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EISENHUT,D.G. Division of Operating Reactors

SUBJECT: Forwards response to NUREG-0578,Lessons Learned Task Force short-term requirements,per NRC 791030 ltr.Designs subj to change in response to NRC clarifications.

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	3 LPDR	1	1	4 NSIC	1	1
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FLORIDA POWER & LIGHT COMPANY

January 11, 1980
L-80-17

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: St. Lucie Unit 1
Docket No. 50-335
NUREG-0578 Short Term Requirements

Pursuant to Mr. Denton's letter of October 30, 1979, as verbally amplified by the NRC Staff on December 3, 1979, Florida Power & Light Company is submitting herein additional details concerning the actions being taken to meet our commitments to implement the Category A items contained in NUREG-0578. The drawings associated with this submittal are being copied and will be forwarded to you within a few days.

FPL is committed to meeting the intent of NUREG-0578. However, the actions and engineering designs described herein represent our best current thinking about how to implement the NUREG recommendations. The designs are subject to change in response to additional clarifications from the NRC Staff, changes in the state of the art, and changes in commercially available hardware.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/MAS/ah

Attachment

cc: J. P. O'Reilly, Region II
Harold F. Reis, Esquire

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ST. LUCIE UNIT 1

IMPLEMENTATION ACTIONS TO MEET NUREG-0578

CATEGORY A ITEMS

2.1.1 EMERGENCY POWER SOURCES

Our reviews have shown that St. Lucie Plant as originally designed meets the Category A requirements of NUREG-0578 for this item.

2.1.2 RELIEF AND SAFETY VALVE TESTING

FPL will be participating in the EPRI program submitted to you by letter dated December 17, 1979, from Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force, under the title "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems", December 13, 1979. Reference FPL's letter L-79-362 from R. E. Uhrig to H. Denton dated 12-21-79.

2.1.3.a DIRECT INDICATION OF VALVE POSITION

The implementation schedule for this modification is being held in abeyance pending our review of the NRC's Show Cause Order for St. Lucie Unit 1 dated January 2, 1980. However, equipment required for installation is on-site and installation of cable and components outside containment is in progress. The modification itself is described below:

Design Bases

Valve positions will be monitored acoustically and indicated in the Control Room.

Valve position monitors are powered from a vital instrument bus and are seismically qualified.

The sensors are designed to be qualified for the postulated post accident environment.

Valve position indication is safety grade.

As backup to this single channel environmentally qualified system, limit switch position indication for the PORV pilot valves and discharge pipe temperature indication for the pressurizer safety relief valves are used. The PORV's also are provided with an RTD in the piping downstream of the valves.

Function

The acoustical valve flow monitoring system provides the operator with positive

flow indication for each pressurizer safety valve and power operated relief valve (PORV).

System Description

The means of detecting pressurizer safety relief and power operated relief valve position is by continuously and automatically detecting acoustical signals (noise levels) generated by flow through the valve.

This is accomplished by utilizing accelerometers mounted on each valve's discharge piping. The accelerometer converts flow-generated acoustical noise into an electrical charge which is then converted to a voltage by the charge converter. This proportional voltage is then processed and a relative flow indication is obtained.

Five valve position monitors are provided, three for the pressurizer safety relief valves and two for the PORV. A common audio-visual alarm alerts the operator when flow through any of the five valves exceeds a pre-established set point. These set points are front panel adjustable.

The system is powered from a 120V AC 60 Hz Class 1E power supply. An alarm is initiated upon loss of instrument power. The indicator modules are located on the Post Accident Panel in the Control Room. The system is qualified in accordance with applicable IEEE codes.

The accelerometers and charge converters are located inside the containment and are subjected to the containment environment during and following a LOCA. These components are designed and tested to withstand and remain operable following the postulated accident.

The indicator modules are located in the Control Room where the environmental conditions during and following an accident are the same as that prior to the postulated accident.

The components of the Accoustical Valve Flow Monitoring System have been designed as Seismic Category I. The system components remain operable following seismic loads.

System Components

The following equipment is furnished as part of the Accoustical Valve Flow Monitoring System:

5 Accoustical Sensors	-	tested and qualified to IEEE* for the containment environment
5 Charge Converters	-	tested and qualified to IEEE* for the containment environment

* The vendor is currently performing qualification test.

Cable	-	furnished as Class 1E. 50 feet of low noise, high temperature cable connects each valve sensor to its charge converter
5 Indicator Modules	-	tested and qualified to IEEE* for the control room environment
1 Alarm Module	-	tested and qualified to IEEE* for the control room environment

Testing and Inspection

Vendors substantiate through test, calculations and/or operational data that the system components remain operable following seismic loads. Vendors also qualify the equipment to the service environment.

* The vendor is currently performing qualification test.

2.1.3.b INSTRUMENTATION FOR INADEQUATE CORE COOLING

Installation of a Subcooled Margin Monitor is currently in progress. Recent problems regarding the qualification of Class 1E isolators has caused a delay in the shipment of the isolators, but all other material is on-site. We currently schedule complete installation of the monitor on or before February 15, 1980.

Both the need for and the functional requirements of a reactor vessel level instrument are under development by CE and the CE Owners Group. When finalized, the information will be provided to the NRC for review.

Subcooling Margin Monitor Design Bases

Continuous display of reactor coolant margin to saturation will be provided in the Control Room.

The inputs to the Subcooling Margin Monitor are from safety grade sensors qualified for the post accident environment.

Safety grade temperature signals from each RCS hot and cold leg will be utilized.

The monitor will be a highly reliable, non-redundant, environmentally qualified system using the steam tables as a backup.

Subcooling margin monitor will be isolated from existing reactor protection or engineered safety features.

Function

The subcooling margin monitor provides the Control Room operator with continuous indication of margin to saturated conditions in the reactor coolant system.

System Description

The Subcooled Margin Monitor is an on-line microcomputer based system which uses reactor coolant process signals to provide a continuous indication of the margin to saturation. The system consists of a calculator module, a panel display module and cabling between the modules.

The calculator module receives temperature and pressure analog signals from the following primary loop parameters:

a)		Pressurizer Pressure (2 Signals)
b)	RCS	Cold Leg Temperature
	Loop 1	Cold Leg Temperature
		Hot Leg Temperature

c) RCS Cold Leg Temperature

Loop 2 Cold Leg Temperature

Hot Leg Temperature

These signals are interfaced to a microcomputer. The microcomputer contains steam tables and interpolation routines to calculate saturation temperatures and pressures. By comparing the saturation temperature or pressure to the actual coolant temperature or pressure, a margin to saturation is calculated. Either the temperature or pressure margin can be displayed on the digital panel meter. The margin is compared to a setpoint for a low margin alarm. Input sensor ranges and setpoints are listed in Table 2.1.3b-1 attached. See Figure 2.1.3b-1 for the instrument block diagram.

Software diagnostics are periodically executed to detect observable software malfunctions. A diagnostic failure is indicated by a flashing meter display. The Subcooled Margin Monitor system is provided with a 120V AC 60 Hz safety grade power supply. An alarm actuates in case of a power failure.

The system is qualified in accordance with applicable codes and will be upgraded to meet requirements of Regulatory Guide 1.97 prior to January 1, 1981. The display module is located on RTGB 103 in the Control Room.

The only portions of the system which will be subjected to the containment environment associated with a LOCA are the input sensors. These components are part of the Reactor Protection instrumentation and are designed to withstand the service environment during and following a LOCA as specified in Section 3.11 of the FSAR. The remaining portions of the system are located in the Control Room where the environmental conditions during and following an accident are the same as that prior to the postulated accident.

System Components

The following equipment is furnished as part of the Subcooled Margin Monitor System:

1 Calculator Module	-	tested and qualified to IEEE* for the control room environment
1 Panel Display Module	-	tested and qualified to IEEE* for the control room environment
Cable	-	furnished as Class 1E, 15 feet interconnecting cable
7 E/E Signal Isolators	-	tested and qualified to IEEE* for the control room environment. These isolators are required to maintain the separation requirements between the different Reactor Protection channels and safeguard channels

* IEEE-323-1974
IEEE-344-1975

Testing and Inspection

Vendors will substantiate through test, calculations and/or operational data that the system components will remain operable under seismic loads. Vendors will also qualify the equipment to their service environment.

SUBCOOLED MARGIN MONITOR
INPUT RANGES AND ALARM SETPOINT

TABLE 2.1.3b-1

Process Input	Process Signal Range (°F, psia)	Electrical Signal Range (V, mA)		
PY-1102A-1	1500-2500 psia	1-5 Vdc		
PY-1103	0-1600 psia	1-5 Vdc		
TY-1112CA	465-615°F	1-5 Vdc		
TY-1112CB	465-615°F	1-5 Vdc		
TY-1112HA	515-665°F	1-5 Vdc		
TY-1122HA	515-665°F	1-5 Vdc		
TY-1122CB	465-615°F	1-5 Vdc		
TY-1122CA	465-615°F	1-5 Vdc	Set °F	Reset °F
		Low Margin Alarm	30° Subcooling	35° Subcooling

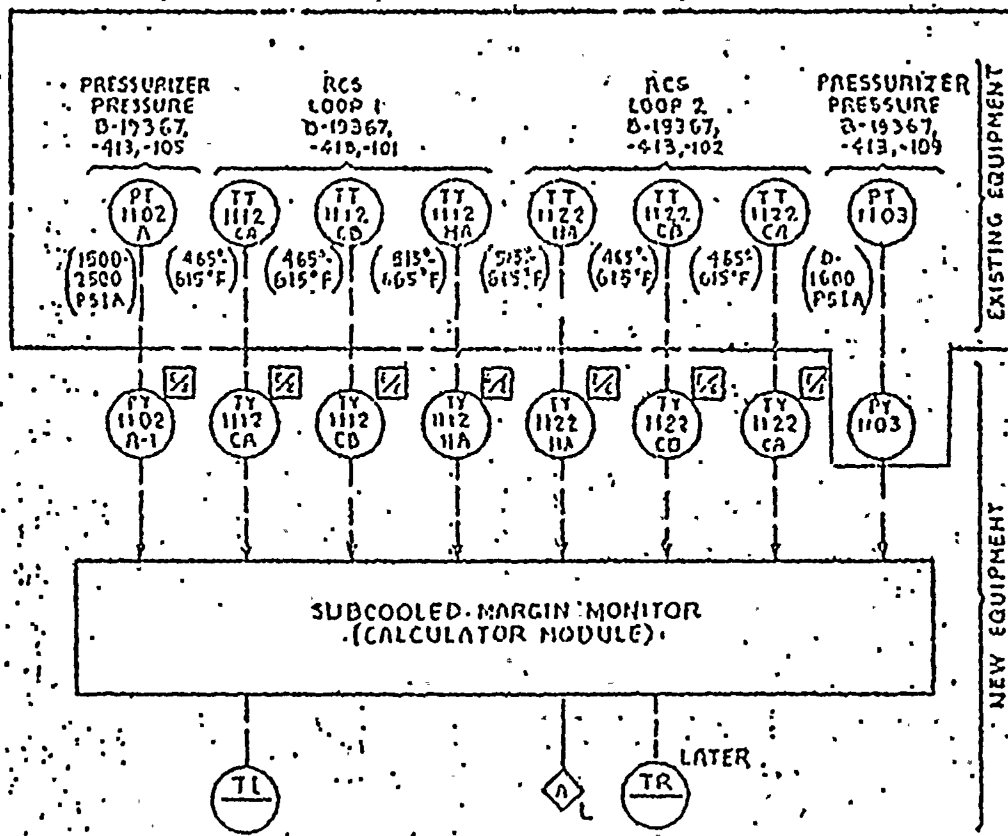

FIGURE 2.1.3.b-1

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
8				4			
7				3			
6				2			
5				1			

EBASCO SERVICES INCORPORATED
NEW YORK

DIV. ELEC DR. A.M APPROVED
SCALE ~ CH.
DATE 12-13-79

FLORIDA POWER & LIGHT CO.
ST. LUCIE PLANT #1
BLOCK DIAGRAM
SUBCOOLED MARGIN MONITOR

SK-SMM-01
SHEET 1

2.1.4 DIVERSE CONTAINMENT ISOLATION

The original plant design provides for containment isolation on the receipt of either a high containment pressure signal or a high containment radiation signal. Additionally, the design has been changed to add the Safety Injection Actuation Signal (SIAS) as an input to the containment isolation circuitry (CIAS). Receipt of SIAS will now automatically initiate containment isolation. Neither SIAS nor CIAS can be reset while an initiating signal is present, and resetting of either SIAS or CIAS will not result in the reopening of any valves.

This change resulted from a review of all lines penetrating containment, and the following lines, not previously isolated on SIAS were found to be non-essential from a safety standpoint:

1. Steam Generator Blowdown (2 lines)
2. Primary Makeup Water
3. Instrument Air Supply
4. Containment Pure Air Exhaust
5. Containment Purge Supply
6. Nitrogen Supply to Safety Injection Tanks
7. Reactor Coolant Sample
8. Pressurizer Surge Line Sample
9. Pressurizer Steam Space Sample
10. Steam Generator Blowdown Sample (2 lines)
11. Containment Vent Header
12. Reactor Drain Tank Pump Suction

13. Reactor Coolant Pump Controlled Bleedoff

14. Containment Atmosphere Radiation Monitoring System (3 lines)

All other lines penetraing containment are either essential to core cooling capability, have locked closed manual valves, or isolate already on a safety injection actuation signal.

We conclude that St. Lucie Unit 1 meets requirement 2.1.4.

2.1.5.a DEDICATED PENETRATIONS FOR EXTERNAL HYDROGEN RECOMBINERS

St. Lucie Unit 1 is equipped with 2 hydrogen recombiners located within the containment. This requirement is therefore not applicable.

2.1.5.b <u>INERTING BWR CONTAINMENTS</u>

NOT APPLICABLE TO PWR's

2.1.5.c HYDROGEN RECOMBINER OPERATING PROCEDURES

A detailed review of the existing bases and procedures for recombiner use at St. Lucie Unit 1 has been conducted in response to NUREG-0578 and other NRC documents. Our review has concluded that no changes are necessary at this time, and that St. Lucie Unit 1 meets the requirements of this item.

2.1.6.a SYSTEM INTEGRITY FOR HIGH RADIOACTIVITY

In order to implement the recommendations of NUREG-0578, St. Lucie Plant has recently developed a program to identify and minimize leakage in systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The procedure "RAB Fluid Systems Periodic Leak Test", OP-1300054, which has been written and approved, establishes and implements a leak test program to provide baseline data pertaining to operational leakage characteristics and information from which leak reduction measures can be instituted. The following systems are included within the scope of this program:

(1) ECCS (HPSI, LPSI, and Containment Spray systems)

(2) RCS Sample System

(3) Charging and Letdown System (CVCS)

(4) Waste Gas System

In order to perform the leak test, these systems are pressurized for a minimum period of one hour during which physical walk-downs and visual inspections of individual components are performed and all observed leakage is recorded. Gaseous systems are tested using Volumetric Leakrate Monitors or the equivalent, bubble testing, and airborne activity sampling in conjunction with physical walk-downs and visual inspections. This surveillance program is presently scheduled to be performed at 18 month intervals (maximum). The following data was collected during initial testing of individual components and fluid systems:

(1) ECCS Integrated Leak Rate <1.0 gpm

(2) RCS Sample Integrated Leak Rate <1.0 gpm

(3) CVCS Integrated Leak Rate <1.0 gpm

(4) Waste Gas System Integrated Leak Rate <0.5 SCFM

2.1.6.b DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS

Design Bases

Radiation fields are considered resulting from all equipment and piping outside the containment that might contain primary coolant or radioactive gases during post-accident operations.

Radiation fields are considered as resulting from gaseous nuclides released from the core into the containment atmosphere as the result of a postulated LOCA.

Source terms for liquid systems assume 100% of the core inventory of noble gases, 50% of the core inventory of halogens, and 1% of the core inventory of other nuclides are contained in the primary coolant. Approprite dilution and decay factors are applied.

Source terms for containment atmosphere follow the guidance of Regulatory Guide 1.4 and assume that 100% of the core inventory of noble gases and 25% of the core inventory of halogens are released into the containment atmosphere. Appropriate decay factors are applied.

Source terms for gaseous systems are derived from the appropriate activity in either the primary coolant or containment atmosphere. Dilution and decay factors shall be applied.

Dose rates at all vital areas are reviewed to ensure that personnel access to perform required tasks will be possible. A vital area is defined as any area which will or may require occupancy to permit an operator to aid in the mitigation of, or recovery from, an accident.

Radiation exposure to personnel is deemed unacceptable if exposures exceed the limits of 10CFR20, and GDC 19, 60 and 64 of Appendix A to 10CFR50. Radiation exposure to the control room operators is limited to 5 Rem whole body in 30 days following the accident. Exposure to other operating personnel performing necessary functions in vital areas is limited to 3 Rem whole body and 18 3/4 Rem to the extremities.

Acceptance criteria for dose rates in vital areas consider the appropriate dose limit, required occupancy time, round-trip access route, required frequency of access, and a safety margin.

Integrated doses are estimated for safety equipment required for post-accident operation to ensure that GDC 4 requirements are met. Acceptance criteria for equipment doses are determined either by equipment specification, actual equipment qualification, or generic material damage data. Doses are found by integrating dose rates over the time the equipment is required to function.

The radiation fields present do not impair the function of counting room instruments or other Health Physics equipment required for sample analyses.

Function

The design review evaluates the radiological environment of the plant following an accident in which significant core damage has occurred. The objective of this review is to:

a) Insure that required access to vital areas and equipment needed for post-accident operations will not be compromised by high radiation fields, and

b) Insure that the operation of safety equipment needed for post-accident operations will not be impaired by high radiation fields.

Shielding Description

Shielding is presently provided around the majority of radioactive and potentially radioactive equipment and piping. The original design criteria and shielding descriptions are found in Section 12.1 of the FSAR. Shielding protection is generally provided by concrete walls which are shown on Reactor Auxiliary Building General Arrangement sketches 2.1.6b-12 thru 2.1.6b-17.

Design Evaluation

The following systems or portions of systems have been identified as containing reactor coolant or radioactive gases:

a) Containment Spray System

b) Safety Injection System

- Low Pressure Safety Injection

- High Pressure Safety Injection

c) Shutdown Cooling System

d) Sampling System

- Liquid

- Containment H_2 Analyses

These systems or portions of systems that have the potential for containing highly radioactive fluids are indicated on Flow Diagram sketches 2.1.6b-1 thru 2.1.6b-11.

Source Terms

The total core inventory at the time of the accident is shown in Table 2.1.6b-1 and was taken from FSAR Table 15.4.1-1 for Noble Gas and Halogens. Core inventory of other isotopes were taken from Table 4.3-1 of Combustion Engineering document SYS80-PE-RG, Rev. 4, Radiation Design Guide, with a factor of .643 applied to account for power difference.

Multigroup source strengths were calculated at various times following the accident for Reactor Coolant (undiluted), Sump Water (diluted reactor coolant), and Containment Atmosphere. These source strengths are shown in Tables 2.1.6b-2, 2.1.6b-3, and 2.1.6b-5. Source strengths were calculated from the original isotopic inventory data by separating the specific activities of the radionuclides in the various fluids into appropriate energy groupings. The GROUP[1] code performed the grouping in some instances.

The specific activity of the reactor coolant was used as the source term for the liquid sample system at one hour. The specific activity of the sump water was used as the source term for all other liquid systems, and for the sample systems at times greater than one hour following the accident. The specific activity of the containment atmosphere was used as the source term for the Hydrogen Analyzer and gas sample.

Dose Rate Calculations

Dose rate calculations were performed in areas identified as vital areas requiring occupancy and in areas containing safety equipment. Sources were determined as stated in Section 4.2 and appropriate geometry factors were applied to the piping and equipment of the systems identified in Section 4.1 as containing radioactive sources. The shielding effect of the equipment, the fluid, and the present shield wall arrangement was considered. The effect of rebar, embedded plates, and structural steel was neglected, thereby giving high (conservative) dose rates. Dose rates were calculated primarily by the ISOSHLD[2] point-kernel integration code for "simple" geometry cases, such as single pipes and tanks. The SPAN-IV[3] point-kernel integration code was used to find dose rates for selected "complex" geometry cases involving a number of pipes, tanks and shields.

Vital Areas Requiring Occupancy

Vital areas requiring occupancy include:

-Control Room	-continuous occupancy
-Sample Room	-1 minute, 1 time occupancy starting at T=1 hr.
-Counting Room	-Single 1 time occupancy to move equipment
-Hydrogen Analyzer	-2-3 minutes occasional occupancy starting at T=1 hr.
-Charging Pump 1A cubicle	-requires 2 men, each for 5 min.
-Charging Pump 1B cubicle	-requires 1 man for 5 min.
-Charging Pump 1C cubicle	-requires 1 man for 5 min.
-Letdown Heat Exchanger Area (Hot Leg Injection)	-requires 4 men, each for 5 min.

-LPSI Pump 1A cubicle	-requires 3 men, each for 5-10 min.
-LPSI Pump 1B cubicle	-requires 3 men, each for 5-10 min.
-LPSI Pump 1B cubicle (Shut Down mode)	-requires 1 man for 10 min.
-Electric Penetration Room (El.19'6)	-requires 2 men, each for 5 min.
-Shutdown Heat Exchangers 1A, 1B (RAB, EL-5, Corridor)	-requires 4 men, each for 5 min.

<u>Potential Vital Areas</u>

-Power to IC-HPSI pump and IC - charging pump - from switchgear "C" located on RAB, elev. 19.5', near the CEDM MG sets (adjacent to the Sample Room)

-Valve station outside Equipment Drain Tank

-Valve station outside Chemical Drain Tank

-Valve station - Waste gas decay tank area

Equipment Requiring Protection

Radiation damage was addressed on a generic basis. The majority of the dose following an accident is experienced in the first few days, and 90% of the dose occurs in the first 30 days. Thirty day integrated doses were determined for selected areas, and one-year post accident doses estimated from these. Doses from direct shine only are considered; airborne activity could increase the levels.

Integrated doses to safety equipment located outside the containment were estimated. One-year post accident doses to equipment located outside the highly radioactive cubicles on elevations - 0.5' and 19.5' are estimated as less than 10^5 rads in all cases, and doses of 10^4 rads or less are typical. For elevation 43', the maximum dose in most areas is estimated as 10^3 rads. However, doses of 10^4 rads may be experienced near the containment structure. Doses on elevation 62' are limited to a few rads.

Safety equipment housed inside the highly radioactive cubicles in the Reactor Auxiliary Building or inside the containment could experience a post-accident dose higher than 10^5 rads. Outside the containment, this would mainly apply to instruments, valves, pumps, and motors located directly on the lines handling the primary coolant fluids. These systems are identified in section 3.1.

Sampling System

A redesign of the sample system is under evaluation and is discussed in the response to requirement 2.1.8.a.

Miscellaneous

Access may be required, for flexibility of operations, to "C" switchgear located on RAB Elevation 19.5' near the CEDM motor generator sets. The primary source of radiation is equipment and tubing in the nearby sample room, which could contribute a dose rate of approximately 290 rem/hr 1 hour after the accident, 130 rem/hr 10 hours later, and 40 rem/hr 100 hours later. This would normally preclude access to the switchgear; However, the existing sample room will be used only once following an accident. An initial, small (1 ml) sample will be drawn and the sample line then flushed with demineralized water. The flush will eliminate the sample lines as the primary source of radiation, and the primary source of radiation then becomes the ECCS piping and equipment on the floor below, elevation - 0.5'. This source far exceeds the remaining sources (excluding the sample room) on the 19.50' level; i.e., the hydrogen analyzer and the letdown heat exchanger. The dose rates and doses (assuming five minute exposure) in the switchgear area as a function of time are:

Time, hr.	Dose Rate, rem/hr.	Dose, (mr)
1	11	920
10	4	330

100	1	80

These doses are sufficiently low to permit brief access to the "C" switchgear.

Estimates of Exposure

Dose rates were calculated for numerous areas in the RAB and are shown in Table 2.1.6b-6. Several time periods from 1 hour to 1000 hours following the accident were considered to allow for decay until access is required. Personnel access and exposure are summarized below.

Primary sources of radiation on elevation -0.5' of the Reactor Auxiliary Building are the Shutdown Cooling Heat Exchangers, and the ECCS pumps. Dose rates inside cubicles with equipment containing primary coolant range from several hundred to several thousand rem per hour, and these cubicles could be expected to be inaccessible for a period of time following an accident. Dose rates decline to the neighborhood of 1 to 10 rem/hr after 30 days, and access to these cubicles would be limited to a few minutes. Dose rates in the main access corridor at elevation -0.5' range from 1 to 10 rem/hr after one hour, decaying to less than 1 rem/hr after three days. Access to equipment in this corridor is possible, but also must be limited to short periods of time. Equipment requiring frequent access will be relocated, redesigned or provided with additional shielding.

Primary radiation sources on the 19.5' elevation of the RAB include the sample station, letdown heat exchanger, and the Volume Control Tank and associated piping. In addition, a significant contribution to the radiation level originates from equipment and piping on the floor below.

Cubicles housing equipment containing primary coolant have dose rates similar to the cubicles on elevation -0.5' and are considered inaccessible. As discussed earlier, the dose rate in the sample room could preclude the use of the sample room with the presently designed sample panel. Redesign of the sample panel is discussed in section 2.1.8a.

Dose rates in the main access corridor of elevation 19.5' is calculated at a maximum of 14 rem/hr at one hour near Staircase RA3 decaying to less than 1 rem/hr after 30 hours. Dose rates in the corridor are generally less than 1 rem/hr after 10 hours. Access to the corridor is possible although limited in certain areas.

Dose rates in the vicinity of the containment hydrogen analyzer exceed 10 rem/hr and access would be limited. Use of this analyzer for grab samples will require modifications and this is discussed in section 2.1.8a. Although readout is in the Control Room, access may be required to this panel for calibration purposes. Dose rates in the radiochemistry lab and the counting room allow access, but may negate the room's main function. Therefore, a procedure has been written and approved which provides, in case of accident, for the relocation of needed equipment from the counting room to one of four specified locations on-site.

Dose rates on elevation 43' of the RAB are generally less than 100 mrem/hr and access is generally possible except in the vicinity of the containment ventilation filters and containment building wall. A dose rate of approximately 10 rem/hr would be experienced within 10 feet of the containment wall.

Exposure in the Control Room at elevation 62' of the RAB will not exceed a total integrated dose of 5 rem for the duration of the accident in accordance with 10CFR50 Appendix A GDC 19.

The above dose rates for all elevations assume the Holdup Tanks do not contain an appreciable volume of primary coolant. The use of these tanks to store liquid wastes with specific activity of the sump water will significantly decrease the accessibility of elevation 19.5' and 43'.

Many locations requiring access to operate manual valves are located in areas expected to be in extremely high radiation fields following an accident, such as the ECCS room and the pipe tunnel. Since these valves are surrounded by radioactive equipment and piping, and are themselves located in radioactive pipe, shielding does not appear to be feasible. Therefore, if our ongoing evaluation determines that it is necessary to operate these valves, then remote actuation will be required. Other valves and locations, such as in the waste management system, may be accessible following an accident with the possible addition of local shielding providing sufficient time has elapsed to allow for radioactive decay. Further classification will be required to establish exactly when a particular valve or area must be accessed.

The accessibility of specific valves and locations are discussed below. Access routes and vital areas are indicated on figures 2.1.6b-18 and 2.1.6b-19.

SDCS Valve Lineup:

Prior to operation of the shutdown cooling system, several 10" valves must be manually operated: I-V-07-008 (bypass); V3452 and V3453 (Shutdown Cooling Heat Exchanger inlet isolation valves); and V3456 and V3457 (heat exchanger outlet isolation valves). The bypass valve is located in the LPSI cubicle, and will, therefore, be inaccessible.

The four shutdown cooling heat exchanger valves are located in the heat exchanger cubicles, but are manually operated with reach rods from outside the South wall of the cubicles. Thus, the operator would be shielded from both the shutdown cooling and ECCS sources. It has been estimated that one person can operate each valve in from 5-10 minutes. The dose rates (from all sources) and exposures for all valve operations are as follows:

Time, hr.	Dose Rate, rem/hr.	Exposure, rem
1	17	5.7 - 11
10	6.3	2.1 - 4.2
100	1.7	.57- 1.1

In addition, contributions to the total dose while travelling to and from the valves must be considered. Table 2.1.6b-6 lists dose rates at various locations in the Reactor Auxiliary Building (RAB). For example, the dose rate in the vicnity of staircase RA-7 (near these valves) is 7.8 rem/hr at 1 hour, 2.3 rem/hr at 10 hours, and 0.43 rem/hr at 100 hours. From the foregoing exposure values, it appears possible for two people to operate these valves 10 hours after an accident. Local shielding near the valve operators is being evaluated.

<u>References</u> - Section 2.1.6b

1. E. Ochoa, D. Vories, "GROUP - An Isotopic Source Generation Program", (1977).

2. R. L. Engel, J. Greenberg, M. M. Hendrickson, "ISOSHLD - A Computer Code for General Purpose Isotope Shielding Analysis", BNWL-236 (1966).

3. O. J. Wallace, "SPAN-IV: A Point-Kernel Computer Program for Shielding", WAPD-TM-809(L) (1972).

TABLE 2.1.6b-1

TOTAL CORE INVENTORY

NOBLE GASES

Nuclide	Activity (Curies)
Kr-85	7.45 E5
Kr-85m	2.95 E7
Kr-87	5.75 E7
Kr-88	8.10 E7
Xe-131m	3.86 E5
Xe-133	1.52 E8
Xe-133m	3.64 E6
Xe-135	1.43 E8
Xe-135m	4.10 E7
Xe-138	1.34 E8

HALOGENS

Nuclide	Activity (Curies)
I-131	6.62 E7
I-132	9.69 E7
I-133	1.53 E8
I-134	1.75 E8
I-135	1.41 E8

TABLE 2.1.6.b-1 (Contd)

TOTAL CORE INVENTORY

OTHER NUCLIDES

Nuclide	Activity (curies)	Nuclide	Activity (curies)
Se-84	1.40 E7	Mo-99	1.38 E8
As-85	~~[illegible]~~ 2.42 E6	Tc-99m	1.19 E8
Se-85	~~[illegible]~~ 8.62 E6	Mo-103	1.21 E8
Se-87	1.38 E7	Tc-103	1.23 E8
Rb-88	5.05 E7	Ru-103	1.24 E8
Rb-89	6.56 E7	Tc-106	5.11 E7
Sr-89	7.01 E7	Ru-106	3.50 E7
Rb-90	~~[illegible]~~ 6.19 E7	Sn-129	8.04 E7
Sr-90	4.89 E6	Sb-129	2.50 E7
Y-90	5.13 E6	Te-129m	6.49 E6
Rb-91	7.97 E7	Te-129	2.37 E7
Sr-91	8.62 E7	Sn-131	2.22 E7
Y-91m	4.96 E7	Sb-131	6.11 E6
Y-91	9.13 E7	Te-131m	1.14 E7
Sr-95	9.19 E7	Te-131	6.56 E7
Y-95	1.21 E8	Sn-132	~~[illegible]~~ 1.29 E7
Zr-95	1.27 E8	Sb-132	3.62 E7
Nb-95	1.28 E8	Te-132	1.08 E8
Zr-99	1.25 E8	Sn-133	4.49 E6
[illegible]	1.31 E8	Sb-133	4.10 E7

TABLE 2.1.6b-1 (Contd)

TOTAL CORE INVENTORY

OTHER NUCLIDES

Nuclide	Activity (Curies)	Nuclide	Activity (Curies)
Te-133m	5.45 E 7	Ba-140	1.32 E 8
Te-133	8.68 E 7	La-140	1.36 E 8
Cs-134	1.43 E 7	Cs-143	2.52 E 9
Sb-134	7.26 E6	Ba-143	1.00 E 8
Te-134	1.15 E8	La-143	1.13 E8
Sb-135	4.55 E6	Ce-143	1.14 E8
Te-135	5.99 E7	Pr-143	1.12 E8
Cs-136	4.00 E6	Cs-144	7.72 E 6
Cs-137	6.56 E6	Ba-144	7.46 E 7
Ba-137m	6.22 E6	La-144	9.84 E 7
Cs-138	1.29 E7	Ce-144	9.00 E 7
Cs-140	1.17 E8	Pr-144	9.07 E7

TABLE 2.1.6b-2

SOURCE STRENGTH

REACTOR COOLANT

Group	Energy (Mev)	Source T=1 hr* (γ/cm^3-sec)
1	.3	1.68 E10
2	.63	1.59 E10
3	1.10	1.63 E10
4	1.55	5.76 E9
5	1.99	2.89 E9
6	2.38	3.01 E9
7	2.75	5.68 E7
8	3.25	2.33 E7
9	3.70	3.93 E7
10	4.22	6.55 E1
11	4.70	2.70 E2
12	5.25	9.26 E1

* Source 1 hour after accident

TABLE 2.1.6b-3

SOURCE STRENGTH – SUMP WATER

Group	Energy (Mev)	Source Strength (γ/cm^3-sec) T = 1 hr *	T = 10 hr	T = 100 hr	T = 1000 hr
1	.3	2.49 E9	1.82 E9	9.97 E8	3.53 E7
2	.63	8.75 E9	4.30 E9	1.52 E9	7.41 E7
3	1.10	2.41 E9	1.01 E9	2.17 E8	1.41 E6
4	1.55	8.52 E8	2.89 E8	8.19 E7	3.34 E6
5	1.99	4.27 E8	1.15 E8	1.24 E7	2.39 E5
6	2.38	4.45 E8	6.85 E7	1.02 E6	1.13 E3
7	2.75	8.70 E6	2.89 E5	1.45 E3	8.76 E2
8	3.25	3.45 E6	2.16 E4	3.01 E1	2.76 E1
9	3.70	5.81 E6	3.70 E4	0	0
10	4.22	9.69 E0	0	0	0
11	4.70	1.14 E2	0	0	0
12	5.25	1.37 E1	0	0	0

TABLE 2.1.6b-4

ASSUMPTIONS AND PARAMETERS

CONTAINMENT ATMOSPHERE SOURCE

1. Core Inventory - Table 2.1.6b-1

2. Core Inventory release fraction to containment air

   Noble Gas - 100%

   Halogen - 50%

3. Halogen plate out factor = .5

4. Delay time of NaOH injection: 2 minutes

5. First order spray removal coefficients

   Elemental Iodine - 10 hr^{-1}

   Organic Iodine - 0

   Particulate Iodine - .45 hr^{-1}

6. Max allowable spray DF - 100

7. Containment free volume - 2.506×10^6 cuft

8. Fraction of free volume reach by sprays - 86%

Table 2[illegible].6b-5

Containment Activity (γ/sec)

N	E, MeV	T=0 Atmos.	T=0 Plateout	T=0 Total	T=2min Atmos.	T=2min Plateout	T=2min Total	T=46min Atmos.	T=46min Plateout	T=46min Total
1	.150	6.44+18*	1.54+16	6.46+18	6.37+18	1.54+16	6.39+18	5.64+18	1.54+16	5.66+18
2	.250	8.35+18	3.62+16	8.37+18	8.15+18	3.62+16	8.19+18	6.00+18	3.62+16	6.04+18
3	.350	1.21+18	1.05+18	2.26+18	1.19+18	1.03+18	2.22+18	2.20+17	8.00+17	1.02+18
4	.475	5.81+18	1.65+18	7.46+18	5.53+18	1.65+18	7.18+18	1.35+18	1.59+18	2.94+18
5	.650	1.77+18	1.59+18	3.36+18	1.75+18	1.57+18	3.32+18	2.80+17	1.26+18	1.54+18
6	.825	4.55+18	3.88+18	8.43+18	4.46+18	3.80+18	8.26+18	7.71+17	2.50+18	3.27+18
7	1.000	4.83+17	4.83+17	9.66+17	4.82+17	4.82+17	9.64+17	4.53+16	4.59+17	5.04+17
8	1.225	1.35+18	1.35+16	2.70+18	1.33+18	1.33+18	2.66+18	1.01+17	1.02+18	1.12+18
9	1.475	5.46+17	1.27+17	6.73+17	5.43+17	1.27+17	6.70+17	3.59+17	1.25+17	4.84+17
10	1.700	1.73+18	6.96+17	2.43+18	1.63+18	6.87+17	2.32+18	1.89+17	5.30+17	7.19+17
11	1.900	2.33+16	2.33+16	4.66+16	2.33+16	2.33+16	4.66+16	2.29+15	2.31+16	2.54+16
12	2.100	1.14+18	0	1.14+18	1.06+18	0	1.06+18	3.61+17	0	3.61+17
13	2.300	1.71+18	0	1.71+18	1.69+18	0	1.69+18	1.41+18	0	1.41+18
14	2.500	3.41+17	3.92+16	3.80+17	3.36+17	3.91+16	3.75+17	2.04+17	3.62+16	2.40+17
15	2.700	0	0	0	0	0	0	0	0	0

N	E, MeV	T=1hr Atmos.	T=1hr Plateout	T=1hr Total	T=2hr Atmos.	T=2hr Plateout	T=2hr Total	T=8hr Atmos.	T=8hr Plateout	T=8hr Total
1	.150	5.57+18	1.53+16	5.59+18	5.38+18	1.53+16	5.40+18	4.82+18	1.49+16	4.83+18
2	.250	5.75+18	3.60+16	5.79+18	5.06+18	3.60+16	5.10+18	2.84+18	3.51+16	2.88+18
3	.350	2.08+17	7.48+17	9.56+17	1.76+17	6.13+17	7.89+17	9.43+16	4.91+17	5.86+17
4	.475	1.08+18	1.57+18	2.65+18	6.01+17	1.50+18	2.10+18	1.37+17	1.05+18	1.19+18
5	.650	2.63+17	1.19+18	1.45+18	2.24+17	1.01+18	1.23+18	1.48+17	8.17+17	9.65+17
6	.825	6.99+17	2.23+18	2.93+18	4.99+17	1.47+18	1.97+18	1.45+17	7.94+17	9.39+17
7	1.000	4.32+16	4.52+17	4.95+17	4.05+16	4.26+17	4.67+17	2.88+16	1.97+17	2.26+17
8	1.225	9.08+16	9.50+17	1.04+18	7.06+16	7.39+17	8.10+17	3.42+16	1.31+17	1.65+17
9	1.475	3.39+17	1.25+17	4.64+17	2.66+17	1.24+17	3.90+17	6.73+16	1.13+17	1.80+17
10	1.700	1.27+17	4.95+17	6.22+17	5.45+16	3.86+17	4.41+17	1.74+16	3.43+16	5.17+16
11	1.900	2.20+15	2.31+16	2.53+16	2.18+15	2.31+16	2.53+16	2.07+15	2.16+16	2.37+16
12	2.100	3.00+17	0	3.00+17	1.96+17	0	1.96+17	3.95+16	0	3.95+16
13	2.300	1.33+18	0	1.33+18	1.03+18	0	1.03+18	2.29+17	0	2.29+17
14	2.500	1.80+17	3.53+16	2.15+17	1.07+17	3.19+16	1.39+17	5.89+15	3.31+15	9.20+15
15	2.700	0	0	0	0	0	0	0	0	0

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* read as 6.44×10^{18}

Table 2.1.66-5 (Cont.)

Containment Activity (γ/sec)

		t = 1 day			T = 4 days			T = 30 days		
N	E, MeV	Atmos.	Plateout	Total	Atmos.	Plateout	Total	Atmos.	Plateout	Total
1	.150	4.20+18	1.41+16	4.21+18	2.79+18	1.09+16	2.50+18	8.71+16	1.17+15	8.83+16
2	.250	7.94+17	3.32+16	8.27+17	8.88+15	2.58+16	3.47+16	2.43+14	2.75+15	2.99+15
3	.350	4.89+16	4.62+17	5.11+17	3.38+16	3.59+17	3.93+17	3.38+15	3.83+16	4.17+16
4	.475	6.23+16	6.28+17	6.90+17	5.35+15	5.57+16	6.11+16	1.12+14	5.91+7	1.12+14
5	.650	8.82+16	7.14+17	8.02+17	3.68+16	3.90+17	4.27+17	4.65+14	5.28+15	5.75+15
6	.825	6.38+16	6.60+17	7.24+17	3.05+16	3.25+17	3.56+17	1.12+14	1.27+15	1.38+15
7	1.000	1.63+16	1.72+17	1.88+17	7.07+15	7.53+16	8.24+16	2.62+13	2.96+14	3.22+14
8	1.225	9.39+15	9.89+16	1.08+17	3.92+14	4.18+15	4.57+15	3.89+5	4.40+6	4.79+6
9	1.475	9.82+15	9.26+16	1.02+17	4.09+15	4.35+16	4.39+16	1.49+13	1.68+14	1.83+14
10	1.700	3.20+15	3.37+16	3.69+16	1.92+12	2.07+13	2.26+13	0	0	0
11	1.950	1.79+15	1.89+16	2.07+16	9.40+14	1.00+16	1.09+16	3.51+12	3.97+13	4.32+13
12	2.10	7.11+14	0	7.11+14	1.05+7	0	1.05+7	0	0	0
13	2.300	4.24+15	0	4.21+15	6.23+7	0	6.23+7	0	0	0
14	2.500	3.15+14	3.31+15	3.63+15	1.89+11	2.03+12	2.22+12	0	0	0
15	2.700	0	0	0	0	0	0	0	0	0

Note: Containment free volume = 2.506×10^6 ft^3

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PROJECT SL1

TABLE 2.1.6b-6

DOSE RATES - RAB

SUBJECT Time Leaving Leaving - Summary of Results

Component or location	Dose Rate mrem/hr 1hr	10 hr	100 hr	1000 hr
1. Shutdown Cooling Heat Exchanger Cubicle				
A. side - no shielding - contact	3.5+7	1.5+7	4.9+6	1.9+5
B. " - " - 2 ft distance	9.0+6	4.0+6	1.3+6	4.9+4
C. " - " - 10 ft "	1.7+6	7.5+5	2.4+5	9.2+3
D. " - 2 ft concrete - just outside shield wall	6.9+3	2.0+3	3.8+2	1.0+1
2. Sampling Room				
A. tubing - 4 lines - 1 ft distance	1.4+6	6.1+5	1.9+5	7.6+3
B. Sample sink - 300 ml bomb - " "	4.1+6	1.8+6	5.8+5	2.3+4
C. Sample panel - contact	2.0+6	8.8+5	2.8+5	1.1+4
D. Sample Cooler - 1 ft distance	1.4+7	6.0+6	1.9+6	7.5+4
E. Doorway - without cooler	1.5+5	6.8+4	2.2+4	8.5+2
" with cooler	2.9+5	1.3+5	4.1+4	1.6+3
F. General room dose rate from "external" sources (primarily Shutdown Cooling HT. Exs. + ECCS room)	1.1+4	4.8+3	1.5+3	6.1+1
3. Letdown Heat Exchanger Cubicle				
A. side - no shielding - contact	1.3+8	5.7+7	1.8+7	7.0+5
B. " " - 5 ft distance	4.2+6	1.8+6	5.9+5	2.3+4
C. " " - 10 ft "	1.3+6	5.7+5	1.8+5	7.0+3
D. " - 1½ ft conc. - just outside shield wall	2.0+4	6.8+3	1.5+3	5.0+1
4. Hydrogen Analyzer				
A. No shielding - 1 ft distance	1.1+5	4.8+4	1.5+4	5.9+2
" - 2 ft "	2.7+4	1.2+4	3.8+3	1.5+2
" - 3 ft "	1.2+4	5.3+3	1.7+3	6.5+1
" - 4 ft "	6.9+3	3.0+3	9.7+2	3.7+1

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BY J. OSTR.. DATE 12/17/77 SHEET 2 OF 7

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PROJECT SL1

SUBJECT TMI Licensing Basis - Summary of Results

TABLE 2.1.6b-6 (Contd)

DOSE RATES - RAB

Component or Location	Dose Rate, mrem/hr			
	1 hr	10 hr	100 hr	1,550 hr
5. Radiochemistry Lab (are rate mainly from -0.50' wall)				
A. General Area Rate - Excluding Sample Room	1.2+4	2.8+3	3.7+2	9.8+0
B. " - Sample Room - no sample coolers (1)	1.3+4	3.0+3	4.0+2	1.1+1
C. " - " + sample coolers (1)	1.3+4	3.2+3	4.2+2	1.2+1
6. Containment Atmosphere Monitor (midheight distances)				
A. No shielding - contact (with reactor bldg)	1.1+4	-	-	-
B. " - 10 ft distance (from Reactor Bldg)	9.7+3	-	-	-
C. " - 20 ft "	8.4+3	-	-	-
D. " - 50 ft "	5.8+3	-	-	-
E. " - 100 ft "	3.5+3	-	-	-
7. Volume Control Tank Cubicle				
A. Side - no shielding - contact	7.2+8	3.1+8	9.6+7	3.8+6
B. " - 3 ft concrete - 8 ft distance just outside shield wall	1.4+4	2.9+3	3.3+2	8.2+0
8. Holdup Tank Cubicle (single tank - midheight)				
A. Side - no shielding - contact	1.5+9	6.4+8	2.0+8	7.9+6
B. " - 3 ft concrete - just outside shield wall	2.3+4	4.7+3	5.3+2	1.3+1

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PROJECT SL-1

SUBJECT TMI Lessons Learned - Summary of Results

TABLE 2.1.6b-6 (Cont'd)

DOSE RATES - RAB

Component or Location	Dose Rate, mrem/hr			
	1 hr	10 hr	100 hr	1000 hr
9. RAB Corridor at EL -0.50' (Holdup Tks. not incl.)				
A. West end of corridor near side of Shutdown Cooling Ht. Ex. Cubicle (Ht. exch. main source)	1.9+3	5.0+2	7.7+1	2.0+0
B. Near Staircase RA-7 (ECCS piping main source)	7.8+3	2.3+3	4.3+2	1.2+1
C. Radwaste panel 1B-2 next to Holdup Tk. 1A (note - Holdup Tanks not included; ECCS main source)	1.3+4	3.8+3	7.0+2	2.0+1
D. Radwaste panel 1A-2 next to charging pump area (ECCS + charging pump piping - main source)	3.5+3	1.2+3	2.5+2	8.0+0
E. Near Staircase RA-3 (charging pump piping - main source)	1.0+3	3.4+2	7.2+1	2.0+0
F. East end of corridor near gas decay tank corridor entrance (decay tanks not considered)	6.0+1	1.8+1	3.0+0	1.0-1
10. RAB Corridor at EL. 19.50' (Holdup Tks. not incl.)				
A. Near Staircase RA-7 (ECCS + shutdown cooling main source)				
with Sample Room	6.9+2	1.7+2	2.6+1	7.8-1
without " "	5.6+2	1.7+2	1.6+1	4.4-1
B. Near waste management panels in front of Holdup Tks (note: Holdup Tks not incl.)				
with Sample Room	1.7+3	6.3+2	1.6+2	5.7+0
without " "	1.7+2	4.4+1	6.8+0	1.8-1
C. Near Staircase RA-3				
with Volume Control Tk.	1.4+4	2.9+3	3.3+2	8.2+0
without " "	3.1+0	6.2-1	6.2-2	1.4-3
11. Control Room No new NUREG-0578 sources beyond LOCA considered in FSAR. ∴ < 5 rem dose for duration of accident from airborne + direct shine.	1.8+1 (direct)	-	-	-

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2.1.7a AUTO INITIATION OF THE AUXILIARY FEEDWATER (AFW) SYSTEM

As discussed in our earlier responses, as well as in numerous NRC information requests and telephone conferences, there is a significant concern regarding the applicability of current FSAR safety analyses if the NRC requirements in this area are implemented. To resolve this issue, Florida Power & Light Company, in conjunction with other owners of the CE NSSS, has contracted CE to perform analyses which are expected to yield acceptable results which will resolve the outstanding questions and allow full implementation of an automated auxiliary feedwater system. Pending the completion of these analyses and subsequent NRC approval, we do not intend to implement the fully automatic system required by NUREG-0578 at this time. However, installation of the semi-automatic system described below is in progress. The isolator qualification problems discussed in response to requirement 2.1.3.b also apply to this system, but implementation is expected on or before February 15, 1980.

Design Bases

The design provides automatic start of the AFW pumps. A failure in the automatic circuit does not result in loss of manual capability.

The automatic pump starting circuit is designed so that single failure does not result in loss of system function.

Manual capability of pump start from the Control Room is retained. Single failure in the manual circuits does not result in loss of system function.

The AC powered pumps and valves are included in the diesel generator sequence.

The system is designed as control grade.

System Function

The Auxiliary Feedwater System (AFWS) pumps are automatically started to satisfy the requirements of AFWS automatic start set forth in GDC 2 Appendix A to 10CFR50.

System Description

The AFW pumps are started once any of the steam generators reach a preset low level. The automatic AFW initiation signal starts the two motor driven AFW pumps and opens the two steam supply valves to the turbine driven pump.

In case of loss of offsite power the AFW pumps are automatically loaded on the diesel generators.

The system is designed as a control grade system. Separation between safety grade components and the new control grade equipment is maintained with safety grade isolation devices.

System Components

The following major equipment is furnished as part of the control grade system:

(8) Level Bistables - commercial grade

(8) Signal Isolators - tested and qualified to IEEE for the control room enviroment

(9) Relays - tested and qualified to IEEE for the control room enviroment

Testing and Inspection

The equipment and components are purchased as specified in "System Components" above for the safety grade equipment. Vendors substantiate through test, calculations and/or operational data that the components remain operable under the seismic loads.

2.1.7b AUXILIARY FEED FLOW INDICATION

The St. Lucie Unit 1 design incorporates safety-grade auxiliary feedwater flow indication in the Control Room which meets the intent of this requirement.

2.1.8a POST - ACCIDENT SAMPLING CAPABILITY

A plant procedure (C-100) has been written to provide interim guidelines and instructions for sampling and analysis of reactor coolant, containment, gas decay tanks, liquid waste, plant vent and fuel building after a loss of coolant or other similar accident.

Design Bases

Plant operators shall be able to obtain a sample under accident conditions without incurring a radiation exposure in excess of 3 Rems whole body and 18 3/4 Rems to extremities.

Plant operators shall be able to obtain samples within one hour after start of accident.

Release of fission products shall correspond to assumptions made in Regulatory Guide 1.4 (Bases for Source Terms).

Effects of direct radiation from piping and components in the Auxiliary Building shall be considered.

Effects of direct radiation from airborne effluents shall be considered.

Samples shall indicate the degree of cladding failure, high fuel temperatures and degree of fuel melting.

The sampling system shall have the capability to analyze RCS samples for pH, activity, Boron, Hydrogen and Oxygen.

The sampling system shall have the capability to analyze the containment air under positive and negative pressure.

Function

Provide information related to the extent of core damage that has occurred or may be occurring during an accident.

Determine the types and quantities of fission products released to the containment in the liquid and gas phase.

Provide information on coolant chemistry and containment hydrogen.

System Description

The criteria used in the design of the modified sampling system to have a post accident sampling capability shall be based on the calculated post accident radiation dose in the area where all the sample taking, handling and analysis activities are conducted during normal operation.

All radiochemical and chemical analyses will be done in the laboratory facilities away from high radiation zones.

Sample collection, and handling, as well as in line monitoring of reactor coolant is accomplished with equipment and instrumentation that are solely dedicated to function during accident and post accident conditions. This facility is designated as the Post Accident Sampling System.

The Hot Leg Loop & Shutdown Cooling Suction line are sampled. Figure 2.1.8.a-1 thru 3 shows the routing. Table 2.1.8.a-2 lists the chemical and radiochemical analyses.

The guideline used in the sampling and monitoring of reactor coolant is to limit the quantity of the reactor coolant in the post accident sampling facility to a minimum at all times. Sample collection and in-line monitoring is done in sequence with an intervening step consisting of a demineralized water purge. These procedures are graphically shown in Figures 2.1.8a-4 through 2.1.8.a-7.

Reduction of sample collected to less than 1 ml for radiological spectrum analysis and Boron concentration.

Intermittent inline monitoring of pH and dissolved gases (oxygen and hydrogen).

Provisions for complete purging of the sampling lines and associated equipment with demineralized water. Purging is done by remote manual operation.

Provisions for use of specially designed shielded sample container that can be safely transported to a remote laboratory for analysis.

The hotleg sample line (3/8" stainless steel) ties in downstream of the primary containment isolation valve (V-5208). The high pressure and high temperature reactor coolant sample is cooled to 120°F and depressurized to approximately 25 psig. The sample line then branches out to supply a grab sample station, inline analyzers and monitors. The pressure reduction station and inline probe holders will be shielded.

The shutdown cooling suction line is routed in the same manner as the hotleg sample. The pressure reducing station and in-line probe holders will be common to both sample lines. Sample coolers will be moved to the penetration room. Remote purging of the system with demineralized water is provided.

The sample lines emerging from the sample station shall be discharged back into the flash tank.

Dose rate data for the optimum combination consistent with the NUREG-0578 requirements will be provided. This combination shall be adopted as the final design of the post accident sampling system.

System Components

The following system components are designed to quality group D and nonseismic category I.

Piping, valves and associated components - constructed of stainless steel.

Sample Coolers - Shell and tube type heat exchange sample stream flows through the tube while the cooling water passes through the shellside. Cooling water is supplied from the Component Cooling Water System.

Dissolved Oxygen, Hydrogen and pH Analyers - analyzers are in-line type with remote indicator transmitter, for readout, and/or recording.

Instrumentation - Pressure, temperature, and flow instrumentation will be provided to monitor system performance.

TABLE 2.1.8.a-1

CALCULATED DOSE RATES

AT ACCIDENT CONDITIONS EXISTING FACILITIES

AREA	DOSE RATE
1. Primary Sample Room (General Dose)	1,000 R/HR
a) From a single sample cooler	14,000 R/HR one ft dist.
b) From single 3/8" sample line	350 R/HR "
2. Chemistry Counting Room (Hot Lab)	12 R/HR
3. Radiochemistry Laboratory	12 R/HR

REFERENCE 1. Marked up drawing No. 8770-G-070. General Arrangement Drawing RAB elevation 19.5. Figure 2.1.8.a-1

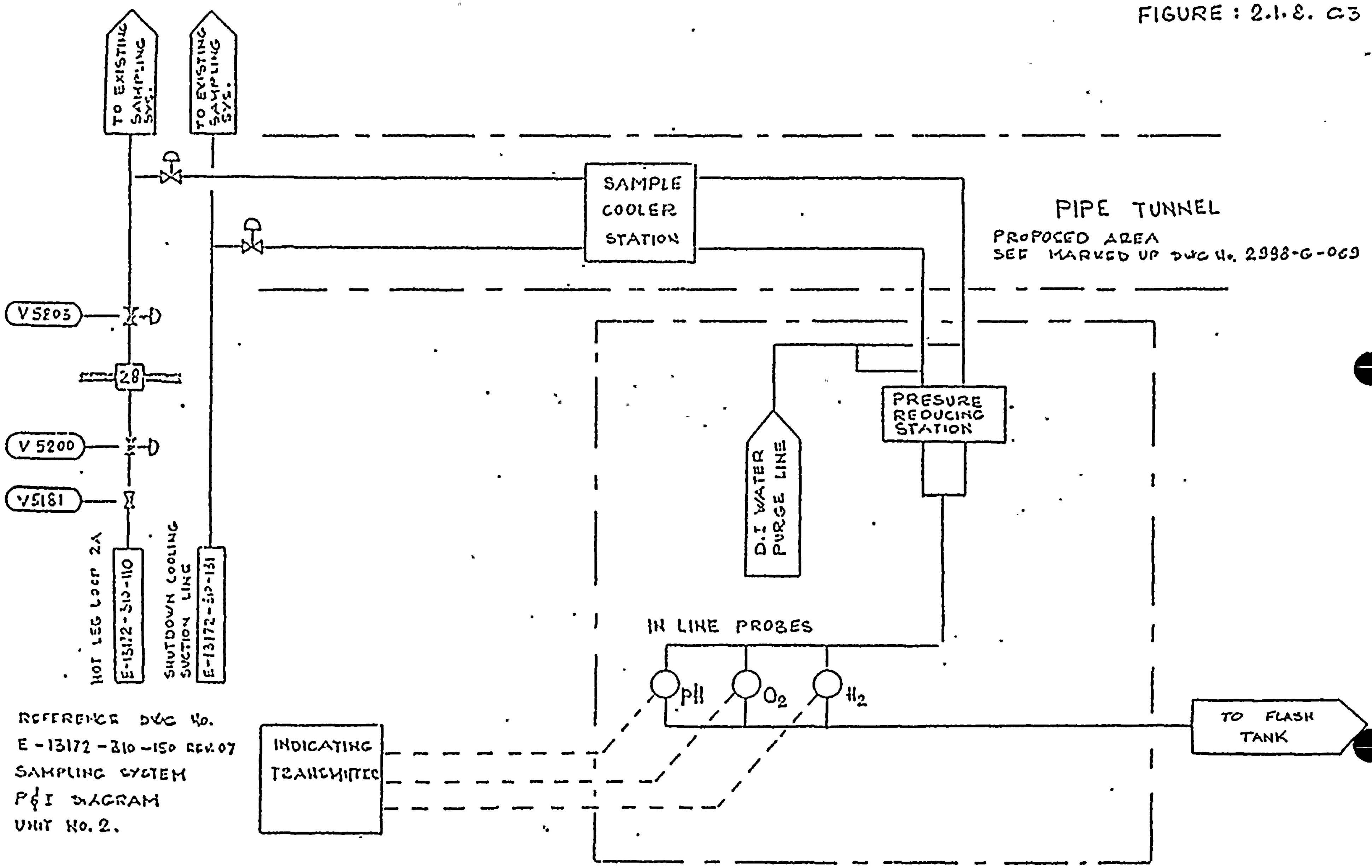
TABLE 2.1.8.a-2

POST ACCIDENT SAMPLING

REACTOR COOLANT

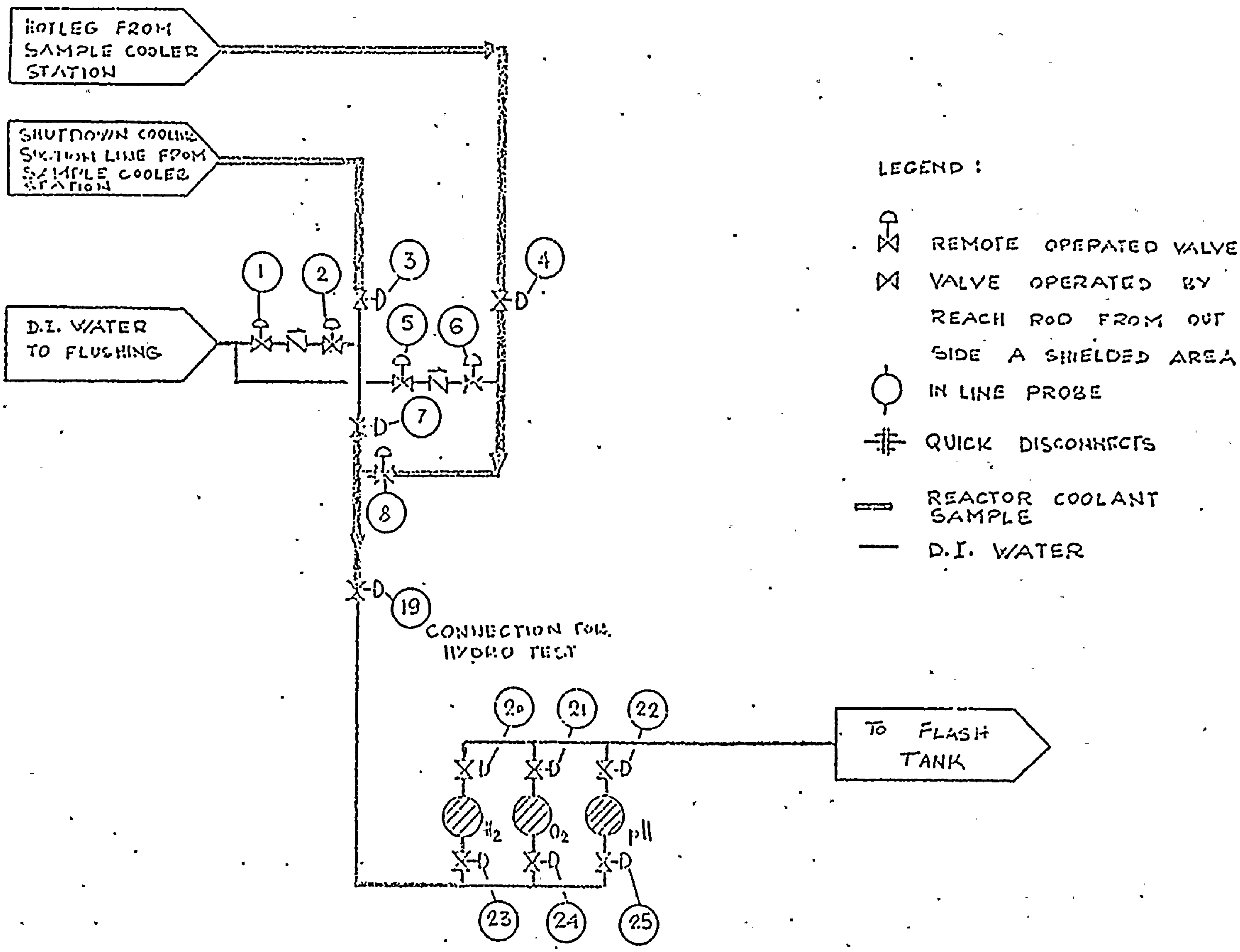
CONSTITUENT ITEM	METHOD OF SAMPLING	SAMPLE VOLUME OR FLOWRATE	REQUIREMENT TIMING	REQUIREMENT RESULTS
1. Radiological Analysis	GRAB		All Samples taken within one hour after accident	Results required within one hour after sampling.
Iodine Cesium Non Volatile Isotopes		< 1 ml		
2. Chemical Analysis				
a) Diss Oxygen	In line Monitoring	200 mlpm		
b) Diss Hydrogen	"	"		
c) pH	"	"		
d) Boron	Grab	< 1 ml		

FIGURE : 2.1.8. C3



REFERENCE DWG No.
E-13172-310-150 REV.07
SAMPLING SYSTEM
P&I DIAGRAM
UNIT No. 2.

POST ACCIDENT SAMPLING SYSTEM
ROUTING SCHEMATIC



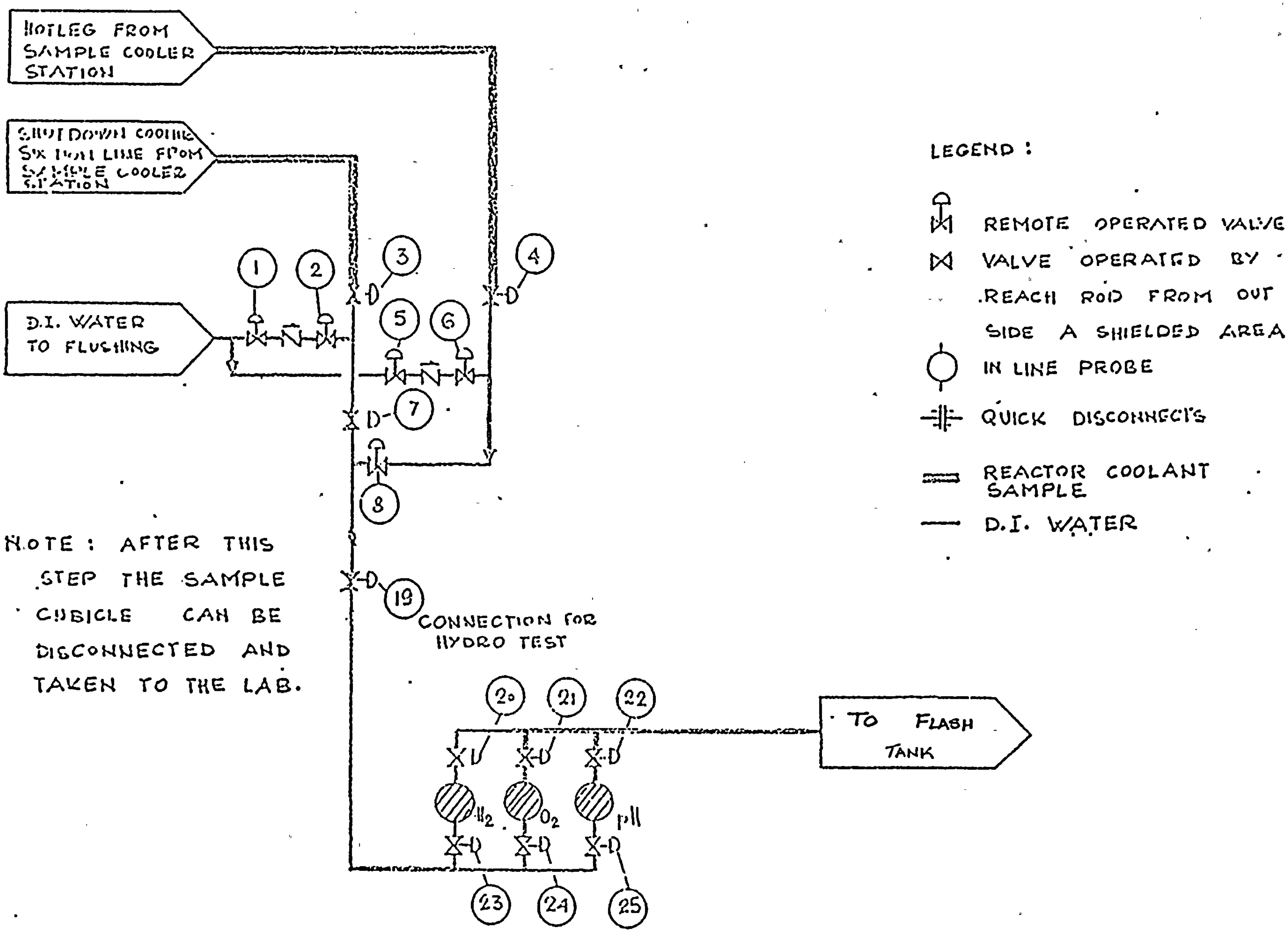
STEP No I . SAMPLE COLLECTION FLOW PATH

VALVE CLOSE : 6 7 19

VALVE OPEN : 4 8

SAMPLE COLLECTION PROCEDURE

FIGURE : 2.1.8. a5.



NOTE : AFTER THIS STEP THE SAMPLE CUBICLE CAN BE DISCONNECTED AND TAKEN TO THE LAB.

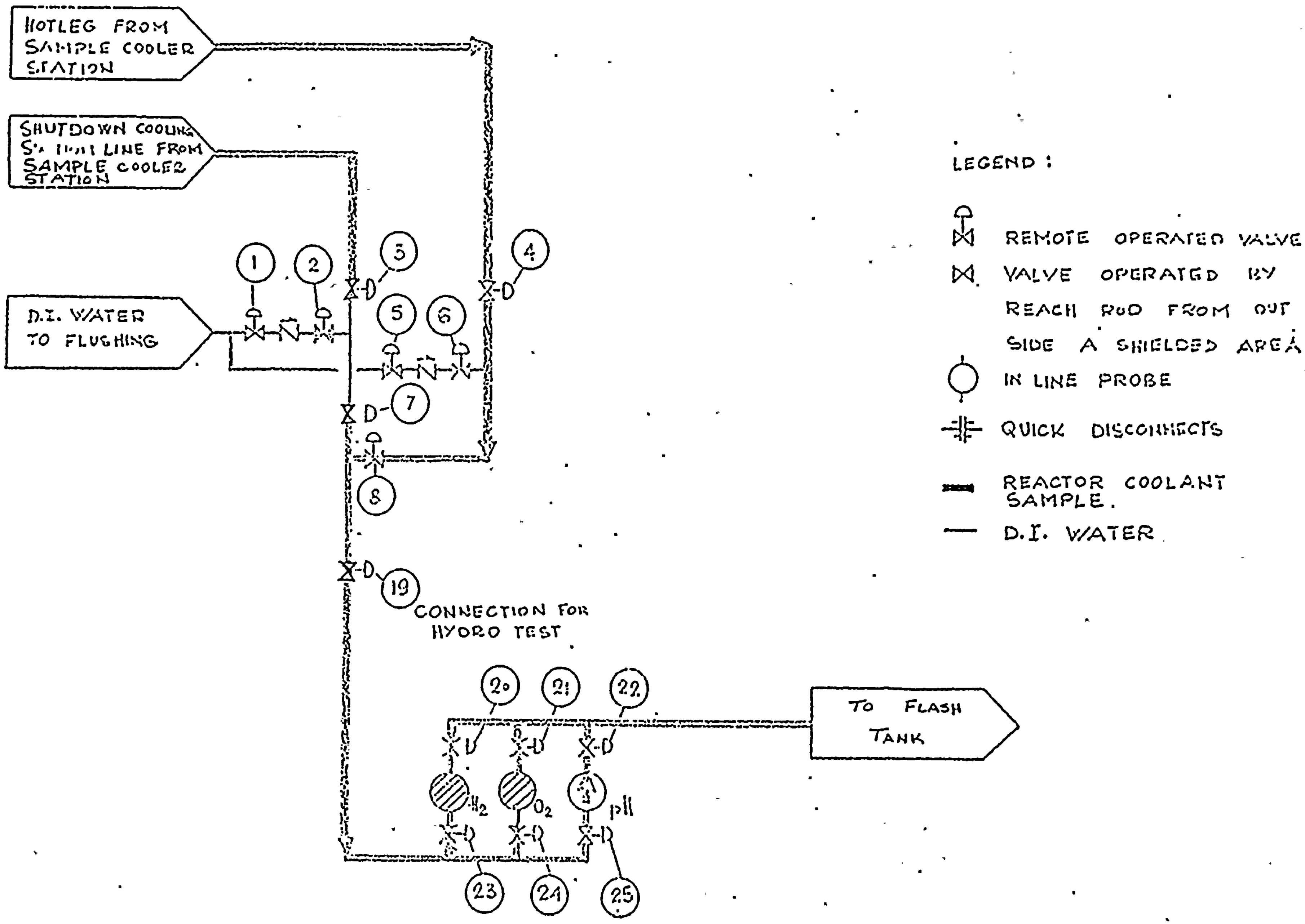
STEP No 2, REMOTE PURGING OF SAMPLE LINE AFTER SAMPLE IS ISOLATED

VALVE CLOCE : 4

VALVE OPEN : 5 6

FIGURE : 2.1.8.06.

IN LINE MONITORING PROCEDURE

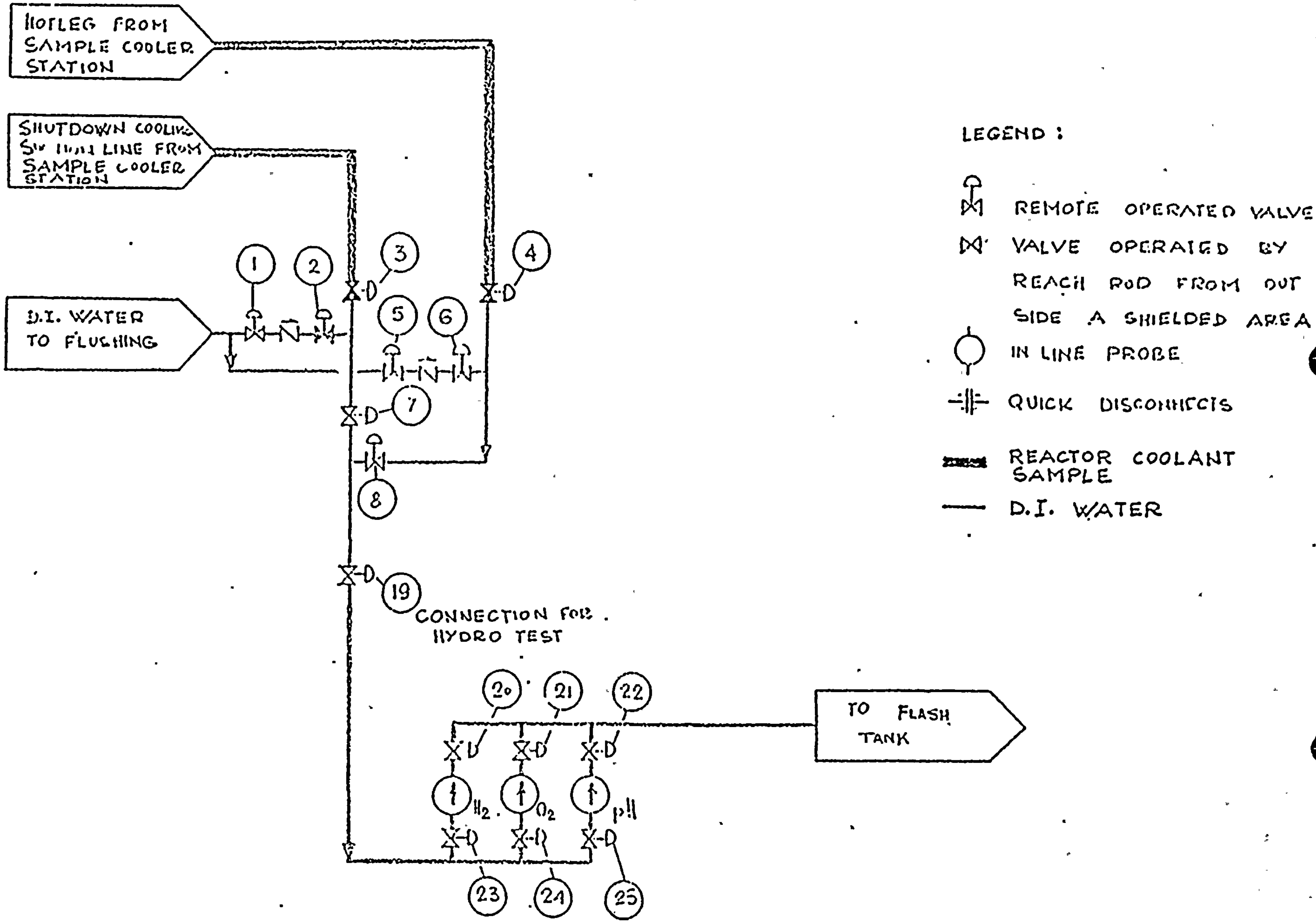


STEP No I FLOW PATH FOR pH MONITORING

VALVE . CLOSE : 6 7 9 20 21 23 24

VALVE OPEN : 4 8 19 22 25

IN LINE MONITORING PROCEDURE



STEP No 2. FLOW PATH FOR REMOTE PURGING

VALVE CLOSE : 4

VALVE OPEN : 5 6 20 21 23 24

2.1.8b INCREASED RANGE OF RADIATION MONITORS

A plant procedure (C-101) has been written to provide interim capabilities to measure high level noble gases through the use of standard instrumentation by providing sample dilution with precision flowmeters. Under accident conditions, radiation from the exhaust air is measured through the Plant Vent and the Fuel Building Effluent Monitors utilizing the Condenser Air Ejector and the Drumming Station Monitors, respectively. In both cases continuous Control Room readout is provided and portable NMC monitors will be used for backup. Appropriate graphs will be used to determine the activity release from the monitor readouts.

Permanent high range monitors are scheduled for installation by January 1, 1981.

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION

A plant procedure has been written which provides that, if required during the interim period (1/1/80 to 1/1/81), air samples will be drawn through silver zeolite filters, purged of noble gases by bypassing air over the silver zeolite, and analyzed for Iodine on a GeLi detector.

Permanent, portable monitors are being evaluated for installation by January 1, 1981.

2.1.9 REACTOR COOLANT SYSTEM VENTING

The following is a description of the proposed design package.

Design Bases

The vent flow rate capability shall be based upon the following considerations:

> The vent rate shall be sufficient to vent one-half of the RCS volume in one hour.
>
> The vent mass rate shall not result in heat loss from the RCS in excess of the normal pressurizer heater capacity.

The vent system shall be designed Seismic Category I and safety related.

The vent system shall be designed for a single active failure.

The system shall be designed to limit RCS mass loss to below the definition of a LOCA in 10CFR50, Appendix A.

The vent system shall be operable following design basis events and loss of offsite or onsite AC power.

The vent system shall be capable of venting directly to containment or to the quench tank.

The vent system shall be designed to vent superheated steam, steam water mixtures, water, fission gases, helium, nitrogen and hydrogen. The system design parameters are as follows:

Flow	100 scfm
Design Temperature	700°F
Design Pressure	2500 psia
Line Size	1"

Control Room position indication is provided for the power operated valves.

The system is designed not to interfere with refueling maintenance operations.

Function

The system allows for remote venting of the RCS via the reactor vessel head vent or pressurizer steam space vent during post accident situations when large quantities of non-condensible gases may collect in these high points.

The system shall also be used in normal maintenance procedures.

System Description

The RCS Vent System design as shown in Figure 2.1.9-1 is consistent with the design philosophy used in all engineered safety feature systems. The system is redundant and functions with allowance for a single active failure. The system is not designed to be used during plant normal operation. The power operated valves are provided with Class 1E power supplies. The system valves are disconnected from their power sources and administratively locked to eliminate the possibility of inadvertent system actuation during normal operation.

In the event of an accident in which large quantities of non-condensible gases are generated within the RCS, operating procedures allow the RCS vent system valves to be connected to their power sources. The valves are also capable of being connected to an emergency power source in the event of loss of offsite power. Pressure reduction of the gas is accomplished through the use of orifice plates in the piping system. Although the primary function of the orifice plate is to reduce vented gas pressure, they are sized such that in the event of system failure or rupture, the RCS mass loss through the orifice is less than a LOCA as defined in 10CFR50, Appendix A.

The RCS Vent System will also be used in conjunction with plant maintenance procedures. Venting of the pressurizer and reactor vessel venting can be accomplished by directing vent flow to the quench tank for this operation.

A single failure analysis verifies that the failure of any single power operated valve to open or close is not detrimental to system operation. This is accomplished through the redundancy in system valves.

Consistent with NRC requirements, the system shall be designed to limit mass loss to less than a LOCA as defined in 10CFR50, Appendix A, and thus a separate analysis of inadvertent system operation or pipe breakage is not required. A Failure Mode and Effects Analysis (FMEA) is provided by the NSSS Supplier.

The components, piping and supports of the RCS vent system are specified as Seismic Category I.

Piping and valves are constructed of austenitic stainless steels. The system shall be designed to Seismic Category I and ASME Code Class 1 or 2 as indicated on Figure 2.1.9-1. Power operated valves are solenoid operated type designed to fail closed. Redundancy in valve configuration and power supplies is designed to meet single failure criteria.

Instruments, Controls, Alarms and Protective Devices -

The system shall be designed to be controlled remotely from the Control Room. Instrumentation shall be powered from emergency power sources and alternate sources are used as necessary to meet single failure criteria. Position indication (open/shut) shall be provided for the remotely operated valves and displayed in the Control Room. Pressure instrumentation shall be also provided to monitor system performance and valve leakage. Pressure indication and a high pressure alarm are also located in the Control Room.

Figure 2.1.9-1

REACTOR COOLANT GAS VENT SYSTEM

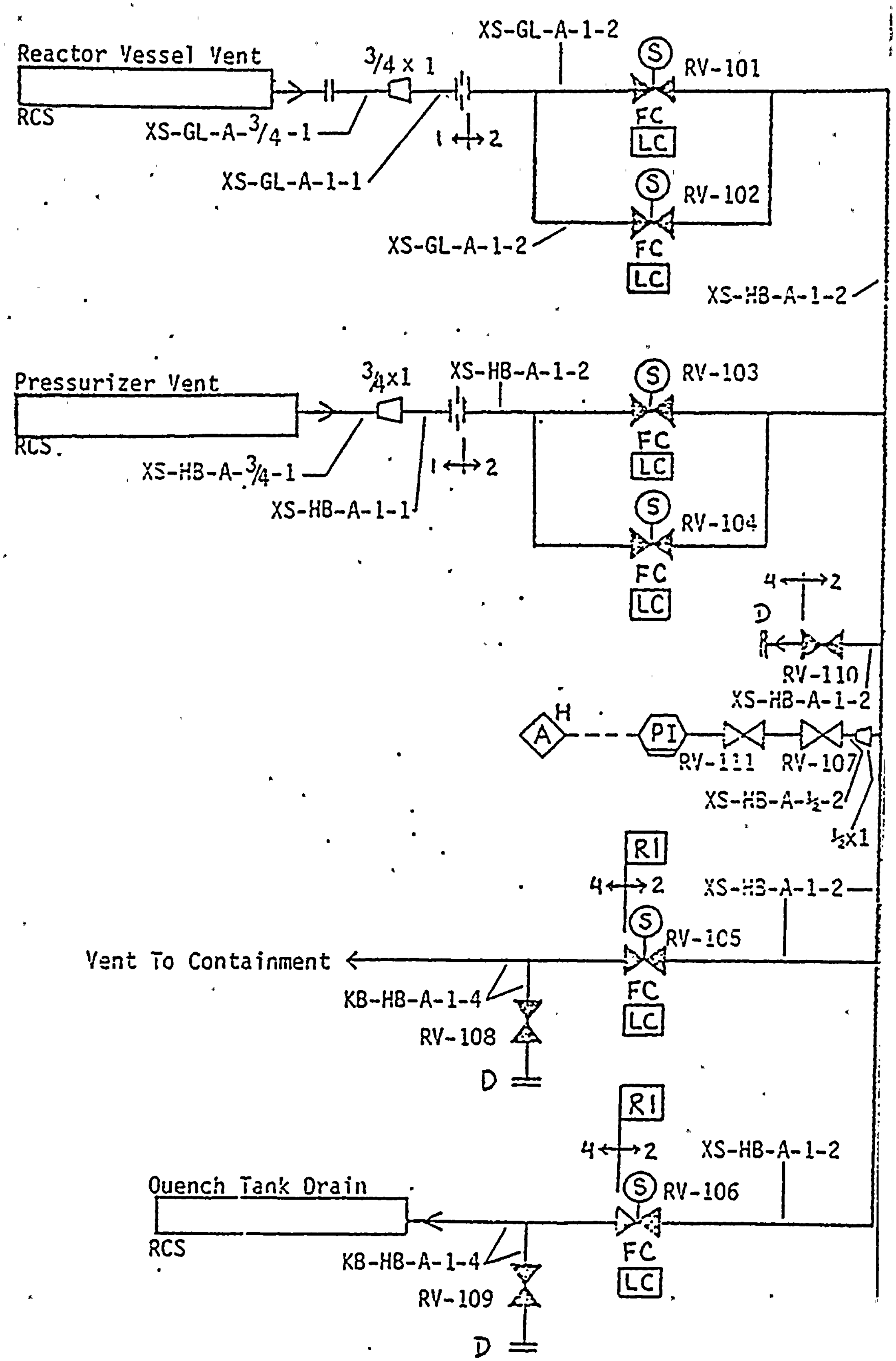


Figure 2.1.9-1

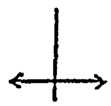
Symbols and Abbreviations

Line Designation: XX - XX - X - X - X

- Safety Class (1,2,3,4)
- Size (Nominal Pipe)
- Material: A-Stainless Steel
- Design Temp (°F)
- Design Pressure (psig)

Pressure/Temperature Code

	1st Letter	2nd Letter
B	-	00
G	6	-
H	7	-
K	10	-
L	-	50
S	-	85
X	24	-

Class Break: Class  Class

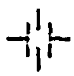
Fail Close: FC

Lock (Key, Electrical) Close: LC

Reactor Coolant Pressure Boundary: RI

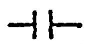
Drain: D

Solenoid Operator: S

Orifice: 

Reducer/Expander: 

Remote Pressure Instrument Readout and High Alarm: PI - - - A H

Flanged Connection: 

Valve Designation: RV-XXX

2.1.9.a ANALYSIS OF DESIGN AND OFF-NORMAL TRANSIENTS AND ACCIDENTS

Transient and Accident Analysis

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling have been performed on a generic basis by the Combustion Engineering Owners Group of which Florida Power & Light Company is a member. The small break analyses have been completed and are reported in CEN-114, which was submitted to the Bulletins and Orders Task Force by the Owners Group. The guidelines of these analyses have been incorporated into plant procedures and initial training of operators on the new procedures has been completed.

The inadequate core cooling analyses have been completed and are reported in CEN-117, which was submitted to the Bulletins and Orders Task Force by the Owners Group. The guidelines of these analyses have been incorporated into the appropriate plant procedures and training of operators on the new procedures has been completed.

2.2.1a SHIFT SUPERVISOR RESPONSIBILITIES

The existing plant procedure, Duties and Responsibilities of Operators on Shift (OP-0010120) has been revised to comply with the requirements of NUREG-0578.

Additionally, a corporate directive has been issued which delineates Shift Supervisor duties and responsibilities and the role of the new Shift Technical Advisor position.

2.2.1b PROVIDE ON-SHIFT TECHNICAL ADVISOR

Personnel have been selected to fill the Shift Technical Advisor positions and have been given initial training to the extent possible. The Shift Technical Advisor position was filled on a continuous, round the clock basis beginning January 1, 1980.

2.2.1c SHIFT TURNOVER PROCEDURE

Appropriate plant procedures have been revised to institute a system of shift turnover checklists which have been developed for each operating station to be completed by the off-going watch stander and reviewed by the oncoming operator.

2.2.2a CONTROL ROOM ACCESS

Appropriate plant procedures have been revised to include means for limiting Control Room access to meet the intent of this requirement.

2.2.2b ON-SITE TECHNICAL SUPPORT CENTER (TSC)

St. Lucie Plant has designated the training classroom area (formerly the Unit 2 Control Room located adjacent to the Unit 1 control room) as the On-Site Technical Support Center (TSC). A new St. Lucie Plant procedure, which has been drafted and reviewed by the Facility Review Group, delineates the activation, manning, and use of the TSC during emergencies.

The following is a description of the proposed design package.

Design Bases

Accomodation for twenty-five occupants shall be provided.

Telephone communications shall be provided to reach: the offsite emergency response center, the NRC, the control room, the operational support center, Combustion Engineering and the FPL General Office building.

The following documents shall be available to the TSC:

- System Descriptions
- FSAR & Emergency Plan
- Emergency Procedures
- P&ID's and Electrical Schematics
- General Arrangement Drawings

Water shall be provided for the twenty-five occupants for 72 hours following a LOCA & DBE (75 gal. @ 1 gal/man/day).

Food shall be provided for the twenty-five occupants for 72 hours following a LOCA & DBE.

Two sanitation kit III's shall be provided. (One kit provides for 25 persons for two weeks).

First aid supplies shall be provided in a 36 unit industrial first aid kit.

Portable fire protection shall be provided.

Several systems are being evaluated for the display of plant data in the TSC.

The HVAC system shall be upgraded to maintain a "Control Room Level" environment during POST/LOCA environmental conditions.

Portable monitors will be provided in the TSC to monitor both direct radiation and airborne radioactive contaminants.

Function

The Technical Support Center shall be an area outside of and adjacent to the Control Room for the use of technical and management personnel in support of the operating staff during Post Accident conditions. Figure 2.2.2.b-1 shows short and long term general arrangement of the TSC.

System Description

While several systems for the display of plant data in the Technical Support Center are under consideration, only one is sufficiently developed and reviewed at this time to the point that a detailed system description can be provided. This is the closed-circuit television system described in the following paragraphs. Other systems undergoing evaluation include a computer-based system designed by our NSSS vendor and other designs which duplicate, in the TSC, selected Control Room meters, recorders and associated equipment.

The television system under consideration for the Technical Support Center includes six high resolution black and white cameras and one color camera. Each camera will be complete with monitor and camera control, for the purpose of monitoring meters, gages, recorders, controls, switches and status lights mounted on control and protection panels in the main Control Room.

The television cameras, located in the Control Room, are ceiling mounted on separate pan - tilt mounts. Each camera, including the color camera, is equipped with a 10:1 zoom lens and with the necessary amplification for the purpose of close observation of panel meters and control status (1/8 of an inch minimum character size).

The remote controls for the pan - tilt and zoom functions are located in the Technical Support Center. The color camera is provided to ascertain the status of certain color recorders on the Post Accident Monitoring Panel.

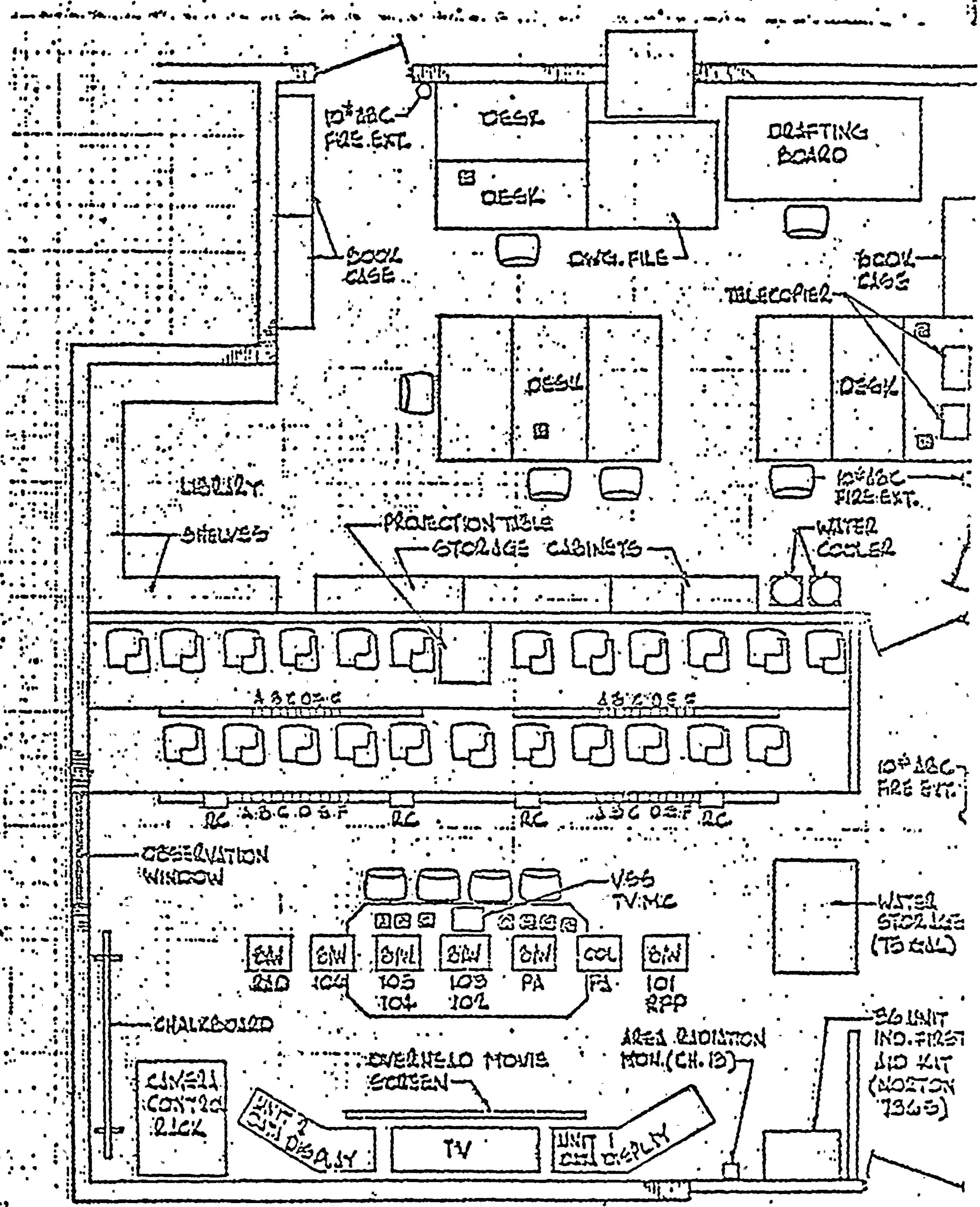
A 17 inch high-resolution monitor is provided for each black and white camera output, as well as a 17 inch color monitor for the color camera output. These monitors are ceiling mounted in the Technical Support Center. Corresponding remote pan - tilt and zoom controls are located beneath each monitor. This mode of operation provides easy access for the viewing of any one of the critical control panels in close-up or expanded view.

A five foot, high resolution, screen television projection system is provided in the Technical Support Center, and a 14 x 1 video switcher is provided for the purpose of viewing any one of the high resolution camera outputs.

The remote camera electronics are located in a separate cabinet in the Technical Support Center, and will be used to align the cameras and maintain adequate picture quality.

Since the Technical Support Center is designed for two unit operation (St. Lucie Unit 1 and the St. Lucie Unit 2, now under construction), a 4 x 1 switcher is provided for each monitor and control location to select between the two Control Rooms. The switcher for the large screen display includes the expanded facility for two Control Rooms. These additions assume the duplication of the cameras and their respective controls.

The CCTV system shall have the capability to historically record all monitored parameters via CCTV video tape playback. Initiation of the historical record shall be from the Control Room.



GENERAL ARRANGEMENT - TECHNICAL SUPPORT CENTER

FIGURE 2.2.2.b-1

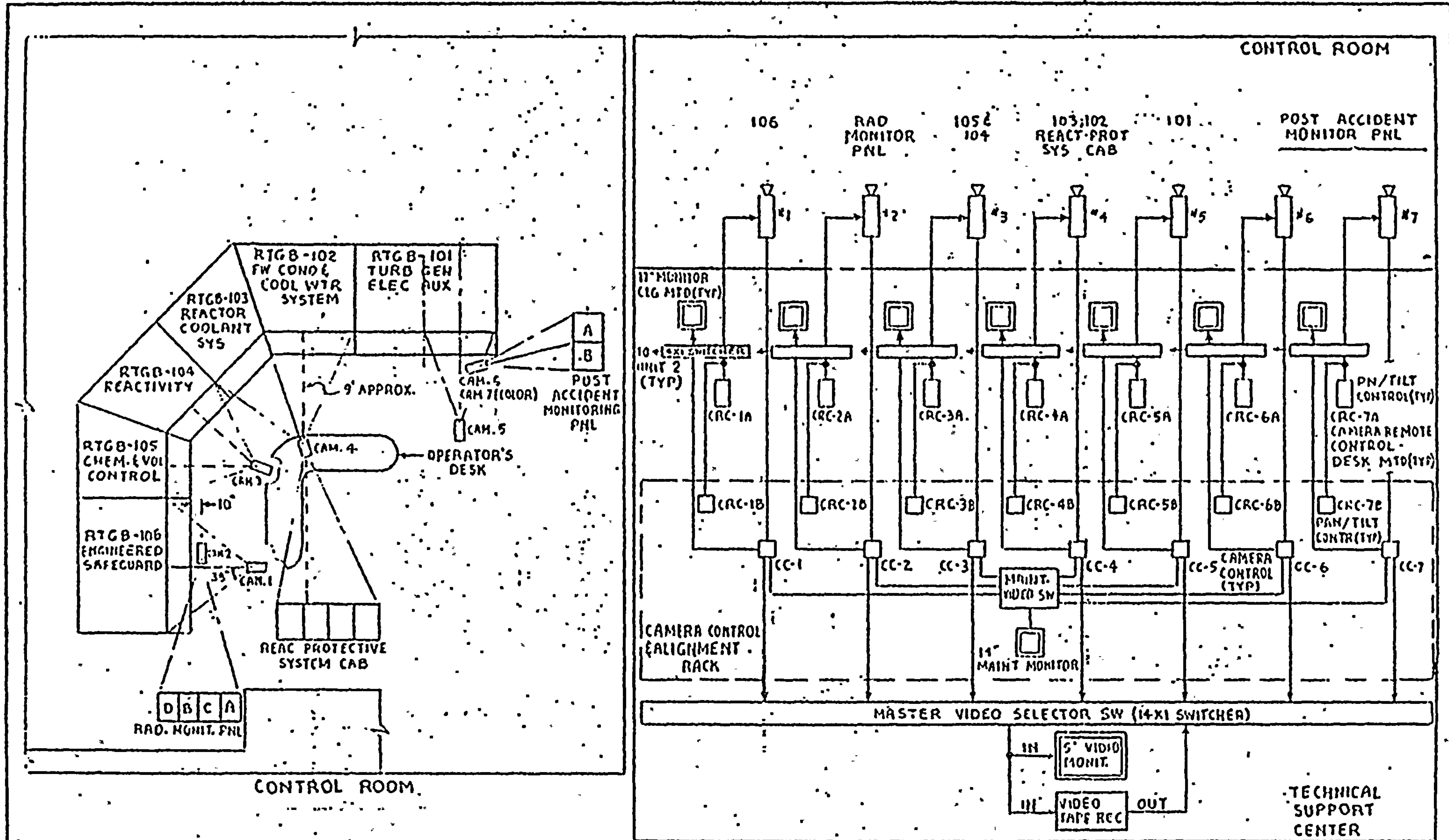


FIGURE 2.2.2.b-2

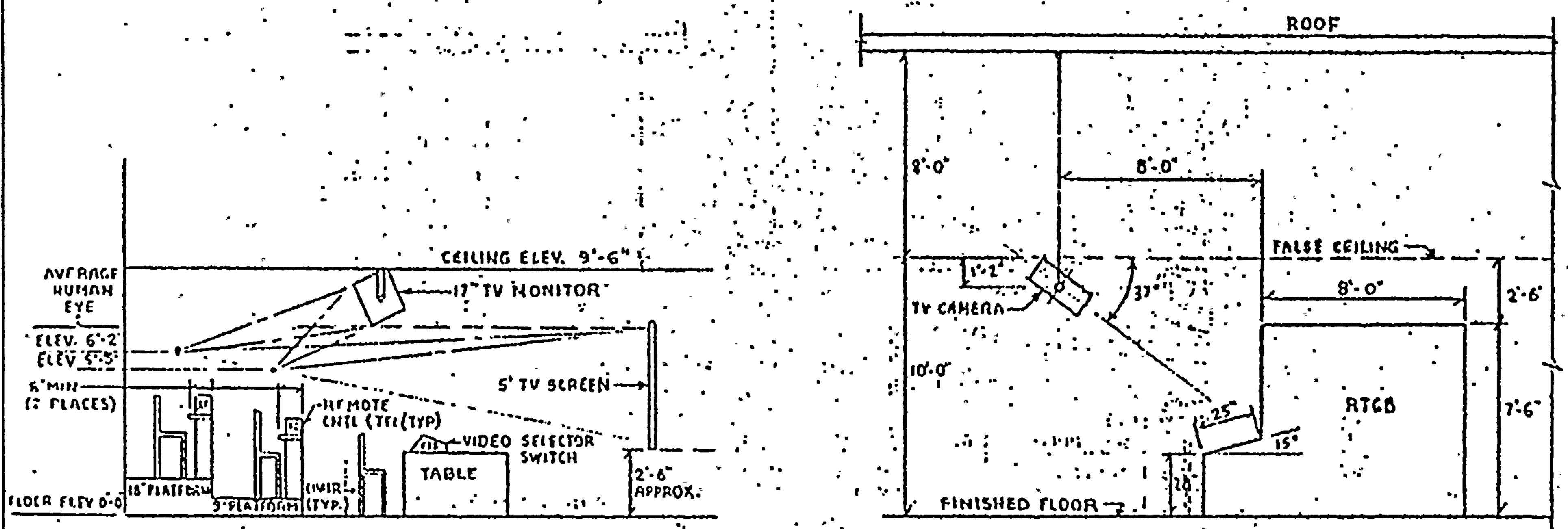
REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
8				4			
7				3			
6				2			
5				1			

EBASCO SERVICES INCORPORATED
NEW YORK

DIV. ELEC DR. A.M.
SCALE ~ CH.
DATE DEC 17 1979
APPROVED

FLORIDA POWER & LIGHT CO.
ST. LUCIE PLANT #1
TECHNICAL SUPPORT CENTER
CCTV SYSTEM

SK-TSC-01
SHEET 1



SUGGESTED MONITOR MOUNTING

(TECHNICAL SUPPORT CENTER)

SUGGESTED CAMERA MOUNTING (TYP.)

(CONTROL ROOM)

NOTE
1. DIMENSIONS ARE APPROXIMATE

FIGURE 2.2.2b-3

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
8				4			
7				3			
6				2			
5				1			

EBASCO SERVICES INCORPORATED
NEW YORK

FLORIDA POWER & LIGHT CO.
ST LUCIE PLANT #1
TECHNICAL SUPPORT CENTER
CCTV SYSTEM

SK-TSC-01
SHEET 2

2.2.2c ONSITE OPERATIONAL SUPPORT CENTER

St. Lucie Unit 1 has designated the first floor maintenance area of the Service Building as the on-site Operational Support Center. Communications equipment to and from the Control Room presently exists in this area. The Emergency Plan will be revised to reflect the existence of this center and to establish the methods and lines of communication and management.