel exposure limits and equipment irradiation qualification required by NUREG 0578 and subsequent clarifications:

1. The post-accident hydrogen recombiner vault requires shielding modifications to limit radiation exposure to the operators at the vault while realigning and operating the recombiner, and to reduce the levels in the continuous occupancy areas.

2. Manual valves, located in high radiation zones, which must be operated to line up and operate the post-accident hydrogen recombiner, will be fitted with remote operators or replaced with environmentally qualified remotely operated valves, such as direct-acting solenoid valves or motor-operation valves.

3. In order to adjust the cooldown rate of the nonaccident unit, access to the components cooling water pumps and heat exchangers is required. Additional shielding in the lower level auxiliary building may be required to permit access to that area within 24 hours after an accident.

4. Shielding of portions of the lines added as part of the new post-accident sampling system may be required.

5. Shielding for the Post-Accident Sampling Facility will be required.

6. The drain system for the auxiliary building sump and the safeguards building sump will be modified such that these sumps can be pumped to the affected unit's containment instead of to the high or low level waste tanks. This would eliminate a significant potential source of activity in the basement of the auxiliary building.

7. Shielding will be added in or around the auxiliary building sump to reduce accident level dose rates.

8. Sampling procedures have been modified and temporary shielding employed to limit dose rates at the present sample facility.

9. Additional shielding, area relocation, or procedural modifications are being evaluated to limit radiation dose rates in the technical support center, the operational support center, the counting lab, and the security control center.

10. The letdown line will be modified to divert letdown to the containment sump under post-accident conditions.

11. System modifications to permit interfacing with external process systems designed and shielded after the accident is being evaluated. The external process sytem design would be based on the extent of the accident and would utilize the most current technology available at the time of the accident.

2.1.7.a. Auto Initiation of the Auxiliary Feedwater System

Our responses of October 24, 1979 and October 25, 1979 identified the need for one minor modification to establish full compliance with the requirements of this section. This modification involved the addition of alarms to alert the operator in the event of an auxiliary feed pump discharge pressure control valve malfunction. This modification has been completed on North Anna Units 1 and 2.

2.1.7.b. Auxiliary Feedwater Flow Indication to Steam Generators

Our responses of October 24, 1979, October 25, 1979 and November 26, 1979 identified the need for modifications to establish full compliance with the requirements of this section. These modifications involved the relocation of power supplies to the auxiliary feed flow indicators to meet the diversity requirements. These modifications are now complete on North Anna Units 1 and 2.

2.1.8.a. Improved Post-Accident Sampling Capability

A design and operational review of the reactor coolant and containment sampling systems has been performed for North Anna power station. This review was conducted to determine the capability of promptly obtaining samples under post-accident conditions without incurring a radiation dose to any individual in excess of 3 Rem to the whole body or 18.75 Rem to the extremities. The source terms for these fluid systems are provided in section 2.1.6.b.

This review has determined that there are major difficulties with the existing sampling facility in obtaining representative samples and with performing the required analysis on these highly radioactive samples. Therefore, the sample system will be modified, in

2.1.8.a. (cont'd.)

the long term, to provide the capability to obtain and analyze post-accident samples from the reactor coolant system, containment atmosphere and containment sump.

Short Term Sampling

Special procedures have been developed for monitoring of the sample area prior to entry and for taking samples under accident conditions. Lead shielding and a small sample bomb will be utilized to minimize exposure. These procedural and equipment changes to facilitate sampling in the short term will be completed by January 31, 1980.

Proposed Long Term Design Basis

The shielded sample area will be located in an area of relatively low background activity. It is proposed that all analysis and delution will be performed within the shielded area. However, provisions will be made to remove samples for remote analysis. During post-accident operation the discharge would be collected and processed or routed back to the containment. The sampling facility airborne activity will be controlled by the use of a ventilation system which contains prefilter, charcoal and HEPA filters. A decontamination capability will be provided to reduce personnel exposure when access for sample acquisition, maintenance, or calibration of equipment is required.

The sample system piping and components, up to the second isolation valve, will conform to the QA Category and seismic requirements of the system to which each sample line is connected. The piping and components downstream of the second isolation valve will be designed to Quality Group D as defined in Regulatory Guide 1.26 and nonseismic Category I requirements. All sample lines will be 3/8 inch tubing to limit reactor coolant or containment air leakage due to a failure of the sample line. For all sample lines which may contain large inventories of activity connections outside the containment or sample enclosure will be welded or capable of being leak tested.

Shielding of the new sample lines will be provided where required and will be consistent with the shielding survey conducted in accordance with section 2.1.6.b. The design basis for shielding the sampling area is to limit the dose rate to 500 mrem/hr to an operator when drawing a sample one hour after an accident. Radiation exposure to personnel in transient will not exceed GDC 19 limits. To ensure representative samples and to reduce radiation exposure to personnel and equipment provision will be made to purge and flush highly radioactive fluid back to the containment or to a specially designed waste handling system.

On-line equipment will be utilized to perform much of the analysis. In addition, a backup provision for obtaining and analyzing these samples manually, by means of manipulator arms and a viewing window is being evaluated.

The selection of equipment is not final. Proposals for sampling techniques and equipment are still under review for equipment availability, reliability and qualification. However, the sample system will provide for the samples and analyses as outlined below.

Reactor Coolant Sample

Provisions will be made to draw within the first hour a pressurized reactor coolant sample from either the hot leg of one loop or the cold leg of a different loop. Provisions will exist to analyze this sample within one hour after sample acquisition and to quantify the following constituents: radioisotopes, boron concentration and gross dissolved gas. Provisions will be made to draw a sample from the reactor containment sump and analyze it for radionuclides.

Containment Atomospheric Sample

Provisions will be made to draw, under negative or positive pressure, within one hour after an accident a representative sample of the containment atmosphere. Provision will be made to analyze the sample for radionuclides. Hydrogen concentration in the containment atmosphere will be determined in accordance with the requirements of Section 2.1.9.

The above listed modifications will be completed by January 1, 1981.

2.1.8.b. Increased Range of Radiation Monitors

Our responses of October 24, 1979 and November 26, 1979 identified two (2) areas where actions were necessary to meet all requirements of this section.

1. In our previous responses, we expressed our concerns that fully qualified commercially available monitors may not be obtainable for the extended ranges specified. We still believe this to be the case but we will make every effort to meet the requirements for inplant monitoring capabilities by Janaury 1, 1981.

2. In our November 26, 1979 response we committed to provide interim methods for the quantification of high level releases.

   Shielded area radiation monitors that meet the requirements specified in the NRC letter of October 30, 1979 have been purchased to measure the radiation levels on the main steam safety valves, the process vent and the ventilation vent. The monitors are NRC Industries Model TA-600 detectors (10^{-4} to 10^{+4} R/hr) with check source and Model TA-900 controllers to be installed in the control room to provide a continuous readout. Normal AC power with a backup power supply will be provided.

Procedures are being developed for estimating release rates. The procedures use predetermined calculational methods to convert the measured radiation level to radioactive effluent release rates.

Installation of monitors and finalization of procedures will be completed by January 31, 1980 on Unit 1 and prior to operation on Unit 2.

2.1.8.c. Improved In-Plant Iodine Instrumentation Under Accident Conditions

Our previous responses of October 24, 1979 and November 26, 1979 identified two actions required to establish full compliance with the requirements of this section. The status of these items is as follows:

1. An adequate stock of "silver zeolite" sampling cartridges has been purchased and is in stock at North Anna Power Station. Included in Attachment D is a manuscript entitled "Retention of Noble Gases By Silver Zeolite Iodine Samples" which documents the adequacy of the silver zeolite cartridges under accident conditions.

2. Procedures for the use of silver zeolite cartridges in iodine sampling have been completed.

2.1.9. Analysis of Design and Off-Normal Transients and Accidents

Our responses of October 24, 1979 and November 26, 1979 identified three (3) areas where action was required in order to establish full compliance with the requirements of this section. The status of these items is as follows:

1. Station emergency procedures have been revised to include the new generic guidelines developed by Westinghouse. This includes small break LOCA emergency procedures and procedures for recognition of and recovery from inadequate core cooling conditions. Operator training on the procedure revisions has been completed. Unit 1 procedures have been reviewed by the Bulletins and Orders Task Force. Unit 2 procedures are identical to the Unit 1 procedures with minor exceptions due to differences in mark numbers or minor differences in plant design.

2. In our response of November 26, 1979, we committed to the installation of a reactor vessel head vent. A conceptual design of the proposed reactor vessel head vent system is included in Attachment A, Figure A-1.

   The system is designed such that no single failure will prevent vessel gas venting or prevent venting isolation. The

3/8 inch orifice restricts the flow rate from a pipe break downstream of the orifice to within the makeup capacity of one charging pump.

Procedures and techniques for operation of the vessel vent are still under evaluation pending selection of a reactor vessel level instrument. Procedures for the use of the vessel vent system will be finalized within a short time following the final selection and design of the level instrument and prior to operation following installation of the vent.

We also will install a vent system on the pressurizer. A conceptual design of the proposed pressurizer vent system is included in Attachment A, Figure A-2. The system is designed such that no single failure will prevent pressurizer venting or prevent vent isolation.

Procedures for the use of the pressurizer vent system will rely on the existing pressurizer level instrumentation and will be prepared prior to operation following installation of the vent.

3. In our November 26, 1979 response we committed to install additional containment instrumentation including increased range pressure indication and improved hydrogen indication. Work on these instruments is proceeding toward completion by the required date of January 1, 1981. We will keep you informed of our progress.

2.2.1.a. Shift Supervisors Responsibilities

Full compliance with the requirements of this section has been established by completion of the following actions.

1. A management directive from the Vice President-Production Operations and Maintenance has been issued to emphasize the duties and responsibilities of the shift supervisor. A copy of the directive is included in Attachment E. This directive will be reissued on an annual basis.

2. The station administrative procedures which delineate the duties and responsibilities of shift supervisors and control room operators have been reviewed and revised to address the concerns expressed in NUREG 0578.

3. Future SRO training programs and retraining programs will include emphasis on the responsibility for safe operation

and the management function the shift supervisor is to provide for assuring station safety. Details of this additional training are included in Attachment F.

4. The Director of Nuclear Operations has fully participated in the review and revision of administrative procedures with specific emphasis on the delegation of miscellaneous duties to personnel other than the Shift Supervisor.

2.2.1.b. Shift Technical Advisor

Our responses of October 24, 1979, October 25, 1979 and November 26, 1979 included our commitments and specific methods for providing the two functions of the Shift Technical Advisor.

Since that time, we have made significant progress in the development of a training program for Shift Technical Advisors. Specific details of the training program are included in Attachment F.

2.2.2.a. Shift and Relief Turnover Procedures

Our responses of October 24, 1979 and October 25, 1979 identified several areas in which our existing shift turnover practices and procedures could be improved. Revisions to the turnover procedures to incorporate these improvements have been completed.

2.2.2.b. Control Room Access

In our responses of October 24, 1979 and October 25, 1979 we committed to make revisions to existing administrative procedure to reflect more stringent control room access requirements and to establish the authority of the Shift Supervisor to limit access. The revisions are now complete.

2.2.2.c. Onsite Technical Support Center

Following are responses to information item 1a through 1g as requested in the NRC letter of October 30, 1979.

A. A temporary Onsite Technical Support Center (OTSC) has been established in the Records Building, which is a two-story building inside the Protected Area security fence adjacent to the main facility. The first level of this building contains the record processing and storage areas, with records and drawing describing the as-build condition of the facility available in the records fileroom (see E below). The second level contains an area which has been designated as the assembly area for technical support personnel during an emergency. Additional communications equipment has been installed in this area to allow communications between the OTSC and the Control

Room, the Onsite Operations Support Center, the Offsite Emergency Support Center, the NRC, Vepco headquarters, etc. (see C below). A typewriter, paralleled with the Control Room computer typewriter, has been installed in this area to allow direct display of plant parameters necessary for assessment by technical support personnel (see E below).

B. The existing Emergency Plan Implementing Procedures (EPIP's) have been revised to cover engineering and management support and staffing of the OTSC.

C. Dedicated communications lines have been installed to allow communications between the OTSC and the following:

1) Control Room
2) Offsite Emergency Operations Center
3) NRC Emergency Response Center

Additional communications lines allow communications with Vepco headquarters, NRC Region II headquarters, Vepco System Operator, and the Westinghouse Emergency Response Center, as well as various locations within the station.

These communications systems employee redundant networks such as the NRC Health Physics Network, the C & P Telephone System, the Vepco microwave system, and the station PBX system. The use of these various systems provides redundancy of communications links.

D. Procedures have been revised to provide for the installation of portable radiation and airborne radioactivity monitoring equipment in the OTSC when it is activated.

E. A typewriter paralleled with the Unit 1 utility typewriter in the Control Room has been installed in the OTSC. This provides direct display of plant parameters necessary for evaluation and assessment. The records fileroom located in the OTSC contains the technical information such as general arrangement drawings, piping isometrics, electrical drawings, system specifications, and plant procedures that might be needed during the emergency condition.

F. Procedures have been revised to cover performance of the accident assessment function from the Control Room should the OTSC become uninhabitable.

G. In the longer term, it will be necessary to construct a new building in order to meet all requirements for the TSC. We are currently developing a design and schedule for completion. Additional information and an estimated schedule for completion are included in Attachment G.

2.2.2.d. Onsite Operational Support Center

In our responses of October 24, 1979 and October 25, 1979 we designated the location of the Onsite Operational Support Center and committed to revise procedures to address the role of the center under emergency situations. The procedure revisions have been completed.

Attachment C

<u>Emergency Power Supply Requirements for Pressurizer Heaters</u>

EMERGENCY POWER SUPPLY REQUIREMENTS FOR THE PRESSURIZER HEATERS

The NRC concern deals with the need to use pressurizer heaters following a loss of offsite power to maintain reactor coolant system pressure, thus keeping the primary coolant subcooled and providing core cooling via natural circulation. An alternate means of core cooling is through use of the ECCS; the NRC desires to reduce the frequency of challenges to the ECCS by assuring the availability of pressurizer heaters following a loss of offsite power.

A study was performed to determine the heater capacity required to maintain RCS pressure with a loss of offsite power. The study established the minimum capacity and the time frame when emergency power supplies must be available.

Pressurizer heat losses can be divided into two basic components: 1) losses through the pressurizer walls, insulation, supports, connections, etc., and 2) losses due to continuous spray flow. Spray flow is driven to the top of the pressurizer by reactor coolant pump head; without offsite power, the pumps will coast down and no spray flow will be supplied. Thus, without offsite power only heat losses through insulation, supports, etc. must be offset by heaters.

A review of several pressurizer heat loss calculations has resulted in the following minimum heater requirements without offsite power:

Pressurizer Size, Ft^3	Heater Capacity, Kw
1000	100
1400	125
1800	150

These capacities will conservatively cover heat losses from the pressurizer at or below normal operating pressure with no allowance for continuous spray.

A transient analysis of the loss of offsite power event was performed to establish the time frame when heaters would be required to maintain RCS subcooling. The analysis was performed for a four loop plant with an 1800 ft^3 pressurizer. A heat loss from the pressurizer of 100 Kw was assumed with no credit taken for the heat capacity of the pressurizer metal. Decay heat was removed via the steam generator safety valves which results in highest RCS temperature and least margin to subcooling (no credit for steam dump system). Continued operation of charging-letdown-level control was assumed since this results in a decrease in pressurizer level to no-load value and consequentially a decrease in system pressure, again giving least margin to subcooling. This also gives the least mass in the pressurizer and the least heat capacity, resulting in a more rapid decrease in pressure due to heat losses. Results of this analysis are shown in Figures 1A and 1B. After pressurizer level has stabilized, heat losses cause a reduction in system pressure at about 90 psi/hour with T_{sat} dropping at 7°F/hour. Loss of subcooling would then occur between 5 and 6 hours. Heater input at any time of 150 Kw as specified above would more than offset the heat loss and allow system pressure to be stabilized at any desired value.

Ability to supply Emergency power to the heaters within four hours will prevent loss of subcooling in the primary following a loss of offsite power.

FIGURE 1A.
Loss of Offsite Power, 100 Kw Heat Loss
Four Loop Plant, 1800 ft³ Pressurizer

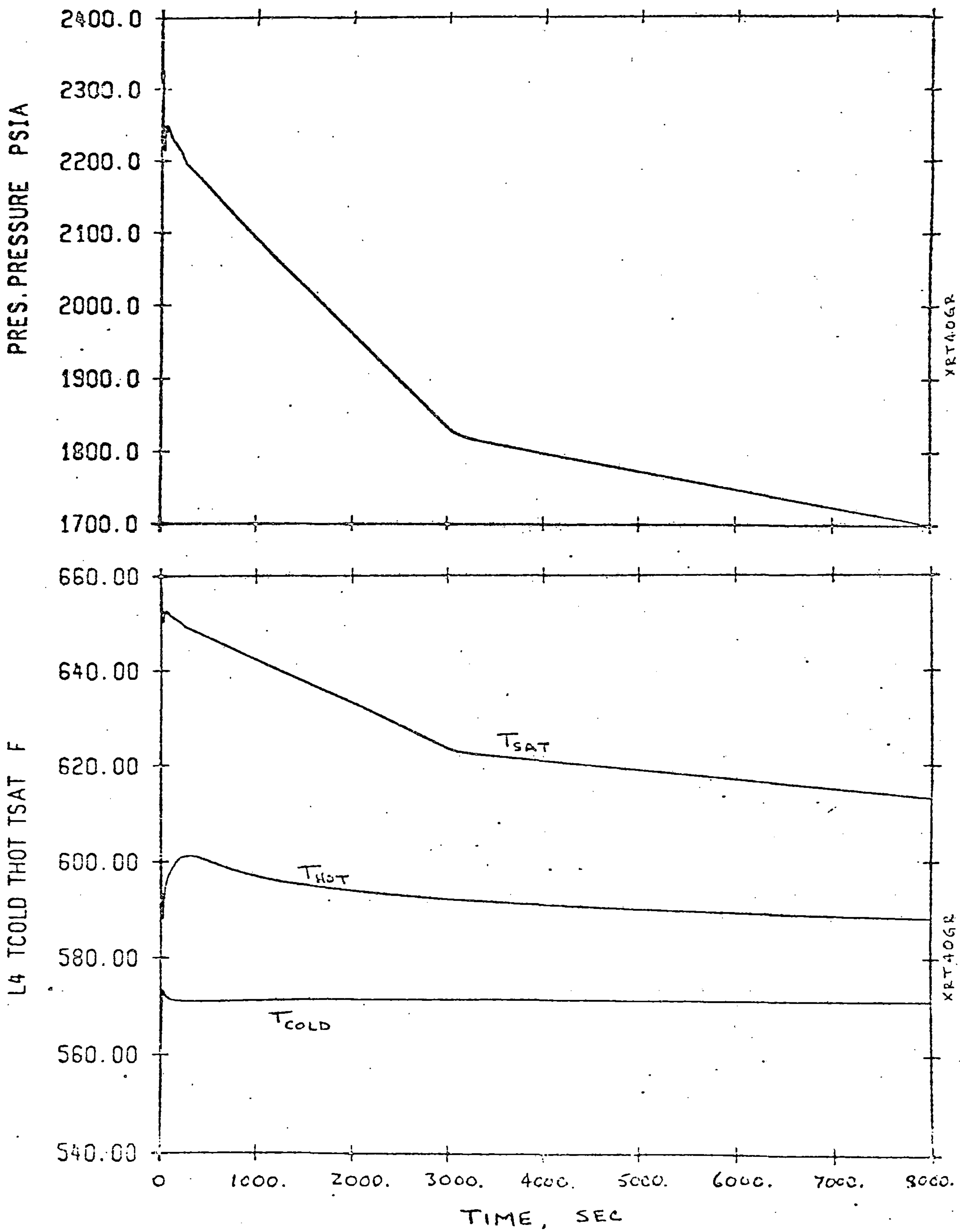
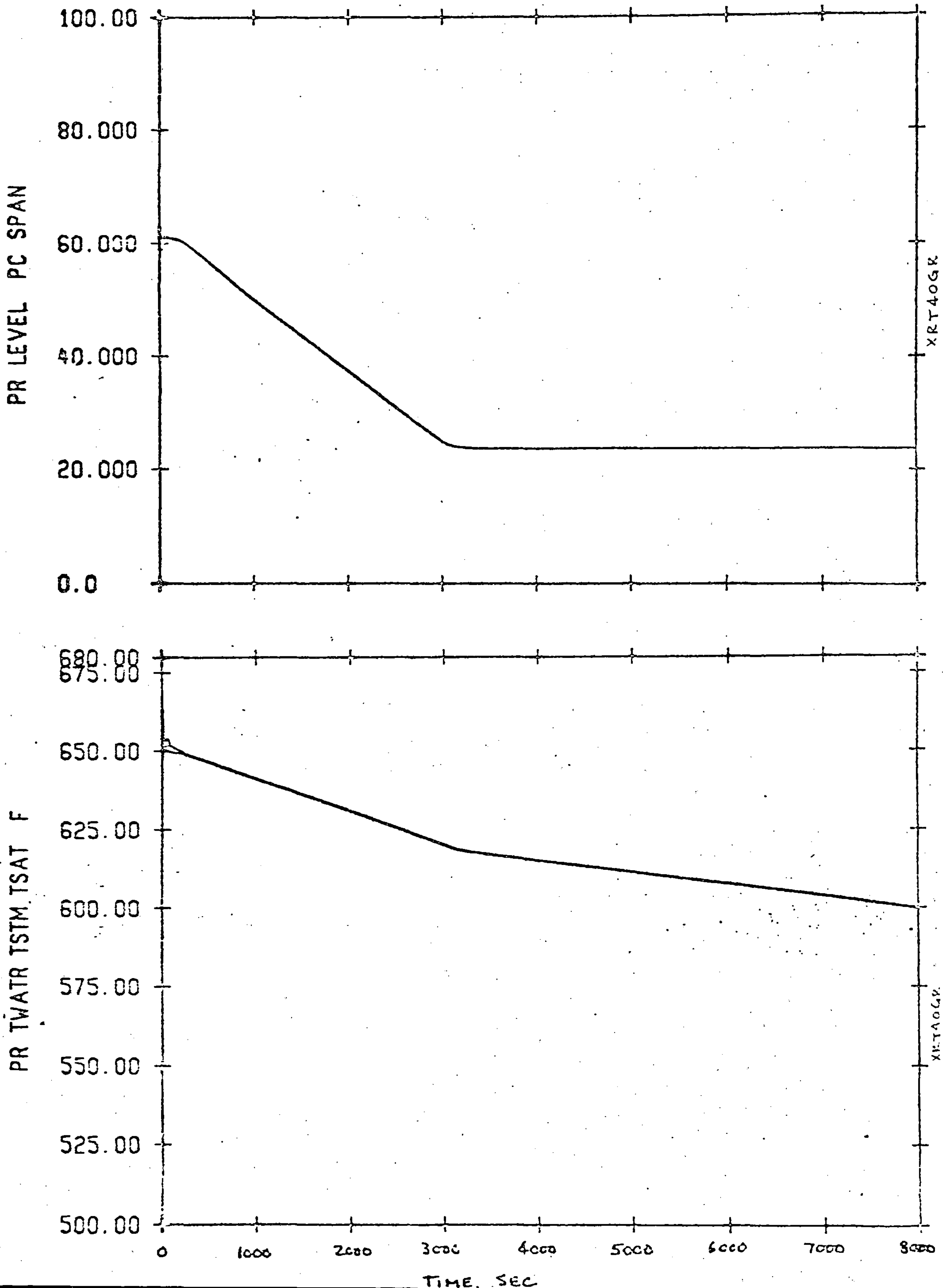


FIGURE 1B.


100.00
80.000
60.000
40.000
20.000
0.0
PR LEVEL PC SPAN
XRT40GR
680.00
675.00
650.00
625.00
600.00
575.00
550.00
525.00
500.00
PR TWATR TSTM TSAT F
XRT40GR
0
1000
2000
3000
4000
5000
6000
7000
8000
TIME, SEC

Appendix D

"Retention of Noble Gases By Silver Zeolite Iodine Samples"

RETENTION OF NOBLE GASES BY SILVER ZEOLITE IODINE SAMPLERS

James E. Cline

Science Applications, Inc.
Rockville, MD

SUMMARY:

Analyses have been made of the retention of radioactive xenon by sample cartridges loaded with silver zeolite. No evidence of any xenon retention was found by these measurements. A maximum value of 1/15 000 was measured for the retention of xenon in silver zeolite relative to that in TEDA-impregnated charcoal in cartridges through which air containing the gas was drawn for one hour. This represented a retention efficiency for xenon in silver zeolite of less than 5×10^{-6} percent.

Manuscript submitted for publication as a letter to the *Journal of Health Physics*.

1.0 BACKGROUND

The accident at the Three Mile Island nuclear station resulted in airborne radioactive xenon and iodine concentrations in the plant reactor containment and auxiliary buildings. Measurements of the radioiodine concentrations, normally done by drawing air through charcoal cartridges, were severely complicated by the presence of the radioxenon retained by the charcoal. This resulted in:

1. On-line iodine monitors were paralyzed due to the high counting rate from the retained xenon, resulting in the untimely loss of these direct reading instruments,
2. Readings from iodine survey instruments were misinterpreted because of pulse pile-up problems due to the high counting rate from the xenon,
3. The presence of the xenon in the sample cartridges meant that the off-line analyses by Ge(Li) spectrometry had to be done with higher source-detector distances (many systems were not calibrated for these large distances) with resulting lower sensitivity for the desired iodine activity and longer analysis times.

Hence, it would be desirable to use an alternate iodine absorbing material that does not retain noble gases for emergencies such as occurred at Three Mile Island. Silver zeolite can serve as such a material and was used in many samplings at the Three-Mile-Island facility to measure radioiodine in the presence of considerably higher concentrations of radioactive xenon.

2.0 MEASUREMENTS AND ANALYSES

RADeCO charcoal (model CP-100) and silver zeolite (model GY-130) cartridges were used in the measurements. Characteristics of the material in these cartridges are as follows:

<u>Charcoal</u>: 40-50 mesh TEDA-impregnated (5%) charcoal

<u>Silver Zeolite</u>: 30-50 mesh AgX zeolite (fully exchanged) (for some measurements, AgY material was used).

Air was pulled through the two cartridges in series. After the particulate filter, air was drawn through the silver zeolite followed by charcoal. Sample flow rates were approximately 28 liters per minute (1 SCFM) through the 5.4-cm diameter cartridges. The residence time in each cartridge was about 120 milliseconds. Sampling times for these measurments varied from 10 minutes to 120 minutes. No differences were seen in results that could be attributed to the length of the sampling times. Samples were <u>not</u> purged with clean air following their collection. Similarly, no differences were seen between the results from the measurements using AgX and those that used AgY.

Exposed cartridges were analyzed with a calibrated Ge(Li) gamma-ray spectrometer. The efficiency calibration of this unit was made and verified using standards from NBS. Cartridges were carefully positioned in the spectrometer and in each experiment, the silver zeolite and the charcoal cartridges were counted in precisely the same geometry so the results could easily be compared. Emission rates (gammas per sec) were obtained for the 80-, 284-, 364-, 637- and 723-keV gamma rays from the data analyses for each cartridge. An 80-keV gamma ray is emitted in the decays of both ^{131}I and ^{133}Xe; hence, it is necessary to partition the observed intensity of this gamma ray into its two components in order to determine the amount of ^{133}Xe retained by the cartridge. This can best be done by determining the amount of ^{131}I on the cartridge through analyses that use the other four gamma rays and then determining the intensity of the 80-keV transition that can be ascribed to the decay of ^{131}I. A best value for the ^{131}I activity on each cartridge was obtained through a weighted average of activities as independently determined from the other four gamma-ray energy groups. The weighting factor for each value was the reciprocal of the statistical uncertainty in the observed intensity of that gamma ray. The expected intensity of the 80-keV transition resulting from the decay of ^{131}I was then calculated using this best value and the absolute decay intensity for the 80-keV emission.

3.0 RESULTS AND DISCUSSION

Table 1 lists the results from the analyses of two typical measurements. Results are listed from both tests for the particulate filter and the silver zeolite and charcoal cartridges. The particulate filter is included to demonstrate the ability of the analysis technique to partition properly the measured 80-keV transition. The particulate filter should retain little xenon; only the binder in the glass fiber material might be expected to retain any amount. In the two samples shown in the table, the 80-keV activity on the particulate filter can be accounted for, within statistics, as being from the ^{131}I on the filter. This indicates that the branching intensity for the 80-keV transition in the decay of ^{131}I and that the relative efficiency for the detection of the 80-keV gamma ray are both adequately known.

Within the experimental statistics of the measurements, no 80-keV radiation was emitted from the silver zeolite cartridges that could not be assigned to the decay of the ^{131}I in the cartridge. Hence, only an upper limit can be placed on the amount of ^{133}Xe retained in these measurements by the silver zeolite. This limit can be expressed as a minimum in the ratio of ^{133}Xe activities on the silver zeolite and on the charcoal. From the data in Table 1, this ratio can be seen to be:

$$\frac{^{133}\text{Xe retention on AgX}}{^{133}\text{Xe retention on charcoal}} < \frac{1}{15000} .$$

Simultaneously with sample 1, a grab sample of gaseous activity was taken and analyzed for the concentration of ^{133}Xe. The results, when compared to the analyses of cartridge activities gave a retention efficiency of charcoal for xenon of 0.03% for the 17-minute sampling time. This value, then, gives a maximum retention of silver zeolite for xenon of 2×10^{-6}%.

Table 1. Analyses of two typical samples for ^{133}Xe Retention on Silver Zeolite

		Gamma-Ray Energy (keV)				
Cartridge		80	284	364	637	723
			SAMPLE I			
Particulate Filter	Counts/sec	12.8±1.4	34.9±1.2	500.7±4.0	41.7±1.6	11.2±0.9
	μCi	--	0.0156±0.0005	0.0167±0.0001	0.0156±0.0006	0.0168±0.0014
	c/s (^{131}I)	11.8	--	--	--	--
AgX	Counts/sec	556. ±11	1789.±13	24553.±49	2048.±16	516.4±9.3
	μCi	--	0.800±0.006	0.817±0.002	0.764±0.006	0.776±0.014
	c/s (^{131}I)	568	--	--	--	--
Charcoal	Counts/sec	16675±17	6.4±0.6	89.6±0.2	6.7±0.7	1.9±0.5
	μCi	--	0.0029±0.0003	0.0030±0.0001	0.0025±0.0003	0.0029±0.0007
	c/s (^{131}I)	2.9	--	--	--	--
			SAMPLE II			
Particulate Filter	Counts/sec	9.9±1.1	28.5±1.1	382.3±3.8	31.4±1.4	9.53±0.82
	μCi	--	0.0128±0.0005	0.0128±0.0001	0.0117±0.0005	0.0143±0.0012
	c/s (^{131}I)	9.1	--	--	--	--
AgX	Counts/sec	52.1±2.0	133.4±2.3	1874.0±7.5	160.3±2.9	40.1±1.7
	μCi	--	0.0596±0.0010	0.0626±0.0003	0.0600±0.0011	0.0629±0.0026
	c/s (^{131}I)	48.3	--	--	--	--
Charcoal	Counts/sec	4100.4±8.2	--	2.8±0.4	--	--
	μCi	--		(9±1) x 10^{-5}		
	c/s (^{131}I)	0.01	--	--	--	--

Appendix E

Management Directive Regarding Shift Supervisor Responsibilities

Vepco

VIRGINIA ELECTRIC AND POWER COMPANY, RICHMOND, VIRGINIA 23261

December 26, 1979

TO: All Nuclear Station Personnel

Subject: SHIFT SUPERVISOR RESPONSIBILITIES

It is essential that all employees understand their responsibilities and the responsibilities and authorities of others in the operation of our nuclear facilities. The station organization, job descriptions of key personnel and lines of communication and authority are explained in the station Administrative Procedures. The responsibilities and authorities of station personnel under emergency situations are defined in the station Emergency Plan as explained in employee orientation and in the periodic reinstruction in the plan.

It is imperative that all employees recognize the responsibilities and authority of the Shift Supervisor under normal and emergency conditions. The Shift Supervisor has primary management responsibility for the safe operation of the plant under all conditions. The role of the Shift Supervisor is to be cognizant of station conditions and to direct operations. He has full authority to direct operations in a manner which assures safe operation of the station.

The Shift Supervisor will not become involved in any single operation in times of emergency when multiple operations are required in the control room. The Shift Supervisor can be relieved only by certain designated individuals who hold Senior Reactor Operator licenses. No other personnel can relieve the Shift Supervisor or direct licensed operators. The Shift Supervisor has the authority to limit access to the control room or to expel non-essential personnel from the control room at any time. While the Shift Supervisor may call upon members of the station or corporate staff for information or advice, the ultimate decision making responsibility for station operations under normal or emergency operations rests with the Shift Supervisor.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

Appendix F

Training of Shift Technical Advisors
and
Shift Supervisors

Shift Supervisors and Shift Technical Advisor Training
Surry Power Station Units 1 and 2
North Anna Power Station Units 1 and 2

Introduction

In our responses to NUREG 0578, dated October 24, 1979, and October 25, 1979 we made the following commitments to provide improved on-shift accident assessment capability.

1. Provide an additional SRO on each shift
2. Provide a Shift Technical Advisor on each shift. The Shift Technical Advisor will be one of a group of SRO's who will receive special training during 1980 to qualify them as STA's.
3. Long term upgrading of SRO training so that all SRO's will be qualified as STA's.

In addition, in response to section 2.2.1.a. of NUREG 0578 we committed to improve SRO training programs to provide greater emphasis on and reinforcement of the responsibility for safe operation and of the management function the shift supervisor is to provide.

Three new training programs have been developed as required to meet these commitments:

1. Short Term STA Training

   This program will be conducted during 1980 to train the designated STA's and will provide a small number of fully qualified STA's by January 1, 1981.

2. Long Term Shift Supervisors/STA Training

   This is a long term program which will provide an extensive upgrading of SRO and shift supervisor training and which will provide shift supervisors with a Bachelor of Science Degree or equivalent.

3. SRO Supervisory Skills Training
   The program is being developed to provide all SRO candidates with specific training in managerial skills, including communication, leadership, decision making and problem solving.

The short term STA training program will be conducted during 1980 and possibly beyond in order to provide the requisite number of qualified STA's to meet short term staffing requirements. The Long Term Shift Supervisor/STA Training and the SRO Supervisory Skills Training programs will be developed and implemented beginning in 1980. As the two long term programs become fully implemented the Short Term STA Training Program will be discontinued. While the two long term programs are currently being developed independently, they will ultimately be combined in a single comprehensive program. Following are additional details on each of these training programs.

1. Short Term STA Training Program

We have developed a tentative short term STA training program and are currently pursuing arrangements with several consultants for provision of portions of the program. In addition to our use of established training consultants from within the industry we are actively seeking the cooperation of several universities in order to provide college level instruction as part of the program. Since these arrangements with outside parties are incomplete, a more specific outline and schedule is not available at this time. The following is a tentative program outline.

Academics (8 weeks)

I. Mathematics (1 week)
   a. Basic Operations
   b. Algebraic Operations
   c. Special Techniques
   d. Logarithms
   e. Geometrical Applications
   f. Differential Calculus
   g. Integral Calculus
   h. Applied Differential Equations

II. Physics (2 weeks)
   a. Classical
   b. Nuclear Physics
   c. Nuclear Engineering
   d. Reactor Operations

III. Thermodynamics (2 weeks)
   a. Fundamental Concepts (includes First Law)
   b. Second Law of Thermodynamics
   c. Thermodynamics of Steam
   d. Properties of Gas

IV. Fluid Flow (1 week)
   a. Fluid Flow Basics
   b. Fluid Statics
   c. Mathematical Models and Equations of Fluid Motion
   d. Flow in Pipes and Ducts
   e. Two Phase Flow
   f. Critical Flow
   g. Pump Characteristics

V. Heat Transfer (1 week)
  a. Heat Transfer Modes
  b. Material Properties of Importance in Heat Transfer
  c. Heat Conduction in One Dimenson
  d. Convective Heat Transfer
  e. Boiling Heat Transfer
  f. Radiation Heat Transfer
  g. Heat Exchangers
  h. Heat Transfer in Reactor Core

VI. Miscellaneous (1 week)
  a. Instrumentation and Controls
    1. Fundamentals
    2. Transducers
    3. Control Elements
  b. Chemistry
    1. Fundamentals
    2. Effects on Radiation on Water
    3. Fission Products in Reactor Coolant
  c. Materials
    1. Reactor Construction Materials
    2. Mechanical Properties of Materials
  d. Structural Analysis
    1. Fundamentals
    2. Pressure Vessel Stress

<u>Design Review</u> (2 weeks)

I. Design Consideration
II. Reactor Coolant System
III. Secondary System
IV. Control Systems
V. Reactor Protection System
VI. Auxiliary Systems

<u>Systems Dynamic Behavior</u> (5 weeks)

I. Transient Analysis
II. Techniques for Transient Identification

2. <u>Long Term Shift Supervisor/STA Training</u>

In the longer term our training programs for SRO's and Shift Supervisors will be upgraded to provide a far greater scope and depth of technical and managerial training.

We are currently working with several universities to develop a program whereby our SRO's/Shift Supervisors will receive college level technical training leading to a Bachelor of Science degree. This program would provide extensive coverage of all topics listed above for the short term training program plus additional college level courses in general and technical electives.

We will provide additional details on the longer term program as soon as arrangements are finalized.

3. SRO Supervisory Skills Training

We have developed a tentative SRO Supervisory Skills Training program and are in the process of finalizing arrangement with consultants to provide this training. Following is a tentative course outline.

I. Communication (2 days)
   a. Channels of Communication
   b. Effective Communication
   c. Perception and Frame of Reference
   d. Data Gathering
   e. The Communication Model
   f. Types of Communication
   g. Problems of Communication
   h. Listening

II. Motivation (2 days)
   a. Managerial Motivation Approaches
   b. Elements of Employee Satisfaction
   c. Motivational Links to Positive Employee Relations
   d. Behaviorial Theories of Motivation

III. Leadership (2 days)
   a. Understanding Individual Approaches to Leadership
   b. Responsive Supervision
   c. Effective Leadership
   d. Approaches to Leadership
   e. Concepts of Leadership

IV. Problem Solving and Decision Making (3 days)
   a. The Nature of Problem Solving
   b. Problem Solving Toxonomy
   c. Problem Solving as a Type of Decision Making
   d. Methods of Problem Solving
   e. Types of Problems
   f. Decision Making in the Real World
   g. Planning Decision Making
   h. Management Decision Process
   i. Basics of Decision Making
   j. Decision Criteria
   k. Approaches to Decision Making
   l. Group Decision Making

V. Program Summary (1 day)
   a. Program Review and Analysis
   b. Exercises Linked to each Module
   c. Evaluation of Supervisor Data Retention

Attachment G

Onsite Technical Support Centers

Surry Power Station Units 1 and 2
North Anna Power Station Units 1 and 2

Onsite Technical Support Centers

Temporary onsite Technical Support Centers (TSCs) have been established at Surry and North Anna Power Stations. In the longer term it will be necessary to construct additional facilities in order to provide a TSC which meets all the requirements established in NUREG 0578.

Design work is in progress for TSC's at both Surry and North Anna. Exact designs are incomplete at this time; however, each TSC will be designed and constructed to meet all requirements of NUREG 0578 Section 2.2.2.b.

Following is preliminary information on TSC design.

1. Each station will add or convert building space to provide a TSC of approximately 4500 to 5000 square feet.

2. The TSC will include the following work spaces.

   a. Operations center: Approximately 750 square feet will be provided for equipment for the receipt, monitoring, and transmission of data on critical plant parameters and operating conditions and other communications equipment.
   b. Technical Support Work Area: Adjacent to the Operations Center will be additional work space for technical support personnel. This will be approximately 600 square feet.
   c. Records Area: A small records and reference room will be provided which will house the necessary drawings and records for use by the technical support personnel.
   d. Conference Rooms and Office Space: Offices will be provided for technical support management and for operations support management.
   e. Personnel Amenities: The TSC will include shower and locker room facilities, and lunch room facilities.
   f. Personnel Access: Entrances for use under accident conditions will include appropriate personnel monitoring equipment.

As discussed above, the Operations Center, within the TSC, will include communications and data link equipment. Since many questions regarding data transmission to other locations, extent of data required in the TSCs, and types of display required are not yet resolved we have not firmly established the exact specifications of equipment to be included in the TSC in the long term. We encourage and will participate in industry and NRC efforts toward establishing firm parameter monitoring and data transmission requirements. We are currently planning the installation of data links to existing plant computers and the use of closed circuit TV to monitor control room activities and displays. The use of CCTVs is an interim solution which is being used because of the short term availability of equipment. Additional equipment will be provided as requirements are finalized and equipment is available.

The preliminary schedule for design and construction of the TSCs is as follows:

1.	Finalize Design Criteria	February 1, 1980
2.	Begin Construction	May 15, 1980
3.	Complete Structural Design	June 15, 1980
4.	Complete Construction	December 15, 1980

We will provide additional information as it is available and will be glad to meet with the staff to discuss the TSCs.