

Attachment 9

C-E POST TMI EVALUATION TASK 2

CONCEPTUAL DESIGN FOR A

REACTOR VESSEL LEVEL MONITORING

SYSTEM

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ABSTRACT

In March 1979 an incident occurred at the Three Mile Island Unit 2 Nuclear Power Plant which resulted in considerable damage to the reactor core. As a result of investigations into the cause of this incident, areas where improvements in nuclear power plant administration, design and operation are necessary have been identified. With respect to nuclear power plant design, the Nuclear Regulatory Commission (NRC) has requested licensees to evaluate the use of a reactor vessel level measurement system as a means of providing additional useful information to power plant operators or as a means for directly determining the adequacy of core cooling.

Subsequently, the C-E Owners' Group authorized Combustion Engineering to develop the functional requirements and a conceptual design for a system to monitor reactor vessel level.

This report provides the result of the C-E investigation of reactor vessel level monitoring system designs. The report includes a description of the design basis and functional requirements for a Reactor Vessel Level Monitoring System (RVLMS); a description of the designs and design configuration recommended by C-E; and a brief discussion of other designs that were evaluated.

The recommended design will not require major plant structural changes and has a high probability of being installed by January 1, 1981.

In developing the functional requirements and the conceptual design recommended in this report, it was assumed that the RVLMS is technically required to provide a valid indication of reactor vessel level or the adequacy of core cooling. This is an assumption upon which the design is based - analyses have not been performed to substantiate such an assumption.

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1.0

PURPOSE

The purpose of this report is to document C-E's research and to recommend a Reactor Vessel Level Monitoring System Design. The design effort was requested by the C-E Owners Group in response to a Nuclear Regulatory Commission Lessons Learned Task Force requirement.

2.0

SCOPE

This report provides C-E's conceptual design for a Reactor Vessel Level Monitoring System. The system functional requirements, description of the designs considered, a design recommendation, and the recommended system configuration comprise the report.

3.0

REFERENCES

3.1

TMI-2 Lessons Learned Task Force Status Report And Short Term Recommendations

3.2

United States Nuclear Regulatory Commission, "Follow-up Actions Resulting From The NRC Staff Reviews Regarding The Three Mile Island Unit 2 Accident", To All Operating Nuclear Power Plants, Dated September 13, 1979.

BACKGROUND

As a result of reviewing the TMI incident, the NRC Lesson Learned Task Force issued a status report and short term recommendations document (Reference 3.1). This document consisted of recommendations to be implemented to further insure the public health and safety. One of these recommendations was associated with instrumentation for detection of inadequate core cooling in PWR's and BWR's. In its assessment of instrumentation adequacy, the task force concludes that sufficient instrumentation existed at TMI-2 to provide adequate information to indicate reduced reactor vessel coolant level, core voiding and deteriorated core thermal conditions. However, in order to preclude failure to recognize such conditions in the future, the task force proposed to address this problem in two stages. The first stage further evaluated the use of installed instrumentation. The second stage was to study and develop system modifications that would not require major structural changes to the plant and that could be implemented in a relatively rapid manner to provide more direct indication of certain plant parameters than that available with present instrumentation. This second stage includes the study of PWR vessel level detectors. In stating its position with respect to this recommendation, the task force required licensees to provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement the currently installed instrumentation including a description of the functional design requirements.

In reference 3.2 the NRC imposed the Lessons Learned Task Force's recommendations upon all Operating Nuclear Power Plants and provided a final schedule for implementation.

The Owners Group subsequently authorized C-E to develop the functional requirements and conceptual design for a system to monitor reactor vessel level. This report documents the results of that study.

In developing the functional requirements and the conceptual design

recommended in this report, it was assumed that the RVLMS is technically required to provide a valid indication of reactor vessel level or the adequacy of core cooling. This is an assumption upon which the design is based - analyses have not been performed to substantiate such an assumption.

In addition to clearly understanding what the results of this task are, it is important to understand what is not provided as part of this task.

1. The described system does not monitor reactor vessel level below the bottom of the hot leg, i.e. into the active core region. Various options appear to be open in this area. It is believed that already installed instrumentation--the incore thermocouples and self powered neutron detectors--can be used to provide an indication of level in the core. Further analyses are required to confirm this. Alternatively, the heated junction thermocouples described in this report could be adapted to extend down into the core region.
2. No analyses or studies have been undertaken as part of this task to correlate the output of the described RVLMS to an "unambiguous indication of inadequate core cooling". The system monitors and displays reactor vessel level above the core, only. Further analyses would be required to show how the indication of reactor vessel level relates to core cooling adequacy.
3. No procedures or guidelines have been developed as part of this task to show how the RVLMS indication should be used.
4. No analyses have been done as part of this task to determine if a system to directly measure reactor vessel water level is required.

5.0

DESCRIPTION OF ASSUMPTIONS, DESIGN BASIS AND FUNCTIONAL REQUIREMENTS

5.1

ASSUMPTIONS

The following assumptions were made in the conceptual design effort of the RVLMS.

1. The reactor vessel inventory between the bottom of the hot leg to the top of the reactor vessel head is to be monitored. This range was specified at the beginning of the task and represents that area of the reactor vessel where no instrumentation exists which can be used to measure vessel inventory. In addition, the response of existing in-core instrumentation may be useful in measuring core inventory.
2. The RVLMS will be useful to the plant operator, therefore, it must be reliable throughout the operating cycle.

5.2

DESIGN BASIS

A Reactor Vessel Level Monitoring System (RVLMS) will supplement the existing NSSS instrumentation that gives an indication of the adequacy of core cooling. The NSSS design addresses the need to supply water to the reactor coolant system during an unlikely event that affects reactor coolant system inventory. However, under these conditions, the RVLMS will inform the plant operator via its alarm that the reactor vessel level inventory is affected, specifically, and additional effort to assure the adequacy of core cooling may be necessary. The continuous recording of reactor vessel level will provide both a permanent record of this parameter and a means of evaluating the results of evolutions which may affect reactor vessel level by indicating trends. In addition, the recording of reactor vessel inventory, in conjunction with other plant instrumentation, will provide the plant operator with a means of determining core cooling adequacy. The reactor vessel level alarm setpoint will be such as to alert the plant operator in time to allow for evaluation of overall NSSS response to the cause for the alarm initiation. The alarm setpoint will also be such as to preclude spurious alarms during plant operation.

In the event the system should suffer a loss of electrical power or a major failure, a unique indication shall be provided to inform the plant operator that the system is not monitoring reactor vessel level and it should not be used to make decisions which could affect core cooling adequacy.

The RVLMS Design Basis is as follows:

1. Sense reactor vessel level and continuously record the parameter in the control room.
2. Provide an alarm and annunciation in the control room when reactor vessel water level decreases to a preset value to alert the plant operator prior to further degradation of reactor vessel inventory.
3. Monitor and record reactor vessel water level during the following plant operating modes:
 - a. Power operation
 - b. Start-up
 - c. Hot standby
 - d. Hot shutdown
 - e. Cold shutdown
4. Monitor and record reactor vessel water level during post accident conditions where reactor vessel inventory may be lost. The Design Basis Events under which the system must meet its design requirements are listed in Table I.

5.3 FUNCTIONAL DESIGN

5.3.1 General

Figure 1 provides a functional block diagram of the Reactor Vessel Level Monitoring System (RVLMS) and its interfaces. The RVLMS monitors changes in the reactor vessel bulk water level. This input is utilized to perform two basic functions by the RVLMS; Display and Alarm.

The first function is to display the reactor vessel water level to the plant operator by employing a strip chart recorder. The second function is to provide an alarm to the Annunciation System when reactor vessel water level reaches a low preset level.

5.3.2 Display Function

The reactor vessel water level is the input to the RVLMS. The RVLMS continuously displays the reactor vessel water level on a strip chart recorder in the control room.

5.3.3 Alarm Function

The RVLMS provides its output signal to the plant Annunciator System. The plant Annunciator System will initiate an alarm when the sensed reactor vessel water level reaches a low preset value.

5.4 FUNCTION REQUIREMENTS

This section specifies the requirements to be met in the design of the RVLMS.

5.4.1 Design Criteria

The following Regulatory Guides, IEEE Standards and 10 CFR 50 requirements will be met in the design of the RVLMS.

5.4.1.1 Regulatory Guides

1. 1.97 Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident - Revision 1, August 1977.
2. 1.29 Seismic Design Classification - Revision 3, September 1978.
3. 1.100 Seismic Qualification of Electric Equipment for Nuclear Power Plants - Revision 1, August 1977.

4. 1.89 Qualification of Class 1E Equipment for Nuclear Power Plants - Original, November 1974.
5. 1.75 Physical Independence of Electric Systems - Revision 2, September 1978.
6. 1.53 Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems - Original, June 1973.
7. 1.118 Periodic Testing of Electric Power and Protection Systems - Revision 2, June 1978.
8. 1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - Revision 3, November 1978.
- *9. Draft Revision 2 to 1.97 Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident. This includes the draft standard ANS-4.5 Functional Requirements for Post Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generating Station.

5.4.1.2 IEEE Standards

1. 323-74 - Qualifying Class 1E Equipment for Nuclear Power Generating Stations.
2. 338-77 - Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems.
3. 344-75 - Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.
4. 384-77 - Standard Criteria for Independence of Class 1E Equipment and Circuits.
5. 379-77 - Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems.

*These items are included to represent those requirements which should be considered in the design but are not presently required.

5.4.1.3 10 CFR 50 Appendix A

1. CRITERION 1 - Quality Standards and Records
2. CRITERION 2 - Design Bases for Protection Against Natural Phenomenon
3. CRITERION 3 - Fire Protection
4. CRITERION 13 - Instrumentation and Control

5.4.1.4 10 CFR 50 Appendix B

5.4.2 Interface Requirements

The following interface requirements shall be accommodated:

5.4.2.1 Input Requirements

Electrical Power Source

The RVLMS power sources shall be from two Class 1E power buses. One power source shall supply electrical power to one of the RVLMS channels. The other, separate, power source shall supply electrical power to the other, separate, RVLMS channel.

5.4.2.2 Output Requirements

Annunciation and Alarm

The RVLMS shall provide output signals to the Plant Annunciator system when the sensed reactor vessel water level reaches a low preset value. A separate output signal shall be provided for each separate RVLMS channel. These signals shall alarm separate indications on the Plant Annunciator system. The Plant Annunciator shall provide an audible and visible indication to the plant operators of the RVLMS low reactor vessel water level condition.

5.4.3

Sensor Requirements

Reactor Vessel Level

The RVLMS shall provide signals via reactor vessel level transmitters which represent the reactor vessel bulk water level. The range of the signals shall be at least from the top of the reactor vessel head to the bottom of the reactor vessel outlet nozzle.

Sensor Requirements Locations

The RVLMS sensors may be located internal to or external to the reactor vessel.

Sensor Requirements Range

The RVLMS sensors shall monitor the range from the bottom of the reactor vessel outlet nozzle to the top of the reactor vessel head, as a minimum.

Number of Channels

The RVLMS shall have at least two redundant channels for sensing reactor vessel level.

5.4.4

System Accuracy and Response Time Requirements

System Accuracy

The RVLMS system accuracy shall be better than $\pm 10\%$ of the required range discussed above. This accuracy applies to both the alarm set-point and to the indication to the plant operator as read on the RVLMS recorders.

System Response Time

The RVLMS system response time shall be at most 6.0 seconds from the time a change in reactor vessel level is sensed until the change is indicated at the recorder. It shall also be at most 6.0 seconds from the time the reactor vessel water level reaches the low preset setpoint until the alarm signal is present at the output of the RVLMS.

5.4.5 Display Requirements

Recording

The RVLMS shall continuously monitor and simultaneously record the signal from each reactor vessel level sensor. Separate recorders shall be provided for each separate RVLMS channel. The recorders shall be a permanently mounted component located in the control room and uniquely identified as post accident monitoring equipment.

5.4.6 Alarm and Annunciation Requirements

The RVLMS shall provide alarm signals to the Plant Annunciator System upon determining that reactor vessel level has reached a low preset value. A separate alarm signal shall be provided for each RVLMS channel. This setpoint will be determined later during the setpoint effort.

5.4.7 Testing Requirements

Instrument Channel Checks

The RVLMS channel outputs shall be displayed such that channel operability may be verified by comparing readings between channels.

Functional Tests

The RVLMS channels shall have the capability to be tested by injecting a test signal as close to the sensor as is practical to assure that each channel is capable of performing its design function. This shall include having the capability of testing the accuracy of the setpoint used for the alarm function as well as testing the accuracy of the display available to the operator. The ability to test the system response time shall also be provided.

Sensor Calibration Verification Tests

The RVLMS sensors shall have the capability to be tested such that with a known precision input the instrument gives an output signal of the required accuracy and response time.

5.4.8 Operational Requirements

The RVLMS is required to function within the required accuracy and time response limits during the modes of plant operation specified in the Design Basis section.

5.4.9 Post Accident Monitoring Requirements

The RVLMS is required to perform its required functions with the required accuracy and time response limits during post accident monitoring conditions where reactor vessel level may be affected.

5.4.10 Qualification Requirements

The RVLMS shall be qualified to perform its required functions with the required accuracy and time response before, during and after the most adverse environmental conditions associated with the Design Basis Events listed in Table I. With respect to seismic qualification, the RVLMS shall continue to perform its required functions with required accuracy and time response following the seismic event.

5.4.11 Indication of Loss of Power or System Failure Requirements

An indication of loss of electrical power or a system failure to each RVLMS channel shall be provided.

5.4.12 Accessibility & Mounting Requirements

The RVLMS recorders shall be permanently mounted in the control room and the recorders shall be clearly displayed for operator use.

6.0 RECOMMENDATION

This section of the report provides Combustion Engineering's recommendation of the best means by which to measure reactor vessel level and the reason for the recommendation.

6.1 RECOMMENDED DESIGN

6.1.1 Heated Junction Thermocouple

The heated junction thermocouple sensing means measures the change in thermocouple output voltage as a result of the difference in the thermal conductive properties between steam and water. A series of heated junction thermocouples will be located at different axial positions above the core. Figures 2 and 3 depict heated junction thermocouple installations in the reactor vessel. Each discrete location would have two thermocouples connected in series but in electrical opposition to each other. One of these thermocouples will be heated, thereby establishing a reference differential voltage output which is a function of the heater input. When the water surrounding the heated junction thermocouple is replaced by steam or air the voltage generated by the heated thermocouple will change because the heat generated will no longer be removed from the area by the water. Thus the temperature of the heated thermocouple will increase relative to the unheated thermocouple. This change will be used to determine the steam water interface in the reactor vessel.

6.1.1.1 Evaluation of the Heated Junction Thermocouple

The heated junction thermocouples are adaptable to the operating reactors by insertion through an In Core Instrument (ICI) nozzle port or a Part Length CEA location. This method of determining vessel level is presently in use in a research facility in Idaho. The resolution of the system is adjustable over the range of measurement based upon the distance between thermocouples. The accuracy of the system is expected to exceed the $\pm 10\%$ of span

presently required. The heated junction thermocouples may be compatible with existing core designs for measuring level in the core. However, this application has not been thoroughly evaluated. Since the application of heated junction thermocouples for monitoring vessel level has been demonstrated, there is a high probability of system installation by the January 1981 deadline imposed by the Nuclear Regulatory Commission.

Three disadvantages associated with the use of heated junction thermocouples are as follows:

1. Use of the ICI nozzle port for thermocouple insertion will require removal of in-core instrumentation. With respect to the Technical Specifications, the plant will then be operating with a certain percentage of unavailable in-core-instruments based upon the in-core-instrument uncertainty analysis.
2. Use of a Part Length CEA location will require removal of a Part Length CEA in order to accommodate the heated junction thermocouples.
3. The resolution of the system is restricted by the number of electrical leads which can physically be accommodated by the ICI nozzle port and/or spare containment electrical penetrations.

6.2 RECOMMENDED SYSTEM CONFIGURATION

6.2.1 Description of the Recommended System

Figure 4 shows the recommended system configuration for the Reactor Vessel Level Monitoring System as discussed below.

The recommended system configuration will require that four Part Length CEA's be removed to provide locations for the sensing devices.

Two Heated Junction Thermocouple level sensor groups will comprise one channel of vessel level indication. Each group of Heated Junction Thermocouples will be inserted into a vacated Part-Length CEA location. The output signal will be sent to a recorder located in the control room where each output signal can be continuously recorded. A selector switch may be provided to allow the average of the two signals or any one or both of the two level sensor outputs to be recorded at a time.

The remaining two Part-Length CEA locations will also contain a Heated Junction Thermocouple level sensor in each location. This second channel will be identical to the first channel. The second Heated Junction Thermocouple channel will record its sensor output on a separate recorder.

6.2.2 Reasons for the Recommended System

The advantage of the specific type of sensor recommended has previously been discussed. Following is a discussion of the reasons for the recommended system configuration.

It must be emphasized that in developing the functional requirements and conceptual design recommended in this report, it was assumed that the RVLMS is technically required to provide a valid indication of reactor vessel level or the adequacy of core cooling. This assumption was required to be made in order that the task could proceed. No analyses

have been performed as part of this task to substantiate such an assumption.

Making the assumption that the RVLMS is technically required, it is believed that the recommended design is the optimum method of performing its required functions. A system of lesser complexity (i.e. no redundancy) may be acceptable from a licensing standpoint. However, if a RVLMS is not required for a technical reason, it is recommended that no system be installed.

During the course of developing a conceptual design for a Reactor Vessel Level Monitoring System, certain basic criteria emerged as being necessary or very desirable for such a system. Meeting these criteria will go a long ways toward ensuring that the system will not only indicate level to the required degree of accuracy but will also have sufficient reliability and clarity so as to be genuinely useful to a plant operator. The criteria and how the recommended system configuration meets the criteria is discussed below.

Redundancy

Redundancy is a desirable design feature to incorporate because it is a means by which the availability of the system can be assured throughout the operating cycles. If one has a system of two separate channels, one has redundancy. If one sensor fails, the redundant sensor is still operable. However, it is very desirable to have more than two degrees of redundancy in any system that is to be relied upon. If there are two channels indicating different levels, which one is correct? It is thus desirable to have at least three degrees of redundancy to alleviate this problem. If there are four degrees of redundancy, the system can continue operating reliably with one redundant indication system failed. This still leaves the capability to identify a failure that may develop as a result of the event the system is needed for.

The recommended system has four degrees of redundancy. There are two redundant channels, each of which has two separate sensors. This is

the optimum system from a redundancy standpoint and is easily implemented because the designs require removal of part length CEAs and they are arranged in symmetric groups of four.

Non-Hydraulic

During the events that the RVLMS is required to operate, the hydraulic forces in the area of the reactor vessel head can be extremely dynamic and unpredictable. Although this concern may not be insurmountable, much analytical and design work would be required to ensure proper indication of level using a sensor system that worked on a hydraulic principle. The recommended design does not utilize this principle to determine level.

No Moving Parts

A level system that relies on physically moving components should be avoided if possible. Historically, moving components tend to fail by virtue of their movement. The recommended design uses no moving parts to sense vessel level.

7.0 OTHER DESIGNS CONSIDERED

This section provides a brief description of other designs considered for use in measuring Reactor Vessel Level along with the advantages and disadvantages associated with each design.

7.1 Radio Frequency (RF) Probe

The R-F probe design (Figure 5) consists of a Time Domain Reflectometry unit which sends a voltage step along the length of the R-F Probe. The pulse is reflected back to the Time Domain Reflectometry unit when it encounters a discontinuity along the probe such as water. The time between pulse initiation and reflection is measured by the circuitry and translated into a level signal.

7.1.1

Advantages

1. It is highly feasible since the design was used successfully in the C-E pot boiler blowdown test conducted in 1976.
2. Test Report, TR-ESE-164, "R-F Probe for Two Phase Liquid Level Measurements Description and Calibration", dated September 29, 1977, indicated this system is suitable for measurement of reactor vessel level. Accuracy ± 1.5 "; Response Time 60 msec.
3. It is adaptable to operating operators.
4. There are no moving parts.
5. Corrections to problems are documented in the test report. Therefore, development time is reduced making the January 1981 installation date realistic.

7.1.2

Disadvantages

1. Requires removal of part length CEA.
2. May impact existing refueling procedures (probe must be withdrawn).
3. Probe and accessories must be manufactured.
4. The effect of radiation, coolant temperature changes and coolant chemical concentration changes on system stability and reliability must be further evaluated.

7.2 FLOATING SOURCE

The floating source concept is shown in two variations on Figure 6 and 7. It consists of a low strength Beryllium-Plutonium neutron source double encapsulated within a thin wall stainless steel sphere. The sphere is pressurized with helium to minimize the shell thickness and for detecting leaks. In one variation (Figure 6) the sphere is 7" in diameter and is contained within a CEA shroud. The only addition required to the upper guide structure (UGS) is the removal of CEA guide brackets and the addition of a "top hat" which is mechanically joined to the top of the CEA shroud.

This arrangement of course means that a part length CEA must be removed. The second variation (Figure 7) utilizes a 3" dia sphere contained within a perforated sleeve. The sleeve is installed within the CEA shroud and mechanically locked in place with no machining or cutting required. During normal operation, the sphere is pinned against the top of the sleeve by the buoyant force of the coolant. When the water level drops, the sphere follows the water level. A neutron detector mounted on the R.V. closure head directly above the source sleeve measures the position of the sphere as a function of neutron reception.

7.2.1 Advantages

1. It is feasible as determined by review of existing technology.
2. It is expected to operate under normal and accident conditions.
3. It is adaptable to operating reactors by modifying vessel components.
4. It has minimum impact on existing refueling procedures.
5. Its accuracy is expected to meet the functional requirements.

7.2.2 Disadvantages

1. Requires removal of part length CEA and modification to CEA guide tube.
2. Design must ensure that ball and/or source remain in CEA guide tube.
3. Floating ball movement against the CEA guide tube may wear through ball surface.
4. Problems associated with design development may prevent system installation by January 1981.
5. Source, ball and CEA guide tube modifications must be manufactured.
6. Vessel Head Removal is required to correct any mechanical problems.

7.3 FIXED NEUTRON SOURCE AND DETECTOR

The fixed neutron source and detector concepts are shown on Figures 8 and 9 and utilize the ICI routing conduit. The two schemes under consideration include a fixed separate source conduit and two assemblies of small neutron detectors axially located within an adjacent conduit (Figure 8), and an alternate scheme where the source and detectors are located in the same conduit (Figure 9). The detectors measure the neutron flow from adjacent sources and, therefore, accurately predict water level and steam quality above the water layer.

7.3.1 Advantages

1. It is feasible as determined by review of existing technology.
2. It is expected to operate under normal and accident conditions.
3. It is adaptable to operating reactors by inserting detector string through ICI port.
4. Its accuracy is expected to exceed the functional requirements.
5. There are no moving components.
6. Detectors exist.

7.3.2 Disadvantages

1. Requires use of ICI Nozzle Port. If there are no spare ports, ICI removal is necessary.
2. There is a history of failure associated with the detector.
3. Problems associated with design development may prevent system installation by January 1981.
4. Source must be manufactured.

7.4 FLOATING DIP STICK - FLOATING SPHERES

The floating dip stick concept is shown in Figure 10 and consists of large diameter thin walled stainless steel spheres connected to a shaft. The top of the shaft contains a magnet which can be located by a reed switch assembly in the same manner as CEA position is determined. The large diameter spheres serve as a floatation device for the stick. This concept requires that the density of the coolant be constantly monitored by means of temperature recording equipment.

An alternate scheme to the floating dip stick concept is shown in Figure 11 and consists of a long column of 1-1/2" diameter thin walled spheres. The column of spheres is supported by the buoyant force acting on about the 15 lower spheres. The entire column is enclosed within a perforated sleeve. The upper sphere contains a magnet similar to that used in the floating dip stick scheme. This scheme is much simpler than the floating dip stick when it comes to special refueling procedures because the upper containment sleeve and spheres are handled along with the closure head, i.e., the column of spheres are held in place by battery operated electromagnetic means.

7.4.1 Advantages

1. It is feasible as determined by review of existing technology.
2. It is expected to operate under normal and accident conditions.
3. It is adaptable to operating reactors by removal of a part length CEA.
4. Its accuracy is expected to meet the functional requirements.
5. The existing reed switch position transmitter may be used.

7.4.2

Disadvantages

1. Requires removal of part length CEA and modification to CEA guide tube.
2. Accuracy is dependent upon water density.
3. More than one floating ball is required.
4. Movement of the floating ball against the CEA guide tube may wear through the surface of the ball.
5. Floating balls, dipstick and CEA guide tube modifications must be manufactured.
6. Vessel Head Removal is required to correct any mechanical problems.
7. Problems associated with design development may prevent system installation by January 1981.
8. It may impact existing refueling procedures (dipstick must be removed).

7.5

ULTRASONIC PROBE

The ultrasonic probe design (Figure 12) consists of a pulse transmitter which sends a pulse down a stepped shaft wave guide which is located within a Part Length CEA guide tube. The wave guide is reduced in circumference (stepped) at regular intervals to provide a distinct interface upon which the signal will act. The pulse will reflect back when it encounters the discontinuity in the wave guide. The amplitude of the back reflection will be larger if a step is covered with steam than with water. The difference in the amplitude of reflected signals is analyzed by the circuitry for water level determination.

7.5.1

Advantages

1. It is feasible as determined by review of existing technology.
2. It is expected to operate under normal and accident conditions.
3. It is adaptable to operating reactor by removal of part length CEA.
4. There are no moving parts.
5. Head removal not necessary for repair.

7.5.2 Disadvantages

1. Requires removal of part length CEA and modification to CEA guide tube.
2. Problems associated with design development may prevent system installation by January 1981.
3. May impact existing refueling procedures (wave guide must be withdrawn).
4. Wave guide and CEA guide tube modification must be manufactured.
5. Steam bubbles may collect on shaft steps thus causing false level indication.

7.6 BUOYANT FORCE TRANSMITTER

The buoyant force transmitter is shown in Figure 13 and consists of a 15 feet long - 2" diameter light weight hermetically sealed stainless steel tube. The shaft is connected to a linear voltage differential transmitter (LVDT) cell which monitors change in weight of the tube as a function of the buoyant force. The shaft is positioned within a CEA shroud.

7.6.1 Advantages

1. It is feasible as determined by review of existing technology.
2. It is expected to operate under normal and accident conditions.
3. It is adaptable to operating reactors by removing part length CEA.
4. Its accuracy is expected to meet the functional requirements.

7.6.2 Disadvantages

1. Requires removal of part length CEA.
2. Output must be compensated for fluid density changes to meet accuracy functional requirements.
3. Problems associated with design development may prevent system installation by January 1981.
4. Probe and accessories must be manufactured.
5. Vessel head removal is required to correct any mechanical problems.

6. Receiving an accurate indication of the buoyant force on the LVDT cell is complicated by the system pressure, the dynamic forces on the tube and the frictional resistances.

7.7 EXTERNAL STANDPIPE WITH FLOAT SENSOR OR DP CELL

The external standpipe approach (Figure 14) would make use of existing connections on the primary system. The standpipe would be connected between the head vent and a flange connection on the bottom of the hotleg piping. The standpipe, which is filled with reactor coolant, would contain a float mechanism which contains a source or a magnet. A suitable detector would be located alongside the standpipe to detect the position of the float. The position of the float would reflect the reactor vessel water level.

A DP cell (Figure 14) may be added to or substituted for the standpipe.

7.7.1 Advantages

1. It is feasible as determined by existing technology.
2. It is expected to operate under normal and accident conditions.
3. It is adaptable to operating reactors by connecting to the head vent and flange on the hotleg.
4. Installation will not require modification of reactor internals or removal of the vessel head.
5. DP cells are used to measure level and inaccuracies associated with level measurement are understood.
6. Accuracy associated with float in the standpipe is expected to meet the functional requirements.
7. System is accessible.

7.7.2

Disadvantages

1. External standpipe must be removed and stored during refueling.
2. DP cell must account for flow in the reactor vessel, changes in reactor coolant temperature, and the difference between reference leg temperature and reactor coolant temperature.
3. DP cell reference leg flashing effects on accuracy must be considered.
4. Standpipe layout problems are plant specific.
5. Increases primary system piping which can experience leakage.
6. Leakage at flanged connections will affect system accuracy.
7. Problems associated with standpipe design and installation may prevent system installation by January 1981.

TABLE I

RVLMS DESIGN BASIS EVENTS

The RVLMS shall be designed to function during the following Design Basis Events (including the environmental effects associated with these events).

- I. Loss of Coolant Accident (LOCA) up to and including a double-ended rupture of the largest pipe of the Reactor Coolant Pressure Boundary (RCPB) as defined in 10 CFR 50.2.
- II. Steam System pipe rupture up to and including a double-ended rupture of the largest pipe in the system.
- III. Feedwater System (main, auxiliary or emergency, and steam generator blowdown system) pipe rupture up to and including a double-ended rupture of the largest pipe in the system.
- IV. Depressurization due to inadvertent actuation of a pressurizer or secondary safety valve.

RVLTS FUNCTIONAL BLOCK DIAGRAM

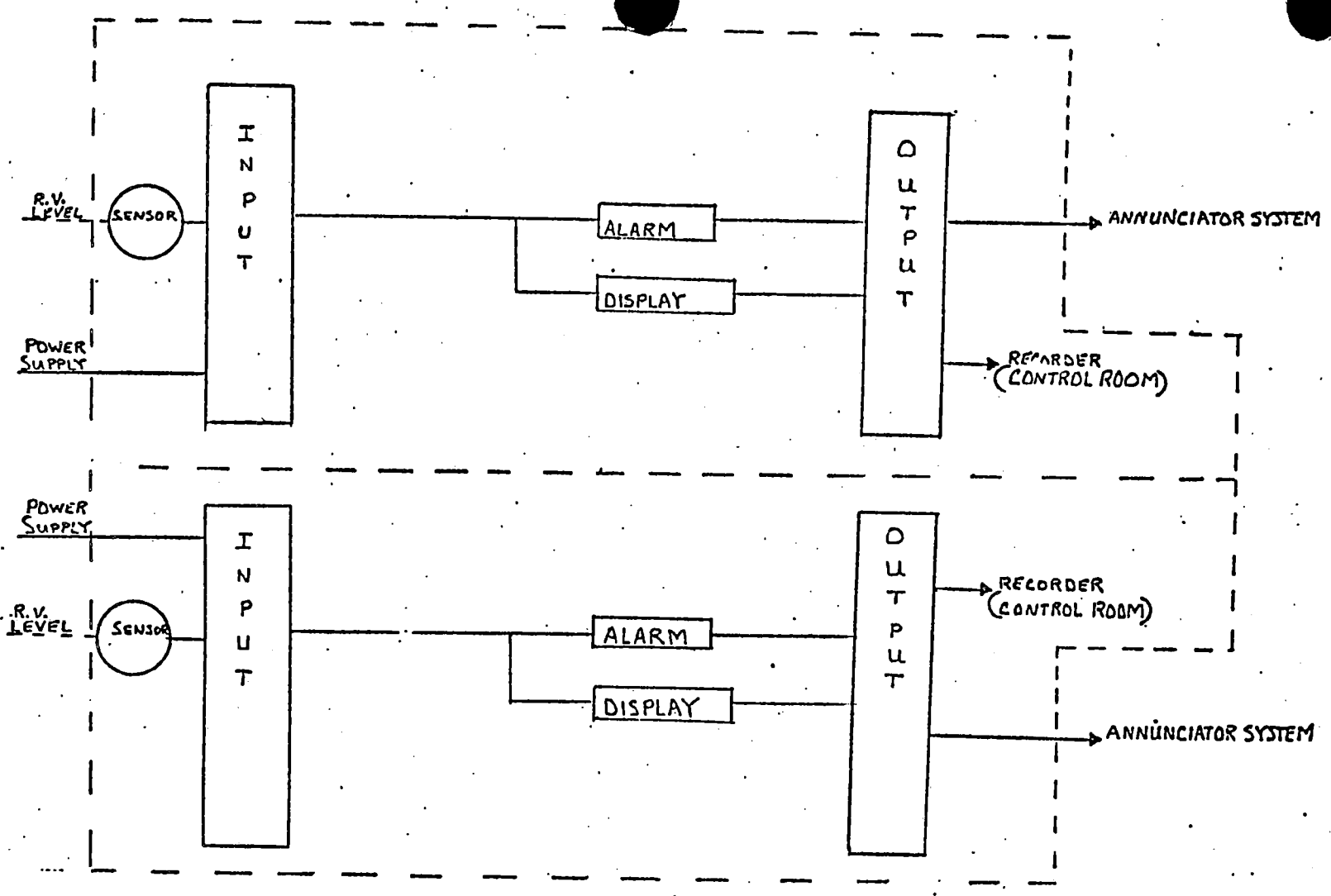
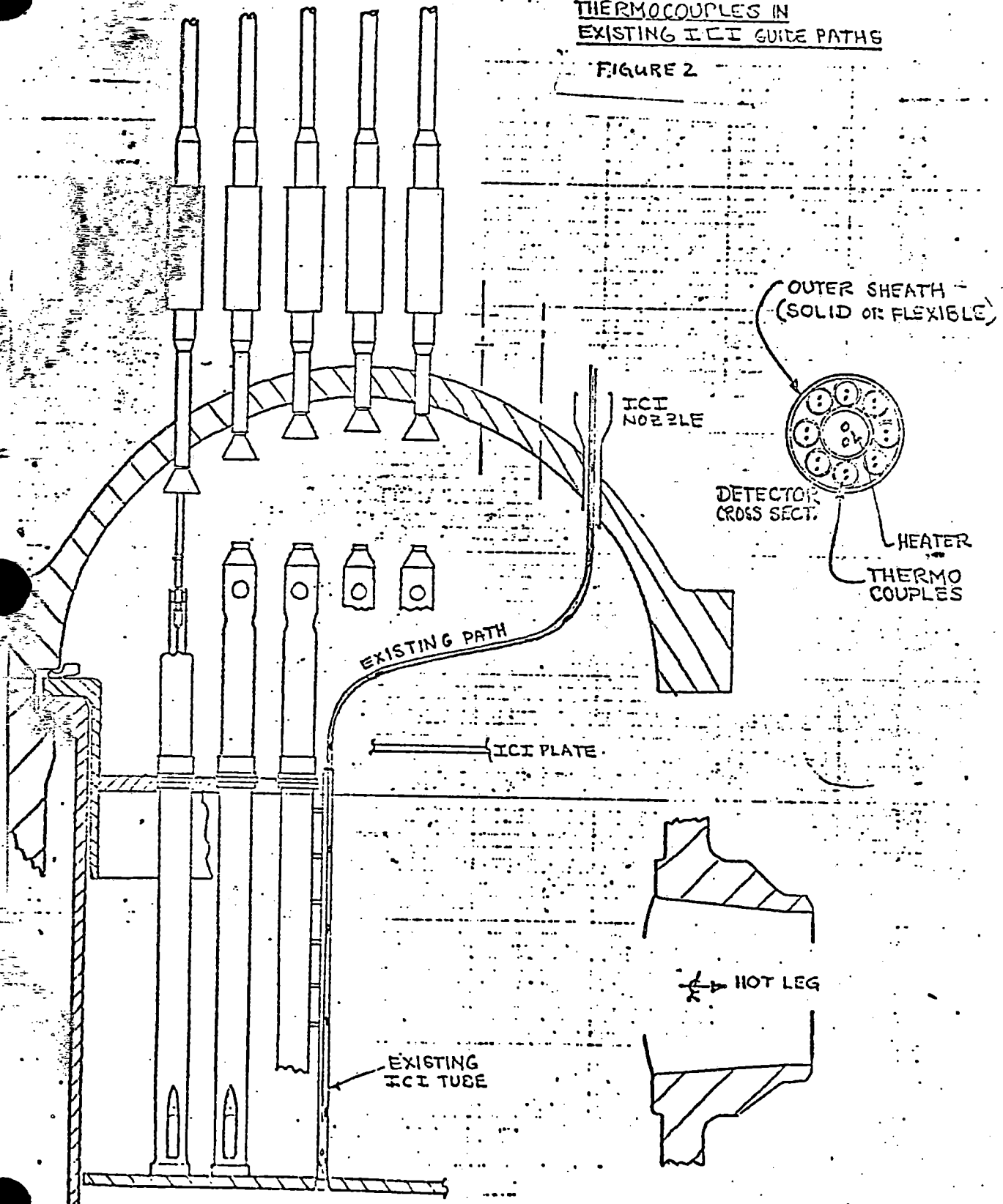


FIGURE 1

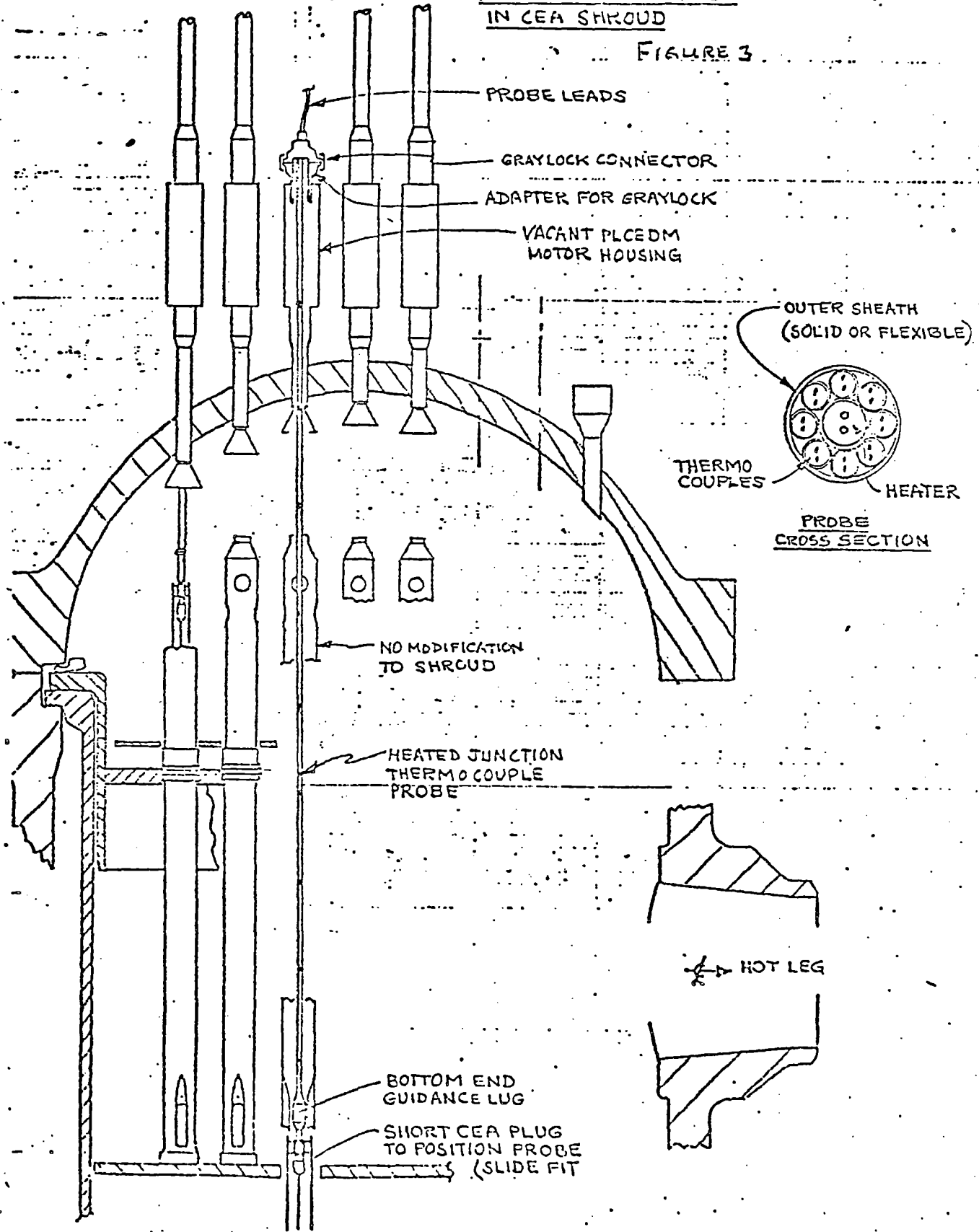
HEATED JUNCTION
THERMOCOUPLES IN
EXISTING ICI GUIDE PATHS

FIGURE 2

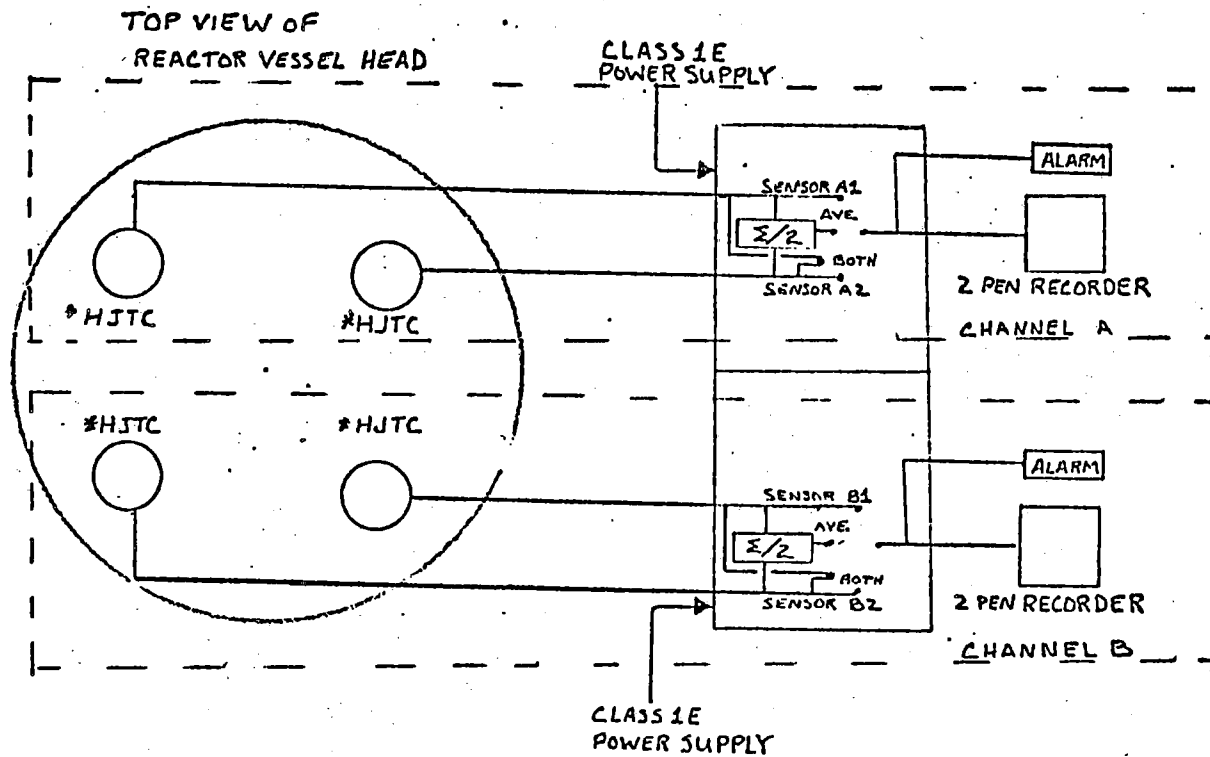


HEATED JUNCTION
THERMOCOUPLE PROBE
IN CEA SHROUD

FIGURE 3



SYSTEM CONFIGURATION FOR A RVLMS

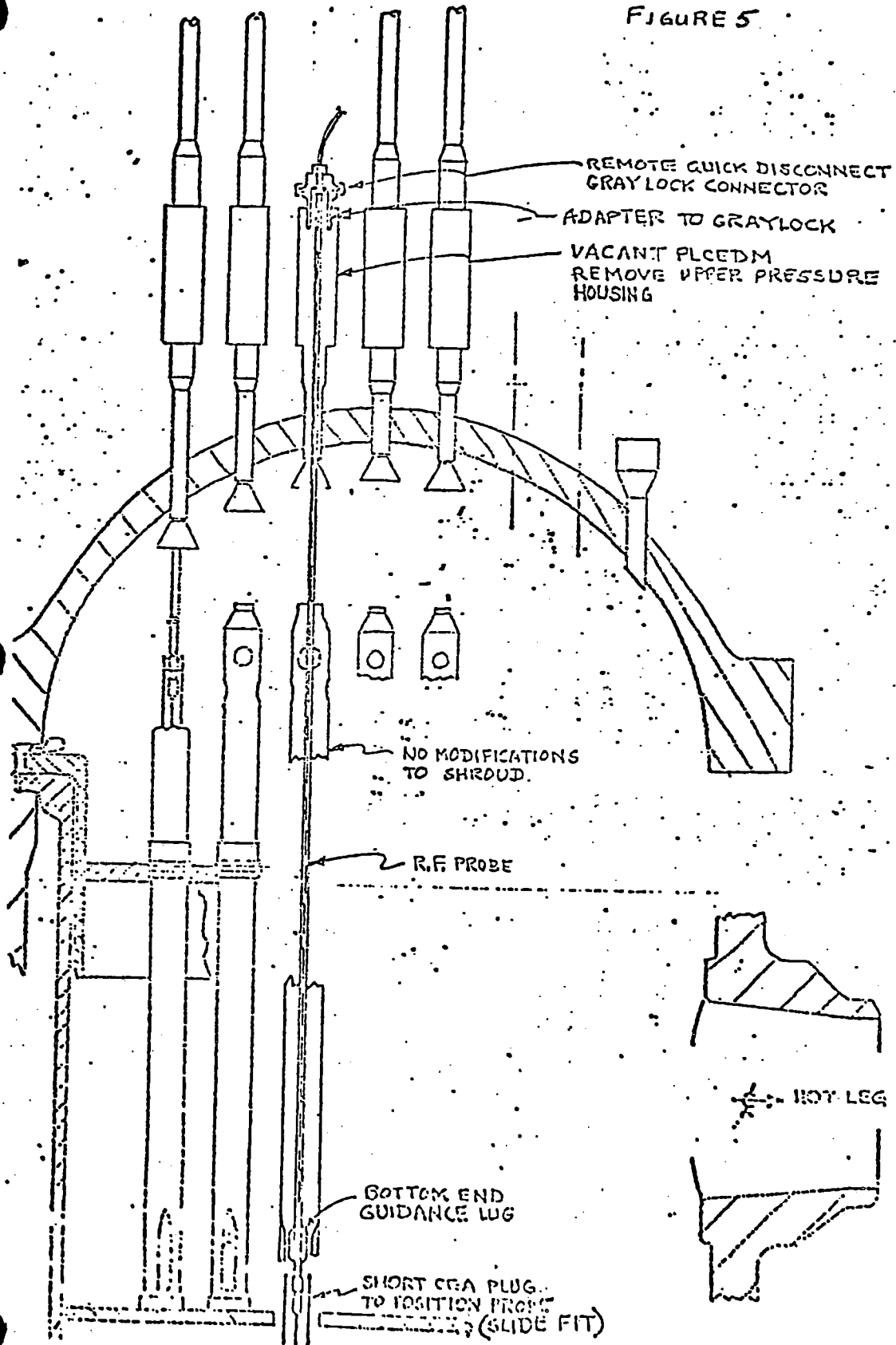


*HJTC - HEATED JUNCTION THERMOCOUPLE

FIGURE 4

RADIO FREQUENCY (R.F) PROBE

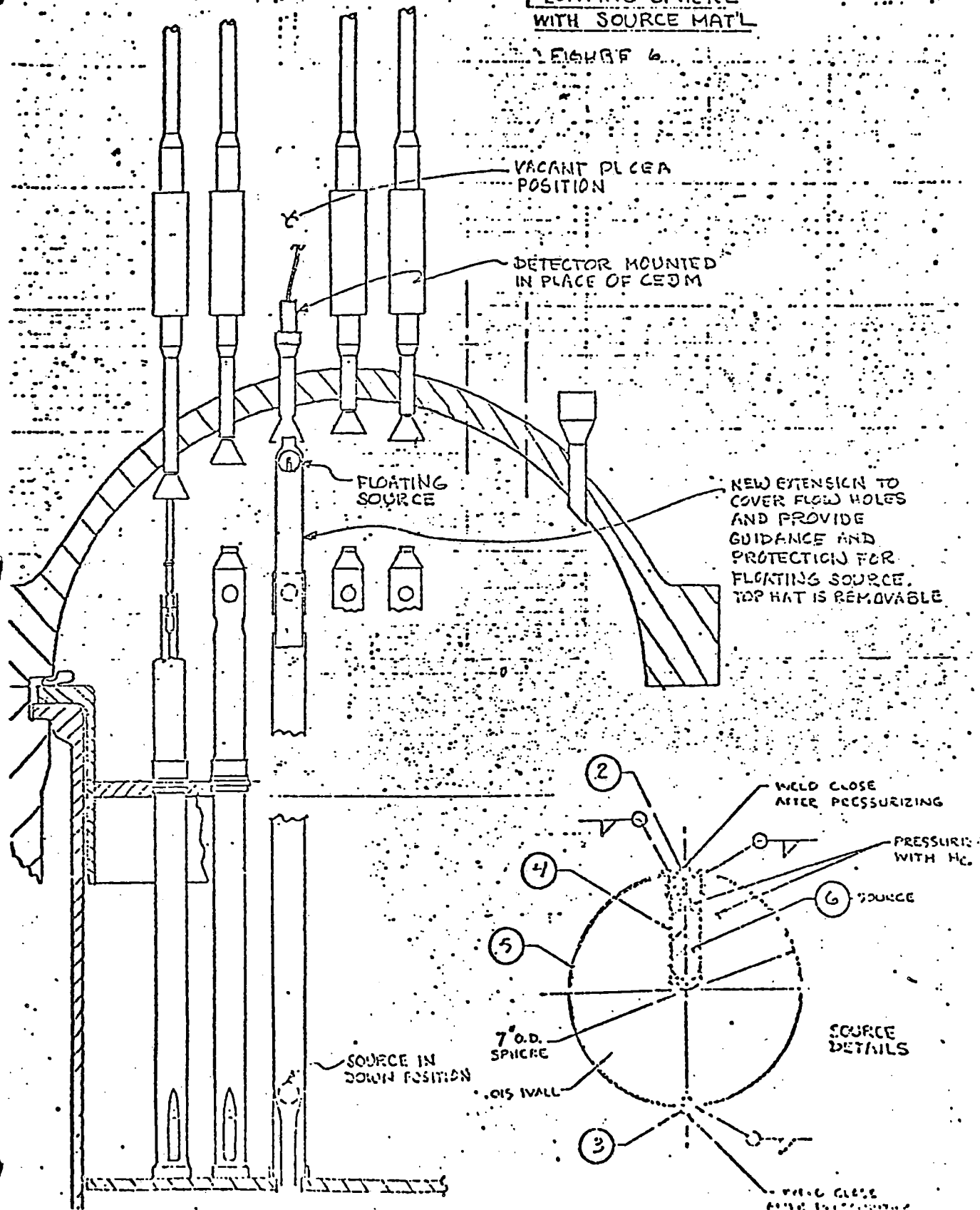
FIGURE 5



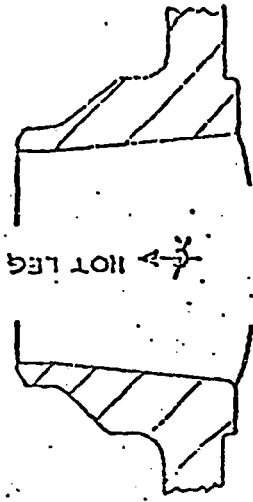
A. BROWN

FLOATING SPHERE WITH SOURCE MAT'L

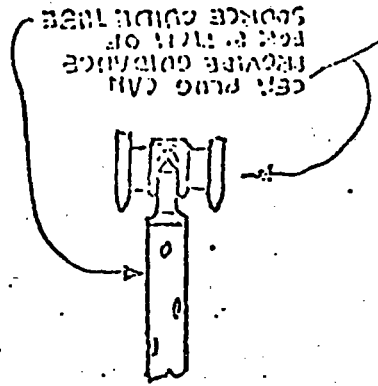
FIGURE 6



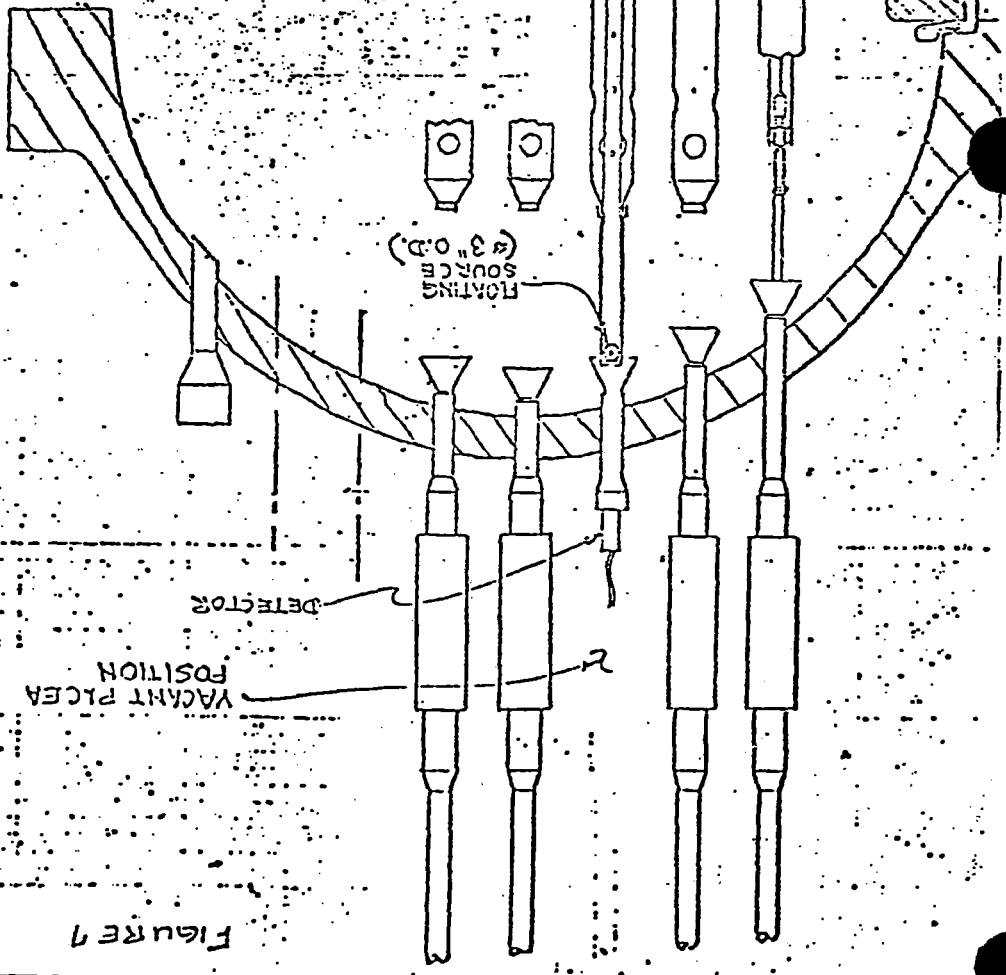
J. Bussler



PERFORATED SOURCE GUIDE TUBE WITH EXTENSION IN HEAD AREA. TOP HAT IS REMOVABLE. (MINOR CHANGES TO IRRADIATED INTERVALS)



FLOATING SOURCE (3" O.D.)

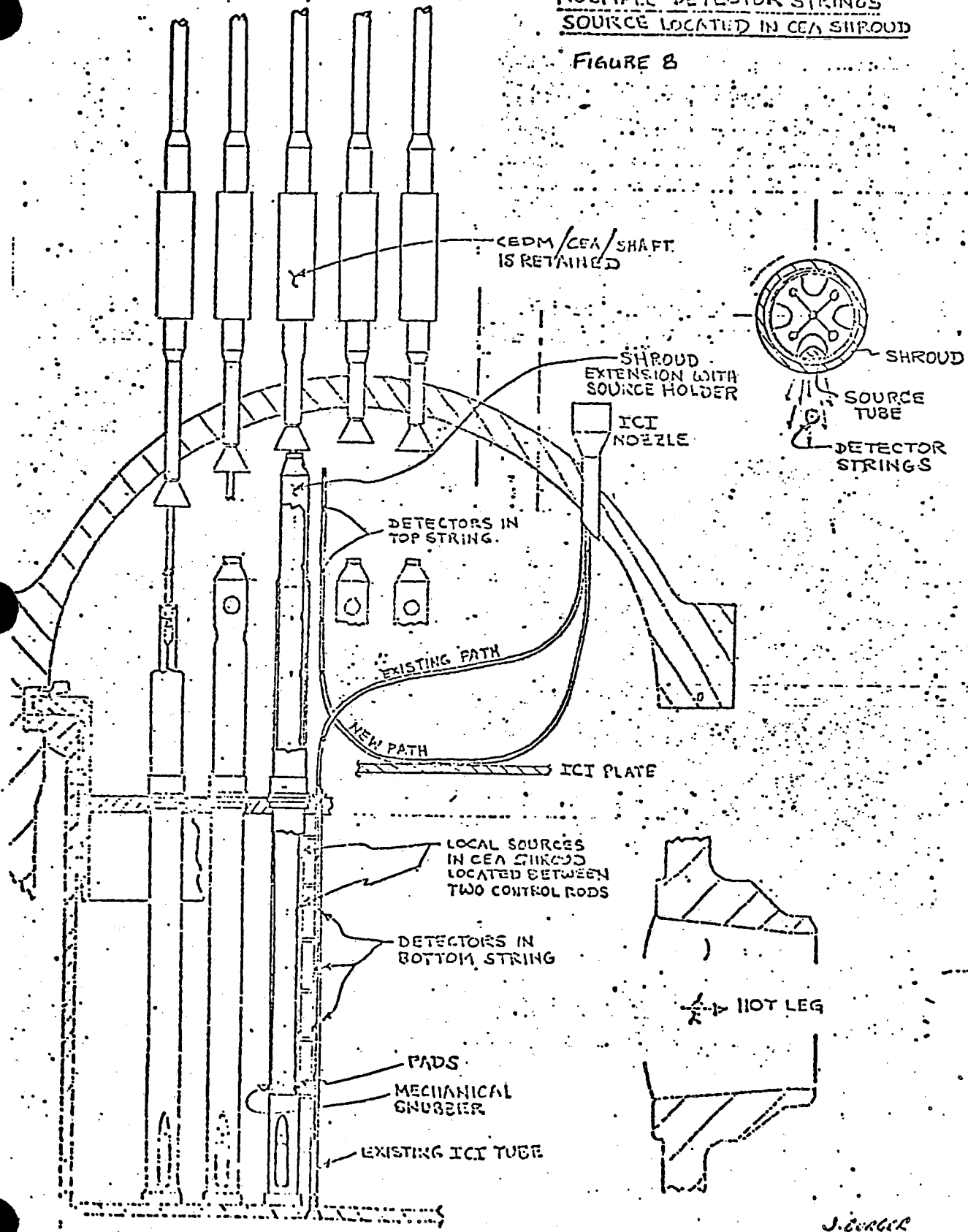


SMALL SPHERICAL SOURCE

FIGURE 1

MULTIPLE DETECTOR STRINGS
SOURCE LOCATED IN CEA SHROUD

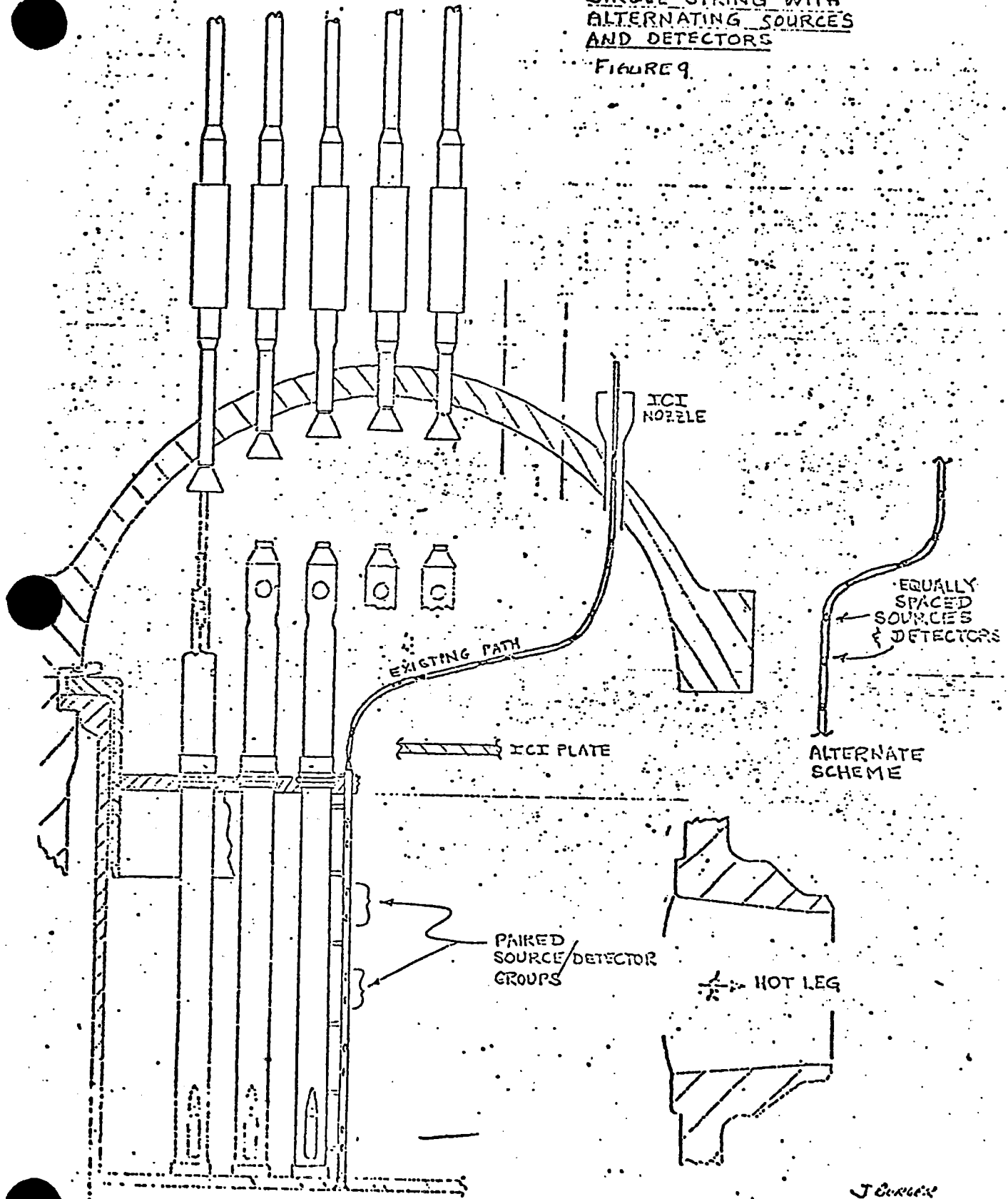
FIGURE 8



J. Berger

SINGLE STRING WITH
ALTERNATING SOURCES
AND DETECTORS

FIGURE 9



J. BURTON
12-2-79

FLOATING DIP STICK/REED SWITCH SYSTEM

FIGURE 10

WATER PRESSURE HOUSING

REED SWITCH POSITION INDICATION SYSTEM

MAGNET

VACANT PLACE

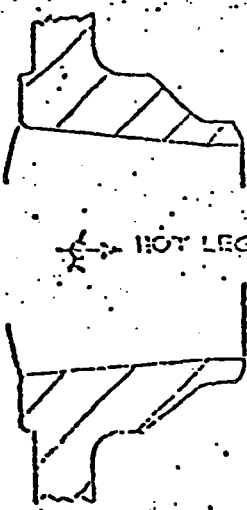
LIGHTWEIGHT MAGNET SUPPORT POLE

NEW EXTENSION SHAFT GUIDE WITH BOTTOM EXTENSION TO COVER FLOW HOLES AND PROVIDE GUIDANCE AND PROTECTION FOR FLOATS. TOP HAT IS REMOVABLE

PREPRESSURIZED INCO 718 OR TITANIUM SPHERES

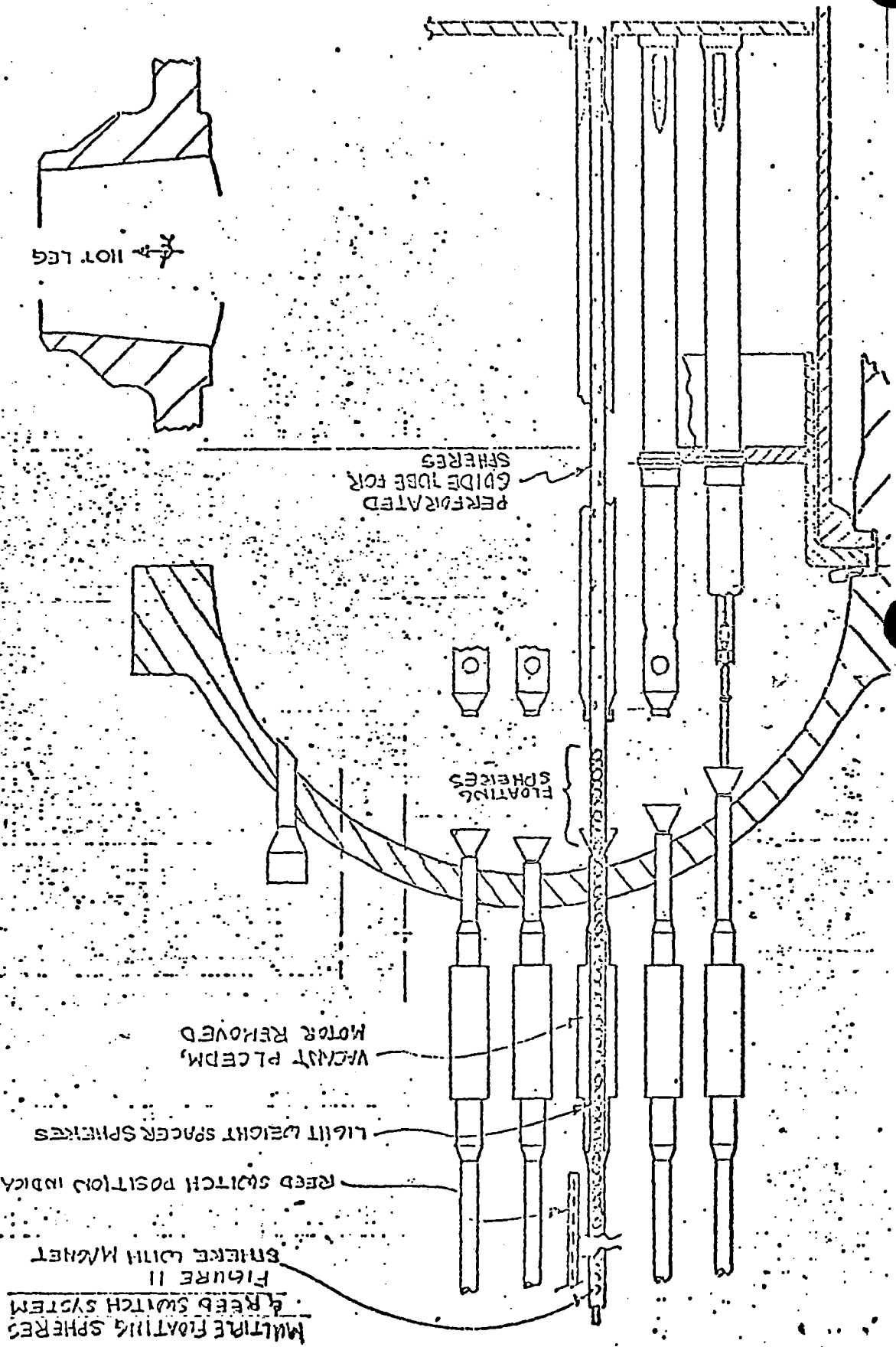


FLOATS IN DOWN POSITION

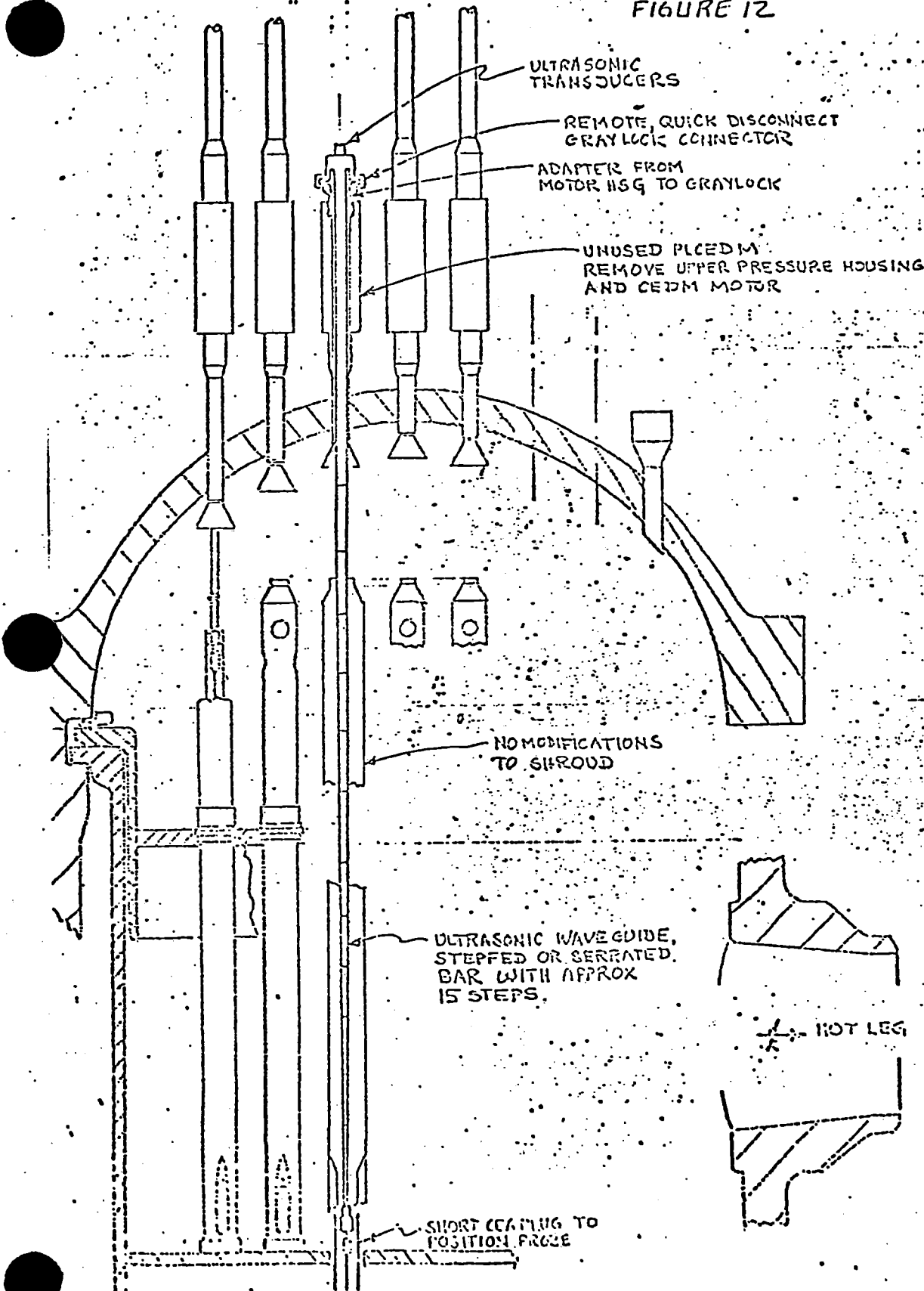


J. BURGER
12-2-77

Source

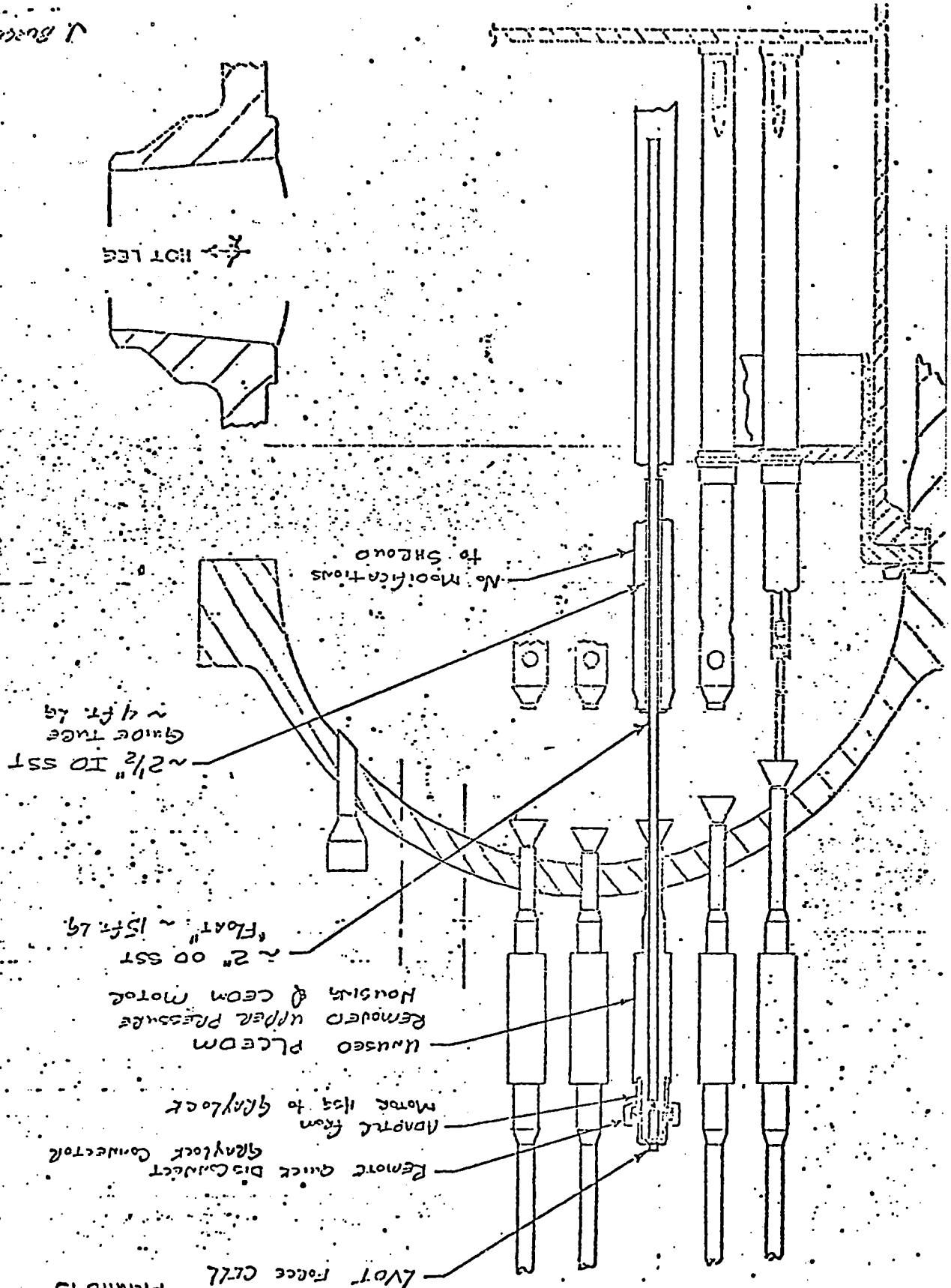


ULTRASONIC PROBE
FIGURE 12



J. FINGER
12-2-79

J. Bussler

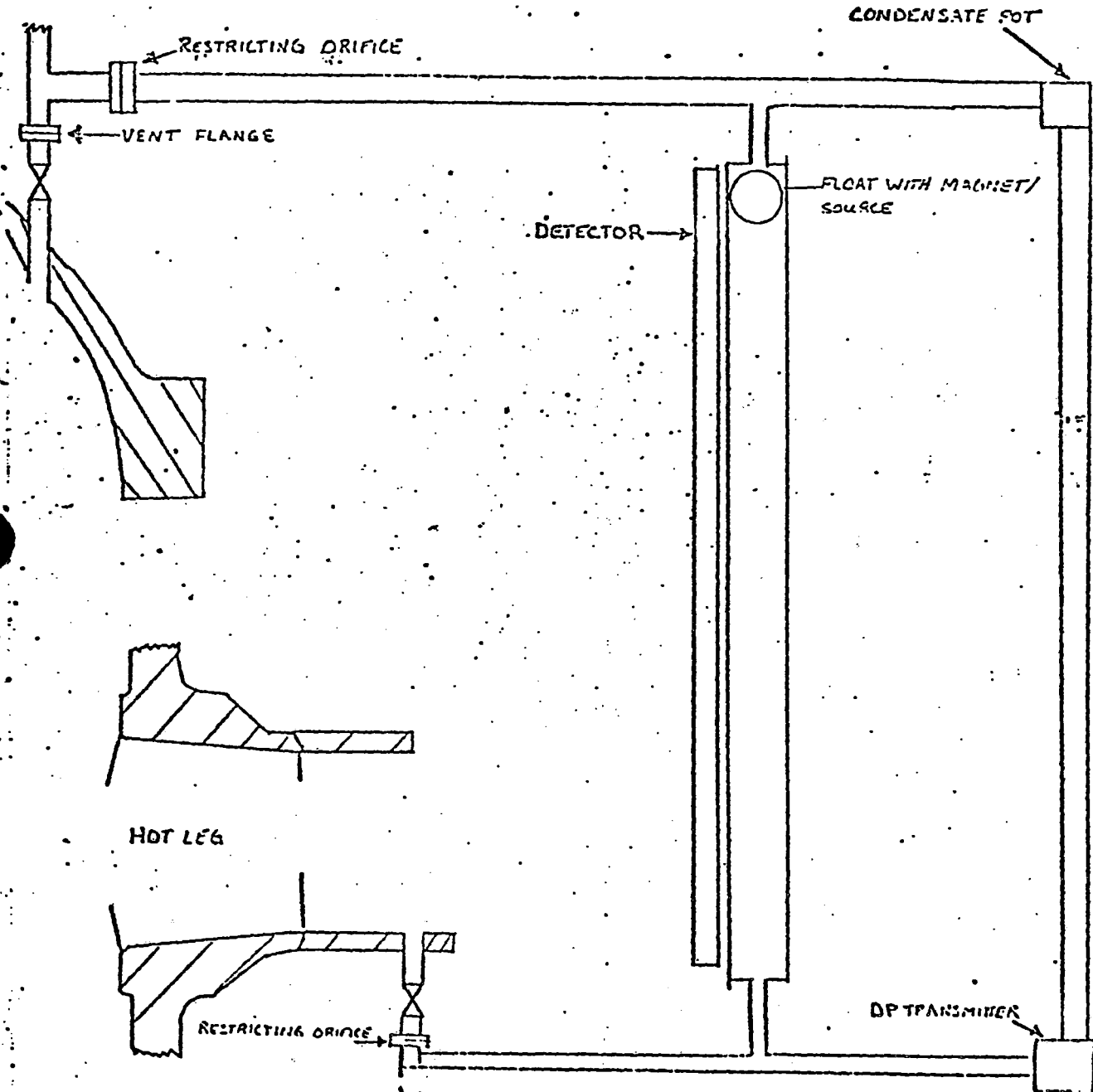


BUCCANT FORCE INDICATION PROBE

FIGURE 13

EXTERNAL STANDPIPE WITH A
FLOAT SENSOR OR DP CELL

FIGURE 14



MPR ASSOCIATES, INC.

December 4, 1979
P-490

Mr. R. A. Vincent
Consumers Power Company
Palisades Nuclear Power Plant
Route 2 - Box 154
Covert, Michigan 49043

Subject: Palisades Systems Outside Containment which
Could Contain Radioactive Fluids

Dear Mr. Vincent:

At your request, we have performed a review to determine which of the systems located outside containment would or could contain highly radioactive fluids during a serious transient or accident. In particular, this review identifies the systems included in the leakage reduction program being implemented in accordance with Nuclear Regulatory Commission Recommendation 2.1.6.a in "TMI-2 Lessons Learned Task Force Report (Short Term)," NUREG-0578.

The results of our review indicate that Consumers Power Company compliance with the NRC recommendation should be accomplished as follows:

1. A series of plant modifications would be made to reduce the number of systems outside containment which would be contaminated following an accident.
2. A leakage reduction program will be implemented on the remaining systems which may be required to process contaminated fluids following an accident, specifically the engineered safeguards systems and the sampling systems.

These results were discussed with Consumers Power Company on November 9, 1979, and it was agreed this was a technically desirable approach. The specific modifications agreed to, together with the pertinent background and basis for the modification, are described in Attachment A. For convenience, the recommended modifications are summarized below:

° Reactor Coolant System Degassing

Modifications will be made to permit venting the pressurizer effluent to the containment atmosphere, most likely via the quench tank. These modifications would permit degassing of the primary coolant system within containment, if required following an accident. Currently, this pressurizer effluent is cooled in the sampling system and routed to the volume control tank or to the vacuum degasifier.

° Sampling Systems

- Sampling system effluents will be returned to containment, rather than to chemical and volume control system and waste gas system components.
- Sampling lines from engineered safeguards equipment to the existing liquid sampling panel will be deleted or isolated within the safeguards rooms.
- A single sampling line from the engineered safeguards pump discharge to the new liquid sampling system will be considered. This line would enable a reactor coolant sample to be obtained when reactor coolant system pressure is low following a major accident.

° Safeguards Room Sump Effluent

The sump effluent will be returned to containment in the event of a serious accident. The effluent is currently routed to the dirty waste tank outside the safeguards room. An alternate route is to the auxiliary building equipment drain tank. Either of these routes could cause contamination of numerous additional auxiliary building components and will not be used following an accident.

° Safeguards Equipment Relief Valve Discharge

The discharges from engineered safeguards component relief valves will be routed to the safeguard room sumps. These discharges are currently routed to the auxiliary building equipment drain tank located outside of the safeguards room.

Mr. R. A. Vincent

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December 4, 1979

° High Pressure Air Supply to Safeguards Equipment Valve Operators

A modification will be made to prevent air from the safeguards rooms to enter the turbine building via the intertie between the high pressure air compressor located in the east safeguards room and a similar compressor located in the turbine building.

° Letdown Line Isolation based on Failed Fuel Monitor

Plant personnel indicated that they are considering an additional possible modification to ensure that the letdown line is isolated whenever significant quantities of radioactive material are present in the reactor coolant. This would involve adding one additional isolation signal derived from the failed fuel monitor.

In addition to Attachment A, this letter contains the following two attachments:

° Attachment B -- A checklist of each line which penetrates containment. This checklist was used to identify each system which could potentially contain radioactive material in the event of a severe accident, presuming no plant modifications are accomplished.

° Attachment C -- A checklist of each potentially radioactive system identified in Attachment B. This checklist was prepared based on review of system P&ID's, plant operating procedures, and detailed discussions with Consumers Power Company personnel. It presumes that plant modifications are made, and is used to identify the systems or portions of systems which could contain radioactivity. This checklist also identifies various functions which must be performed following an accident, and how they would be accomplished.

As part of developing the plan outlined in Attachment A, we performed several evaluations which will be transmitted by separate correspondence. These include:

- ° An evaluation of pressurizer degassing capability during potential accident conditions.
- ° An evaluation of the off-site doses resulting from operation of the steam generators with a tube leak.

MPR ASSOCIATES, INC.

Mr. R. A. Vincent

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December 4, 1979

- ° An evaluation of the off-site doses resulting from leakage of contaminated water from safeguards equipment.

It is our understanding that various plant modifications described in Attachment A are currently being detailed by various Consumers Power personnel or by other organizations. We would be pleased, if you wish, to review these designs, and to assist you in the design of specific modifications.

Finally, as we have discussed in recent telephone conversations, you will establish a mutually convenient date for us to visit Palisades to continue with our efforts on the leakage reduction program.

If you have any questions concerning this letter or its attachments, please do not hesitate to contact us.

Sincerely,

Alex Jancubak for:
David G. Strawson

cc: R. B. DeWitt
CPCo, Jackson, MI
J. G. Lewis
CPCo, Covert, MI

REVIEW OF
SYSTEMS OUTSIDE CONTAINMENT OF THE PALISADES PLANT
WHICH COULD CONTAIN RADIOACTIVE FLUIDS
IN THE EVENT OF SIGNIFICANT CORE DAMAGE

A review has been performed to determine which of the systems located outside containment would or could contain highly radioactive fluids during a serious transient or accident. In particular, this review identifies the systems included in the leakage reduction program being implemented in accordance with Nuclear Regulatory Commission Recommendation 2.1.6.a in "TMI-2 Lessons Learned Task Force Report (Short Term)," NUREG-0578, and subsequent clarifications of NUREG-0578 contained in the NRC letter dated October 30, 1979, "Discussions of Lessons Learned Short Term Requirements."

The review also identifies the systems which will not be included in a leakage reduction program. It is considered that the delineation of systems to be covered in the leakage program is in full compliance with the Nuclear Regulatory Commission recommendations.

The type accidents considered for this review include a TMI-2 type of accident in which the initial path for loss of reactor coolant was isolated. Also considered were non-isolable breaks in the reactor coolant pressure boundary, either large or small in size, which would eventually require recirculation of reactor coolant from the containment sump to achieve reactor

cooling and containment spray. For each type of accident, the functions which must be performed by various plant systems outside of containment to achieve a safe shutdown condition were considered.

In summary, the review indicates that a number of systems outside containment could contain radioactivity following an accident, based on their current configuration. Some of these systems, such as the emergency core cooling systems, would clearly be required under certain accident conditions, and cannot reasonably be modified to avoid contamination. Accordingly, such systems will be included in the leakage reduction program. The systems in this category are identified in Paragraph A below.

Contamination could be avoided in other plant systems, such as the chemical and volume control system, by implementing plant modifications. It is considered that such modifications, which permit radioactive materials to be kept inside containment, are the technically desirable means for minimizing off-site exposures. Such modifications would eliminate significant off-site exposure as a result of leakage, and would also avoid potential exposures as a result of operator error (e.g., valve misalignments), or equipment malfunctions (e.g., decay tank overpressurization). Such modifications would also minimize exposure of on-site personnel because they significantly reduce the potential exposure from radioactive components -- pumps, tanks,

pipings, etc. Accordingly, it is planned, where practicable, to implement appropriate modifications. Clearly, leakage from such systems will be kept as low as practical as a matter of normal plant practice, in order to minimize waste processing requirements and personnel exposure during operation and maintenance. The specific modifications which will be made are identified in Paragraph B below. Paragraph C summarizes the systems which can avoid contamination as a result of the modifications, and identifies the means which would be used to accomplish various functions which may be required following an accident.

A. Systems Included in Leakage Reduction Program

1. Safety Injection and Containment Spray

The following systems located outside containment would be required to operate in the recirculation mode (draw potentially radioactive water from the containment sump and return it to the reactor coolant system or containment) following a non-isolable break in the reactor coolant pressure boundary. Accordingly, these systems will be included in the leakage reduction program:

- ° High Pressure Safety Injection (HPSI)
- ° Low Pressure Safety Injection (LPSI)
- ° Containment Spray

2. Shutdown Cooling

Operation of the LPSI system in the shutdown cooling mode (draw potentially radioactive water from the reactor coolant system and return it to the reactor coolant system) would be employed as a backup to the steam generators to achieve core cooling following an isolable break in the reactor coolant system pressure boundary. Accordingly, the additional portions of LPSI associated with the shutdown cooling mode will be included in the leakage reduction program.

3. Sampling

The systems employed for obtaining samples of reactor coolant and of the containment atmosphere will be required following an accident and, accordingly, will be included in the leakage reduction program. In this regard, these systems will be modified in accordance with the TMI-2 Lessons Learned Task Force Recommendation 2.1.8.a. New sample lines and panels will be employed. Accordingly, the new systems, rather than the existing sampling systems, would be included in the leakage reduction program.

Modifications to sampling systems which minimize the likelihood that radioactive sample materials could escape to the environment are identified in Paragraph B below.

B. Plant Modifications

Plant modifications to minimize transport of radioactive materials to systems outside containment are identified below.

1. Degassing of Primary Coolant

Degassing could be required as it was at TMI-2 to remove excess hydrogen from the reactor coolant. The current degassing method employed at Palisades during normal shutdowns is to withdraw steam and excess hydrogen from the pressurizer gas space via a sampling line to the sample system outside containment. The fluid is cooled by sampling system coolers and is then returned to the volume control tank in the chemical and volume control system. The cooled fluid may also be returned to the vacuum degasifier. In either case, the gas is then passed to the waste gas surge tank, compressed, and stored in the gas decay tanks.

This degassing method will be modified to avoid unnecessary contamination of portions of the chemical and volume control system, the vacuum degasifier, and waste gas system components. In particular, effluents from the pressurizer gas space during degassing will be vented to the containment atmosphere, most likely via the quench tank. This modification will be accomplished in conjunction with

the installation of a pressurizer vent in accordance with TMI-2 Lessons Learned Task Force Recommendation 2.1.9, Reactor Coolant System Venting.

2. Sampling Systems

Modifications to the new and existing sampling systems are as follows:

- ° The new sampling systems will contain provisions to return liquid and gas sampling system effluents to containment rather than to systems outside containment.
- ° Sampling lines from engineered safeguards equipment to the existing liquid sampling panel will be deleted or isolated within the safeguards rooms.
- ° A single sampling line to the new liquid sampling system from the engineered safeguards pump discharge will be installed. This line would enable a reactor coolant sample to be obtained when reactor coolant system pressure is low.

3. Safeguards Room Sump Effluent

The two rooms which contain the HPSI, LPSI, and containment spray systems each contain a sump which collects any leakage from the equipment. As currently designed, effluent from these sumps is pumped

automatically on high sump level to tanks outside of the safeguards room. The normal route is to the dirty waste tank, which is vented via the auxiliary building gas collection header to the plant stack. An alternate sump discharge path, not normally used, is to the equipment drain tank, which is vented to the waste gas surge tank. In either case, noble gases from sump pump effluent, as well as the liquid, would be transported to numerous additional auxiliary building components.

Accordingly, a modification will be made to return safeguards room sump effluent to containment, and thus avoid extensive contamination of other auxiliary building components.

4. Safeguards Equipment Relief Valve Discharge

The relief valves for various components of the engineered safety systems currently discharge to the equipment drain tank which, as noted above, is located outside the safeguards room, and is vented to the waste gas surge tank. A modification will be made to reroute these discharges to the safeguards room sumps. In conjunction with the sump effluent modification described above, these discharges would be returned to containment.

5. High Pressure Air Supply to Safeguards Equipment Valve Operators

The high pressure air supply to various control valves for engineered safeguards equipment is currently intertied with a similar system in the turbine building. The current system allows the air compressors located in the turbine building and the safeguards room to serve as backups to each other. The turbine building compressor can supply engineered safeguards system valves and, alternatively, the safeguards room compressor can supply valves in the turbine building.

With such an arrangement, contaminated air from the safeguards room could enter the turbine building. Accordingly, a modification will be made to prevent flow of air from the safeguards room to the turbine building.

C. Systems which Avoid Contamination

The modifications identified above prevent contamination of various plant systems as noted above and, accordingly, it is considered that each of these systems need not be included in the formal leakage reduction program. These are:

- The Chemical and Volume Control System
- The Liquid Waste System
- The Vacuum Degasifier System

° The Waste Gas System, including the waste gas compressors and decay tanks, and the auxiliary building gas collection header

These systems perform various functions during normal operation, some of which would be required in the event of a serious accident. The modified method of performing various required functions following an accident is summarized below. In the following discussion, "design basis" accident refers to breaks in the reactor coolant system which cannot be isolated, and "TMI type" accident refers to breaks which can be isolated.

1. Chemical and Volume Control System (CVCS)

a. Letdown of Reactor Coolant

During normal operation, letdown of reactor coolant is accomplished in order to (1) purify the reactor coolant in the purification ion exchangers, (2) deborate the reactor coolant in the deborating ion exchangers during plant start-ups, and (3) remove excess reactor coolant to maintain pressurizer water level. Following a major accident, the letdown line would be isolated by high containment radiation (CHR) or high containment pressure (CHP). Operator action would be required to re-establish the letdown flow. This action will not be needed during or immediately following an accident. In particular:

Purification: Purification of the primary coolant would not be required during the course of an accident or immediately following an accident. It is expected that, as at TMI, specialized systems would be required to achieve eventual cleanup of highly contaminated fluids. These systems would utilize highly shielded process components, special spent waste handling equipment, etc., and would be thoroughly leak tested before use. Reactor coolant purification would not be accomplished in the CVCS.

Deboration: This function would not be required following a design basis or TMI type of accident.

Excess Coolant Removal: Removal of coolant would not be required following a design basis or TMI type of accident. During both a design basis accident and a TMI type accident, there is a requirement to add water to the PCS to replace coolant which has been lost, rather than to letdown. If an unforeseen requirement to release fluid from the PCS arises, this may be accomplished through the pressurizer relief or vent valve to the quench tank or to the containment sump.

b. Reactor Coolant Pump Seal Leakage Collection

During normal operation, reactor coolant pump seal leakage is diverted to the volume control tank. On high containment pressure or radiation, the leakoff line is isolated and the leakage is automatically directed to the primary system drain tank inside containment. This flow path would be maintained following an accident so as to keep the radioactive fluids inside containment. (Note that there is no seal injection system at Palisades such as at TMI.)

In this regard, it is understood that the current Combustion Engineering recommendation is to keep two pumps operating in the event of an accident. If this recommendation is implemented, the leakage flow would be approximately 1/2 to 1 gpm (about 700 to 1,400 gallons per day). Depending on the duration of pump operation, this could cause release of drain tank contents to the containment sump. This is considered preferable to permitting transport of the leakage flow outside of containment.

c. Degassing and Sampling

The degassing and sampling effluents during normal operation are routed to the volume

control tank. As discussed in Paragraph B above, plant modifications will be made to ensure that these effluents are kept within containment.

d. Reactor Coolant Makeup

Makeup of reactor coolant via the charging pumps would be performed following a design basis or TMI type of accident. However, the sources of makeup water would not be radioactive. These sources include the primary system makeup storage tank or, alternatively, the safety injection and refueling water (SIRW) storage tank.

e. Reactor Coolant System Boration

Boration of the reactor coolant system by CVCS equipment could be required following a TMI type accident to achieve cold shutdown. (This function would be accomplished following a design basis accident by safety injection components drawing borated water from the SIRW tank.) Boration would be accomplished by supplying highly borated water from the concentrated boric acid tanks. These tanks are required by the Technical Specifications to contain at least 6,540 gallons at 10,936 ppm, which is adequate to achieve cold shutdown. The borated water

would not be significantly radioactive as a result of the accident.

It is also noted that addition of the water would not impose a letdown requirement on the CVCS. Reactor coolant shrink as a result of cooldown amounts to about 22,000 gallons, which is significantly greater than the volume of borated water.

f. Boron Concentration Monitoring

A boronometer is employed during normal operation to monitor the boron concentration of the reactor coolant. This function would be performed following an accident by the new sampling system.

2. Liquid Waste System

During normal operation, this system collects liquids from various equipment and sumps both inside and outside containment, and processes those liquids as required to permit their discharge or re-use in the plant. Our review indicates the following:

- ° In the event of a serious accident, each line from collection tanks or sumps inside containment is isolated by CHR and/or CHP, and need not be reopened.

- ° Based on the modifications discussed in Paragraph B above (rerouting of engineered safeguards room sump effluent, engineered safeguards component relief valve discharge, and liquid sampling system effluents back to containment), none of the equipment outside of containment which could contain highly radioactive fluids would transmit fluids to the liquid waste system.

Accordingly, the liquid waste system would not be contaminated following an accident and its use would not be required. As noted above, it is expected that eventual cleanup of highly contaminated fluids would be accomplished by specialized equipment such as at TMI so that the liquid waste system would not be required for this function.

3. Vacuum Degasifier System

This system is normally employed to remove gases from various contaminated liquids before they are transported to liquid waste storage tanks. These include liquids from equipment drain tanks, both inside and outside containment, and sample system liquid discharges. Our review indicates:

- ° The line from the primary system drain tank inside containment is isolated by CHP and/or CHR following a serious accident, and need not be reopened.

- ° The line from the auxiliary building equipment drain tank will not contain highly radioactive materials, based on the modifications to the engineered safeguards room sump effluent routing and relief valve discharge routing.
- ° The line from the sampling system will not contain highly radioactive fluid, based on rerouting the sampling system discharge effluent.

Accordingly, the vacuum degasifier system will not be contaminated following an accident.

4. Gaseous Waste System

This system contains two subsystems for collection of potentially radioactive waste gases. One subsystem consists of the waste gas surge tank, compressors, and decay tanks. The other subsystem consists of the auxiliary building gas collection header. Inputs to these subsystems are as follows:

Waste Surge Tank Inputs

- ° Containment vent header, which isolates on CHP and/or CHR.
- ° Nitrogen addition.
- ° Dirty waste drain tank relief valve discharge.
- ° Auxiliary building equipment drain tank vent.
- ° Vacuum degasifier gas discharge.
- ° Containment atmosphere sampling gas discharge.

- CVCS purification and deborating demineralizer vents.
- Radwaste evaporator vents.
- CVCS volume control tank vent.
- Gas decay tank relief valve discharge.
- Waste gas compressor discharge header relief valve effluent.

Gas Collection Header

- Containment clean waste receiver tank vent, which isolates on CHP and/or CHR.
- Dirty waste tank vent.
- Controlled chemical lab tank vents.
- Radwaste demineralizer vents.
- Filtered waste monitor tank vent.
- Treated waste monitor tank vent.
- Volume control tank N₂ and H₂ supply header relief valve vent.
- Component cooling water surge tank vent.
- Charging pump seal loop tank and well vents.
- Radwaste building vent gas collection header.

Based on our review, each input from containment is isolated following a serious accident. As a result of the plant modifications discussed above, none of the other inputs would be radioactive. Accordingly, the waste gas system will not be contaminated following an accident.

In summary, our review has indicated that a relatively few systems outside of containment could become highly radioactive following a serious accident. These include the engineered safeguards systems and the sampling systems, which will be included in a formal leakage reduction program. In addition, the specific plant modifications discussed above will be made to ensure that highly radioactive fluids will be kept within containment. This is considered a technically desirable approach to minimize both off-site and on-site radiation exposures. It is considered that this approach is fully responsive to the Nuclear Regulatory Commission recommendation.

REVIEW OF CONTAINMENT PENETRATIONS

A review was made of each line which penetrates containment to determine, for the current plant configuration, which systems outside containment could potentially contain radioactive material in the event the reactor coolant or containment atmosphere were highly contaminated.

The containment penetration configurations have been recently verified, as described in MPR Report 639, "Palisades Nuclear Plant - Independent Review of Containment Penetrations," dated November 1979. The penetration configurations described in MPR-639 were employed for the current review. The results of that review are documented in the enclosed checklist, which indicates for each penetration:

- ° The containment penetration number.
- ° The system name and associated P&ID number.
- ° The signals, if any, which would isolate the line.
- ° An indication of whether or not fluids in the line would be potentially radioactive.
- ° A conclusion as to whether or not the associated system outside containment would potentially contain highly radioactive materials in the event of a serious accident, and the basis for that conclusion.

In summary, the review indicates that, without any plant modifications, the following systems outside containment could contain radioactive materials:

- ° High pressure safety injection system in the recirculation mode.
- ° The containment spray system in the recirculation mode.
- ° Low pressure safety injection system in the recirculation mode.
- ° The containment air sampling system.
- ° The chemical and volume control system.
- ° Reactor coolant sampling system.

- The vacuum degasifier system.
- The liquid waste system. (1)
- The gaseous waste system. (1)

(1) These systems are not directly contaminated by fluids which escape through containment penetrations, but would be contaminated by processing fluids from other systems outside containment.

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
1	Purge air supply (M-218)	Blower shuts on CHP* and CHR*; valves normally closed	No	Not a source (not radioactive)
2	Main steam line (M-205, M-207)	Steam Generator A or B low pressure	No (except if tube leak at high pressure)	Not a source (can isolate if a tube leaks - evaluations indicate doses due to short-term operation with a leak are acceptable)
3	Main steam line (M-205, M-207)	Steam generator A or B low pressure	No (except if tube leak at high pressure)	Not a source (can isolate if a tube leaks - evaluations indicate doses due to short-term operation with a leak are acceptable)
4	Purge air exhaust (M-218)	CHP and CHR	Yes	Not a source (isolates)
4A	Purge air exhaust sample line (M-218)	Locked closed	Yes	Not a source (locked closed)
5	SG(A) bottom blow (M-207/M-226)	CHP and CHR (also normally closed)	No (except if tube leak at high pressure)	Not a source (normally non-radioactive and closed; also isolates)
6	SG(B) bottom blow (M-207/M-226)	CHP and CHR (also normally closed)	No (except if tube leak at high pressure)	Not a source (normally non-radioactive and closed)
7	Feedwater to SG(A) (M-207)	Open	No	Not a source (not radioactive)
8	Feedwater to SG(B)	Open	No	Not a source (not radioactive)
9	Spare	---	---	---
10	Service air (M-212)	Locked closed	No	Not a source (not radioactive; locked closed)

*CHP = Containment High Pressure

CHR = Containment High Radiation

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
11	Condensate to shield cooling surge tank (M-221)	CHP and CHR	Yes (when clean fluid not being pumped in)	Not a source (isolates)
12	Service water supply (M-208) (to containment air coolers)	Remain open	No	Not a source (not radioactive)
13	Service water return (M-209)	Remain open	No	Not a source (not radioactive)
14	Component cooling water in (M-209) (letdown HX, RC pump HX)	Remain open	No	Not a source (not radioactive)
15	Component cooling water out (M-209)	Remain open	No	Not a source (not radioactive)
16	SG(A) surface blowdown (M-207 and M-226)	CHP and CHR	No (unless a tube leak)	Not a source (isolates)
17	Containment pressure instruments (M-218)	Locked open	Yes	Is a source, but tested during containment ILRT
18	Fuel transfer tube	Locked closed	Yes	Not a source (locked closed)
19	Personnel lock	Double isolation doors	Yes	Not a source (double doors)
20	Spare	---	---	---
21	Spare	---	---	---

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
22	Redundant HP safety injection (M-203 and M-204)	Open on SIS	Yes (in recirculation mode)	Is a source
23	Redundant HP safety injection (M-203 and M-204)	Open on SIS	Yes (in recirculation mode)	Is a source
24	Spare	---	---	---
25	Clean waste receiver tank vent to stack (M-210)	CHP and CHR	Yes	Not a source (isolates)
26	Nitrogen to quench tank (M-222)	CHP and CHR	Yes (if at low pressure)	Not a source (isolates)
27	ILRT fill line	Blind flange during normal operation	Yes	Not a source (blind flange)
28	Containment air sample (M-224)	Normally closed solenoid valves	Yes	Is a source (would need for sampling)
29	Spare	---	---	---
30	Containment spray pump discharge (M-203)	Opens on high containment pressure	Yes (in recirculation mode)	Is a source
31	Containment spray pump discharge (M-203)	Opens on high containment pressure	Yes (in recirculation mode)	Is a source
32	LP safety injection	Opens on SIS	Yes (in recirculation mode)	Is a source

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
33	Safety injection tank drain (M-204)	Locked closed	No	Not a source (locked closed)
34	Spare	---	---	---
35	Shutdown cooling return (M-204) from primary loop 2	Can be opened	Yes	Is a source
36	Letdown purification ion exchanger (M-202)	CHP and CHR	Yes	Not immediately, could be if use letdown
37	Primary system drain tank pump return (M-210)	CHP and CHR	Yes	Not a source (isolates)
38	Condensate return from steam heating units (M-215)	CHP and CHR	No	Not a source (isolates, not radioactive)
39	Containment heating system steam supply (M-215)	CHP and CHR, also spool piece	No	Not a source
40	Reactor coolant sample line (M-219) (pressurizer vapor and liquid, loop 2 hot leg, quench tank vapor and liquid)	CHP and CHR	Yes	Is a source (isolates initially, but needed for sampling)
41	Degassifier pump discharge (M-210)	CHP and CHR	Yes, from sampling system	Is a source
42	Demineralizer water to quench tank (M-204)	CHP and CHR	No	Not a source (isolates, not radioactive)

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
43	Spare	---	---	---
44	Controlled bleedoff from RCPs (M-202)	CHP and CHR	Yes	Not a source, unless conclude controlled bleedoff to letdown instead of primary coolant drain tank is needed
45	Charging pump discharge (M-202)	Remains open	No, unless use letdown	Not a source, unless use letdown
46	Containment vent header (M-211)	CHP and CHR	Yes	Not a source (isolates)
47	Primary system drain tank pump suction (M-210)	CHP and CHR	Yes	Not a source (isolates)
48	Containment pressure instrumentation	Stays open	Yes	Is a source, but covered by containment ILRT
49	Clean waste receiver tank circulating pump suction (M-210)	CHP and CHR	Yes	Not a source (isolates)
50	Emergency access	Double doors	Yes	Not a source (double doors)
51	Equipment door	Closed	Yes	Not a source (closed)
52	Containment sump drain to dirty waste tank (M-211)	SIS and CHR	Yes	Not a source (isolates)

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
53	Containment spray pump suction (M-204)	Opens on SIRW tank low level	Yes, in recirculation mode	Is a source
54	Containment spray pump suction (M-204)	Opens on SIRW tank low level	Yes, in recirculation mode	Is a source
55	SG(B) surface blowdown (M-207 and M-226)	CHP and CHR	No, unless SG tube leak	Not a source
56	Spare	---	---	---
57	Spare	---	---	---
58	Spare	---	---	---
59	Spare	---	---	---
60	Spare	---	---	---
61	Spare	---	---	---
62	Spare	---	---	---
63	Spare	---	---	---
64	Reactor cavity fill and recirculation (M-221)	Locked closed	No	Not a source (locked closed)
65	Instrument air (M-212)	Stays open	No	Not a source (not radioactive)
66	ILRT instrument line	Locked closed	Yes	Not a source (locked closed)
67	Clean waste receiver tank pump recirculation return (M-210)	CHP and CHR	Yes	Not a source (isolates)

REVIEW OF LINES WHICH PENETRATE CONTAINMENT AS POTENTIAL SOURCES OF LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

CONTAINMENT PENETRATION NUMBER	SYSTEM NAME (P&ID NO.)	ISOLATION SIGNAL	POTENTIALLY RADIOACTIVE	CONCLUSION
68	Air supply to air room (M-218)	CHP and CHR	No	Not a source
69	Clean waste receiver tank pump suction (M-210)	CHP and CHR	Yes	Not a source
70	Spare	---	---	---
71	Spare	---	---	---
72	Reactor refueling cavity drain (M-221)	Locked closed	Yes	Not a source (locked closed)
73	Spare (with capped pipe in place)	---	---	---

REVIEW OF POTENTIALLY RADIOACTIVE SYSTEMS

A review was made of each potentially radioactive system outside of containment to determine the specific systems or portions of systems which would contain radioactive materials following a severe accident. This review also identified the various functions which must be performed following an accident, and the means for accomplishing these functions with available equipment.

The results of the review are documented in the enclosed checklist. This checklist presumes that the plant modifications described in Attachment A will be implemented. The checklist includes each potentially radioactive system identified in Attachment B, and for each includes:

- The system name and associated P&ID number.
- The subsystem involved, e.g., specific lines, or portions of the system which perform a unique function.
- The review results for a design basis accident, which is considered to be a break in the reactor coolant system pressure boundary, either large or small in size, which cannot be isolated. The results indicate whether subsystem components would be radioactive, and defines how various functions are accomplished.
- The review results for a TMI type accident, which is considered to be an isolable break in the reactor coolant pressure boundary.

The results of the review indicate that the following systems could contain radioactivity following a severe accident:

- The high pressure safety injection system.
- The containment spray system.
- The low pressure safety injection system.
- The reactor coolant and containment atmosphere sampling systems.

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT	TMI-TYPE INCIDENT
<p>Safety Injection, Containment Spray Shutdown Cooling (M-203, M-204)</p>	<p>HP safety injection: from containment sump through HP pumps to HP safety injection containment penetration 23</p>	<p>Contents will be radioactive: operated in recirculation mode (suction from containment sump) following depletion of SIRW tank.</p>	<p>Contents not expected to be radioactive; recirculation from containment sump is not required.</p>
	<p>Redundant HP safety injection: from HP pumps to redundant HP safety injection containment penetration 22</p>	<p>Contents will be radioactive; may be required to operate in recirculation mode.</p>	<p>Contents not expected to be radioactive; recirculation from containment sump is not required.</p>
	<p>LP safety injection: from containment sump through LP pumps through shutdown cooling heat exchangers to LP injection containment penetration 32</p>	<p>Contents will be radioactive: operation in recirculation mode may be required for long-term cooling of primary coolant system.</p>	<p>Contents may be radioactive: operation in recirculation mode may be required for long-term cooling.</p>
	<p>Containment spray: from containment sump through containment spray pumps, through shutdown heat exchangers to containment spray containment penetrations 30 and 31</p>	<p>Contents will be radioactive: operation in recirculation mode may be required for reduction of containment pressure and for iodine reduction in containment atmosphere.</p>	<p>Contents not expected to be radioactive; recirculation from containment sump not required.</p>
	<p>Containment spray recirculation line from shutdown heat exchangers to SIRW tanks</p>	<p>Contents not radioactive: this line is used for periodic pump test but is normally locked closed and would be closed during and following an accident.</p>	<p>Contents not radioactive (same as for DBA)</p>

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT	TMI-TYPE INCIDENT
Safety Injection Containment Spray Shutdown Cooling (M-203, M-204) (continued)	Connecting lines from discharge of shutdown heat exchangers to suction of HP safety injection pumps	Contents radioactive: operation in recirculation mode may be required for long-term cooling of primary coolant system	Contents not expected to be radioactive: recirculation from containment sump not required.
	Recirculation (minimum flow) lines from discharge of HP, LP and containment spray pumps to SIRW recirculation inlet valves.	Contents are radioactive up to SIRW recirculation inlet valves.	Contents are radioactive (same as for DBA).
	SIRW tank	Contents not radioactive, SIRW inlet lines are isolated when recirculation taken from containment sump.	Contents not radioactive (same as for DBA)

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT	TMI-TYPE INCIDENT
Chemical and Volume Control [M-202]	Letdown system: from letdown heat exchanger through containment penetration 36 through purification filters to volume control tank	Not radioactive: these lines are isolated upon high radiation signal and would not be re-opened as long as primary coolant is highly radioactive.	Not radioactive: letdown not required following TMI-type accident.
	Volume control tank	Not required	<p>Volume control tank is normally used in degassing and sampling primary system and collecting primary coolant pump seal leakage. Plant modifications will be made to allow:</p> <ul style="list-style-type: none"> - Degassing to be accomplished in pressurizer with venting to containment atmosphere (addition of remotely operated control valve probably required for pressurizer). - Sampling to be accomplished with special sample system (see table for sample system) <p>In addition, primary coolant pump seal leakage will be allowed to collect in primary coolant drain tank and containment sump if required.</p>
	Line from volume control tank through charging pumps to containment penetration 45 (to regenerative heat exchangers)	Not radioactive: although charging pumps can be actuated by safety injection signal, suction will be from non-radioactive supply. Boron addition is accomplished by the engineered safeguards equipment.	Not radioactive as long as volume control tank is not radioactive (see above). Boron addition is accomplished by the high concentration boric acid tanks.
	Borometer piping	Not radioactive: boron concentration of primary coolant will be determined by primary coolant sample analysis (see table for sample system)	Not radioactive: (same as for (DBA))

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT	TMI-TYPE INCIDENT
Chemical and Volume Control (M-202) (continued)	Purification Ion Exchangers	Not required	Not required
	Deborating Ion Exchanger	Not required	Not required

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT	TMI-TYPE INCIDENT
<p>Sampling System (M-219 and M-224)</p>	<p>(M-219) NSSS sample line from containment penetration 21 through sample coolers to volume control tank (M-202) or vacuum degasifier tank (M-210)</p>	<p>Samples for radioactive primary coolant will be required. New sample lines and panel will be installed for use with highly radioactive fluid. Sample fluid will be returned to containment.</p>	<p>Samples of radioactive primary coolant will be required (same as for DBA). New sample panel will be used.</p>
	<p>Miscellaneous liquid sample lines from the following: containment spray pump discharge, LP injection pump discharge, HP injection pump discharge, SI tanks drain</p>	<p>These lines could be radioactive to the first isolation valve (near sample panel). Modifications will be made to eliminate these lines or isolate within engineered safeguards rooms. One special sample line at pump discharge should be considered for use in the event that PCS pressure is too low to provide PCS sample.</p>	<p>These lines could be radioactive to the first isolation valve (near sample panel). Modifications will be made to eliminate these lines or isolate within engineered safeguards rooms. These lines will not be used following TMI-type accident.</p>
	<p>Purification ion exchanger inlet purification filter outlet, purification ion exchanger outlet SIRW tank recirculation</p> <p>(M-224) Containment building atmosphere sample line from containment penetration 28 through vacuum pumps and moisture separators, H₂ and O₂ analyzers to waste gas surge tank (M-211) and equipment drain tank (M-210)</p>	<p>Not radioactive: sources will not be contaminated (see table on chemical and volume control system)</p> <p>Samples of radioactive containment atmosphere will be required. New sample lines and panel will be installed for use with highly radioactive gas. Sample fluid will be returned to containment.</p>	<p>Not radioactive: (same as for DBA)</p> <p>Samples of radioactive containment atmosphere will be required; (same as for DBA)</p>

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT	TMI-TYPE INCIDENT
Sampling System (M-219 and M-224) (continued)	Miscellaneous gas sample lines from the follow- ing: Volume control tank Waste gas surge tank Waste gas decay tanks	These lines will not be radioactive since the sources will not be contaminated (see tables on chemical and volume control system and waste gas system)	These lines will not be radioactive (same as for DBA)

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT (DBA)	TMI-TYPE INCIDENT
Waste Gas - Vacuum Degasifier (M-210)	Input from primary system drain tank (located inside containment)	Not radioactive -- the line is isolated by CHP or CHR, and does not need to be reopened.	°Same as DBA.
	Input from auxiliary building equipment drain tank	Not radioactive -- modifications will be made to prevent radioactive inputs from: -- Safeguards room sump, and -- Safeguards equipment relief valves	°Same as DBA.
	Letdown from chemical and volume control system upstream of volume control tank (letdown can go to either the VCT or vacuum degasifier)	Not radioactive -- the letdown system will not be used following a DBA.	°Same as DBA. Note that reactor coolant degasification will not require use of the letdown system. This will be accomplished by venting the pressurizer inside containment.
	Nitrogen addition line	Not radioactive.	°Not radioactive.
	Nuclear steam supply sampling system discharge line	Not radioactive -- modifications will be made to return sample system effluent to containment.	°Same as DBA.
	Degasifier tank to degasifier pump suction	Not radioactive, based on modifications noted above.	°Same as DBA.
	Degasifier pump discharge to containment	Not radioactive, based on modifications noted above.	°Same as DBA.
Vacuum pump and associated seal coolant circuit	Not radioactive, based on modifications noted above.	°Same as DBA.	

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

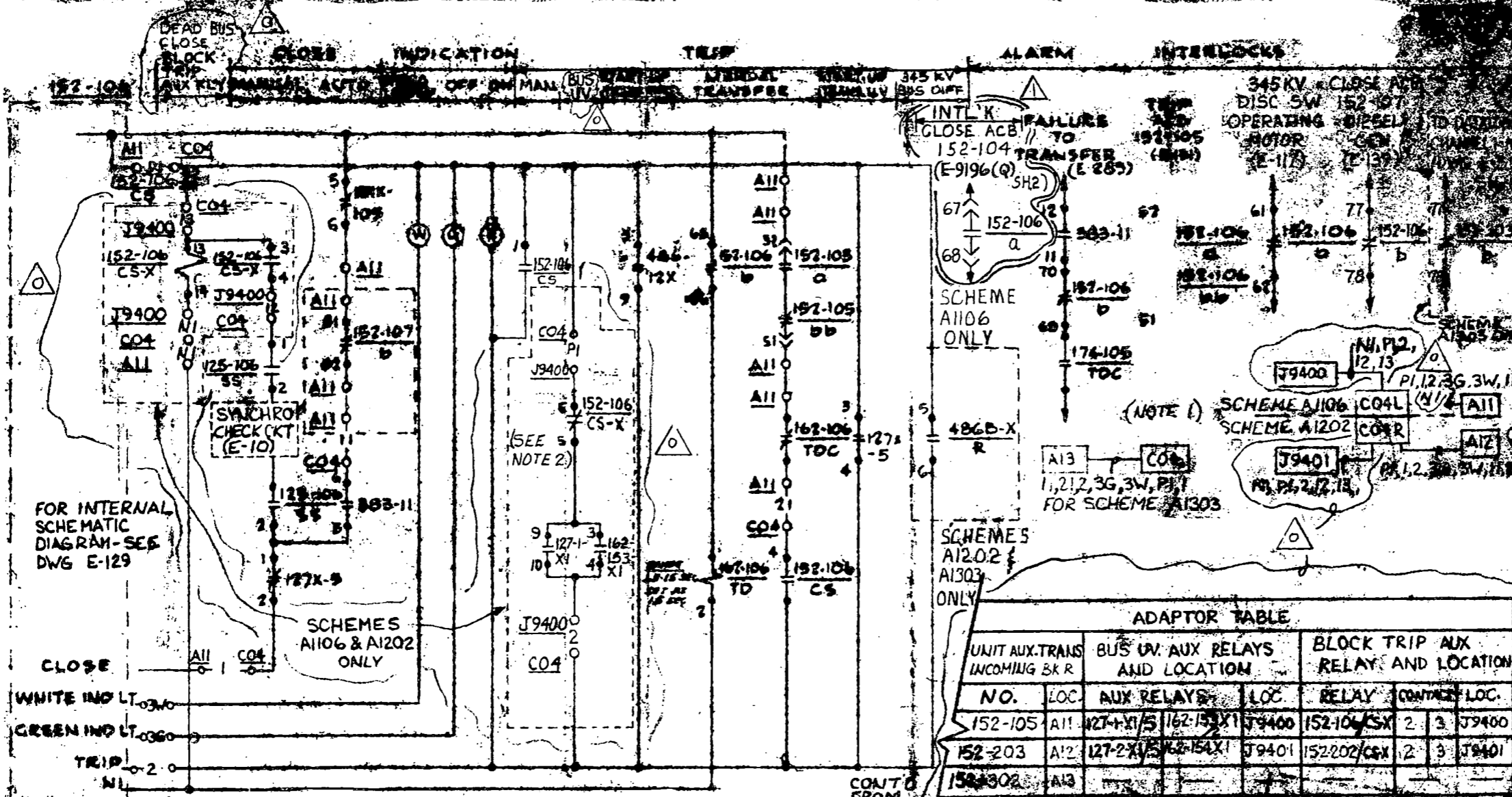
SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT (DBA)	TMI-TYPE INCIDENT
Waste Gas - Surge Tank	°Containment vent header input to surge tank	°Not radioactive. The header isolates on CHP or CHR, and does not need to be re-opened.	°Same as DBA.
	°Nitrogen addition line input to surge tank	°Not radioactive.	°Same as DBA.
	°Dirty waste drain tank relief valve discharge to surge tank	°Not radioactive. A modification will be made to ensure safeguards room sump effluent is not routed to the dirty waste tank.	°Same as DBA.
	°Equipment drain tank vent to surge tank	°Not radioactive. A modification will be made to ensure safeguards room sump effluent is not routed to the equipment drain tank.	°Same as DBA.
	°Vacuum degasifier vacuum pump discharge to surge tank	°Not radioactive, based on sampling system modifications.	°Same as DBA. Also modifications to permit coolant degasification inside containment will eliminate this source.
	°Auxiliary building gas analyzers input to surge tank	°Not radioactive, based on sampling system modifications.	°Same as DBA.
	°Purification and de-borating demineralizer vents to surge tank	°Not significantly radioactive; normally closed.	°Same as DBA.
	°Radwaste evaporator vent	°Not significantly radioactive.	°Same as DBA.
	°Volume control tank vent to the surge tank	°Not radioactive, based on sampling system modifications.	°Same as DBA.
°Waste gas relief valve input to surge tank	°Not radioactive.	°Same as DBA.	
°Waste gas compressor discharge header relief valve input to surge tank	°Not radioactive.	°Same as DBA.	

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT (DBA)	TMI-TYPE INCIDENT
Waste Gas - Surge Tank (continued)	°Surge tank to waste gas compressor suction	°Not radioactive.	°Same as DBA.
	°Waste gas compressors and decay tanks	° Not radioactive.	°Same as DBA.
Waste Gas - Gas Collection Header	°Clean waste receiver tank vent	°Not radioactive. Isolates on CHP or CHR	°Same as DBA.
	°Dirty waste drain tank vent	° Not radioactive. A modification will be made to ensure safeguards room sump effluent is not routed to this tank.	°Same as DBA.
	°Controlled chemical lab tank vents	°Not significantly radioactive.	°Same as DBA.
	°Radwaste demineralizer vents	°Normally closed. Not significantly radioactive.	°Same as DBA.
	°Filtered waste monitor tank vents	°Not significantly radioactive.	°Same as DBA.
	°Treated waste monitor tank vent	°Not significantly radioactive.	°Same as DBA.
	°Volume control tank N ₂ + H ₂ supply header relief valve vent	°Not radioactive.	°Same as DBA.
°Component cooling surge tank vent	°Not radioactive.	°Same as DBA.	

IDENTIFICATION OF SYSTEMS OUTSIDE CONTAINMENT WHICH
COULD CONTAIN RADIOACTIVE FLUID FOLLOWING AN ACCIDENT

SYSTEM	SUBSYSTEM	DESIGN BASIS ACCIDENT (DBA)	TMI-TYPE INCIDENT
Waste Gas - Gas Collection Header (continued)	°Charging pump - seal loop tank vents and well vents	°Not significantly radioactive.	°Same as DBA.
	°Radwaste addition vent gas collection header	°Not significantly radioactive unless waste water transferred to radwaste.	°Same as DBA.



FOR INTERNAL SCHEMATIC DIAGRAM - SEE DWG E-129

CLOSE WHITE IND LT. 0360
 GREEN IND LT. 0360
 TRIP NI 2

ADAPTOR TABLE

UNIT AUX. TRANS INCOMING BK R	BUS W/ AUX RELAYS AND LOCATION	BLOCK TRIP AUX RELAY AND LOCATION
NO.	LOC. AUX RELAYS	RELAY CONTACT LOC.
152-105	AI1 127-1X1/5 162-153X1	J9400 152-106/CSX 2 3 J9400
152-203	A12 127-2X1/5 162-154X1	J9401 152-202/CSX 2 3 J9401
152-302	A13	

CONTACTS POSITIONS

CONTACTS	POSITION
HANDLED	ON OFF
1	
2	
3	
4	

SYNCHRONIZING SWITCH
 G.L. TYPE 3BM, MAINTAINED CONTACT LOCATION 524

2400V. BUS #1C START-UP TRANSFORMER INCOMING BREAKER 152-106 (1536)

SCHEME AI106 ADAPTOR TABLE

DESCRIPTION	C.B.C.O. NO.	ACS NO.	LOC	OCN.	INTLK	TRANSFER	LOCKOUT	START-UP
2400V BUS #1C START-UP TRANSFORMER INCOMING BREAKER 152-106 (1536)	152-106	AI106	106/CS	AI106	106/CS	AI106	106/CS	AI106
2400V BUS #1D START-UP TRANSFORMER INCOMING BREAKER 152-203 (1536)	152-203	A1202	203/CS	A1202	203/CS	A1202	203/CS	A1202
2400V BUS #1E START-UP TRANSFORMER INCOMING BREAKER 152-302 (1536)	152-302	A1303	302/CS	A1303	302/CS	A1303	302/CS	A1303

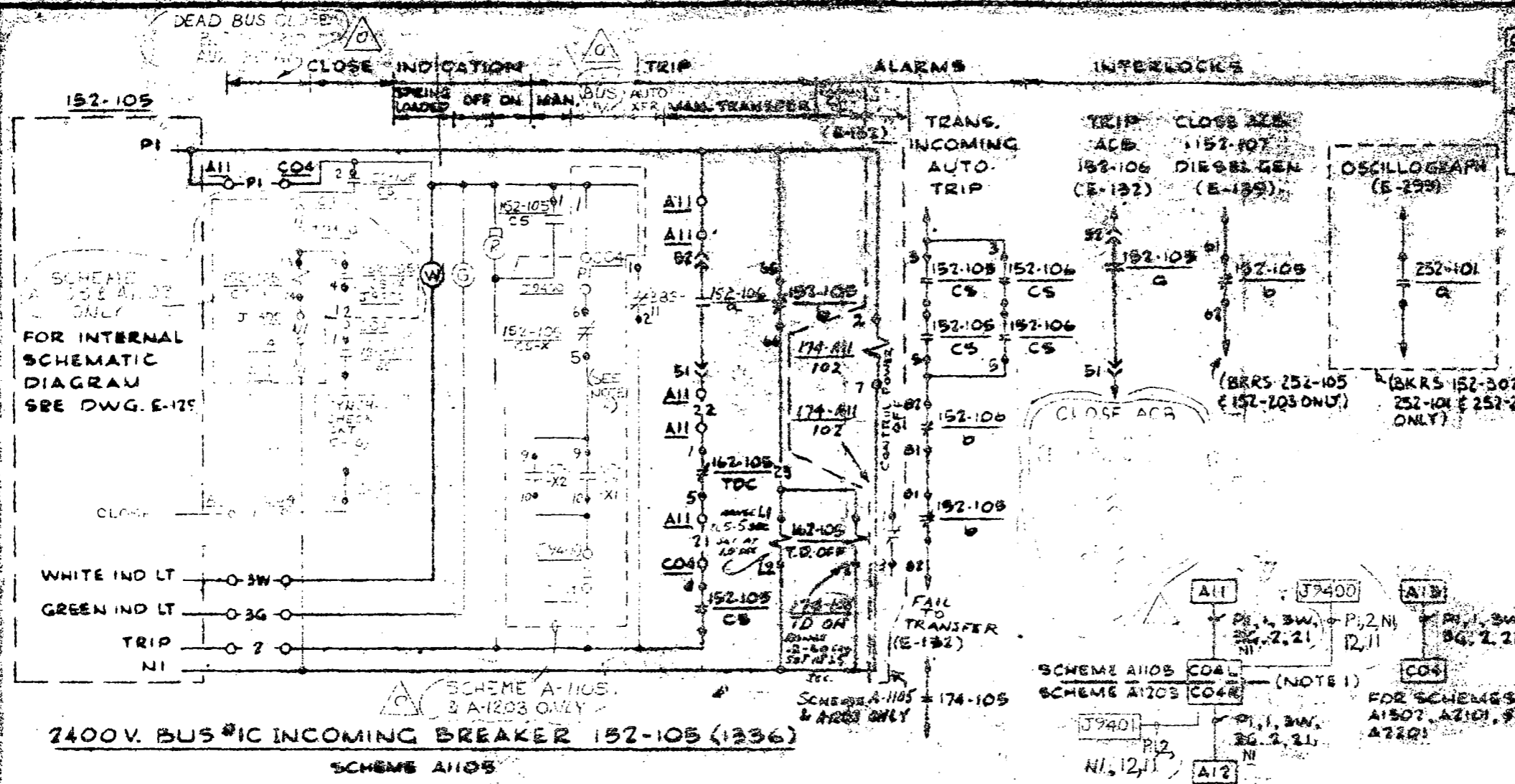
GWO 8428 SHOWN THUS-
 GWO 8538 SHOWN THUS-

WORK ITEM #3

NOTE: GWO 8428 SYSTEMS MODIFICATIONS PER NUREG 0578
 This drawing is controlled by Revision 1 in 9000 series for the duration of the project and will revert to the original number after project completion.

12447-039
 022-91320

1. ADDED INTL. ISSUED FOR CONSTRUCTION (GWO 8538) 1/8/78
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CONTACT	POSITION	LOCATION	TYPE
A11	1		
A110	2		
A111	3		
A12	4		
A13	5		
A14	6		

CONTROL SWITCH 152-105
 G.E. TYPE SEM SPRING RETURN
 LOCATION CO4 * NOT USED

OFF	CONTACTS	POSITIONS	
	HANDLE	ON	OFF
	1	X	
	2	X	
	3	X	
	4	X	
	5	X	
	6	X	

SYNCHRONIZING SWITCH 152-105
 G.E. TYPE SEM MAINTAINED
 CONTACT LOCATION CO4 * SEE DWG. E-10

DESCRIPTION	CP CO. SKR NO.	ACB NO.	LOC.	SCHEME	CONTROL SW		SYNCHRONIZING SW		TRANSFER REL		START/STOP		TRIP		DELAYS		CONTROL PURV			
					NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.
7400V. BUS [#] C INCOMING BRK	(1836)	152-105	A11	A1105	152-105/CS	CO4	152-105/CS	CO4	152-105/CS	CO4	152-105/CS	CO4	152-105/CS	CO4	152-105/CS	CO4	152-105/CS	CO4	152-105/CS	CO4
7400V. BUS [#] D INCOMING BRK	(1836)	152-106	A12	A1206	152-205/CS	CO4	152-205/CS	CO4	152-205/CS	CO4	152-205/CS	CO4	152-205/CS	CO4	152-205/CS	CO4	152-205/CS	CO4	152-205/CS	CO4
7400V. BUS [#] E INCOMING BRK	(1836)	152-302	A13	A1302	152-302/CS	CO4	152-302/CS	CO4	152-302/CS	CO4	152-302/CS	CO4	152-302/CS	CO4	152-302/CS	CO4	152-302/CS	CO4	152-302/CS	CO4
7400V. BUS [#] A INCOMING BRK	(1836)	152-101	A11	A1101	152-101/CS	CO4	152-101/CS	CO4	152-101/CS	CO4	152-101/CS	CO4	152-101/CS	CO4	152-101/CS	CO4	152-101/CS	CO4	152-101/CS	CO4
7400V. BUS [#] B INCOMING BRK	(1236)	152-201	A12	A1201	152-201/CS	CO4	152-201/CS	CO4	152-201/CS	CO4	152-201/CS	CO4	152-201/CS	CO4	152-201/CS	CO4	152-201/CS	CO4	152-201/CS	CO4

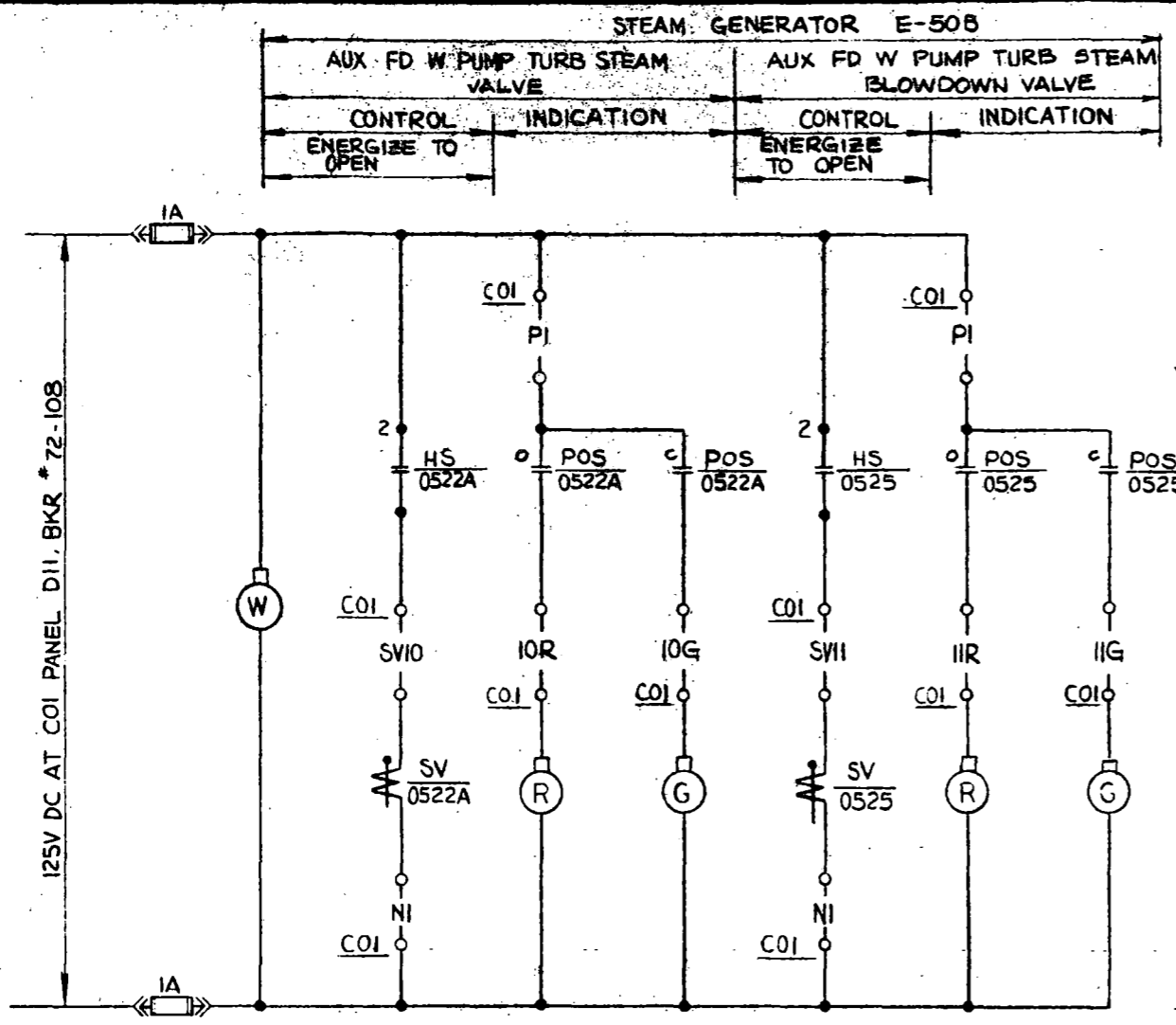
NOTE
 SEPARATE RACEWAY SYSTEM
 FIELD TO
 BETWEEN CONTACTS

DESCRIPTION	LOCATION	RELAY	BLK. NO.	WIRE NO.	BLK. NO.	WIRE NO.
152-105/CS-X	J7400	174-X2/5	162	153/5	J7400	
152-205/CS-X	J7401	174-X2/5	162	153/5	J7401	

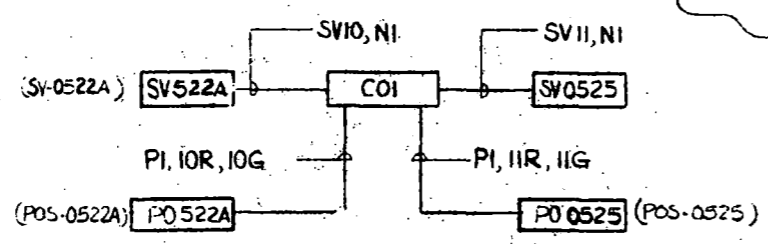
REVISIONS
 12447-007

SCHEMATIC DIAGRAM
 STATION POWER TRANSFORMER
 152-105 INCOMING BREAKERS
 12447-007

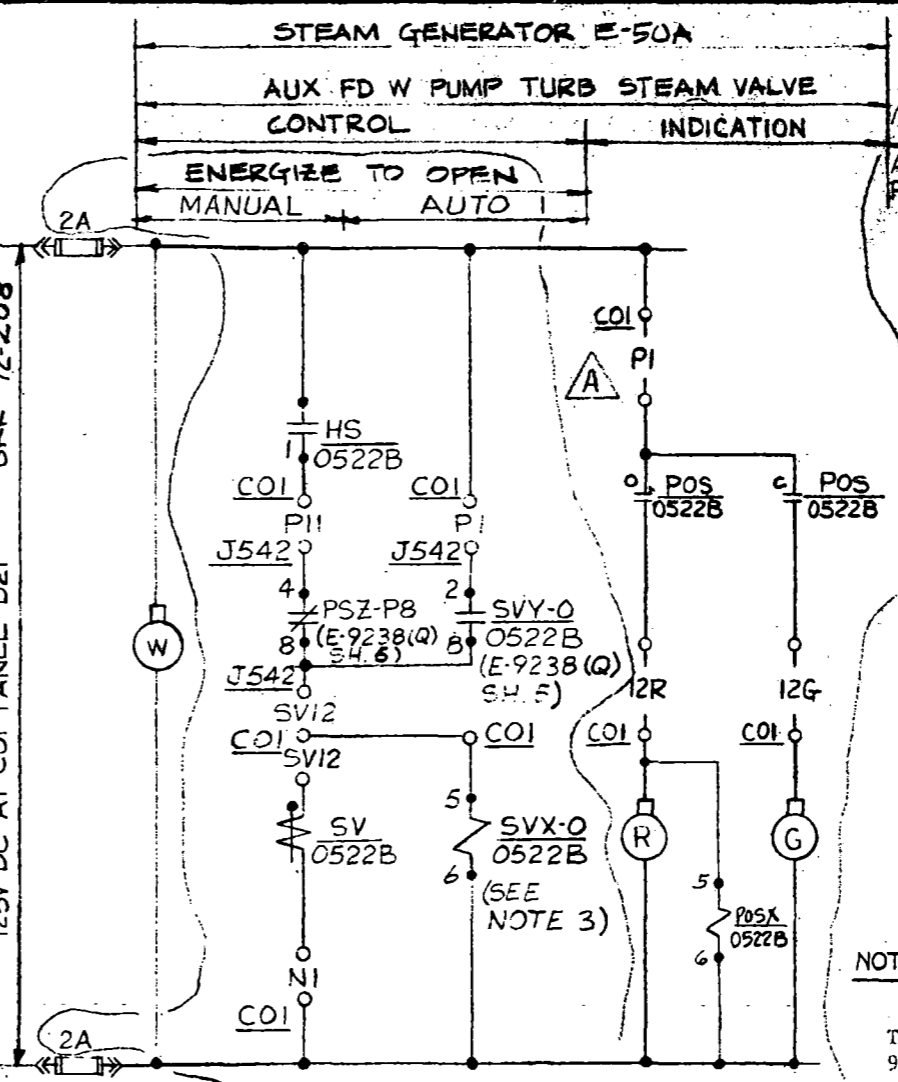
REV.	DESCRIPTION	DATE
A	ISSUED FOR REVIEW	
LX	ENG	
CK	CHK	
SRV	SRV	
APPROVALS	DATE	



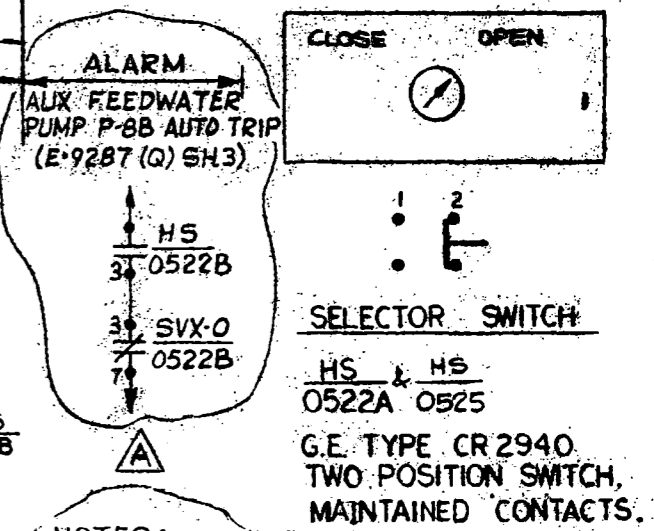
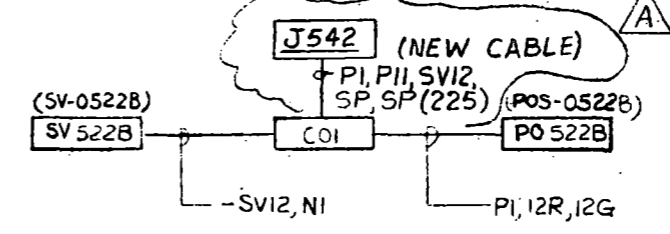
AUX FEED WATER PUMP TURB STEAM VALVE
SCHEME W005 (LEFT CHANNEL)*



* SEPARATE RACEWAY SYSTEM



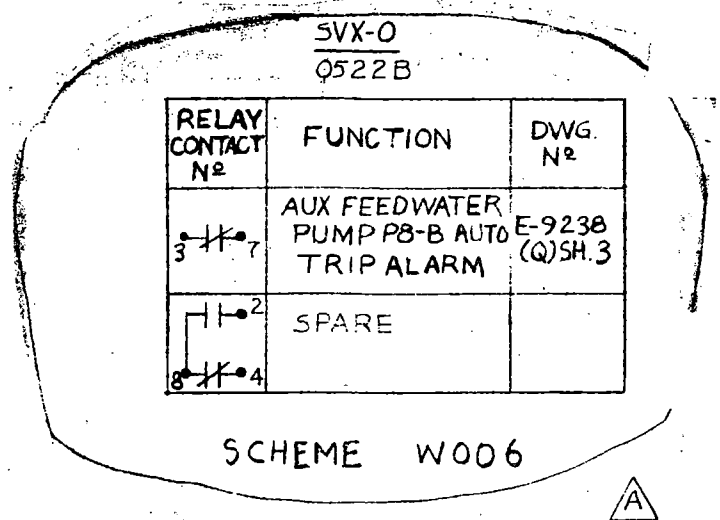
AUX FEED WATER PUMP TURB STEAM VALVE
SCHEME W006 (RIGHT CHANNEL)*



- NOTES:**
- FOR CONTACT DEVELOPMENT OF AUX. RELAYS SVX-0 & POSX SEE RELAY DEVELOPMENT TABLE ON DWG. E-9238(Q) SH. 4.
 - FOR SWITCH HS DEVELOPMENT SEE E-9238(Q) SH. 7.
 - AUX RELAY SVX-0 REPLACES THE EXISTING LATCHING RELAY SVX-0522B ON PNL COI.
- NOTE: GWO 8428 WORK ITEM # 3
SYSTEMS MODIFICATION PER NUREG 0578

This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

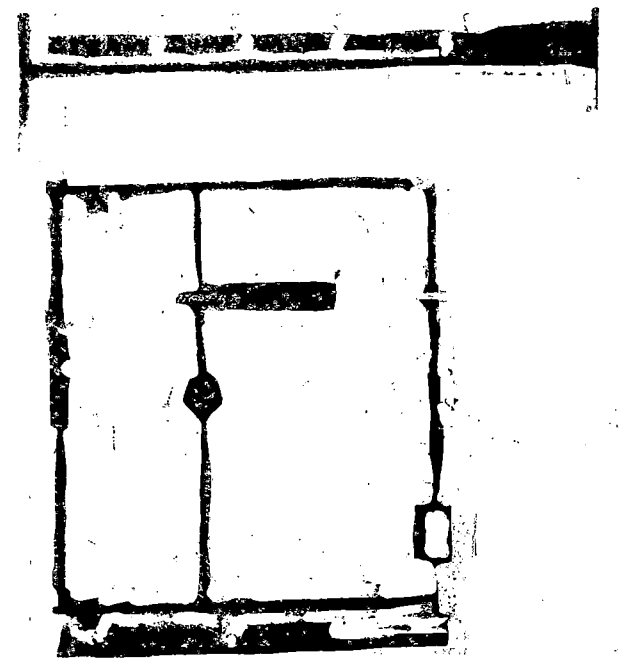
SCALE	DESIGNED KRISHNASAMI	DRAWN WBI/RAN
BECHTEL COMPANY GWO 8428		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MAIN STEAM ISOLATION VALVES		
JOB No.	DRAWING No.	REV.
12447-039	E-9238(Q) SH. 3	A



PSX-0523 PSX-0522B		PSZ-0741A (NOTE-1)	
CONTACT	FUNCTION	NO.	NO.
7-1	SPARE	—	—
8-1	SPARE		SPARE

NOTE 1. EXISTING RELAY PSX-0741A ON PANEL CO1 TO BE RETAGGED AS PSZ-0741A.

RELAY TABLE

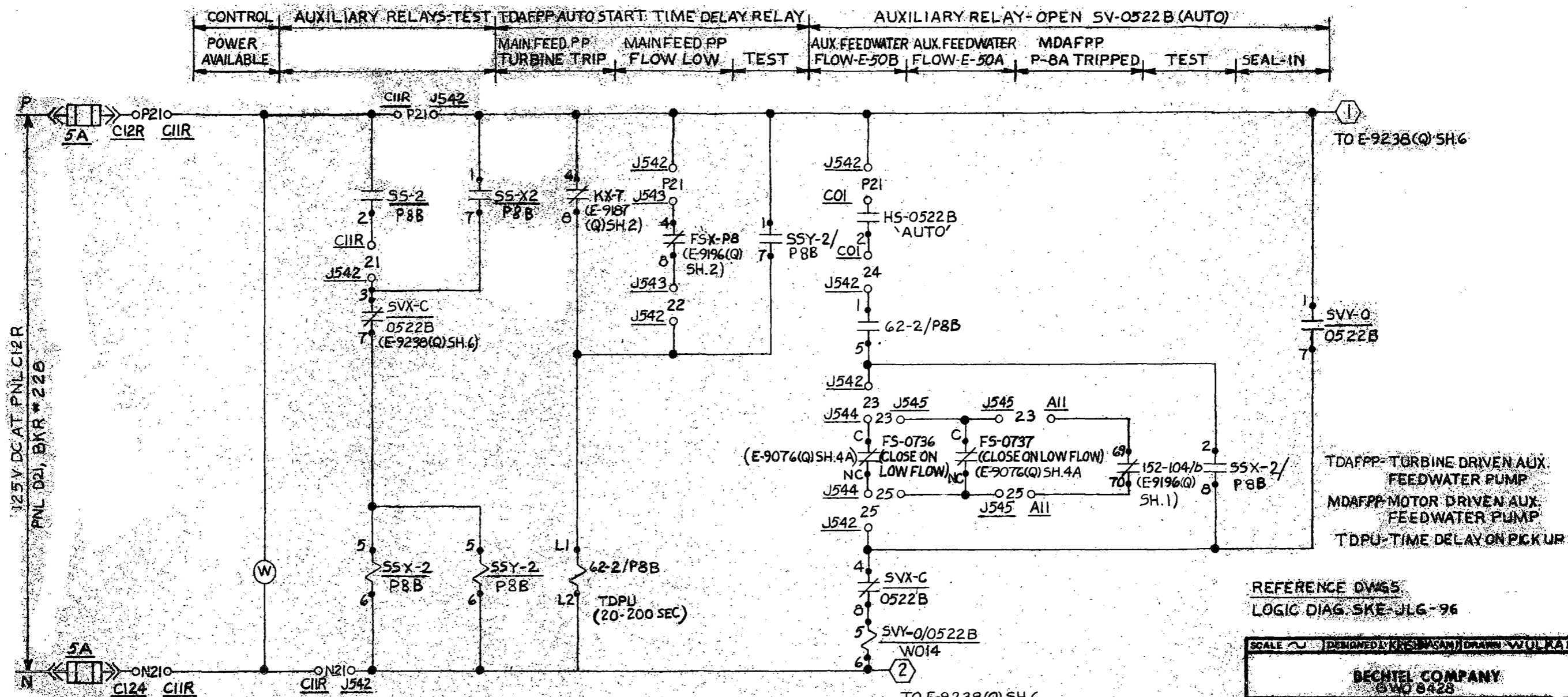


ISSUED FOR REVIEW DATE

KRISHNASAM LX
GWO 8428

Notes: WORK ITEM # 3
GWO: 8428 SYSTEMS MOD PER. NUPSS-0578
This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

42447-039 9238(Q)SH.4 A



AUXILIARY FEEDWATER PUMP TURBINE STEAM VALVES - SV0522B CONTROL-AUXILIARY RELAYS
SCHEME W014

NOTES

1. FOR BLOCK DIAGRAM SEE E-9238(Q)SH.6
2. FOR CONTACT DEVELOPMENT OF AUX. RELAYS
SS-2, SS-X2, 62-2/P8B & SVY-0/O522B
W014, W014, W014, W014
SEE RELAY TABLE ON E-9238(Q)SH.0
3. FOR SWITCH DEVELOPMENT OF
SS-2 & HS-0522B, SEE E-9238(Q)SH.7
P8B

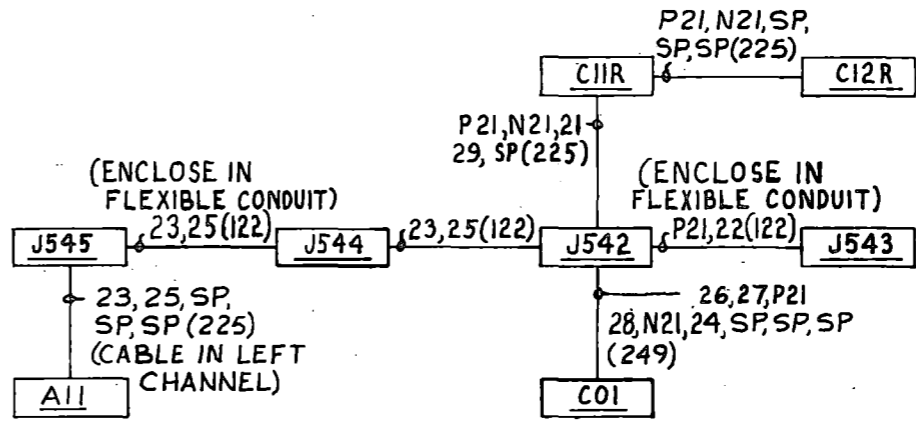
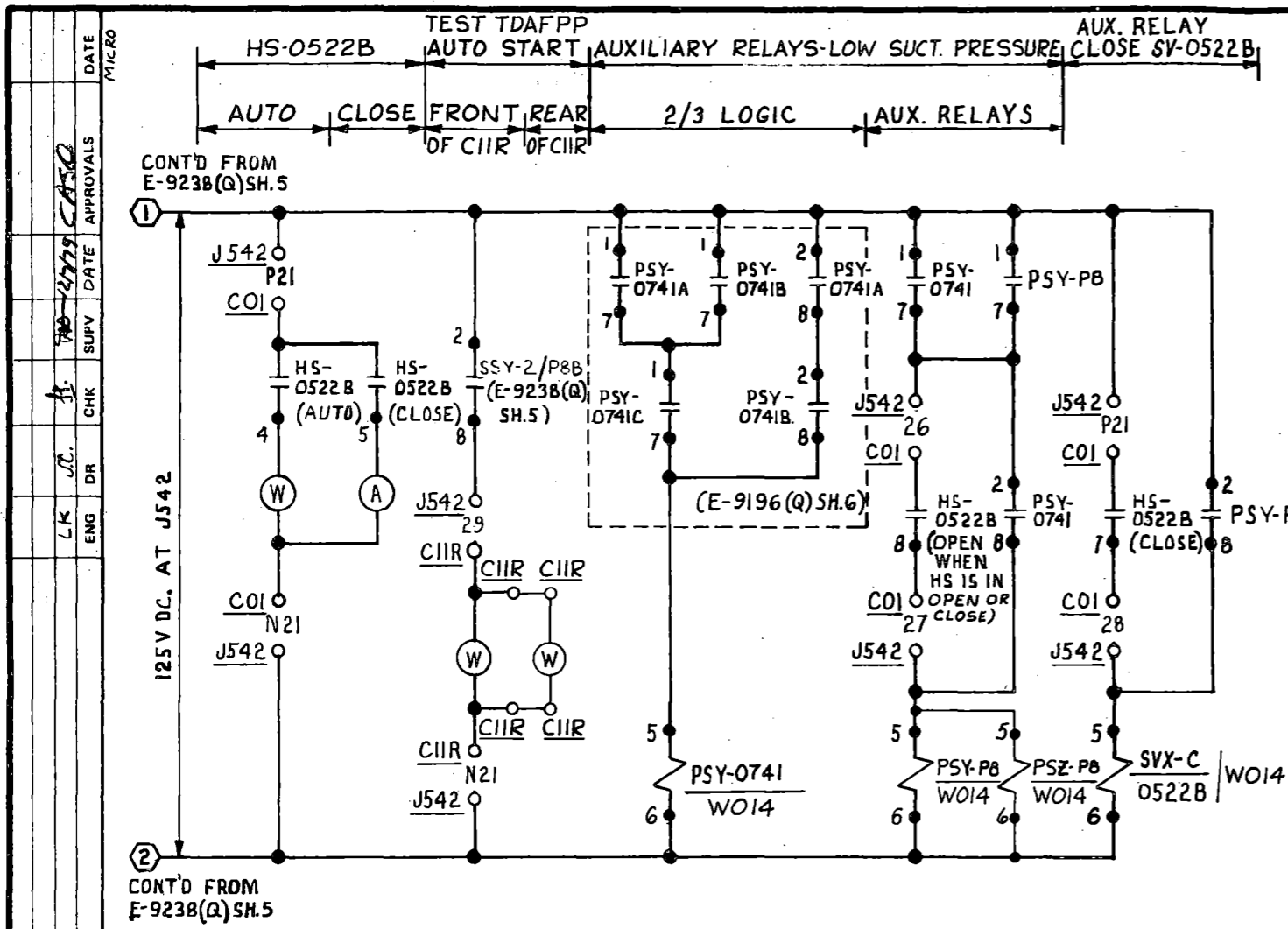
WORK ITEM #3

NOTE: GWO 8428
SYSTEMS MODIFICATION PER NUREG 0578
This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

REFERENCE DWGS
LOGIC DIAG. SKE-JLG-96

SCALE ~	DESIGNED BY KRISHNAN	DRAWN SWULKA N
BECHTEL COMPANY GWO 8428		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MAIN STEAM ISOLATION VALVES		
	JOB No.	DRAWING No.
	12447-039	E-9238(Q)SH.5
		REV.
		A

DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS
DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS
DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS
DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS
DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS	DATE APPROVALS



NOTES:

- FOR CONTACT DEVELOPMENT OF AUX. RELAYS PSY-0741 WO14
- PSY-P8, PSZ-P8 & SVX-C/0522B WO14, WO14, WO14
- SEE RELAY TABLE ON DWG. E-9238(Q)SH.8

REFERENCE DWGS:

- LOGIC DIAGRAM: SK-JLG-96.
- CONNECTION DIAGRAMS: Later

AUXILIARY FEEDWATER PUMP TURBINE STEAM VALVE SV-0522B CONTROL-AUXILIARY RELAYS
SCHEME-WO14

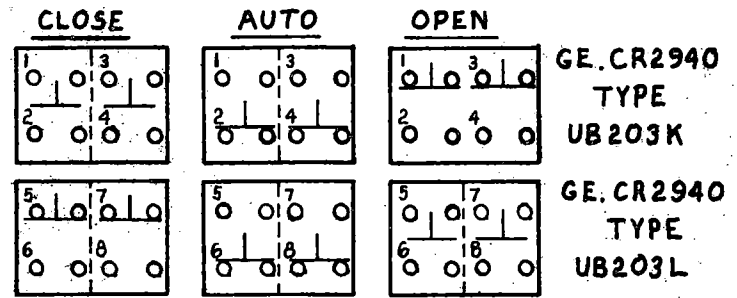
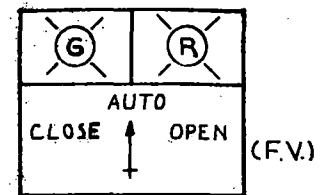
WORK ITEM #3

Note: GWO: SYSTEMS MODIFICATIONS PER NUREG 0578
This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

ISSUED FOR REVIEW	DATE	APPROVALS
REV. DESCRIPTION	DATE	APPROVALS
A		
LK	DR	CHK
ENG	DR	CHK
CHK	DR	CHK
SUPV	DATE	APPROVALS
DATE	APPROVALS	

SCALE	DESIGNED L.K.	DRAWN J.COLPEAN
BECHTEL COMPANY GWO 8428		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MAIN STEAM ISOLATION VALVES		
	JOB No. 12447-039	DRAWING No. E-9238(Q)SH. 6
		REV. A

DATE	MICRO
APPROVALS	CASO
DATE	1/17/77
SUPV	JK
CHK	JK
DR	JK
ENG	JK
REV.	A
ISSUED FOR REVIEW	
DESCRIPTION	

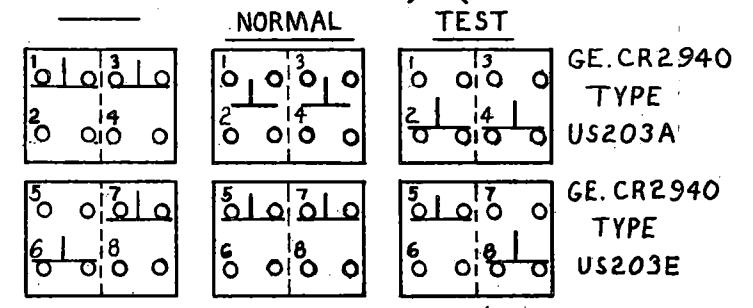
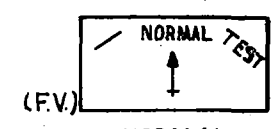


CONTROL SWITCH - HS-0522B
 MAINTAINED CONTACTS TURBINE
 DRIVEN AUX. FEED WATER PP AUTO
 START SELECTOR. LOC-COI
 (SEE NOTE 1)

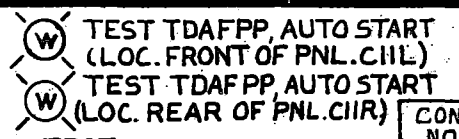
HS-0522B

CONT. NO.	CLOSE	AUTO	OPEN
1			X
2		X	
3			X
4		X	
5	X		
6		X	
7	X		
8		X	

* DENOTES SPARE
 (SEE NOTE 1).



TEST SELECTOR SWITCH SS-2/P8B
 SPRING RETURN TO NORMAL KEY
 REQUIRED TO TURN TO TEST
 LOC. REAR OF PANEL CII-R



SS-2/P8B

CONT. NO.	—	NORMAL	TEST
1	X		
2			X
3	X		
4			X
5		X	X
6	X		
7	X	X	
8			X

* DENOTES SPARE

AUXILIARY FEED WATER PP TURBINE STEAM VALVE-SV0522B CONTROL
 CONTROL SWITCH DEVELOPMENT
 SCHEME W014

NOTES:
 1. THIS CONTROL SWITCH REPLACES THE EXISTING CLOSE-OPEN SWITCH HS-0522B (G E TYPE SBM) ON PANEL-COI.

WORK ITEM #3

Note: GWO: 842B SYSTEMS MODIFICATIONS PER. NUREG 578
 This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

SCALE —			DESIGNED L. KRISHNASAMI DRAWN J. COLPEAN		
BECHTEL COMPANY GWO 842B					
PALISADES PLANT CONSUMERS POWER COMPANY					
SCHEMATIC DIAGRAM MAIN STEAM ISOLATION VALVES					
JOB No.		DRAWING No.		REV.	
12447-039		E-9238(Q) SH.7		A	

RELAY TABLE

62-2/P-8B
WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE AUX. RELAY SVY-0 0522B (OPEN SV-0522B)	E-9238 (Q)SH.5
	SPARE	

SSX-2/P8B
WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	SEAL-IN	E-9238 (Q)SH.5
	ENERGIZE AUX. RELAY SVY-0/0522B	E-9238 (Q)SH.5

SSY-2/P8B
WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE TIME DELAY RELAY 62-2/P8B	E-9238 (Q)SH.5
	INDICATION (TEST TDAFPP AUTO START)	E-9238 (Q)SH.6

SVY-0
0522B /WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	SEAL-IN	E-9238 (Q)SH.5
	ENERGIZE SV-0522B (START TDAFPP, P-8B)	E-9238 (Q)SH.3

PSY-0741 /WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE AUX RELAY PSY-P8	E-9238 (Q)SH.6
	ENERGIZE AUX RELAY PSY-P8	E-9238 (Q)SH.6

PSY-P8 /WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	SEAL-IN	E-9238 (Q)SH.6
	ENERGIZE AUX RELAY SVX-C/0522B (TRIP TDAFPP P-8B)	E-9238 (Q)SH.6

SVX-C /WO14
0522B

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	BLOCK TEST	E-9238 (Q)SH.5
	BLOCK OPEN SV-0522B (ENERGIZE AUX. RELAY SVY-0) 0522B	E-9238 (Q)SH.5

PSZ-P8 /WO14

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	SPARE	
	DEENERGIZE SV-0522B (TRIP TDAFPP, P-8B)	E-9238 (Q)SH.3

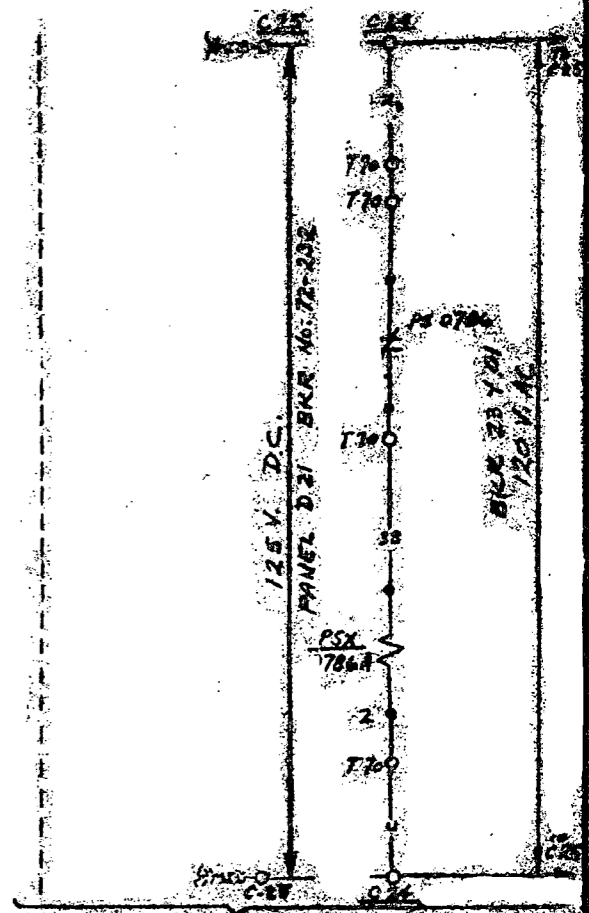
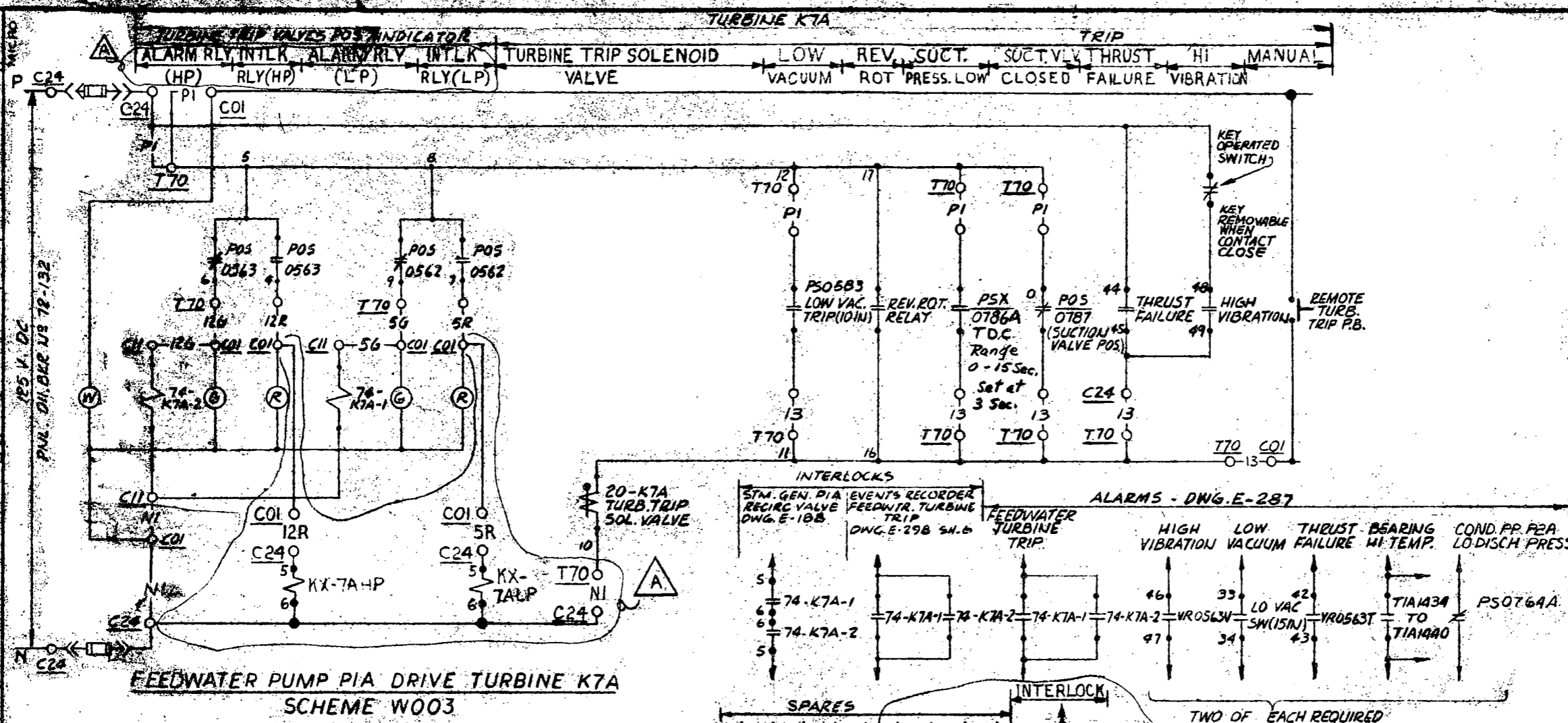
AUXILIARY RELAY TABLE-AUXILIARY FEEDWATER PUMP TURBINE STEAM VALVE - SV0522B
SCHEME WO14

TDAFPP-TURBINE DRIVEN AUXILIARY FEEDWATER PUMP

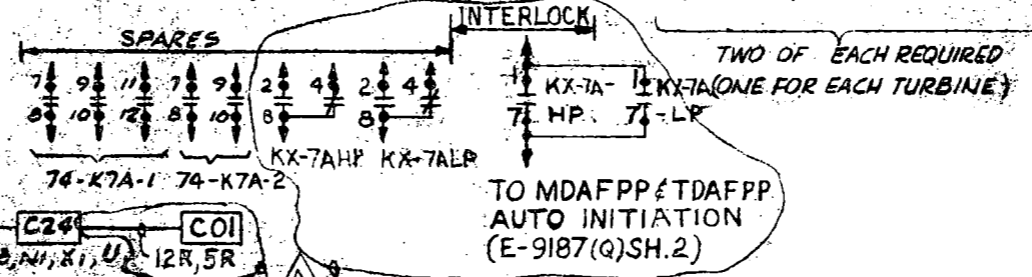
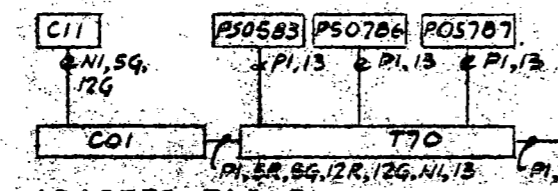
DATE	
APPROVALS	
DESIGN	
CHECK	
DR	
ENG	
REV	
DESCRIPTION	

WORK ITEM N^o 3
NOTE: GWO 8428
SYSTEMS MODIFICATION PER NUREG 0578
This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

SCALE	DESIGNED L. KRISHNASAMI	DRAWN W. L. KAN
BECHTEL COMPANY GWO 8428		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MAIN STEAM ISOLATION VALVES		
	JOB No. 12447-039	DRAWING No. E-9238 (Q)SH.8
		REV. A



- NOTES:**
1. ALL WIRES #14 AWG.
 2. REF. DWG. WORTHINGTON LD-11672 (5985-MIB-22) & LD-11676 (5985-MIB-12)



DESCRIPTION	SCHEME NO.	LOCATION	TURB. L.P. TRIP VALVE POS. SW.	TURB. H.P. TRIP VALVE POS. SW.	AUX. TRIP RELAY	PRESS. & POS. SWITCHES	TURB. TRIP SOLENOID VALVE	ALARM HI TEMP. RELAYS	BEARING HI TEMP. ALARMS	LO VAC. P.S.	MDAFPP & TDAFPP AUTO INITIATION INTLK. RELAYS
FEEDWATER PUMP PIA DRIVE TURBINE K7A	W003	T70 C24	0562	0563	0786A	0787 0786	0764B	74-K7A-1 74-K7A-2	TIA1434 TO TIA1440	0583	KX-7AHP KX-7ALP
FEEDWATER PUMP PIA DRIVE TURBINE K7B	W004	T71 C25	0565	0566	0786B	0787 0786	0765B	74-K7B-1 74-K7B-2	TIA1441 TO TIA1447	0586	KX-7BHP KX-7BLP

MDAFPP - MOTOR DRIVEN AUX. FEEDWATER PUMP
TDAFPP - TURBINE DRIVEN AUX. FEEDWATER PUMP

Note: GWO:8428 SYSTEMS MODIFICATION PER NUREG 0578
This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

SCALE 1/8" = 1"

DESIGNED BY: L. KRISHNASAN
CHECKED BY: J. COLPEAN

BECHTEL COMPANY
GWO 8428

FALSADES PLANT
CONSUMERS POWER COMPANY

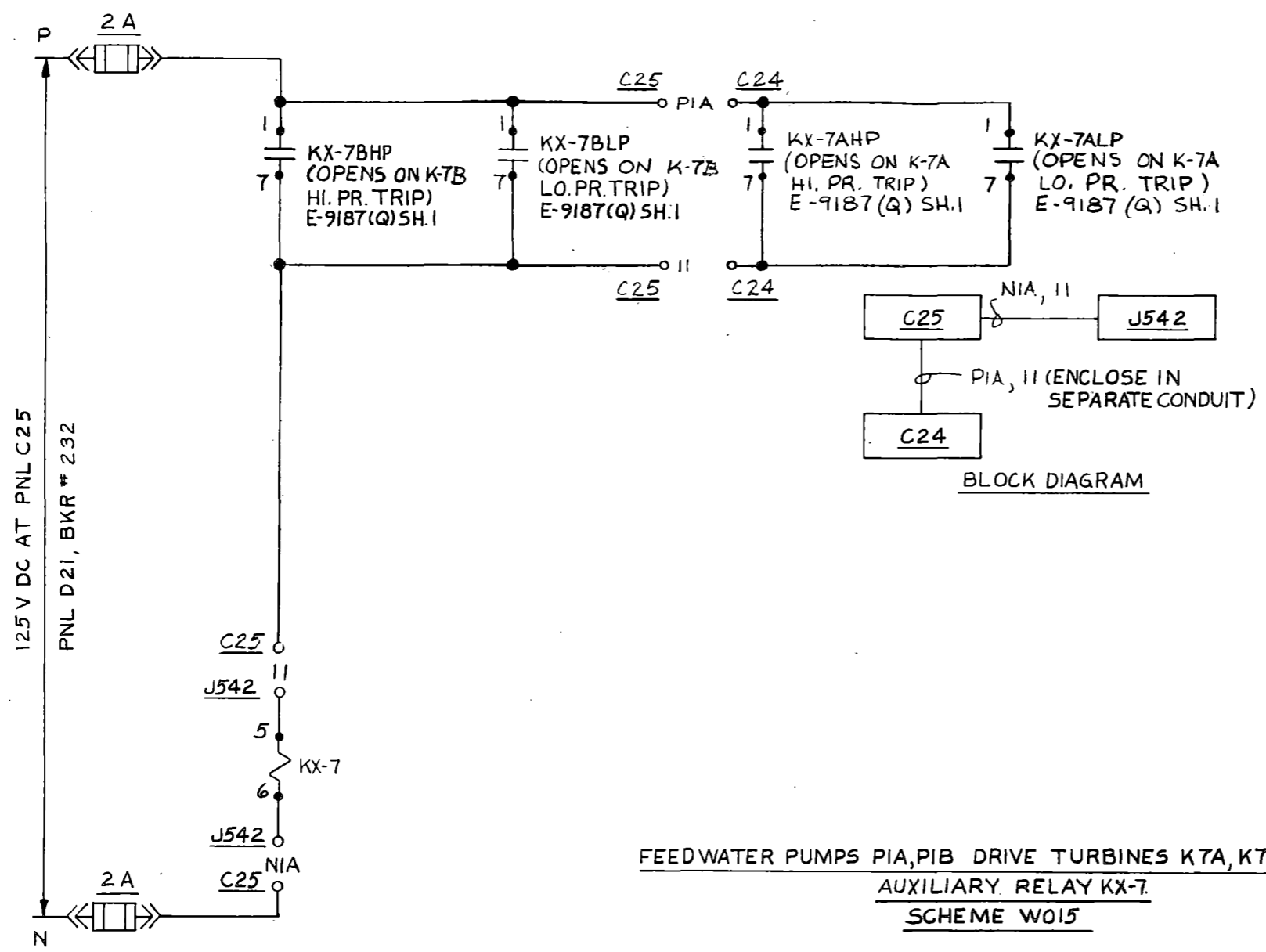
SCHEMATIC DIAGRAM
FEEDWATER TURBINE CONTROL

NO. 12447-039 E-9187(Q) SH. 1 A

ISSUED FOR REVIEW

WORK ITEM #3

REV.	DESCRIPTION	DATE	APPROVALS	MICRO
A	ISSUED FOR REVIEW			
L.K.	ENG	DR	CHK	SUPV
L.V.	DR			



FEEDWATER PUMPS PIA,PIB DRIVE TURBINES K7A, K7B TRIP
 AUXILIARY RELAY KX-7
 SCHEME W015

KX-7/W015

RELAY CONTACT No	FUNCTION	REF. DWG.
3/7	START MDAFPP P-8A	E-9196 (Q)SH.2
4/8	START TDAFPP P-8B	E-9238 (Q)SH.5

RELAY TABLE

TDAFPP-TURBINE DRIVEN AUXILIARY FEEDWATER PUMP
 MDAFPP-MOTOR DRIVEN AUXILIARY FEEDWATER PUMP

WORK ITEM #3

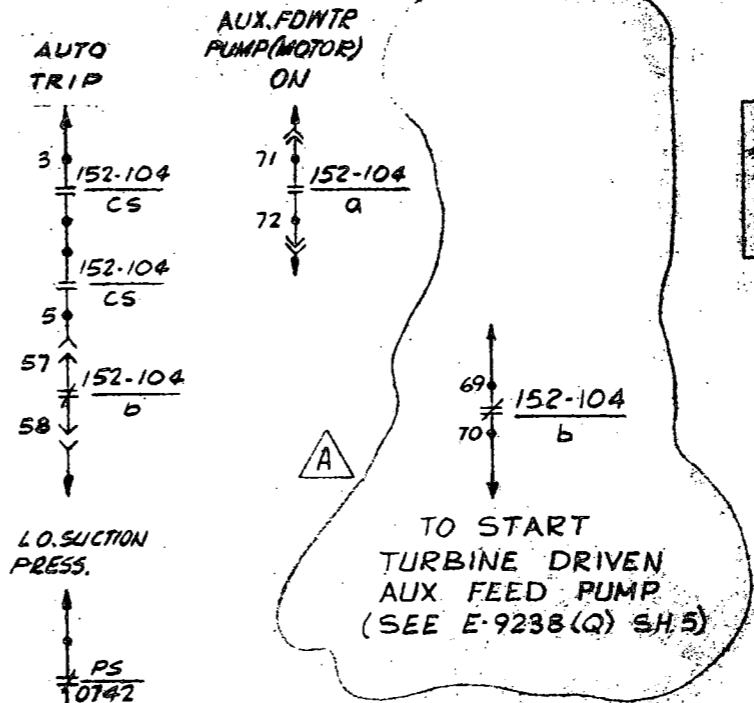
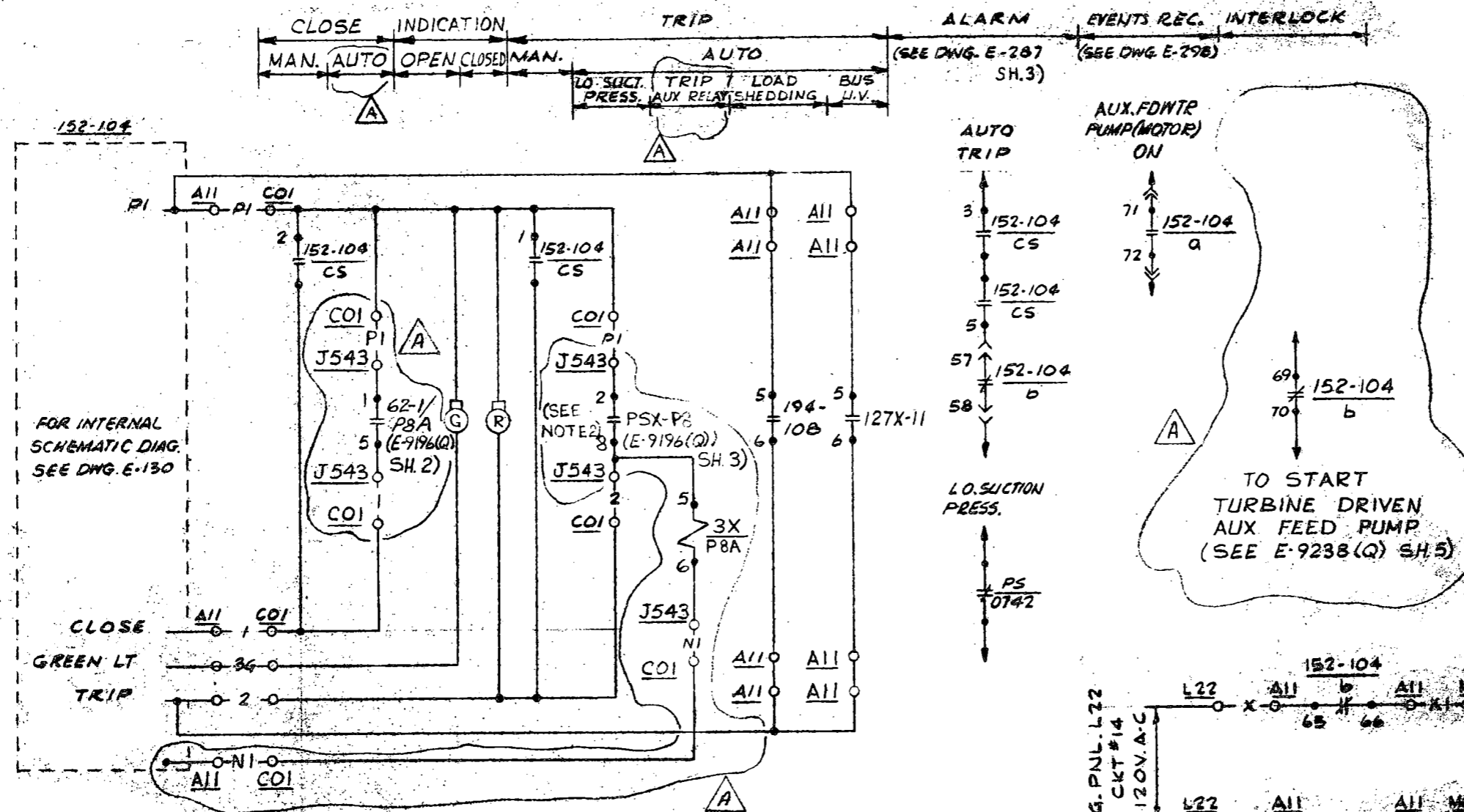
NOTE: GWO 8428

SYSTEMS MODIFICATION PER NUREG 0578

This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

SCALE	DESIGNED L. KRISHNASAMI	DRAWN WULKAN	
BECHTEL COMPANY GWO 8428			
PALISADES PLANT CONSUMERS POWER COMPANY			
SCHEMATIC DIAGRAM FEEDWATER TURBINE CONTROL			
	JOB No.	DRAWING No.	REV.
	12447-039	E-9187(Q) SH.2	A

DATE	APPROVALS
DATE	DATE
CHK	DATE
DR	DATE
ENG	DATE
LK	DATE
ISSUED FOR REVIEW	
DESCRIPTION	

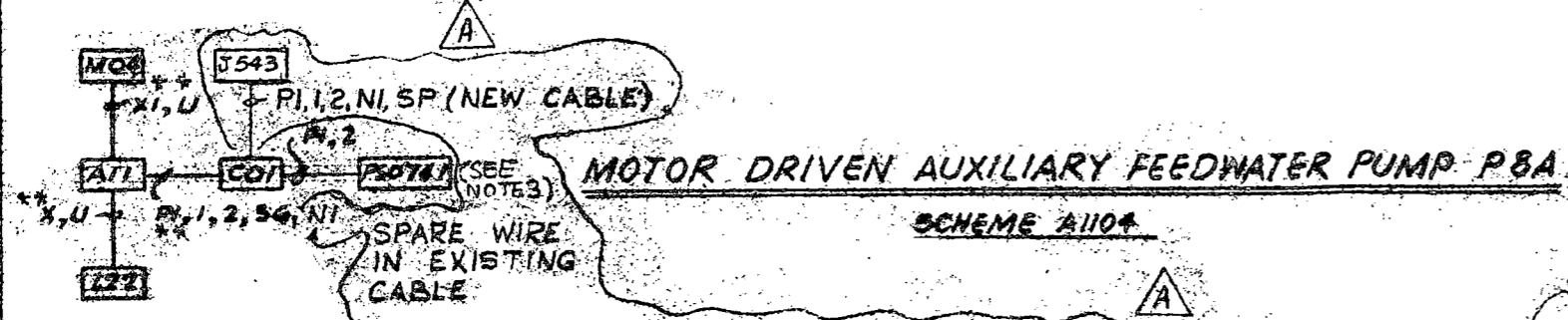
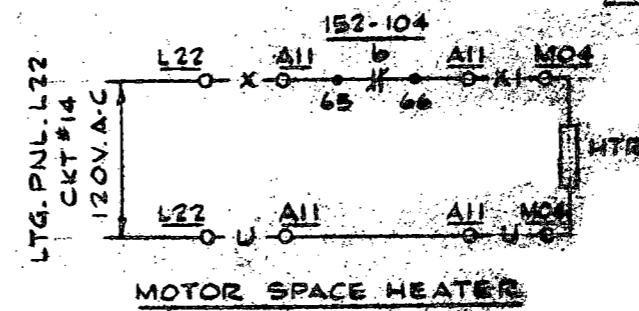


CONTACTS (HANDLE END)	POSITIONS		
	CLOSE	TRIP	TRIP
1-1-1			X
2-1-1	X		
5-1-1		X	X
6-1-1		X	X
5-1-1	X	X	
6-1-1	X	X	

CONTROL SWITCH 152-104 CS
G.E. TYPE SBM, SPRING RETURN LOCATION COI

NOT USED
REFERENCE DWGS:
I. LOGIC DIAGRAM SK-JLG-95

- NOTE**
1. ALL WIRES #18 AWG.
 2. PSX-P8 REPLACES PS-0741 WHICH WAS ORIGINALLY USED TO TRIP ACB 152-104 ON L.O. SUCT. PRESS. FOR PS-0741 SEE E-9196(Q) SH.6
 3. FIELD TO USE THIS EXISTING CABLE A1104/COI-PS0741 IN SCHEME WO13.DWG. E-9196(Q) SH.6



** DENOTES EXISTING CABLE FOR RELAY TABLE, SEE E-9196(Q) SH.5

WORK ITEM # 3
Note: CWO: 8428 SYSTEMS MOD. PER WIRING 9275
This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

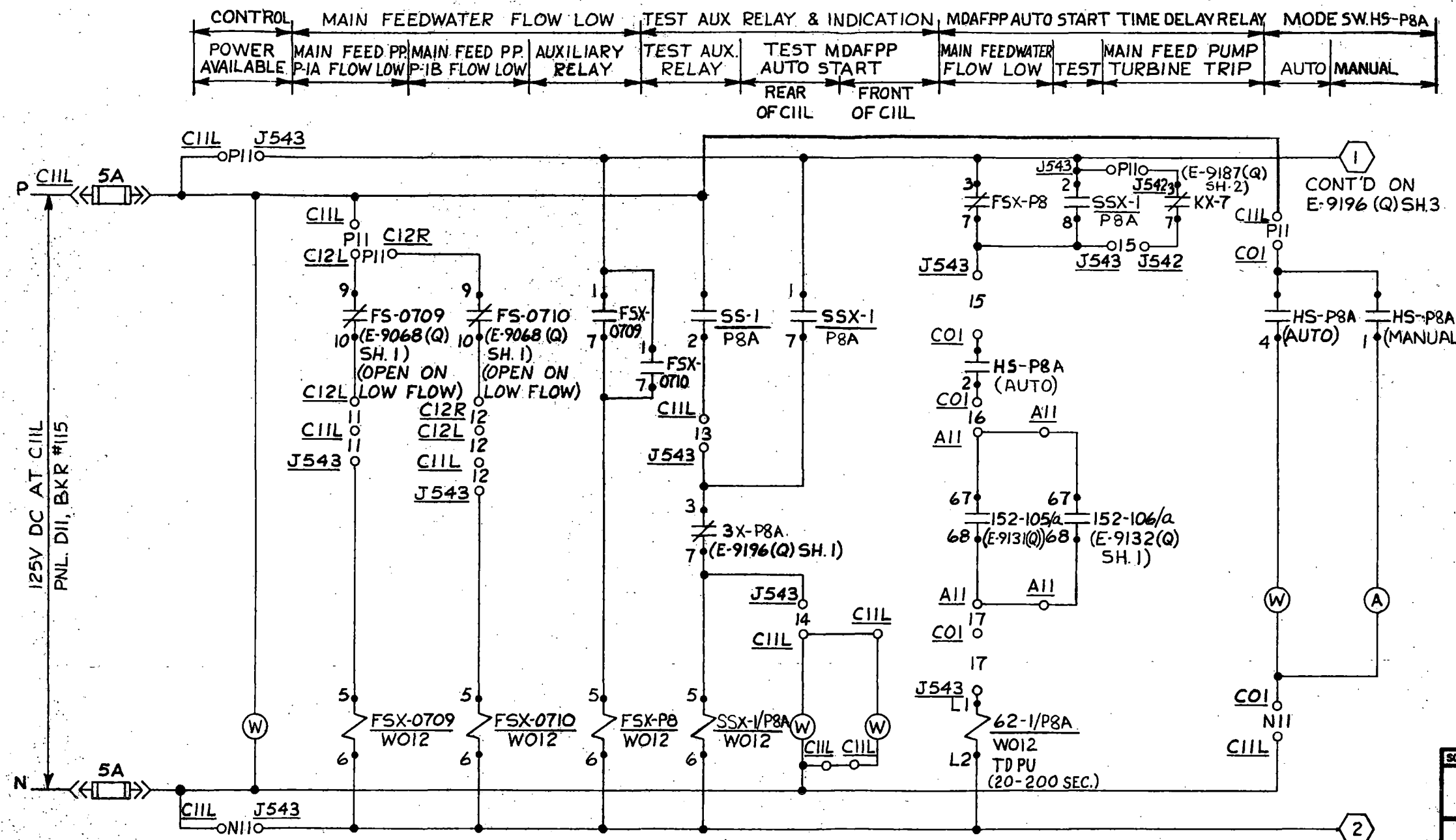
REVISION COMPANY

PARADES PLANT CONSUMERS POWER COMPANY

SCHEMATIC DIAGRAM MOTOR DRIVEN AUXILIARY FEEDWATER PUMP

12447-039 E-9196(Q) SH.1 A

DATE	APPROVALS	DATE	APPROVALS
20-11-77	CASO		
CHK	DATE	CHK	DATE
1			
DR		DR	
176			
ENG		ENG	
ISSUED FOR REVIEW			
REV. DESCRIPTION			
A			



MOTOR DRIVEN AUXILIARY FEEDWATER PUMP P-8A CONTROL-AUXILIARY RELAYS
SCHEME WO12

- NOTES:
- FOR BLOCK DIAGRAM, SEE E-9196(Q) SH.3.
 - FOR CONTACT DEVELOPMENT OF AUX. RELAYS FSX-0709, FSX-0710, FSX-P8, SSX-1/P8A, WO12, WO12, WO12, WO12, 62-1/P8A SEE RELAY TABLE ON E-9196 WO12 (Q) SH.5
 - FOR SWITCH DEVELOPMENT OF HS-P8A, SS-1/P8A, SEE E-9196(Q) SH.4
- MDAFFP-MOTOR DRIVEN AUX. FEEDWATER PUMP.
TDPU-TIME DELAY ON PICKUP.
- REFERENCE DWG:
1. LOGIC DIAGRAM SK-JLG-95

WORK ITEM N^o 3
 Note: GWO:8428 SYSTEMS MOD. PER NUREG 0578
 This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

SCALE 1/2" = 1'-0" / DESIGNED L. KRISHNASAMI / DRAWN M. GUENTHER		
BECHTEL COMPANY GWO 8428		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MOTOR DRIVEN AUXILIARY FEEDWATER PUMP		
JOB No.	DRAWING No.	REV.
12447-039	E-9196(Q) SH.2	A

RELAY TABLE

FSX-0709/W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE AUX. RELAY FOR MAIN FEEDWATER FLOW LOW, FSX-P1	E-9196 (Q)SH.2
	ALARM-MAIN FEEDWATER FLOW LOW	E-9287 (Q)SH.3

FSX-0710/W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE AUX. RELAY FOR MAIN FEEDWATER FLOW LOW, FSX-P1	E-9196 (Q)SH.2
	ALARM-MAIN FEEDWATER FLOW LOW	E-9287 (Q)SH.4

FSX-P1/W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE TIME DELAY RELAY 62-1/P-8A (START MDAFPP P-8A)	E-9196 (Q)SH.2
	ENERGIZE TIME DELAY RELAY 62-2/P-8B (START TDAFPP P-8B)	E-9238 (Q)SH.5

3X-P8A/W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE TEST RELAY SSX-1/P8A	E-9196 (Q)SH.2
	SPARE	

SSX-1/P8A W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	SEAL-IN	E-9196 (Q)SH.2
	ENERGIZE TDR 62-1/P8A (START MDAFPP P8A)	E-9196 (Q)SH.2

PSX-0741/W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	ENERGIZE AUX. RELAY PSX-P8	E-9196 (Q)SH.3
	ENERGIZE AUX. RELAYS PSX-P8	E-9196 (Q)SH.3

PSX-P8/W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	SEAL-IN	E-9196 (Q)SH.3
	TRIP MDAFPP P-8A	E-9196 (Q)SH.1

62-1/P-8A W012

RELAY CONTACT N ^o	FUNCTION	DWG. N ^o
	CLOSE 2400V ACB 152-104 (START MDAFPP P-8A)	E-9196 (Q)SH.1
	SPARE	

MOTOR DRIVEN AUXILIARY FEEDWATER PUMP P-8A CONTROL-AUXILIARY RELAYS

SCHEME W012

TDR-TIME DELAY RELAY
MDAFPP-MOTOR DRIVEN AUXILIARY FEEDWATER PUMP
TDAFPP-TURBINE DRIVEN AUXILIARY FEEDWATER PUMP

WORK ITEM N^o 3

NOTE: GWO 8428

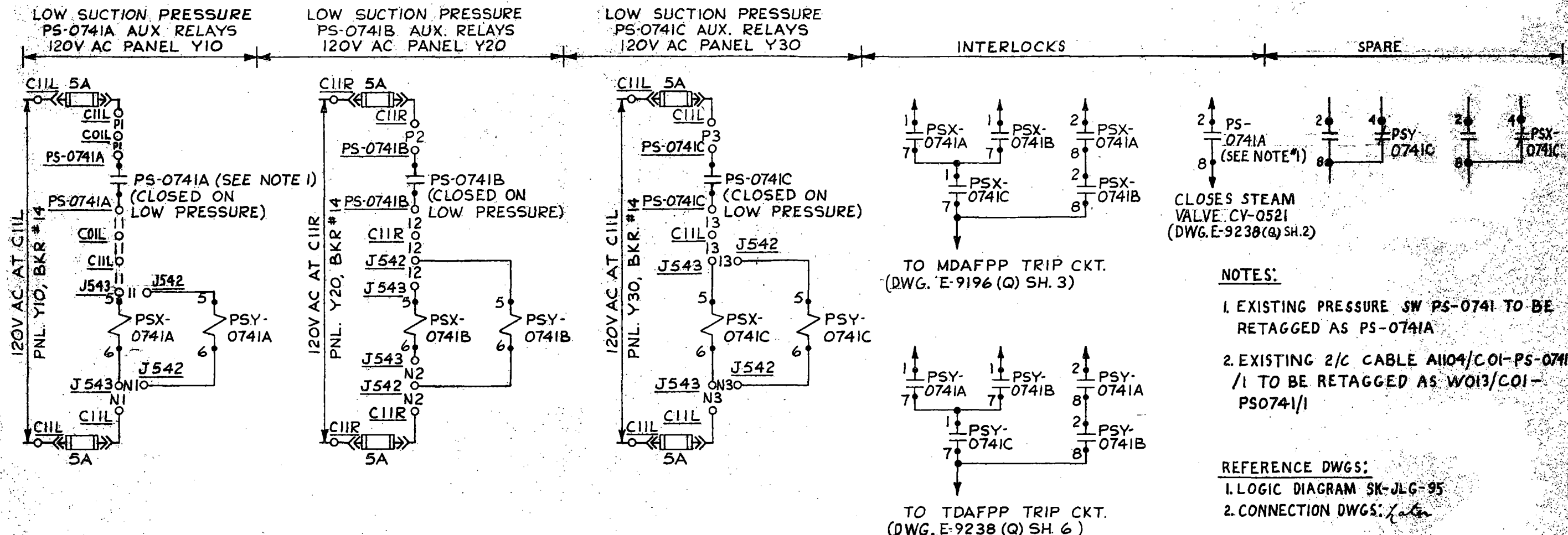
SYSTEMS MODIFICATION PER NUREG 0578

This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

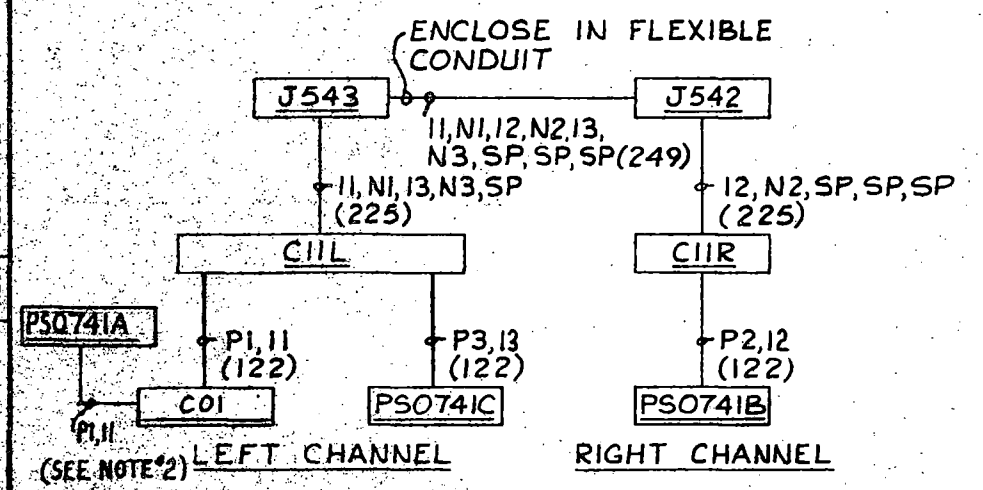
MICROFILM
 DATE APPROVALS
 DATE SUPV
 DATE CHK
 L. W. DR
 ENG
 ISSUED FOR REVIEW
 REV. DESCRIPTION

SCALE	DESIGNED: KRISHNASAMI	DRAWN: WULKAN
BECHTEL COMPANY		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MOTOR DRIVEN AUXILIARY FEEDWATER PUMP		
	JOB No. 12447-039	DRAWING No. E-9196(Q)SH.5
		REV. A

ISSUED FOR REVIEW	DATE	APPROVALS	MICROFILM
	REV	DESCRIPTION	
	1	AS ISSUED	
	2	ISSUED FOR REVIEW	
	3	ISSUED FOR REVIEW	
	4	ISSUED FOR REVIEW	
	5	ISSUED FOR REVIEW	
	6	ISSUED FOR REVIEW	
	7	ISSUED FOR REVIEW	
	8	ISSUED FOR REVIEW	



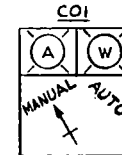
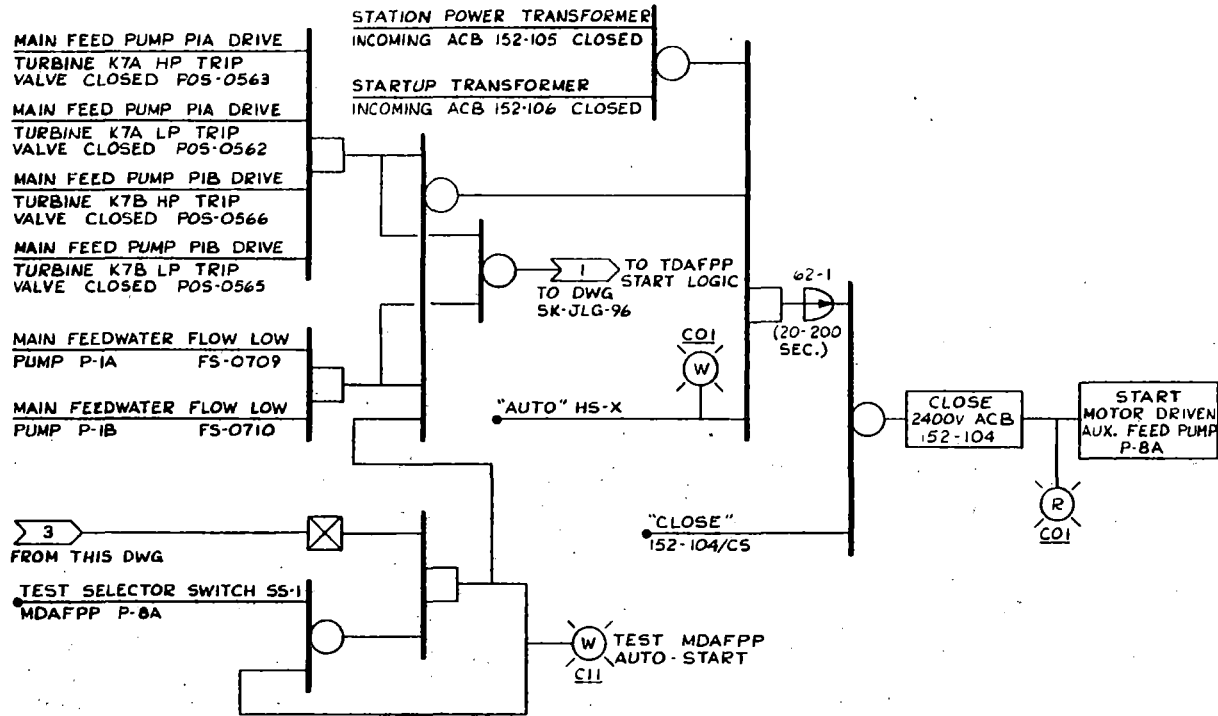
AUXILIARY FEEDWATER PUMPS LOW SUCTION PRESSURE TRIP- 2/3 LOGIC AUX RELAYS
SCHEME W013



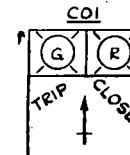
WORK ITEM NR 3
 Note: GWO: 8428 SYSTEMS MODIFICATION PER NUREG 0578
 This drawing is temporarily renumbered in the 9000 series for the duration of the project and will revert to the original number when the "Issued for Record" revision is issued.

SCALE	DESIGNED L. KRISHNASAMI	DRAWN M. GUENTHER
BECHTEL COMPANY GWO 8428		
PALISADES PLANT CONSUMERS POWER COMPANY		
SCHEMATIC DIAGRAM MOTOR DRIVEN AUXILIARY FEEDWATER PUMP		
JOB No.	DRAWING No.	REV.
12447-039	E-9196(Q) SH 6	A

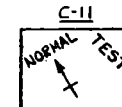
ATTACHMENT 12



MODE SWITCH HS-X,
MAINTAINED CONTACTS
MOTOR DRIVEN AUX.
FEEDWATER PUMP P-8A
AUTO START SELECTOR



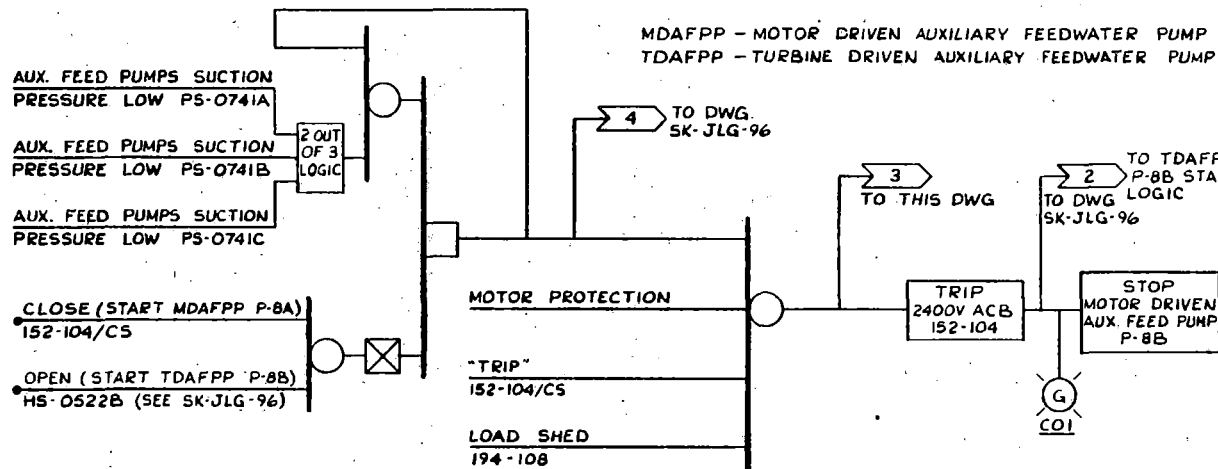
CONTROL SWITCH
SPRING RETURN
152-104/CS



TEST SELECTOR SWITCH SS-1
SPRING RETURN TO NORMAL
KEY REQUIRED
TO TURN TO TEST

"MANUAL" HS-X

MOTOR DRIVEN
AUX FEEDWATER PUMP
P-8A IN MANUAL
POSITION



WORK ITEM NO 3

C	11-14-79	REDRAWN & REISSUED FOR REVIEW	MG	DL	ST	TH	CSO
B	11-5-79	REDRAWN & REISSUED FOR REVIEW	MG	LK	ST	TH	CSO
A	10/17/79	ISSUED FOR REVIEW	MG	ST	TH	CSO	CSO
No. DATE		REVISIONS	BY	CHK	GROUP	PROJ	MICROFILM
SCALE NONE		DESIGNED D. VASSALLO	DRAWN M. GUENTHER				

BECHTEL

ANN ARBOR, MICHIGAN

PALISADES PLANT
CONSUMERS POWER COMPANY
GWO 8428

NUREG 0578 IMPLEMENTATION

MOTOR DRIVEN
AUXILIARY FEEDWATER PUMP P-8A



JOB No.	DRAWING No.	REV.
12447-039	SK-JLG-95	C

LOGIC DIAGRAM