POWER AUTHORITY OF THE STATE OF NEW YORK

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February 3, 1980

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
TMI Lessons Learned Supplemental Response

Dear Sir:

This letter supplements our January 8, 1980 letter entitled, "Short Term Requirements of TMI Lessons Learned", which was submitted in response to the Commission's October 30, 1979 letter.

The supplemental responses contained in the Attachment and indicated by bar lines, are provided as a result of the Commission's January 23, 24, and 30, 1980 inspection of the IP3 facility.

Very truly yours,

for Paul J. Early
Assistant Chief Engineer-
Projects

A039
3/1

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

WAJ/jd

8002070 345

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 Motor Control Center (MCC) and a constant voltage transformer The MCC can also be tied directly to the emergency diesel from the control room.

In order to independently verify the above, please find enclosed.

1. Westinghouse Drawing 617F645 - Main One Line
2. Consolidated Edison Dwg. A210662 - 480V one line
3. United Engineers & Const. Dwg. 9321-F-30063 - 480V & Instrument bus one line
4. United Engineers & Const. Dwg. 9321-F-30083 - DC System one line 1
5. United Engineers & Const. Dwg. 9321-F-31993 - Wiring diagram - Instrument Bus 31 & 32
6. United Engineers & Const. Dwg. 9321-F-32003 - Wiring diagram - Instrument Bus 33 & 34
7. Foxboro Dwg. FA-1 - Rack Layout
8. Foxboro Dwg. PW-1 - Terminal block wiring

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 1
Energy Corporation" (TEC) of Knoxville, Tenn. was 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 1
converts to voltage. The clamping device used to secure the
accelerometer to the discharge pipe has been thermally analysed by
TEC to assure that it will remain firmly attached after valve
actuation. 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. 1
The bar graph is labeled in relative valve flow with 1.0 conser-
vatively representing a value of less than full flow and is located
in the control room. Discrete valve flow indications are .01, .04, 1
.09, .16, .25, .36, .49, .64, .81, and 1.0. The present alarm
setpoint as set by the manufacturer is 0.25. TEC has selected this
as an optimum setting based on their own testing program. Should
there be any significant leakage from any valve, this will initiate
the alarm. As more operating experience is gained with this unit,
the alarm setpoints may require resetting.

The annunciator setpoints for each valve can be adjusted by adjusting strap wires on the PC boards. The annunciator will be activated when any one of the three (3) safety or 1
two (2) relief valves reach their alarm setpoint. The operator will know which of the five (5) safety/relief valves have opened from the visual flow reading (bar graph) on the front panel. Cross-talk between certain valves will occur, however, the open valve will give a considerably higher flow reading than the valve which is experiencing mere cross-talk. Those valves which do open will be clearly obvious.

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, TEC has adopted the design position that the system will one day be required to meet full 1E requirements. The design and testing program, which TEC has followed, assures that this system will meet environmental qualifications as per IEEE-323-1974 and seismic qualifications as per IEEE-344-1975. TEC's program is at this time still being documented for their customers, however, we expect to receive a copy of 1
their report on environmental and seismic testing performed to date and a schedule for completion of all final testing within the very near future. Similar systems provided by other manufacturers are also in the same situation in that their systems have yet to be post qualified to Class 1E standards.

Final installation of the TEC system was performed during the month of January and it has already been tested and determined to be operational. In addition, an alarm system was added in the control room to the existing Pressurizer Power Operated Relief Valves as was requested by the NRC staff. The plant operating procedures have been revised to reflect these new systems.

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. Subcooling Meter

1. P-250 - Saturation Program

The plant computer has been programmed to display the pressure above saturation condition. This will be our primary system until our analog saturation system is completed. The program computes a saturation pressure based on the average incore thermocouple temperature. The reactor coolant pressure (taken from the lowest reliable value of 2 sensors) is then compared to this saturation pressure and the difference is stored as an addressable calculated value which can be displayed on the operator's CRT, computer visual display or logged on the computer trend recorders or trend printer. One of these continuous display methods (i.e., CRT, Trend recorder or computer visual display) will be utilized to continuously display the saturation margin when the reactor is above cold shutdown.

Two alarms are provided in the system. The first one is "Approaching Saturation Condition" which is set at 300 psi above saturation. The second one is "At Saturation". Both conditions would result in a message on the alarm printer and the activation of the appropriate board annunciator.

This system has been installed, tested and fully complies with all the NUREG requirements.

2. Analog Saturation System

A subcooling meter will be installed which
uses four wide range hot leg RTD's (1 per
loop) and two pressure inputs. The temperature
inputs will be channeled to a high temperature
selector unit which feeds a saturation pressure
calculator unit. The two pressure inputs
will be channeled to a low pressure selector
unit which will be compared to the output of
the calculator unit. The difference between
these two analog signals will be continuously
displayed by a strip chart recorder. The
system's power will be supplied via a vital
power supply. Testability has also been
designed into the system. The majority of
the system has been installed and tested,
i.e., a single temperature and pressure input
system. However, due to component lead time,
redundant temperature and pressure inputs 1
have not been installed.

Two alarms are provided in the system. The first one is "Approaching Saturation Condition" which is set at 300 psi above saturation. The second one is "At Saturation". We believe that the alarm setpoints as chosen, will alert the operator adequately that a saturation condition is approaching and provide ample time for the operator to take corrective action before saturation conditions are achieved.

It is our opinion that the final system with four RTD inputs, one from each hot leg, as opposed to redundant RTD's for each loop is adequate temperature input redundancy to provide reliable indication of the subcooled condition of the reactor core. Indian Point Unit #3 does not have reactor coolant loop isolation valves and only has one direct emersion RTD in each hot leg. It is our opinion that four temperature inputs combined with the high selector device provides adequate redundant temperature inputs and we would like for you to reconsider your position in light of this plant specific information.

The high and low selector devices have 16-18 week lead time, therefore, when these components become available they will be installed. No plant outage is required to install these devices.

SINGLE TEMP./PRESS. METER

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 (°F, PSI)	∿ 31 psi
Range of Display	1500 psi above saturation
Qualifications (seismic, environmental, IEEE323)	Per original plant design 1

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)	Single pressure and temperature
Qualifications (seismic, environmental, IEEE323)	Per Original Plant Design
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

Input

Temperature (RTD's or T/C's)	RTD
Temperature (number of sensors and locations)	1 RTD, hot leg piping
Range of temperature sensors	0-700 °F

Uncertainty of temperature sensors (°F at 1)	3.5°F
Qualifications (seismic, environmental, IEEE323)	Note 1
Pressure (specify instrument used)	PT-403
Pressure (number of sensors and locations)	1 RCS Hot Leg Piping
Range of Pressure sensors	0-3000 psig 1
Uncertainty of pressure sensors (PSI at 1)	15 psi

Backup Capability

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

Note 1: 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, Westinhouse 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 and IEEE 344.

The RTD temperature sensor was qualified via Westinghouse WCAP 9157.

REDUNDANT TEMP./PRESS. METER

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 (F, PSI)	*
Range of Display	1500 psi above saturation
Qualifications (seismic, environmental, IEEE323)	Per original plant design

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)	Lowest pressure & Highest temperature
Qualifications (seismic, environmental, IEEE323)	Per Original Plant Design
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

Input

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

*Will be supplied at a later date.

Uncertainty of temperature sensors (°F at 1)	3.5°F
Qualifications (seismic, environmental, IEEE323)	Note 1
Pressure (specify instrument used)	PT-403, PT-402 1
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)	15 psi (for PT-403) 12.5 psi (for PT-402)

Note 1: 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, Westinhouse 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 and IEEE 344.

The RTD temperature sensor was qualified via Westinghouse WCAP 9157.

COMPUTER SATURATION METER

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	P-Psat
Display Type (Analog, Digital, CRT)	CRT + Digital
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Computer Operator's Desk or Console
Alarms (include setpoints)	Saturation and 300 psi above s
Overall Uncertainty (psi)	~ +6.4, -22 psi
Range of Display	99999.9 psig above sat.
Qualifications (seismic, environmental, IEEE323)	Per Original Plant Design

Calculator

Type (process computer, dedicated digital or analog calc.)	Process Computer
If process computer is used specify availability. (% of time)	95% +
Single or redundant calculators	Single
Selection Logic (highest T., lowest press)	Average T, lowest pressure
Qualifications (seismic, environmental, IEEE323)	Per Original Plant Design
Calculational Technique (Steam Tables, Functional Fit, ranges)	Functional Fit 100-700 °F

Input

Temperature (RTD's or T/C's)	T/C's
Temperature (number of sensors and locations)	65 core exit
Range of temperature sensors	0-700 °F

Uncertainty of temperature sensors (°F at 1)	18.3 °F
Qualifications (seismic, environmental, IEEE323)	Note 1
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)	15 psi for PT-403, 12.5 psi for PT-402

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Note 1: 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 and IEEE 344.

Section 2.1.4

Containment Isolation Provisions for PWRs and BWRs

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 non-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.

The selection of each system as essential or non-essential is based on the following criteria;

Essential - Lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also lines which, if available, would be used in the short term (24-36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation.

Non-Essential - Lines which are not required to mitigate or limit an accident, which if required at all would be required for long term recovery only, i.e., days or weeks following an accident.

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 have been revised to re-emphasize to the operating personnel that these asterisked (*) valves may be opened only when absolutely essential for plant operations or testing, that an operator must be dedicated to the operation of these valves as long as they are in the open position, that their first response to any emergency condition while the valve is open is to insure that the valve is returned to the closed position and that these valves must be immediately returned to the shut position upon completion of the task.

Additionally, these manual valves located on non-essential systems do not receive a containment isolation signal but comply with the intent of NUREG 0578 with respect to the definition in the FSAR section 5.2.2 subparagraph 3 which states:

"Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operations qualify as automatic trip valves".

A) N_2 to PRT Valve No. 550

This valve is used to supply N_2 to the PRT so as to 1
insure an inert atmosphere exists in the PRT. This valve is maintained in the locked shut position until its use is deemed necessary by the senior reactor operator on shift who then orders the valve to be opened so that N_2 can be admitted into the PRT. Immediately upon completion of this task, the valve is returned to the locked shut position. Additionally, it is noted that downstream of this valve is a containment isolation check valve which would prevent any back flow through this line should containment be subjected to the incident pressure. Lastly, the N_2 pressure exceeds the accident pressure thus any leakage would be into containment and not out to the environment.

B) RHR Loop Sampling System Valves 958, 959 & 990C.

These valves are used to sample the RHR loop for boron concentration prior to placing the RHR system in service. This sample is usually taken as the unit is being shutdown and is rarely if ever used during power operations. As such, these valves are maintained in the shut position during normal operations.

C) Safety Injection Pump Test Line Valves 859A & 859C.

These valves are used to provide a flow path from the SIS pumps to the RWST through a flow indicator so as to measure pump flow during periodic testing of the pumps as required by technical specifications. These valves are maintained in the locked shut positions during all other times and are verified shut upon completion of the periodic test.

D) N Supply to the Accumulator Valves 891A, 891B, 891C, 891D and 863.

These valves are used to supply N_2 to the accumulators so as to maintain pressure as required by the technical specification. When required, the senior reactor operator on shift directs the opening of valve 863 and then opens the isolation valve to the appropriate accumulator. Upon completion of the N_2 pressurization task, all valves are returned to the shut position. Additionally, a modification has been performed which provides a check valve downstream of valve 863. This valve has been successfully tested as a containment isolation valve and a technical specification change submitted to the NRC to remove the 891A, B, C & D valves as containment isolation valves. This check valve will prevent back flow through this line should containment be subjected to the incident pressure.
Also, the N_2 pressure exceeds the accident pressure 1
thus any leakage would be into containment and not out to the environment. Lastly, all these valves are air operated and fail shut upon loss of air which would occur with a containment isolation signal. These valves are maintained in the shut position except as noted above with indication and control in the central control room.

E) N_2 to RCDT Valve 1610

This valve is similar in function to valve no. 550, N_2 to the PRT, except it supplies N_2 to the RCDT. It too has a check valve located upstream of valve 1610 which performs the same function as described in (A) above. This valve is locked shut during normal operations except as previously noted.

F) Station Air SA-24 (Two valves)

Both these valves are locked shut above cold shut down conditions and are not opened during normal operations.

G) Pressurizer Pressure Dead Weight Calibrator Valves 580A and 580B.

Both these valves are locked shut above cold shut down conditions and are not opened during normal operations.

H) Auxiliary Steam Supply and Return Valves UH-37 and UH-38

Both these valves are locked shut above cold shut down conditions and are not opened during normal operations.

A review of the containment isolation system at Indian Point No. 3 indicated that there were a number of valves which automatically reset to the previous position upon reset of containment Phase A isolation. These valves were under operator control via operating procedures to be placed in the closed position prior to resetting Phase A. These valves have been modified to preclude automatic opening on reset. The modification to the valve circuits entailed the installation of pushbuttons that work in conjunction with the Containment Isolation Reset Switches so that each valve control circuit has to be reset or the valve will be inhibited from opening.

In some instances, these push buttons are associated with more than one valve as well as more than one containment penetration. These are identified and discussed below but it should be noted that the design of our system incorporates the philosophy of NUREG 0578 which states:

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

In no instance will resetting of the containment isolation signal result in the automatic reopening of any containment isolation valves. Deliberate operator action is required to open the containment isolation valves.

The following reset buttons are associated with several valves which have been grouped together because they are associated with the same system.

A) <u>Reset buttons 5 and 11</u> 1

Both these reset buttons control the steam generator blowdown and sample valves. Each button controls only one set of the two containment isolation valves associated with each steam generator. Therefore, resetting only one of the buttons will allow opening of only one of the containment isolation valves in each of the above lines but will not allow the opening of a flow path unless the other button is also depressed. Thus, an operator would have to take two deliberate independent actions after he reset the containment isolation signal before he could establish the steam generator blowdown or sample system.

B) <u>Reset button 17</u>

This button had previously been designed to control valves 1789 and 1787 which are containment isolation valves for the RCDT to the vent header and to the gas analyzer. The revised design and presently installed system is such that reset button 17 controls only valve 1789 and reset button 17A controls valve 1787.

C) <u>Reset button 18</u>

This button controls two "A" containment isolation valves (PRT to the gas analyzer and PW to the PRT). Two additional independent reset buttons are required to open the remaining "B" containment isolation valves (reset buttons 15 and 25). An operator would have to take a minimum of two deliberate independent actions after containment isolation had been reset before a flow path could be established from either the PRT to the gas analyzer or from the PW system to the PRT.

D) Reset buttons 1 and 2

Both these reset buttons control the RCS and accumulator sample valves. Each button will control only one set ("A" or "B") of the containment isolation valves associated with each of these lines. Thus, resetting button 1 will allow opening of the "A" containment isolation valves but will not result in opening a flow path unless reset button 2 is also depressed. Therefore, an operator would have to take two deliberate independent actions after resetting the containment isolation signal before flow could be established from the RCS of Accumulator Sample Lines.

E) Reset buttons 1A and 2A

These buttons parallel the operation of reset buttons 1 and 2 (for RCS sample valves only) and are provided should access to buttons 1 and 2 be prohibited by high radiation.

F) Reset buttons 7 and 13

Both these buttons control the pressurizer steam and liquid
space sample valves. Each button controls only one set of
containment isolation valves associated with these two
lines. Resetting button 7 will allow opening of the "A" 1
containment isolation valves but it will not result in
opening a flow path unless reset button 13 is also depressed.
Thus, an operator must take two deliberate independent
actions after resetting the containment isolation signal
before flow could be established from the pressurizer steam
or liquid space sample lines.

G) Reset buttons 20 and 27

Both these reset buttons control the RCDT pump discharge valves and the VC sump pump discharge valves. Resetting of either button will control only one valve in each of these lines. The other button must also be pushed in order to allow a flow path to exist from either the RCDT or VC sump. Again, as with the previously discussed reset buttons, the operator would have to take two deliberate, independent actions after the containment isolation signal was reset in order to establish flow from these lines.

Again, in all cases, the containment isolation reset system has been designed to meet the criteria as set forth in NUREG 0578. However, as a result of a meeting between the NRC and IP3 Plant personnel at the Site on January 30, 1980, clarification was given of the intent of NUREG 0578 with respect to the requirements of the containment isolation reset system. Based on this information, we will re-review our containment isolation reset system to determine if further modifications are necessary to insure that multiple penetrations are not needlessly reset. As a short-term measure, until this review has been completed, and additional hardware changes made, we have incorporated into our procedures for resetting the containment isolation signal after a bonified containment isolation, the requirement that prior to resetting any of the multiple function reset buttons permission to do so must be secured from the Superintendent of Power or his alternate. Exception was made for those reset buttons which would be required to sample the areas of the plant so that the initiating incident could be properly evaluated. Additional exception was made for those systems which require more than one penetration to be opened in order for the equipment to function, i.e., inlet and outlet lines for the containment air sample.

In summary, in order for a flow path to be established through a containment penetration, the following would occur: 1

1) The operator, in following the emergency procedure PEP-ES-1F (Resetting Phase "A" Containment Isolation) would shut each valve control switch.

2) After verifying the above action, the operator would then reset the containment isolation signal by depressing two pushbuttons on the CCR panel SNF.

   NOTE: This will <u>not</u> result in the opening of any containment isolation valve.

3) The operator is then instructed to:

   1. Get permission to reset a reset button which would affect multiple penetrations if applicable.

   2. Open the individual control switch for the valve desired to be opened.

   NOTE: This does not result in the opening of any containment isolation valve.

4) The operator would then depress not just one reset button but two reset buttons <u>and</u> they must be the matched pair buttons in order to establish any flow through a containment isolation valve.

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.
885A	Safety Injection	RHR Suction	Essential - P.A.	" " " " "
885B	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 Ops.
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 2.1.6.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 Power Authority has established a program to identify and reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident.

I. Systems Included in Program

The following systems are included in the subject program for leak identification and reduction.

1. Chemical and Volume Control System

   A. Volume Control Tank up to the outlet isolation valve including gas space.

2. Residual Heat Removal System 1

   A. Suction to pumps from Loop 32

   B. Discharge of pumps to containment

3. Safety Injection System

   A. Recirculation path from containment sump thru RHR pumps to RHR heat exchanger and to S.I. Pump Suction.

   B. Recirculation path from recirculation pump discharge to S.I. Pump Suction.

   C. S.I. Pump discharge path to containment.

4. Primary Sampling System

   A. Reactor Coolant Hot Leg Sample

   B. Recirculation Pumps Sample

   C. RHR Loop Sample

   D. Volume Control Tank Sample up to the outlet isolation valve.

5. Post Accident Containment Air Sampling System

   A. Sample from containment and return to containment

The initial leak testing of the above listed systems has been completed and the results are as follows:

1)	Volume Control Tank	0
2)	Residual Heat Removal System	31.5 cc/min 1
3)	Safety Injection System	3 cc/min
4)	Primary Sample System	1 drop/hr.
5)	Post Accident Containment* Sample System	0
6)	Modified Post Accident Containment Sample System	10 cc/min.

*Initial test showed that this system leaked 90 bubbles/sec. All leakage was repaired.

The following systems are excluded from the subject program. These systems would not be used to process highly radioactive fluids outside the containment during a serious transient or accident:

Component Cooling[2]

Spent Fuel Pool Cooling[2]

Liquid Waste Disposal[3 & 4]

Primary Makeup Water[1&4]

Steam Generator Secondary Systems[2]

Containment Spray (Injection Phase)[1]

Containment Air Recirculation Cooling and Filtration[5]

Isolation Valve Seal Water System[1] 1

Weld Channel and Cont. Pene. Press.[1]

Hot Penetration Cooling[2]

Hydrogen Recombiners[5]

Containment Ventilation [4]

Service Water[2]

Auxiliary Steam and Condensate[2]

Fire Protection[1]

Gaseous Waste Disposal System[4]

Compressed Air and Gases (Hydrogen, Oxygen, Nitrogen)[1]

Accumulator Samples[3]

Post Accident Containment Venting System[3]

Containment Atmosphere Radiation Monitor (R-11 & R-12)[4]

Those portions of the CVCS, SIS and PSS not included in the program.[3]

NOTES:

1. System provides fluid supply to various systems and would not be used to process highly radioactive fluids outside the containment.

2. System is isolated from highly radioactive fluids and would be isolated in the event of significant contamination and not used to process fluids. 1

3. System would not be used to process highly radioactive fluids.

4. System isolated from radioactive fluids in event of accident.

5. System totally within containment and would not process highly radioactive fluids outside the containment.

Design and operator deficiencies discussed in your letter regarding North Anna and related incidents have 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 + 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 during 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 conjunction 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 for personnel access was based on following limits of acceptability: 1

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

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

All vital areas and equipment requiring access were addressed in the study including paths of access. These components, valves and areas which require access to aid in post accident recovery operations were identified as follows: 1

1. PAB Entrance (EL 55')
2. Waste Disposal Control Panel (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 (EL 41 PAB)
12. Control Room
13. Diesel Generators
14. Auxiliary Feedwater Pumps
15. Health Physics Building
16. Nuclear Plant Operator (NPO) Office

The results of the design review for personnel access to the above areas are presented in Table 1 along with recommendations 1
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.

In conjunction with the design review of personnel accessibility to vital areas following an accident, a review was conducted of equipment which could be unduly degraded by the radiation fields due to the same sources that are identified in Table 1. The results of the equipment qualification review are presented in Tables 2 through 5 for instrumentation, electrical equipment, 1
mechanical equipment and valves, respectively. To date no safety related equipment has been found to be unduly degraded, but as can be seen for the attached tables many items require further investigation to totally document their ability to operate in these postulated radiation fields. It is anticipated that this continuing review will be completed by April 30, 1980, at which time the results will be transmitted to you.

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 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 . 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 with 6 feet of concrete. For a time of 1 day with 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

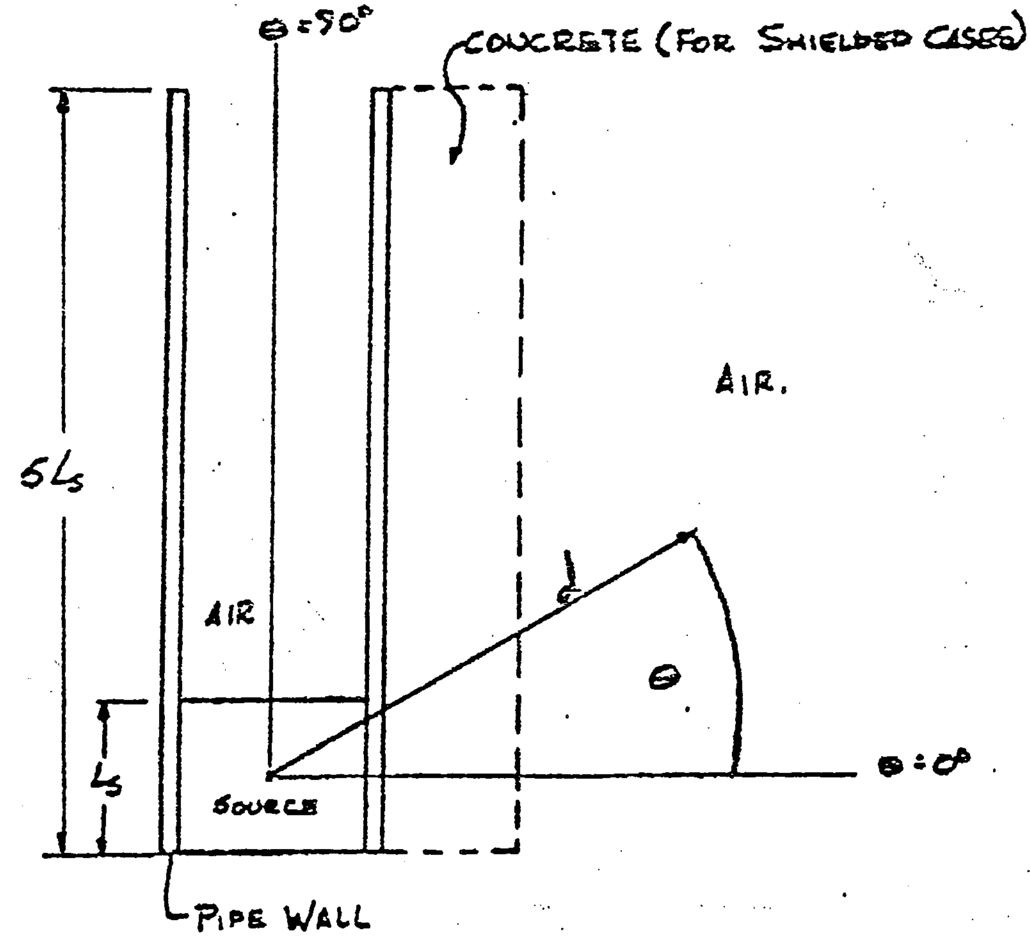
united engineers

DISCIPLINE

NAME OF COMPANY ____________ UNIT/S ________

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

SUBJECT PIPE DOSE COMPUTER MODEL



1

FIGURE 1

TABLE 1

PERSONNEL ACCESS DESIGN REVIEW RESULTS

	Point of Interest	Sources Investigated	Evaluation Criteria (note 1)	Results of Review	Remarks
1.	PAB Entrance	Containment* R.C. Sample* Pipe Trench (RHR, SI etc.)	b	Major Source dose rate is due to the R.C. Sample and the containment. Results are acceptable.	Investigate shielding in sampling room
2.	Waste Disposal Panel	(Same as above)	b	(Same as 1 above)	
3.	Motor Control Centers	(Same as above)	b	(Same as 1 above)	
4.	Sampling Room	(Same as above)	b	Additional dose reduction is required	Possible Fixes Include: Automatic Sampling Syster Local Shielding Revised Sampling Procedures.
5.	Fan Bldg. Stairway	Containment* Pipe Trench (RHR, SI*, etc.)	b	Acceptable	
6.	Hydrogen Recombiner Area	Containment* PAC Air Sample RC Sample Vice Failed Fuel Dectector Pipe Trench (RHR, SI*, etc.)	b	Additional dose reduction is required	Possible Fixes Include: Additional Shielding Remote Operation Rerouting Pipe Panel Relocation
7.	Post Accident Containment Air Sampling Station	Containment* Containment Air Pipe Trench	b	Additional dose reduction is required.	Possible Fixes Include: Local Shielding Valve Operator Extension Automatic Valves Shielded Sample Pig Remote Automatic Analysi

TABLE 1
(Cont'd)

Point of Interest	Sources Investigated	Evaluation Criteria (note 1)	Results of Review	Remarks
8. Manual Containment Isolation Valves	Containment* RHR Piping SI Piping*	b	Additional dose reduction is required.	Possible Fixes Include: Valve Operator Extensions (must confirm with above) Automatic Valve Operators Revised Procedures Shielding
9. Safety Injection Pump Room	SI Piping* Pipe Trench (RHR, CVCS, etc.)		Acceptable	No personnel access required.
10. RHR Pump Room	RHR Piping* Pipe Trench (CVCS, WDS, etc.)		Acceptable	No personnel access required.
11. Laboratory	Pipe Chase	b	Acceptable	
12. Control Room	Containment*	a	Acceptable	
13. Diesel Generators	Containment	b	Acceptable	
14. Aux. Feedwater Pumps	Containment	b	Acceptable	
15. Health Physics Bldg.	Containment	a	Acceptable	
16. NPO Office	Containment	a	Acceptable	

NOTES: 1. a) Continuous Occupancy $\leq$ 15 mr/hr
b) Infrequent Access $\leq$ 5 rem Whole Body

* Source Used

TABLE 2

REVIEW OF POST-ACCIDENT MONITORING INSTRUMENTATION OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	INSTR. TAG NO.	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Pressurizer Pressure Indicator: W Type 252 Spec. Sht. #4.11 P.O. #107489	PT-455B	PAB	55'-0"	F-70403	The integrated dose at this location is less than 10 rads for a one year time period. Equipment is acceptable.
Pressurizer Level Indicator: W Type 252 Spec. Sht. #4.11 P.O.107489	LI-459B	PAB	55'-0"	F-70403	The integrated dose at this location is less than 10 rads for a year time period. Equipment is acceptable.
Containment Bldg. Cooling Water Flow Transmitters: Foxboro Model E-13DM	FT-1121 FT-1122 FT-1123 FT-1124 FT-1125	Pipe Penet. Area	41'-0"	F-70433	Item is acceptable based on Foxboro Test Reports #T2-1075, T3-1068 & T3-1097 (10^7 rads integrated dose). The integrated dose at this location is less than 10 rads for a one year time period.
Containment Bldg. Pressure Transmitters: Foxboro Model E-11GM; W Spec. Sht. #9.17 P.O.#104275	PT-948A PT-948B PT-948C PT-949A PT-949B PT-949C	Pipe Penet. Area	41'-0"	F-70433	Item is acceptable based on Foxboro Test Reports #T2-1075, T3-1068 & T3-;097 (10^7 rads integrated dose). The integrated dose at this location is less then 10 rads for a one year time period.
Main Steam Pressure Transmitters: Foxboro Model E-11GM; W Spec. Sht. #4.90 P.O. #104275	PT-419A PT-419B PT-419C PT-429A PT-429B PT-429C PT-439A PT-439B PT-439C PT-449A PT-449B PT-449C	Aux. P.W. Pump Bldg.	18'-6"	P-70533	Item is acceptable based on Foxboro Test Reports: T2-1075, T3-1068 & T3-1097 (10^7 rads integrated dose). The integrated dose at this location is less than 10 rads for a one year time period.

TABLE 2 (cont'd)

REVIEW OF POST-ACCIDENT MONITORING INSTRUMENTATION OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	INSTR. TAG NO.	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Aux. F.W. Flow Foxboro Model E-13DM; UE&C Spec. #252-34	FT-1200	Aux. F.W. Pump. Bldg.	18'-6"	F-70533	Item is acceptable based on Foxboro Test Report: T2-1075, T3-1068 & T3-1097 (10^7 rads integrated dose). The integrated dose at this location is less than 10 rads for a one year period.
Feedwater Pressure Transmitters: Fischer & Porter Model #50EP1000 Serial # 7010A3163A UE&C Spec. #252-9	PT-1181 PT-1182 PT-1183 PT-1184	Aux. F.W. Pump Bldg.	18'-6"	F-70533	No vendor radiation data available. The integrated dose at this location is less than 10 rads for a one year time period.
Feedwater Flow Transmitters; Foxboro Model E-13DM; W Spec. Sht. #4.49 P.O. #104275	FT-418A FT-418B FT-428A FT-428B FT-438A FT-438B FT-448A FT-448B	Aux. F.W. Pump Bldg.	18'-6"	F-70533	Item is acceptable based on Foxboro Test Reports #T2-1075, T3-1068 & T3-1097 (10^7 rads integrated dose). The integrated dose at this location is less than 10 rads for a one year period.
Aux. F.W. Pump Suction Pressure Transmitter: Foxboro Model E-11GM UE&C Spec. #252-34	PT-1260 PT-1261 PT-1262	Aux. F.W. Pump Bldg.	18'-6"	F-70533	Item is acceptable based on Foxboro Test Reports #T2-1075, T3-1068 & T3-1097 (10^7 rads integrated dose). The integrated dose at this location is less than 10 rads for a one year time period.
Aux. F.W. Pump Discharge Pressure Transmitters: Foxboro Model E-11GM UE&C Spec. #252-34	PT-1263 PT-1264 PT-1265	Aux. F.W. Pump Bldg.	18'-6"	F-70533	Item is acceptable based on Foxboro Test Reports #T2-1075, T3-1068 & T3-1097 (10^7 rads integrated dose). The integrated dose at this location is less than 10 rads for a one year time period.

TABLE 2 (cont'd)

REVIEW OF POST-ACCIDENT MONITORING INSTRUMENTATION OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	INSTR. TAG NO.	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Pressurizer Pressure Indicator: W type 252 Spec. Sht. #4.11 P.O. #107489	FI-455A	Aux. F.W. Pump Bldg.	18'-6"	F-70533	The integrated dose at this location is less than 10 rads for a one year period. Equipment is acceptable.
Pressurizer Level Indicator: W Type 252 Spec. Sht. #4.11 P.O. #107489	LI-459A	Aux. F.W. Pump Bldg.	18'-6"	F-70533	The integrated dose at this location is less than 10 rads for a one year period. Equipment is acceptable.
Steam Generator Level Indicators: W Type 252 Spec. Sht. #4.11 P.O. #107489	LI-417D LI-417E LI-427D LI-427E LI-437D LI-437E LI-447D LI-447E	Aux. F.W. Pump Bldg.	18'-6"	F-70533	The integrated dose at this location is less than 10 rads for a one year time period. Equipment is acceptable.
Main Steam Isolation Valve Control Panel; Atwood-Merrill Order #11762, Dwg. #20702-H; UE&C Spec. #248-1A	Part of MS-1-31 MS-1-32 MS-1-33 MS-1-34	Aux. F.W. Pump Bldg.	69'-8" & 80'-0"	F-70313	The integrated dose at these locations is less than 10 rads for a one year time period. Equipment is acceptable.

TABLE 2 (cont'd)

REVIEW OF POST-ACCIDENT MONITORING INSTRUMENTATION OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	INSTR. TAG NO.	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Isokinetic Stack Gas Monitor; Furnished by W Tracerlab Dwg. #0998164, #D998165	R-14	Stack	124'-5"	F-70453	The integrated dose at this location is approx. 20.4 rads for a one year time period. Equipment is acceptable.
Hydrogen Recombiner Control Panels; Furnished by W		Fan House	67'-6"	F-25183	The integrated dose at this location is 1.275 (10^4) rads for a one year time period. Further investigation is required.

TABLE 3

REVIEW OF ELECTRICAL EQUIPMENT OUTSIDE CONTAINMENT
FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	EQUIP. DESIGNATION*	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Component CLG PP 31 Motor	M43	PAB	41'-0"	9321-F-30843	Note 1
Component CLG PP 32 Motor	M44				
Component CLG PP 33 Motor	M45				
PAB Supply Fan Motor	T4				
PAB Con Ret UN 34 PP 34A Motor	MV7				
PAB Con Ret UN 34 PP 34B Motor	MV8				
Contmnt Spray PP 31 Motor	M52				
Contmnt Spray PP 32 Motor	M53				
PAB Cond Ret Mech Alt	LT8				
Residual Heat PP 32 Motor	M47		15'-0"	9321-F-30863	
Residual Heat PP 31 Motor	M46				

Note 1: Equipment qualified to 10^6 rad or higher, per supplement 21 to the IP-3 FSAR, Section 3.5.C, Page 6F-5, August, 1973. This is acceptable.

*FROM EQUIPMENT LIST FORM 2 OF CASP COMPUTERIZED CONDUIT & CABLE SCHEDULE.

TABLE 3 (cont'd)

REVIEW OF ELECTRICAL EQUIPMENT OUTSIDE CONTAINMENT
FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	EQUIP. DESIGNATION	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Safety Injection PP 31 Motor	M48	PAB	34'	9321-F-30853	Note 1
Safety Injection PP 32 Motor	M49				
Safety Injection PP 33 Motor	M51				
Charging Pump 33 Motor	M71		55'	9321-F-30823	
Charging Pump 32 Motor	M42				
Charging Pump 31 Motor	M41				
Oil Press Sw Chg PP 33	KY5				
Oil Press Sw.Chg PP 32	KY4				
Oil Press Sw Chg PP 31	KY3				
Charging PP Control Panel	PL6				Less than 10 rads integrated dose for 1 year. Acceptable
Charging PP Mtr, Backup GP & Letdown Flow Control Sta.	PL2				Less than 10 rads integrated dose for 1 year. Acceptable

TABLE 3 (cont'd)

REVIEW OF ELECTRICAL EQUIPMENT OUTSIDE CONTAINMENT
FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	EQUIP. DESIGNATION	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS
Heat Tracing Cab 33	PX3	PAB	55'	9321-F-30873	Less than 10 rads integrated dose for 1 year. Acceptable
MCC-36B	-				Note 1
MCC-36A	-				
Boric Acid Transfer PP 31 Motor	MA1			9321-F-30833	
Boric Acid Transfer PP 32 Motor	MA2				
Boric Acid PP Control Sta.	HD3				
Boric Acid Tk Htr 31 Ctl Sta.	HD7				
Boric Acid Tk Htr 32 Ctl Sta.	HD8				
Regen PP Control Sta.	HD4		73'	9321-F-30813	
Regen Tk Mis Starter	HG2				
Regenerant PP Motor	MI5				
Conc. Holding Tk PP 31 Motor	MA5				

REVIEW OF ELECTRICAL EQUIPMENT OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	EQUIP. DESIGNATION	LOCATION	ELEV.	UE&C DWG. NO.	REMARKS	
Conc. Holding Tk PP 32 Motor	MA6	PAB	73'	9321-F-30833	Note 1	
Conc. Holding Tk Htr	RK3					
Comp Clg Bst 31 Cont. Sta.	HC1	Fan Room	67'-6"	9321-F-31593	Note 1	
Comp Clg Bst 32 Cont. Sta.	HC2					
Comp Clg Bst 33 Cont. Sta.	HC3					
Comp Clg Bst 34 Cont. Sta.	HC4					1
Panet. Air Blwr 34	MS1					
Panet. Air Blwr 33	MS2					
Fan Room Contl Pnl	JC1		80'			
PAB Exh Fan 31 Motor	ML8		67'-6"			
PAB Exh Fan 32 Motor	ML9					
Charcl Dampr Contr Pnl	JM3		72'			
Comp Clg Booster PP33 Motor	MX1		67'-6"	9321-F-31593		

TABLE 3 (continued)

REVIEW OF ELECTRICAL EQUIPMENT OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

Comp Clg Booster PP 34 Motor	MX2	Fan Room	67'-6"	9321-F-31593	Note 1
Comp Clg Booster PP 31 Motor	MU7				
Comp Clg Booster PP 32 Motor	MU8				
Motor Oper Louver 311	-		80'		
Motor Oper Louver 312	-				
Fir Prot. Cont Pnl.	-		92'		

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TABLE 4

REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
Spent Resin Stor. Tank. #31	PAB	9321-05-2258	15'-0"	9321-F-25103	9321-F-27193	Radiation environment from a cident not significant com- pared to normal field. Major accident source, RHR system, is contained in RHR cubicles
Large Gas Decay Tanks #31-34		9321-05-2199			9321-F-27303	Same as above
Small Gas Decay Tanks #31-34						
Seal Inj. Filter #31, 32		9321-05-2364				
Ion Exchanger #31, 32		9321-05-20208			9321-F-27373	
Seal Water Filter #31		9321-05-2009			9321-F-27363	
RHR Pump #31, 32		9321-05-2236			9321-F-27513	Integrated dose approx. 10^5 10^6 rads for one year perioc Further investigation is required.
Mix Bed Demineralizer #31, 32		9321-05-2372	34'-0"		9321-F-27363	Radiation environment from a cident not significant compa ed to normal field. Major accident source, S.I. system is contained in individual S.I. cubicles.

TABLE 4 (continued)
REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT
FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
Deborating Demin. #31,32	PAB	9321-05-2372	34'-0"	9321-F-25103	9321-F-27363	Same as above
Cat. Ion Bed Demin. #31		9321-05-2371				
Evap. Feed Ion Exchange #31-34					9321-F-27373	
Safety Inj. Pumps #31,32,33.		9321-05-2228			9321-F-27503	Integrated dose approx. 10^5 t 10^6 rads for one year period. Further investigation is required.
Spray Additive Tank #31		9321-05-2070	41'-0"			Integrated dose less than 100 rads for one year time period. Acceptable.
Primary Make Up Water Pumps #31,32		9321-05-20323			9321-F-27243	
Cont. Spray Pumps #31,32		9321-05-2237				
Component Cool Pumps #31, 32, 33		9321-05-2247			9321-F-27513	

TABLE 4 (continued)

REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION EVVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
Collecting Tank Transfer Pumps	PAB	-	41'-0"	9321-F-25103	9321-F-27283	Same as above
Charging Pumps Leak Collect. Tank						
Flash Evap. Product Cooler		9321-05-20074			9321-F-25103	
Air Intake & Filter		-		9321-F-40363, 25103		
Waste Evap. Pkg. #31		9321-05 2793	55'-0"	9321-F-25153	9321-F-27193	Integrated dose less than 10 rads for one year time period. Accceptabl
Waste Gas Compressor #31,32		9321-05-2935			9321-F-27303	
Press. Air Receiver #31-34					9321-F-27263	
Nitr. (N_2) Stor. Assembly		9321-05-7066			9321-F-27233	

TABLE 4 (continued)

REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
Boric Acid Transf. Pump #31, 32	PAB	9321-05-20104	55'-0"	9321-F-25153	9321-F-27363	Same as above
Comp. Cool. Heat Exch. #31, 32		9321-05-2224			9321-F-27223	
Charging Pumps #31,32, 33		9321-05-2229			9321-F-27363	
N_2 Storage Assembly		9321-05-7066			9321-F-27233	
Seal Water Ht. Exch. #31		9321-05-2218	73'-0"		9321-F-27363	
Boric Acid Evap. Pack. #31, 32		9321-05-2293			9321-F-27513	
Volume Control Tk. #31		9321-05-2205			9321-F-27363	
Nom. Regen. Ht. Exch. #31		9321-05-2217				
Isol. Valve Seal Water Stor. Tk.		9321-05-2203				

TABLE 4 (cont'd)

REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT
FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
Comp. Cool. Surge Tks. #31, 32	PAB	9321-05-2196	73'-0"	9321-F-25163	9321-F-27513	Same as above.
Boric Acid Tank #31, 32		9321-05-2204			9321-F-27363	
Boric Acid Blender		9321-05 2358				
Boric Acid Filter		9321-05-2008				1
O_2 Gas Stand #31	FAN HOUSE	9321-05-2361	67'-6"	9321-F-25183	9321-F-27533	Integrated dose approx. 10^4 rads for one year time period. Further investigation is required.
Roughing & HEPA Filters		Spec. 9321-05-45-12	72'-0"	9321-F-40823, 25183	9321-F-40853	Integrated dose apprx. 10^3 rads for one year period. Further investigation is required.
HECA Filters		9321-05-40823				
CB Purge & PAB Exh. Fans #31, 32						
Filter Unit			100'-0"			Integrated dose less than 100 rads for one year period. Acceptable

TABLE 4 (continued)

REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
Fuel Storage Bldg. Exh. Fan	Fan House		92'-0"	9321-F-25183, 40333	9321-F-40223, 40803	Same as above
Fuel Stor. Emerg. Exh. Carbon Filters		Spec. 9321-05-45,-28 9321-05-40333		9321-F-25183		
Roughing & Abs. Filters		9321-05-40333 Spec.-45, -12		9321-F-25183 & 40333	9321-F-40223, 40333	
Valve Enclosure	Pipe Trench Area	9321-05-20216	35'-0"	9321-F-25123		Integrated dose approx. 10 rads for one year. Further investigation is required.
R.C. Sample Heat Exch.		9321-05-2222, 2476	36'-6"		9321-F-27453	Same as above, except integrated dose is approx. 10^4 rads for one year period.
Boron Inj. Tk.		9321-05-2891	34'-0"		9321-F-27503	Same as above

TABLE 4 (cont'd)

REVIEW OF MECHANICAL EQUIPMENT (EXCEPT VALVES) OUTSIDE CONTAINMENT
FOR POST-ACCDIENT RADIATION ENVIRONMENT

DESCRIPTION	LOCATION	F.P. NO.	ELEV.	LOCATION DWG. REF.	SAFETY CLASS DWG. REF.	REMARKS
H_2 Gas Stand #31, 32	Fan Bldg. Room	9321-05-2361	67'-6"	9321-F-25183	9321-F-27533	Integrated dose approx. 10^4 rads for one year period. Further investigation is required.

TABLE 5

REVIEW OF SAFETY RELATED VALVES OUTSIDE CONTAINMENT

FOR POST-ACCIDENT RADIATION ENVIRONMENT

POTENTIAL RADIATION AREA	NOMINAL POST-ACCIDENT RADIATION ENVIRONMENT	SYSTEMS IN AREA WITH SAFETY RELATED VALVES	REMARKS
RHR Pump Room (15' PAB)	10^5 - 10^6 rads integrated dose for 1 year period	Aux. Coolant	There are no automatic valves located in this area. Manual valves are acceptable.
S.I. Pump Room (34' PAB)	Same as Above	Safety Injection	Manual valves in this area are acceptable. Automatic valves in this area utilize limitorqu Class B operators which have been qualified to 2×10^8 rads.
Pipe Trench, Penetration Area (35' & 54' Fan House)	10^5 rads approx. integrated dose for 1 year period	Station Air Aux. Coolant Chemical Volume Safety Injection Reactor Coolant Nitrogen To Nuclear Equip. Waste Disposal Sampling System Steam Gen. Blowdown	With respect to containment isolation valves located in this area, ASCO solenoids are used and evaluated in Westinghouse letter to the NRC NS-CE-755. Manual and motor operated valves are qualified as stated above. Further detailed review is still required.
Pipe Trench Penetration Area (34' & 32'6")	10^4 rads approx. integrated dose for 1 year period	Safety Injection Aux. Coolant Chemical Volume Waste Disposal Steam Gen. Blowdown	Same as Above
H^2 Recombiner Area (67'-6" Fan House)	Same as Above	Aux. Coolant	Further detailed review is still required.
Sampling Area (55' PAB)	Same as Above	Sampling System Aux. Coolant Chemical Volume	There are no automatic valves located in this area. Manual valves are acceptable.

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 (RCS) and have found that the individual doses received using the existing systems may exceed established radiation exposure limits. In order to comply with NUREG 0578 requirements for 1-1-81 in section 2.1.8.a on sampling of the primary cooling system and containment we have modified the existing sampling systems so that samples can be obtained to perform all chemical analyses specified without any significant radiation exposure to personnel. These modifications are complete.

The system from which a Reactor Coolant sample is being drawn is described on the flow diagram of the sampling system, in UE & C drawing number 9321-F-27453. The sample is being drawn at a connection between valves 961C and 964C as described on this drawing. The modification is described in the Indian Point 3 Modification Number 79-3-129-RCS and is summarized as follows.

A determination of oxygen content will be performed using a Rexnord Oxygen Analyzer within the primary sample line within the sampling room. To assure that a representative sample has been taken, the sample will be purged to the volume control tank with 1
a side stream taken for chemical analysis. A sample of approximately forty milliliters in volume is being taken into a shielded cask from a location outside of the sampling room within a lead shield. Under these conditions, there will be no exposure to the individual collecting a sample above the general design criteria addressed in the NUREG 0578 document. The sample will then be transferred inside of a shielded cask to the analysis location on a separate elevation of the primary auxiliary building and again transferred automatically to another shielded location to perform sample dilutions and chemical analysis.

After the sample has been transferred to the shielded analysis area, a determination of chloride content will be performed on an approximately 1 to 1 dilution of sample using a chloride sensitive electrode for this determination with a sensitivity of 20 parts per billion. The hydrogen analysis will be performed by extracting the gas from an undiluted sample in a modified "Shirley" rig and subsequent gas chromatographic determination of the hydrogen content. A portion of the sample will then be taken with a dilution of approximately one thousand to one in order to do an isotopic determination. This diluted sample will be withdrawn from the shielded area remotely so that analysis can be performed on existing radiochemistry and health physics counting room equipment. The equipment that is used for this analysis is enclosed in an approximately one meter inside dimension cave with four inches of lead shielding including cadmium and copper x-ray attenuation shielding. The counting caves will be continuously purged with bottled air under post accident conditions. A diluted sample will also be analyzed for boron content using a plasma emisssion spectrometer which has a sensitivity for boron analysis better than fifty parts per billion.

Section 2.1.8.b Cont'd

All procedures to perform these analyses have been completed. All instrumental and laboratory analytical techniques have been tested and proven reliable. Installation of the RCS sampling is complete.

In order to allow compliance with requirements of section 2.1.8.a for containment sampling, modifications to existing plant systems had to be performed because of either background interference problems or inaccessability because of high potential radiation field in the situation described in NUREG 0578. To obtain a sample of containment atmosphere the supply line to the containment particulate and gas monitor (R-11,12) is used. A sample is being taken through tubing from this supply line to a remote low background sampling location of the primary auxiliary building, and returned to the containment via R-11,12 return line through tubing. This sample will be analyzed for radioactivity and hydrogen content. 1

Section 2.1.8.b

Increases 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 has been 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 hour reading of this instrument to a microcuries per cc concentration range in the plant vent stack. A second ion chamber has been mounted outside of the shielding to remotely provide information on radiation fields in the area of the monitor.

To obtain a sample from the plant vent, a run of tubing is being performed from the supply line to the stack particulate monitor(R-13) to the same sampling location for the containment sample. These samples will be collected using pumps as the motive force for sample flow within an enclosure in this remote location. This will protect personnel collecting the samples from being exposed to any airborne activity caused in the sampling process and will allow personnel to collect samples keeping exposure within the general 1
design criteria. After small volumes sampled have been obtained for radiogas, particulate, and radioiodine analyses, they will be analyzed using existing analytical methods at the facility.

All releases from the Vapor Containment Building , the Fuel Storage Building, Waste Holdup Facilities, and Auxiliary Buildings are combined into a single discharge point at the main plant vent. The system described here provides sampling for all these potential patways. Upon indication of high radiation in the condenser air ejector, the plant systems are designed such that the condenser air ejector discharge is transferred to be released to the Vapor Containment Building with minimal direct release to the environment, therefore this pathway to the environment is protected against any high level radiation release. Even if release continues samples can be obtained using existing systems. The blowdown flash tank vent on the Indian Point 3 Facility can be continuously sampled for radioactive iodine releases and as such existing equipment will be used to monitor this potential pathway for release to the environment during an accident condition. The steam generator blowdowns are continuously monitored with an alarm circuit isolating the blowdowns upon indication of high radioactivity.

As originally installed the main steam sampling system at Indian Point 3 draws a sample downstream of the main steam isolation valves at a location shortly before the steam enters the high pressure turbine. This system has been modified, to provide for a main steam sample to be taken upstream of the main safety valves on the main steam line and this sample has been connected to the existing sampling system so that a sample may be collected through existing main steam sampling equipment and coolers from a location upstream of the MSIV's as well as the location downstream of the MSIV's.

A procedure has been prepared to establish a method to determine the amount of steam released from the main steam safety and atmospheric valves. This will be used in conjunction with the main steam sample to determine radioactivity release rates during an accident.

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 Radeco air sampling equipment presently in use at the site and Eberline single channel analyzers. 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. 1

In plant area where immediate assessment of radioiodine level is required such as the control room and the technical support center and the emergency control center, air samples will be collected using silver zeolite so that there will be no or minimal noble gas interference in determination of radioiodine concentrations.

Electrical Power to the sample and analyzing stations will be provided from a plant emergency 480 volt bus which can be powered from either the normal AC system or from the emergency diesel generators. This system provides a reliable power source with an alternate back-up power supply required by NUREG-0578.

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 following additional information is provided in response to individual subsections of item 2.1.9.c.

C.1 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 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.

C.2 The Reactor Vessel Head Vent system connects to the reactor vessel head at the existing vent pipe with flow path through a flow reducing orifice. The orifice will be capable of venting up to 17,000 ft./hr. which satisfies the NRC requirement of venting 1/2 of the volume of noncondensibles in the RCS in one hour at expected transient and accident conditions.

C.3 Downstream of the existing 3/4" vent pipe on the reactor head there will be a 3/8" restricting orifice. The 3/8" orifice restricts the flow rate from a pipe break downstream of the orifice to within the makeup capacity of one charging pump. In this situation an actuation of the ECCS does not occur and therefore a block valve which can be closed remotely, is not required. A break in the existing pipe upstream of the 3/8" orifices would require makeup from 2 charging pumps. 1

C.4 & C.5 An indication of the position of the remotely operated isolation valves will be provided in the control room.

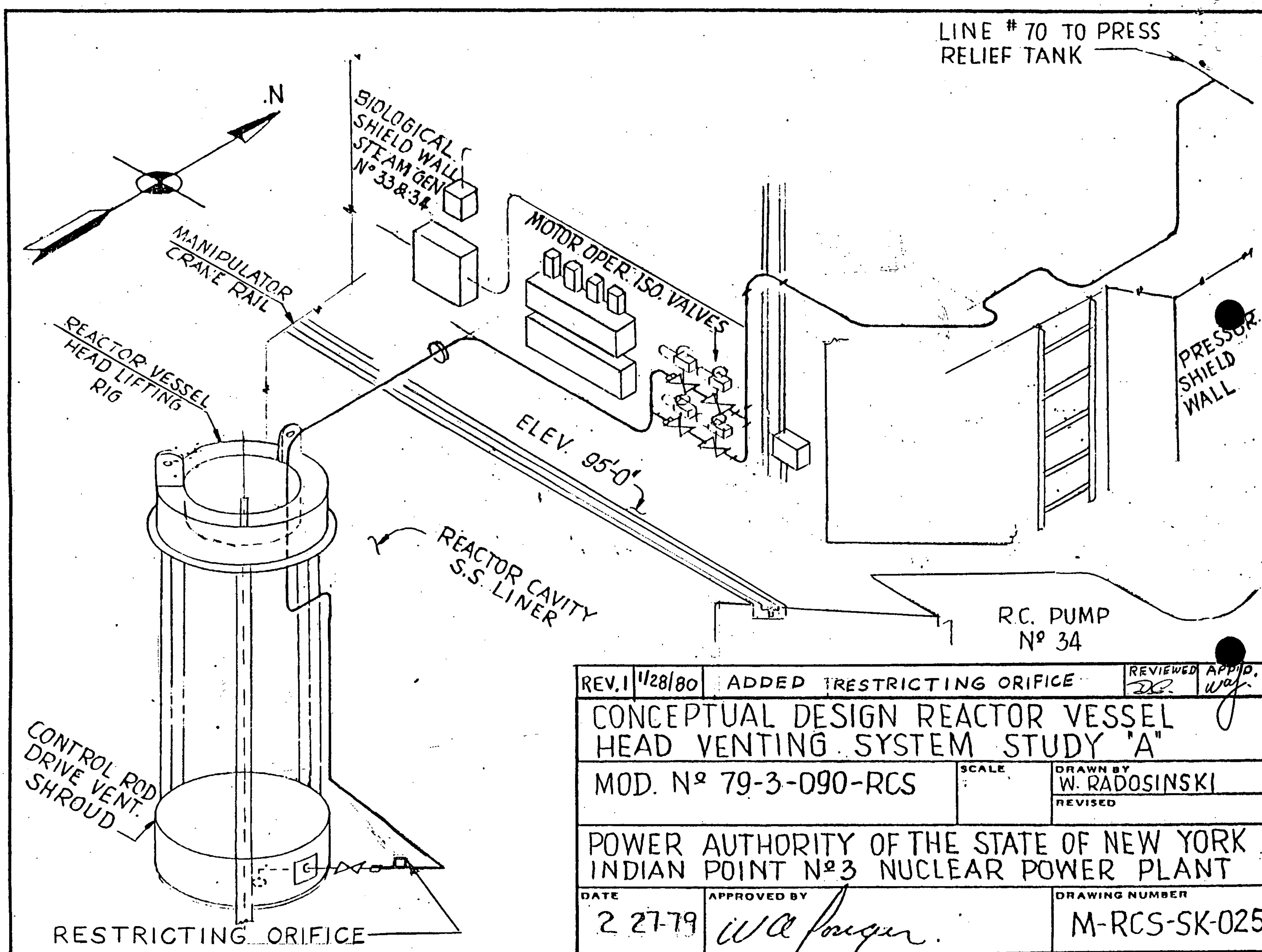
C.6 & C.7 The reactor vessel head vent system is designed to the same criteria, where applicable, that have constituted an acceptable design and licensing basis for safety systems. The design satisfies applicable USNRC and industry nuclear safety criteria; including safety class/seismic design criteria, single failure criteria and post-accident operability criteria. Each vent will be powered from a different emergency bus.

C.8 The present system design does not require the use of block valves.

C.9 The system utilizes four Safety Class 1 "fail as is" isolation valves. To eliminate potential downtime due to isolation valve seat leakage, the system will be normally operated with all valves closed. The series of two normally closed safety class valves in each flow path minimizes the chances of an inadvertent actuation and consequent opening of a flow path. 1

C.10 The reactor vessel head vent system presently discharges into the pressurizer relief tank to assure maximum cooling of the vented gas. By discharging into the pressurizer relief tank, system testing and potential inadvertent releases of water and steam can be accommodated. The pressurizer relief tank is located below #35 containment fan cooler vent thus providing a means of mixing the vented gasses with the containment atmosphere.

C.11 In addition to positive valve position indication, an acoustic flow or down stream temperature leakage monitoring system will be installed.



REV. 1	1/28/80	ADDED RESTRICTING ORIFICE	REVIEWED	APP'D.
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 2 27 79	APPROVED BY	DRAWING NUMBER M-RCS-SK-025		

Section 2.2.1.b

Shift Technical Advisor

The Authority has provided an on shift technical advisor, as required, as of January 1, 1980, whenever the unit is above cold shutdown.

In the interim period, while personnel are being recruited specifically to fill the STA position, the STA will be manned by existing plant personnel on a rotation basis. The personnel selected to fill the STA position during this interim period all have bachelors degrees in engineering or scientific disciplines. They have received additional training in reactor theory, thermal dynamics, primary and secondary systems. Unitl sufficient personnel can be added to the staff and trained specifically for the STA position, suitable quarters have been provided to accomodate the on site STA on a twenty four hour per day basis such that he can be available in the control room within 10 minutes.

To insure a higher level of involvement in plant operations, the STA will be required to review all Plant Operating Review Committee (PORC) minutes, Licensee Event Reports (LER's) and Significant Occurrence Reports (SOR's), as well as other material as appropriate. To further enhance his operational expertise, the STA will be involved in the assesment of plant operating problems and will make recommendations to appropriate management personnel.

One of the responsibilities of the STA is to initiate action to man the Technical Support Center during emergency conditions. To accomplish this function in the most expeditious manner, he will place a single call to the site security force which in turn will contact the appropriate personnel. Procedure revisions to incorporate these improvements in the STA function have been completed.

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. The required communication link between the
operational support center and the control room has 1
been installed.

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%.

In order to independently verify the above, refer to drawings detailed in response to section 2.1.1. 1