REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8202230353 DOC.DATE: 81/12/18 NOTARIZED: NO DOCKET #
FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards responses to Auxiliary Sys Branch Questions 010.043 & 010.054 & Instrumentation & Control Sys Branch Questions 031.117,031.118 & 031.128. Responses will be incorporated into FSAR amend.

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December 18, 1981
G02-81-533
SS-L-02-CDT-81-108

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSES TO REQUESTS FOR INFORMATION

Enclosed are sixty copies of the responses to Auxiliary Systems Branch Questions 010.043 and 010.054 and Instrumentation and Control Systems Branch Questions 031.117, 031.118 and 031.128.

These questions and responses will be incorporated into an amendment to the WNP-2 FSAR.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC

B001
S 1/1

Q. 010.043
(4.6)

Describe the effects on the safety and operability of the control rod drive hydraulic system if the following control rod drive system valves either fail closed or fail open:

1) Drive water pressure control valve (between F060 and F061);

2) Cooling water pressure control valve (between F070 and F071).

Response:

The function of the F003 pressure control valve (PCV) is to provide a means of adjusting the drive water header and cooling water header pressures. The F003 PCV is a manually controlled motor-operated valve which is controllable from the main control room. Indicating lights are provided in the control room for the valve full open and full closed positions. Adjustment of the F003 PCV in concert with adjustments to the F002 flow control valve permit adjustment of the drive water header pressure to approximately 260 psi above vessel pressure while at the same time, maintaining the drive cooling water header pressure at approximately 20 psi above vessel pressure.

If the F003 PCV were to fail to a full open position, the cooling water pressure would increase and the drive water pressure would decrease. The resulting cooling water pressure increase could cause control rods to drift inward. The existence of rod drifts would be alarmed to the control room operator for appropriate action. The resulting drop in drive water pressure would make normal control rod notch movements impossible but would not affect the ability of the scram function.

Conversely, if the F003 PCV were to fail to a full closed position, the cooling water pressure would decrease while the drive water pressure would increase. The reduction in cooling water pressure (and flow) would eventually lead to high CRD temperatures being alarmed in the control room. The CRD system's scram function would not be affected by the increase in drive water pressure. In the limiting case, the resulting increase in drive water pressure would reach up to the shut off pressure of the supply pump (1750 psig). The occurrence of this condition during withdrawal of a drive at zero reactor

pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations and tests indicate that the drive would accelerate from 3 inches per second to no more than 6.5 inches per second. The rod movement would stop after the driving signal is removed or rod block is enforced by the Reactor Manual Control System (RMCS). In the unlikely event where RMCS fails to enforce a rod block, the peak fuel enthalpy for drive speeds of 6.5 inches per second is well below the fuel cladding failure threshold design limit. Therefore, due to provisions in the system design and margin in the fuel design, this postulated scenario will not compromise the integrity of the fuel.

In both of the cases described above, the manually operated bypass PCV (F004) in conjunction with the isolation gate valves located upstream and downstream of the F003 PCV would enable the operators to take corrective action.

In conclusion, although the failure to the full open or full closed position of the drive/cooling water PCV will cause perturbation in the CRD system operation, it does not present a safety problem or affect the scram capability of the CRD system.

The PCV F005 was deleted from the CRD hydraulic system in the process of implementing the CRD return line deletion modifications, therefore, this question is not applicable.

See response to Question 211.130 for additional details on the deletion of the CRD return line.

Q. 010.054
(9.1.1)

Provide the K_{eff} and the density for optimum moderation for the new fuel storage facility, assuming the infinite array of maximum enriched new fuel for the optimum case. Describe the preventive measures taken to assure that $K_{eff} \leq .98$ for the new fuel storage facility for all moderating conditions. Alternately, demonstrate that no moderating condition between 100% water and 0% water densities can credibly exist.

Response:

Moderating conditions between 100% and 0% water cannot credibly occur. The new fuel has a water tight cover.

The new fuel vault cover consists of three segments; each is a one-foot thick concrete plug equipped with water tight seals at the edges. This cover will prevent direct entry of any water spray or foam into the vault and optimum moderation will not occur.*

*Draft FSAR page change(s) attached.

Insert 1 to Page 9.1-2:

The new fuel storage vault is fitted with leak tight gasketed concrete covers, positively preventing any significant water inleakage. The movement of these covers is administratively controlled by approved plant procedures.

Insert 2 to Page 9.1-2:

The vault is provided with concrete, gasketed covers which are capable of positively preventing significant water inleakage into the vault.

Insert to Page 9.1-4:

New fuel storage vault covers prevent optimum moderation in the new fuel vault (water density between 0 and 1 gram per cc).

calculations performed to assure that $k_{eff} \leq 0.95$, the standard lattice methods (Reference 9.1-1) used at General Electric are employed. Under conditions where diffusion theory is valid, the reference 9.1-1 method is used in calculations (i.e., conditions where fuel is flooded with water at a density of between 0.7 and 1.0 g/cc).

Insert 1

It is assumed that the storage array is infinite in all directions. Since no credit is taken for leakage, the values reported as effective neutron multiplication factors are in reality infinite neutron multiplication factors.

The biases between the calculated results and experimental results, as well as the uncertainty involved in the calculations, are taken into account as part of the calculational procedure to assure that the specified k_{eff} limits are met.

9.1.1.1.2 Power Generation Design Bases

a) New fuel storage racks provide for approximately 30% of the full core fuel load.

b) New fuel storage racks are designed and arranged so that the fuel assemblies can be handled efficiently during refueling operations.

9.1.1.2 Facilities Description

The new fuel storage vault containing the new fuel storage racks is a concrete structure adjacent to the spent fuel pool at the refueling floor level of the reactor building (see Figure 1.2-6). The reactor building is built to Seismic Category I requirements and is further discussed in 3.2. Insert 2
The new fuel rack design features are as follows:

a) The new fuel storage vault contains 24 sets of castings which may contain up to 10 fuel assemblies; therefore a maximum of 240 fuel assemblies may be stored in the fuel vault.

b) There are three tiers of castings which are positioned by fixed box beams. This holds the fuel assemblies in a vertical position, the fuel assemblies are supported at the lower and upper tie plate, with additional lateral support at the center of gravity of the fuel assembly.

The floor of the new fuel storage vault is sloped towards a drain located at the low point. This removes any water that may be accidentally and unknowingly introduced into the vault. the drain is part of the floor drain subsystem of the liquid radwaste system.

The radiation monitoring equipment for the new fuel storage area is described in 12.3.4.

9.1.1.3 Safety Evaluation

9.1.1.3.1 Criticality Control

The calculations of k_{eff} are based upon the geometrical arrangements of the fuel array, and that subcriticality does not depend upon the presence of neutron absorbing materials. To meet the requirements of General Design Criterion 62, geometrically-safe configurations of fuel stored in the new fuel array are employed to assure that k_{eff} will not exceed 0.95 if fuel is stored in the dry condition or if the abnormal condition of flooding (with water with a density of 1 g/cc) occurs. In the dry condition, k_{eff} is maintained ≤ 0.95 due to under-moderation (water density of 0 grams per cc). In the flooded condition, the geometry of the fuel storage array assurs the k_{eff} will remain ≤ 0.95 due to over-moderation.

The fuel storage rack is designed using non-combustible materials. Plant procedures and inspections assure that combustible materials are restricted from this area. The primary approach to fire prevention is the elimination of combustible materials. ~~The fire suppressant is water which will not inhibit or negate criticality control. Refer to 9.5.1 for discussion of WNP-2 fire protection systems.~~

9.1.1.3.2 New Fuel Rack Structural Design

The new fuel storage racks are designed to meet Seismic Category I requirements.

The maximum stress in the fully load rack in a faulted condition is 16.5 Kip. (See Table 3.9-2(s)). This is significantly lower than the allowable stress.

The storage rack is designed to withstand horizontal combined loads up to 222,000 lbs., well in excess of expected loads.

Q. 031.117
(7.3)

General Electric and other nuclear steam supply system (NSSS) vendors have reported that post-accident temperature conditions can affect reactor vessel water level instrumentation.

a. Describe the liquid level measuring systems within the WNP-2 containment which are used to initiate safety-related actions or are used to provide post-accident monitoring information. Provide a description of the type of reference leg used; i.e., an open column or a sealed reference leg.

b. Provide an evaluation of the effect of post-accident ambient temperatures on the indicated level which will relate the change in indicated level to the actual water level. This evaluation must include all other sources of error, including the effects on the water level measurements caused by varying fluid pressure and flashing of water in the reference leg to scram.

c. Provide an analysis of the effect that the potential level measurement errors in both the WNP-2 control and protection systems, discussed in Item (b) above, can have with respect to the validity of the assumptions used in your analyses of plant transients and postulated accidents. Your response should include a review of all safety and control setpoints derived from level signals to verify that the setpoints will initiate the safety-related action required by the plant safety analyses. This review should encompass the anticipated range of ambient temperatures that may be encountered by the safety-related instrumentation, including those temperatures which could occur during and after postulated accidents. If this analysis demonstrates that the level measurement errors which could occur in the WNP-2 facility are greater than those assumed in your safety analysis, indicate the corrective action to be taken. The corrective actions which you should consider include design changes that could be made to ensure that the effects of varying containment temperatures are automatically accounted for. These measures may include setpoint changes as an acceptable corrective action for the short term. However, some form of temperature compensation or modification to eliminate or reduce temperature errors should be investigated as a long-term solution.

d. Indicate any required revisions to your emergency procedures to include specific information obtained from the review and evaluation of Items (a), (b), and (c) above to ensure that the reactor operators are instructed on the potential for, and magnitude of, erroneous level signals. Provide a copy of tables, curves, or correction factors that would be applied to post-accident monitoring systems which will be used by plant operators.

Response:

a. Reactor vessel water level is measured by means of a produced differential pressure between a reference leg and a variable leg. The reference leg is connected to the upper part of the vessel (steam zone) and provides the constant head using an overflow type condensing chamber. The variable leg is connected to the lower part of the vessel. The produced differential pressure is therefore a function of water level.

b.
c.
d. General Electric has conducted a WNP-2 review on the effects of high drywell temperature on the ~~Hanford~~ reactor vessel water level instrumentation. Instrument accuracy is not markedly affected by varying drywell temperatures because the vertical drop of the sensing lines within the drywell are approximately equal. This ensures cancellation of temperature effects between lines, should elevated drywell temperature conditions (as in a LOCA) occur, and thereby ensures continued instrument setpoint accuracy under these conditions. Therefore, there would be little or no impact on the scram or other level trip function, nor would post-accident monitoring be affected except for the possible occurrence of reference leg boil-off. The error due to reference leg boil-off will occur only if the operator fails to follow the Emergency Procedure Guideline (EPG) requirement to refill the reference leg after reactor depressurization. If the operator fails to refill the reference leg but does follow the EPG to maintain reactor water level in the normal range post LOCA, there would be a minimum of 11.7 feet of water over the top of the active fuel at all times (assuming a loss of all water in the portion of the reference leg located in the drywell). The level measuring error due to reference leg boil-off is a slowly developing one. Ten hours or more following reactor depressurization would be required to boil the reference leg in the drywell dry. Assuming the reference leg is boiled dry in ten hours, a low reactor water level 2 signal would be received when the actual reactor water level is 6.4 feet above the top of the active fuel. Therefore, the operator would receive a low reactor water level 2 signal approximately 79 minutes prior to initial reactor core uncovery which provides adequate time for the operator to re-initiate core cooling systems.

In summary, it is concluded that the reactor WNP-2 water level measurement instrumentation provided on ~~Hanford~~ will provide all safety related trips at the proper setpoint and adequate information to the operator for the performance of required manual safety actions under all potential high drywell temperature conditions.

Q. 031.118
(7.3.1)
(F7.3-10a)

Your discussion of the reactor building ventilation radiation monitors in Section 7.3.1.1.2 of the FSAR is incomplete. Accordingly, provide additional information to show how the channel trips are connected to initiate isolation. Correct the discrepancy between Figure 7.3-10a and GE Drawing No. 807E168TC, sheets 8 and 9, and Burns and Roe Drawing No. E-519, sheet 33, which show that only the contacts of relay K2 (i.e., the upscale trip) are used to actuate the isolation valves. This discrepancy implies that the process radiation monitor inoperative trip is utilized in the isolation logic.

Response:

Section 7.3.1.1.2 has been revised to add the following paragraph:*

> "There are four radiation monitors arranged in two sets of two channels each, (A,B) and (C,D), which make up the PRM Reactor Building Vent Radiation Monitor Isolation System. Each channel has a single trip signal output used for shutdown and isolation of the Reactor Building Ventilation System, startup of the Standby Gas Treatment System, and trip and/or isolation of various other plant functions. The trip output (relay K2) consists of a high-high radiation signal (upscale) or an instrument in "calibration" signal (inoperative). In addition, an instrument downscale alarm is provided to alert the operator to instrument trouble conditions".

Burns and Roe Drawing No. E-519, sheet 33, has been modified to describe the GE relay (K2) input as "high-high radiation or inoperative" rather than "high radiation" in order to be more clear.

*FSAR draft page change attached.

7. Reactor Building Ventilation Radiation Monitor - Instrumentation and Controls

~~The reactor building ventilation monitoring consists of four sensor and trip units. Each channel has two trips. The upscale trip indicates high radiation and the downscale trip indicates instrument trouble.~~

~~The reactor building ventilation radiation monitor senses reactor building exhaust to the elevated release point. In the event that radiation levels exceed predetermined limits the intake and exhaust dampers are closed.~~

replace with attached

8. Reactor Water Cleanup (RWCU) System-High Differential Flow

High differential flow in the reactor water cleanup system could indicate a breach of the reactor coolant pressure boundary of the cleanup system. The flow at the inlet to the system (suction from Recirc. lines) is compared with the flow at the outlets of the system (flow return to feedwater or flow to the main condenser and/or radwaste).

Two redundant differential flow sensors compare the reactor water cleanup system inlet-outlet flow. Each of the flow monitoring sensors provides an input to one of the two (inboard or outboard) logic trip channels.

When an increase in reactor water cleanup system differential flow is detected, the PCRVICS initiates closure of all reactor water cleanup system isolation valves.

Diversity of trip initiation signals for reactor water cleanup system line break is provided by instrumentation for reactor water level, differential flow, and ambient or differential temperature in RWCU equipment areas.

The reactor water cleanup system high differential flow trip is bypassed by an automatic timing circuit during normal reactor water cleanup system surges. This time delay bypass prevents inadvertent system isolations during system operational changes.

9. Reactor Water Cleanup (RWCU) System-Area High Ambient Temperature and Differential Temperature

High temperature in the equipment room areas of the reactor water cleanup system could indicate a breach in the reactor coolant pressure boundary in the cleanup system.

Insert to Page 7.3-15:

There are four radiation monitors arranged in two sets of two channels each, (A,B) and (C,D), which make up the PRM Reactor Building Vent Radiation Monitor Isolation System. Each channel has a single trip signal output used for shutdown and isolation of the Reactor Building Ventilation System, startup of the Standby Gas Treatment System, and trip and/or isolation of various other plant functions. The trip output (relay K2) consists of a high-high radiation signal (upscale) or an instrument in "calibration" signal (inoperative). In addition, an instrument downscale alarm is provided to alert the operator to instrument trouple conditions.

Q. 031.128
(7.3.1)
(F7.7-7)

Figure 7.3-7 of the FSAR indicates that there are two condensate storage tanks, each with a manually operated discharge valve. The functional control diagram (i.e., Figure 7.3-8) illustrates, and the text discusses, the interlock between the condensate storage tank and the suppression pool suction valves which is intended to provide assurance that the high pressure core spray (HPCS) system pump has an acceptable supply of water at the suction inlet. However, if both manual discharge valves were to be closed, the purpose of this interlock would be defeated. Accordingly, provide justification in section 7.3.1.1.1.1 for the omission of manual discharge valve position switches as initiators in the HPCS pump suction control logic.

Response:

FSAR Figure 7.3-7 has been changed to Figure 6.3-1 and Figure 7.3-8 has been changed to Figures 7.3-8a, 7.3-8b, and 7.3-8c. These figures have been revised to delete the ~~namual~~ manual discharge valve position interlocks. These interlocks were originally provided (when the condensate storage tanks level switches were physically mounted on the tanks) to prevent premature transfer of the HPCS suction to the suppression pool when one of the tanks was out of service (drained) giving a false transfer signal.

Figure 6.3-1 has been revised to indicate the level switches on a Class 1 standpipe in the Reactor Building on the condensate supply line to HPCS which is downstream of the manual suction valves. (These valves are locked open during normal operation.) With this modification, valve position interlocks are no longer required. FSAR Section 7.3.1.1.1.1 has been revised to correctly describe this modification.

FSAR Revision

Revise for BRSCN-81-435

b. Automatic depressurization system (ADS);

c. Low pressure core spray system (LPCS);

d. Low pressure coolant injection (LPCI) mode of the residual heat removal system (RHRS).

The following plant variables are monitored and provide automatic initiation of the ECCS when these variables exceed predetermined limits:

1. Reactor Vessel Water Level

A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the reactor coolant pressure boundary and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes. Refer to Figure 7.3-9 (Nuclear Boiler P&ID) for a schematic arrangement of reactor vessel instrumentation.

2. Drywell Pressure

High pressure in the drywell could indicate a breach of the reactor coolant pressure boundary inside the drywell and that the core is in danger of becoming overheated as reactor coolant inventory diminishes.

7.3.1.1.1.1 High Pressure Core Spray (HPCS) System - Instrumentation and Controls

a. HPCS Function

The purpose of the HPCS is to provide high pressure reactor vessel core spray for small line breaks which do not depressurize the reactor vessel. In addition, HPCS is redundant to the RCIC system for mitigation of the consequences of various events listed in Appendix 15A. Refer also to 6.3.2.2.1.

b. HPCS Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 6.3-1 ~~7.3-7~~ (HPCS P&ID). HPCS system component control logic is shown in Figure 7.3-8 (HPCS FCD) and Figure 7.3-4 (HPCS Power Supply FCD). Instrument specifications are listed in Tables 7.3-1 and 7.3-2. Plant Layout drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure ~~7.3-7~~ 6.3-1 (HPCS P&ID) and Figures 7.3-8 and 7.3-4 (HPCS and HPCS Power Supply ~~FCD~~ FCDs).

Revise for SCN 81-435 Q 31.128

closes. Two level switches mounted on a Class 1 standpipe in the Reactor Building are used to detect low water level in ~~each of~~ the condensate storage tanks. Either switch can cause automatic suction transfer. The suppression pool suction valve also automatically opens if high water level is detected in the suppression pool. Two level switches monitor suppression pool water level and either switch can initiate opening of the suppression pool suction valve. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other closes.

The HPCS provides makeup water to the reactor until the vessel water level reaches the high level trip (Trip Level 8) at which time the injection valve MOF004 is automatically closed. The pump will continue to run on minimum flow recirculation. The injection valve will automatically reopen if vessel level again drops to the low level (Trip Level 2) initiation point.

The HPCS pump motor and injection valve are provided with manual override controls. These controls permit the reactor operator to manually control the system following automatic initiation.

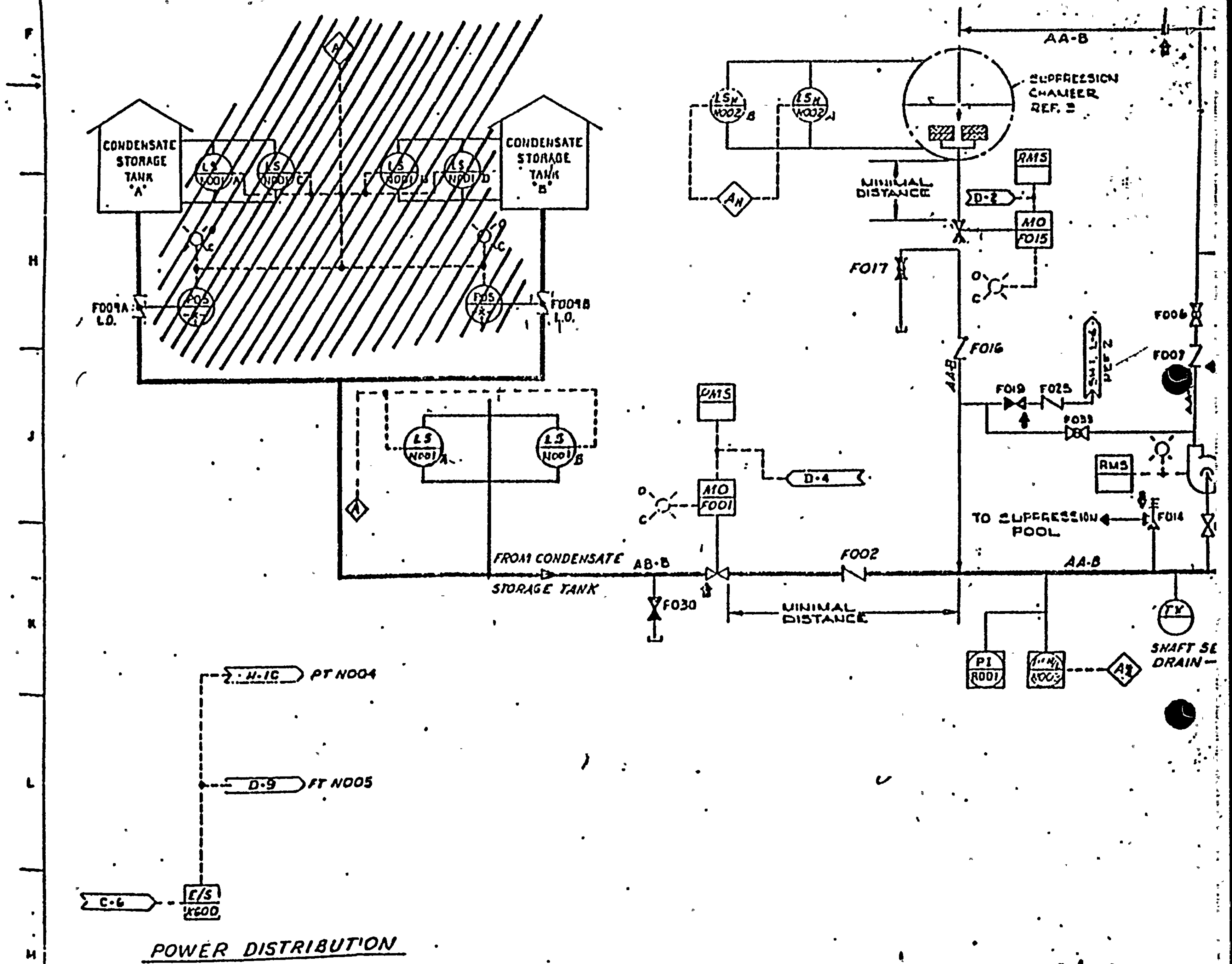
7.3.1.1.1.2 Automatic Depressurization System (ADS)-Instrumentation and Controls

a. ADS System Function

The automatic depressurization system is designed to provide automatic depressurization of the reactor vessel by activating seven safety/relief valves. These valves vent steam to the suppression pool in the event that the HPCS cannot maintain the reactor water level following a LOCA. ADS reduces the reactor pressure so that flow from the low pressure ECCS, LPCI system and LPCS, can inject into the reactor vessel in time to cool the core and limit fuel cladding temperature. Refer also to 6.3.2.2.2.

b. ADS Operation

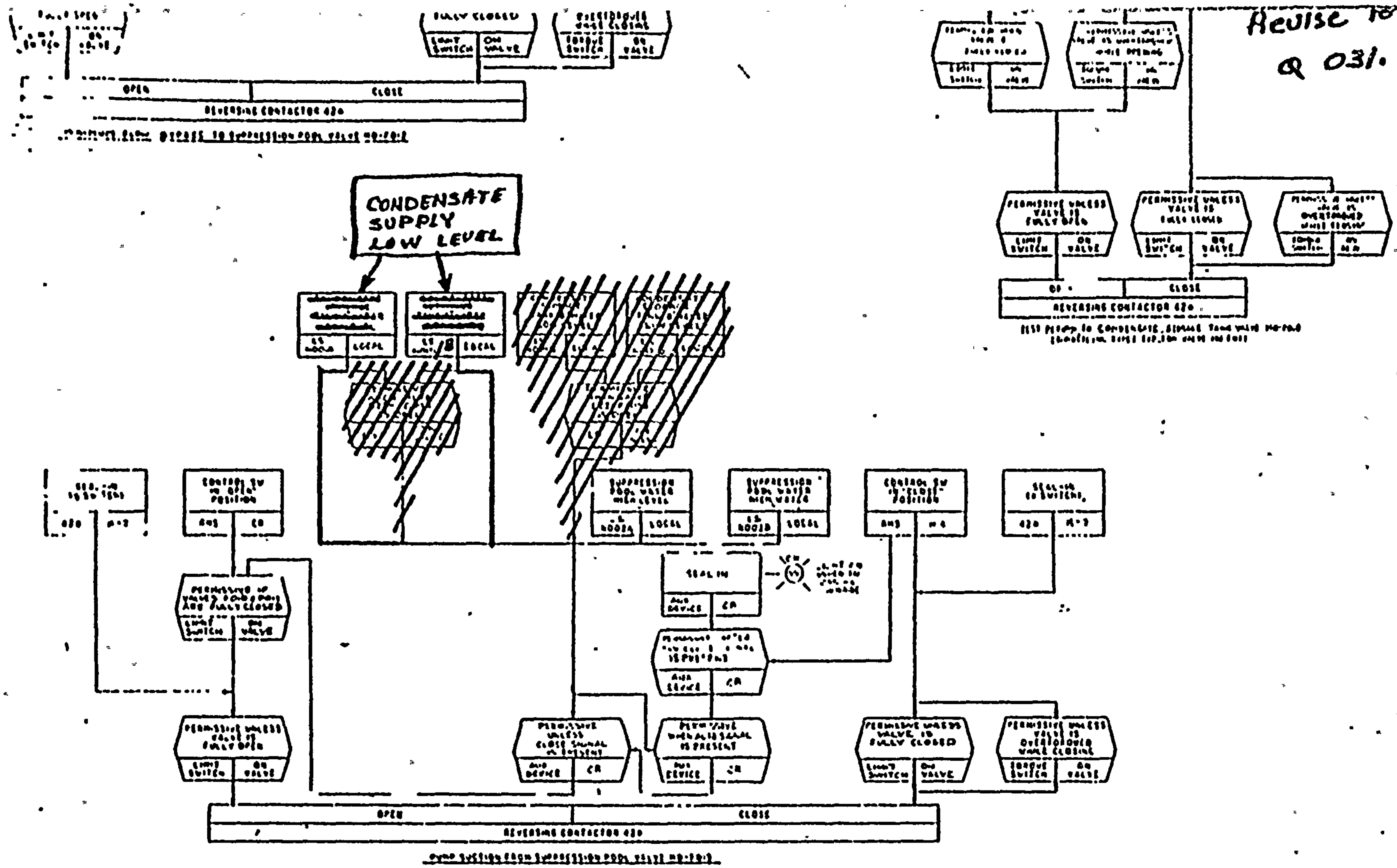
Schematic arrangements of system mechanical equipment is shown in Figure 7.3-9 (Nuclear Boiler P&ID). ADS component control logic is shown in Figure 7.3-10 (Nuclear Boiler FCD). Instrumentation specifications are listed in Tables 7.3-3 and 7.3-4. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-9 (Nuclear Boiler P&ID) and Figure 7.3-10 (Nuclear Boiler FCD).



HPCS P&ID

Revised for RRSCN 81-435

Revise for ... Q 031.128

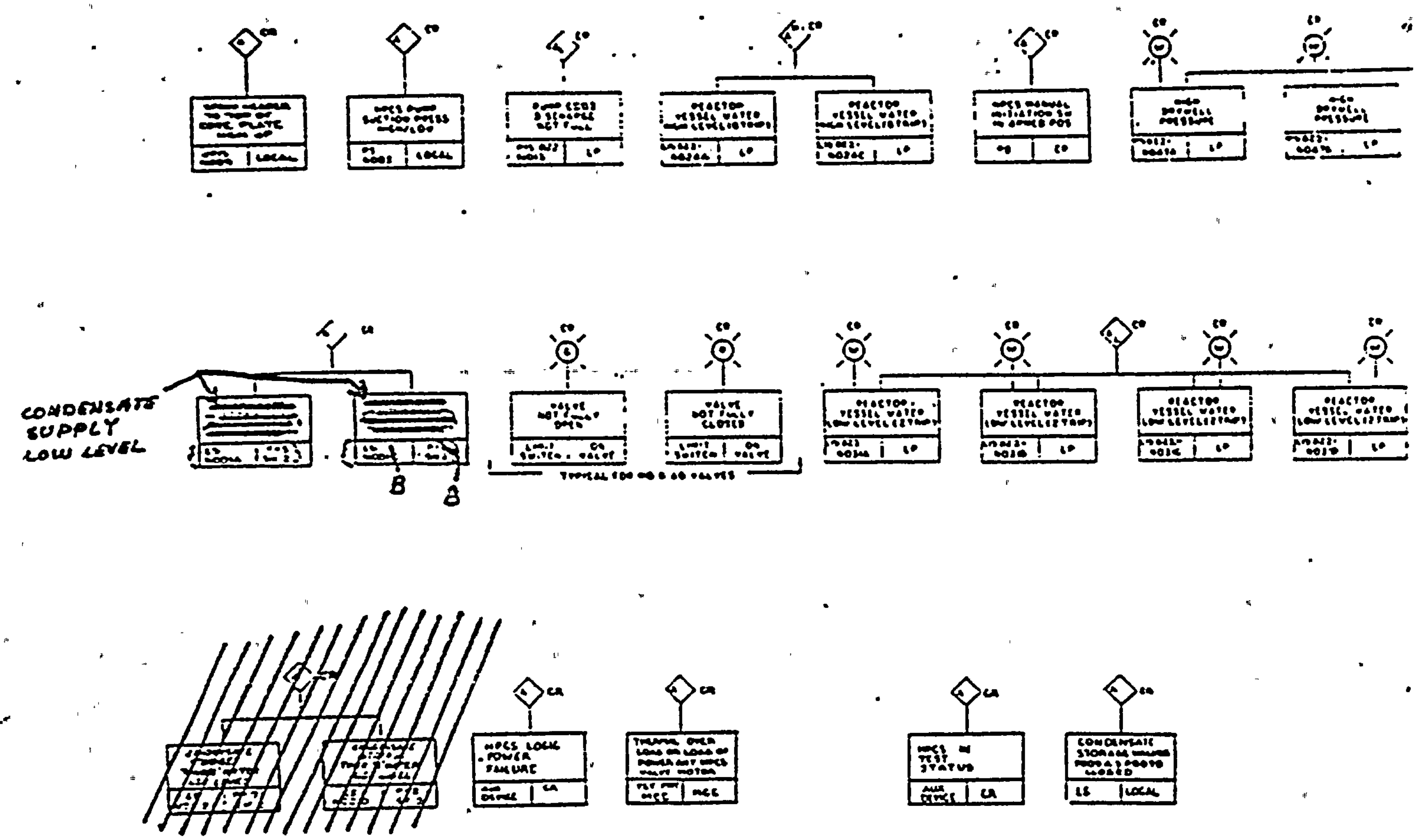


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY FCD SHEET 2

Figure 7.3-E

Revise for BRSCN 81-4

Q 031.128



HPCS FCD
sheet 2

Figure 7.3-8C

WASHIN