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November 21, 1979

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

Gentlemen:

Subject: Docket No. 50-206
Additional Information in Support of Responses to NRC Requirements Related to the Three Mile Island Accident
San Onofre Nuclear Generating Station
Unit 1

By letter dated October 17, 1979, we submitted the report entitled "Responses to NRC Requirements Established To Date Following the Three Mile Island Accident, San Onofre Nuclear Generating Station, Unit 1, October, 1979." The report contains our commitments to each of the actions in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," as modified or supplemented by the NRC letter dated September 13, 1979.

By letter dated October 30, 1979, you provided additional clarification of the NRC requirements contained in NUREG-0578 and your September 13, 1979 letter, and delineated those NRC requirements which require prior NRC review and approval and those for which post implementation NRC review is acceptable. In addition, you outlined action items necessary to allow you to complete your review concerning the acceptability of our responses for the NRC short-term requirements established following the Three Mile Island Accident. The action items are as follows:

1. Submit a revised schedule for implementation of all items which do not meet the established NRC implementation schedules, and for those items which can not be met by January 1, 1980, submit a description of the degree of compliance expected on that date and a detailed justification for the delay within fifteen days from receipt of your October 30, 1979 letter,

2. Submit a detailed description of our methods which are not in complete agreement with the NRC requirements along with justification for any differences within fifteen days of receipt of your October 30, 1979 letter,

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3. Submit design details for those items requiring prior NRC approval in a timely manner so that NRC approval and our implementation can be completed by the required date,

4. Document our method of implementation for those items which do not require prior NRC approval by the required date.

Specific responses addressing each of the above action items are provided below (referenced sections correspond to the enumeration contained in our aforementioned October 17, 1979 report):

1. As requested in your October 30, 1979 letter, we have reviewed the implementation schedules contained in our aforementioned October 17, 1979 report with the purpose of improving the schedules to meet those established by the NRC. Based on this review and discussions with members of the NRC staff, we have improved our implementation schedules as discussed below:

   a. For those items required by the NRC to be implemented by January 1, 1980, a special station outage has been scheduled to commence on or about January 1, 1980 to complete the modifications associated with the backup nitrogen pneumatic supply to the PORV block valves (Section 2.1.1), the PORV, block valve and safety valve position indication (Section 2.1.3.a), the diverse containment isolation signal and prevention of automatic reset of containment isolation valves (Section 2.1.4), the controls grade automatic initiation of the Auxiliary Feedwater System (Section 2.1.7.a), and the controls grade auxiliary feedwater flow indication (Section 2.1.7.b).

      The precise shutdown date is predicated on the timely completion of the engineering, procurement and preoutage construction activities associated with each of the above modifications. As discussed with members of the NRC staff, it is anticipating that the current status of these activities for all of the above modifications will support a station outage on or about January 1, 1980 to complete the modifications, except those associated with the controls grade automatic initiation of the Auxiliary Feedwater System. Based on the current status and projection of activities associated with modifications to the Auxiliary Feedwater System, it may not be possible to complete those modifications during the special January 1980 outage in which case the modifications would be completed during the refueling outage currently scheduled for March-April 1980. The delay is associated with the required sequence of engineering and design activities necessary to implement the design details for the modifications to the Auxiliary Feedwater System which are provided in Enclosure 1 to this letter. During the interim period until the modifications are completed, the operator stationed at the manual Auxiliary Feedwater System isolation valves, as required by IE Bulletin No. 79-06A, will promptly initiate auxiliary feedwater flow when necessary. As described in Enclosure 1 to this letter, controls grade auxiliary feedwater flow indication will be provided at this location and in the control room. It is anticipated that the current status of the activities for this modification will support a station outage on January 1980 to complete the modifications.

Following implementation of the design details described in Enclosure 1 to this letter, we will terminate continuous stationing of an individual to promptly initiate auxiliary feedwater.

We will periodically advise you concerning our progress relative to completing the above modifications. As additional information becomes available, we will advise you of the exact date when the station will be shut down.

In addition to the above modifications, the leak rate testing of systems which would carry radioactive fluids outside containment described in our aforementioned October 17, 1979 report (Section 2.1.6.a) will be completed by January 1, 1980. The procedures prepared to perform the leak rate testing will also identify manual isolation valves which would be closed prior to use of the systems following a TMI type accident. This will limit the potential flow of radioactive fluids within these systems. As required, the results of the leak rate testing will be reported by January 1, 1980.

All other items required by the NRC to be implemented by January 1, 1980 will be so completed as identified in our aforementioned October 17, 1979 report.

b. For those items required by the NRC to be implemented by January 1, 1981, plant modifications associated with some of the items have been proposed to be deferred pending completion of the integrated assessment in connection with the Systematic Evaluation Program (SEP). The remainder will be completed independent of the SEP. Those items for which plant modifications have been proposed to be deferred to SEP include new instrumentation to detect inadequate core cooling (Section 2.1.3.b), combustible gas control (Section 2.1.5.a), plant shielding for personnel and equipment (Section 2.1.6.b), safety grade automatic initiation of the Auxiliary Feedwater System (Section 2.1.7.a), safety grade auxiliary feedwater flow indication (Section 2.1.7.b), improved post accident sampling system (Section 2.1.8.a), improved post accident onsite radiological spectrum and chemical analysis capability (Section 2.1.8.a), improved post accident onsite stack effluent radio-iodine analysis capability (Section 2.1.8.b), and reactor coolant system high point venting (Section 3.2). Although the actual extent of the modifications for each of these items has not yet been established, our judgment is that modifications may be extensive and that each of the items will be affected by potential modifications identified by SEP topic reviews of station design and operation. Accordingly, the basis for deferral of each of these items as described in our aforementioned October 17, 1979 report remain applicable. However, we intend to proceed with the evaluations required by the identified items on the NRC required schedules and re-evaluate our assessment following completion of the evaluations to determine if any of the modifications can be completed outside the SEP. We will advise you of the results of our re-evaluation in a timely manner subsequent to January 1, 1980.

Those items which will be completed independent of the SEP include extended range noble gas effluent monitor (Section 2.1.8.b), high range in-containment radiation monitors (Section 2.1.8.b), onsite technical support center data display and transmit system (Section 2.2.2.b), containment pressure indication (Section 3.1.1), containment hydrogen concentration indication (Section 3.1.2), and containment water level indication (Section 3.1.3). Sufficient information is not yet available to accurately project if these modifications can be completed by January 1, 1981. As the design details for these modifications become available, they will be evaluated with the purpose of establishing a reliable implementation schedule. We will advise you of the implementation schedules and the basis therefore if the modifications can not be completed by January 1, 1981.

2. As described in our aforementioned October 17, 1979 report, some of our methods of implementation are not in complete agreement with the NRC requirements. For each such case, a description of our proposed methods along with justification for the differences is provided below:

   a. The backup nitrogen pneumatic supply to the PORV block valves (Section 2.1.1), the PORV, block valve and safety valve position indication (Section 2.1.3.a), and the diverse containment isolation signal and prevention of automatic reset of containment isolation valves (Section 2.1.4) to be installed during the special outage commencing on or about January 1, 1980 may not be seismically or environmentally qualified. If qualified equipment cannot be installed, a qualification program will be implemented to qualify the equipment consistent with the seismic criteria for the existing component or system to which it is a part and with the environment in which it is expected to operate. It is anticipated that the qualification program will be completed and appropriate action taken by January 1, 1981. We will advise you if we are unable to qualify the equipment by January 1, 1981.

   b. The power operated relief valves (PORV's) and their associated block valves will not be supplied from different emergency power buses (Section 2.1.1). At San Onofre Unit 1, two parallel PORV paths are provided for overpressure relief capability. The control power to the PORV and its associated block valve in one path is fed from the same vital bus, and the control power to the opposite PORV and its associated block valve is fed from a separate vital bus. The PORV's and block valves are air operated, and in the event of loss of power and/or loss of instrument air, the PORV's fail closed and the associated block valves fail open. Based on a failure modes and effects analysis, the current method of supplying power to the PORV's and their associated block valves (i.e., "train aligned") provides higher operational reliability relative to single failure protection and redundancy than the NRC requirement. In addition, reactor overpressurization protection safety considerations would be degraded if the NRC requirement is implemented. Therefore, the NRC requirement will not be implemented.

c. A controls grade primary coolant saturation recorder in conjunction with a saturation temperature/pressure curve (Section 2.1.3.b) will be utilized rather than safety grade, redundant primary coolant saturation meters required by the NRC. As clarified by your October 30, 1979 letter, an acceptable alternative method of complying with the stated NRC requirement involves the use of a highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. Enclosure 2 to this letter provides the information required by your October 30, 1979 letter necessary to demonstrate the reliability of the controls grade primary coolant saturation recorder.

3. Of those items requiring prior NRC review and approval, design details are only provided at this time for the item required to be implemented by January 1, 1980 (i.e, Section 2.1.7.a, Controls Grade Automatic Initiation of Auxiliary Feedwater System). Enclosure 1 to this letter describes our method of providing controls grade automatic initiation of the Auxiliary Feedwater System. Although not required as part of this response, Enclosure 1 also describes our method of providing controls grade auxiliary feedwater flow indication to the steam generators (Section 2.1.7.b) for completeness.

   The remaining items requiring prior NRC review and approval are to be implemented by January 1, 1981. The schedule for submitting the design details for these items is as follows:

   a. The functional design of any new instrumentation for inadequate core cooling (Section 2.1.3.b) will be submitted by January 1, 1980. The design details for the new instrumentation will be submitted in a timely manner subsequent to that date when available.

   b. The design details for the high range radiation monitors (Section 2.1.8.b) will be submitted in a timely manner subsequent to January 1, 1980 when available.

   c. The functional design for reactor coolant system high point venting (Section 3.2) will be submitted by January 1, 1980. The design details for the venting system will be submitted in a timely manner subsequent to that date when available.

4. Our method of implementation for those items which do not require prior NRC review and approval will be documented and maintained on file available for NRC inspection.

During a meeting in Bethesda, Maryland on October 24, 1979 with members of your staff, two such items were discussed: (1) shift technical advisor (Section 2.2.1.b), and (2) onsite technical support center (Section 2.2.2.b). For completeness, our method of implementation for these items as discussed at the October 24, 1979 meeting, as modified following additional considerations, are provided below:

a. Commencing on January 1, 1980, the position of Shift Technical Advisor (STA) will be filled in the following manner:

   o During all off-normal hours (i.e., grave and swing shifts, weekends and holidays), San Onofre Units 2 and 3 operations personnel with previous experience at San Onofre Unit 1 will function as the STA. In addition, a Station Engineer with a bachelor's degree knowledgeable in the basic engineering and science subjects will be on-call and available at the station within one hour after reactor trips.

   o During normal working hours or as otherwise required, either a Station Engineer with a bachelor's degree knowledgeable in the basic engineering and science subjects or station personnel holding a senior reactor operating license on San Onofre Unit 1 will function as the STA.

   o Station personnel functioning as STA's or providing on-call assistance will have experience in station design and response.

   Commencing on January 1, 1981, the position of STA will be filled in the following manner:

   o The STA will have a bachelor's degree in a scientific or engineering discipline or will have equivalent education and experience. The STA will have received specific training in the response and analysis of the station for transients and accidents and in station design and layout, including capabilities of instrumentation and controls in the control room.

   o Once San Onofre Units 2 and 3 become operational, the STA will be assigned to function at these units as well and will have specific training in the station design and response of the units.

b. The Onsite Technical Support Center (OTSC) is currently established as described in our aforementioned October 17, 1979 report. However, we cannot provide a direct display of plant parameters by January 1, 1980 since San Onofre Unit 1 does not have a plant process computer and thus cannot utilize a teletype to provide data acquisition in the OTSC.

However, the OTSC is immediately adjacent to the Control Room and contains a viewing window through which some of the plant parameters and system's status can be directly observed. In addition, this close proximity allows ready access to technical information in the form of recording charts, incore thermocouple maps and incore flux data. Procedures will be prepared to allow key OTSC personnel access to the control room for the purpose of obtaining technical information.

If you have any questions, or desire additional information, please contact me.

Very truly yours,

KP Baskin/JH

Enclosures (2)

ENCLOSURE 1

CONTROLS GRADE
AUTOMATIC INITIATION OF AUXILIARY FEEDWATER SYSTEM
AND
AUXILIARY FEEDWATER FLOW INDICATION

Introduction

The existing manually controlled Auxiliary Feedwater (AFW) System will be modified to achieve automatic initiation of the system along with the remote control capability and AFW flow indication in the main control room (MCR) to each of the steam generators.

Flow Path

The AFW flow path that will be modified will consist of electric driven AFW pump supplying flow through a newly added motor operated valve (MOV) installed in the path connecting the AFW pump header to the west main feedwater line which ultimately supplies each of the steam generators. Flow to the steam generators is controlled by the main feedwater line by-pass valves CV-142, CV-143 and CV-144 which can be remotely throttled from the MCR. The attached Figure 1 depicts the flow path to be utilized. An appropriate flow indication system will be added downstream of each of the CV's 142, 143 and 144 to indicate remotely in MCR the flow to the steam generators.

Automatic Initiation Logic

The AFW system will have the capability to be initiated automatically or remote manually from the MCR. Automatic initiation will be based on a low steam generator level signal processed from each of the new level transmitters (LT 450X, LT 451X and LT 452X) associated with each of the steam generators. In order to avoid any spurious and unwanted actuations, the automatic initiation logic will be based on two out of three (2/3) low level signal logic. Remote manual initiation will be accomplished by manual actuation of a control switch in the MCR. Appropriate indication and annunciation will be provided in the MCR when automatic or manual initiation of the AFW system occurs. In addition, remote control capability to operate components of the system from the MCR will be provided. The attached Figure 2 depicts the system operation.

Automatic Initiation Control Panel and Logic Rack

The steam generator level transmitters will supply a 4-20 MA signal to the logic rack installed in MCR area, where a low level signal and a 2/3 logic will be developed. A control board (panel) in the MCR will provide monitoring and manual control capability of the system. If the 2/3 logic is initiated, the operator will be informed through the status indication on the control board; however, no action will be required by the operator to initiate flow since the system will be automatically initiated. Since auxiliary feedwater flow to the steam generators will be controlled by throttling of main feedwater by-pass control valves CV-142, CV-143 and CV-144, annunciation and status indication will be provided alerting the operator that "Auto System is Initiated" and that he should control the flow to the steam generators through these valves. The logic is designed such that once the Automatic Initiation Signal is initiated, it will go through and latch the output which must be manually reset if individual component control is desired. In case the Automatic System does not initiate actuation of the components, system level manual initiation will achieve this function also. At present, valves CV-142, CV-143 and CV-144 are closed on SIS. However, the control circuit for each of these valves will be modified to override SIS if the Auxiliary Feedwater Automatic Signal is initiated during or after the occurrence of SIS. This implies that the control valves will always be available to the operator whenever this system is desired.

If the auxiliary feedwater system was initiated prior to the occurrence of SIS simultaneously with LOP, the system will be tripped so that the emergency diesel generators are placed on line on a dead bus, then SIS related loads are automatically initiated and after 21 seconds of time delay (adjustable), the auxiliary feedwater loads will be placed on the diesel generator and operation resumed automatically. However in case of SIS with no LOP the AFW system will not be disturbed. The logic and the monitoring panel along with the individual components will be powered from the emergency diesel generators.

Flow Indication

Since AFW flow to the steam generators will be controlled by manual throttling of CV-142, CV-143 and CV-144, an appropriate flow measurement system will be provided downstream of these valves so that flow through each of the lines to the steam generators will be indicated in the MCR.

This AFW flow indication will provided with diverse back up steam generator level indicators in the MCR.

The existing manual capability of the AFW System will be retained and in addition new flow measurement devices (FC 2002A, FC 2002B and FC 2002C) will be installed with a flow indication locally and in the MCR so that this system is also available should the automatic initiation of AFW not be operable. By appropriate communication between the MCR operator and an operator at the AFW isolation valves, the manual valves can be controlled to obtain flow to the steam generators through this system also. Figure 1 also depicts this system.

Periodic Testing

The system will be designed to be testable. In order to avoid unnecessary component operation during periodic testing, the operator will be able to disable the initiation logic by means of "Auto Enable/Disable" control switch. Whenever the system is disabled, annunciation will be provided indicating this condition. However, if during testing the Automatic System is initiated, the operator will still be alarmed of such event so that he can either place the "Enable/Disable" control switch in the "Enable" mode and control the AFW flow through the control valves or he will be directly able to operate individual components (namely the motor driven pump and cross connect MOV) from the MCR and again control the AFW flow through the control valves.

ENCLOSURE 2

Information Required On The Subcooling Meter

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	Hot Leg Temp., Pressurizer Pressure, T-Tsat
Display Type (Analog, Digital, CRT)	Analog
Continuous or on Demand	Continuous
Single or Redundant Display	Single
Location of Display	Control Room (J-console)
Alarms (include setpoints)	T-Tsat $\leq$ 50 F
Overall uncertainty ($^{\circ}$F, PSI)	Accuracy of T-Tsat = $\pm$ 1.3% of span
Range of Display	Hot Leg Temp. 450 to 700°F Pressurizer Pressure 400 to 2400 psi T-Tsat -100 to +100°F
Qualifications (seismic, environmental, IEEE323)	See Note 1

Calculator

Type (process computer, dedicated digital or analog calc.)	Dedicated analog calculator
If process computer is used specify availability. (% of time)	N.A.
Single or redundant calculators	Single
Selection Logic (highest T., lowest press)	Manual - any sensor
Qualifications (seismic, environmental, IEEE323)	See Note 1
Calculational Technique (Steam Tables, Functional Fit, ranges)	Calculator duplicates Steam table Tsat curve within $\pm$0.5% over range 450 to 700°F

Input

Temperature (RTD's or T/C's)	RTD's
Temperature (number of sensors and locations)	Three, one per loop on hot leg
Range of temperature sensors	32-1000^{o}F
Uncertainty of temperature sensors (^{o}F at 1)	Accuracy is $\pm$.25% of span
Qualifications (seismic, environmental, IEEE323)	Qualified commensurate with equipment to which it is attached
Pressure (specify instrument used)	Foxboro Ell Series
Pressure (number of sensors and locations)	One, pressurizer
Range of pressure sensors	0 to 3000 psig
Uncertainty of pressure sensors (PSI at 1)	Accuracy is 1/2% of span
Qualifications (seismic, environmental, IEEE323)	Qualified commensurate with equipment to which it is attached

Backup Capability

Availability of Temp. & Press.	Incore thermocouples, three cold leg RTD's, three Taug indicators and four pressurizer pressure indicators other than those used for the saturation recorder are available
Availability of Steam Tables etc.	Yes, control room
Training of Operators	Yes
Procedures	Yes

NOTE 1

Qualifications of the Display and Calculator

These items are located in the control room and are not subject to adverse environmental conditions. The display recorder is a Foxboro Spec 200 Series 220S similar to fully qualified safety grade units utilized on San Onofre Units 2 and 3. The calculator utilizes high quality, reliable components (e.g. $\pm$ 1% resistors). All components are installed in safety related consoles and instrument racks which meet the original seismic design criteria for the plant.

LEGEND

(M)	MOTOR-OPERATED OPEN/CLOSE VALVE
[S]	SOLENOID-OPERATED OPEN/CLOSE VALVE
A.I.S.	AUTOMATIC INITIATION SIGNAL
MCR	MAIN CONTROL ROOM
*	NEW EQUIPMENT

SAFETY NOTES:

1. IT IS PLANNED TO INSTALL NEW LEVEL TRANSMITTER PER STEAM GENERATOR FOR AUTOMATIC INITIATION OF AUX. FEED WATER SYSTEM.
2. THE LEVEL INDICATION WILL BE DISPLAYED IN THE MAIN CONTROL, AS A BACK UP TO THE AUX. FEED WATER FLOW INDICATION.
3. THE INSTRUMENTATION BEING ADDED FOR EACH STEAM GENERATOR MAIN FEEDWATER BYPASS FLOW AND AUXILIARY FEEDWATER LINE FLOW SHALL BE PHYSICALLY AND ELECTRICALLY INDEPENDENT SO AS TO AVOID A COMMON MODE FAILURE BY HAVING POWER SUPPLIED FROM SEPARATE VITAL BUSES AS SHOWN.
4. THE LOGIC RACK AND PANEL INSTRUMENTATION POWER SUPPLY SHALL BE FROM VITAL BUS #1.

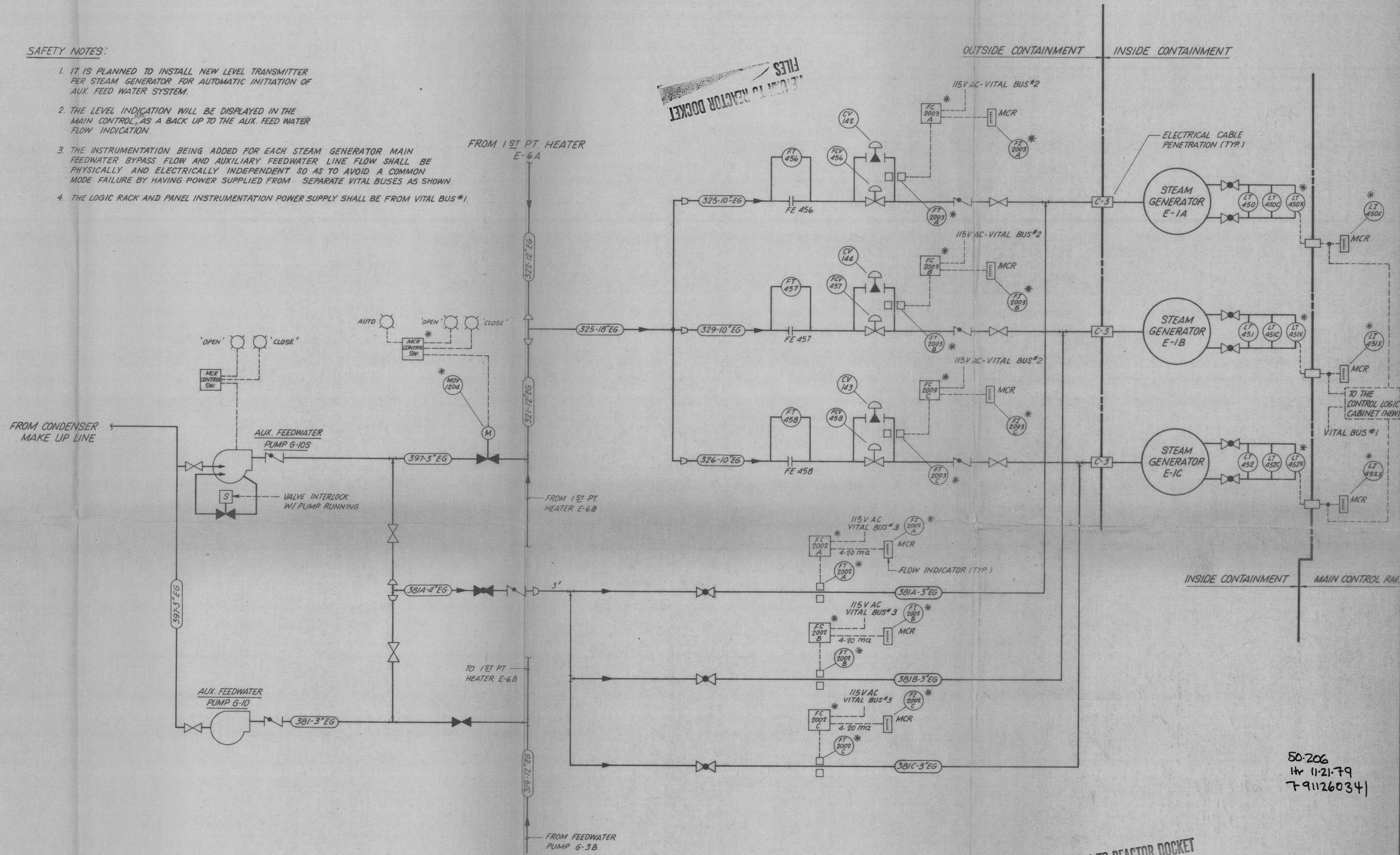

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FIGURE 1

TEMPORARY SAFETY RELATED TMI

Reference Drawings		Reference Drawings	
451283	CONTROL BLOCK DIAGRAM		
5159696	FUNCTIONAL SCH. (G10S & MOV1204)		
M-33753	LEVEL INDICATOR		
M-33702	FLOW TRANS/CONTROLLER		
M-33704	CONTROL LOOP DIAGRAM	M-33761	LEVEL TRANSMITTER
568779	P & I DIAGRAM-FEEDWATER & COND.	M-33764	CONTROL SWITCH
M-33754	REMOTE FLOW INDICATOR	M-33906	INDICATOR LIGHT

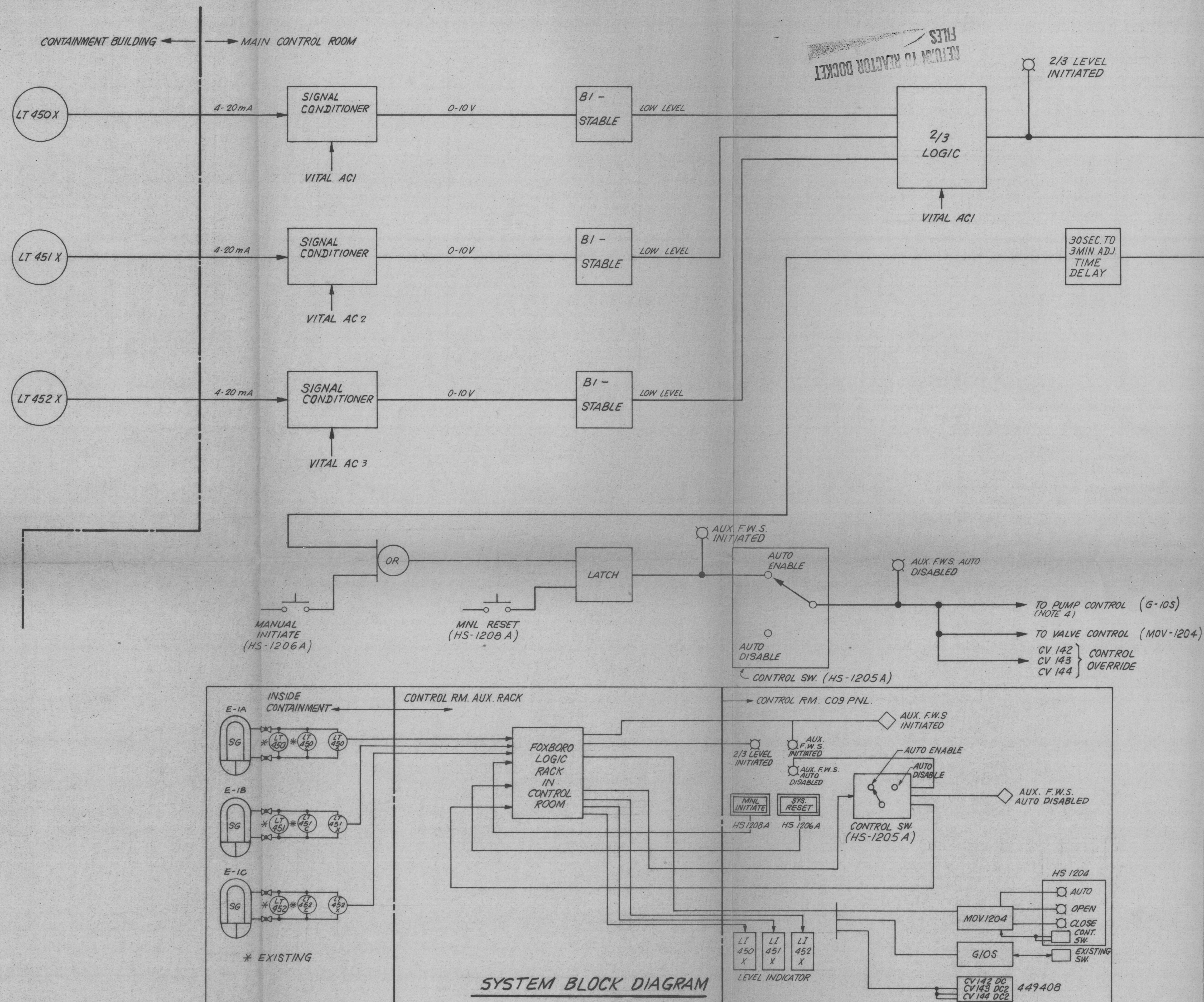
No.	Revisions	M	Date	P.E.	Q.A.E.	Disc Supv.	Approved	Resp Engr.	Ck'd.	Made	J.O. No.
0	ISSUED FOR CONSTRUCTION		11/9/79							AS	7764

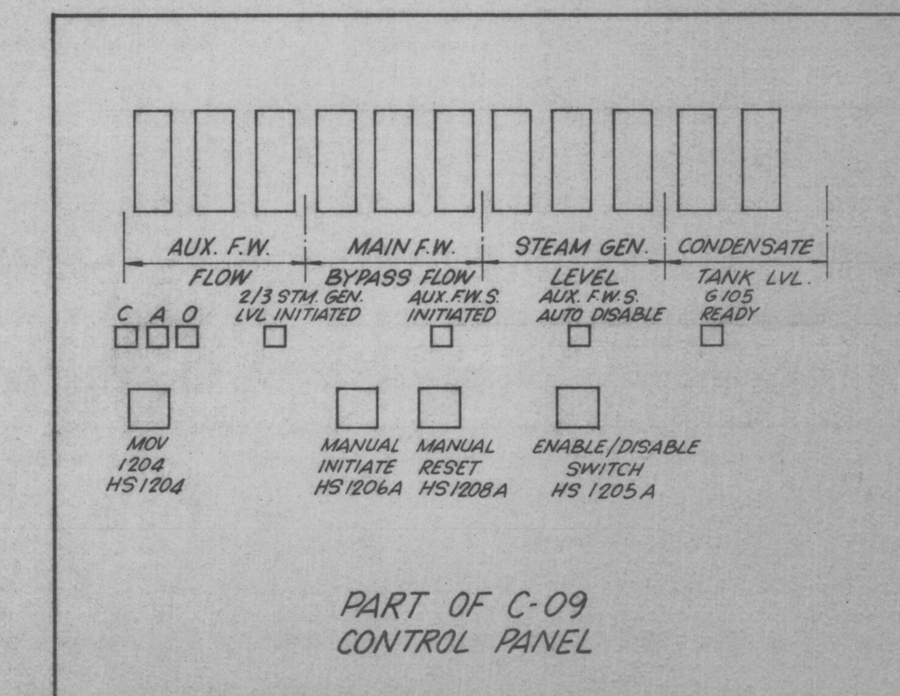
Location SAN ONOFRE NUC. GEN. STA. UNIT 1

CONTROL BLOCK DIAGRAM
AUXILIARY FEEDWATER
AUTOMATIC INITIATION

SCE EDISON Rosemead California

Scale:
D/L: 451283-0 C





SAFETY NOTES:

1. IF THE SYSTEM WAS RUNNING BEFORE SAFETY INJECTION OCCURS CONCURRENT WITH LOSS OF OFF SITE POWER (SIS/LOP) THEN DURRING DIESEL GENERATOR LOADING, PUMP & VALVE CONTROL WILL BE BY PASSED UNTIL SIS RELATED LOADS ARE ON LINE & THEN THE SYSTEM WILL BE AUTOMATICLY RE-INITIATED IF DEMAND FOR THE SYSTEM EXISTS.

2. MANUAL SYSTEM INITIATION WILL BE AVAILABLE BEFORE AUTO INITIATION AND AFTER THE AUTO INITIATION IF THE SYSTEM WAS RESET.

3.

4. IN THE EVENT OF LOP, THE PUMP AND MOV MUST NOT ENERGIZE UNTIL AFTER THE LAST LOAD HAS BEEN ADDED BY SEQUENCER. (SEE DRAWING NUMBER 5159696 FUNCTIONAL SCHEMATIC)
NOTE: UNDER VOLTAGE RELAY CONTACTS AVAILABLE IN PUMP MOTOR SWGR. SUFFICIENTLY ACHIEVE THE FUNCTION WITHOUT ANY SEQUENCER INTERFACE.

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FIGURE 2

TEMPORARY
SAFETY RELATED
TMI

SAFETY RELATED

Reference Drawings	
451283	CONTROL BLOCK DIAGRAM
5159696	FUNCTIONAL SCHEMATIC (G10S & MOV 1204)
M-33761	LEVEL TRANSMITTER
M-33753	LEVEL INDICATORS
M-33906	INDICATOR LIGHT
M-33764	CONTROL SWITCH

No.	Revisions	M	Date	Approved	J.O. No.
0	ISSUED FOR CONSTRUCTION		11/19/79		7764

Location SAN ONOFRE NUCLER GEN. STA. UNIT 1

LOGIC DIAGRAM
AUXILIARY FEEDWATER
AUTOMATIC INITIATION

SCE EDISON Rosemead California

D/L: 451285-0 C

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