SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

October 13, 1982

DIRECTOR OF NUCLEAR REACTOR REGULATION
ATTENTION JOHN F STOLZ CHIEF
OPERATING REACTORS BRANCH 4
U S NUCLEAR REGULATORY COMMISSION
WASHINGTON DC 20555

DOCKET 50-312
RANCHO SECO NUCLEAR GENERATING
STATION UNIT NO 1
NUREG 0737 ITEM II.B.3. POST ACCIDENT SAMPLING SYSTEM

Your letter dated July 12, 1982, requested information on the Post Accident Sampling System for Rancho Seco Unit 1, (NUREG 0737, Item II.B.3). Attached is our response to your information request.

John J Mattimoe

John J. Mattimoe
General Manager

Attachment

A046

ATTACHMENT NO. 1

POST ACCIDENT SAMPLING SYSTEM
NUREG-0737 ITEM II.B.3
DESCRIPTION OF COMPLIANCE

ITEM NO. 1 : Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of offsite power (i.e., designate an alternative backup power source, not necessarily the vital (Class 1E) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

Response : The Rancho Seco Post Accident Sampling System (PASS) will have the capability to obtain and analyze samples of the reactor coolant system, reactor building atmosphere and reactor building emergency sump within three (3) hours from the time a decision to take a sample is made.

As shown on the attached drawings, the reactor coolant sample will be obtained via the pressurizer liquid sample line by way of a tie to a cold leg loop drain. The PASS sample is diverted from the normal path, in the decay heat cooler room, -20 feet elevation. The reactor building emergency sump sample is taken directly from the decay heat removal system (low pressure injection system) sample line and is diverted from its normal path. The sample line then ties into the RCS sample in the decay heat cooler room to form a single liquid sample line. The reactor building atmosphere sample will be obtained by a tie into the dedicated Post Accident Hydrogen Monitoring System also in the decay heat cooler room. From the decay heat cooler room on the -20 feet elevation of the auxiliary building all sample supply and disposal lines follow a common path of approximately 200 feet to the grade elevation to a remotely controlled sample collection and analysis station (SCAS).

Samples will be collected by relying primarily on system pressure as the driving force, if system pressure is inadequate, two (2) small pumps internal to the SCAS can be utilized to obtain the required sample.

The samples will be taken one at a time as necessary. Each sample will be flushed through the SCAS to provide a representative sample. Liquid waste will be disposed of by discharge to the reactor coolant drain tank (RCDT). Reactor building atmosphere and gaseous waste will be returned to the reactor building.

By calculations using bounding conditions of pressures, temperatures, and flows, including analytical times it has been shown the PASS is capable of providing the required analysis well within the three hour time frame.

Currently, no provisions have been made to provide backup power, as this is not a design criteria within NUREG 0737, Item II.B.3. The feasibility of providing backup as applicable to a nonclass 1E system is currently being investigated.

ITEM NO. 2 : (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation to minimize personnel radiation exposures (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and nonvolatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (See Vol. II, Part 2, pp. 524 - 527 of Rogovin Report for further information).
2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

(b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

(c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

(d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected online instrument is appropriate for this application. (See (8) and (10) below relative to backup grab sample capability and instrument range and accuracy).

Response : (a) As stated in Item 1 above, the PASS is a remotely controlled system capable to complete inline remote monitoring and analysis. This system utilizes time, distance and shielding as the primary means of reducing personnel exposure. In addition, once an aliquot has been collected for analysis the remainder of the system is flushed with demineralized water to limit the inplant dose rate. Procedures are currently being prepared for the operation of the system to obtain the necessary analyses. These procedures will be complete prior to startup after the next refueling of Rancho Seco and will contain guidance for determining the extent of core damage.

(b) Grab samples are capable of being collected under any desired condition, i.e., pressurized and hot, diluted, etc. The methods of handling, transporting and analyzing are currently being developed.

(c) The system is capable of analyzing all species listed in NUREG 0737 as well as RG 1.97 Rev. 2, with the exception of ECCS pump room sumps and oxygen percentage in the containment atmosphere.

(d) All inline analytical equipment is of the best available quality and reliability. The equipment used is designed for the specific

application and is commonly used throughout industry for the same analytical purposes.

ITEM NO. 3 : System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Response : As shown in the attached drawings, an isolated system, e.g., the pressurizer sample line,will be used. The safety feature's valves are controlled from the control room, thereby providing positive operator control. This line was used because it provides the desired containment penetration; is of small size,3/4 inch; and does not carry fluid outside of the decay heat cooler rooms, a high radiation area.

ITEM NO. 4 : Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is less than 0.1 ppm by measurement of a dissolved hydrogen residual of less than or equal to 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Response : Total dissolved gases will be measured using expansion of the pressurized coolant sample between two known volumes and measurement of pressure changes (Henry's Law). We anticipate that chlorides will not exceed 0.15 ppm; however, grab samples should be adequate to check dissolved oxygen.

ITEM NO. 5 : BWR's on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers, (e.g., shutdown cooling), that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform a chloride analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ___ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system; and (2) that dissolved oxygen can be verified at less than 0.1 ppm, consistent with the guidelines above in Clarification No. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Response : Chloride concentration will be measured using an Ion Chromatograph using chromatographic comparisons with standard solutions. Additionally, grab samples will be available for laboratory analysis. Analysis time will be well within the required 96 hours by either method.

ITEM NO. 6 : Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on personmotion for sampling,transport,and analysis of all required parameters.

Response : The system is designed to maintain personnel exposure ALARA, and within the limits of GDC 19. As stated previously the system utilizes remote operation, time, distance shielding and dilution to minimize personnel exposure. Worst case analysis based on RG 1.4 source terms would give an exposure to a single operator of 900 millirem for the first sample taken at the time of the accident.

ITEM NO. 7 : PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Response : Boron analysis in the range of 0 to 6000 ppm is performed by the use of an Ion Chromatograph. Boron concentration is determined by chromatographic comparison to a known standard.

ITEM NO. 8 : A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an offsite laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Response : Grab sample collection flasks are provided to allow the acquisition of any type of sample conditions required to perform the desired analysis. As stated previously, methods are still under development for the handling, transport and analysis of the grab samples.

ITEM NO. 9 : (a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident sampling capabilities.

(b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

Response : (a) The sampling and analysis system is capable of infinite dilution and can be adjusted to achieve the necessary sensitivity. Because of this capability, the system can perform radioisotope analysis in the range from normal operations up to and including those concentrations corresponding to RG 1.4 source terms.

(b) The background radiation levels in the area of the Sample Collection and Analysis Station are predicted to be approximately 100 millirem per hour, and 25 millirem per hour at the Remote Control Panel. Analytical equipment was selected, assembled and protected to operate sufficiently under these conditions. The presence of a "hot" sample would affect the operability of certain equipment but the sample will be diluted prior to contact with this equipment.

Background levels in the analytical lab and counting room are predicted to be less than 10 millirem per hour. See also our response to Item 10.

ITEM NO. 10 : To demonstrate that the selected procedures and instrumentation will achieve the required accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing a standard test matrix or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

Response : As stated previously, the sensitivity and accuracy are a function of the dilution ratio.

For gamma spectroscopy the accuracy will be a function of dilution, at the predicted 1000 to 1 dilution to meet the sensitivity of the Intrinsic Germanium detector the error is estimated to be near the factor of two.

The accuracy of Boron and Chloride analysis using the Ion Chromatograph since it is a comparison to a known standard by calculating the area under the curve plotted by the recorder is estimated to be within ±5% taking into consideration dilution error, standard solution error and plotter error.

Total dissolved gases are measured by controlled volume expansion and pressure change comparison, the error will depend on the accuracy of the volume and reading of pressure indication. The exact volume of the "Controlled" volume will have to be measured after fabrication to accurately determined the error contribution. It is expected that the total error will be within the recommended ranges.

The pH meter used to determine pH will provide an accuracy ±1% over the entire range.

Once the system is installed a test matrix similar to that listed will be used to verify the System Performance Operator (technician). Training will be conducted prior to startup after the next refueling to provide familiarity and efficiency in using the system.

ITEM NO. 11 : (a) A description of the provisions which address each of the items in clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e., sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

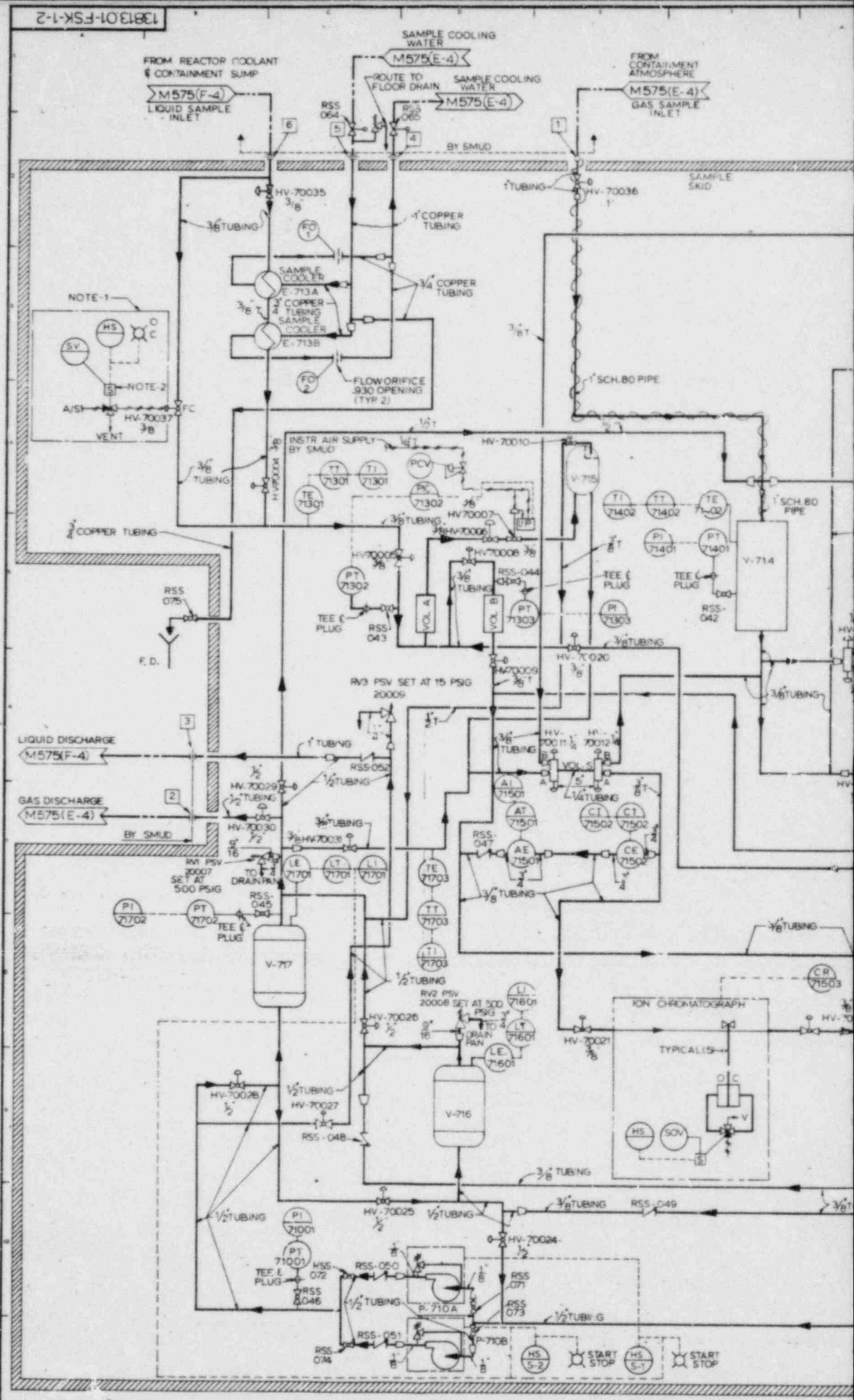
(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

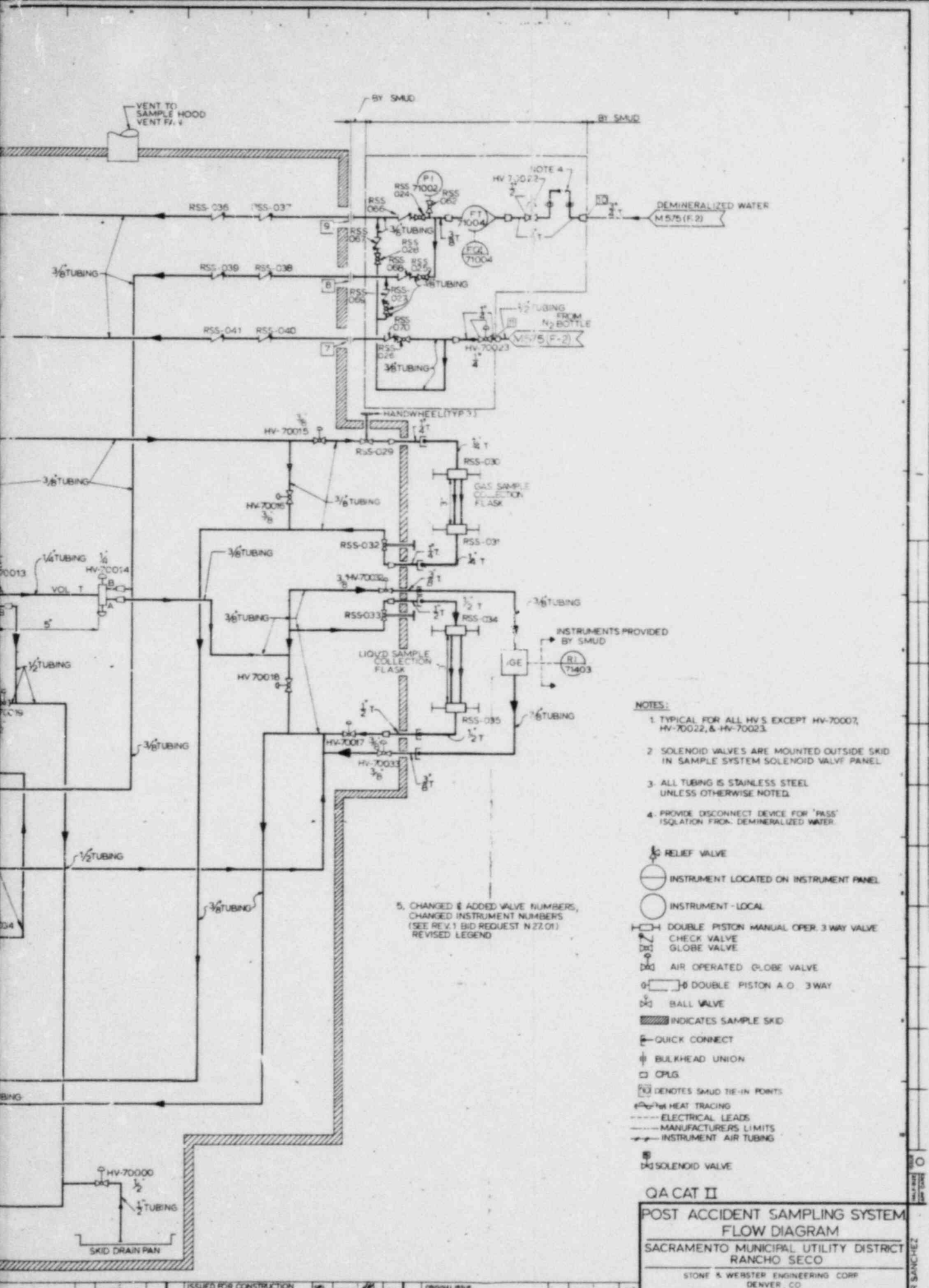
Response : (a) The system is designed for the protection of associated systems by segregation by multiple check valves. A flow passive restrictor has been provided on the reactor coolant sample tie-in to limit flow under a possible line break accident. Additionally where applicable, relief valves have been provided to protect PASS system piping from overpressurization.

The systems are provided with demineralized water flush connections to limit in plant radiation levels, mud buildup sample distortion and loose materials. The sump sample is provided with a strainer to prevent blockage due to possible loose materials in the containment sump. The containment atmosphere sample line is being heat traced to limit iodine plate out.

Waste fluids are to be returned to the containment (gases) or collected in the reactor coolant drain tank (liquid).

(b) To provide complete system protection and limit the buildup of radiation in the area of sampling station the SCAS is a totally sealed enclosure provided with a drain sump should any internal valve leaks occur. Fluids in this sump will automatically drain when disposing of waste fluids. Additionally, the enclosure is ventilated to the Auxiliary Building radwaste ventilation system which is filtered by charcoal and a HEPA filter system.


13813O1-FSK-1-2
SAMPLE COOLING WATER
M575(E-4)
FROM REACTOR COOLANT & CONTAINMENT SUMP
M575(F-4)
LIQUID SAMPLE - INLET
ROUTE TO FLOOR DRAIN
SAMPLE COOLING WATER
M575(E-4)
FROM CONTAINMENT ATMOSPHERE
M575(E-4)
GAS SAMPLE INLET
BY SMUD
SAMPLE SKID
HV-70035
HV-70036
SAMPLE COOLER E-713A
SAMPLE COOLER E-713B
FLOW ORIFICE .930 OPENING (TYP 2)
NOTE-1
NOTE-2
HV-70037
VENT
INSTR AIR SUPPLY BY SMUD
HV-70010
V-715
V-714
HV-70007
HV-70006
HV-70005
HV-70008
HV-70020
HV-70009
RV3 PSV SET AT 15 PSIG 20009
LIQUID DISCHARGE
M575(F-4)
GAS DISCHARGE
M575(E-4)
HV-70029
HV-70030
HV-70031
HV-70011
HV-70012
RV1 PSV 20007 SET AT 500 PSIG
DRAIN PAN
V-717
RV2 PSV 20008 SET AT 500 PSIG
HV-70026
HV-70028
HV-70027
V-716
ION CHROMATOGRAPH
TYPICAL (5)
HV-70021
HV-70025
HV-70024
P-710A
P-710B
START
STOP
SCH.80 PIPE
COPPER TUBING
TUBING
TEE & PLUG
F.D.



NOTES:

1. TYPICAL FOR ALL HV'S EXCEPT HV-70007, HV-70022, & HV-70023.
2. SOLENOID VALVES ARE MOUNTED OUTSIDE SKID IN SAMPLE SYSTEM SOLENOID VALVE PANEL.
3. ALL TUBING IS STAINLESS STEEL UNLESS OTHERWISE NOTED.
4. PROVIDE DISCONNECT DEVICE FOR 'PASS' ISOLATION FROM DEMINERALIZED WATER.
5. CHANGED & ADDED VALVE NUMBERS, CHANGED INSTRUMENT NUMBERS (SEE REV.1 BID REQUEST N.22.01) REVISED LEGEND

- RELIEF VALVE
- INSTRUMENT LOCATED ON INSTRUMENT PANEL
- INSTRUMENT - LOCAL
- DOUBLE PISTON MANUAL OPER. 3 WAY VALVE
- CHECK VALVE
- GLOBE VALVE
- AIR OPERATED GLOBE VALVE
- DOUBLE PISTON A.O. 3 WAY
- BALL VALVE
- INDICATES SAMPLE SKID
- QUICK CONNECT
- BULKHEAD UNION
- CPLG.
- DENOTES SMUD TIE-IN POINTS
- HEAT TRACING
- ELECTRICAL LEADS
- MANUFACTURERS LIMITS
- INSTRUMENT AIR TUBING
- SOLENOID VALVE

QA CAT II

POST ACCIDENT SAMPLING SYSTEM FLOW DIAGRAM

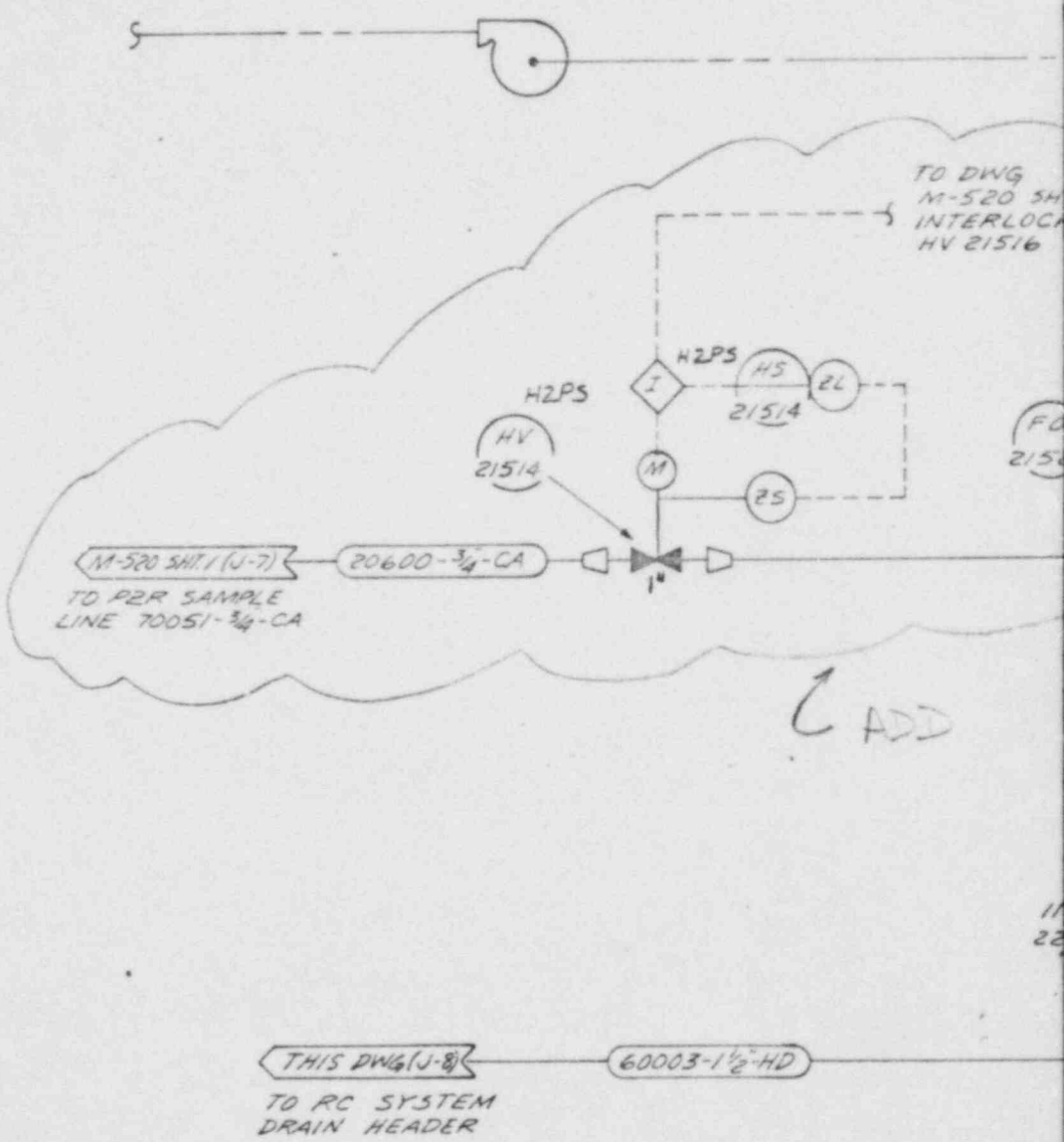
SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO

STONE & WEBSTER ENGINEERING CORP
DENVER, CO

DRAWING NUMBER 13813.01-FSK-1-2

REV	DESCRIPTION	DATE
2	ISSUED FOR CONSTRUCTION SEE NOTE 5	
1	ISSUE FOR BIDS	

DESIGNED BY R.SANCHEZ
DRAWN BY SCHRAMPFLO
CHKD R.BEACH
CHKD ALVAREZ



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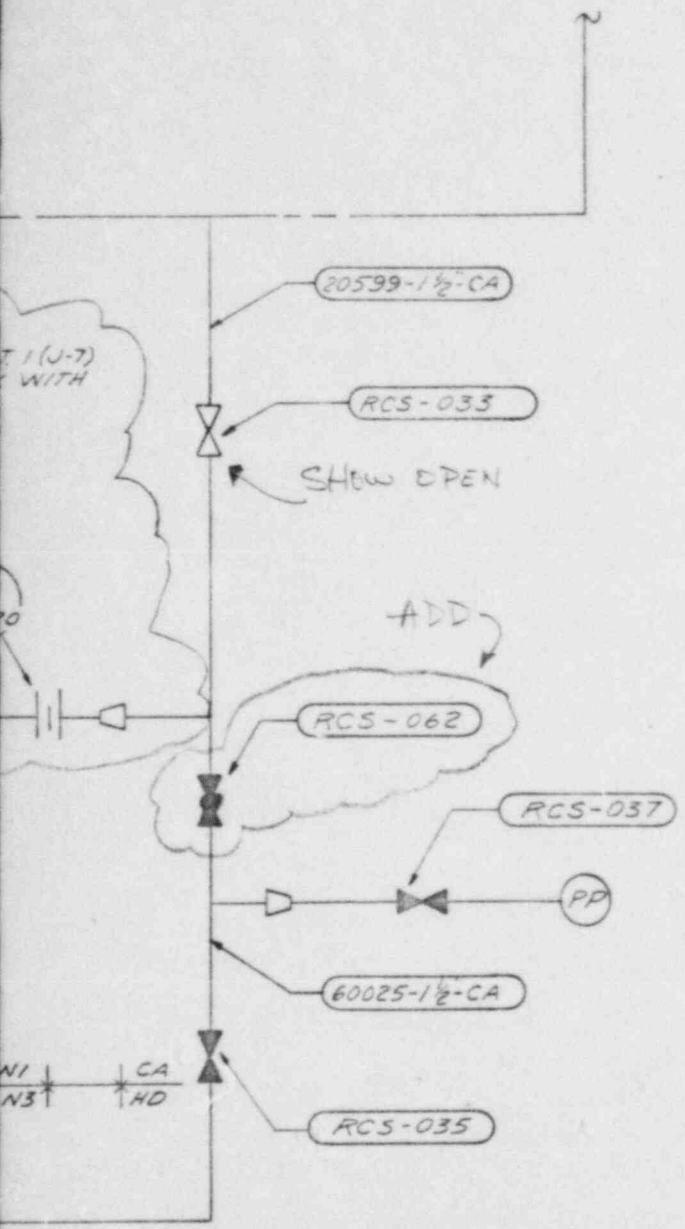
	OR REASON FOR CHANGE: TASK 561	YES ☐ NO ☑	DESIGN ENGINEER:	
	NCR NO. BECHTEL		SUPERVISING ENGINEER:	



FOR INFORMATION ONLY

NO.	DATE	REVISIONS	DR.	CHK	DGS	ENG GS	CHF ENG	PROJ ENG

BECHTEL
LOS ANGELES

SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO NUCLEAR STATION

DCN NO.	DRAWING NUMBER	DRAWING SHT. NO.	DRAWING REV.	APERTURE CARD NO.
	M-520	2	0	NEW

FOR ECC USE ONLY
DWG. REV. BY _____ DATE _____
DCN
SHEET 2 OF 2

SMUD
APPROVED | DATE
SMUD DRAWING NO.

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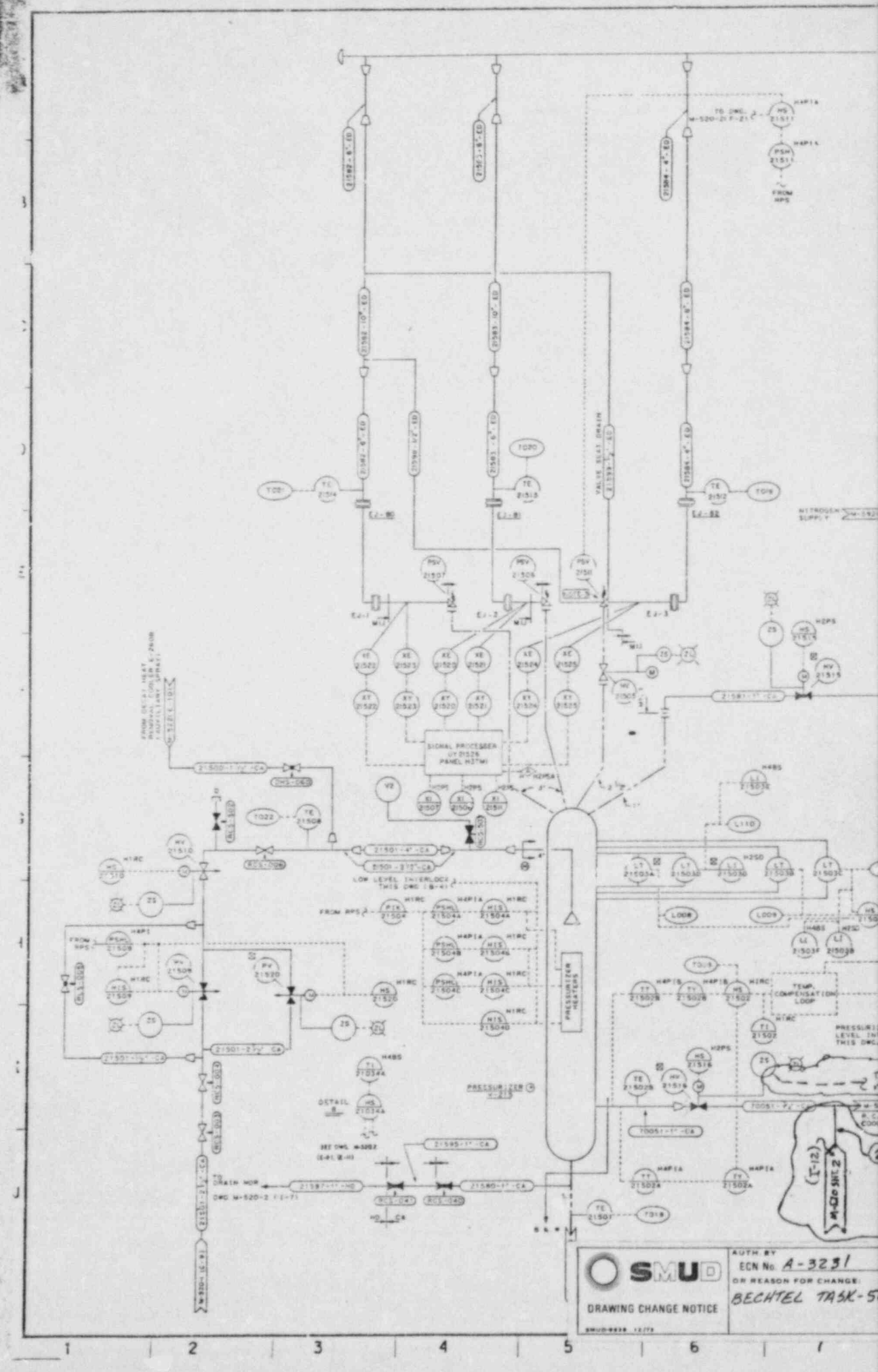
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AUTH BY
ECN No. A-3231
OR REASON FOR CHANGE:
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SIGNAL PROCESSER
PRESSURIZER HEATERS
TEMP. COMPENSATION LOOP
DETAIL B
PRESSURIZER
LOW LEVEL INTERLOCK
FROM RPS
TO DWG.
NITROGEN SUPPLY
VALVE SEAT DRAIN
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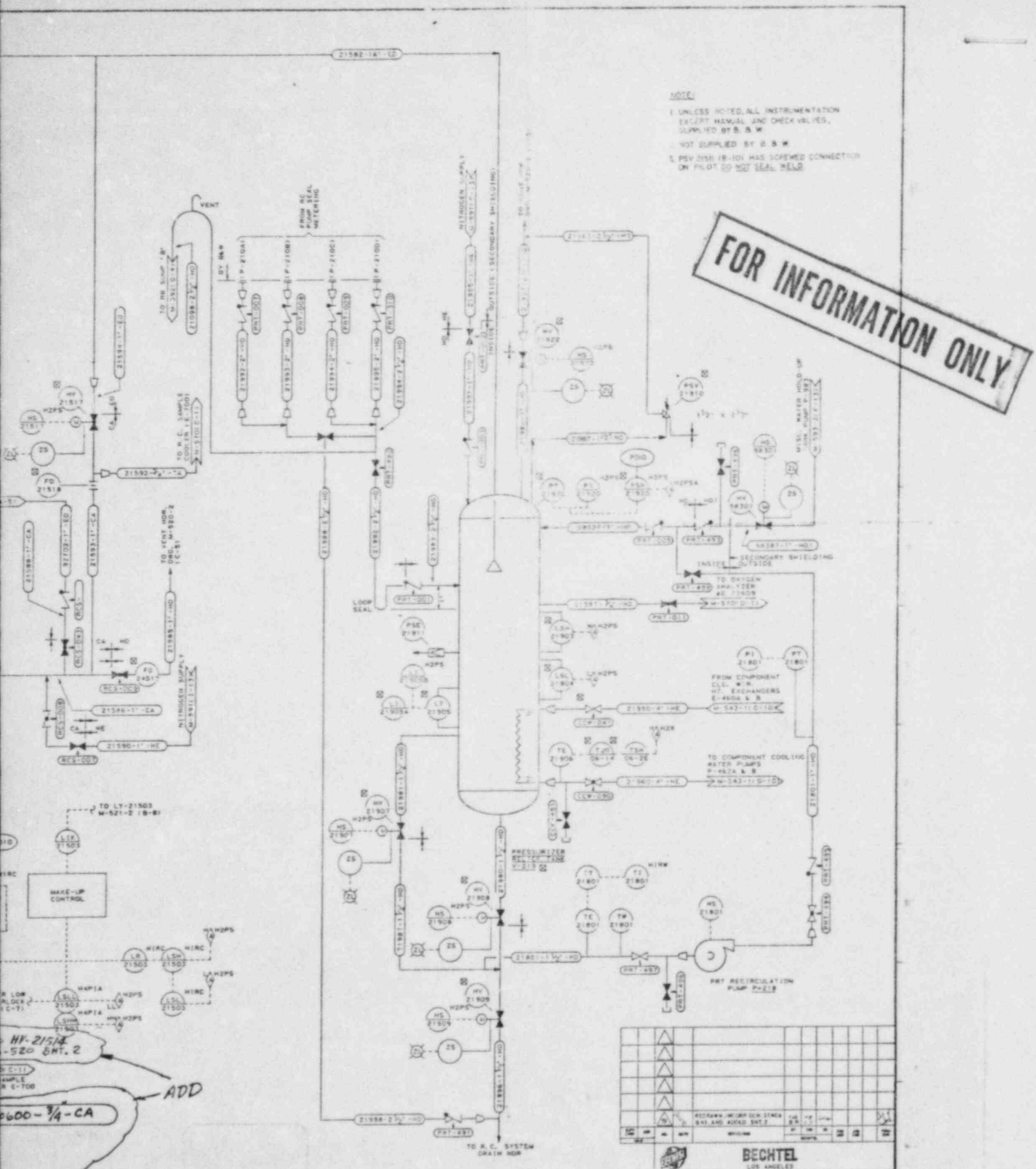
3
4
5
6



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BECHTEL
LOS ANGELES

SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO NUCLEAR STATION

P & I DIAGRAM
REACTOR COOLANT SYSTEM

UNIT	DRAWING NO.	REV
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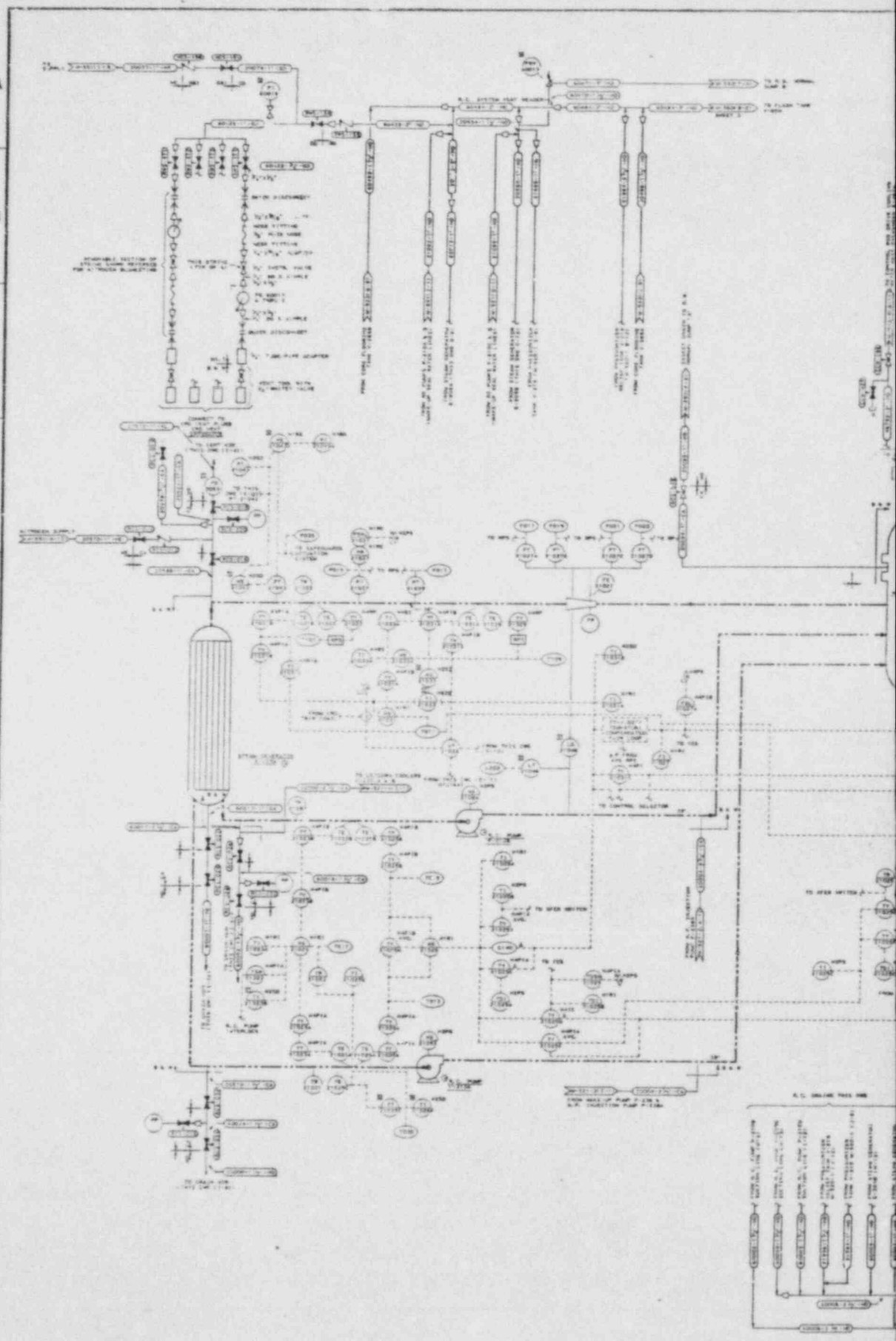

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AUTH. BY ECN No. A-3231

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NCN NO.

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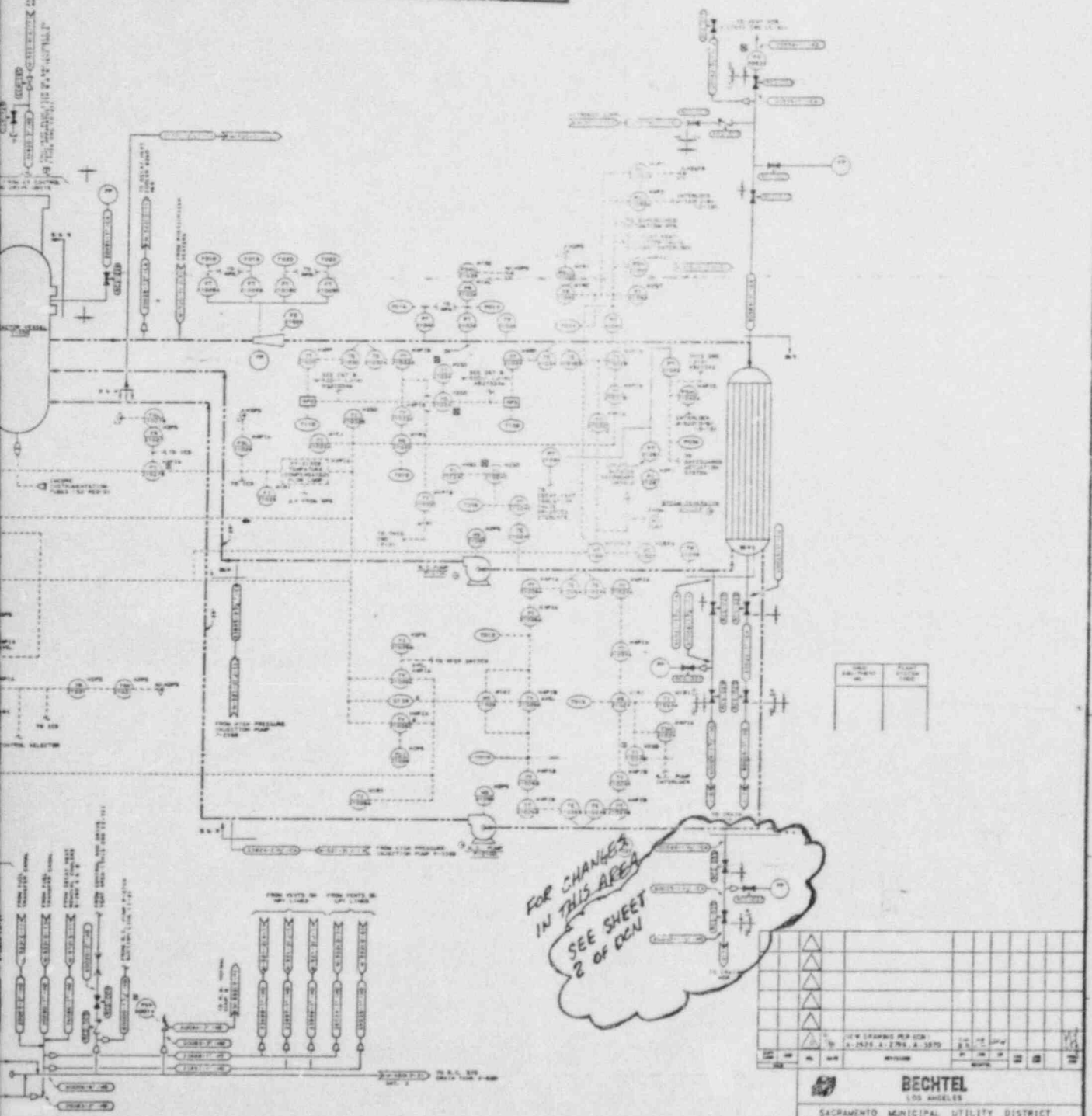

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YES ☐ NO ☑	D.P.	FB 3/5/82		M-520	2	0	NEW		

DESIGN ENGINEER: Orlando [illegible]

SUPERVISING ENGINEER: [illegible] 3/10/82

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DCN SHEET 1 OF 2

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BECHTEL
LOS ANGELES

SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO NUCLEAR STATION

P & I DIAGRAM
REACTOR COOLANT SYSTEM

UNIT	JOB NO.	DRAWING NO.	REV.
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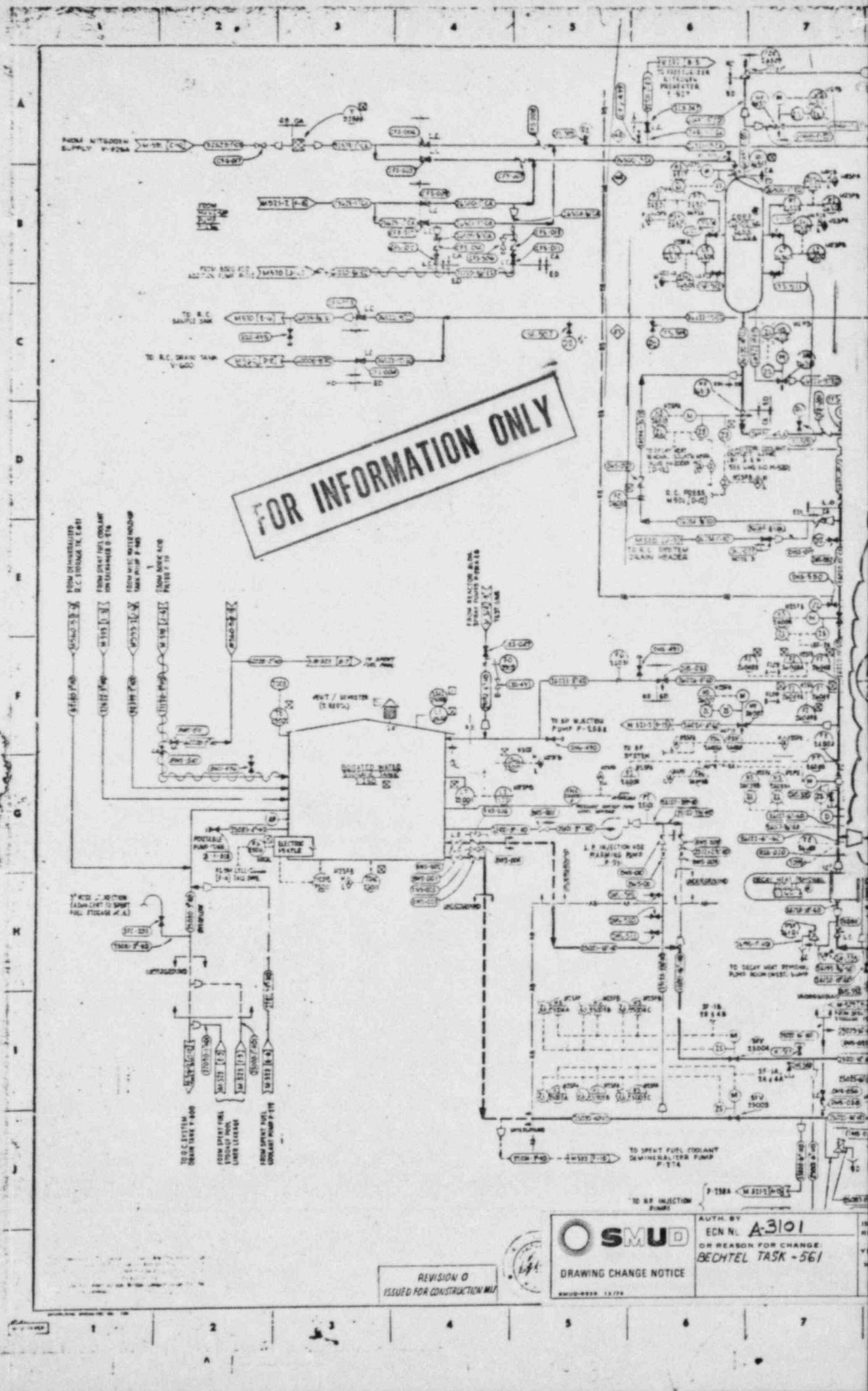
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AUTH. BY
ECN No. A-3101
OR REASON FOR CHANGE
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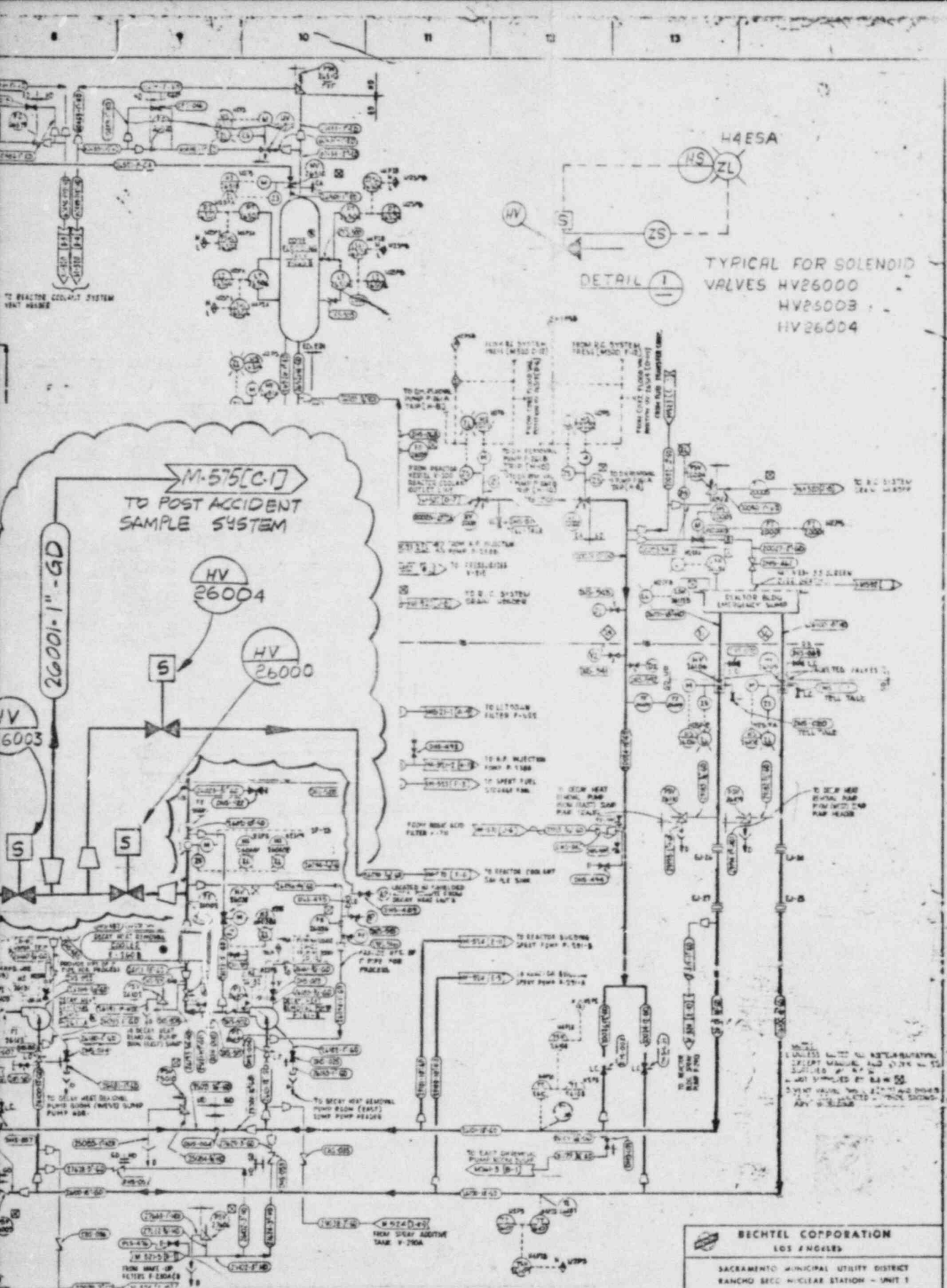
REVISION 0
ISSUED FOR CONSTRUCTION MLP



PREPARED BY GARY SELCER	CHKD BY	DCN NO.	DRAWING NUMBER M-522	DRAWING SHT No. REV. 18	APERTURE CARD NO. ABB281
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BECHTEL CORPORATION
LOS ANGELES

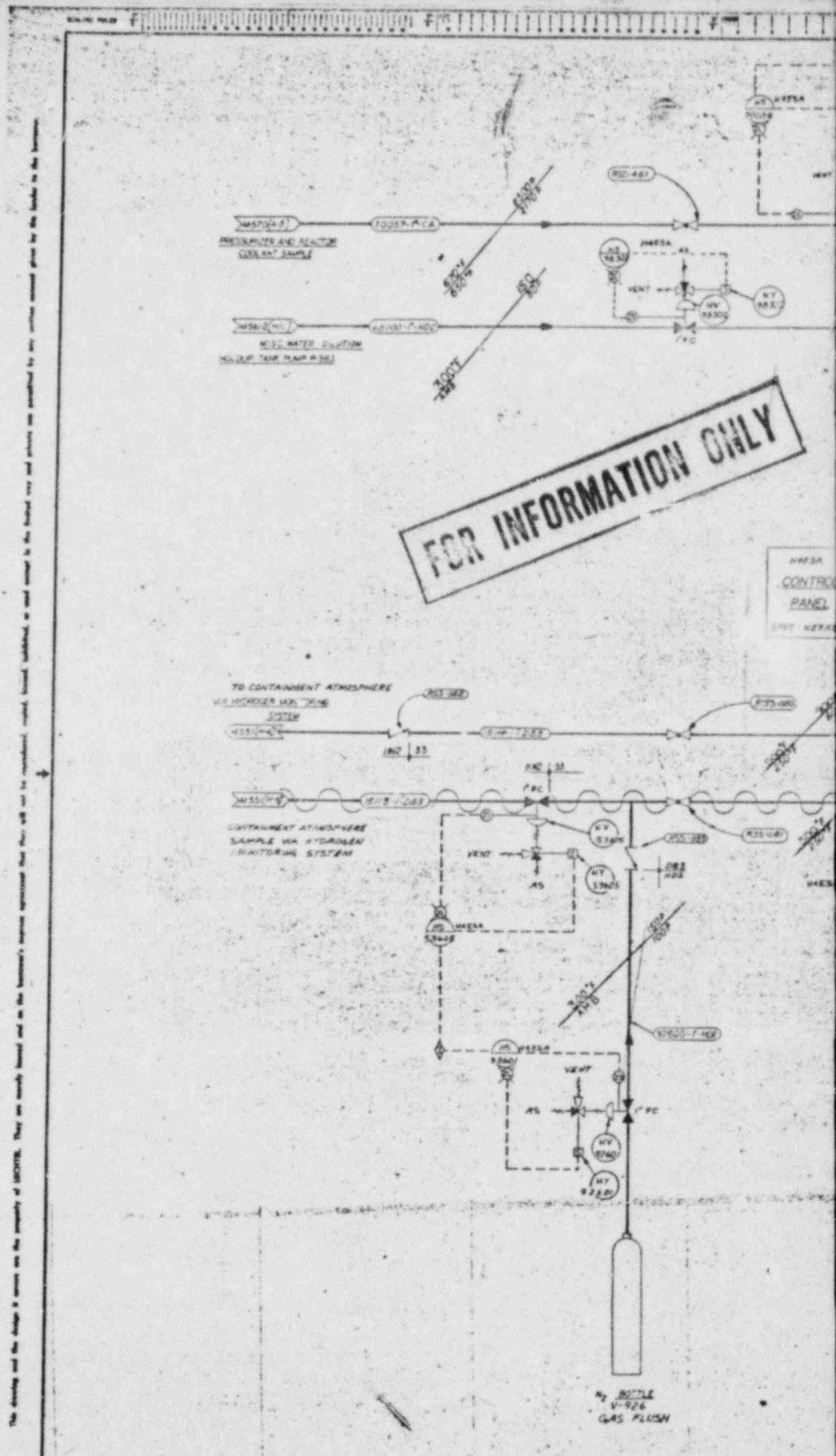
SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO NUCLEAR STATION - UNIT 1

P & I DIAGRAM
DECAY HEAT REMOVAL SYSTEM

SMUD

ABB281

M-522



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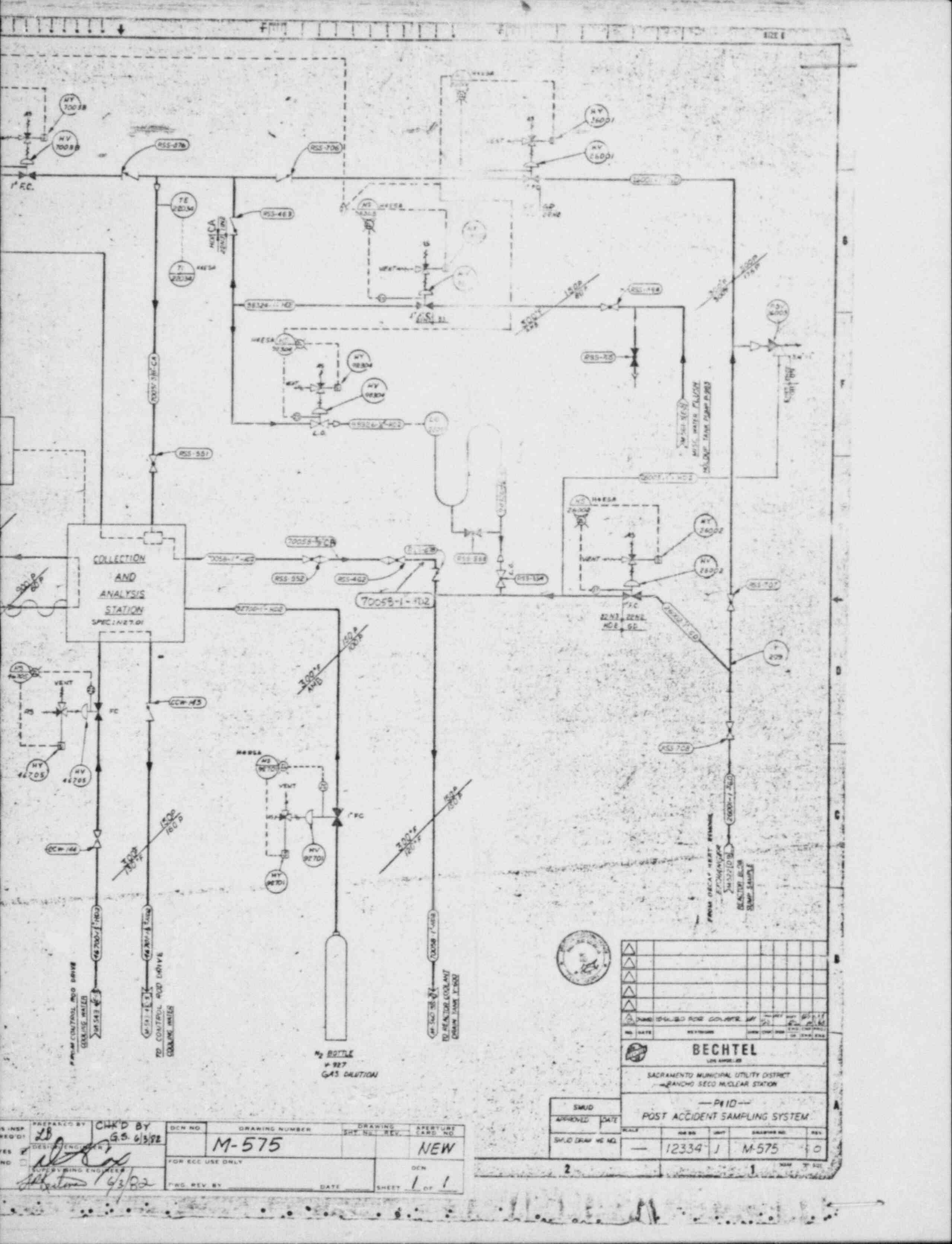

COLLECTION AND ANALYSIS STATION
SPEC 12127.01
N2 BOTTLE
V-927
GAS DILUTION
TO CONTROL ROD DRIVE COOLING WATER
FROM CONTROL ROD DRIVE COOLING WATER
TO REACTOR COOLANT DRAIN TANK T-600
MISC. WATER FLUSH
FROM DECAY HEAT REMOVAL EXCHANGER
REACTOR BLDG SUMP SAMPLE
70058-1-1D2
VENT
BECHTEL
LOS ANGELES
SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO NUCLEAR STATION
— P&ID —
POST ACCIDENT SAMPLING SYSTEM
SMUD
APPROVED
DATE
SMUD DRAW. NO.
12334
J
M-575
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PREPARED BY
CHK'D BY
G.S. 6/3/82
DESIGN ENGINEER
SUPERVISING ENGINEER
6/3/82
DCN NO.
DRAWING NUMBER
M-575
DRAWING SHT. NO.
REV.
APERTURE CARD NO.
NEW
FOR ECC USE ONLY
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DATE
SHEET 1 OF 1