POWER AUTHORITY OF THE STATE OF NEW YORK

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January 8, 1980
IPN-80-3

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

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Short Term Requirements of TMI Lessons Learned

Dear Sir:

Enclosed, as Attachment 1 of this letter, is the Power Authority of the State of New York response to the NRC letter of clarification on TMI Lessons Learned dated October 30, 1979 for the Indian Point 3 Nuclear Power Plant.

As you will note during your review of the attached responses, all short term items will be completed prior to returning this Unit to service.

Very truly yours,

PJ Early

Paul J. Early
Assistant Chief Engineer-
Projects

Att.
cc: Mr. T. Rebelowski w/ att

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ATTACHMENT 1

THREE MILE ISLAND LESSONS LEARNED
STATUS AND COMMITMENTS - 1/1/80

Section 2.1.1

Pressurizer Heaters

The pressurizer heaters, one control group and their back up groups, are all powered from a separate safety related 480 volt bus. The heaters will be stripped from that bus during a safety injection (SI) signal or loss of bus voltage. After resetting S.I., their power supply can be re-established manually from the control room. The appropriate operating procedure highlights this point. The 480 volt safety buses and diesels are sized for full safeguard loads during a DBA LOCA. For smaller breaks, all the safeguards equipment is not required and redundant components would be secured thus providing the needed capacity for the energizing of the pressurizer heaters. The pressurizer heaters have safety related circuit breakers for main and control power.

A recent study performed by Westinghouse under the auspices of the TMI-2 Owner's Group to determine the ability to maintain subcooling conditions for a 4-loop plant with an 1800 cubic foot pressurizer indicates that loss of subcooling would occur between five and six hours with no pressurizer heaters. Any one group of heaters at any time during this interval with a capacity of 150 KW would more than offset the heat losses from the pressurizer and allow the system pressures to be stabilized to maintain subcooling conditions.

Training to incorporate the above will be completed before the unit is returned to service

Pressurizer Level and Relief Block Valves

The pressurizer PORV's solenoids are powered from the battery supplied 125v DC system. The valves use nitrogen as a motive force via solenoids. The nitrogen is stored in accumulators and tanks which are independent of offsite power. The block valves are powered from the emergency diesels in the event of loss of outside power. This changeover is accomplished automatically. The design of the PORV's and block valves are such that they can be opened in addition to being closed in the event of a loss of offsite power. The existing PORV's have been upgraded to include environmentally qualified components. The design of the PORV's are such that they rely on nitrogen rather than instrument air.

Three of the pressurizer level instrumentation channels are powered from a vital instrument bus which is fed from a battery inverter system for reliability. The battery charger can be powered from the diesel generators when offsite power is not available. The fourth is fed from a vital instrument bus which derives its power from a 480V Moter Control Center (MCC) and a constant voltage transformer. The MCC can also be tied directly to the emergency diesel from the control room.

Section 2.1.2.

Performance Testing for PWR Relief and Safety Valves

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force submitted a program titled "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems", dated December 13, 1979.

The Power Authority of the State of New York considers the program to be responsive to the requirements presented in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations" dated July, 1979. Item 2.1.2. which recommended in part, "Commit to provide performance verification by full scale prototypical testing for all relief and safety valves. Test conditions shall include two-phase slug flow and subcooled liquid flow calculated to occur for design basis transients and accidents."

The EPRI Program Plan provides for a completion of the essential portions of the test program by July, 1981. The Power Authority of the State of New York will be participating in the EPRI program to provide program review and to supply plant specific data as required.

Section 2.1.3.a

Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs

All of the pressurizer power operated relief valves (PORV) and their associated motor-operated block valves (MOV) have positive position indication in the control room. In addition, downstream temperature indication, Pressurizer Relief Tank temperature, level and pressure indication will also indirectly indicate valve position.

While we firmly believe that the existing instrumentation is sufficient for detecting leakage from the code safety relief valves, an acoustic monitoring system for position indication of these valves will be installed. A description of the Acoustical Monitoring System follows:

An Acoustical Monitoring System manufactured by "Technology for Energy Corporation" (TEC) of Knoxville, Tenn. is to be installed in order to monitor RCS Pressurizer Safety/ Relief Valve operation. This modification is being implemented in order to comply with NRC requirements under NUREG-0578 (Item 2.1.3.a.) for pressurizer safety and relief valve position or reliable flow indication.

This system, which is designated as the "TEC Model 914 Valve Flow Monitor Module", is designed to be used with a remote charge converter (i.e., TEC Model 500). A piezoelectric accelerometer mounted on the discharge side of the valve to be monitored converts acceleration (i.e., g's) to charge, which the charge converter then converts to voltage. The TEC 914 processes this voltage signal and indicates the relative valve flow based on the phenomenon of acoustical accelerations set up by flow through the discharge piping near the valve.

For each valve being monitored, a lighted bar graph on the TEC 914 module senses RMS output voltage and indicates relative valve flow. The bar graph is labeled in relative valve flow with 1.0 being full flow and is located in the control room. Discrete valve flow indications are .01, .04, .09, .16, .25, .36, .49, .64, .81, and 1.0. The annunciator alarm set points for each valve can be adjusted by adjusting strap wires on the P.C. boards. The annunciator is activated when one or more valves reach their alarm set point.

According to NUREG-0578, this equipment is to be safety-grade and have environmental qualification consistent with the accident conditions to be monitored. While this system is currently not qualified to Class 1E standards, previous and ongoing qualification tests indicate that the system can be qualified to these standards. The necessary documentation for post qualification of this system will be provided by TEC in the near future. This same situation prevails for similar systems being supplied by other manufacturers.

In addition, an alarm system will be added in the control room to the existing Pressurizer Power Operated Relief Valves as requested by the NRC staff. The components for the alarm and acoustic monitoring system have been received and installation will be completed prior to the unit being returned to service.

Section 2.1.3.b

Instrumentation for Detection of Inadequate Core Cooling in PWR's

A. Procedures and Description of Existing Instrumentation

The Westinghouse Owner's Group, of which the Power Authority of the State of New York is a member, has performed analyses as required by Item 2.1.9 to study the effect of inadequate core cooling. These analyses were provided to the NRC "Bulletins and Orders Task Force" for review on October 31, 1979. As part of the submittal made by the Owners' Group, an "Instruction to Restore Core Cooling during a Small LOCA" was included. This instruction provides the basis for procedure changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. The Power Authority of the State of New York will incorporate the key considerations of this instruction into our LOCA procedures, and will provide training to the operators in this area prior to returning the unit to service.

B. A subcooling meter has been designed and components procured. It will consist of selector switches to choose one out of two safety grade pressure sensors, one out of four safety grade temperature sensors or selection of one out of a number of incore thermocouples. The difference in pressure from saturation will be indicated on a recorder.

Design/installation is being expedited because of lead time of components. Due to this fact, the entire system will not be completed by January 1, 1980. However, installation of a subcooling meter with single channel temperature and pressure inputs has been completed with alarms set at saturation and 300 psi above saturation.

In addition to the above system, a supplemental means, utilizing the plant computer has been installed that will continuously monitor the subcooling condition. This system will compute the saturation pressure due to either the average hot leg temperature or the average incore thermocouples. The reactor coolant pressure is then compared to this and the difference displayed on the Cathrode Ray Tube (CRT) or logged on the computer recorders.

Two annunciated control room alarms have also been added to the computer. The first alarm is presently set to activate when RCS pressure decreases to 300 psig above saturation. The second alarm will actuate when RCS pressure decreases to saturation. These two alarms provide additional warning to the operator if the RCS approachs saturation.

The above mentioned plant modification will not adversely impact the reactor protection or engineering safeguards system.

Refer to the attached sheets for further detailed information.

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	P-Psat.
Display Type (Analog, Digital, CRT)	Analog
Continuous or on Demand	Continuous
Single or Redundant Display	Single
Location of Display	Control Room Flight Panel
Alarms (include setpoints)	Saturation and 300 psi above Sat
Overall uncertainty (^{O}F, PSI)	*
Range of Display	1500 psi above saturation
Qualifications (seismic, environmental, IEEE323)	Note I

Calculator

Type (process computer, dedicated digital or analog calc.)	Analog
If process computer is used, specify availability (% of time).	N/A
Single or redundant calculators	Single
Selection Logic (highest T., lowest press)	Manual redundant pressure and temperature
Qualifications (seismic, environmental, IEEE323)	N/A
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

Input

Temperature (RTD's or T/C's)	RTD and T/C's
Temperature (number of sensors and locations)	4 T/C, Incore 4 RTD, hot leg piping
Range of temperature sensors	0-700 ^{O}F

* All components are accurate to $\pm$ 1/2%, except the Indicator, which is $\pm$ 2%.

Uncertainty of temperature sensors (^{o}F at 1)	*
Qualifications (seismic, environmental, IEEE323)	Note I
Pressure (specify instrument used)	PT-403, PT-402
Pressure (number of sensors and locations)	2, RCS Hot Leg Piping
Range of Pressure sensors	0-3000 psig, 0-2500 psig
Uncertainty of pressure sensors (PSI at 1)	*

<u>Backup Capability</u>

Availability of Temp & Press	In Control Room
Availability of Steam Tables etc.	In Control Room
Training of operators	In Progress
Procedures	In Progress

* All components are accurate to ± 1/2%, except the Indicator, which is ± 2%.

<u>Note I:</u> The sensors (both pressure and temperature) are part of the initial plant design and therefore qualified to the standards in effect at the time of construction. The pressure transmitters were tested by Franklin Institute/ Foxboro and documented in report Q9-6005, T2-1075 and T3-1097. In addition, Westinghouse performed tests on a similar transmitter and documented it in WCAP 7410-L Volume I. We presently are planning to upgrade the transmitters with a modified Barton Transmitter qualified to IEEE 323 1971 and IEEE 344.

The RTD temperature sensor was qualified via Westinghouse WCAP 9157.

C. Additional Instrumentation to Indicate Inadequate Core Cooling

The submittal referenced in 2.1.3.b.A above described the capabilities of the core exit thermocouples in determining the existence of inadequate core cooling conditions and their superiority in some instances to the loop RTD's for measuring true core conditions. Other means of determining the approach to or existence of inadequate core cooling could be:

1) Reactor vessel water level
2) Incore detectors
3) Excore detectors
4) Reactor coolant pump motor currents
5) Steam generator pressure

A discussion of the possible use of these measurements is addressed below.

The use of incore movable detectors to determine the existence of inadequate core cooling conditions appears doubtful. The detectors could be driven in to the tops of the incore thimbles, which are located at the top of the core, following an accident in which concern for inadequate core cooling exists. The problem comes in the lack of sensitivity of the detectors to very low neutron levels and changes that would occur due to core uncovery. Gamma detectors could perhaps be employed, but they suffer from similar sensitivity problems, and the fact that gamma levels in the fuel region change insignificantly between the covered and uncovered condition. As a result, it does not appear worthwhile to pursue incore movable detectors as a means of determining inadequate core cooling conditions.

The use of excore detectors has been mentioned as a possibility in responding to core uncovery. The only detectors which would have the required sensitivity are the source range monitors, since the intermediate and power range monitors are not sensitive enough to the low level changes resulting from vessel voiding. The use of the source range monitors will be investigated further as part of the more in-depth study of inadequate core cooling being performed by the Westinghouse Owners' Group. However, their use is probably limited to those instances when significant voiding exists in the downcomer region, since normally, water in the downcomer would effectively shield the detectors from the core region whether or not the voids exist.

Reactor coolant pump motor current, which could be indicative of core voiding, is inappropriate as a reliable means of determining inadequate core cooling, since a loss of offsite power pump trip due to a LOCA blowdown shuts the pumps down.

Steam generator pressure, which already exists, is useful in the case where heat transfer from primary to secondary is interrupted due to loss of natural circulation. This, however, does not satisfy requirements to indicate the approach to inadequate core cooling, nor does it indicate the true condition of the core.

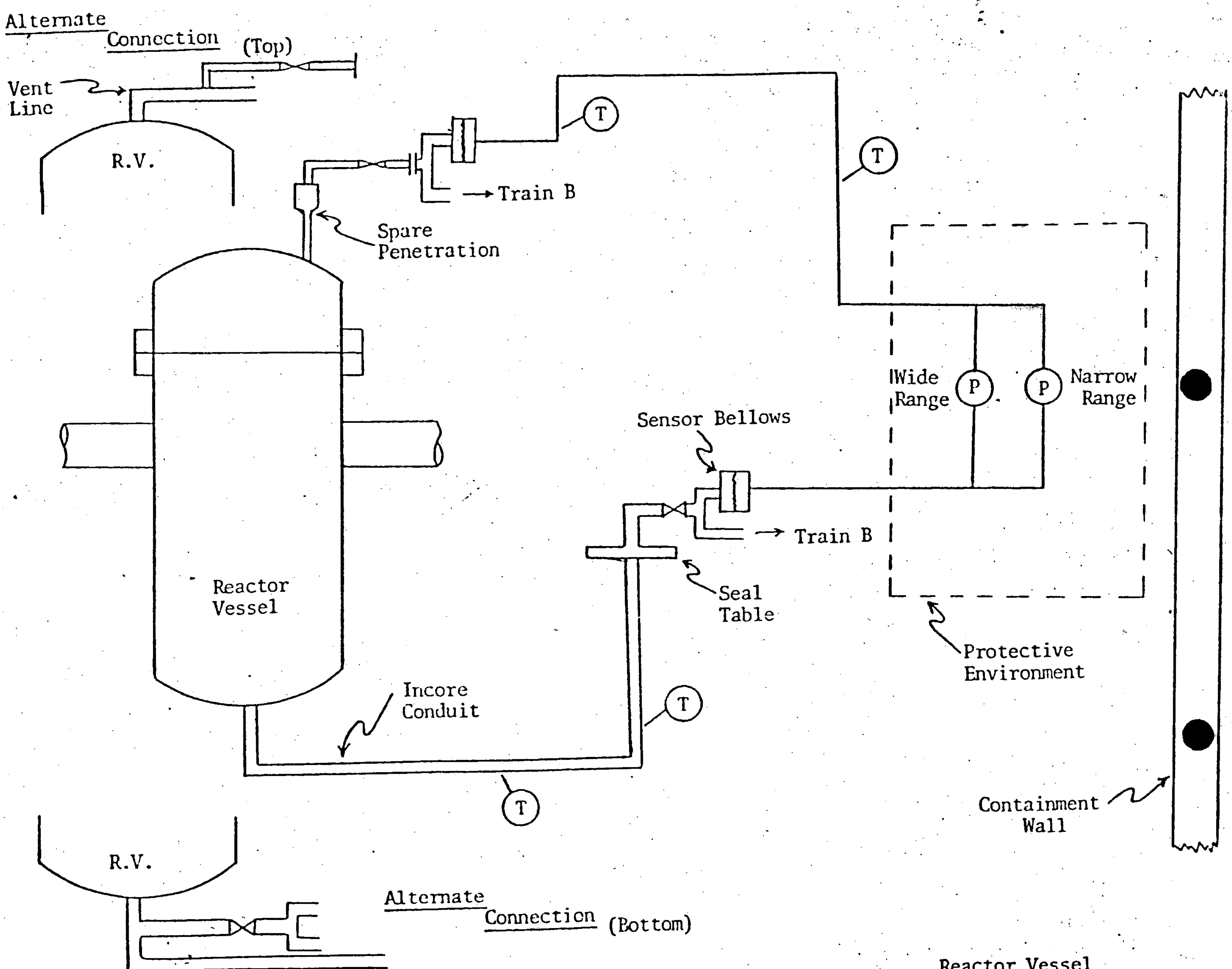
Reactor vessel water level determination is the most promising of the items discussed to provide additional capability of determining the approach to and the existence of inadequate core cooling. Several systems for determining water level are under review by the Westinghouse Owners' Group. A conceptual design of one system is given below:

Vessel Level System Description

After examining many different methods and principles for determining the water level in the reactor vessel, a basic delta pressure measurement from the bottom of the vessel to the top of the vessel appears to provide the most meaningful and reliable information to the operator. One of the reasons for choosing this system is that the sources of potential errors are better known for this system than for any other new or untested system.

The attached figure shows a simplified sketch of the proposed vessel level instrumentation system. The bottom tap of the instrument would use a thimble of the incore movable detector system, either at the seal table or in the thimble below the vessel. Use of the thimble as part of the incore flux monitoring would not be lost. The flux thimble guide tube would be tapped below the vessel and an instrument line connection made. The instrument line would have an isolation valve and slope down to a hydraulic coupler connected to a sealed reference leg. For connection at the seal table, a special fitting would be utilized which would be connected to an isolation valve and sealed reference leg. The sealed reference leg would go to the differential pressure transmitter located at a higher elevation above the expected level of containment flooding. A similar sealed leg would go to the top of the vessel and penetrate the head using the vent line or a special connection on a spare RCC mechanism penetration. Two trains of vessel level instrumentation would be provided.

The behavior of the signal generated by this level instrument under normal and accident conditions is being evaluated. The usefulness of this instrument to provide an unambiguous indication of inadequate core cooling is being evaluated as part of NUREG Section 2.1.9. The potential errors and accuracy of a final system configuration are being evaluated to assess its usefulness to provide information to the operator for proper operation of a vessel venting system and for normal water level control during periods when the primary system is open and a water level may exist in the vessel. The connection of the level system to the vessel head should be designed to be compatible with the head vent system. Operation of the vent system should not upset all indications of vessel level. This can easily be avoided by using a separate instrument tap or by using more than one location.



Reactor Vessel Level Instrumentation

Section 2.1.4

Containment Isolation Provisions for PWRs and BWRs

1. The Containment Isolation System at Indian Point 3 satisfies the recommendations of SRP 6.2.4 that there be diversity in the parameters sensed for the initiation of containment isolation. Automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals (Phase A) isolates essential systems and is derived in conjunction with automatic safety (or manual) injection initiation based on the following parameters:

   a. Containment Pressure (high)
   b. Low Pressurizer Pressure
   c. Steam Line Differential Pressure
   d. High Steam Line Flow (coincident with low steam line pressure or low Tavg)

   The second signal (Phase B) isolates essential reactor coolant pump auxiliary systems and is derived in conjunction with automatic containment spray actuation based on containment pressure (high-high). In addition to the Phase A and B signals a high radiation signal inside containment also isolates the containment purge supply and exhaust ducts and the containment pressure relief venting system.

2. The results of the re-evaluation of the manual containment isolation valve classification is as shown in the attached table. Additional information can be found in Chapter 5 of the FSAR.

   Manual containment isolation valves are strictly administratively controlled in accordance with station operation procedure SOP-CB-1 "Containment Integrity".

   As can be seen from the attached table, all manual non-essential valves are locked shut during normal operations with the exception that the valves marked with an asterisk (*) may require periodic cycling to maintain plant operating parameters and/or to perform periodic tests. The plant operating procedures are presently being revised to re-emphasize to the operating personnel that these asterisked (*) valves must only be opened when absolutely essential for plant operations and must be immediately returned to the locked shut position upon completion of the task.

3. A review of the containment isolation system at Indian Point No. 3 indicates that there are a number of valves which automatically reset to the previous position upon reset of containment Phase A isolation. At present, these valves are under operator control via operating procedures to be placed

in the closed position prior to resetting Phase A. These valves will be modified to preclude automatic opening on reset prior to the unit returning to service. The modification to the valve circuits entails the installation of pushbuttons that work in conjunction with the Containment Isolation Reset Switches so that each valve control circuit will not reset unless both the Containment Isolation Reset Switch and the dedicated pushbutton have been operated.

MANUAL VALVES

VALVE I.D. NUMBER	SYSTEM	DESCRIPTION	CLASSIFICATION	REMARKS
550	Nitrogen	N_2 to PRT	Non-essential	Locked shut during normal Ops. *
744	Auxiliary Coolant	RHR Discharge	Essential	
1870	Auxiliary Coolant	RHR Miniflow	Essential	
743	Auxiliary Coolant	RHR Miniflow	Essential	
732	Auxiliary Coolant	RHR Suction	Essential - P.A.	Locked shut during normal Ops.
855A	Safety Injection	RHR Suction	Essential - P.A.	" " " " "
855B	Safety Injection	RHR Suction	Essential - P.A.	" " " " "
205	Chemical & Volume Control	Charging Pump Discharge	Essential	
226	Chemical & Volume Control	Charging Pump Discharge	Essential	
227	Chemical & Volume Control	Charging Pump Discharge	Essential	
250A	Chemical & Volume Control	RCP 31 Seal Injection	Essential	
241A	Chemical & Volume Control	RCP 31 Seal Injection	Essential	
250B	Chemical & Volume Control	RCP 32 Seal Injection	Essential	
241B	Chemical & Volume Control	RCP 32 Seal Injection	Essential	
250C	Chemical & Volume Control	RCP 33 Seal Injection	Essential	
241C	Chemical & Volume Control	RCP 33 Seal Injection	Essential	
250D	Chemical & Volume Control	RCP 34 Seal Injection	Essential	
241D	Chemical & Volume Control	RCP 34 Seal Injection	Essential	
869A	Safety Injection	31 Cont.Spray PP Discharge	Essential	
869B	Safety Injection	32 Cont.Spray PP Discharge	Essential	

VALVE I.D. NUMBER	SYSTEM	DESCRIPTION	CLASSIFICATION	REMARKS
851A	Safety Injection	Safety Injection PP Disch.	Essential	
850A	Safety Injection	Safety Injection PP Disch.	Essential	
1610	Nitrogen	N_2 to RCDT	Non-essential	Locked Shut During Normal Ops. *
1835A	Safety Injection	Safety Injection PP.Disch.	Essential	
1835B	Safety Injection	Safety Injection PP.Disch.	Essential	
990A	Sampling System	Recirc. Pump Discharge	Essential - P.A.	Locked Shut During Normal Op
990B	Sampling System	Recirc. Pump Discharge	Essential - P.A.	Locked Shut During Normal Ops.
752F	Auxiliary Coolant	Aux. Comp.Cooling PP Disch.	Essential	
753F	Auxiliary Coolant	Aux. Comp.Cooling PP Disch.	Essential	
752J	Auxiliary Coolant	Aux. Comp.Cooling PP Disch.	Essential	
753J	Auxiliary Coolant	Aux. Comp.Cooling PP.Disch.	Essential	
891A	Nitrogen	N_2 to Accumulator 31	Non-essential	Closed *
891B	Nitrogen	N_2 to Accumulator 32	Non-essential	Closed *
891C	Nitrogen	N_2 to Accumulator 33	Non-essential	Closed *
891D	Nitrogen	N_2 to Accumulator 34	Non-essential	Closed *
863	Nitrogen	N_2 to Accumulators	Non-essential	Closed *
878A	Safety Injection	Safety Injection PP Suct.	Non-essential	Locked Closed
878B	Safety Injection	Safety Injection PP Suct.	Non-essential	Locked Closed
PCV-1111	Weld Channel & Pen Press.	Weld Channel Press Supply	Essential	
PCV-1111	Weld Channel & Pen Press.	Weld Channel Press Supply	Essential	

VALVE I.D. NUMBER	SYSTEM	DESCRIPTION	CLASSIFICATION	REMARKS
1814A	Safety Injection	Cont Press Trans.Isol.	Essential	
1814B	Safety Injection	Cont Press Trans.Isol.	Essential	
1814C	Safety Injection	Cont.Press Trans.Isol.	Essential	
859A	Safety Injection	S.I. Pump Test Line	Non-essential	Locked Closed *
859C	Safety Injection	S.I. Pump Test Line	Non-essential	Locked Closed *
1833A	Safety Injection	Boron Injection TK Bypass	Non-essential	Locked Closed *
1833B	Safety Injection	Boron Injection TK Bypass	Non-essential	Locked Closed *
SA-24	Station Air	Containment Sta.Air Sup.	Non-essential	Locked Closed
SA-24	Station Air	Containment Sta.Air Sup.	Non-essential	Locked Closed
580A	Reactor Coolant	Dead Weight Cal.Isol.	Non-essential	Locked Closed
580B	Reactor Coolant	Dead Weight Cal.Isol.	Non-essential	Locked Closed
958	Sampling System	Residual Heat Removal Loop	Non-essential	Locked Closed *
959	Sampling System	Residual Heat Removal Loop	Non-essential	Locked Closed *
990C	Sampling System	Residual Heat Removal Loop	Non-essential	Locked Closed *
SWN-41	Service Water-Nuclear	FCU 31 Inlet	Essential	
SWN-43	Service Water-Nuclear	FCU 31 Inlet	Essential	
SWN-41	Service Water-Nuclear	FCU 32 Inlet	Essential	
SWN-43	Service Water-Nuclear	FCU 32 Inlet	Essential	
SWN-41	Service Water-Nuclear	FCU 33 Inlet	Essential	
SWN-43	Service Water-Nuclear	FCU 33 Inlet	Essential	

VALVE I.D. NUMBER	SYSTEM	DESCRIPTION	CLASSIFICATION	REMARKS
SWN-41	Service Water-Nuclear	FCU 34 Inlet	Essential	
SWN-43	Service Water-Nuclear	FCU 34 Inlet	Essential	
SWN-41	Service Water-Nuclear	FCU 35 Inlet	Essential	
SWN-43	Service Water-Nuclear	FCU 35 Inlet	Essential	
SWN-44	Service Water-Nuclear	FCU 31 Outlet	Essential	
SWN-51	Service Water-Nuclear	FCU 31 Outlet	Essential	
SWN-44	Service Water-Nuclear	FCU 32 Outlet	Essential	
SWN-51	Service Water-Nuclear	FCU 32 Outlet	Essential	
SWN-44	Service Water-Nuclear	FCU 33 Outlet	Essential	
SWN-51	Service Water-Nuclear	FCU 33 Outlet	Essential	
SWN-44	Service Water-Nuclear	FCU 34 Outlet	Essential	
SWN-51	Service Water-Nuclear	FCU 34 Outlet	Essential	
SWN-44	Service Water-Nuclear	FCU 35 Outlet	Essential	
SWN-51	Service Water-Nuclear	FCU 35 Outlet	Essential	
SWN-71	Service Water-Nuclear	FCU Motor Cooler 31	Essential	
SWN-71	Service Water-Nuclear	FCU Motor Cooler 32	Essential	
SWN-71	Service Water-Nuclear	FCU Motor Cooler 33	Essential	
SWN-71	Service Water-Nuclear	FCU Motor Cooler 34	Essential	
SWN-71	Service Water-Nuclear	FCU Motor Cooler 35	Essential	
UH-37	Auxiliary Steam	Cont. UH Stm. Supply	Non-essential	Locked closed.

VALVE I.D. NUMBER	SYSTEM	DESCRIPTION	CLASSIFICATION	REMARKS
UH-38	Auxiliary Steam	Cont. UH Cond. Return	Non-essential	Locked closed
1882A	Hydrogen Recombiner	Cont. O_2 Supply	Essential - P.A.	Locked Shut during Normal Ops.
1875A	Hydrogen Recombiner	H_2 to 31 Combustor	Essential - P.A.	Locked Shut during Normal Ops.
1875B	Hydrogen Recombiner	H_2 to 32 Combustor	Essential - P.A.	Locked Shut during Normal Ops.
1876A	Hydrogen Recombiner	H_2 to 31 Combustor	Essential - P.A.	Locked Shut during Normal Ops.
1876B	Hydrogen Recombiner	H_2 to 32 Combustor	Essential - P.A.	Locked Shut during Normal Ops.
PS-7	Post Acc. Cont. Venting	Filter Supply	Essential - P.A.	Locked Shut during Normal Ops.
PS-8	Post Acc. Cont. Venting	Filter Supply	Essential - P.A.	Locked Shut during Normal Ops.
PS-9	Post Acc. Cont. Venting	Filter Supply	Essential - P.A.	Locked Shut during Normal Ops.
PS-10	Post Acc. Cont. Venting	Inst. Air Filter Purge	Essential - P.A.	Locked Shut during Normal Ops.
888A	Safety Injection	S.I. Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
888B	Safety Injection	S.I. Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
1890A	Post Accident Cont. Sampling	Vacuum Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
1890B	Post Accident Cont. Sampling	Vacuum Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
1890C	Post Accident Cont. Sampling	Vacuum Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
1890D	Post Accident Cont. Sampling	Vacuum Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
1890E	Post Accident Cont. Sampling	Vacuum Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.

VALVE I.D. NUMBER	SYSTEM	DESCRIPTION	CLASSIFICATION	REMARKS
1890F	Post Accident Cont. Sampling	Vacuum Pump Suction	Essential - P.A.	Locked Shut during Normal Ops.
1890G	Post Accident Cont. Sampling	Sample Device Disch.	Essential - P.A.	Locked Shut during Normal Ops.
1890H	Post Accident Cont. Sampling	Sample Device Disch.	Essential - P.A.	Locked Shut during Normal Ops.
1890J	Post Accident Cont. Sampling	Sample Device Disch.	Essential - P.A.	Locked Shut during Normal Ops.

Section 2.1.5.a

Dedicated Hydrogen Control Penetrations

Indian Point 3 has redundant hydrogen recombiners located in containment. No requirement for external recombiners is associated with this facility, therefore, no further action is required on this item.

Section 2.1.5.b

Rulemaking to Require Inerting BWR Containments

Not applicable to IP-3.

Section 2.1.5.c

Capability to Install Hydrogen Recombiners at Each Light

Water Nuclear Power Plant

A confirmatory review of the existing hydrogen recombiners systems has been completed. The procedure for recombiner operation will be revised in view of the shielding analysis results of Section 2.1.6.b before the unit returns to service.

The redundant hydrogen recombiners are located in containment. The recombiner control panels and support systems are located external to containment at El. 67'-6" of the Fan Building. Table 1 of Section 2.1.6.b indicates that additional shielding or other modifications are required in this area to reduce the doses to acceptable levels.

A review of the Hydrogen Recombiner System design criteria documented in Section 14.3.6-1 of the FSAR reveals that the hydrogen recombiners are designed to handle the hydrogen generated from the following sources after the DBA.

1) 5% Zirconium - water reaction
2) Corrosion of plant materials
3) Radiolysis of Core and Sump waters

Figures 14.3.6-6 and 14.3.6-7, attached, show the volume percent of hydrogen in containment for the AEC Safety Guide 7 model and Westinghouse model respectively. Both of these figures show that based on a 5% Zirconium-Water reaction, the hydrogen recombiners are not needed until several days after the accident.

In light of TMI, we will however revise our post-accident hydrogen recombiner operating philosophy to require early sampling of the containment atmosphere and early activation of the recombiner systems if conditions warrant. The appropriate operating procedures will be revised prior to returning the Unit to service.

As a long-term fix, the Power Authority has already committed to install permanent containment hydrogen indicators. These will provide continous indication of containment hydrogen concentration and will provide the operator with additional information so that he can evaluate the need for early recombiner system actuation.

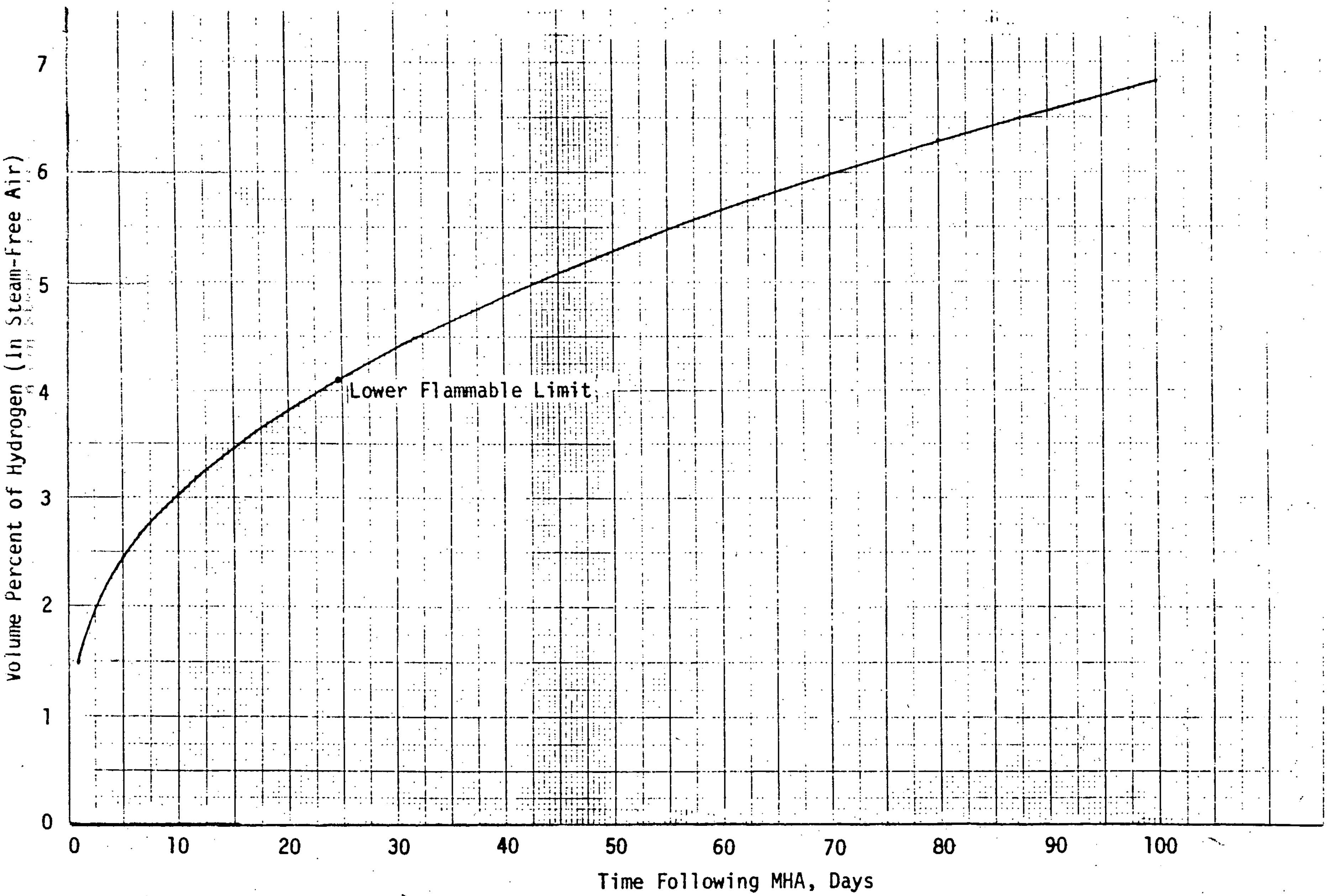


Figure 14.3.6-6 Concentration of Hydrogen in Containment, Safety Guide 7 Model

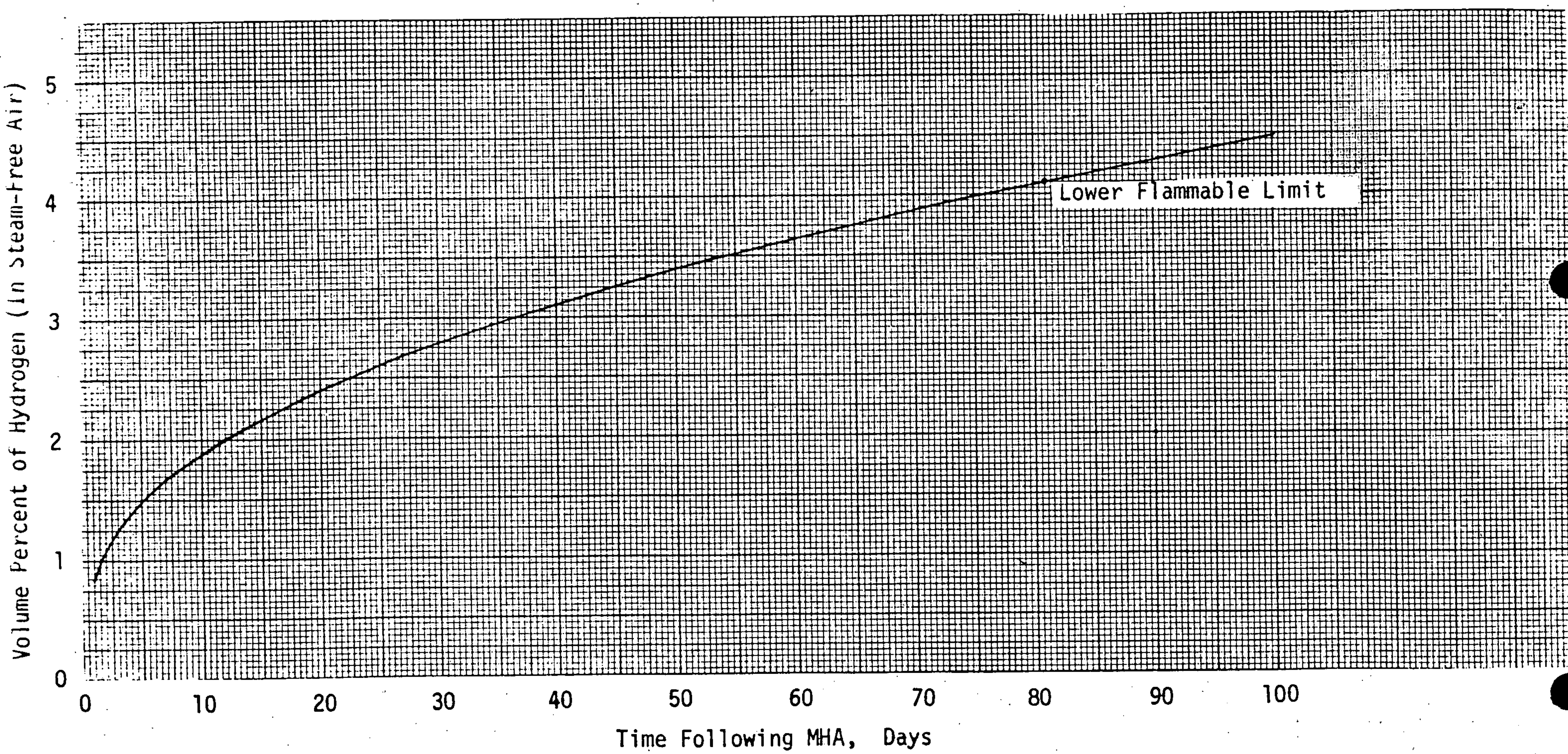


Figure 14.3.6-7 Concentration of Hydrogen in Containment, Westinghouse Model

Section 2.1.6.a

Integrity of Systems Outside Containment likely to contain Radioactive material for PWR's

At Indian Point 3 the components utilized for the recirculation phase of safety injection are located inside the Containment. This equipment has been designed for operation under an accident environment and consists of redundant recirculation pumps and RHR heat exchangers. This results in reducing dose rates from fluid systems external to containment during severe accident conditions.

A review of containment external lines that normally transfer low level or non-radioactive material has been completed. The following is a listing of those lines that could become contaminated with highly radioactive fluid during the post accident recovery phase.

1) Residual Heat Removal System.
2) Cross connect between low head recirculation system and high head safety injection system.
3) High head safety injection system (partial).
4) Reactor coolant sampling system.
5) Post accident containment system.

A program is now being implemented to periodically inspect these systems for leakage and perform the necessary preventative maintenance to reduce any leakage to as low as possible prior to the unit returning to service.

Design and operator deficiencies discussed in your letter regarding North Anna and related incidents has been considered for the above listed systems and no plant modifications are required.

Section 2.1.6.b

"Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post Accident Conditions"

A design review of Indian Point 3 has been conducted to identify areas, components and access paths which may require occupancy during post accident recovery operation. This review was conducted in accordance with NUREG-0578 Section 2.1.6.b as clarified by the NRC letter of October 30, 1979.

The source terms used in the dose calculations were those listed in the Westinghouse Radiation Design Manual Revision 3, dated 11/78, resulting from a "Maximum Credible Accident". For liquid containing systems 100% of the core noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of the remaining fission products were assumed to be released to the reactor coolant. For gas containing systems 100% of the core eqilibrium noble gas inventory and 25% of the core equilibrim halogen inventory were assumed to be released to the containment atmosphere. The liquid source terms were considered to be mixed with the reactor coolant and safety injection volumes including the accumlators, boron injection tank and the refueling water storage tank. The gaseous source terms were considered mixed with the containment atmosphere. These source terms were then used in conjunction with QAD-GEOM (360) to determine dose rates at the selected location.

The code QAD-CYL is the cylindrical version of QAD-GEOM, one in a series of QAD codes developed by Los Alamos Scientific Laboratory to calculate the fast-neutron and gamma-ray penetration through various shield confirgurations. The point-kernel method utilized by QAD-CYL involves representing the source volume by a number of point isotropic sources and computing the line-of-sight distance from each of these source points to the detector point. From the distance through the shielding region and the attenuating characteristics of the shielding materials, the geometric attenuation and material attenuation are determined. For further information, see RSIC publication CCC-48.

QAD-GEOM was used to model various length segments of different pipe sizes and to compute unshielded doses at 30 minutes as a function of distance and angle from the mid point of the segment. The same computation was also run for a representative pipe diameter and varying concrete thicknesses to develop shielded doses which were then used to generate does reduction factors. Four inches was selected as the representative pipe diameter as it was found to result in only approximately $\pm$ 10% error for the smallest and largest pipe diameters respectively.

The unshielded doses and dose reduction factors were plotted and used in conjunction with a segmented model of pipe routing and intervening shielding to develop individual dose contributions from all applicable pipe segments and to compute an approximate total direct beaming dose. In areas where exposure to a single bounce exists, the dose contribution from this will be taken into account.
Tanks which could contain large amounts of radioactivity, and the containment building were modeled and computed separately and the resulting contribution considered with the piping to determine the total dose at the point of interest. Piping and tubing smaller than 1" in diameter was evaluated using the dose rates computed for 1" pipe scaled by the volume ratio.

The dose rate in each area of interest was evaluated considering the effect of a single source in conjuction with the containment dose. The sources used were selected as being the worst case of probable combinations of operating systems. This source was then evaluated by segments until either the segment contributions became negligible or the applicable limit was reached. If the first source did not exceed the limit, additional sources were evaluated to verify the habitability of the area.

The evaluation criteria was based on the occupancy and equipment qualification requirements as follows:

a. Continuous occupany - $\leq$ 15 mr/hr

b. Infrequent access - $\leq$ 5 rem whole body does considering the required occupancy for the duration of the accident

c. Equipment Qualificaton - $\leq$ 10^6 Rad integrated dose

Components, valves and areas which would require access to aid in post accident recovery operations and equipment operating areas were identified as follows:

1. PAB Entrance (EL 55')
2. Waste Disposal Control Panal (EL 55' PAB)
3. Safety Related Motor Control Centers 36A and 36B (EL 55' PAB)
4. Sampling Room (EL 55" PAB)
5. Fan Building Stairway
6. Hydrogen Recombiner Operating Area (EL 67'-6" Fan Bldg.)

7. Post Accident Containment Air Sampling System (EL 67'-6" Fan Bldg.)
8. Manual Containment Isolation Valves (EL 54' Pipe Trench)
9. Safety Injection Pump Room (EL 15 PAB)
10. RHR Pump Room (EL 15 PAB)
11. Laboratory (EL41 RAB)
12. Control Room
13. Diesal Generators
14. Auxiliary Feedwater Pumps
15. Health Physics Building
16. Nuclear Plant Operator (NPO) Office

The results of the design review are presented in Table 1 along with recommendations for possible solutions which will be evaluated during the first part of 1980. The most practical solution will be determined and then installed prior to January 1, 1981.

It should be noted that inherent conservatisms exist in the data due to the computer modeling techniques used and the limited time available to develop the information. The basic computer model used for each pipe segment is shown in Figure 1. This model is conservative for angles approaching 90^{o} in that self shielding which would be provided by the adjacent segment has been neglected. In addition, the effect of the pipe wall was not considered for dose points at 90^{o}. Additional conservatism exists in the shield reduction factors as they were calculated at time 30 minutes. As time increases the dominent source term energy levels shift from the higher to the lower energy groups and the concrete shield worth will increase. This effect has not been included and could result in conservative factors of up to 50 at 1 year and 6 feet of concrete. For a time of 1 day and 3 feet of concrete this factor is approximately 10. The combined effect of these conservatisms indicates that the results obtained from this analyses should be considered as "order of magnitude" approximations. Therefore, appropriate areas will be examined in more detail as part of the engineering activities to be conducted during 1980.

GENERAL COMPUTATION SHEET

united engineers & constructors inc.

DISCIPLINE

NAME OF COMPANY ______ UNIT/S ______

SUBJECT Pipe Dose Computer Model

CALC. SET NO.		REV	COMP. BY	CHK'D BY
PRELIM		0	[illegible] DATE 12/9/79	DATE
FINAL				
VOID				
SHEET	OF		DATE	DATE
J.O.				

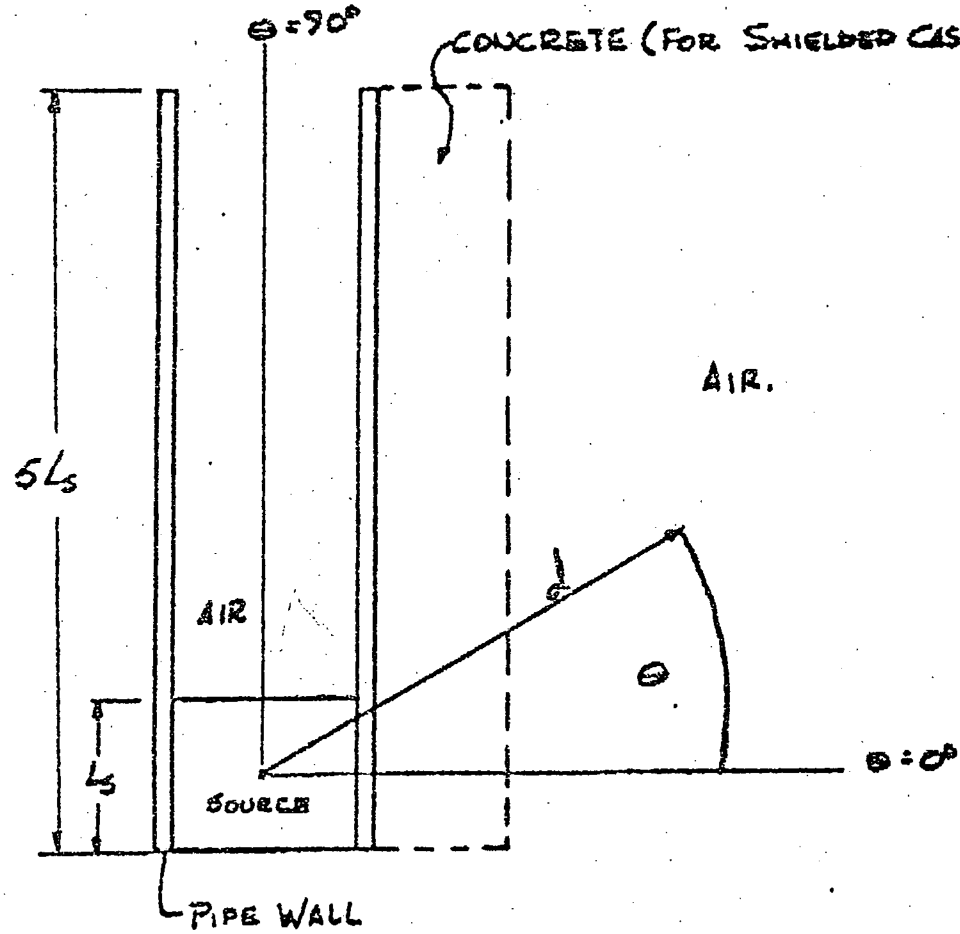


FIGURE 1

TABLE 1

Design Review Results

Point of Interest	Sources Investigated	Evaluation Criteria (note 1)	Results of Review	Remarks
PAB Entrance	Containment* R.C. Sample* Pipe Trench (RHR, SI etc.)	b	Major source of dose rate is due to the R.C. Sample and the containment. Results are acceptable.	Invesitgate shielding in sample room.
Waste Disposal Canal	(Same as 1 above)	b	(Same as 1 above)	
Motor Control Centers	(Same as 1 above)	b	(Same as 1 above)	
Sampling Room	(Same as 1 above)	b	Additional dose reduction is required	Possible Fixes Include: Automatic Sampling System Local Shielding Revised Sampling Procedures
Fan Bldg. Stairway	Containment* Pipe Trench (RHR, SI*, etc.)	b	Acceptable	
Hydrogen Recombiner Area	Containment* PAC Air Sample RC Sample vice Failed Fuel Detector Pipe Trench (RHR, SI*, etc.)	b/c	Additional dose reduction is required	Possible Fixes Include: Additional Shielding Remote Operation Rerouting pipe Panel Relocation
Post Accident Containment Air Sampling Station	Containment* Containment Air Pipe Trench	b/c	Additional dose reduction is required.	Possible Fixes Include: Local Shielding Valve Operator Extensions Automatic Valves Shielded Sample Pig Remote Automatic Analysis

TABLE 1
(Cont'd)

Point of Interest	Sources Investigated	Evaluation Criteria (Note 1)	Results of Review	Remarks
Manual Containment Isolation Valves	Containment* RHR Piping SI Piping*	b/c	Additional dose reduction is required.	Possible Fixes Include: Valve Operator Extensions (must confirm with 7 above) Automatic Valve Operators Revised Procedures Shielding
Safety Injection Pump Room	SI Piping* Pipe Trench (RHR, CVCS, etc.)	c	Acceptable	
RHR Pump Room	RHR Piping* Pipe Trench (CVCS, WDS, etc.)	c	Acceptable	
Laboratory	Pipe Chase	b	Acceptable	
Control Room	Containment*	a/c	Acceptable	
Diesel Generators	Containment	b/c	Acceptable	
Auxiliary Feedwater Pumps	Containment	b/c	Acceptable	
Health Physics Bldg.	Containment	a	Acceptable	
NPO Office	Containment	a	Acceptable	

NOTES: 1. a) Continuous Occupancy $\leq$ 15 mr/hr
b) Infrequent Access $\leq$ 5 rem Whole Body
c) Equipment Qualification $\leq$ 10^6 Rad Integrated Dose

*Source Used

Section 2.1.7.a

Auto Initiation of the Auxiliary Feedwater System

A confirmatory review has been conducted of the initiation logic design, operational and administrative procedures of the Auxiliary Feedwater System for compliance with the NRC's clarification letter of October 30, 1979. This review has concluded that the system complies with Section 2.1.7.a. The results of this review are presented below:

1. The AFWS has auto/manual initiation.

2. Testability exists for the initiating signal and circuits.

3. Initiating signals and circuits are powered from emergency buses.

4. Necessary pumps are automatically sequenced on to the emergency buses as part of the normal safety injection scheme. The addition of these loads will not compromise the diesel's generating capacity. The power to the valves are always available via a vital instrument bus.

5. Failure of the automatic circuits will not result in loss of manual initiation capability from the control room.

6. The instrument air supply to necessary valves in the Auxiliary Feedwater Building is backed up by a N_2 bottle system. In addition, the valves can be manually operated.

Section 2.1.7.b

Auxiliary Feedwater Flow Indication to Steam Generators for PWR's

A confirmatory review of the Auxiliary Feedwater flow indication to steam generators in the control room has been conducted.

Based on the clarification letter issued by the NRC dated 10/30/79, the results of this review are as follows.

The existing auxiliary feedwater flow instrumentation provides local and remote indication which are powered from the vital instrument buses.

Flowmeters are provided in the common auxiliary feedwater connections to the main feedwater line of each steam generator. These indicate flow remotely in the control room and also in the auxiliary feed pump room. Steam generator water level indicators also are located in the control room and auxiliary feed pump room which are powered from the vital instrument buses. The present configuration meets the single failure criteria considering the S.G. level as a backup to the flow indicators in the control room. It is worth noting that loss of flow indication (and flow) to one steam generator will not compromise the heat removal capability from the reactor. Testability exists for the flow and level indications. The instrument error inaccuracy of the flow and level indicator channels are less than 10%.

Section 2.1.8.a

Improved Post Accident Sampling Capability

We have completed a design review of the Indian Point 3 facility to assure the capability of obtaining a reactor coolant sample and a vapor containment sample within one hour of any incident involving deterioration of the fission product boundaries in the reactor coolant system and have found that the individual doses received using the existing systems may exceed established radiation exposure limits. Conceptual designs of the modifications necessary to reduce the personnel exposure have been completed and the necessary equipment has been ordered. We are vigorously pursuing installation of these major modifications with a target completion date of February 1, 1980, even though these modifications are not required by the NUREG-0578 clarification document (page 27) to be completed until January 1, 1981. Facilities have also been designated and will be operable in the same time frame to allow for analysis of reactor coolant samples for all gamma emitting isotopes and also hydrogen, oxygen, chloride and boron within one hour after receipt of sample. Facilities are also being provided to allow for analysis of hydrogen levels in containment atmosphere in the range of zero to ten volume percent in this same time frame. Procedures to use these facilities and accomplish these analyses will be implemented within the same time frame.

A brief description of the modified sampling system is as follows:

The RCS sample system will be modified to allow drawing a remote shielded 40 ml sample of reactor coolant in a container having 4 inches of 4π radians lead shielding. This sample will then be transferred to a lead shielded frame hood for chemical analysis. Chloride will be determined with an ion selective electrode, hydrogen will be determined by gas chromatography and diluted samples will be analyzed for boron by plasma emission spectrometry and isotopic analysis by gamma spectrometry. Oxygen will be determined with an in-line oxygen probe within the primary sampling system.

Section 2.1.8.b

Increased Range of Radiation Monitors

Engineering analyses has been performed and procedures have been drafted to provide an ability to monitor plant vent stack releases up to 100,000 microcuries per cc. This is being accomplished by installing an Eberline RD-17a ion chamber attached directly to the R-13 plant vent particulate sampling line in the containment penetration areas of the Indian Point 3 facility. This chamber is being provided with shielding to minimize background interference. This ion chamber will then allow for a readout remotely on an Eberline RM-16 readout device. Engineering calculations have been performed to provide a conversion of the mR per reading of this instrument to a microcuries per cc concentration range in the plant vent stack.

Provision is also being made to draw a sample from the R-13 plant vent sample line to allow for isotopic analysis of individual isotopes involved in a release. This will be a grab sample point in a low background area between the vapor containment building and the fan building of the Indian Point 3 facility. This modification and associated procedures will be implemented prior to the unit returning to service.

Section 2.1.8.c

Improved In Plant Iodine Instrumentation

Samples of Iodine can be obtained and analyzed using existing equipment and procedures at the Indian Point 3 facility. This will be accomplished using Eberline and NNC air sampling equipment presently in use at the site and Eberline single channel analyzers with built in stabilization circuits. This instrumentation in conjunction with potassium iodide impregnated charcoal cartridges will be used for initial assessment of radioiodine exposure potentials. Noble gas interference in these charcoal cartridges can be eliminated by use of either clean air flushing of the charcoal cartridges or heating in laboratory drying ovens to strip the noble gases from the cartridges leaving the iodine alone intact on these cartridges. Any modification necessary to existing procedures will be accomplished prior to returning the unit to service.

Section 2.1.9

Analysis of Design and Off-Normal Transients and Accident Analysis

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analysis of transient and accident scenarios including operator actions not previously analyzed are being performed on a generic basis by the Westinghouse Owners' Group of which the Power Authority of the State of New York is a member. The small break analyses have been completed and were reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners' Group on June 29, 1979. Incorporated in that report were guidelines that were developed as a result of small break analyses. These guidelines have been reviewed and approved by the B&O Task Force and have been presented to the Owners' Group utility representatives in a seminar held on October 16-19, 1979. Following this seminar, each utility has developed plant specific procedures and trained their personnel on the new procedures. Revised procedures and training are in place in accordance with the requirement in Enclosure 6 to Mr. Eisenhut's letter of September 13, 1979, and Enclosure 2 to Mr. Denton's letter of October 30, 1979.

The work required to address the other two areas--inadequate core cooling and other transient and accident scenarios--has been performed in conjunction with schedules and requirements established by the Bulletins and Orders Task Force. Analysis related to the definition of inadequate core cooling and guidelines for recognizing the symptoms of inadequate core cooling based on existing plant instrumentation and for restoring core cooling following a small break LOCA were submitted on October 31, 1979. This analysis is a less detailed analysis than was originally proposed, and will be followed up with a more extensive and detailed analysis which will be available during the first quarter of 1980. The guidelines and training will be in place by December 31, 1979, as required by the B&O Task Force.

With respect to other transient accidents contained in Chapter 14 of the Indian Point #3 FSAR, the Westinghouse Owners' Group has performed an evaluation of the actions which occur during an event by constructing sequence of event trees for each of the non-LOCA and LOCA transients. From these event trees a list of decision points for operator action has been prepared, along with a list of information available to the operators at each decision point. Following this, criteria have been set for credible misoperation, and time available for operator decisions have been qualitatively assessed. The information developed was then used to test Abnormal and Emergency Operating Procedures against the event sequences and determine if inadequacies exist in the AOP's and EOP's. The results of this study will be provided to the Bulletins and Orders Task Force prior to March 31, 1980.

The Owners' Group has also provided test predictions analysis of the LOFT L3-1 nuclear small break accident. This analysis was provided on December 15, 1979, in accordance with the schedule established mutually with the Bulletins and Orders Task Force.

Containment Pressure Monitor

A review of the existing containment pressure monitoring system design is being conducted to check compliance with Regulatory Guides 1.89 and 1.97. The implementation of the required modifications will be made consistent with the January 1, 1981 date, as specified in NUREG-0578, except where component lead time and equipment availability preclude meeting this date.

Containment Water Level Monitor

A review of the containment water level monitoring system will be conducted to check compliance with the NUREG-0578. Implementation schedule of modifications, if required, will be made consistent with the January 1, 1981 date, as specified in NUREG-0578, except where component lead time and equipment availability preclude meeting this date.

Containment Hydrogen Monitor

A review to ascertain the extent of modifications required to provide continuous indication of hydrogen concentration in the control room will be undertaken. The implementation of the required modifications will be made consistent with the January 1, 1981 date, as specified in NUREG-0578, except where component lead time and equipment availability preclude meeting these dates.

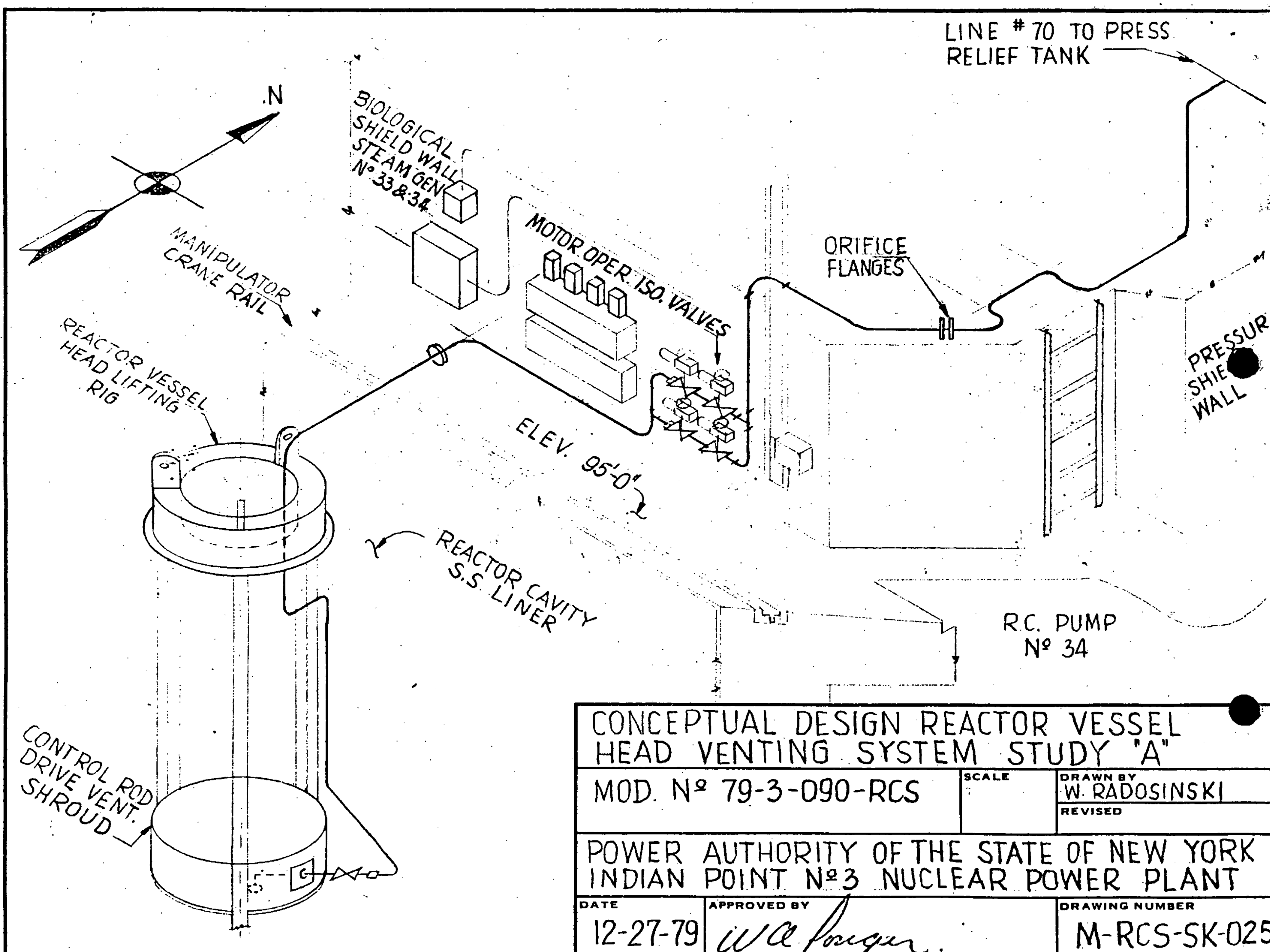
Reactor Head Venting System Description

The reactor Head Venting System is designed to remove gases from the reactor head via remote manual operations from the control room.

A conceptual design of the piping layout has been made to permit purchasing of pipe and fittings. The valves have been built and are now undergoing final testing before shipment to the Indian Point 3 site.

The system connects to the reactor vessel head at the existing vent pipe. From a welded connection at the existing head vent valve, a 3/4" SCH. 160 S.S. vent line runs vertically up the reactor head leg assembly and inside the platform assembly at top of the head. Above the platform assembly, the line runs horizontally toward the steam generator biological sheild. This portion of the piping run contains a disconnect which will be used during removal of the reactor head. Running parallel to the shield wall, the line tee's into the two 1½" SCH. 160 S.S. lines and four Limitorque Motor operated gate valves (two per line). Downstream of the valves the lines reduce back to 3/4" and tee together. This common 3/4" vent line runs atop the shield wall to a flanged restricting orifice and on into the pressurizer relief line and the pressurizer relief tank. By discharging into the pressurizer relief tank, system testing and potential inadvertent releases of water and steam can be accommodated.

The system mainly consists of piping, four 1½" motor operated gate valves and a restricting orifice. Piping upstream and between the valves is Safety Class I and downstream of the valves is Safety Class 2. The four safety Class I valves are provided with Limitorque Model SMB-000 motor operators and meet the requirements of ASME B & PV Code Section III. These valves are remotely operated from the control room and will be maintained in the closed position during normal operations. The restricting orifice is sized to limit letdown flow rate to a maximum of 50 GPM at a reactor coolant system pressure of 1500 psig.



CONCEPTUAL DESIGN REACTOR VESSEL HEAD VENTING SYSTEM STUDY "A"		
MOD. Nº 79-3-090-RCS	SCALE	DRAWN BY W. RADOSINSKI REVISED
POWER AUTHORITY OF THE STATE OF NEW YORK INDIAN POINT Nº 3 NUCLEAR POWER PLANT		
DATE 12-27-79	APPROVED BY W A Josiger	DRAWING NUMBER M-RCS-SK-025

Section 2.2.1.a

Shift Supervisor Responsibilities

1. The Power Authority has had the senior officer for plant operation issue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties. This document will be periodically reviewed and reissued by the highest level of corporate management.

2. Plant procedures have been revised to assure that the duties, responsibilities and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. The shift supervisor has the responsibility and authority to command the control room during normal operation. When he is absent from the control room, he shall designate the senior reactor operator to be in charge.

   During accident conditions the shift supervisor will remain in the control room to direct the activities of the control room operators and the overall operation of the plant.

3. The training program for shift supervisors in the responsibility for safe operation and the management function will be expanded to include the requirements of ANS 3.1 (Draft) Section 5.2.1.8.

4. The Power Authority has had the administrative duties of the shift supervisor reviewed by the senior officer responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuming the safe operation of the plant will be delegated to other operations personnel not on duty in the control room.

Section 2.2.1.b

Shift Technical Advisor

The Authority will provide a shift technical advisor, as required, by January 1, 1980, whenever the unit is above cold shutdown.

Section 2.2.1.c

Shift and Relief Turnover Procedure

1. A check list has been provided for the control room operators as required in this item.

2. A checklist or log has been provided for the auxiliary operators but it is felt not necessary for technicians. Operations personnel remove equipment from service for work done by technicians or authorize removal, and subsequently restore the equipment to service or verify restoration, following completion of the work.

3. Presently independent verification of selected system alignments is periodically made. However, a more formal system has been established to evaluate the effectiveness of the shift and relief turnover procedure.

Section 2.2.2.a

Control Room Access

Administrative procedures have been revised to limit control room access and establish clear lines of authority and responsibility consistent with the requirements of Section 2.2.2.a of NUREG-0578.

Section 2.2.2.b

Onsite Technical Support Center

TEMPORARY ONSITE TECHNICAL SUPPORT CENTER DESCRIPTION

Section 1A. TSC Description

The interim Technical Support Center (TSC) is to be located in the temporary office area[1] immediately adjacent to the Documentation Control Department (See Figure I), thereby providing direct access to drawings, reports or whatever information is needed for emergency assessment.

Plant parameters may be continuously monitored on a cathode-ray (CRT) terminal which will link with the Prodac-250 plant computer.

The TSC is presently scheduled to be complete and operable in its temporary location prior to the plant returning to service.

Section 1.B. Plans and Procedures for TSC Support and Staffing

The TSC will be under the supervision of the site Shift Technical Advisor (STA), who will coordinate all activities therein, until relieved of this responsibility by a senior member of the TSC staff. Notification of necessary personnel to activate the TSC upon declaration of emergency by the Shift Supervisor will also be his responsibility. Procedures outlining operation of the TSC will be used as guidelines.

Section 1.C. Communications

In addition to the computer linkage, there will be dial-up telephone lines between the TSC, the control room and near site emergency center and dedicated lines to the NRC and control room. Furthermore, communication by radio on dedicated frequencies will also be utilitzed.

Section 1.D. Radiation Monitoring

Radiation monitoring equipment consisting of hand-held gamma rate-count meters, alarming rate meters (ARM-16) and air sampling machinery will be available at all times. When not in use, they are to be securely locked in a cabinet by the TSC. Should radioactivity increase beyond certain setpoints defined in the procedures governing the operation of the TSC, evacuation of TSC personnel to the Unit 3 control room will be organized by the STA. Adequate facilities will be provided in the control room whereby the emergency evaluation may be continued.

[1]The temporary offices are located in the south west corner of the site property approximately 500 ft. from Unit #3.

Section 1.E. Access to Technical Data and Plant Parameters

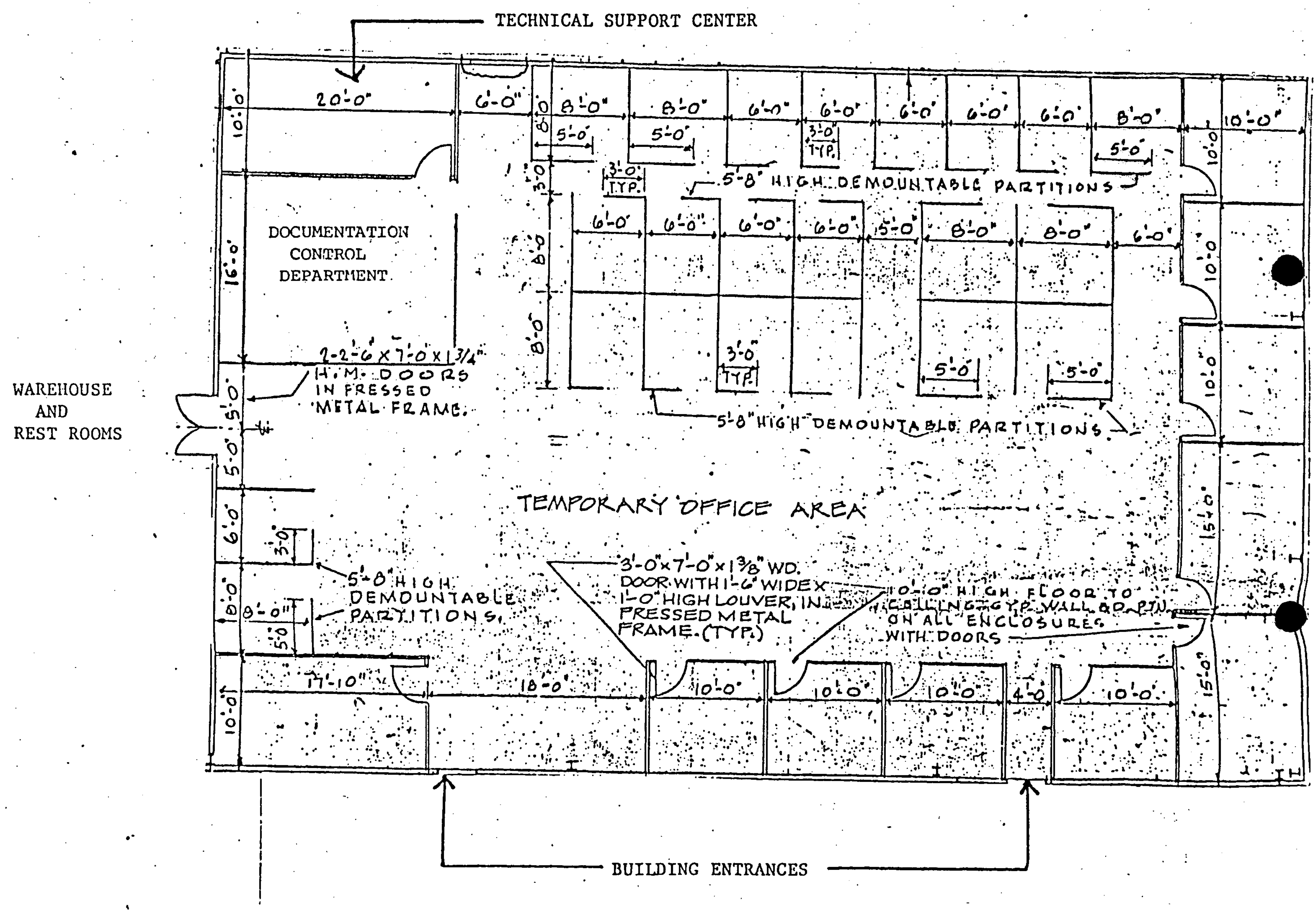
As was previously mentioned, the TSC is adjacent to the Documentation Control Department. In addition to this, its location in the office area enables quick access to the Performance/Reliability and Technical Service Departments, in the event that historical data concerning plant performance is needed.

The plant computer, as accessed through the TSC terminal, is capable of listing or trending much of the information relevant to plant performance which might be needed in an emergency assessment.

Section 1.F. TSC Relocation Procedures

In the event that evacuation to the Unit 3 control room should become necessary, there will be procedures to assist the STA in conducting the accident evaluation from this alternate location. The control room will be provided with the needed documentation to facilitate this analysis.

FIGURE I - TEMPORARY TECHNICAL SUPPORT CENTER LOCATION



PERMANENT ONSITE TECHNICAL SUPPORT CENTER DESCRIPTION

Sections 2 and 3 Location, Physical Size and Staffing

According to present plans, the permanent site of the Technical Support Center (TSC) is to be located on the southwest corner of the second floor of the new PASNY Administration Building.[1] The TSC, covering an area of approximately 3200 square feet, will contain all the instrumentation and documentation required to monitor and assess any plant emergency. The space provided will be enough to house 25 people comfortably, as well as provide an uncluttered work space. (Area has been allocated to provide an independent HVAC).

The TSC will be under the control of the Shift Technical Advisor (STA), who will activate the TSC and coordinate all operations therein until relieved of this responsibility by a senior member of the TSC staff.

Sections 4 and 5 Activation and Instrumentation

Upon activation of the TSC, the staff will be able to monitor any needed plant parameters by way of a CRT display and hard copy output attached to the plant computer. Dedicated telephone lines and closed circuit video monitoring will provide direct communication with the control room. The capacity of the system will be such that historical data from the time the accident began can be retrieved and recorded.

The interface between plant and TSC monitoring equipment will be designed so as to interfere in no way with any control room instrumentation, nor to degrade its function.

Section 7 Technical Data

All needed documentation will be available in the TSC or from the document storage area located one floor above by which the event may be assessed and analyzed. This will include drawings, graphs, and other references containing technical information. All of the parameters relevant to the safety of a Westinghouse plant will be continuously monitored in the TSC.

[1]The new Administration Building is being constructed on the south of the turbine hall.

Section 6 Instrumentation Power Supply

All of the power to operate the TSC instrumentation will come from an essential 480 v bus powered by diesel generators in the event of loss of offsite power. Habitability and air conditioning systems are presently planned to be provided by independent equipment which is to be located in a separate HVAC room adjacent to the TSC.

Section 8 Data Transmission

In addition to direct link with the Control Room, there will also be a dedicated line to the Near Site Emergency Center and to the NRC. The Power Authority is presently investigating the technical possibilities of transmitting plant parameter data to other locations, such as Westinghouse, or other offsite agency.

Section 9 Structural Integrity

The present plans for the TSC place it in the PASNY Administration Building, which is classified as Seismic Category II.

In the event that external conditions (radioactivity, natural phenomena) require evacuation of the TSC, efficient backup facilities, including necessary telephone lines and documentation, will be available in the Control Room, which is easily accessible from the planned TSC site.

Section 10 Habitability

At all times during operation of the TSC, permanent radiation monitoring will be utilized. Monitoring systems with alarms in both the TSC and the Control Room will also be used. The TSC will have an independent air conditioning system including particulate and charcoal filters.

Under no conditions will the inhabitants of the TSC be permitted to receive doses greater than those specified in GDC 19 and SRP 6.4(i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the incident).

In addition to the ventilation system and radiation monitoring equipment, breathing apparatus and potassium iodide tables will be available to all TSC personnel.

In the event that radiation levels reach the point at which the TSC becomes uninhabitable, as defined by procedures which will govern its operation, the STA will coordinate evacuation to the Control Room, from which accident assessment may be continued.

Section 2.2.2.c

Onsite Operational Support Center

The onsite operational support center will be located in the lunch room building adjacent to the Unit 3 turbine building. It will have telephone communications with the control room and the I & C and maintenance offices whose management personnel will coordinate the efforts of the mechanics and technicians in the center. Communications will be established prior to the unit returning to service.

Items Covered by Enclosures 7 and 8
to the September 13, 1979 NRC Letter

Near Term Emergency Preparedness Improved Implementation

1) Upgrade Emergency Plan

A meeting was held in Westchester County with the NRC at which time twenty-five (25) requests for modifications to the Authority's plan was made by the Commission. These requests will be incorporated into the plan by 2/1/80.

2) Short Term Actions Recommended by Lessons Learned Task Force

See responses given in Sections 2.1.8.a, 2.1.8.b and 2.1.8.c of this attachment.

3) Emergency Operation Center for Federal, State and Local Officials

The temporary emergency control center for Federal, State and Local Officials has been established for the Indian Point 3 Nuclear Plant. The Authority will upgrade this center in conjunction with the development of the permanent in-plant technical support center by January 1, 1981.

4) Improved Off-Site Monitoring Capability

The additional requirement specified by the NRC during the 12/18/79 meeting in Westchester County is being incorporated into the off-site monitoring program and will be implemented prior to returning this unit to service.

5) Adequacy of State/Local Plans

The New York State Radiological Response Plan is an approved NRC plan. It only has to be upgraded in some areas. The Authority in cooperation with Consolidated Edison Co., has offered the local counties technical expertise to write a model emergency plan that could be made county specific by them. Work on the model plan for the counties has commenced.

6) Conduct of Test Exercises

Section 8.1.2 Drills and Exercises of the revised Indian Point 3 Emergency Plan provides for a joint exercise involving Federal, State and Local response organizations to be conducted once every five years in addition to emergency plan tests already specified in Book 1 of the plan itself.