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AUTH.NAME AUTHOR AFFILIATION
UTLEY,E.E. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Forwards response to NRC 791030 request for documentation re method of implementation of Category A requirements.All Category A items implemented.Requests meeting w/NRC to discuss implementation of Category B Section 2.1.6.b.

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Carolina Power & Light Company

December 31, 1979

File: NG-3514(R)

Serial: GD-79-3306

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

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
LESSONS LEARNED SHORT TERM REQUIREMENTS

Dear Mr. Denton:

In accordance with your request of October 30, 1979, Carolina Power & Light Company (CP&L) provides, as an attachment to this letter, documentation of the method of implementation of all Category A requirements for our H. B. Robinson plant. Definition of the implementation category is consistent with Enclosure 2 to your letter. As has been reviewed with your staff, all Category A items have been completely implemented in accordance with the positions and clarifications contained in Enclosure 1 of your letter, as modified by subsequent clarifications by members of your staff in discussions with utility owners' group or CP&L representatives.

Regarding one item, Plant Shielding Review - Section 2.1.6.b, CP&L has completed the review and has found a number of areas requiring action in order to complete the requirements for Category B implementation. Because of the potentially severe impact on plant structures that these actions might impose, if the proper approach is not taken to resolve the items identified in our review, we believe it is important to discuss our results with you as soon as possible after January 1, 1980. The purpose of this discussion is to ensure acceptance of our methodology, our proposed means to reduce radiation exposure in certain plant areas, and the assumptions used in arriving at radiation source terms. Accordingly, we request that your staff review this item as expeditiously as possible. Our staff will be calling soon after the first of the year to establish a meeting date.

We trust the attached information is suitable to meet your request.

Yours very truly,

M A McDuffie

for E. E. Utley
Executive Vice President
Power Supply & Customer Services

JJS/jc
Attachments

8001310287

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Mr. W. B. Kincaid
Mr. S. McManus
Mr. R. B. Starkey, Jr.
Mr. C. W. Woods (LIS)
File: RC/A-2
File: R-2-561

EMERGENCY POWER SUPPLY (2.1.1)

Pressurizer Heaters

POSITION

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

CLARIFICATION

1. In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class 1E division power supply.

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situtation.

   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.

   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

7. Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See item 5.b. above)

CP&L IMPLEMENTATION OF ITEM 2.1.1 - PRESSURIZER HEATERS

CP&L Response To NRC Position No. 1:

At H. B. Robinson Unit No. 2, the pressurizer heaters are normally powered from the offsite power source (grid) via the unit auxiliary and startup transformers. Following a loss of the grid, three banks (150 KW) of pressurizer heaters may be manually transferred to the emergency busses for a source of emergency power to aid in establishing and maintaining natural circulation at the hot shutdown condition.

The number of heaters transferred to the emergency bus is based on a Westinghouse study which indicated that for a 1300 cubic foot pressurizer, a minimum of 125 KW of pressurizer heaters is required to maintain natural circulation in hot shutdown. Since the pressurizer heaters are composed of 50 KW banks, 3 banks (150 KW) are required to maintain the natural circulation condition.

To ensure a redundant emergency power supply capability exists, the emergency instruction which effects the shift of heaters to an emergency power supply address two independent banks of heaters: Backup Group "A" to be powered from emergency bus E1 and the Control Group which is to be powered from emergency bus E2.

CP&L Response To NRC Position No. 2:

Procedures & training which make the operator aware of when and how the pressurizer heaters are to be connected to the emergency bus exist. The existing procedures and training do not address the shedding of loads from the emergency busses as the diesel generators possess sufficient capability to power 150 KW of pressurizer heaters in addition to the normal load expected during a site blackout. Should a safety injection actuation signal occur, the pressurizer heaters will automatically be tripped, thus preventing an overload of the diesel generators following the starting of the engineered safeguards system.

CP&L Response To NRC Position No. 3:

The emergency procedure for loss of offsite power requires that 3 banks of pressurizer heaters be energized by an emergency source of power within 60 minutes of a loss of normal offsite power. The 60-minute time limit is based on a Westinghouse study which indicates that for a 1300 cubic foot pressurizer, power must be restored to the pressurizer heaters within one hour if natural circulation is to be maintained.

CP&L Response To NRC Position No. 4:

The pressurizer heater motive and control power circuits are powered from the nonemergency 480 volt busses. A one-line diagram of the 480 volt electrical distribution systems is shown in Figure 8.2-5 of the H. B. Robinson FSAR. Following a loss of offsite power, the pressurizer Backup Group "A" heaters or Control Group heaters will receive their respective control and motive power through circuit breakers 18B or 28B, each of which have been qualified in accordance with safety grade requirements.

CP&L Response To NRC Clarification No. 1:

The method which H. B. Robinson Unit No. 2 uses to provide emergency power to the pressurizer heaters provides the redundant capability without compromising independence between emergency busses. Three banks of pressurizer heater Backup Group "A" can be powered from emergency bus 1. Should the aforementioned heaters or bus be unavailable, three banks of pressurizer control group heaters can be powered from emergency bus 2.

CP&L Response To NRC Clarification No. 2:

This item has been discussed with NRC Position No. 1.

CP&L Response To NRC Clarification No. 3:

The emergency diesel generators do not have the capacity required to power the pressurizer heaters and all engineered safeguard systems. Therefore, the pressurizer heaters will be automatically shed from the emergency busses should a safety injection actuation signal occur.

CP&L Response To NRC Clarification No. 4:

Following the loss of offsite power, all breakers which supply power to the nonemergency 480 volt busses will open on undervoltage. The pressurizer heaters are transferred to the emergency busses as a source of power by shutting the bus tie breakers between the emergency busses and the nonemergency 480 volt busses. This evolution is performed manually in the Control Room.

CP&L Response To NRC Clarification No. 5:

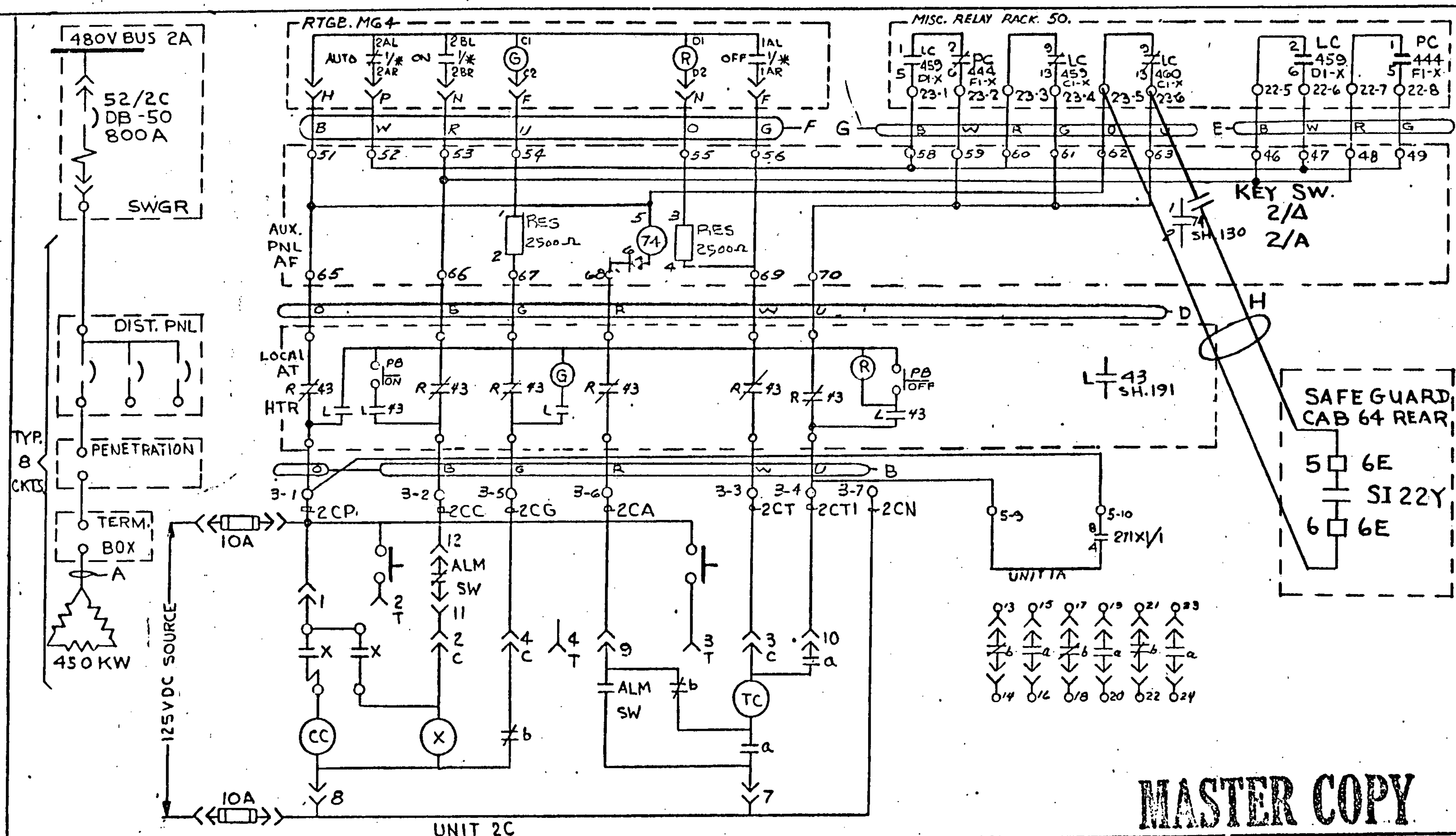
The procedure for manually reloading the pressurizer heaters onto the emergency power sources requires that the safety injection actuation signal be reset prior to energizing the pressurizer heaters. In addition, this procedure describes the instrumentation and criteria which the operator is to use to prevent overloading the diesel generators. Following the reset of the safety injection actuation signal and the stopping of the safety injection pumps, both diesel generators are unloaded sufficiently to support the 150 KW of pressurizer heaters.

CP&L Response To NRC Clarification No. 6:

This item has been discussed with NRC Position No. 4.

CP&L Response To NRC Clarification No. 7:

A modification which trips the non class 1E pressurizer heaters from the emergency power source following a safety injection actuation signal has been implemented at H. B. Robinson Unit No. 2. This modification installed a key switch and a set of safety injection action signal contacts in the pressurizer heater trip circuitry. A simplified diagram of this circuitry is shown in the attached figure. The key switch is closed by an operator as a step in providing emergency power to the heaters. Once the switch is closed, the heaters will trip automatically on a safety injection actuation signal as required.



NOTE
2/Δ SEE SWITCH DEV No 5 DET A SH (37)

NOTE
1/✱ SEE SWITCH DEV. No 17 SH(37) DET. B. ✱= PHBA.

SUB	1	2	3	4	5	6	7	8	9
	S.O. CPL-380	[illegible]	[illegible]	ECN 4737	ECN 4E41	[illegible]	E.C.N. 5241	[illegible]	SYSTEM REVIEW MOD #

Westinghouse Electric Corporation

TITLE: CAROLINA POWER & LIGHT CO.

H.B. ROBINSON STEAM PLANT-UNIT NO. 2

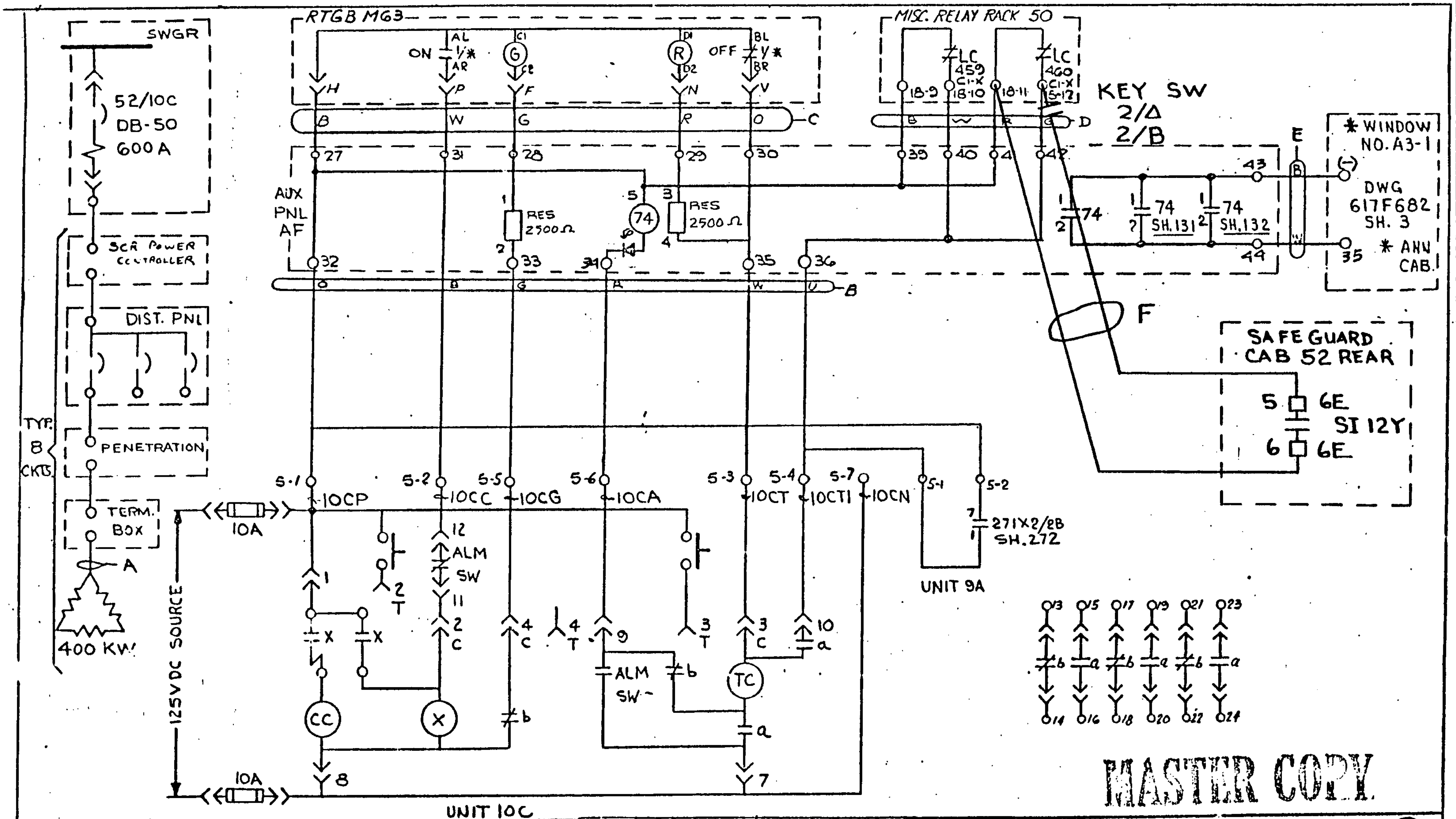
ELEMENTARY WIRING DIAG — PRESS. HEATER BACK-UP GROUP A

Shearer 5/9/68

500B452

SHEET 131

ATOMIC POWER DIV., PITTSBURGH, PA., U.S.A.



Westinghouse Electric Corporation	
TITLE: CAROLINA POWER & LIGHT CO.	
H.B. ROBINSON STEAM PLANT - UNIT NO. 2	
ELEMENTARY WIRING DIAG	PRESS. HEATER CONT. GROUP
Shearer 5/9/68	500B452
JMKurr 7/4/68	SHEET- 130
ATOMIC POWER DIV.	PITTSBURGH, PA., U.S.A.

S.O. CPL-380

SUB 1 2 3 4 5 6 7 8

SYSTEM REVIEW MOD# MSW 3-17-75

NOTE.
2/Δ SEE SWITCH DEV No 5
DET A SH (37)

NOTE.
Y✱ SEE SWITCH DEV. No 3.
DET. A. (SH.37) ✱= PHC.

Emergency Power Supply (2.1.1)

Pressurizer Level and Relief Block Valves

POSITION

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

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

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being suppied from either the offsite power source or the emergency power source when offsite power is not available.

CLARIFICATION

1. While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirement should be capable of being manually connected to the emergency power sources.

CP&L IMPLEMENTATION OF ITEM 2.1.1 - PRESSURIZER LEVEL AND RELIEF BLOCK VALVES

CP&L Response To NRC Position No. 1:

Control power for the power operator relief valves is supplied from the 125 VDC busses. These busses are normally powered by the station batteries which are charged by the emergency busses.

The motive power for the power operated relief valves is supply by the instrument air system which is powered from Emergency Busses E1 and E2.

CP&L Response To NRC Position No. 2:

The motive and control power for the PORV blocking valves is supplied by MCC6. As such these valves are capable of being supplied control and motive power from offsite or the emergency diesel generators.

CP&L Response To NRC Position No. 3:

The motive and control power connections to the emergency busses are through devices which have been qualified in accordance with safety grade requirements.

CP&L Response To NRC Position No. 4:

The pressurizer level indication instrument channels are powered from the instrument busses and as such can be powered from on or off-site sources of power.

CP&L Response To NRC Clarification No. 1:

The prevalent consideration with respect to the PORV/Blocking Valves is to close the valves and shut off the relief path when conditions allow. This is reinforced in the system design by use of a fail safe PORV in series with a motor operated blocking valve. Should the PORV fail to shut, the condition would be recognized by a valve position indicator and the motor operated blocking valve could be shut thus eliminating the relief path.

Although the system has been designed to close the PORV/Blocking Valves, the ability to open these valves is maintained through use of two separate relief paths with their associated valves powered from diverse emergency power supplies.

CP&L Response To NRC Clarification No. 2:

This item has been discussed in NRC Position 1 & 2.

CP&L Response To NRC Clarification No. 3:

The changeover of PORV and blocking valve motive and control power from offsite power to emergency power is accomplished automatically as each of these valves are powered from an emergency bus. The blocking valves, powered from MCC6, will temporarily lose power on a site blackout, but will be reenergized as the diesel generator is automatically started and connected to emergency bus E2 restoring power to MCC6. The PORV's will see an uninterrupted source of power as they are powered from 125 VDC busses which are constantly backed up by the station batteries.

CP&L Response To NRC Clarification No. 4:

The PORV's are air operated relief valves. Should the supply of air to these valves be removed, the valves will fail shut as designed. To maximize the ability to open these valves as discussed in Clarification No. 1, the instrument air compressors which supply air to these valves are powered from Emergency Busses E1 and E2.

PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES (2.1.2)

POSITION

Pressurized Water Reactor and Boiling Water Reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

CLARIFICATION

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70.

2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.

3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.

4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.

5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

6. A description of the test program and the schedule for testing should be submitted by January 1, 1980.

7. Testing shall be complete by July 1, 1981.

CP&L Implementation of Item 2.1.2

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

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

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

DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3.a)

POSITION

Reactor System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

CLARIFICATION

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis and action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualificiation.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification program for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environmental qualification program should be provided.

CP&L IMPLEMENTATION OF ITEM 2.1.3.a

PRESSURIZER POWER OPERATED RELIEF VALVE

The pressurizer power operated relief valve (PORV) position indication system is a direct position indication system which employs limit switches actuated by the valve stem as a sensor for valve position. This system is part of the original design of the H. B. Robinson Plant and does not qualify as "safety grade" in the strictest sense of current standards; however, it does satisfy the short term lessons learned requirements as clarified in Mr. Denton's letter of October 30, 1979 in that it is a reliable, single channel, direct indication powered from a vital instrument bus. In addition, this system is backed up by indirect means of determining valve position such as temperature and pressure downstream of the valves. A visual indication of valve position and an audible alarm is provided in the Control Room to alert the operator should the valve open.

SAFETY RELIEF VALVES (SRV)

In its asbuilt condition, the H. B. Robinson Plant relied on indirect indications of SRV position downstream of the SRV's. The indirect method of determining valve position was used because the SRV's are spring operated relief valves whose body design is not conducive to direct position indication systems such as limit switches. Therefore, an acoustic system which senses flow through the SRV's has been installed.

This newly installed position indication system provides the capability to continuously and automatically detect acoustic signals generated by flow through the valves. These signals are transmitted to an instrument panel mounted in the Control Room where the system compares the current noise level to a quiescent level determined during calibration. When the quiescent level is exceeded by a predetermined amount, visual indication, and audible alarm informs the operator that the valve is open.

This system utilizes individual accelerometers, preamplifiers, and signal conditioners for each SRV. Each valve position monitor is powered from a vital instrument bus which is backed up by battery power thus maximizing the reliability of the power source.

The valve position indication has builtin features that allow at power testing of individual channels. Each channel of the acoustic monitor is provided with a front panel low alarm function which indicates a loss of signal. In addition, test switches on the signal conditioner allow continuity checks of the cable and preamps.

Although this system will ultimately be classified "Safety Grade", it is presently designated "Control Grade" as seismic and environmental testing of the system in accordance with IEEE 344-1974 and IEEE 323-1975 has not been completed. The seismic and environmental testing conducted to date indicates that this system can easily be qualified to the proper standards. The qualification of this system is being coordinated by the vendor, Babcock & Wilcox, and is scheduled for completion by mid 1980.

In the interim, Carolina Power and Light Company has provided a reliable single channel direct indication powered from a vital instrument bus which is backed up by an indirect indication, temperature and pressure downstream of the valves.

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

SUBCOOLING METER

POSITION

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation " (see Section 2.1.9 of NUREG-0578)

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

CLARIFICATION

1. The analysis and procedures addressed in paragraph one above will reviewed and should be submitted to the NRC "Bulletins and Orders Task Force" for review.

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety grade temperature input from each hot leg (or use of multiple core exit in T/C's) are required.

4. Redundant safety grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

6. Each PWR should have: (A.) Safety grade calculational devices and display (minimum of two meters) or (B.) a highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident) which is under development.

8. In all cases appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

9. The attachment provides a definition of information required on the subcooling meter.

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

POSITION

Licensees shall provide a decription of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

CLARIFICATION

1. Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements to or a synthesis of existing measurements which meet safety-grade criteria.

2. The evaluation is to include reactor water level indication.

3. A commitment to provide the necessary analysis and to study advantages of various instruments to monitor water level and core cooling is required in the response to the September 13, 1979 letter.

4. The indication of inadequate core cooling must be unambiguous, in that, it should have the following properties:

   a) it must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high void fraction pumped flow as well as stagnant boil off).

   b) it must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.

5. The indication must give advanced warning of the approach of inadequate core cooling.

6. The indication must cover the full range from normal operation to complete core uncovering. For example, if water level is chosen as the unambiguous indication, then the range of the instrument (or instruments) must cover the full range from normal water level to the bottom of the core .

CP&L IMPLEMENTATION OF ITEM 2.1.3.b

I. PROCEDURES AND DESCRIPTION OF EXISTING INSTRUMENTATION:

The Westinghouse Owners' Group, of which the Carolina Power and Light Company is a member, has performed analyses as required by Item 2.1.9 to study the effects 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 H. B. Robinson S.E.G. Plant has incorporated the key considerations of this instruction into our procedure concerning incidents involving reactor coolant system depressurization and have provided training to the operators in this area. The key considerations which were incorporated include: indications which are available to identify inadequate core cooling (subcooling monitor, incore thermocouples, abnormal steam generator pressure, core differential temperature, and source or intermediate range nuclear instrumentation), methods to restore core cooling, and methods to eliminate a non-condensible gas bubble from the reactor vessel.

II. SUBCOOLING MONITOR:

A core subcooling monitor which provides a continuous online indication of the primary coolant saturation condition has been installed at H. B. Robinson Unit No. 2. The heart of this system is a microprocessor which receives inputs from sixteen core exit thermocouples, eight loop RTD's and four primary system pressure transmitters and outputs a margin to saturation in psi subcooling or degrees F superheat. For a subcooled condition, the margin to saturation in psi is the auctioneered lowest system pressure minus the saturation pressure. In this case, the saturation pressure is determined from either the highest RTD (Thorte) if the selector switch is in the RTD position or the highest thermocouple if the selector switch is in the T/C position. In a superheated condition, the margin to saturation in ^{o}F is the difference between the highest temperature input (RTD or thermocouple depending on whether RTD or T/C is selected) and the saturation temperature based on the lowest pressure input.

The subcooling monitor possesses the required redundancy in that it is comprised of two channels which operate completely independent of each other. Each channel is powered from a vital instrument bus which receives its power from offsite or the emergency diesels. In addition, the warning lights, alarms, meter movements and associated electronics are testable through use of front panel test switches.

The core subcooling monitor makes use of redundant control grade temperature inputs from each hot and cold leg RTD. Since these RTD's are control grade, they are backed up by multiple core exit thermocouples. In addition, each channel is provided with three pressure signals , one narrow range safety grade pressure and two wide range control grade pressures. All safety grade sensors are isolated from the subcooling monitor by isolation amplifiers.

II. Continued

Although this system does not currently meet the safety grade criteria, it is a highly reliable, redundant channel, testable system which is backed up by procedures for the use of steam tables. The use of steam tables are described in the Emergency Instructions for a reactor trip and reactor coolant system depressurization. Following the issuance of Regulatory Guide 1.97, this system will be qualified to the appropriate environmental standards.

The required list of information required on the subcooling monitor is attached.

INFORMATION REQUIRED ON SUBCOOLING METER

Display

Information Displayed	below T-sat: Psat - P
	above T-sat: T - Tsat
Display Type	control board: analog - continuous
	system panel: digital - on demand
Location of Display	rack in Control Room
Alarms	RTD level 1 alarm: 25°F below Tsat
	level 2 alarm: 0°F (Tsat)
	T/C level 1 alarm: 15°F below Tsat
	level 2 alarm: 0°F (Tsat)
Overall Uncertainty	RTD digital alarm: ± 17 psi, ± 1.7F
	T/C digital alarm: ± 21 psi, ± 2.1F
	RTD analog alarm: 44 psi
	T/C analog alarm: 45 psi
Range of Display	1000 psi subcooled to 2000°F superheated
Qualifications	none

Calculator

Type	redundant dedicated digital
Selection Logic	highest T, lowest pressure
Qualification	none
Calculation Technique	functional fit and steam tables

Input

Temperature	2 - hot leg RTD's
	2 - cold leg RTD's
	8 - incore T/C's
Range of Sensors	RTD: 0 - 700°F
	T/C: 0 - 2500°F
Uncertainty of Sensors	RTD: 3.5°F
Qualifications	environmental
Pressure	1 - narrow range transmitter (pressurizer pressure)
	2 - wide range transmitter (overpressure protection)
Range of Sensors	wide range: 0 - 3000 psig
	narrow range: 700 - 2500 psig
Uncertainty of Sensors	wide range: 15 psig
	narrow range: 9 psig
Qualifications	wide range: environmental
	narrow range: IEEE-279

Backup Capability

Availability of Temp. & Pressure	100%
Availability of Steam Tables	100%
Training of Operators	review of applicable procedures
Procedures	EI-1 - Incident Involving RCS Depressurization
	EI-14 -Reactor Trip Part A

III. Additional Instrumentation to Indicate Inadequate Core Cooling

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

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

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

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

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

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

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

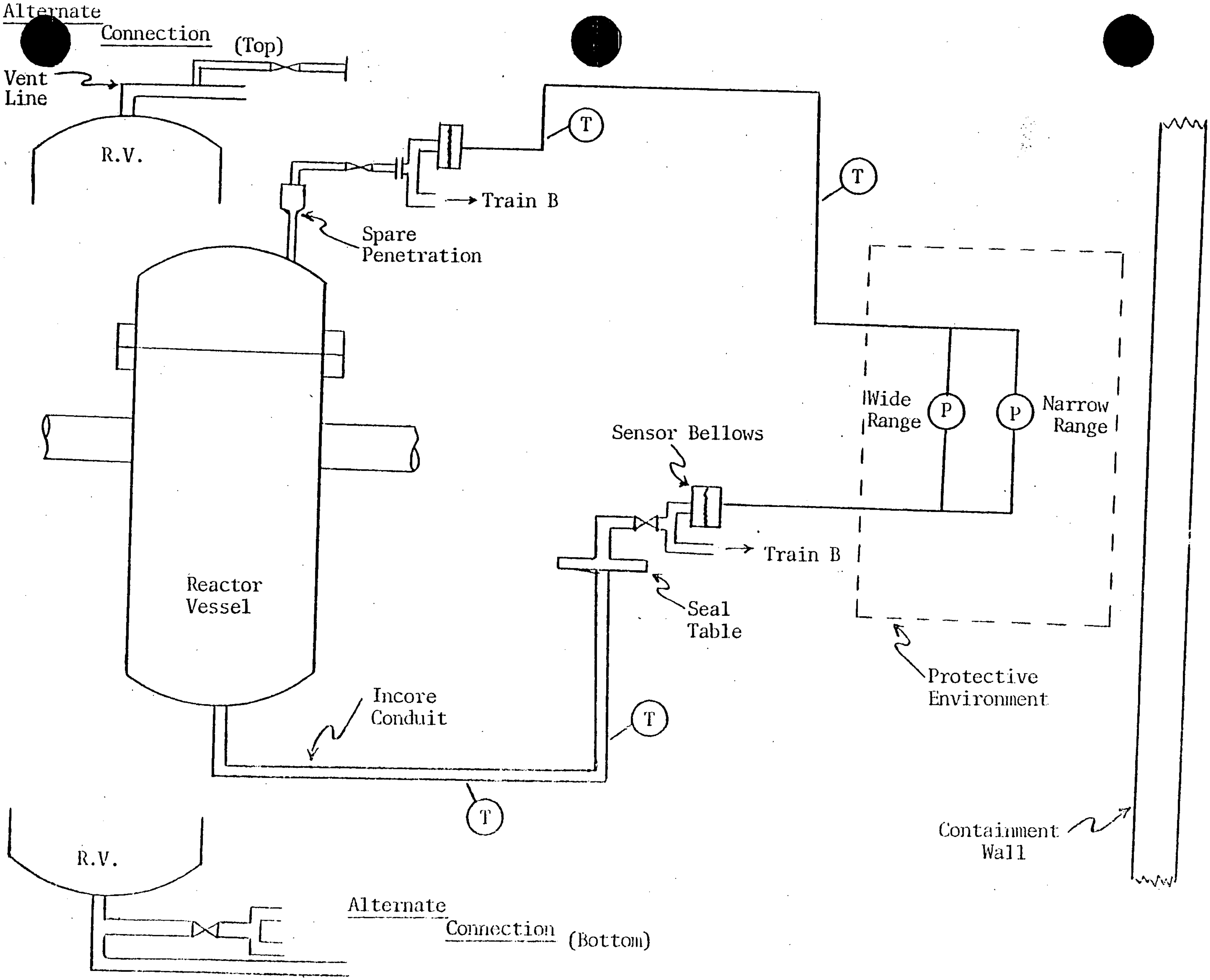
Reactor vessel water level determination is the most promising of the items discussed to provide additional capability of determining the approach to and the existence of inadequate core cooling. Several systems for determining water level are under review by the Westinghouse Owners' Group. A conceptual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

Vessel Level System Description

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

The attached figure shows a simplified sketch of the proposed vessel level instrumentation system. The bottom tap of the instrument would use a thimble of the incore moveable detector system either at the seal table or in the thimble below the vessel. Use of the thimble as part of the incore flux monitoring would not be lost. The flux thimble guide tube would be tapped below the vessel and an instrument line connection made. The instrument line would have an isolation valve and slope down to a hydraulic coupler connected to a sealed reference leg. For connection at the seal table, a special fitting would be utilized which would be connected to an isolation valve and sealed referenced leg. The sealed reference leg would go to the differential pressure transmitter located at a higher elevation above the expected level of containment flooding. A similar sealed leg would go to the top of the vessel and penetrate the head using the vent line or a special connection on a spare RCC mechanism penetration. Two trains of vessel level instrumentation would be provided. The behavior of the signal generated by this level instrument under normal and accident conditions is being evaluated. The usefulness of this instrument to provide an unambiguous indication of inadequate core cooling is being evaluated as part of Item 2.1.9. The potential errors and accuracy of a final system configuration are being evaluated to assess its usefulness to provide information to the operator for proper operation of a vessel venting system and for normal water level control during periods when the primary system is open and a water level may exist in the vessel. The connection of the level system to the vessel head should be designed to be compatible with the head vent system. Operation of the vent system should not upset all indications of vessel level. This can easily be avoided by using a separate instrument tap or by using more than one location.

Carolina Power and Light Company is currently seeking proposals on systems which will provide additional information on inadequate core cooling. Although the subcooling meter provides valuable information as the primary system approaches a saturated condition, additional instrumentation is required to properly evaluate the magnitude of the voided condition. To date, the major NSSS suppliers have not yet made available to the utilities a system which meets the NRC requirements. This is due in part to the NRC's additional requirements issued as clarifications in H. R. Denton's letter of October 30, 1979. This letter required that if a reactor vessel water level detector were used, it must cover the full range of vessel level. This requirement invalidated the currently available instrumentation. A system which meets these requirements will be proposed by Westinghouse in early January, 1980.



Reactor Vessel Level Instrumentation

CONTAINMENT ISOLATION (2.1.4)

POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. 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.

CLARIFICATION

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Non-essential systems should be automatically isolated by containment isolation signals.

4. Resetting of containment isolation signals shall not result in the automatic loss of containment isolation

CP&L IMPLEMENTATION OF ITEM 2.1.4

CP&L Response To NRC Position 1:

The present CP&L H. B. Robinson Plant logic for generation of Phase "A" containment isolation "T" signal does provide for "diversity in the parameters sensed". Any time a safety injection "S" signal is initiated, Phase "A" containment isolation is also initiated. The parameters sensed which provide these signals are steamline pressure, pressurizer pressure, or containment pressure. The "T" & "S" signals are generated by: low pressurizer level (1/3); high differential steam line pressure; high steam line flow in 2/3 steam generator outlet lines coincident with low tave in 2/3 loops or low pressure in 2/3 steam generator outlet lines; or higher containment pressure (2/3). Additionally, these may also be initiated manually by the plant operator from the main control board.

In addition to this diversity, the hardware design also provides redundancy such that these signals will be generated even with a single failure of an individual instrument channel or a complete logic train. Phase "A" containment isolation automatically closes all remotely operable containment isolation valves in lines penetrating containment, except those lines required to mitigate an accident, or to provide component cooling water to reactor coolant pumps.

CP&L Response To NRC Position 2:

This task was performed by first defining terms (owners group publication "Classification of Lines Penetrating Containment..."), then performing a review of the containment penetrations listed in the FSAR assigning the type of system to each of the penetrations.

The definition of these terms is:

1. Essential: Lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event.

2. NonEssential: Lines which are not required to mitigate or limit an accident, and which if required at all would be required for long term recovery only; i.e., days or weeks following an accident.

Table 1 is a listing of containment penetrations derived from H. B. Robinson's FSAR tables. The classification of penetration column identifies each mechanical penetration as Essential (E) or NonEssential (NE) based on the previous definition of terms.

The H. B. Robinson's Emergency Operating Instructions E.I.1 & flow diagrams were utilized as a basis for determining lines used to recover from an accident.

Criteria for terminating safety injection and restoring desirable systems to operation is found in these emergency instructions. In conjunction with the H. B. Robinson Emergency Procedure review, a review was performed against the Westinghouse Owners Group publication titled "Classification of Lines Penetrating Containment and a Review of Containment Isolation Logic and Philosophy".

CP&L Response To NRC Position 3:

Review of Table 1 has shown lines classified as NonEssential are automatically isolated, or are normally isolated by administrately controlled valves, with the exception of the charging path and the N2 supplies to the pressurizer relief tank and the reactor coolant drain tank. If the charging pump continues to run, flow will continue to the RCS via this path, as well as to the seal injection path. In the event that a charging pump is not running containment isolation would be initially effected by a double check valve inside containment in each seal injection line and a single check valve in the charging line, followed by operator action to close manual valves. The N2 supply lines to the pressurizer relief tank and the reactor coolant drain tank are manually isolated as a specific step in EI-1. In the interim between the initiation of a phase "A" containment isolation signal and the manual shutting of the N2 isolation valves, containment isolation is guaranteed by a single check valve in each N2 supply line.

CP&L Response To NRC Position 4:

Remote operated valves in "Essential" containment penetration paths are not designed to receive automatic isolation signals. Remote operated valves in "NonEssential" containment penetration paths are designed to receive automatic isolation signals.

The primary design objective of remote operated containment isolation valves in "NonEssential" containment penetration paths is to close upon receipt of containment isolation signal. To ensure that this design objective is satisfied, the containment isolation signal overrides all other control signals & causes all remotely operable "NonEssential" containment isolation valves to close.

Following reset of the containment isolation signal, the valves remain in the closed position with any automatic opening logic blocked. This includes the Gas Analyzer Sample Line to the Pressurizer Relief Tank and to the Reactor Coolant Drain Tank which have been removed from the Gas Analyzer automatic sampling cycle until a modification blocking their automatic reopening can be implemented. The PRT and RCDT are now sampled manually once a shift.

P = High High Containment Pressure
S = S/I
T = Trip [illegible] Auto S/I

Table 1

H. B. ROBINSON UNIT NO. 2 REACTOR CONTAINMENT BUILDING PIPING PENETRATION

	System Piping Description	Classification Of Penetration	Post LOCA Condition
1.	Pressurizer Relief Tank Gas Analyzer	NE	Closed
2.	Pressurizer Relief Tank N_2 Supply	NE	Closed
3.	Pressurizer Relief Tank Make Up Primary Water	NE	Closed
4.	Primary System Vent Header	NE	Closed
5.	Reactor Coolant Drain Tank Gas Analyzer	NE	Closed
6.	Drain Header - Reactor Coolant Tank	NE	Closed
7.	Main Steam Header	NE	Closed
8.	Main Steam Header	NE	Closed
9.	Main Steam Header	NE	Closed
10.	Main Feedwater Header	NE	Closed
11.	Main Feedwater Header	NE	Closed
12.	Main Feedwater Header	NE	Closed
13.	Steam Generator Blowdown Line	NE	Closed
14.	Steam Generator Blowdown Line	NE	Closed
15.	Steam Generator Blowdown Line	NE	Closed
16.	Residual Heat Removal Loop Out	E	Open
17.	Residual Heat Removal Loop In	E	"S" Open

P = High High Containment Pressure
S = S/I
T = Trip [illegible] Auto S/I

Table 1 (continued)

H. B. ROBINSON UNIT NO. 2 REACTOR CONTAINMENT BUILDING PIPING PENETRATION

System Piping Description	Classification Of Penetration	Post LOCA Condition
18. Reactor Coolant Pump Cooling Water In	E	Close on P
19. Reactor Coolant Pump Cooling Water Out	E	Close on P
20. Reactor Coolant Pump Cooling Water Out	E	Close on P
21. Excess Letdown Heat Exchanger Cooling Water In	NE	Closes on "T" SI
22. Excess Letdown Heat Exchanger Cooling Water Out	NE	Closes on "T" SI
23. Letdown Line	NE	"T" Closed
24. Charging Line	E*	Open
25. Reactor Coolant Pump Seal Water Supply Line (Loop 3)	E	Open
26. Reactor Coolant Pump Seal Water Supply Line (Loop 2)	E	Open
27. Reactor Coolant Pump Seal Water Supply Line (Loop 1)	E	Open
28. Reactor Coolant Pump Seal Water Return Line	NE	Closed on "P"
29. Reactor Coolant System Sample Line (Pressurizer Steam Sample)	NE	Closed
30. Reactor Coolant System Sample Line (Pressurizer Liquid Sample)	NE	Closed
31. Reactor Coolant System Sample Line (Loops 2 & 3)	NE	Closed
32. Fuel Transfer Tube	NE	Closed
33. Instrument Air Header	NE	Closed
34. N_2 Supply to H_2 Vent System Valves	NE	Closed
35. Containment Air Sample In	NE	Closed
36. Containment Air Sample Out	NE	Closed

P = High [illegible]h Containment Pressure
S = S/I
T = Trip on Auto S/I

Table 1 (continued)

H. B. ROBINSON UNIT NO. 2 REACTOR CONTAINMENT BUILDING PIPING PENETRATION

	System Piping Description	Classification Of Penetration	Post LOCA Condition
37.	Containment Purge Supply Duct	NE	Closed
38.	Containment Purge Exhaust Duct	NE	Closed
39.	Plant Air Supply Header	NE	Closed
40.	Containment Purge	NE	Closed
41.	Containment Pressure Relief	NE	Closed
42.	Containment Vacuum Relief	NE	Closed
43.	Safety Injection Line	E	"S" Open
44.	Containment Spray Header	E	Open
45.	Containment Spray Header	E	Open
46.	Containment Sump Recirc. Line	NE	Closed
47.	Containment Sump Recirc. Line	NE	Closed
48.	Safety Injection Test Line - High Head	NE	Closed
49.	Ventilation System Cooling Water In	E	Open
50.	Ventilation System Cooling Water In	E	Open
51.	Ventilation System Cooling Water In	E	Open
52.	Ventilation System Cooling Water In	E	Open
53.	Ventilation System Cooling Water Out Fan/Motor	E	Open

P = High High Containment Pressure
S = S/I
T = Trip [illegible] Auto S/I

Table 1 (continued)

H. B. ROBINSON UNIT NO. 2 REACTOR CONTAINMENT BUILDING PIPING PENETRATION

	System Piping Description	Classification Of Penetration	Post LOCA Condition
54.	Ventilation System Cooling Water Out Fan/Motor	E	Open
55.	Ventilation System Cooling Water Out Fan/Motor	E	Open
56.	Ventilation System Cooling Water Out Fan/Motor	E	Open
57.	Emergency Feedwater Header	E	Open
58.	Emergency Feedwater Header	E	Open
59.	Emergency Feedwater Header	E	Open
60.	Accumulator Sample Line	NE	"T" Closed
61.	Pump Discharge - Containment Sump	NE	Closed
62.	Boron Injection - Loop 2 Cold Leg	E	Open
63.	Boron Injection - Loop 1 Cold Leg	E	Open
64.	Boron Injection - Loop 3 Cold Leg	E	Open
65.	N_2 Supply	NE	Closed
66.	Containment Test Channel Pressure	NE	Closed
67.	Containment Test Controlled Leakage	NE	Closed
68.	Containment Pressure Sensing Line	NE	Closed
69.	Containment Pressure Sensing Line	NE	Closed
70.	Containment Pressure Sensing Line	NE	Closed
71.	Penetration Pressure Air Supply	NE	Closed
72.	Pressurizer PT-458 To Dead Weight Cal.	NE	Closed

* Seal cooling is essential
pb-46 (6)

DEDICATED H_2 CONTROL PENETRATIONS (2.1.5.a)

POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that the redundancy and single failure requirements of General Design Criterion 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

CLARIFICATION

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

CP&L Implementation of Item 2.1.5.a

H. B. Robinson Unit No. 2 uses a post-accident containment venting system which permits controlled venting of the containment atmosphere to maintain the hydrogen concentration at a safe level.

The system consists basically of two full-capacity supply lines through which hydrogen-free air can be admitted to the containment, two full-capacity exhaust lines through which hydrogen-bearing gases may be vented from the containment and associated valving and instrumentation. During accident conditions, these lines are dedicated to purge service only. Drawing HBR-2-6933, enclosed with this report, is an engineering flow diagram depicting the Post-Accident Containment Venting System.

Piping and valving in the supply lines and in the exhaust lines are seismic Class 1 starting inside the containment and proceeding up to and including the isolation valve outside the containment. Exhaust piping is adequately sized to handle flow rates from 100 scfm to 500 scfm with a design flow rate of 240 scfm being used from either of two independent supply systems. The system has been designed to meet the redundancy and single failure criteria of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50.

The above description supports the conclusion that the existing system fully complies with the requirements stated from Item 2.1.5.a in NUREG-0478, including the clarification provided in Mr. H. R. Denton's letter of October 30, 1979. Therefore, no system modifications are planned or required.

CAPABILITY TO INSTALL HYDROGEN RECOMBINER

AT EACH LIGHT WATER NUCLEAR POWER PLANT (2.1.5.c)

POSITION

The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering sheilding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

CLARIFICATION

1. This requirement applies only to those plants that included Hydrogen Recombiners as a design basis for licensing.

2. The shielding and associated personnel exposure limitations associated with recombiner use should be evaluated as part of licensee response to requirement 2.1.6.B, "Design review for Plant Shielding."

3. Each licensee should review and upgrade, as necessary, those criteria and procedures dealing with recombiner use. Action taken on this requirement should be submitted by January 1, 1980.

CP&L Implementation of Item 2.1.5.c

This item is not applicable to the H. B. Robinson Plant. A hydrogen recombiner was not included as part of the design basis for the H. B. Robinson Plant.

INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS FOR PWRs AND BWRs (2.1.6.a)

POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

CLARIFICATION

Licensees shall, by January 1, 1980, provide a summary description of their program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. Examplesof such systems are given on page A-26 of NUREG-0578. Other examples include the Reactor Core Isolation Cooling and Reactor Water Clea Cleanup (Letdown function) Systems for BWRs. Include a list of systems which are excluded from this program. Testing of gaseous systems should include helium leak detection or equivalent testing methods. Consider in your program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to you regarding North Anna and Related Incidents dated ctober 17, 1979.

CP&L Implementation of Item 2.1.6.a

A program designed to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident has been implemented by the H. B. Robinson Plant. This program has been initiated in two phases: the first being the Immediate Leak Reduction Program which is complete; the second being the Continuing Leak Reduction Program which has been established.

Both programs were designated to detect leakage to atmosphere from systems which would be used to bring the plant to a safe shutdown following a serious transient or accident. A thorough review of the plant operating procedures and emergency instructions has resulted in the following systems being included in the leak reduction programs:

1. Residual Heat Removal

2. Safety Injection

3. Containment Spray

4. Chemical & Volume Control

    a. Make-up letdown function
    b. CVCS Purification System
    c. CVCS hold-up tanks and gas stripper feed pumps

5. Liquid Waste Disposal

    a. From Reactor Coolant Drain Tank
    b. Sump Tank A & Transfer Pump
    c. Residual Heat Removal Pump Pit Sump Pumps
    d. Waste Holdup Tank

6. Gaseous Waste Disposal

7. Sample System

    a. Pressurizer Steam Space
    b. Pressurizer Liquid Space
    c. Reactor Coolant System Loop 3
    d. Reactor Coolant System Loop 2
    e. Residual Heat Removal Loop
    f. Volume Control Tank
    g. Letdown Line
    h. Letdown Line

8. Post-Accident Containment Ventilation

A list of systems excluded from the program is not provided, but is available onsite for review, along with justification for the systems included and excluded.

Both the immediate leak reduction program and the continuing leak reduction program employ visual inspections of the mechanical joints and seals of the system in test to detect leakage. These inspections are conducted with the system pressurized to normal system pressure using the system fluid as a test medium. The observed leakage is documented as the "as-found" leakage. Should it be possible to reduce the "as-found" leakage, the "as-left" leakage will also be documented. At the completion of the test, the "as-left" leakage is compared to the maximum allowable leakage. Should the "as-left" leakage be less than the maximum allowable leakage, the test is complete. If the "as-left" leakage is greater than the maximum allowable leakage, a trouble report will be issued, leak repairs initiated, and the leak test repeated for that portion of the system which was effected by repairs.

The maximum allowable system leakage is based on industry and manufacturer suggested allowable design base leakage and Section II of In-service Inspection Pump and Valve Test Code 1WV-3426, Analysis of Leak Rates.

The Immediate Leak Reduction Program has served as the basis from which a continuing leak reduction program has been developed. The Immediate Leak Reduction Program has been completed and the "as-left" leakages for each individual system are totalized below. It should be noted that each of these leak totals are less than the maximum allowable for that system. The continuing leak reduction program will be performed as a periodic test with an annual periodicity.

IMMEDIATE LEAK REDUCTION PROGRAM
TEST RESULTS

		As Left
1.	Residual Heat Removal	25 cc/min
2.	Safety Injection	118 cc/min
3.	Containment Spray	10 cc/min
4.	Chemical & Volume Control	No Detectable leakage
5.	Waste Disposal System-Liquid	No Detectable Leakage
6.	Waste Disposal System-Gaseous	No Detectable Leakage
7.	Sample System	12.7 cc/min
8.	Post Accident Containment Vent	No Detectable Leakage

CP&L RESPONSE TO NRC LETTER REGARDING NORTH ANNA INCIDENT

A review of the facilities at H. B. Robinson Unit No. 2 with respect to the North Anna Unit No. 1 incident as described in the Nuclear Regulatory Commission letter dated October 17, 1979, (Reference IE Circular 79-21) was conducted. This review indicated that a release similar to that experienced at North Anna Unit No. 1 is not probable at H. B. Robinson Unit No. 2.

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

POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids, are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

CLARIFICATION

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. In order to assure that personnel can perform necessary post-accident operations in the vital areas, we are providing the following guidance to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. Source Term

   The minimum radioactive source term should be equivalent to the source terms recommended, in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plant 15.6.5. with appropriate decay times based on plant design.

a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and liquids injected by HPCI and LPCI.

b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For gas containing lines connected to the primary system (e.g., BWR steam lines) the concentration of radioactivity shall be determined assuming the activity is contained in the gas space in the primary coolant system.

2. <u>Dose Rate Criteria</u>

The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria

with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. These doses are design objectives and are not to be used to limit access in the event of an accident.

a. Areas Requiring Continuous Occupancy: ≤15mr/hr. These areas will require full time occupancy during the course of the accident. The Control Room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

b. Areas Requiring Infrequent Access: GDC 19. These areas may require access on a regular basis, but not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant Radiochemical/Chemical Analysis Laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples where occupancy may be needed often but not continuously.

CP&L Implementation of Item 2.1.6.b

SUMMARY

In accordance with Section 2.1.6.b of NUREG-0578, a radiation and shielding design review of areas outside the H. B. Robinson Unit 2 containment building that could become highly radioactive following a severe accident has been conducted. The postulated accident used as the basis for calculating

the source terms included a loss-of-coolant accident (LOCA) with an uncovering, subsequent damage, and release of fission products from the core. Radionuclide releases from the core inventory used in calculating the source terms were as suggested by the NRC.

Using these conservatively high calculated source terms, an analysis was made of the liquid and air systems that would be operated following the postulated accident to cool down the reactor and bring the plant to a safe condition. Several systems, including the Safety Injection, Containment Spray, Residual Heat Removal and Sampling systems would become highly radioactive. Areas containing equipment and piping from these systems have been identified. Calculated dose rates in these areas indicate plant operators may not be able to enter some of these areas to perform required post-accident actions without additional shielding and/or other radiation protection measures utilizing time and distance. Areas which do not require additional radiation protection measures, such as the Control Room which must be occupied during and after the accident, were also identified.

Alternate methods for providing required radiation protection, utilizing time, distance and/or shielding to minimize operator exposures during and following the accident, have been identified. These methods are being analyzed in detail with consideration being given to such factors as cost-benefit, material availability and time available in 1980 to implement the chosen methods before January 1, 1981.

SOURCE TERMS USED AND PLANT CONDITIONS USED FOR ANALYSIS

The gamma source strengths (gammas/cc-sec) have been derived from four separate sources which are listed below:

1. The liquid source of the Containment Sump.

2. The liquid source of the Primary Loop.

3. The gaseous source within the Reactor Building.

4. The plateout source on the interior surfaces of the Reactor Building.

The water level in the Containment Sump accumulates until there is a water depth of approximately 1.5 feet on the Containment floor. This water volume is derived from the three accumulators and a portion of the Refueling Water Storage tank, which is about 215,000 gallons, over a period as short as 30 minutes. With an accident condition involving a major break in the primary loop, the injection of this water in the cold legs is followed by the spilling of a mixture of this water and the reactor loop water into the Containment Sump. By the time the injection phase is completed, it is assumed that the water in the Primary Loop and the water in the sump are uniformly mixed. Therefore, the isotopic core releases (fission products) are diluted by the combined volumes of water or 285,300 gallons.

The isotopic core releases are specified in USNRC Regulatory Guide 1.4 and NUREG-0578 as 100 percent of the noble gas inventory, 50 percent of the halogen inventory, and 1 percent of all the other fission products. For conservatism, 50 percent of the rubidium and cesium were also included to account for the decay of the gaseous krypton and xenon, respectively. Data from the Three Mile Island plant have indicated that only about 10 percent of the noble gas inventory remained in the water at some hours after the incident. Therefore, the source strength as used in this analysis are considered to be conservative.

A second accident condition was considered, which differs from the first by considering a small break or leak of the Primary Loop. With this condition the isotopic core releases are diluted only by the water volume of the Primary Loop, which is approximately 70,300 gallons. Thus, the source strengths for this condition are about a factor of four times greater than for condition one.

The locations in the plant where each of these conditions were used for the analysis are indicated in a later section. The actual conditions will fall between these two limiting cases, and judgment will be required to determine the choice of condition when the shielding design is undertaken. The source strengths for condition one are tabulated in Table 1 for 30 minutes, 1 hour, 1 day and 1 month following an incident.

The gaseous source strengths in the Reactor Building are based on the instantaneous release of 100 percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen inventory. These source strengths have been calculated for 5 minutes, 1 hour, 1 day and 1 week following an incident, and are listed in Table 2.

The plateout source strengths are assumed to contain 25 percent of this core equilibrium iodine inventory and the rubidium and cesium daughter products from the decay of the gaseous krypton and xenon, respectively. These source strengths have been calculated for 5 minutes, 1 hour, 1 day, 1 week and 1 month following an incident and are listed in Table 3.

The source strengths as derived above use 200 percent of the noble gas inventories, 100 percent of the halogen inventory and more than 1 percent of all the other fission product inventories. Therefore, the analysis has a very conservative basis.

No consideration of equipment leakage or spillage due to equipment failure has been included in this analysis. Only direct radiation effects from systems containing highly radioactive liquids and gases have been considered.

OPERATIONS AT H. B. ROBINSON, UNIT 2, FOLLOWING A MAJOR ACCIDENT

H. B. Robinson, Unit 2, Emergency Instruction E.I.-1 (Reference c) discusses and contains specific instructions concerning both the short and long term operator actions to be followed after an accident postulated to include a severe:

a. Loss of primary coolant.

b. Loss of secondary coolant.

c. Steam generator tube rupture.

As a result of any such accidents, E.I.-1 assumes rapid depressurization of the primary system with coolant discharged directly into the containment building sumps and floor. Although there are specific cautions and statements in E.I.-1 related to the seriousness of depressurization and subsequent possibility of forming voids and thus compromising core cooling capability, for sake of analysis and conformance with NRC TMI-2 Lessons Learned analysis requirements core damage and release of fission products is also assumed to occur.

E.I.-1 describes all of the automatic actions triggered by a safety injection signal resulting from the accident briefly described above. These begin with a reactor trip, resultant turbine trip, feedwater pump trips, steam and feedwater block and bypass valve closures, emergency diesel start, tripping of non-essential breakers, automatic alignment of numerous normal and safeguard valves and start or shutdown of numerous other pumps and fans. In the event of a high-high containment pressure signal, which might accompany such an accident, spray pumps automatically start also with automatic alighment of various valves.

All of the above automatic operations are observed and verified by the operators in the Control Room. Seciton D.7 of E.I.-1 states that in the event that any of the safeguards equipment has not properly been activated, an operator will place the equipment in operation manually. Most of the automatically activated equipment is in the Reactor Auxiliary Building (RAB) and, if necessary, can be activated manually by the Auxiliary Operator who is normally stationed in Area J-13, as shown on Figure 1, appended.

A detailed recovery procedure following a loss of coolant is appended to E.I.-1. This procedure is used to insure that proper cooling is provided to the core after a safey injection occurs due to a loss of primary coolant. All sections in the injection phase of the Appendix A, E.I.-1 procedure down to and including Section 2.8 are performed or verified by the Control Room Operators. Section 2.9.1 requires Operator or Chemical Technician action in the RAB, namely, sampling the steam generators. This is performed in the Sample Room, Area K-14 of Figure 1.

Additional Operator action is required in the Pipe Alley of the RAB, Area L-11. This is in accordance with Section 2.10 of Appendix A, requiring manual isolation of nitrogen from the Reactor Coolant Drain Tank and the Pressurizer Relief Tank.

In the initial recirculation phase of Appendix A, Operator action in the RAB requires manual closing of a number of breakers at MCC5 per Section 3.3 in Area I-9 on Figure 1. In accordance with Section 3.5 an Operator is then required to enter the RHR Heat Exchanger Room, Area K-11, to open four designated valves. Section 3.6 then requires an Operator to close designated breakers at MCC6, located in Area F-13 on Figure 1.

Section 5.0 of Appendix A describes a long term recirculation flow path. Section 5.3.8 later notes, in relation to alignment of additional RHR system valves, that the radiation levels may be prohibitive in the Pipe Alley and RHR pit for valve alignment.

Section 6.0 of Appendix A, describing operation of the Reactor Coolant System with a non-condensible bubble in the system, does not call for Operator actions in the RAB.

Appendix B of E.I.-1 provides a detailed recovery procedure following a steam line rupture. Operator action is performed from the Control Room. No Operator action in the RAB is specified. Appendix C of E.I.-1 provides a detailed recovery procedure following a steam generator tube rupture. Section 3.1 of Appendix C provides the option of monitoring the ruptured steam generator either by use of blowdown monitor R-19 or by manual sampling, requiring an Operator or Chemical Technician to enter the Sample Room, Area K-14 of the RAB.

Emergency Instruction E.I.-16 (Reference d) provides detailed procedures for pressurizing, venting, and sampling the containment atmosphere for hydrogen generated by a severe loss of coolant accident. Sections 3.2.1 and 3.3.1 require that nitrogen instrument gas bottle pressures be checked (by an Operator) once each 8 hours. The nitrogen gas bottles are located in Area F-9 of the RAB. Manual and remote operation of valves for pressurizing

and venting of the containment, described in Sections 2.4 and 3.4 of E.I.-16, are performed in the vicinity of the Hydrogen Venting Panel located in Area F-8 of Figure 1.

Sampling the containment, described in Section 4.2 of E.I.-16, requires Operator and/or Chemical Technician entries into three areas of the RAB. These are the Pipe Alley (Area L-11) for manually opening or closing containment pressure tap isolation valves, the Gas Analyzer Room (Area K-9) for obtaining grab samples of containment atmosphere, and the R-11 and R-12 containment monitors (Area F-9).

Access paths to all areas of the RAB where Operator action is required, as described above, are shown on Figure 1.

SYSTEMS AFFECTED BY RAPID INCREASE OF RADIOACTIVITY
DUE TO MAJOR ACCIDENT CONDITIONS

The scenario of a postulated loss of coolant accident (LOCA), is described in the Final Safety Analysis Report, Section 6.2.2 and Tables 6.2-4, 6.2-5 and 6.2-6. Assuming that fission products and noble gases will be released from the coolant to the containment sump and into the air inside the containment from the coolant released, systems outside the containment will build up to high radioactive levels in the approximate sequence:

-The Liquid Sampling System, if it is monitoring the reactor hot leg early in the accident sequence in accordance with TMI-2 Lessons Learned Item 2.1.8, may give the first indication of radioactivity entering the RAB. Buildup of radioactivity in these sample lines should be rapid where they leave the containment penetration in the RAB Pipe Alley (Area L-11) and progress down the Alley to the Sample Room (Area K-15).

-The Containment Atmosphere Sampling System, if activated to monitor hydrogen buildup inside the containment, may also rapidly become radioactive as airborne activity is circulated with the gas sample through the penetrations in the RAB Pipe Alley (Area L-11) and into the Gas Analyzer Room (Area K-9).

- The Safety Injection, Core Spray and/or that portion of the RHR systems aligned for core deluge will rapidly become highly radioactive when pump suctions for these systems are transferred from the Refueling Water Storage Tank to the containment sumps. Suction lines to these pumps from the containment enter the Pipe Alley (Area L-11) and proceed northerly to the RHR Pump Area Penetration (Area L-6) and also to the Safety Injection Core Spray Pump Room (between I-3 and L-3). The discharge lines from these systems generally parallel the paths of the return or suction lines.

- The Post Accident Containment Venting System, if used to ventilate the containment at any time during the course of the accident should also become highly radioactive, significantly increasing the dose rates in the vicinity of the Hydrogen Venting System Panel and the R11, R12 Radiation Monitor (Area F-9).

As the accident recovery proceeds from the injection phase to the initial and longer term recirculation phase, valve alignments will be made remotely and manually on the RHR system, recirculating coolant from the containment floor by the RHR pumps through the RHR heat exchangers (Area K-12).

If the Containment Vent Filters are used at any time after the accident, some buildup of radioactivity on the filters can also be expected to contribute to Area F-9. Assuming that leakage from components and systems in the RAB will be minimized due to action required by TMI-2 Lessons Learned Item 2.1.6.a, radioactive buildup in the Auxiliary Building Vent Filters (Area E-10) is expected to be minimal.

ANALYSIS OF DOSE RATES IN AND AROUND PLANT WHERE OPERATOR ACTION IS REQUIRED FOLLOWING A MAJOR ACCIDENT

The dose rate calculations for the H. B. Robinson, Unit No. 2 Shield Design Review were done with the SPAN-4 model. SPAN-4 (Reference e) is a computer program which enables the user to model the sources and shields very

accurately in the 3 spatial dimensions. The calculation technique uses the point-kernel equation, and the dose rate from a volume source is the integration over that volume from many of the point-kernel calculations. This is the standard method for detailed gamma dose rate calculations (without resorting to transport theory or Monte Carlo methods which are very expensive and difficult to use). The dose rates calculated for this study are a result of this model, and there is a high degree of confidence in their accuracy for the specified assumptions. Thus, these calculated values can be used in determining shielding configurations and thicknesses. In addition to the data shown in this report, the dose rates from individual components (pipe segments, portions of tanks, etc.) are known, which will enable local shielding to be effectively employed. There are also some refinements in the model used which can be expanded to reduce shielding requirements on a local basis. The dose rates apply only to the sources from which they were calculated. If two adjacent systems are in operation simultaneously, then the dose rates of common points are additive.

The calculated dose rates are the expected dose rates with the present arrangements and shielding. The dose rates considered are the maximum expected in each area; however, it should be noted that the most likely dose rates are somewhat lower. The liquid source strengths were derived for two distinct conditions following an accident. The first condition is that a major break occurs in the primary loop and the water from the Refueling Water Tank is pumped into the cold legs until it is exhausted. This will result in a dilution of the core isotopic releases (fission products) by the total volume of these two water sources. The second condition is that the primary loop does not have a major break or leakage, but most of the core isotopic releases are contained within the primary loop. This second condition results in isotopic concentrations (for the fission product release from the core) which are four times higher than for condition one. The accident condition used for each of the liquid systems is listed below, and the most likely dose rates lie somewhere between the two conditions. It should also be noted that no credit has been taken for cleanup of the liquid by demineralizers or filters.

Area of Plant	Accident Condition
1. SI and CS Pump Room	1
2. Liquid Sampler Room	2
3. RHR Heat Exchanger Room	2
4. Nonregenerative Heat Exchanger	2
5. Volume Control Tank	2
6. Pipe Alley Area (RAB)	1 & 2

The extent of dilution for each system will depend upon the nature of the accident, time after the accident, operator responses, etc., and judgments of the conditions will need to be made when the sheilding modifications are designed. The gaseous and plateout source strengths are not subject to these differences of dilution.

The Safety Injection (SI) and Containment Spray (CS) pumps could take suction on the containment sump water (accident condition 1) as early as 30 minutes following an incident. The dose rates within the room and adjacent areas, including Diesel Generator Room B, are very large and would prohibit operator entry.

The USNRC requires a liquid sample to be taken in less than one hour. The calculated dose rates indicate that this condition will be very difficult to achieve with the existing hot leg sample system. The particular sources that contribute to these high dose rates are the Sample Heat Exchanger and the trap below the sample sink.

The Residual Heat Removal (RHR) systems may be used in the early hours following an incident. Both of the RHR heat exchangers are assumed to be in operation simultaneously. The Operators station in the corridor has an almost line of sight path with RHR heat exchanger A (approximately 6 inches of concrete corners intervene), which result in very high dose rates in the corridor by direct radiation. In addition, there is a large scatter component along the corridor and into the Hot Chemistry Laboratory. The dose rates through the 2 feet thick concrete shield walls are likewise very large, both on the first floor and second floor.

The Nonregenerative Heat Exchanger and the Volume Control Tank are parts of the Chemical and Volume Control System which may possibly be used in the short term following an incident. These components have been included in this section of the report so that their function may be considered in more detail when the detailed design is undertaken. The dose rates from these components are additive to the other dose rates in the vicinity.

The Pipe Alley in the RAB in the vicinity of the Gas Analyzer room and the RHR heat exchanger room has access requirements for manual operation of some valves. This area has a large number of SI, CS and RHR pipes which contain sump or reactor water. The maximum dose rate estimated for this area precludes entry for manual valve operation.

The sources in the pipe alley also have an effect on the dose rates in the Gas Analyzer room. The magnitude of the dose rates through the shield wall preclude access to the Gas Analyzer room for purposes of obtaining a gas grab sample.

The Containment and Stack Radiation Monitors R11/R12 are located on the seoncd floor of the RAB adjacent to the stack. In addition to the monitors, the control valves for the Hydrogen Purge system and some of its 3 inch piping is also in this area. This area is also a location in which a containment grab sample can be obtained.

The Control Room is separated from the radioactive sources in the RAB by several floor levels, and it is offset an appreciable distance to the south and east of the sources. Thus, the dose rate to the Control Room from the RAB sources are negligible. However, the top of the Control Room is exposed to direct radiation from the upper portion of the Reactor Building. The Reactor Building sources consist of the noble gases and halogen gaseous atmosphere, and the plateout of iodine, rubidium and cesium on the interior surfaces of the Reactor Building. The dose rate in the Control Room from the Reactor Building source is calculated by a conservative model to be 15 mrem/hr at 5 minutes. The dominant portion of the dose rate is from the upper cylindricial portion (above the operating floor). The dose rate drops off rapidly to less than 0.1 mrem/hr at 1 day after the incident.

ALTERNATIVE CORRECTIVE ACTIONS REQUIRING DETAILED EVALUATION TO MINIMIZE EXPOSURES IN VITAL AREAS

Operations performed in vital areas of the Reactor Auxiliary Building (RAB) during the initial recovery operations following a LOCA and the approximate sequence when high levels of radioactivity would enter the RAB via systems activated during the injection and recirculation phases following this LOCA have been discussed above. A qualitative analysis of dose rates in the vital areas in and around the RAB has also been presented. The vital areas requiring corrective action to minimize operator exposure during entrance to these areas have thus been identified and the dose rates calculated.

The radiation level criteria used for guidance in determining the need for radiation protection measures to minimize personnel exposures in vital areas other than the containment are:

Category		Dose Rate Guidance
A	Areas requiring continuous occupancy during all phases of the accident.	15mr/hr
B	Areas requiring possible frequent access to perform specific or vital operations and inspections to assure safe plant shutdown.	100mr/hr
C	High radiation areas where access is required infrequently to perform specific or vital operations essential to safe plant shutdown.	10 CFR 20 limits

Each of these areas will be investigated in detail to determine alternate radiation protection measures, using time, distance and shielding radiation protection methods, that will be designed and implemented by January 1, 1981 to meet the criteria set forth above. Procedures to be employed in reducing radiation exposure to equipment and personnel, such as temporary shielding and flushing of highly radioactive water from secured systems, will also be investigated and developed as necessary.

REFERENCES

(a) TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, U. S. Nuclear Regulatory Commission, July, 1979

(b) Regional Meeting Agenda, TMI Followup, NRC Short Term Implementation Program, Atlanta, Georgia - September 28, 1979

(c) CP&L, H. B. Robinson - Unit 2 Emergency Instruction-1, "Incident Involving Reactor Coolant System Depressurization", Revision 17, October 5, 1979

(d) CP&L, H. B. Robinson - Unit 2 Emergency Instruction EI-16, "Post Accident Containment Venting System", Revision 1, November 30, 1978

(e) O. J. Wallace, "SPAN-4, A Point-Kernel Computer Program for Shielding", WAPD-TM-809(1), Bettis Atomic Power Laboratory, Pittsburgh, Pennsylvania, October, 1972

AUTO INITIATION OF THE AUXILIARY

FEEDWATER SYSTEM (AFWS) (2.1.7.a)

[illegible]SITION

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

CLARIFICATION

Control Grade (Short-Term)

1. Provide automatic/manual initiation of AFWS.

2. Testability of the initiating signals and circuits is requied.

3. Initiating signals and circuits shall be powered from the emergency buses.

4. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not comprimise the emergency diesel generating capacity.

5. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the AFWS from the control room.

6. Other Considerations

   a. For those designs where instrument air is needed for operation, the electric power supply requirement should be capable of being manually connected to emergency power sources.

CP&L IMPLEMENTATION OF ITEM 2.1.7.a

The auxiliary feedwater system consists of two motor driven and one steam driven auxiliary feed pumps. The capacity of any one of the three auxiliary feed pumps is sufficient to meet the decay heat removal requirements of the plant.

1. The auxiliary feedwater system is automatically initiated as a result of:

    a. Safety injection initiation.

    b. Loss of offsite power.

    c. Loss of two main feedwater pumps.

    d. 2/3 steam generator level channels sensing lowlow level (15%) on 1/3 steam generator will automatically start both motor driven AFW pumps and open discharge valves.

    e. 2/3 level channels sensing lowlow level on 2/3 steam generators will automatically start the steam driven AFW pump and open discharge valves.

2. The automatic initiation signals and circuits are designed so that a single failure will not result in the loss of auxiliary system function.

3. Testability of the initiating signals and circuits is possible. Automatic initiation of the auxiliary feedwater system is assured by the Safeguard Protection Logic System and is verified by the Periodic Testing Program. Manual initiation of this system is also assured by the Periodic Testing Program.

4. The initiating signals and circuits are powered from the emergency buses E1 and E2.

5. Manual capability to initiate the auxiliary feedwater system from the Control Room is possible with a single failure in the manual circuits and will not result in the loss of system function.

6. The AC motor driven pumps and valves in the auxiliary feedwater system are automatically sequenced onto the emergency buses following a loss of all power.

7. Manual capability to initiate the AFWS from the Control Room is still possible when failures occur in the automatic initiating signals and circuits.

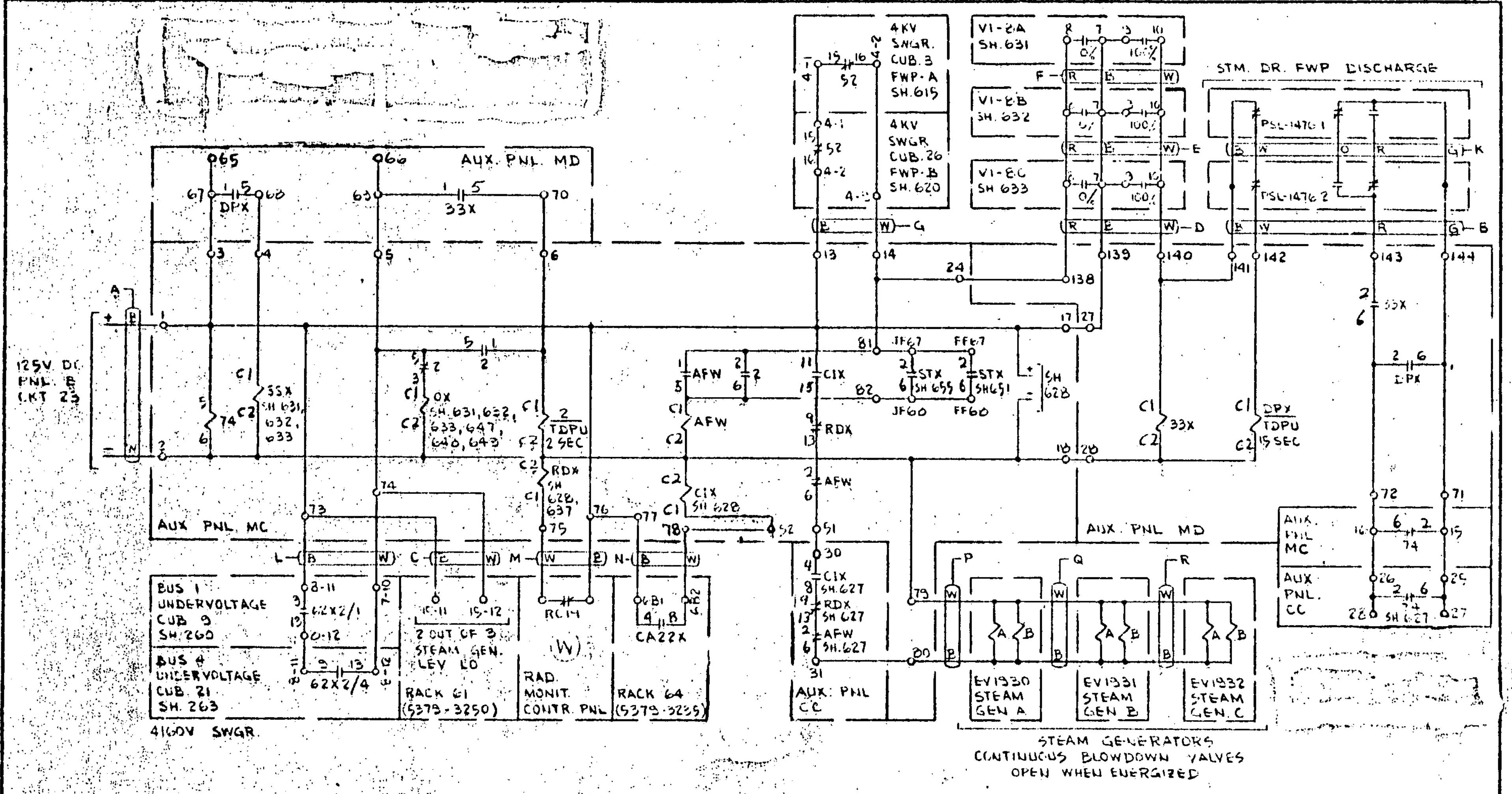
The automatic initiation signals and circuits were installed when the plant was built in accordance with safety grade requirements.

Safeguards actuation drawings 110E198, Sheets 6 & 12 as well as the following control wiring diagrams are included for your information and use:

Drawing No. 500B452:

Sheet No.:	Title
630	Steam Driven Feedwater Pump
631	Steam Driven FWP - Steam Shutoff Valve, V1-8A
632	Steam Driven FWP - Steam Shutoff Valve, V1-8B
633	Steam Driven FWP - Steam Shutoff Valve, V1-8C
642	Feedwater Control & Bypass Valves
647	Steam Driven FWP Discharge Valve, V2-14A
648	Steam Driven FWP Discharge Valve, V2-14B
649	Steam Driven FWP Discharge Valve, V2-14C
651	Auxiliary FWP "A"
655	Auxiliary FWP "B"
660	Auxiliary FWP - Section Valve, V2-20A
661	Auxiliary FWP - Section Valve, V2-20B
662	Auxiliary FWP Header Discharge Valve, V2-16A
663	Auxiliary FWP Header Discharge Valve, V2-16B
664	Auxiliary FWP Header Discharge Valve, V2-16C

These drawings illustrate the actuation of the Auxiliary Feedwater System, relays, contacts of relays, connector pin and terminal designations.



EBASCO DWG B-190628

EBASCO SERVICES INCORPORATED
NEW YORK

STEAM DRIVEN FEEDWATER PUMP

Westinghouse Electric Corporation

TITLE: CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT

CONTROL WIRING DIAGRAM

500B452

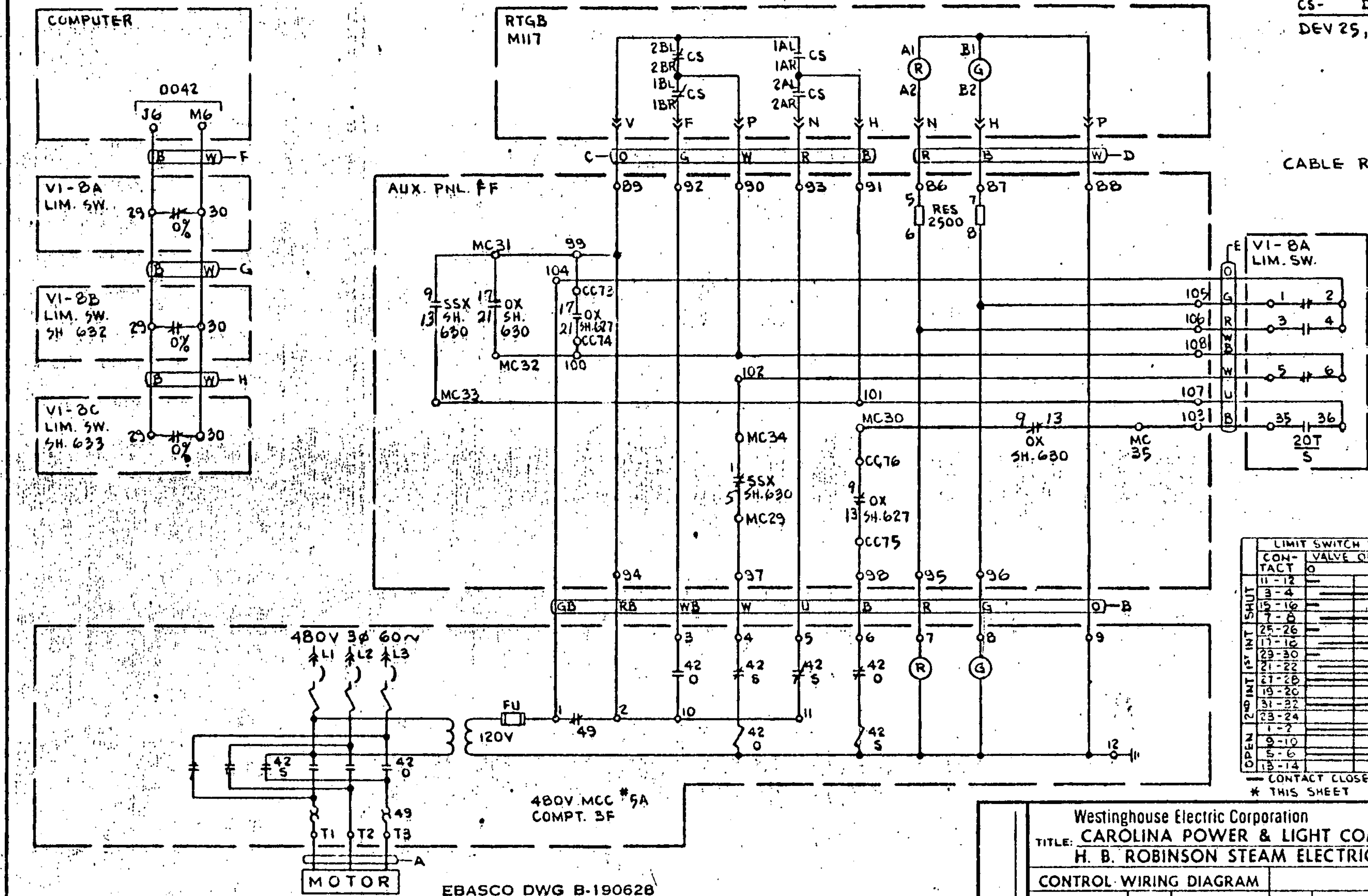
SHEET 630

ATOMIC POWER DIV. PITTSBURGH, PA. U.S.A.

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
8				4	7-19-71		
7				3	2-23-71		
6				2	1-6-70		
5				1	8-15-69		

DATE JUL 11 1969

CS- DEVELOPMENT
DEV 25, DET H., SH. 37



	LIMIT SWITCH DEVELOPMENT		
	CON-TACT	VALVE OPENING % (0 – 100)	CWD SH.
SHUT	11 - 12		635
SHUT	3 - 4		*
SHUT	15 - 16		
SHUT	7 - 8		630
1ST INT	25 - 26		627
1ST INT	17 - 18		
1ST INT	29 - 30		*
1ST INT	21 - 22		
2ND INT	27 - 28		
2ND INT	19 - 20		627
2ND INT	31 - 32		
2ND INT	23 - 24		
OPEN	1 - 2		*
OPEN	9 - 10		630
OPEN	5 - 6		*
OPEN	13 - 14		

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EBASCO DWG B-190628

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
8				4			
7				3	6-23-78	TW	
6				2	3-16-70	GSC	ZB
5				1	1-6-70	GSC	EB

EBASCO SERVICES INCORPORATED
NEW YORK
DIV. ELEC. GR. GSC
SCALE — CH. VB
DATE JUL. 11 1969
APPROVED K Bricknell

STEAM DRIVEN FWP
STEAM SHUT-OFF VA.
VI-8A

Westinghouse Electric Corporation
TITLE: CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM

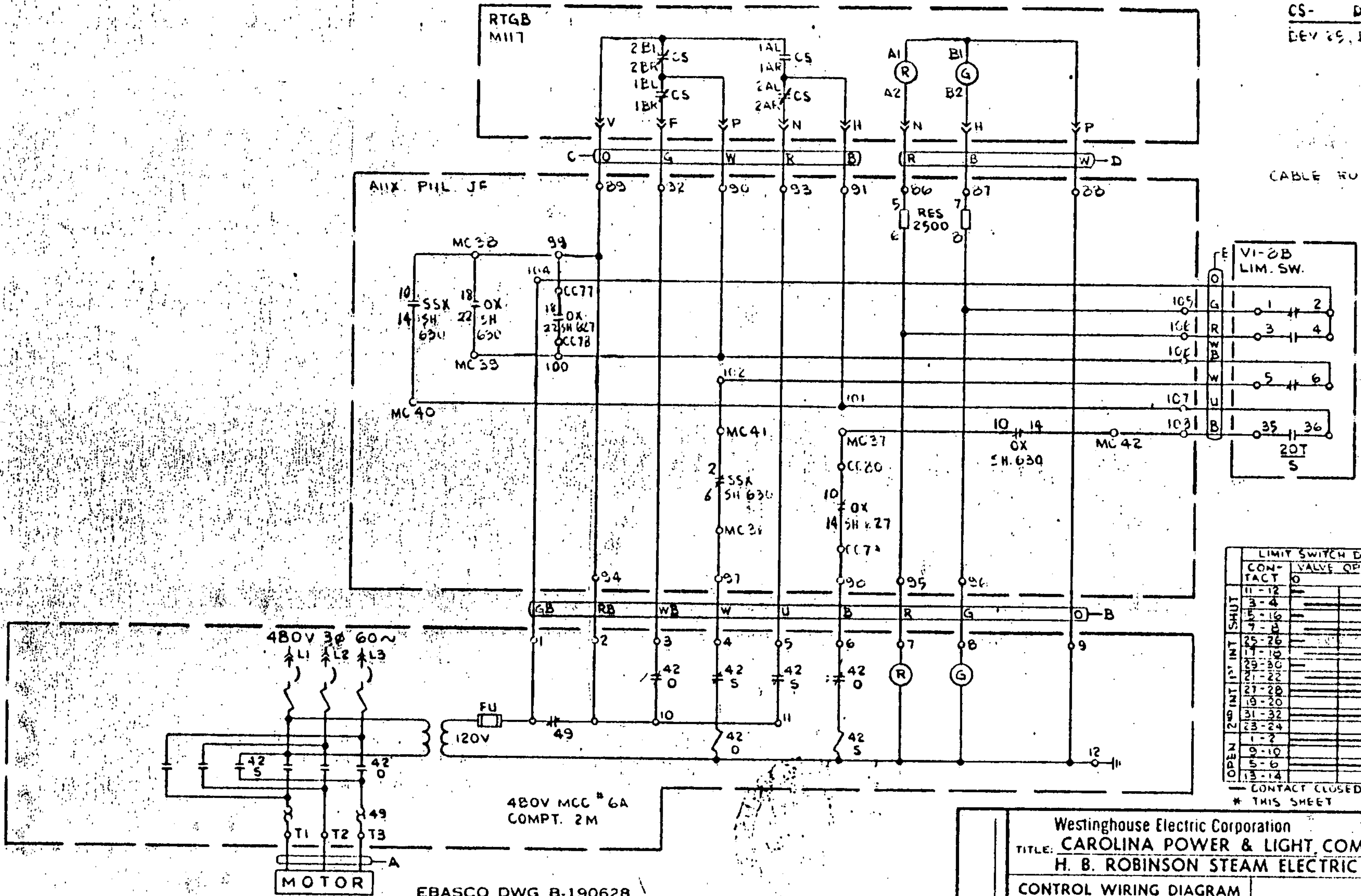
S.O.
SUB 1

500B452
SHEET 631

ATOMIC POWER DIV., PITTSBURGH, PA., U.S.A.

4

CS- DEVELOPMENT
DEV 65, DET II, SH 37



	LIMIT SWITCH DEVELOPMENT		
	CONTACT	VALVE OPENING % (0 – 100)	CWD SH.
SHUT	11-12		[illegible]
SHUT	3-4		*
SHUT	15-16		[illegible]
SHUT	7-8		[illegible]
1ST INT	25-26		[illegible]
1ST INT	17-18		[illegible]
1ST INT	29-30		[illegible]
1ST INT	21-22		
2ND INT	27-28		
2ND INT	19-20		627
2ND INT	31-32		
2ND INT	23-24		
OPEN	1-2		*
OPEN	9-10		[illegible]
OPEN	5-6		*
OPEN	13-14		

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EBASCO SERVICES INCORPORATED
NEW YORK

DIV. ELEC. DR. GSC
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DATE [illegible]
APPROVED K Brickwell

STEAM DRIVEN FWP
STEAM SHUT-OFF VA
VI-8B

Westinghouse Electric Corporation
TITLE: CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM

S.O.
1 SUB

500B452
SHEET 632

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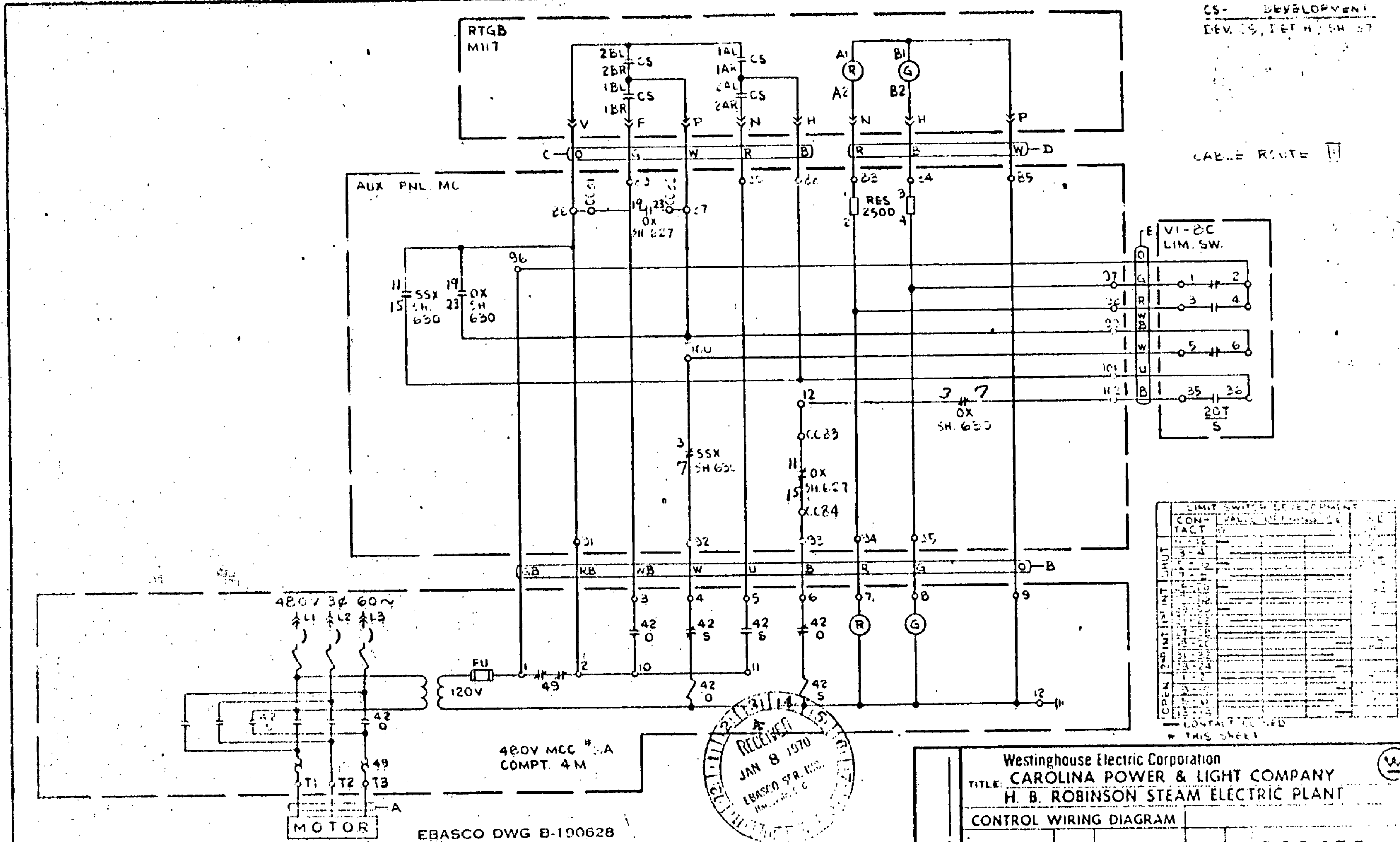
PITTSBURGH, PA., U.S.A.



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CONTROL WIRING DIAGRAM

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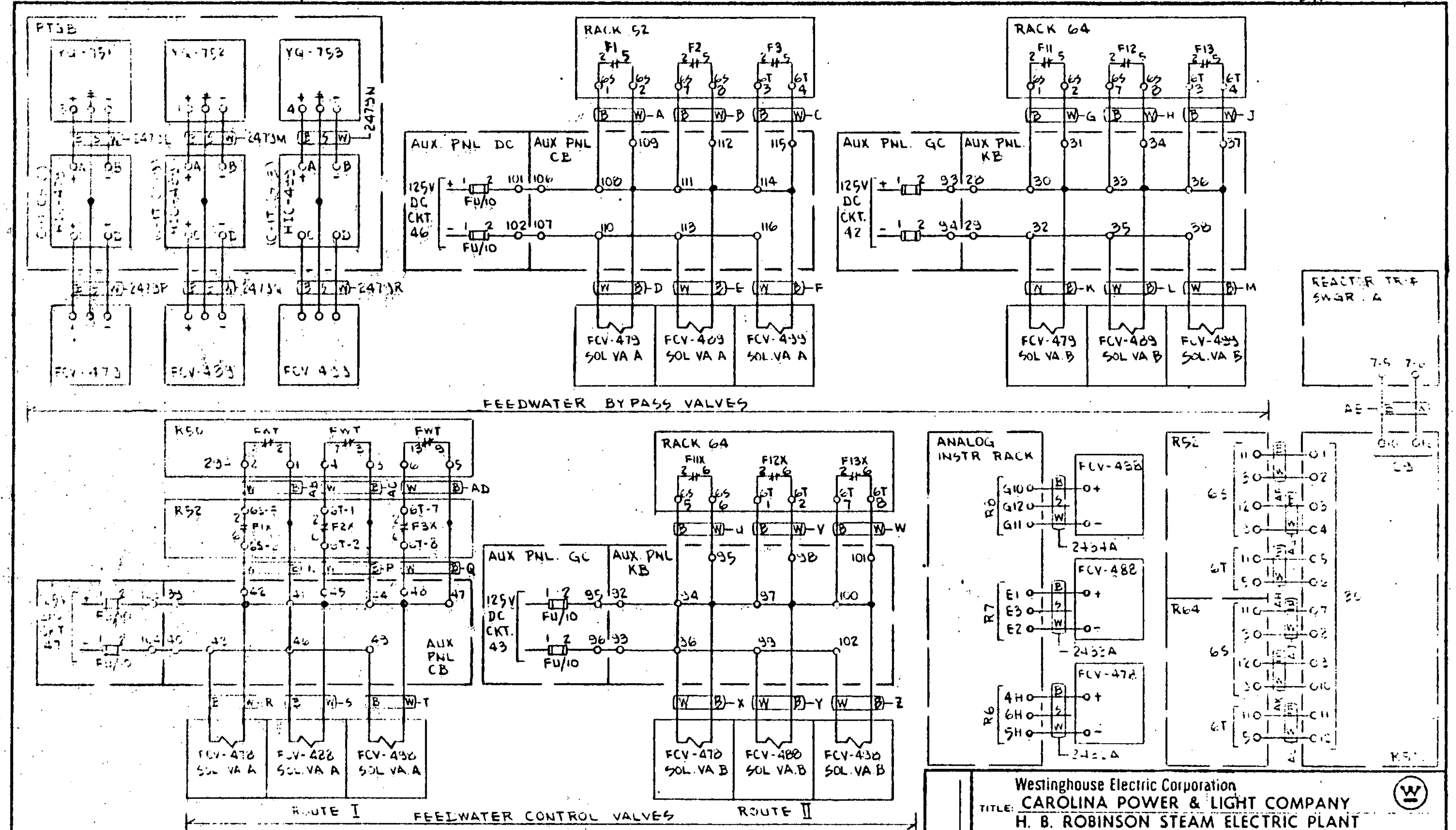

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FEEDWATER CONTROL & BYPASS VALVES

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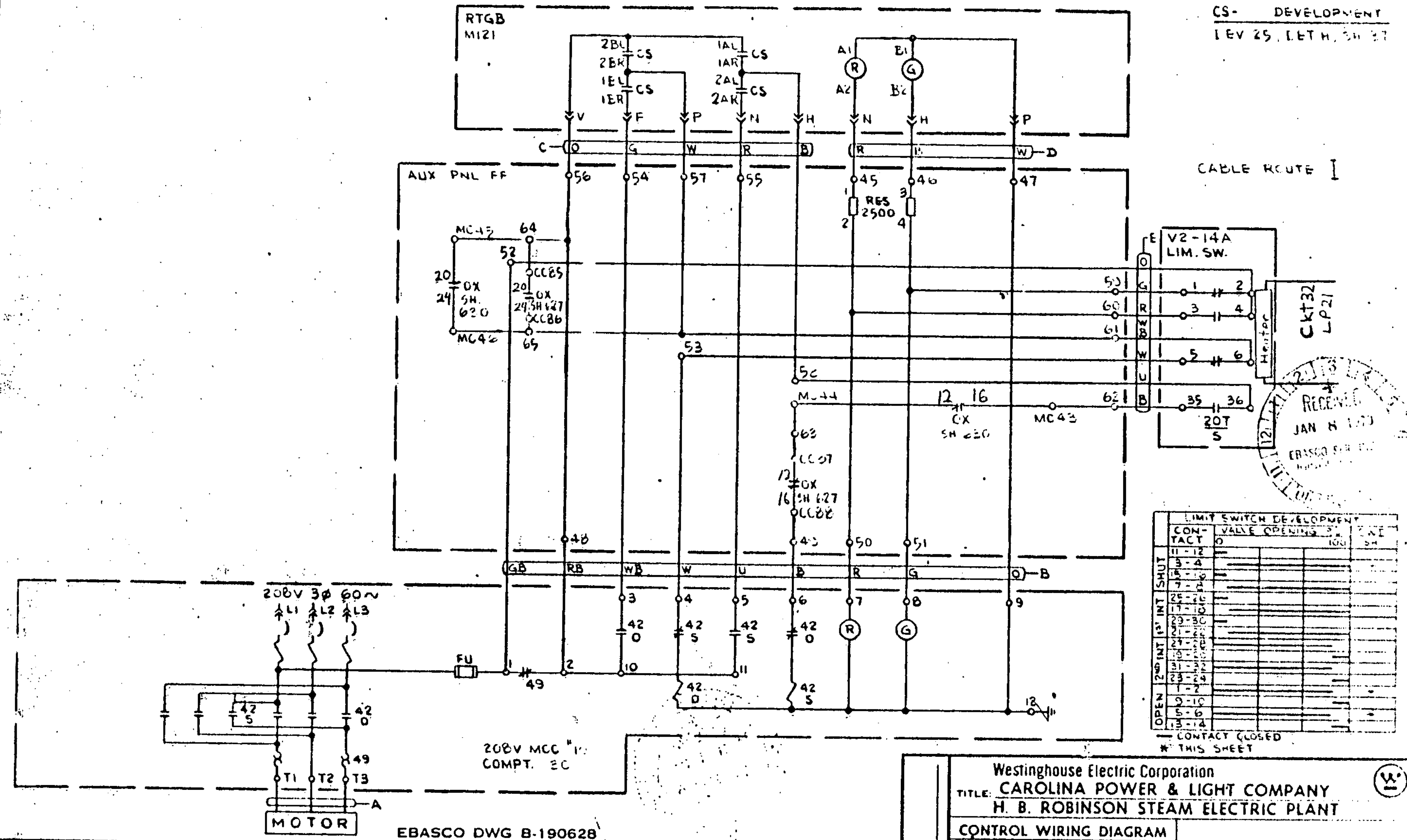

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STEAM DRIVEN FWP
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CONTROL WIRING DIAGRAM

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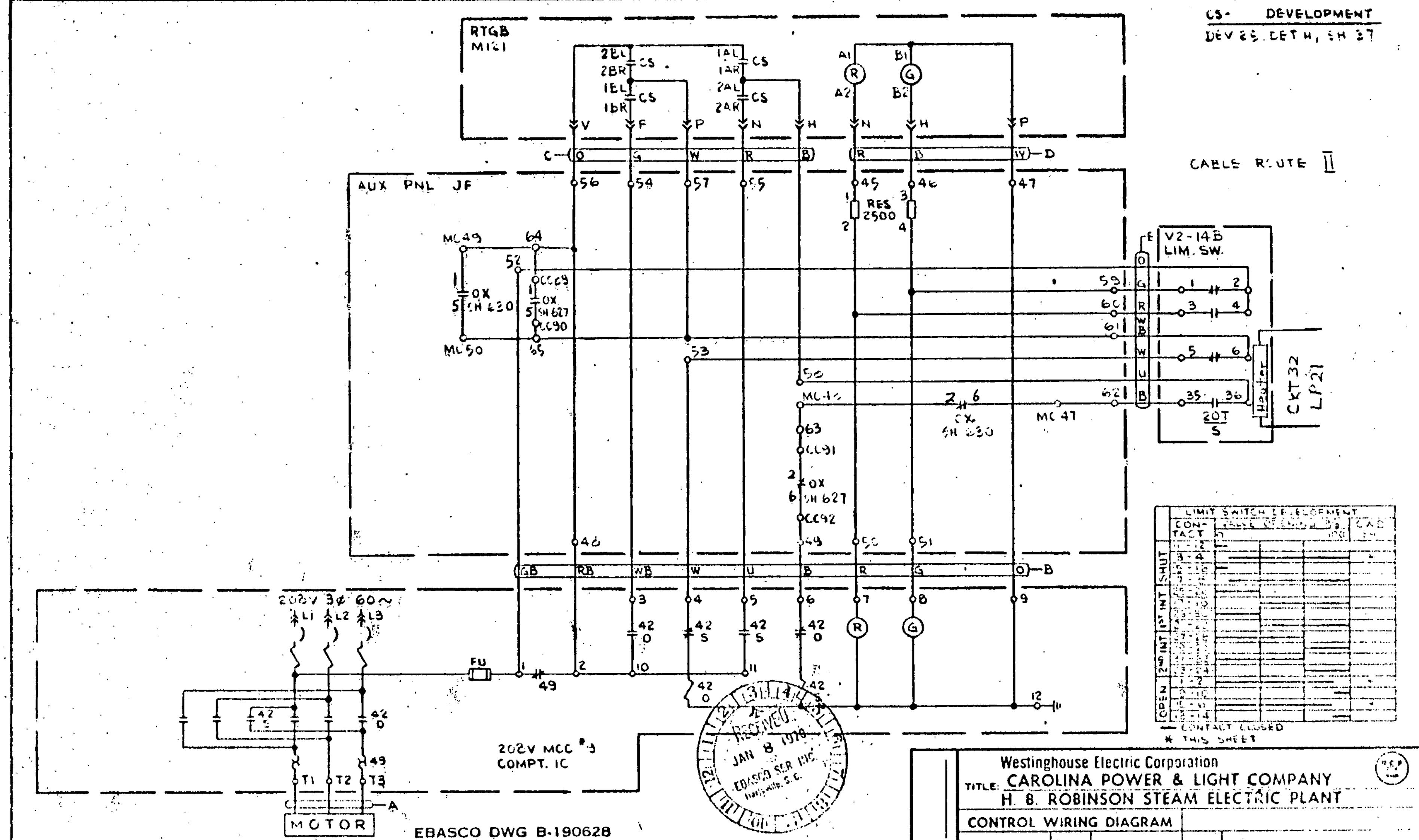

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DATE JUL 11 1969
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CONTROL WIRING DIAGRAM

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PITTSBURGH, PA. U.S.A.

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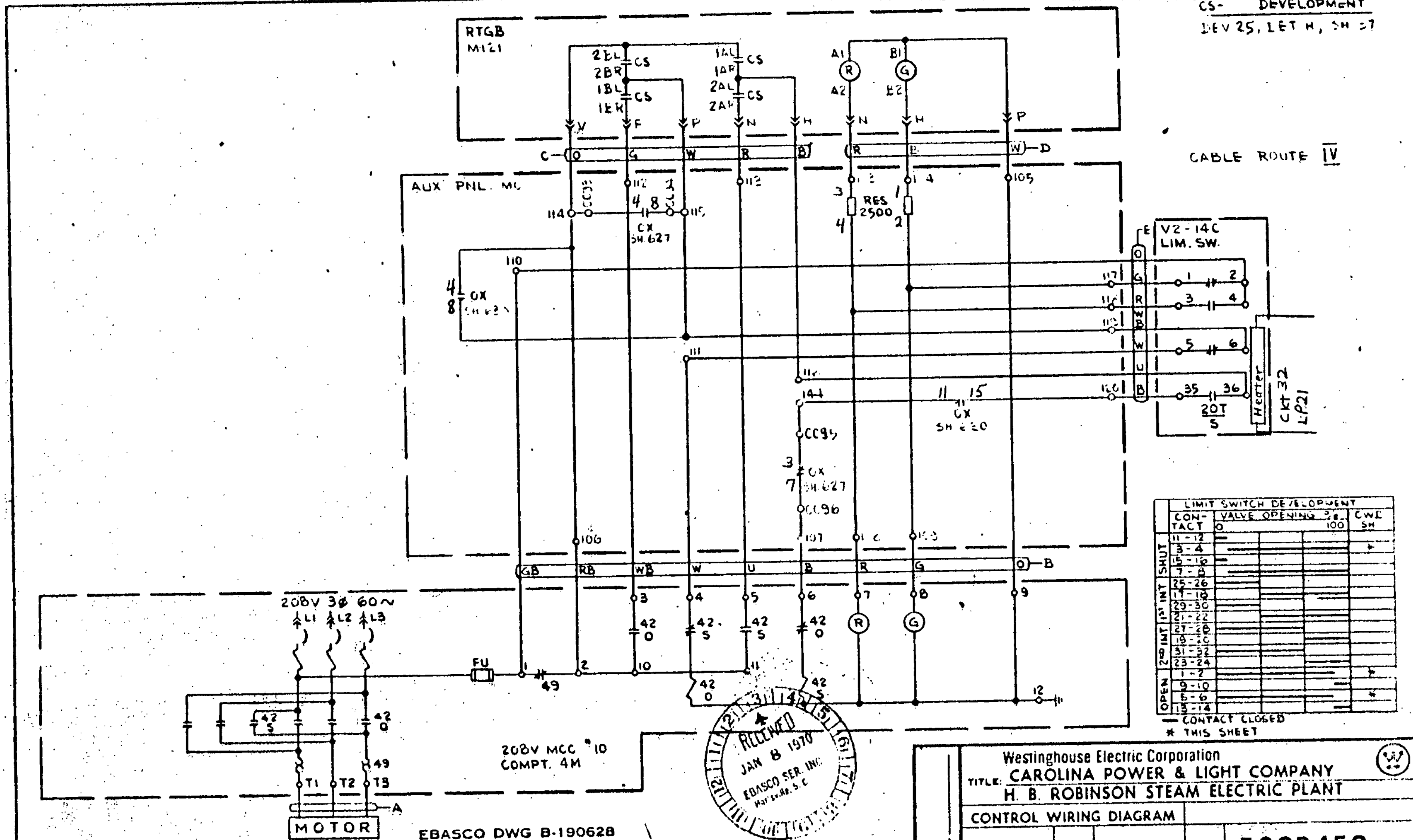

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H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM

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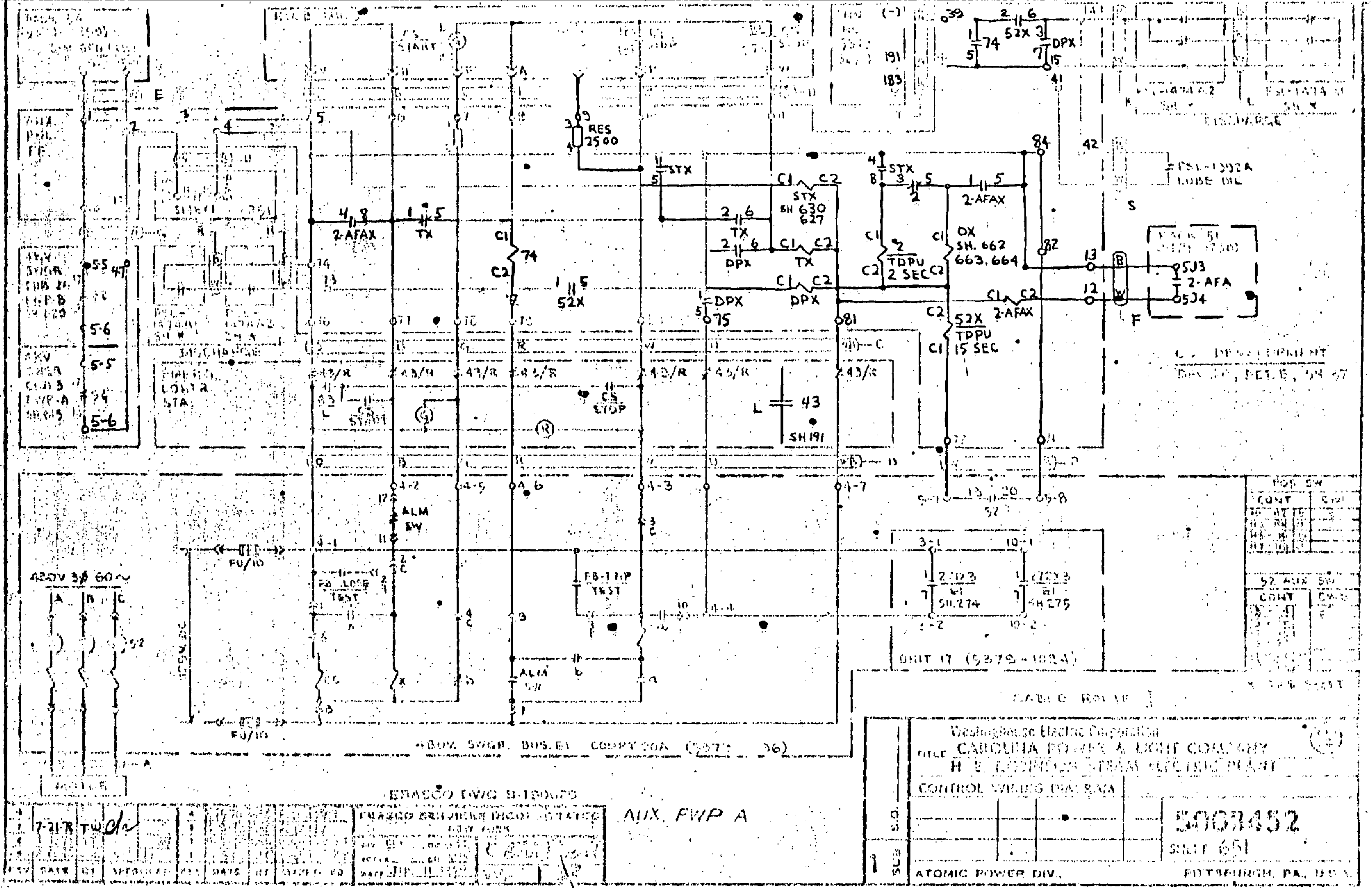
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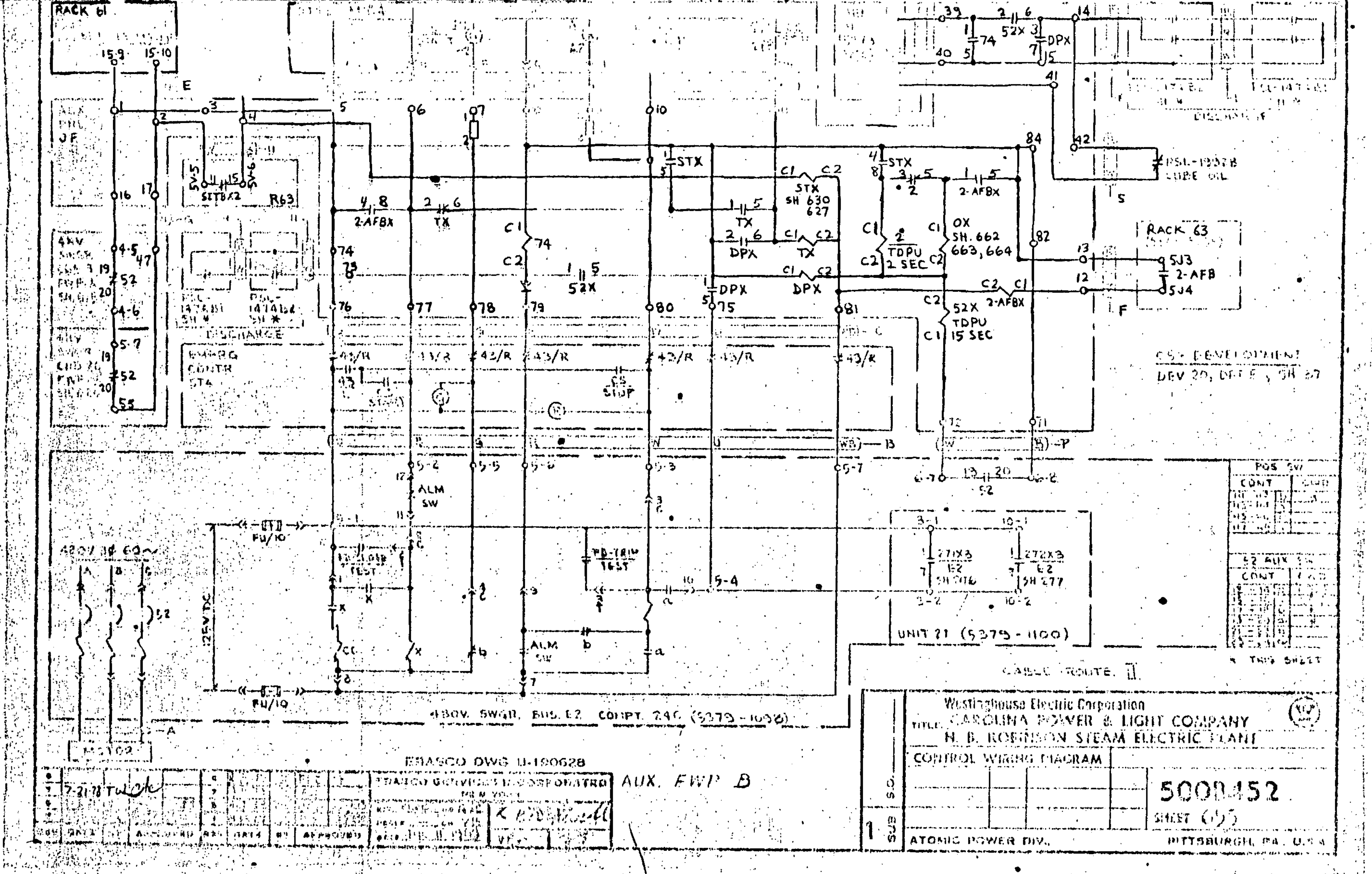
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CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM
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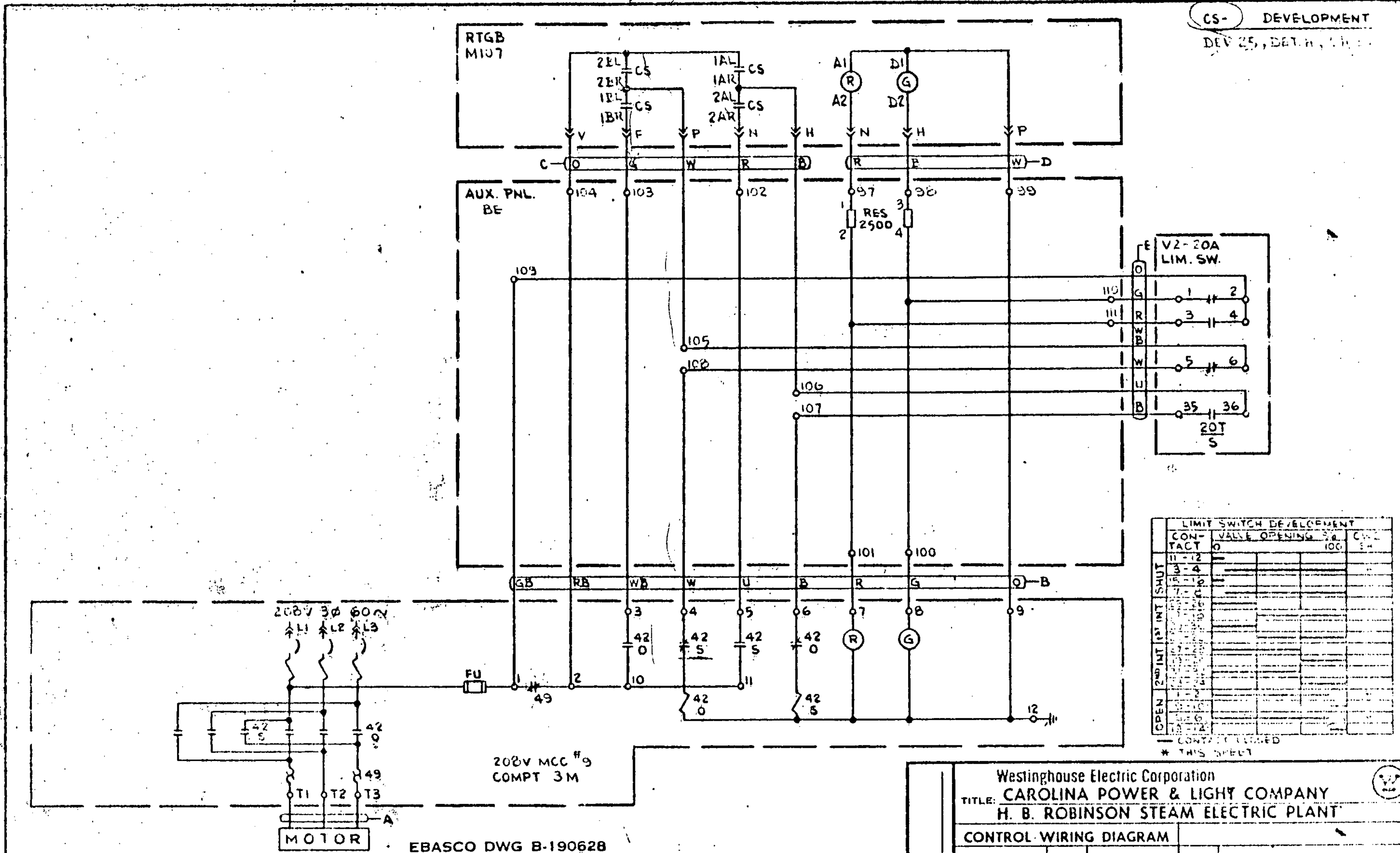

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Westinghouse Electric Corporation

TITLE: CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT

CONTROL WIRING DIAGRAM

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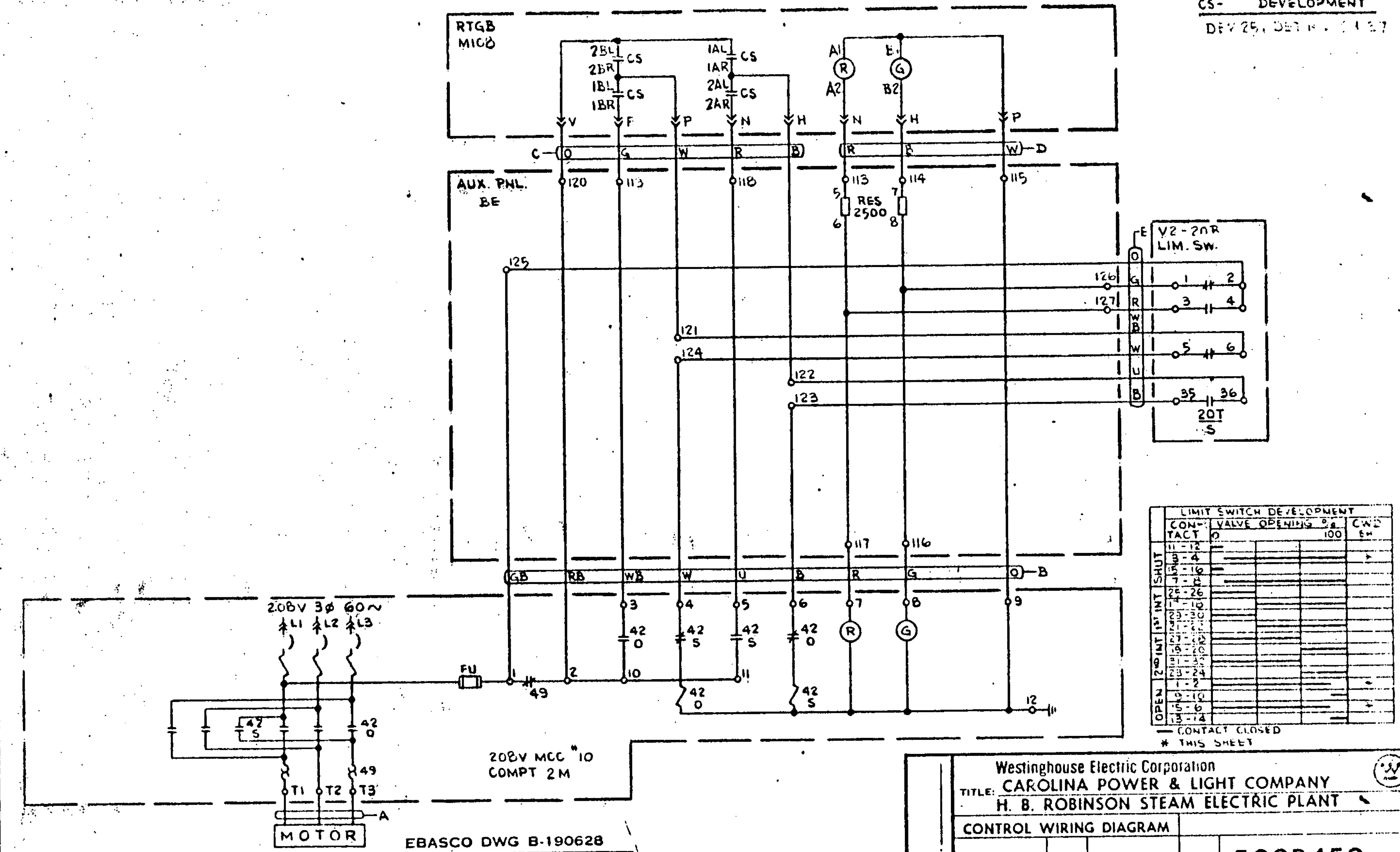
S.O. 1 SUB

ATOMIC POWER DIV.. PITTSBURGH, PA., U.S.A.

CS- DEVELOPMENT



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APPROVED K Brockwell

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H. B. ROBINSON STEAM ELECTRIC PLANT

CONTROL WIRING DIAGRAM

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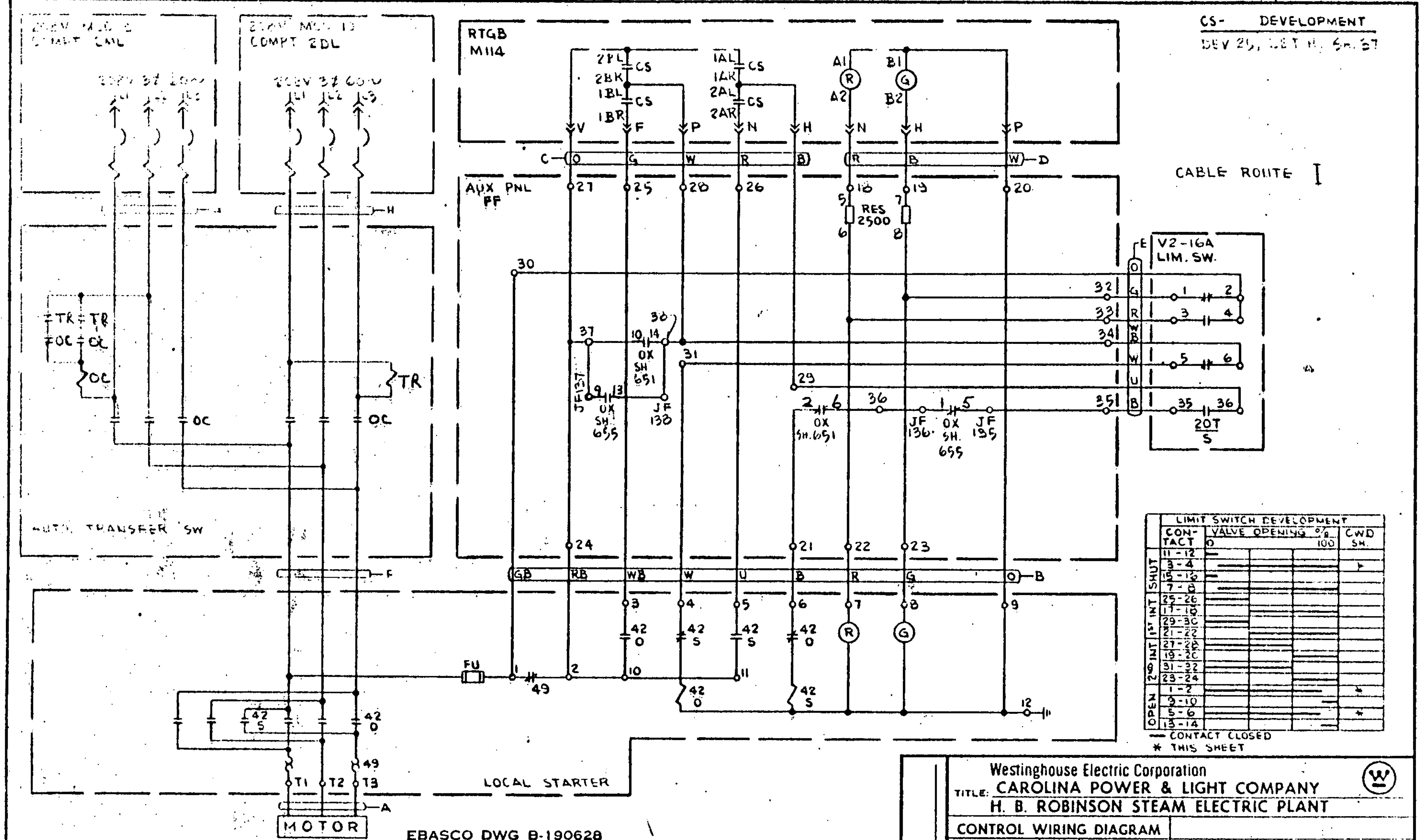

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NEW YORK
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DATE JUL 11 1969
APPROVED K Brockwell

AUX. FWP HEADER
DISCH. VA. V2-16A

Westinghouse Electric Corporation
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H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM

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PITTSBURGH, PA., U.S.A.

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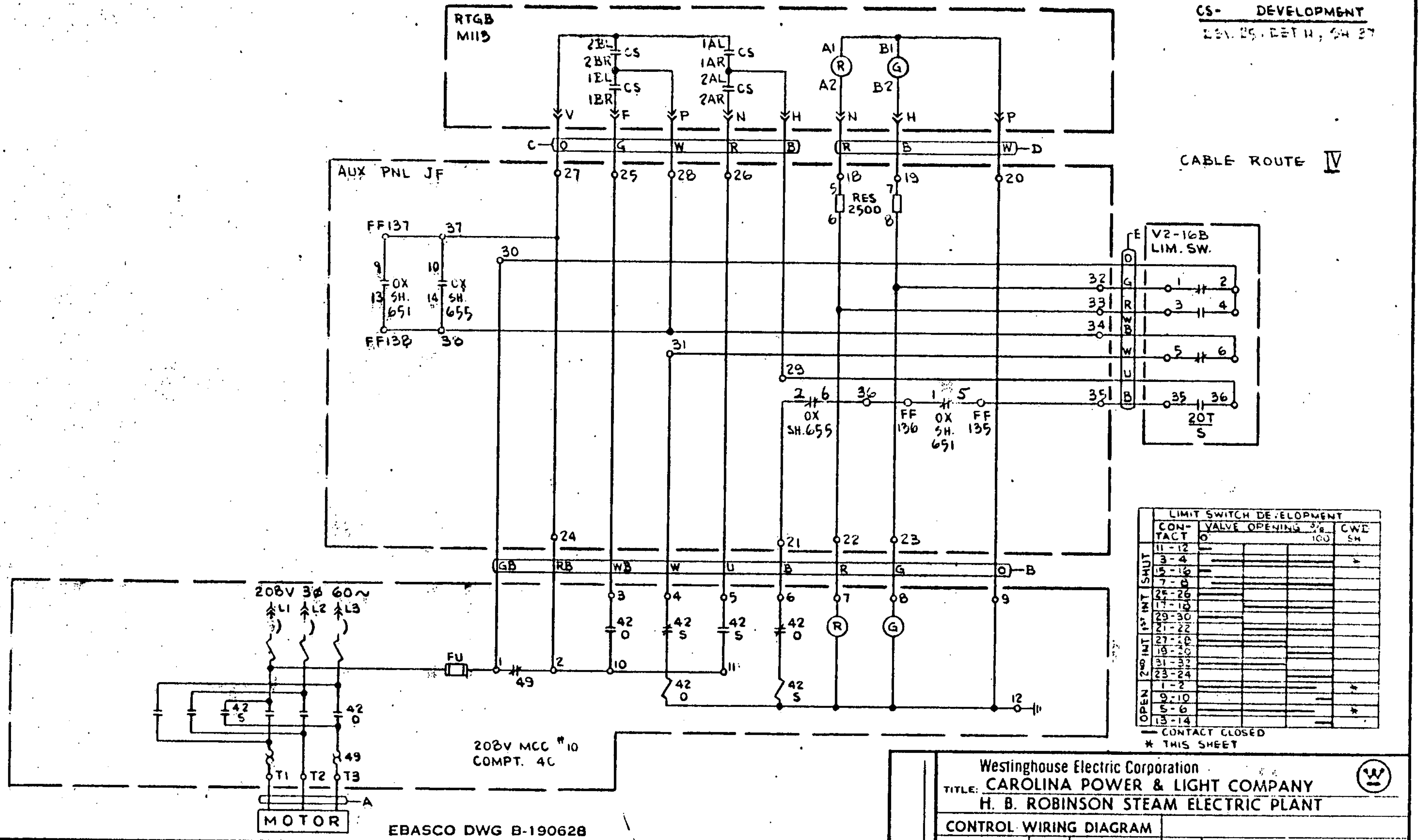

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Westinghouse Electric Corporation

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H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM

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ATOMIC POWER DIV., PITTSBURGH, PA., U.S.A.

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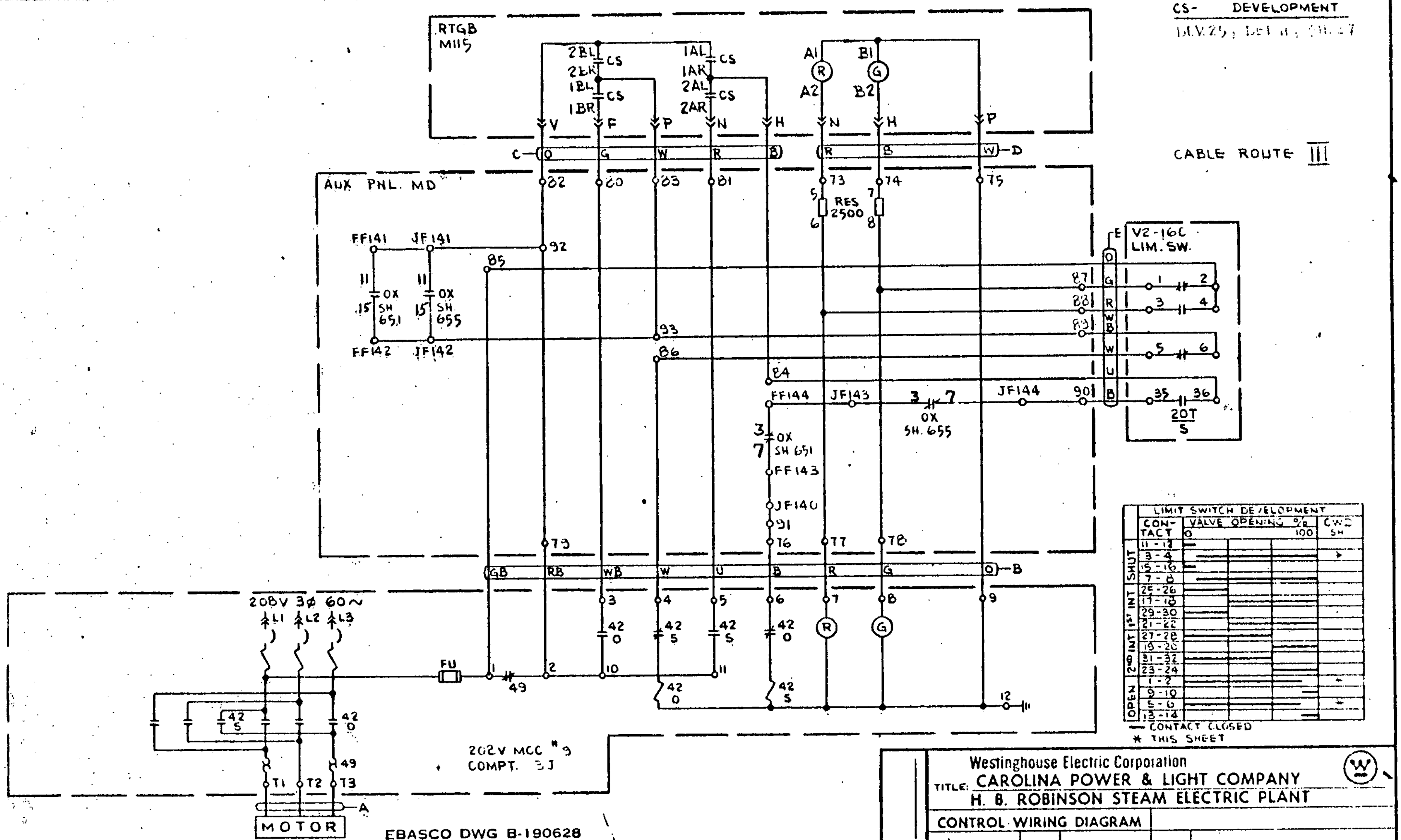
CABLE ROUTE III



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AUX. FWP HEADER
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Westinghouse Electric Corporation
TITLE: CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT
CONTROL WIRING DIAGRAM

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500B452
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ATOMIC POWER DIV.
PITTSBURGH, PA., U.S.A.

AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS (2.1.7.b)

POSITION

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

CLARIFICATION

A. Control Grade (Short-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy the single failure criterion.

2. Testability of the auxiliary feedwater flow indication channels shall be a feature of the design.

3. Auxiliary feedwater flow instrument channels shall be powered from the vital instrument buses.

B. Safety-Grade (Long-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy safety-grade requirements.

C. Other

1. For the Short-Term the flow indication channels should by themselves satisfy the single failure criterion for each steam generator. As

a fall-back position, one auxiliary feed water flow channel may be backed up by a steam generator level channel.

2. Each auxiliary feed water channel should provide an indication of feed flow with an accuracy on the order of ± 10%.

CP&L Implementation of Item 2.1.7.b

Auxiliary feedwater flow instrumentation is installed with indication in the Control Room. The instrumentation consists of six channels of flow instrumentation, three channels, one for each of the three steam generators for both the steam driven and motor driven auxiliary feedwater pumps. A block diagram of the system is attached.

Each channel of the auxiliary feedwater flow instrumentation consists of two transducers, a local mounted flow display computer and an analog edge meter in the Control Room.

The AFW Flowmeter senses the flow rate of a liquid from outside the pipe, and does not require the intrusion of a sensing device in the pipe. It is intended for use of any liquid that conducts sound, which is a property of all homogeneous liquids. Detection of the flow rate is performed by two transducers which are connected to a flow display computer by coaxial cables. An ultrasonic pulse is transmitted through the pipe wall and through the liquid from one transducer to the other. When the pulse is received, the transducer sends an identical pulse back through the pipe wall and liquid to the first transducer, enabling comparison of the effect of flow on both upstream and downstream sound beam velocity. The flow display computer analyzes the received signals and converts the flow information to a pulse rate proportional to liquid volumetric flow rate. In this form, the data is used to operate the analog display on the control board in the Control Room.

A. Control Grade (Short-Term)

1. Auxiliary feedwater flow indication satisfy the single failure criterion since there are two flow indicators for each generator (one for motor driven and one for the steam driven AFW pumps). Flow from one pump to one steam generator is sufficient to shut down the plant.

2. Each of the six instruments can be tested and calibrated. An installation/calibration procedure has been instituted to assure proper operation and accuracy ($\pm$ 5%).

3. Each of the AFW flow instruments is powered from an emergency (vital) bus.

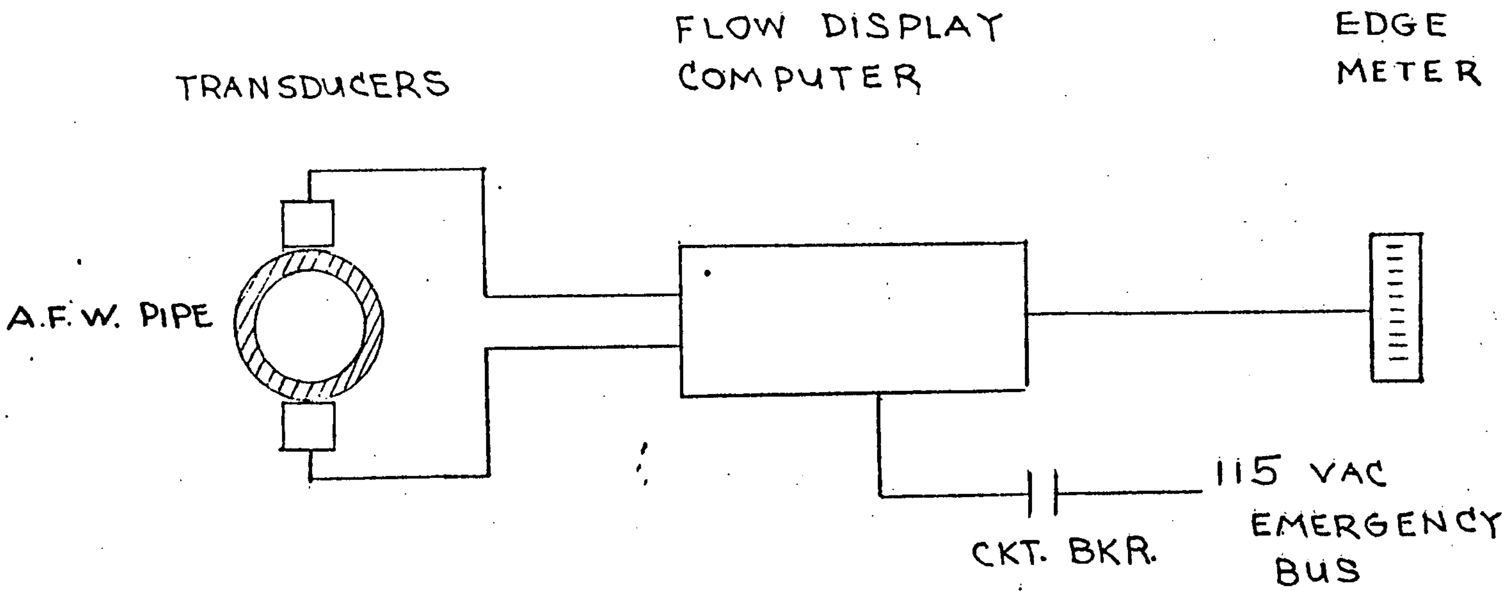
B. Safety-Grade (Long-Term)

1. These AFW flow instruments will be upgraded to safetygrade requirements prior to January 1, 1981.

C. Other

1. The channels satisfy the single failure criterion as discussed in A.1. above and each channel is also backed up by steam generator level indication.

2. Each auxiliary feedwater flow instrument provides an indication of flow with an accuracy of $\pm$ 5%.



AUXILIARY FEEDWATER FLOW METER

POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

DISCUSSION

The primary purpose of implementing Improved Post-Accident Sampling Capability is to improve efforts to assess and control the course of an accident by:

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

2. Determining the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment;

3. Providing information on coolant chemistry (e.g., dissolved gas, boron and pH) and containment hydrogen.

The above information requires a capability to perform the following analyses:

1. Radiological and chemical analyses of pressurized and unpressurized reactor coolant liquid samples;

2. Radiological and hydrogen analyses of containment atmosphere (air) samples.

CLARIFICATION

The licensee shall have the capability to promptly obtain (in less than 1 hour) pressurized and unpressurized reactor coolant samples and a containment atmosphere (air) sample.

The licensee shall establish a plan for an onsite radiological and chemical analysis facility with the capability to provide, within 1 hour of obtaining the sample, quantification of the following:

1. certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums and non-volatile isotopes),
2. hydrogen levels in the containment atmosphere in the range 0 to 10 volume percent,
3. dissolved gases (i.e., H_2, O_2) and boron concentration of liquids.

or have in-line monitoring capabilities to perform the above analysis. Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples and a design review and procedural modifications (if necessary) shall be completed by January 1, 1980. Major plant modifications shall be completed by January 1, 1981.

During the review of the post accident sampling capability consideration should be given to the following items:

1. Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

2. The licensee shall consider provisions for purging samples lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

3. If changes or modifications to the existing sampling system are required, the seismic design and quality group classification or sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and and piping downstream of the second isolation valve can be designed to quality Group D and nonseismic Category I requirements.

The licensee's radiological sample analysis capability should include provisions to:

a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Lessons Learned Item 2.1.6.b. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel

exposure, should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/gm to the upper levels indicated here.

b. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

c. Maintain plant procedures which identify the analysis required, measurement techniques and provisions for reducing background levels.

The licensees chemical analysis capability shall consider the presence of the radiological source term indicated for the radiological analysis.

In performing the review of sampling and analysis capability, consideration shall be given to personnel occupational exposure. Procedural changes and/or plant modifications must assure that it shall be possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as reasonably achievable and not in excess of GDC 19. In assuring that these limits are met, the following criteria will be used by the staff.

1. For shielding calculations, source terms shall be as given in Lessons Learned Item 2.1.6.b.

2. Access to the sample station and the radilogical and chemical analysis facilities shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples in the analysis facilities shall be such that the radiation dose criteria are met. This may involve sufficient shielding of personnel from the samples and/or the dilution of samples for analysis. If the existing facilities do not satisfy these criteria, then additional design features, e.g., additional shielding, remote handling etc. shall be provided. The radioactive sample lines in the sample station, the samples themselves in the analysis facilities, and other radioactive lines of the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

4. High range portable survey instruments and personnel dosimeters should be provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

The licensee shall demonstrate their capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria given in this section.

CP&L Implementation of Item 2.1.8.a

A design and operational review of the reactor coolant and containment atmosphere sampling systems has been completed by the H. B. Robinson Plant. This review was conducted in conjunction with the Radiation Shield Design Review of NUREG-0578, Item 2.1.6.b. The combined efforts of both of these studies has shown that the design of Reactor Coolant Sampling System and the Containment Vessel Atmosphere Sampling System will not allow a sample of the Regulatory Guide 1.4 magnitude to be drawn wtihout overexposing the personnel who drew the sample. Although the sample can be drawn within an hour, the operator will receive greater than 3 Rem whole body and 18 3/4 Rem to his extremities. This is due solely to the magnitude of the source term for the sample and volume of sample in the unshielded sample lines. As discussed in the response to NUREG-0578, Item 2.1.6.a, the H. B. Robinson Plant is currently evaluating the alternatives which are available to minimize the dose rate at the sampling stations. The required changes will be completed prior to January 1, 1981.

A design and operational review of the radiological spectrum analysis facilities and chemical analysis facilities has been completed for the H. B. Robinson S.E.G. Plant. This review evaluated the effect of radiation fields from the Auxiliary Building following the postulated accident and the radiation fields from the sample being analyzed and their resultant effect on the lab technician and counting equipment. This review indicated that the analysis equipment and procedures are adequate but due to the magnitude of the activity of the sample, the lab technician who performed the analysis could receive more than 3 Rem whole body or 18 3/4 Rem to the extremities while analyzing the sample. Therefore, the H. B. Robinson Plant is evaluating several alternatives to the current counting laboratory design. The alternatives being considered are:

a. Use of a mobile hot cell to transport, store and handle the sample.

b. Automatic dilution equipment.

c. In line sampling equipment.

d. Additional shielding in sample sink and counting lab areas.

The design changes which are determined to be necessary to allow a technician to analyze a highly contaminated sample within two hours of an accident without exceeding 3 Rem whole body and 18 3/4 Rem to his extremities will be completed prior to January 1, 1981.

INCREASED RANGE OF RADIATION MONITORS (2.1.8.b)

POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident", which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

    a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

    b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

DISCUSSION

The January 1, 1980 requirement, were specifically added by the Commission and were not included in NUREG-0578. The purpose of the interim January 1, 1980 requirement is to assure that licensees have methods of quantifing radioactivity releases should the existing effluent instrumentation go offscale.

CLARIFICATION

1. Radiological Noble Gas Effluent Monitors

    A. January 1, 1980 Requirements

    Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for

quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to serve only as a provisional fix with the more detailed, exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble gases from all potential release points, (e.g., auxiliary building, radwaste building, fuel handling building, reactor building, waste gas decay tank releases, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves and BWR turbine buildings) and any other areas that communicate directly with systems which may contain primary coolant or containment gases, (e.g., letdown and emergency core cooling systems and external recombiners). Measurements/analysis capabilities of the effluents at the final release point (e.g., stack) should be such that measurements of individual sources which contribute to a common release point may not be necessary. For assessing radioiodine and particulate releases, special procedures must be developed for the removal and analysis of the radioiodine/particulate sampling media (i.e., charcoal canister/filter paper). Existing sampling locations are expected to be adequate; however, special procedures for retrieval and analysis of the sampling media under accident conditions (e.g., high air and surface contamination and direct radiation levels) are needed.

It is intended that the monitoring capabilities called for in the interim can be accomplished with existing instrumentation or readily available instrumentation. For noble gases, modifications to existing monitoring systems, such as the use of portable high range survey

instruments, set in shielded collimators so that they "see" small sections of sampling lines is an acceptable method for meeting the intent of this requirement. Conversion of the measured dose rate (mR/hr) into concentration (µCi/cc) can be performed using standard volume source calculations. A method must be developed with sufficient accuracy to quantify the iodine releases in the presence of high background radiation from noble gases collected on charcoal filters. Seismically qualified equipment and equipment meeting IEEE-279 is not required.

The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

1. Noble Gas Effluents

   a. System/Method description including:

      i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

      ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

      iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, Control room read-out is perferred: however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b. Procedures for conducting all aspects of the measurement/analysis including:

i) Procedures for minimizing occupational exposures

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration.

B. January 1, 1981 Requirements

By January 1, 1981, the licensee shall provide high range noble gas effluent monitors for each release path. The noble gas effluent monitor should meet the requirements of Table 2.1.8.b.2. The licensee shall also provide the information given in Sections 1.A.1.a.i, 1.A.1.a.ii, 1.A.1.b.ii, 1.A.1.b.iii, and 1.A.1.b.iv above for the noble gas effluent monitors.

2. Radioiodine and Particulate Effluents

A. For January 1, 1980 the licensee should provide the following:

1. System/Method description including:

a) Instrumentation to be used for analysis of the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

b) Monitoring/sampling location.

c) Method to be used for retrieval and handling of sampling media to minimize occupational exposure.

d) Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

e) If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous read-out for 7 consecutive days.

2. Procedures for conducting all aspects of the measurement analysis including:

a) Minimizing occupational exposure

b) Calculational methods for determining release rates

c) Procedures for dissemination of information

d) Calibration frequency and technique

B. For January 1, 1981, the licensee should have the capability to continuously sample and provide onsite analysis of the sampling media. The licensee should also provide the information required in 2.A above.

3. Containment Radiation Monitors

Provide by January 1, 1981, two radiation monitor systems in containment which are documented to meet the requirements of Table 2.1.8.b.2. It is possible that future regulatory requirements for emergency planning interfaces may necessitate identification of different types of radionuclides in the containment air, e.g., noble gases (indication of core damage) and non-volatiles (indication of core melt). Consequently, consideration should be given to the possible installation or future conversion of these monitors to perform this function.

TABLE 2.1.8.b.1

INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL ACCIDENTAL RADIOACTIVITY RELEASES

- Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

- Examples of major elements of a highly radioactive effluent release special procedures (noble gas).
  - Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.
  - Provide shielding to minimize background interference.
  - Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.
  - Predetermined calculational method to convert the radiation level to radioactive effluent release rate.

TABLE 2.1.8.b.2

HIGH RANGE EFFLUENT MONITOR

- NOBLE GASES ONLY
- RANGE: (Overlap with Normal Effluent Instrument Range)

- UNDILUTED CONTAINMENT EXHAUST	10^{+5}	μCi/CC
- DILUTED (>10: 1) CONTAINMENT EXHAUST	10^{+4}	μCi/CC
- MARK I BWR REACTOR BUILDING EXHAUST	10^{+4}	μCi/CC
- PWR SECONDARY CONTAINMENT EXHAUST	10^{+4}	μCi/CC
- BUILDINGS WITH SYSTEMS CONTAINING PRIMARY COOLANT OR GASES	10^{+3}	μCi/CC
- OTHER BUILDINGS (E.G., RADWASTE)	10^{+2}	μCi/CC

- NOT REDUNDANT - 1 PER NORMAL RELEASE POINT
- SEISMIC - NO
- POWER - VITAL INSTRUMENT BUS
- SPECIFICATIONS - PER R.G. 1.97 AND ANSI N320-1979
- DISPLAY*: CONTINUOUS AND RECORDING WITH READOUTS IN THE TECHNICAL SUPPORT CENTER (TSC) AND EMERGENCY OPERATIONS CENTER (EOC)
- QUALIFICATIONS - NO

*Although not a present requirement, it is likely that this information may have to be transmitted to the NRC. Consequently, consideration should be given to this possible future requirement when designing the display interfaces.

TABLE 2.1.8.D.3

HIGH RANGE CONTAINMENT RADIATION MONITOR

- RADIATION: TOTAL RADIATION (ALTERNATE: PHOTON ONLY)
- RANGE:
  - UP TO 10^8 RAD/HR (TOTAL RADIATION)
  - ALTERNATE: 10^7 R/HR (PHOTON RADIATION ONLY)
  - SENSITIVE DOWN TO 60 KEV PHOTONS*
- REDUNDANT: TWO PHYSICALLY SEPARATED UNITS
- SEISMIC: PER R. G. 1.97
- POWER: VITAL INSTRUMENT BUS
- SPECIFICATIONS: PER R.G. 1.97 REV. 2 AND ANSI N320-1978
- DISPLAY: CONTINUOUS AND RECORDING
- CALIBRATION: LABORATORY CALIBRATION ACCEPTABLE

*Monitors must not provide misleading information to the operators assuming delayed core damage when the 80 KEV photon Xe-133 is the major noble gas present.

CP&L IMPLEMENTATION OF ITEM 2.1.8.b

I. High Range Noble Gas Effluent Monitors

A. January 1, 1980 Requirements

There are three potential high level release paths at the the H. B. Robinson Unit No. 2 site; namely, the plant vent stack, the ventilation exhaust from the basement of the Fuel Handling Building and the steam safety valves and atmosphere steam dump valves. For the interim, the following actions have been taken:

1. Plant Vent Stack

The plant vent stack combines exhaust effluents from the Reactor Auxiliary Building, the condenser air ejector (on initiation of high alarm), the upper levels of the Fuel Handling Building, and the Containment purge.

At present the noble gases from the plant vent stack are monitored by the Plant Vent Gas Monitor (R-14) which consists of four thin-walled GM tubes mounted on a horizontal probe within the stack. The instrument sensitivity ranges from 5×10^{-7} uCi/cc to approximately 5×10^{-2} uCi/cc.

In order to monitor potential noble gas releases from the vent stack in the magnitude of 10,000 Ci/sec (or 4.3×10^{2} uCi/cc), a high range monitor has been installed on the plant vent stack in the vicinity of the existing R-14 monitor.

a. Instrumentation

The high range vent stack monitor consists of a gamma sensitive GM tube, set in a 3" lead shielded collimator and positioned facing the vent stack so as to "see" a small volume of the vent stack. The GM tube has a range of 1 mR/hr to 1×10^{5} mR/hr and energy dependence of 100 kev to 10 Mev. Calibration will be accomplished via (1) an electronics calibration, (2) a standard area monitor calibration with known sources, and an "in place" calibration with (3) three radioactive sources of known energies and strengths, reflective of the postulated accident source spectrum. A calibration will be conducted during each refueling, and in conjunction with the calibration of the normal radiation monitoring system.

b. Monitoring/sampling locations

The high range vent stack monitor is located on a section of stack which is 5 stack diameters downstream from the closest substream, as in accordance with the guidelines set forth in ANSI N13.1. Three inches of lead shield the sides and back of the detector from postulated high back-

ground activity, and provide a minimum source to background difference of 100.

c. Access to Radiation readings

Continuous readout in the control room and in the vicinity of the detector has been provided. Readout at both locations is in mR/hr.

d. Source of power

The high range vent stack monitor receives its power from the vital instrument bus. In the event of a loss of power, the plant diesel generator will provide an alternate back-up power supply.

e. Procedures for conducting all aspects of the measurements/analysis.

i) Occupational exposures have been minimized by providing a continuous monitor readout in the Control Room. During an accident, access to the detector itself is not required.

ii) A calculational method for converting instrument readings (mR/hr) into release rates (Ci/sec) has been developed using standard volume source calculations. The calculation (Attachment No. 8 of Plant Modification No. 505), assumes an isotopic mix which would be representative of the postulated accident, and an exhaust air flow equal to that of the Reactor Auxiliary Building and Containment purge.

iii) A curve for converting the instrument readings (mR/hr) into release rates (Ci/sec) has been derived from ii (above). For ease of dissemination of information, the curve has been placed in Volume 15 of the Plant Operating Manual. Procedures concerning its use have been added to the plant's emergency instructions.

iv) Procedures for calibration are discussed in detail in Attachment No. 10 of Plant Modification No. 505.

2. Fuel Handling Building Basement Exhaust

The Fuel Handling Building basement exhaust sweeps areas of the lower level of the Fuel Handling Building including the spaces for the liquid waste holdup tanks and waste gas decay tanks.

At present, noble gases from the Fuel Handling Building basement exhaust are monitored continuously by an offline monitor (R-20) located at a point near the atmospheric discharge. The instrument sensitivity ranges from 3 X 10^{-6} uCi/cc to 1 X 10^{-1} u Ci/cc.

In order to monitor potential noble gas releases from the Fuel Handling Building Basement Exhaust in the magnitude of 10,000 Ci/sec (or 2.1 X 10^3 uCi/cc), a high range monitor has been installed adjacent to the basement exhaust duct in the vicinity of the existing R-20 monitor.

The instrumentation shielding source of power, procedures and ease of accessibility and dissemination of information are identical to those of the Plant Vent Stack High Range Radiation Monitor.

3. Steam Safety Valves and Atmospheric Steam Dump Valves

A calculational method has been developed for the interim to quantify release rates of up to 10,000 Ci/sec for noble gases and iodines released from steam safety valves and atmospheric steam dump valves. The calculational method is based upon Regulatory Guide 1.4 releases of fission products and the maximum allowable primary to secondary leak rate for continued reactor operation. A curve for estimating the total activity released (Curies) as a function of time for which the valves are open has been added to Volume 15 of the Plant Operating Manual. Procedures concerning the use of the curve have been added to the plant's emergency instructions.

This calculational method is to serve only as an interim solution until high range main steam line radiation monitors can be procured and installed.

B. January 1, 1981 Requirements

By January 1, 1981 H. B. Robinson Unit No. 2 shall have procured and installed high range noble gas effluent monitors for each of the three aforementioned effluent paths. The gas effluent monitors will meet the requirements of NUREG Table 2.1.8.b.2, providing that such instrumentation has been adequately tested and is readily available from qualified radiation monitoring system suppliers.

II. ASSESSING RADIOIODINE AND PARTICULATE EFFLUENTS

A. January 1, 1980 Requirements

1. Plant Vent Stack

A review of both the existing equipment and procedures for assessing radioiodine and particulate releases has been conducted. The sample media can be obtained and handled during accident conditions providing that good health physics practices of time, distance, and shielding are employed. Existing instrumentation and procedures for analyzing the radioiodine/particulate sample media are adequate. Recommendations for reducing personnel exposures and increasing the accuracy of the sample media analysis have been performed.

2. Fuel Handling Building Basement Exhaust

A modification has been made to the existing Fuel Handling Building Exhaust low range monitor (R-20) to provide intermittent particulate/ iodine sampling. Methods and procedures for analyzing the sample media during accident conditions are identical to those of the Plant Vent Stack.

3. Steam Safety Valves and Atmospheric Steam Dump Valves

At present, there is no capability of sampling the main steam lines for particulates and iodines during a Design Basis Accident. However, based on the calculation performed for Section I.A.3 "Steam Safety Valves and Atmospheric Steam Dump Valves", iodine releases are estimated to be less than one percent of the total released from the steam safety valves and atmospheric steam dump valves.

B. January 1, 1981 Requirements

By January 1, 1981, H. B. Robinson Unit No. 2 will have the capability to continuously sample and provide onsite analysis of the sample media for all three potential release paths.

III. HIGH RANGE CONTAINMENT RADIATION MONITORS

A minimum of the high range containment monitoring systems will be procured and installed to meet the intent of NUREG 0578, prior to January 1, 1981.

IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS (2.1.8.c)

POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

CLARIFICATION

Use of Portable versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

Iodine Filters and Measurement Techniques

A. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 365 keV of ^{131}I. A representative air sample shall be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too must activity collected on the sampling cartridge.

B. By January 1, 1981:

The licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble bases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions.

CP&L IMPLEMENTATION OF ITEM 2.1.8.c

a. January 1, 1980 Requirements.

The H. B. Robinson Steam Electric Plant does have the capability to accurately detect the presence of iodine in the region of interest following an accident.

All plant areas requiring access following an accident have been determined. All areas requiring access can be sampled for iodine with existing site air sampling equipment.

The H. B. Robinson Steam Electric Plant does have cart mounted (portable) iodine air sampling equipment with single channel analyzer (SCA) capable of being calibrated to the 364- 365 KEV energy of I131. The existing site equipment has the capability of collecting a representative sample and counting for iodine 131 using the SCA. This will give a conservative estimate of the presence of iodine and can be used, for determination of respiratory protection equipment requirements. Sampling with this equipment will be performed in accordance with established practices that assure sampling system is not inoperative as a result of the sampling cartridge being saturated by too much activity. As the site is currently using the portable iodine air sampling equipment, procedures exist on its use and personnel authorized to use such equipment have received the appropriate training.

b. January 1, 1981 Requirements.

The H. B. Robinson Steam Electric Plant currently has the capability to analyze the sampling cartridge in the counting laboratory which is a low background, low contamination area. This area is currently ventilated with clean air from a unit separate from the Reactor Auxiliary Building air supply and return unit.

The purging of the samples to remove any entrapped noble gases will not be performed in the counting lab as this could contaminate the clean atmosphere. Purging will be performed in the Controlled Sample Room (Chem. Lab). Currently, samples are dessicated using an oven and vacuum system. This process provides a more efficent means of removing entrapped noble gases.

TRANSIENT AND ACCIDENT ANALYSIS (2.1.9)

POSITION

See NUREG-0578, page A-44.

DISCUSSION

The scope of the requied transient and accident analysis is discussed in NUREG-0578. The schedule for these analyses is included in NUREG-0578 and is reproduced in the Implementation Schedule attachment to this letter. The Bulletins and Orders Task Force has been implementing these required analyses on that schedule. The analysis of the small break loss of coolant accident has been submitted by each of the owners groups. These analyses are presently under review by the B&O Task Force. The scope and schedule for the analysis of inadequate core cooling have been discussed and agreed upon in meetings between the owners groups and the B&O Task Force, and are documented in the minutes to those meetings.

The analysis of transients and accidents for the purpose of upgrading emergency procedures is due in early 1980 and the detailed scope and schedule of this analysis is the subject of continuing discussions between the owners groups and the B&O Task Force.

CP&L Implementation of Item 2.1.9

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 Carolina Power & Light Company 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 Robinson Plant, Unit No. 2, Emergency Instruction One (E.I.-1), Incident Involving Reactor Coolant System Depressurization, has been revised to include the key considerations of current Westinghouse provided guidance in the following four areas:

1. Immediate actions and diagnostics for transients involving safety injection initiation (E-0).

2. Loss of reactor coolant (E-1).

3. Loss of secondary coolant (E-2).

4. Steam generator tube rupture (E-3).

Unit No. 2 is a 3 loop Westinghouse PWR with 1400 PSI range safety injection pumps.

The Westinghouse guidance was incorporated into the present unit procedure format while insuring the intent of the guidance was not altered. The key considerations included in the unit emergency procedure are:

1. Diagnostic guidance including event decision trees.

2. Guidance concerning the use of available indications including, but not limited to, the incore thermocouples and the saturation monitor.

3. Reactor coolant pump trip requirements.

4. Safety injection termination criteria including an adequate margin to insure subcooled conditions exist and can be maintained.

5. Procedural guidance to insure that switchover from injection to recirculation mode during a LOCA will be accomplished prior to emptying the refueling water storage tank.

6. Procedure guidance to insure that the safety injection pumps will not be run deadheaded when in the recirculation mode.

7. Procedural guidance to minimize or prevent inadvertent releases of radioactivity during a steam generator tube rupture.

The remainder of the notes and precautions contained in the guidance provided by Westinghouse were incorporated only as applicable to the Robinson Plant, Unit No. 2, in a format consistent with existing unit procedures.

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 and accidents contained in Chapter 14 of the H. B. Robinson FSAR, the Westinghouse Owners' Group is performing 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 will be prepared, along with a list of information available to the operator at each decision point. Following this, criteria will be set for credible misoperation, and time available for operator decisions will be qualitatively assessed. The information developed will then be used to test Abnormal and Energy Operating Procedures against the event sequences and determine if inadequacies exist in the AOPs and EOPs. The results of this study will be provided to the Bulletins and Orders Task Force prior to the end of the first quarter of 1980, as agreed to by the Bulletins and Orders Task Force.

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

ACRS Item - Containment Pressure Indication

This item is not a Category A item. As stated in our submittal of October 18, 1979, CP&L intends to implement the required modifications for this item by January 1, 1981.

ACRS Item - Containment Hydrogen Indication

This item is not a Category A item. As stated in our submittal of October 18, 1979, CP&L intends to implement the required modifications for this item by January 1, 1981.

ACRS Item - Containment Water Level Indication

This item is not a Category A item. As stated in our submittal of October 18, 1979, CP&L intends to implement the required modifications for this item by January 1, 1981.

REACTOR COOLANT SYSTEM VENTING

POSITION

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses deomonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

CLARIFICATION

A. General

1. The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent non-condensible gases will provide additional assurance of meeting the requirements of 10CFR50.46 (LOCA criteria) and 10CFR50.44 (containment criteria for hydrogen generation). For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of non-condensible gas without the loss of core cooling or containment integrity.

2. Procedures addressing the use of the RCS vents are required by January 1, 1981. The procedures should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be based on the following criteria: (1) assurance that the plant can meet the requirements of 10CFR50.46 and 10CFR50.44 for Design Basis Accidents; and (2) a substantial increase in the plants ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since the BWR owners group has suggested that the present BWR designs inherent capability of venting, this question relates to the capability of existing systems. The ability of these systems to vent the RCS of non-condensible gas must be demonstrated. In addition the ability of these systems to meet the same requirements as the PWR vent systems must be documented. Since there are important differences among BWR's, each licensee should address the specific design features of his plant.

2. In addition to reactor coolant system venting, each BWR licensee should address the ability to vent other systems such as the isolation condenser, which may be required to maintain adequate core cooling. If the production of a large amount of non-condensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. The locations for PWR Vents are as follows:

   a. Each PWR licensee should provide the capability to vent the reactor vessel head.

   b. The reactor vessel head vent should be capable of venting non-condensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths). Additional venting capability is required for those portions of each hot leg which can not be vented through the the reactor vessel head vent. The NRC recognizes that it is impractical to vent each of the many thousands of tubes in a U-tube steam generator. However, we believe that a procedure can be developed which assures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Such a procedure is required by January 1981.

   c. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations especially during natural circulation.

2. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly large range of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which we consider reasonable, is to specify a volume of non-condensible gas to be vented and a venting time i.e., a vent capable of venting a gas volume of 1/2 the RCS in one hour. Other criteria and engineering approaches should be considered if desired.

3. Where practical the RCS vents should be kept smaller than the size corresponding to the definition of a LOCA (10CFR50 Appendix A). This will minimize the challenges to the ECCS since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation although it may result in leakage beyond Technical Specification Limits. On PWRs the use of new or existing valves which are larger than the LOCA definition will require the addition of a block valve which can be closed remotely to terminate the LOCA resulting from the inadvertent opening of the vent.

4. An indication of valve position should be provided in the control room.

5. Each vent should be remotely operable from the control room.

6. Each vent should be seismically qualified.

7. The requirements for a safety grade system is the same as the safety grade requirement on other Short Term Lessons Learned items, that is, it should have the same qualifications as were accepted for the reactor protection system when the plant was licensed. The exception to this requirement is that we do not require redundant valves at each venting location. Each vent must have its power supplied from an emergency bus. A degree of redundancy should be provided by powering different vents from different emergency buses.

8. For systems where a block valve is required, the block valve should have the same qualifications as the vent.

9. Since the RCS vent system will be part of the reactor coolant systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of non-condensible gas could be associated with substantial core damage, venting to atmosphere is unacceptable because of the associated released radioactivity. Venting into containment is the only presently available alternative. Within containment those areas which provide good mixing with containment air are preferred. In addition, areas which provide for maximum cooling of the vented gas are preferred. Therefore the selection of a location for venting should take advantage of existing ventilation and heat removal systems.

11. The inadvertent opening of an RCS vent must be addressed. For vents smaller than the LOCA definition, leakage detection must be sufficient to identify the leakage. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46.

CP&L Implementation of Denton Items

Proposals for Reactor Vessel Head and Pressurizer Venting System were requested from four (4) suppliers. Two suppliers responded and an engineering evaluation was performed. The evaluation included requirements outlined in "Discussions of Lessons Learned Short Term Requirements" and clarifications. The results of the evaluation were presented to H. B. Robinson Plant management and a supplier has been selected.

The Reactor Coolant Gas Vent System is designed to remotely (from Control Room) vent gases from the reactor vessel head and pressurizer steam space during post-accident situations when large quantities of non-condensible gases may collect in these high points. Although primarily designed to be available during post-accident conditions, the system can also be used to aid in the RCS venting procedures following a maintenance outage. Detailed operation procedures will be developed by January 1, 1981.

The design criteria for the vent system are as follows:

A. The system permits remote (Control Room) venting from the reactor vessel head or the pressurizer.

B. The vent flow rate capability is based upon the following considerations:

1. The vent rate is sufficient to vent one-half of the RCS volume in standard cubic feet in one hour.

2. Coolant liquid loss through the vent will not exceed makeup capacity.

3. The vent mass rate will not result in heat loss from the RCS in excess of the normal pressurizer heater capacity.

C. The vent path will be safety grade meeting the same qualifications as were accepted for the RCS at time of licensing. The power operated valves will be 1E powered with power removed during normal operation to minimize the possibility of inadvertent operation. Redundant paths will be provided.

D. The design will minimize modification to currently installed safety class equipment.

E. The system will be designed to limit RCS mass loss to below the definition of a LOCA in 10 CFR 50, Appendix A and thus a separate analysis of inadvertent system operation or pipe breakage is not required. A Failure Mode and Effect Analysis (FMEA) will be provided.

F. The vent system will be operable following all design basis events except those requiring evacuation of the Control Room, and loss of all AC power (plant blackout).

G. The vent system will be capable of venting directly to containment.

H. The vent system will be designed to vent superheated steam, steam/water mixtures, water, fission gases, helium, nitrogen, and hydrogen as high as 2500 psia and 700°F.

I. Control Room position indication will be provided for all power operated valves.

J. The system will be designed not to interfere with refueling maintenance actions.

The Reactor Coolant Gas Vent System is designed to vent non-condensible gases from the reactor coolant system during post-accident conditions. The purpose of venting is to prevent possible interference with core cooling. Pressure instrumentation is included in the design to monitor system performance.

Although designed for accident conditions, the system may be used to aid in the pre- or post-refueling venting of the reactor coolant system. Although venting of the CRDMs and RCPs will still be necessary, pressurizer and reactor vessle venting can be accomplished with the system if desired. Vent flow will be directed to the pressurizer relief tank for this operation to prevent inadvertent release of radioactive fluid to the containment.

SHIFT SUPERVISOR RESPONSIBILITIES (2.2.1.a)

POSITION

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly secified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

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

CLARIFICATION

The attachment provides clarification to the above position.

Attachment

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.A)

NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V. P. For Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V. P. for Operations.

CP&L IMPLEMENTATION OF ITEM 2.2.1.a

CP&L Response To NRC Position No. 1:

A management directive that emphasized the primary management responsibilities of the Shift Operating Supervisor at the Brunswick S.E.G. Plant and the Unit Two Shift Foreman at the H. B. Robinson S.E.G. Plant (CP&L equivalents to Shift Supervisor) for the safe operation of the Company's nuclear power plants under all conditions and that clearly established their command duties has been issued by the Vice President, Nuclear Operations. The duties and responsibilities of these individuals (Shift Supervisors) will be reviewed by the office of the Vice President of Nuclear Operations on an annual basis. Following each review, the management directive will be reissued to reinforce Carolina Power and Light Company's commitment to the safe operation of its nuclear power plants.

CP&L Response To NRC Position No. 2:

Procedures at the H. B. Robinson S.E.G. Plant were reviewed and changes incorporated as necessary to insure that the duties, responsibilities, and authority of the Unit No. 2 Shift Foreman and the Control Room operators were properly defined. A definite line of command at the plant has been established, and the command decision authority of the Unit No. 2 Shift Foreman relative to other plant management personnel has been clearly delineated.

CP&L Response To NRC Position No. 2.a:

The concept that the Shift Foreman is responsible for maintaining the broadest perspective of operational conditions affecting the safety of the plant and that this must be a matter of the highest priority at all times is clearly established in the H. B. Robinson Plant procedures. The idea that the Unit No. 2 Shift Foreman should not become totally involved in any single operation in times of emergency when multiple operations are required in the Control Room is also reinforced by plant procedures.

CP&L Response To NRC Position No. 2.b:

H. B. Robinson S.E.G. Plant procedures have been revised to require that the Unit No. 2 Shift Foreman, until properly relieved, remain in the Control Room at all times during accident situations to direct the activities of the Control Room operators. Individuals authorized to relieve the Unit No. 2 Shift Foreman have been specified.

CP&L Response To NRC Position No. 2.c:

H. B. Robinson S.E.G. Plant procedures have been revised to require that, should it be necessary for the Unit No. 2 Shift Foreman to be temporarily absent from the Control Room during routine operations, a lead Control Room operator be designated to assume the Control Room command function. His temporary duties, responsibilities and authority have been clearly defined in Volume 1 of the Plant Operating Manual, Administrative Instructions.

CP&L Response To NRC Position No. 3:

The CP&L training programs for the Unit No. 2 Shift Foreman emphasizes and reinforces his responsibility for the safe operation and his management function in assuring plant safety. Company training programs are continuously being reviewed and revised. It is expected that the Unit No. 2 Shift Foreman Training Program will be revised to meet the standard of Section 5.2.1.8 of the new draft of ANS 3.1 when it is issued.

CP&L Response To NRC Position No. 4:

The administrative duties of the H. B. Robinson S.E.G. Plant Unit No. 2 Shift Foreman have been reviewed by the Vice President, Nuclear Operations. The administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant have been delegated to other operations personnel not on duty in the Control Room.

SHIFT TECHNICAL ADVISOR (Section 2.2.1.b)

POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the Shift Technical Advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

DISCUSSION

The NRC Lessons Learned Task Force has recommended the use of Shift Technical Adviors (STA) as a method of immediately improving the plant operating staff's capabilities for response to off-normal conditions and for evaluating operating experience.

In defining the characteristics of the STA, we have used the two essential functions to be provided by the STA. These are accident assessment and operating experience assessment.

1. Accident Assessment

   The STA serving the accident assessment function must be dedicated to concern for the safety of the plant. The STA's duties will be to diagnose off-normal events and advise the shift supervisor. The duties of the STA should not include the manipulatin of controls or supervision of operators. The STA must be available, in the control room, within 10 minutes of being summoned.

   The qualifications of the STA should include college level education in engineering and science subjects as well as training in reactor operations both normal and off-normal. Details regarding these qualifications are provided in paragraphs A.1, 2 and 3 of Enclosure 2 to our September 13, 1979 letter. In addition, the STA serving the accident assessment function must be cognizant of the evaluations performed as part of the operating experience assessment function.

2. Operating Experience Assessment

The persons serving the opeating experience assessment function must be dedicated to concern for the safety of the plant. Their function will be to evaluate plant operations from a safety point of view and should include such assignments as listed on pages A-50 and A-51 of NUREG-0578. Their qualifications are identical to those described previously under accident assessment and collectively this group should provide competence in all technical areas important to safety. It is desirable that this function be performed by onsite personnel.

CLARIFICATION

1. Due to the similarity in the requirements for dedication to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions. The performance of these two functions may be split if it can be demonstrated the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the STA will be dedicated to concern for the safety of the plant, our position has been that STA's must have a clear measure of independence from duties associated with the commercial operation of the plant. This would minimize possible distractions from safety judgements by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the STA position both in the STA job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor to STA duties as defined herein.

3. It is our position that the STA should be available within 10 minutes of being summoned and therefore should be onsite. The onsite STA may be in a duty status for periods of time longer than one shift, and therefore asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the STA (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent onsite presence. We do not intend, at this time, to specify or advocate a minimum time onsite.

4. The implementation schedule for the STA requirements is to have the STA on duty by January 1, 1980, and to have STAs, who have all completed training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the STAs on duty by that time should enhance the accident and operating experience assessment function at the plant.

CP&L Implementation of Item 2.2.1.b

Our letter of November 16, 1979, provided a summary of our program to implement the requirement for a Shift Technical Advisor (STA) at the Robinson plant. The information contained in that letter is repeated and expanded upon where appropriate to provide a more complete picture of our program and to demonstrate compliance with the requirements above and especially Clarification 4.

The STA concept can be separated into two distinct functions, as set forth by the Nuclear Regulatory Commission. The two functions are defined as 1) operating experience assessment and 2) accident assessment. Both functions will be performed by the same group to ensure proper understanding of the two functions by all individuals involved. This cross-training will be accomplished through the assignment of individuals on a planned program basis, as described below:

1. Operating Experience Assessment

The Operating Experience Assessment function provides additional capability dedicated to the concern for the safety of the plant. The duties to be performed at the Robinson plant that will provide this capability are as follows:

a. Review selected items to be acted upon by PNSC. Present items to PNSC with recommendations for approval or identification of deficiencies.

b. Review plant Licensee Event Reports for completeness, applicability, and meeting of requirements. Also review Licensee Event Reports and operating experiences from other plants to determine applicability of problems reported to our plant.

c. Review plant operations, including forced outages, to determine impact on safety and propose corrective actions that should be implemented. These may include corrective actions that will enhance efficiency and reliability of the plant as well as those which would enhance or prevent the degradation of safety.

d. Review plant modifications, procedure revisions, and new procedures from nuclear safety viewpoint.

e. Review test procedures, maintenance instructions and practices, etc., for adequacy with regard to plant nuclear safety.

f. Disseminate the results of all reviews, including recommendations for corrective action to be taken, to plant management and other members of plant staff.

g. Provide operating experience results for operator training, such as guest lectures, etc.

2. <u>Accident Assessment Function</u>

The accident assessment function is designed to provide additional capability, dedicated to concern for the safety of the plant, for diagnosis of off-normal events. This function of the group will be available onsite continuously during all times that the plant is in other than a cold shutdown condition. In addition to performing duties associated with operating experience assessment, as described above, the STA designated as the accident assessor will perform additional duties. These consist of:

a. Reviewing plant activities in support of plant operations and keeping plant management informed of activities that could adversely affect nuclear safety.

b. Ensuring accomplishment of major maintenance activities and other interface support activities in a safe manner, and obtaining support from other plant personnel and ensuring that proper priority is given when necessary.

c. Keeping constantly aware of plant status and needs.

d. Ensuring the plant is operating within limitations of Technical Specifications by reviewing and evaluating plant operational data and taking necessary steps to maintain plant operation within license limits.

In short, the accident assessor maintains an overall "feel" for the condition of the plant, which would be of aid in his role should an accident occur.

The STA group at Robinson, when fully implemented, will consist of six graduate level engineers or equivalent from the plant and/or General Office staffs. These engineers will be selected based on their level of experience and background in nuclear plant design, maintenance or operations. These six engineers will fulfill the dual role of the STA set forth above in the following manner:

1. Four of the six engineers will provide the continuous coverage necessary to meet the requirements of the accident assessment function. Each engineer will serve for eight weeks in this role, working a rotating shift of 12 hour days for seven straight days and then being off-duty for seven straight days. Thus, two engineers will provide 24 hour coverage for one week, while the other two engineers will provide 24 hour coverage the next week, and so on.

2. The other two engineers will serve onsite for four continuous weeks on a normal eight hour per day, five day per week schedule. They will be dedicated to the operating experience assessment function during this time. At the end of the four weeks, an engineer will rotate into the accident assessor role for eight weeks, while the engineer he replaces will rotate into the operating experience assessor role for four weeks, and so on.

3. Training of the six engineers will be accomplished during 1980 to provide fully trained STAs by the end of 1980. The formal training program has not yet been established, but will provide at least the

level of training and background knowledge recommended in NUREG-0578 and other documents dealing with STA training. The engineers will be trained either while they are on shift during the accident assessor role or in the operating experience assessor role. This training will not interfere with the capability of the accident assessor to be available in the control room within ten minutes of being summoned.

4. The engineers that will fulfill the role of STA beginning January 1, 1980 may not be the same individuals that will fulfill the January 1, 1981 role due to problems encountered in assigning engineers to the position. The number of people and the schedule they follow may also differ from that set forth in items 1-3 above. However, these individuals, whether temporary or permanent, do have an enhanced understanding of transient and accident response resulting from the study of industry analyses performed as a result of the Three Mile Island event, and have reviewed the various procedure changes made as a result of guidelines prepared by the Owners' Group. The STA group will report to the Operations Manager in a staff function, separate from the plant operational function provided by the Shift Operating Supervisors and the Control and Auxiliary Operators. While they have a clear measure of independence from duties associated with the commercial operation of the plant, as established above, their position under the operations manager will ensure that their focus remains toward safe operation of the plant.

Considering the role and intended functions of the STA group, it does not appear necessary that all six engineers must serve as STAs when the plant (both units) is in a cold shutdown condition for refueling or other major maintenance outages. Thus, during these periods some or all of the STAs could be available to assist in the performance of other plant activities. This would be of benefit to the STA by broadening his experience in plant activities, and of benefit to the plant because of the depth of knowledge and experience the STA would bring to the activity.

SHIFT AND RELIEF TURNOVER PROCEDURES (2.2.1.c)

POSITION

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

      (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

CLARIFICATION

No clarification provided.

CP&L Implementation of Item 2.2.1.c

CP&L Response to NRC Position No. 1

At the H. B. Robinson S.E.G. Plant, a checklist has been provided which the oncoming and offgoing Unit No. 2 Shift Foreman (Shift Supervisor) must complete and sign prior to completing shift turnover. The following items have been included in the checklist.

a. Assurance that critical plant parameters, as identified in the plant Technical Specifications, are within allowable limits (parameters and allowable limits are listed on the checklist). These critical plant parameters include vital tank minimum levels (such as the condensate storage tank, diesel fuel oil tank, etc.), penetration pressurization system leakage limit, allowable containment vessel pressure, etc.

b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and acceptance criteria is included on the checklist). This includes systems required to be operable per the Technical Specifications such as Auxiliary Feedwater, Safety Injection, Residual Heat Removal, Containment Spray, Component Cooling Water, etc.

c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode is compared with the Technical Specifications action statement by the completion of a separate form at the beginning of each shift. Due to the impracticality of having four individuals review the same checklist at shift turnover, a separate checklist, which essentially covers the same items as above, has been provided for the oncoming and offgoing control operators to complete and sign.

CP&L Response to NRC Position No. 2

Logs have been provided for completion by the offgoing and oncoming auxiliary operators. Remarks must be included in these logs describing equipment out of service, under maintenance or testing, etc. These logs include what equipment to check (i.e. Auxiliary Feedwater System, Component Cooling, RHR, ER), a check of the various rooms and areas throughout the unit, the frequency of the required inspections, and a separate remarks section in which abnormalities are recorded.

CP&L Response to NRC Position No. 3

A system has been established to evaluate the effectiveness of the shift and relief turnover procedure through Quality Assurance surveillance of shift turnovers.

CONTROL ROOM ACCESS (2.2.2.a)

POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access, and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

CLARIFICATION

No clarification provided.

CP&L IMPLEMENTATION OF ITEM 2.2.2.a

CP&L Response To NRC Position 1:

An administrative procedure has been developed that establishes the authority and responsibility of the individual in charge of the Control Room to limit Control Room access to those individuals directly involved in plant operations or whose skills are needed as an operational aid. During normal operations, access is controlled by the Unit No. 2 Shift Foreman following the guidance provided in the Plant Administrative Instructions. During emergency conditions, access is controlled by the Emergency Coordinator following the guidance provided in the Plant Emergency Plan.

CP&L Response To NRC Position 2:

Procedures have been developed that establish a clear line of authority and responsibility in the Control Room in the event of an emergency. The line of succession for the person in charge of the Control Room has been established and limited to individuals possessing a current Senior Reactor Operators' License. The plan clearly defines the lines of communications and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the Control Room. The normal operations line of command is specified in the position responsibilities section of the Plant Administrative Instructions. The emergency situation line of command is specified in the position responsibilities section of the Plant Administrative Instructions and further defined in the Plant Emergency Plan for senior plant management and corporate management personnel.

ONSITE TECHNICAL SUPPORT CENTER (TSC) 2.2.2.b

POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center. Records that pertain to the as-built conditions and layout of structures, systems and components shall be readily available to personnel in the TSC.

CLARIFICATION

1. By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

   A. Establish a TSC and provide a complete description,

   B. Provide plans and procedures for engineering/management support and staffing of the TSC,

   C. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC,

   D. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminmants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets, or evacuation to the control room),

   E. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC,

F. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable, and

G. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size & Staffing

The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4 Activation

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event" however, the activation of the TSC is discretionary for that class of event.

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be give to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum,

data (historical in addition to current status) should be available to permit the assessment of:

Plant Safety Systems Parameters for:

- Reactor Coolant System
- Secondary System (PWRs)
- ECCS Systems
- Feedwater & Makeup Systems
- Containment

In-Plant Radiological Parameters for:

- Reactor Coolant System
- Containment
- Effluent Treatment
- Release Paths

Offsite Radiological

- Meteorology
- Offsite Radiation Levels

8. Data Transmission

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

9. Structural Integrity

A. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

B. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. Habitability

The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

A. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

B. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

C. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

D. Dose reduction measures such as breathing apparatus and potassium iodide tablets can not be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available.

CP&L Implementation of Item 2.2.2.b

The H. B. Robinson Plant has established a temporary Technical Support Center (TSC) which meets the January 1, 1980 requirements of NUREG-0578 Item 2.2.2.b. The vault and print room of the Main Office Building have been designated as the temporary TSC. This location has been chosen for the following reasons:

a. Records that pertain to the as-built conditions and layout of structures systems and components are readily available in the vault area.

b. This location allows for sufficient space to house the required 25 technical personnel.

c. This building is the farthest permanent structure from the Auxiliary Building of sufficient size that is also contained within the controlled access area. This minimizes effects from direct radiation fields and airborne contamination.

d. Adequate communications are available.

The current site emergency plan requires that all members of the control group report to the Control Room following an accident. The knowledge which the control group gains in personally viewing plant parameters is invaluable in assessing the seriousness of the accident, the current conditions, and required recovery sequence. Therefore, until a direct display of plant parameters is available in the Technical Support Center, the control group will continue to report to the Control Room to assess the seriousness of the accident. After completing an assessment of the accident, the senior member of the control group will direct the appropriate management and engineering support to gather at the Technical Support Center.

The temporary TSC has been provided with the required communications circuits. These communications include:

a. A direct phone line between the TSC, Operational Support and Control Room.

b. A direct phone line between the TSC, Confrol Room and NRC Regional Office.

c. A plant public address system.

d. A Bell telephone system phone line.

A portable airborne sampler and a Beta-Gamma Rate Meter have been made available for use in the Technical Support Center. The radiation shield design review has indicated that no significant radiation fields will exist at the TSC following the postulated accident of Regulatory Guide 1.4. However, the possibility does exist that an airborne contamination problem could exist should a release to atmosphere occur. Therefore, instrumentation to monitor airborne particulate and direct radiation fields have been provided for. Existing plant procedures for donning respiratory equipment and site evacuation will be used for this area.

The H. B. Robinson Plant has been actively pursuing the procurement of a system which will provide a direct display of plant parameters in the TSC. At present, the Robinson Plant believes that a video system using a low light camera equipped with a pan tilt and zoom capability will meet the requirements of NUREG-0578, Item 2.2.2.b. A demonstration of a video system of this type will be presented to the plant staff in early January, 1980. Should this demonstration prove effective, the H. B. Robinson Plant will consider the purchase of that system.

The site emergency plan allows for the control group to operate from the Control Room. Should it be necessary to evacuate the TSC, the remainder of the personnel assigned to the TSC would be transferred to the near site support center to continue their support of the plant recovery.

Carolina Power & Light Company has completed a study of possible locations for the permanent TSC. The selection of a site will mark the completion of the job scope phase. Following the selection of a building location, proposals for the construction of the building will be evaluated. A summary of the construction requirements are as follows:

1. <u>Location</u> - The center will be located within the plant security boundary.

2. <u>Physical Site</u> - The TSC should be large enough to house 25 persons and necessary engineering data and information displays; i.e., TV monitors, recorders, vendor drawings, etc.

3. Activation - The center must be capable of being operated within an hour or two of an accident, although instrumentation in the TSC should provide displays of vital plant parameters from the time the accident begins.

4. Instrumentation - The TSC should contain instrumentation that will be capable of providing displays of vital plant parameters that are comparable to that in the Control Room.

5. Power Supply - The power supply to TSC and instrumentation contained within should be reliable, uninterruptable, and of a quality compatible with the TSC instrumentation requirements.

6. Technical Data - Records that pertain to the as-built conditions and layout of structures, systems, and components shall be readily available to TSC personnel for use in the assessment of an accident.

7. Data Transmission - A data transmission link between the TSC and the Control Room shall be provided.

8. Structural Integrity - The TSC need not be designed to Seismic Category I requirements but should be well built in accordance with sound engineering practice.

9. Habitability - The TSC personnel should be protected from radiological hazards, including direct radiation and airborne contaminants as per General Design Criterion 19 and SRF 6.4.

The construction of this building will be completed prior to January 1, 1981.

ONSITE OPERATIONAL SUPPORT CENTER (SECTION 2.2.2.c)

POSITION

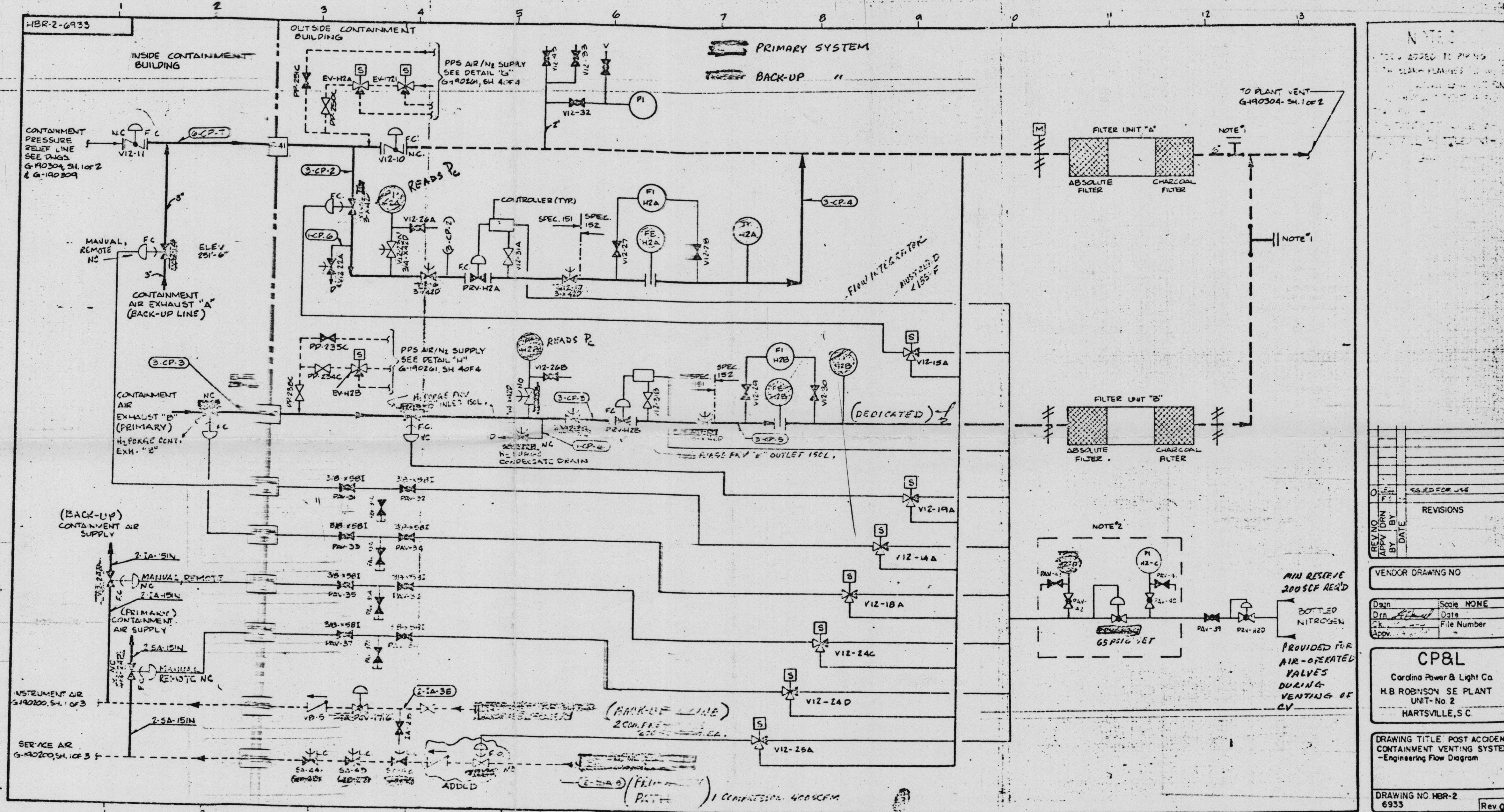
An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

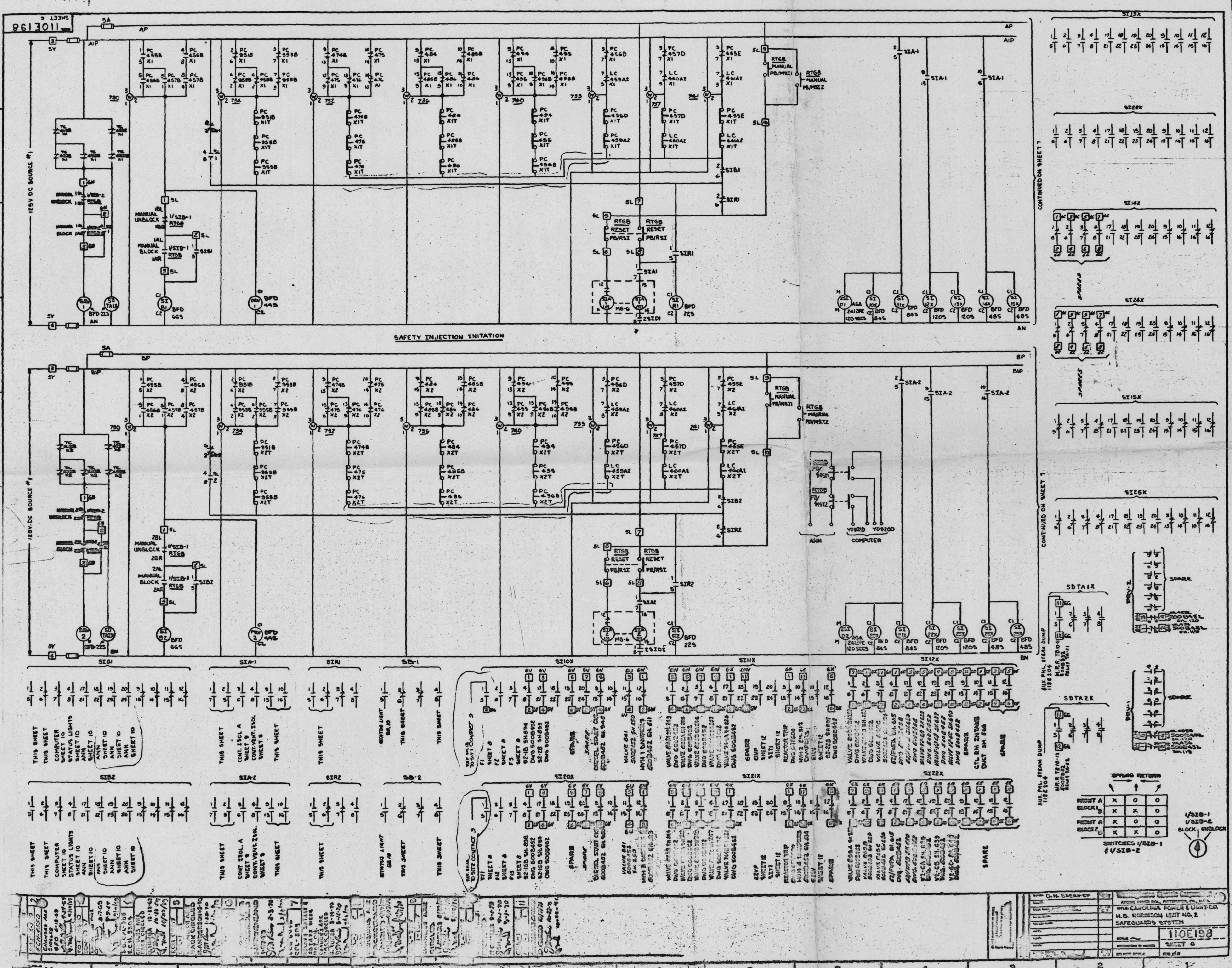
CLARIFICATION

No clarification provided.

CP&L Implementation of Item 2.2.2.c

A portion of the Plant Maintenance Shop has been designated as the On-Site Operational Support Center. It is separate from the Control Room and will be used as the place to which operations support personnel will report in an emergency situation. Three methods of communications with the Control Room have been provided. The Plant Emergency Plan has been revised to reflect the existence of the center and describe the communications available in this center.


HBR-2-6933
INSIDE CONTAINMENT BUILDING
OUTSIDE CONTAINMENT BUILDING
PRIMARY SYSTEM
BACK-UP
PPS AIR/N2 SUPPLY SEE DETAIL "G" G-190261, SH 4 OF 4
CONTAINMENT PRESSURE RELIEF LINE SEE DWGS. G-190304, SH. 1 OF 2 & G-190309
TO PLANT VENT G-190304- SH. 1 OF 2
FILTER UNIT "A"
ABSOLUTE FILTER
CHARCOAL FILTER
NOTE 1
READS Pc
CONTROLLER (TYP.)
MANUAL, REMOTE NC
ELEV 251'-6"
CONTAINMENT AIR EXHAUST "A" (BACK-UP LINE)
FLOW INTEGRATOR
PPS AIR/N2 SUPPLY SEE DETAIL "H" G-190261, SH 4 OF 4
CONTAINMENT AIR EXHAUST "B" (PRIMARY)
H2 PURGE CONT. EXH. "B"
(DEDICATED)
FILTER UNIT "B"
CONDENSATE DRAIN
(BACK-UP) CONTAINMENT AIR SUPPLY
MANUAL, REMOTE NC
(PRIMARY) CONTAINMENT AIR SUPPLY
INSTRUMENT AIR G-190200, SH. 1 OF 3
SERVICE AIR G-190200, SH. 1 OF 3
ADDED
NOTE 2
65 PSIG SET
MIN RESERVE 200 SCF REQ'D
BOTTLED NITROGEN
PROVIDED FOR AIR-OPERATED VALVES DURING VENTING OF CV
V12-15A
V12-19A
V12-14A
V12-18A
V12-24C
V12-24D
V12-25A
REVISIONS
VENDOR DRAWING NO
Scale NONE
File Number
CP&L
Carolina Power & Light Co.
H.B. ROBINSON SE PLANT UNIT-No. 2
HARTSVILLE, S.C.
DRAWING TITLE POST ACCIDENT CONTAINMENT VENTING SYSTEM -Engineering Flow Diagram
DRAWING NO HBR-2 6933
Rev 0

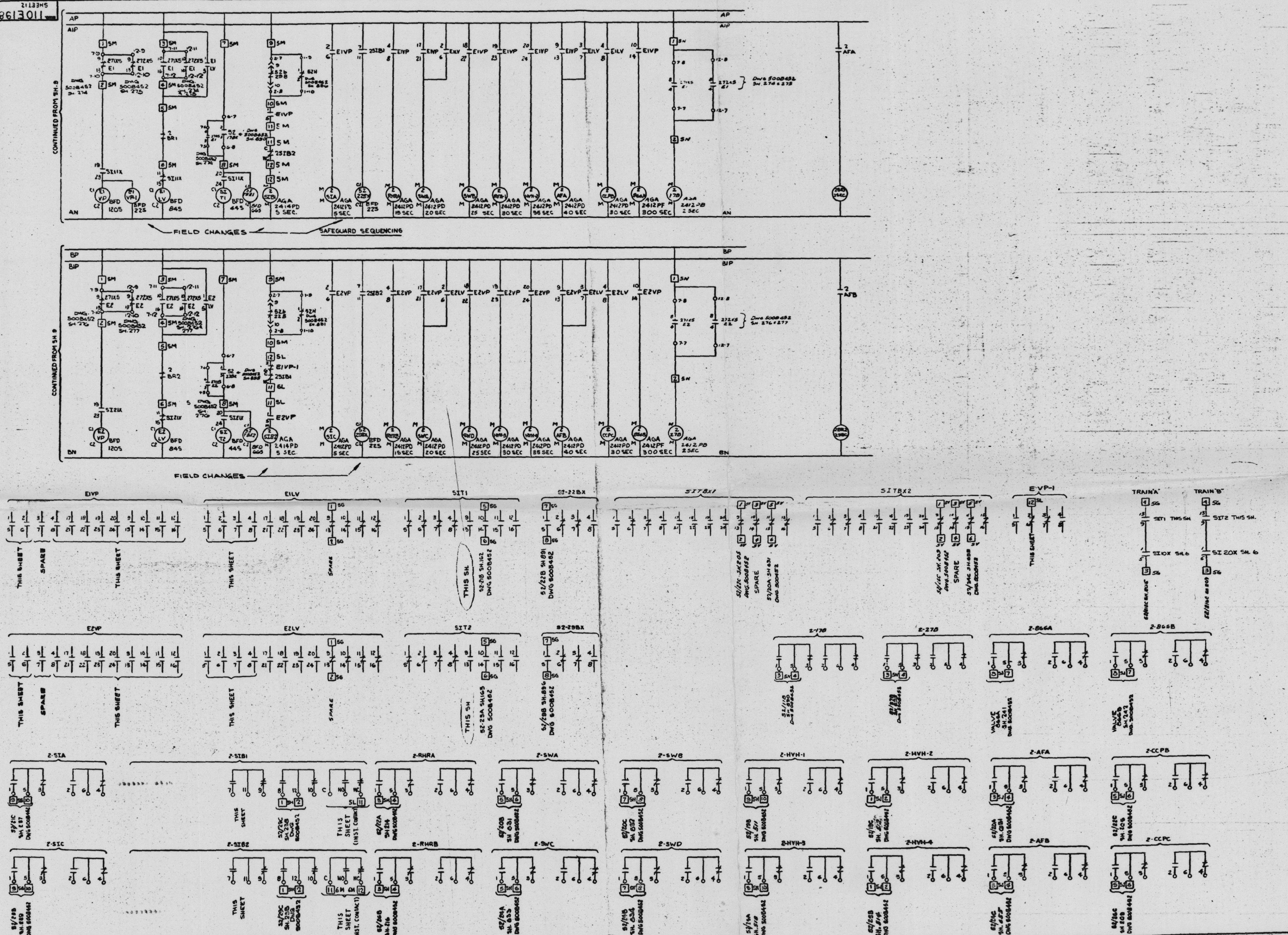


110E198
SHEET 6
125V DC SOURCE #1
125V DC SOURCE #2
SAFETY INJECTION INITATION
MANUAL UNBLOCK
MANUAL BLOCK
RTGB RESET PB/RSI
RTGB MANUAL PB/MSI1
RTGB MANUAL PB/MSI2
CONTINUED ON SHEET 7
SPARES
SPARE
ANN
COMPUTER
SDTA1X
SDTA2X
SPRING RETURN
FRONT A BLOCK
SWITCHES 1/SIB-1 & 1/SIB-2
BLOCK
UNBLOCK
THIS SHEET
COMPUTER
STATUS LIGHTS
CAROLINA POWER & LIGHT CO.
H.B. ROBINSON UNIT NO. 2
SAFEGUARDS SYSTEM
G.H. Shearer

CAROLINA POWER & LIGHT COMPANY
H.B. ROBINSON STEAM ELECTRIC PLANT

1970 - 700,000 KW EXTENSION - UNIT NO. 2
GENERAL ARRANGEMENT
REACTOR AUXILIARY BUILDING - PLANS
ACCESS ROUTES

SCALE: 1/8" = 1'-0" | 11/2/79 | CAR 2762007 | FIGURE I



G.H. Shearer

CAROLINA POWER & LIGHT
H.B. ROBINSON UNIT NO. 2
SAFEGUARDS SYSTEM

110E198
SHEET 12