Washington Public Power Supply System
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G02-81-0427
October 27, 1981

Docket No. 50-397

Mr. A. Schwencer
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 AUXILIARY SYSTEMS BRANCH QUESTIONS

Reference: Letter, R. L. Tedesco to R. L. Ferguson, "Request for Additional Information for the review of the WNP-2 Facility", dated April 27, 1981

Enclosed are sixty (60) copies of the responses to the Auxiliary Systems Branch questions transmitted to the Supply System by the reference letter. These responses will be incorporated into the FSAR in Amendment No. 21.

The three (3) remaining responses will be submitted to the NRC by November 30, 1981.

Very truly yours,

G D Bouchey

G. D. Bouchey
Deputy Director, Safety & Security

GDB/CDT/jca
Enclsoures

cc: WS Chin - BPA
AD Toth - NRC
NS Reynolds - D&L
J Plunkett - NUS Corporation
R Auluck - NRC
OK Earle - B&R RO
EF Beckett - NPI
WNP-2 Files

B001
S 1/39

Q. 010.035
(3.4.1)

The FSAR states that the "Seismic Category I piping and electrical conduit penetrations that are below grade ... are ... not sealed against groundwater pressure." Demonstrate that the safety functions would not be compromised by water flowing into the building through these piping and conduit penetrations as the result of the following events:

a. Another compartment is flooded and water is flowing out of the building through the piping and conduit penetrations, resulting in saturated ground conditions.

b. A non-Seismic Category I tank ruptures emptying all of its contents.

Response:

Please refer to revised 3.4.1.4.2, page 3.4-4 for the information requested.*

The design and installation of the boots at pipe penetrations and of the silicon foam at conduit penetrations, described in 3.4.1.4.2, provide waterproof penetrations that are capable of preventing the compromise of safety functions by events such as those stated in the question.

The response to Question 010.010 has been similarly revised.*

*Draft FSAR page changes attached.

REPLACE WITH ATTACHED INSERT (A)

The lowest floor surface in the reactor building is the top of the foundation mat at elevation 422'-3" MSL. Since this is above the design basis groundwater level, the structure will be unaffected by the force effects of buoyancy and static water due to groundwater at elevation 420 feet MSL.

Groundwater elevation 420 feet MSL has been compared with foundation levels of Seismic Category I structures and it has been found that waterproofing is not required.

Seismic Category I piping and electrical conduit penetrations that are below grade are above the design basis groundwater table and are therefore not sealed against groundwater pressure.

The only materials underlying the site that might exhibit unfavorable response to seismic or other events under saturated soil conditions are the loose to medium dense, fine to coarse sand with scattered gravel, in the upper approximate 40 feet of the soil profile. These are removed and recompacted as structural fill, as described in 2.5.4.8 and 2.5.4.12. Structural fill supports the Seismic Category I structures, including the turbine generator building and service building, in the central plant complex. Structural fill, as required, is also utilized below the other Seismic Category I and safety-related structures including underground piping and electrical duct banks. The structural fill is compacted to a minimum of 75% relative density and an average relative density of not less than 85%.

To evaluate the possible effects of a gross rise in groundwater levels, Shannon and Wilson, soil consultants, performed a series of repetitive triaxial tests in identical soils in the dry and saturated states and concluded that saturation would not necessitate changes in allowable bearing pressures or settlement calculations as discussed in 2.5.4.10.

To provide conservatism in the design of structures, the seismic dynamic reponse of the structures and components is examined over a range of soil shear moduli, as discussed in 2.5.4.7.

According to 3.6, Groundwater, of the Shannon and Wilson Supplementary Soil Investigation in Appendix 2.5F, the compacted backfill discussed above will eliminate

Insert to Page 3.4-4:

Seismic Category I piping and electric conduit penetrations that are below grade are above the design basis groundwater table, and sealing against groundwater pressure is therefore not required. However, all pipes penetrating exterior walls are water-proofed sealed by boots installed on both sides of the wall penetration; all electrical conduit penetrations are through-wall sealed using silicon foam, also a waterproof sealer.

Q 010.10
(3.4.1)

Demonstrate that all piping and electrical penetrations in safety-related structures that are below the level of the Probable Maximum Flood are water-tight.

Response:

As stated in 3.4.1.4.1 the plant site grade is higher than the design basis flood elevation resulting from the probable maximum precipitation (PMP) event. Due to the short duration of the PMP flood, the ground water level at the plant site is not affected. As stated in 3.4.1.4.2, piping and electrical penetrations are above the design basis groundwater level and ~~[illegible]~~ therefore ~~not sealed~~ sealing against groundwater pressure is not required. However, all pipes penetrating exterior walls are water-proofed sealed by boots installed on both sides of the wall penetration; all electrical conduit penetrations are through-wall sealed using silicon foam, also a waterproof sealer.

Q. 010.036
(3.5.1)
RSP

It is the Staff's position that all safety-related equipment shall be appropriately protected against the effects of internally generated missiles in accordance with Title 10, Code of Federal Regulations Part 50, Appendix A, General Design Criteria 4. The effects of internally generated missiles such as valves stems, bonnets, control rod drive mechanisms, and high pressure accumulators impacting onto safety-related equipment must be evaluated. Appropriate protection must be provided to assure that a missile will not prevent a safe shutdown of the plant or result in uncontrolled release of radioactivity during normal operation or during the most severe design basis accident with the most limiting single active failure. Describe the means provided for assuring protection of safety-related equipment from all internally generated missiles.

Response:

The preferred method of protecting safety-related equipment from internally generated missiles is by separation and redundancy. In addition, potential missile sources have been oriented away from essential equipment wherever practical.

A pipe break/missile study evaluation is currently underway at WNP-2. This study will consider the effects of postulated valve stem, bonnet, control rod drive mechanism, and high pressure accumulator missiles. Appropriate protection will be provided where necessary.

Preliminary results scheduled for submittal in the last quarter of 1981 will address effects inside of containment and will also delineate methodology used in the study. A follow-up submittal will identify effects on the remaining areas outside of containment. Upon completion, the study will provide assurance that essential systems have been adequately protected from internally generated missiles.

Q. 010.037
(3.5.1.2)

Regulatory Guide 1.70, Revision 3, Section 3.5.1.2, requires that the structures, systems, and components protected by physical barriers should be identified. The discussion and the figures in the FSAR do not indicate where, if at all, physical missile barriers are used.

Identify all structures, systems, and components that are protected by physical barriers. Provide a description of the types of physical barriers that are employed at your plant.

Response:

At WNP-2, the preferred method of protecting essential systems and components from internally generated missils is by separation and redundancy. In addition, potential missile generating sources have been oriented away from essential systems and components wherever practical. This is discussed in 3.5.1.1 and 3.5.1.2.

The potential missiles described in 3.5.1.2 were investigated and found not to prevent safe cold shutdown. Therefore, specific missile barriers were not required.

The pipe break/missile study evaluation currently underway at WNP-2 will consider some additional types of internall generated missiles. This expanded scope may require barriers for protection of essential systems and components. The expanded approach for internally generated missiles, criteria for required barriers, and systems requiring protection will be provided by amendment to the FSAR in the last quarter of 1981.

Q. 010.038
(3.5.1.2)

Section 3.5.1.1.2 of the FSAR states that missile trajectories are selected to encompass the most adverse conditions. It is not clear from the information provided in the FSAR what the trajectories of the credible primary missiles would be and what systems might be disabled by the missiles.

Provide the bases for selection of the probable missile trajectories and show the trajectories on the appropriate FSAR figure. Include a discussion on the system, component, or structure that could be damaged or disabled by a missile. The extent of damage from each missile should be discussed.

Response:

The pipe break/missile study evaluation currently under way for WNP-2 will provide the answer to NRC Question 010.038. FSAR Section 3.5 will be revised as appropriate in the last quarter of 1981.

Q. 010.039
(3.5.1.2)

Section 3.5.1.1.3.2 states that thermowells and sample probes do not present potential hazards as postulated missiles affecting safe shutdown.

Provide justification to support this position on the thermowells and sample probes.

Response:

As stated in the response to NRC Question 211.013, themowells and sample probes were investigated for their potential of becoming postulated missiles.

Thermowells in high energy systems were found to have connections holding the thermowell to the system that were conservative by many folds. Some thermowells and sample probes are discounted as potential missiles if the piping system pressures are low and there are large factors of safety against failure. The remaining thermowells and sample probes are still postulated as missiles. All equipment that could be contacted by the missile, assuming a 10° half angle cone for the target area, was assumed to fail.

Potential thermowell missiles were found not to prevent safe cold shutdown or the release of unacceptable amounts of radioactive materials assuming a single active component failure and loss of offsite power.

At WNP-2, sample probes are one-inch or less in diameter and meet the requirements of Regulatory Guide 1.11. Pipe breaks were not postulated in lines one-inch or less as recommended in Branch Technical Positions APCSB 3-1 and MEB 3-1.

Q. 010.040
(3.5.1)

The FSAR states that the water lines are "...tornado-hardened". State your criteria for protecting pipes located outside buildings from tornado missiles, including depth below grade requirements and provide drawings which show all pertinent tornado protection features as necessary.

Response:

As stated in Section 3.5.3, buried safety-related piping required for safe shutdown is ensured adequate protection from tornado-generated missiles. Analysis of potential damage is performed using "Tornado Design Considerations for Nuclear Power Plants" by Bates and Swanson, 1967 (Reference 3.5-8). A 5-foot embedment depth is calculated to be acceptable to ensure pipe integrity.

The standby service water piping exits the pumphouses at a centerline elevation of 435'3" and immediately turns down at a 45° angle to elevation 432 feet., where the piping is routed to the reactor building in high relative density Quality Class I backfill. Grade level is at 440'6", providing an embedment depth of over 7 feet from the top of the pipe. Where the pipe exits the pumphouses, a 1-1/2" asphaltic concrete road with a 6" base coarse and 2" leveling coarse bed provides additional protection from tornado-generated missiles. Additionally, the two standby service water loops are separated by at least 20 feet to preclude loss of redundancy. The standby service water pumphouses, shown in Figures 3.5-48 and 3.5-49, are protected from tornado-generated missiles. The standby service water piping and the tower makeup water system from the river are the only safety-related water piping systems outside of tornado-protected buildings. The tower make-up system is only required in the event that the spray ring headers in the ultimate heat sink are lost in the tornado. The tower make-up piping to the river also satisfies the five foot embedment criteria. Protection from tornadoes and tornado missiles in regard to such piping has also been previously addressed in response to Questions 010.024 and 010.027.

Though not technically a piping system in line with this question, the control room remote air intakes are, of course, located remote to tornado-hardened buildings. The intake structures themselves are tornado hardened, however, (see Figure 3.8-59) and the piping from the structures meets the five foot embedment criteria.

It should be noted that Chapters 3.5 and 3.6 will be revised in line with the on-going pipe break/missile study. The information in the response to this question will be more clearly incorporated into the FSAR text at that time.

Q. 010.041
(4.6)

Demonstrate that the scram discharge system meets the criteria enumerated in the Generic Safety Evaluation Report BWR Scram Discharge System, dated December 1, 1980.

Response:

The scram discharge system for WNP-2 has been evaluated against the Generic Safety Evaluation Report, "BWR Scram Discharge System", dated December 1, 1980. In short, the evaluation indicated that the WNP-2 scram discharge system needed upgrading in the following areas:

1) Addition of redundant vent and drain isolation valves;
2) Addition of redundant and diverse level instrumentation for scram;
3) Relocation and repiping of instrument piping directly to the scram instrument volume;
4) Addition of new surveillance and operating procedures.

A summary of our evaluation results is provided below:

FUNCTIONAL CRITERIA

1. The scram discharge volume (SDV) shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting control-rod-drive scram performance.

   WNP-2 Compliance:

   WNP-2's SDV system is currently designed to meet the 3.34 gallons per drive requirement specified in the GE Design Specification 22A4260. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SAFETY CRITERIA

1. No single active failure of a component or service function shall prevent a reactor scram, under the most degraded conditions that are operationally accepted.

   WNP-2 Compliance:

   The WNP-2 system has been designed to meet single failure criteria. The SDV is designed with an integral instrument volume (IV) which provides direct and immediate detection of liquid accumulation. The SDV instrumentation is redundant and single failure proof (including partial loss of service functions).

2. No single failure shall result in uncontrolled loss of reactor coolant.

WNP-2 Compliance:

A redundant air-operated vent valve and drain valve will be added on the SDV in series to insure system isolation during reactor scrams. This includes independent solenoid valves for each set of air-operated vent and drain valves.

3. The scram discharge system instrumentation shall be designed to provide redundancy, to operate reliably under all conditions, and shall not be adversely affected by hydrodynamic forces or flow characteristics.

WNP-2 Compliance:

Six additional diverse level sensors will be added to the SDV system to ensure diversity and redundancy in level monitoring and scram functions. Common cause failures will be considered in the selection of the instruments. This is in agreement with Alternative 3 of the "Acceptable Compliance" statement for this item in the Generic SER.

4. System operating conditions which are required for scram shall be continuously monitored.

WNP-2 Compliance:

The addition of the level switches described in 3 above and periodic surveillance testing of the instruments will provide a continuous means of monitoring the SDV liquid level and insuring instrument reliability. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

5. Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

WNP-2 Compliance:

During routine surveillance testing, instrument repair or calibration the associated logic will be placed in a half-scram (1 out of 2) configuration[1] in accordance with the plants technical specifications. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

OPERATIONAL CRITERIA

1. Level instrumentation shall be designed to be maintained, tested, or calibration during plant operation without causing a scram;

2. The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation;

3. The system shall be designed to minimize the exposure of operating personnel to radiation;

4. Vent paths shall be provided to assure adequate draining in preparation for scram reset;

5. Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

WNP-2 Compliance:

1. The system logic is designed as a one out of two, twice configuration. Each of the associated instrument channels is capable of being separately isolated for maintenance, testing or calibration without inadvertently scramming the reactor.

2. The SDV is provided with a high liquid level alarm on each IV to alert the operator to liquid accumulation in the SDV.

3. The SDV system has been designed in accordance with GE design specification 22A4260 to minimize the exposure of operating personnel to radiation. In addition, the system is being reviewed as part of the WNP-2 ALARA program.

4. The SDV vents directly to the reactor building atmosphere and is independent from other plant vent system.

5. The vent and drain system for the SDV is totally independent from other plant systems, and is therefore not susceptible to blockage or water buildup through system interfaces.

DESIGN CRITERIA

1. The scram discharge headers shall be sized in accordance with GE OER-54 and shall be hydraulically coupled to the instrumented volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on a plant-specific maximum inleakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum inleakage is the maximum flow rate through the scram discharge line without control-rod motion summed over all control rods. The analysis should show no need for vents or drains.

   WNP-2 Compliance:

   WNP-2's IVs have been designed as vertical extensions attached directly to the SDV. This configuration provides a direct hydraulic couple between the SDV and IVs and insures immediate and continuous liquid level monitor in the SDV. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

2. Level instrumentation shall be provided for automatic scram initiation while sufficient volume exists in the scram discharge volume.

   WNP-2 Compliance:

   WNP-2's SDV is adequately coupled to the IV to allow proper instrument operation. The SDV instrument setpoint for scram was established to insure an available volume of 3.34 gallons per drive (185 drives). This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

3. Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

   WNP-2 Compliance:

   All the WNP-2 SDV instrumentation will be relocated and repiped directly to the IV instead of the vent and drain piping. Procedures will be modified to include functional testing of SDV level instrumentation after each scram. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

4. The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

   WNP-2 Compliance:

   The addition of the redundant and diverse instruments described under Safety Criterion 3 and rerouting of the instrument piping to the IV provide an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

5. Structural and component design shall consider loads and conditions including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations, and adverse environments.

   WNP-2 Compliance:

   The WNP-2 SDV design compliance with the latest GE design criteria as outlined in GE Design Specification 22A4260. In addition, the system will be reviewed as part of the equipment qualification program.

6. The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

   WNP-2 Compliance:

   WNP-2's present design configuration meets these requirements.

7. Any reductions in the system piping flow path shall be analyzed to assure system reliability and operability under all modes of operation.

   WNP-2 Compliance:

   WNP-2's SDV header system is designed as a continually expanding path from the 185 3/4" individual scram discharge (withdrawal) lines to one of two integrated SDV/IV systems (one system per approximately half the drives). Each integrated SDV/IV system consists of a continuously downsloping piping run expanding from the SDV (consisting

of seven 6" return headers from the individual hydraulic control unit (HCU) banks to an 8" combined return header) to the 12" vertically oriented IV. The location where blockage need be assumed (piping less than 2" diameter) is in the 3/4" discharge line from the individual HCU. Blockage here would only cause failure of one control rod to insert. This is an acceptable consequence for a single failure and has been evaluated as part of the plant design basis. Accordingly, this design complies with the "Acceptable Compliance" statement for this item in the Generic SER.

8. System piping geometry (i.e., pitch, line size, orientation) shall be such that the system drains continuously during normal plant operation.

   WNP-2 Compliance:

   The WNP-2 SDV has been designed to insure a positive downward slope of scram header and drain piping.

9. Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation.

   WNP-2 Compliance:

   Each IV is provided with high liquid level and rod block instrumentation attached directly to it. The generic SER states that this is acceptable.

10. Vent and drain line valves shall be provided to contain the scram discharge water, with a single active failure and to minimize operational exposure.

    WNP-2 Compliance:

    As stated under Safety Criterion 2, redundant air-operated vent and drain valves will be provided for system isolation. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SURVEILLANCE CRITERIA

1. Vent and drain valves shall be periodically tested.

WNP-2 Compliance:

The vent and drain [illegible] will be tested in accordance with the plant techn[illegible] specification to verify valve closure in less than [illegible] conds (current GE specification). This is an ac[illegible]ptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generi[illegible] SER.

2. Verifying and level detection instrumentation shall be periodically tested in place.

WNP-2 Compliance:

The SDV instrumentation will be tested in accordance with the plants technical specification which will include post[1] scram testing to verify instrument operability.

3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control-rod density.

WNP-2 Compliance:

Surveillance testing will be performed in accordance with the plants technical specifications.

[1] The plant technical specifications will be based on the Standard Technical Specifications for Boiling Water Reactors, NUREG-0123, provided by the NRC.

Q. 010.042
(4.6)
RSP

Demonstrate that a slow or partial loss of air pressure to the scram discharge valves will not result in the following:

1) Rapid filling of both the scram discharge volume and the instrument volume due to the lifting of most or all scram discharge valves, with consequent loss of adequate scram discharge volume.

2) Loss of reactor coolant due to the combination of lifting of most or all scram discharge valves, without compensating closure of the vent and drain valves, with consequent environmental effects inside containment.

Unless it can be demonstrated that no adverse effects can result, a system shall be provided and describeo in this section to protect against these two conditions.

Response:

1) The WNP-2 scram discharge instrument volume (IV) is adequately hydraulically coupled to the scram discharge volume (SDV), i.e., the IV is connected directly to the SDV with piping of a diameter equal to or greater than the diameter of the SDV headers. This allows for direct and immediate detection of liquid buildup so that the ability to scram is ensured, even in the event of lifting of most or all of the scram valves, when the water buildup reaches scram initiation level in the IV.

   Note: The basis of the instrument volume high level scram setpoint and the IV/SDV physical arrangement provides for scram action before significant scram discharge volume reduction occurs which could affect scram capability.

2) The partial loss of air pressure does not result in the uncontrolled release of reactor coolant to the reactor building should all or most of the scram discharge valves lift. When the water buildup reaches scram initiation level in the IV, a scram signal is produced.

This will cause the air supply to the vent and drain valves to vent, thereby ensuring that the vent and drain valves close and isolate. For leakage rates which do not result in buildup in the IV, the leak will drain to the reactor building equipment drain system. The drain system will alarm for leakage rates greater than five (5) gallons per minute. The operator can then take appropriate action, e.g., isolate the leak, scram the reactor, increase air pressure, etc., as required.

See response to Question 010.042 for additional information.

Q. 010.044
(4.6.1.1.2.4)

Table 1.3-8 indicates specific design changes from the PSAR to the FSAR for the CRD system. The design changes for the CRD return line modification addressed in Question 211.019, have not been included in the text description of the FSAR and Figures 4.6-5a, 4.6-5b, and 4.6-6a have not been revised. Revise the text description in the FSAR to reflect the specific design changes in Table 1.3-8 for the CRD system and modify the above figures accordingly.

Response:

Figures 4.6-5a, 4.6-5b, 4.6-6a, 4.6-6b, and 4.6-6c are revised to reflect the CRD return line modification. Figures 4.6-6d and 4.6-6e are deleted. Also, text sections 4.6.1.1.2.4, 4.6.1.1.2.4.1, 4.6.1.1.2.4.2.3, 4.6.1.1.2.4.2.4 and 4.6.2.3.2.2.8 are revised accordingly. Text section 4.6.1.1.2.4.2.5 is deleted.*

*FSAR revised page changes attached.

e. Metal piston rings are Haynes 25 alloy.

f. Certain wear surfaces are hard-faced with Colmonoy 6.

g. Nitriding by a proprietary new Malcomizing process and chromium plating are used in certain areas where resistance to abrasion is necessary.

h. The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code.

4.6.1.1.2.4 Control Rod Drive Hydraulic System

The control rod drive hydraulic system (Figures 4.6-5a, b) supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCU). The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU, the exhaust header, and ~~into the cooling water header.~~ is returned to ~~the~~ reactor vessel via the HCUs of non-moving drives There are as many HCUs as the number of control rod drives.

4.6.1.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown in Figures 4.6-5a, b, and 4.6-6. The hydraulic requirements, identified by the function they perform, are as follows:

a. An accumulator hydraulic charging pressure of approximately 1400 to 1500 psig is required. Flow to the accumulators is required only during scram reset or system startup.

b. Drive water header pressure of approximately 260 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required.

c. Cooling water to the drives is required at approximately 20 psi above reactor vessel pressure and at a flow rate of approximately ~~0.20 to~~ 0.34 gpm per drive unit. (Cooling water to a drive can be interrupted for short periods without damaging the drive.)

d. ~~The drive header pressure will be no more than 5 psi above the cooling water header pressure. (The pressure in this line must be kept as low as possible to avoid interference with normal drive movement.)~~

d ~~e~~. The scram discharge volume is sized to receive and contain all the water discharged by the drives during a scram; a minimum volume of 3.34 gal. per drive is required (excluding the instrument volume).

4.6.1.1.2.4.2 System Description

The CRD hydraulic systems provide the required functions with the pumps, filter, valves, instrumentation, and piping shown in Figures 4.6-5a, b and described in the following paragraphs.

Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from a condensate supply header which takes suction from the condensate treatment system and/or condensate storage tanks depending on plant operation. One spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent immediate pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by two filters in the system. The pump suction filter is a disposable element type with a 25-micron absolute rating. A 250-micron strainer in the filter bypass line protects the pump when the filter is being serviced. The drive water filter downstream of the pump is a cleanable element type with a 50-micron absolute rating. A differential pressure indicator and control room alarm monitor the filter element as it collects foreign materials.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Accumulator charging pressure is established by the discharge pressure of the system supply pump. During scram the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to "run out" (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and closes the flow control valve. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the control room with a pressure indicator and high pressure alarm.

During normal operation the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the ~~drive~~ drive/cooling water pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally passes from the drive water pressure stage through two solenoid operated stabilizing valves (arranged in parallel) and then goes into the cooling water header. The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by ~~while~~ closing the appropriate stabilizing valve, at the sa[me] time, ope[ning] the drive direct[ional] control and exhaust solenoid valves. Thus, flow through the ~~drive~~ drive/cooling water pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

211.180-4

Add Insert 'A'

4.6.1.1.2.4.2.4 Cooling Water Header

~~The cooling water header is located just upstream of the return line control valve. An automatic pressure regulating valve controls the pressure in this header, which is set to produce the desired cooling water flow to the drives. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The temperature of each drive is recorded in the control room, and excessive temperatures are annunciated.~~

~~4.6.1.1.2.4.2.5 Return Line~~

~~The return line routes excess flow from the control rod drive hydraulic system (water not used for charging of accumulators, movement of drives or cooling) through the reactor water cleanup system to the reactor feedwater line. A pressure control valve in this line is manually adjusted from the control room to produce the desired pressure. The flow through this valve is virtually constant. Therefore, once adjusted, the drive pressure control valve and the return water control valve can maintain their required pressure independent of reactor pressure.~~

4.6.1.1.2.4.2.6 Scram Discharge Volume

renumber as req'd

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all [illegible] water discharged by the drives during a scram, indep[illegible] of the instrument volume.

During normal plant operation the scram discharge v[illegible] is empty, and vented to atmosphere through its open v[illegible] and drain valve. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Lights in the control room indicate the position of these valves.

INSERT A

The cooling water header is located downstream from the drive/cooling water pressure valve. The drive/cooling water pressure control valve is manually adjusted from the control room to produce the required drive/cooling water pressure balance.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive/cooling water pressure control valve will maintain the correct drive pressure and cooling water pressure, independent of reactor vessel pressure. Changes in setting of the pressure control valves are required only to adjust for changes in the cooling requirements of the drives, as the drive seal characteristics change with time. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The temperature of each drive is indicated and recorded, and excessive temperatures are annunciated in the control room.

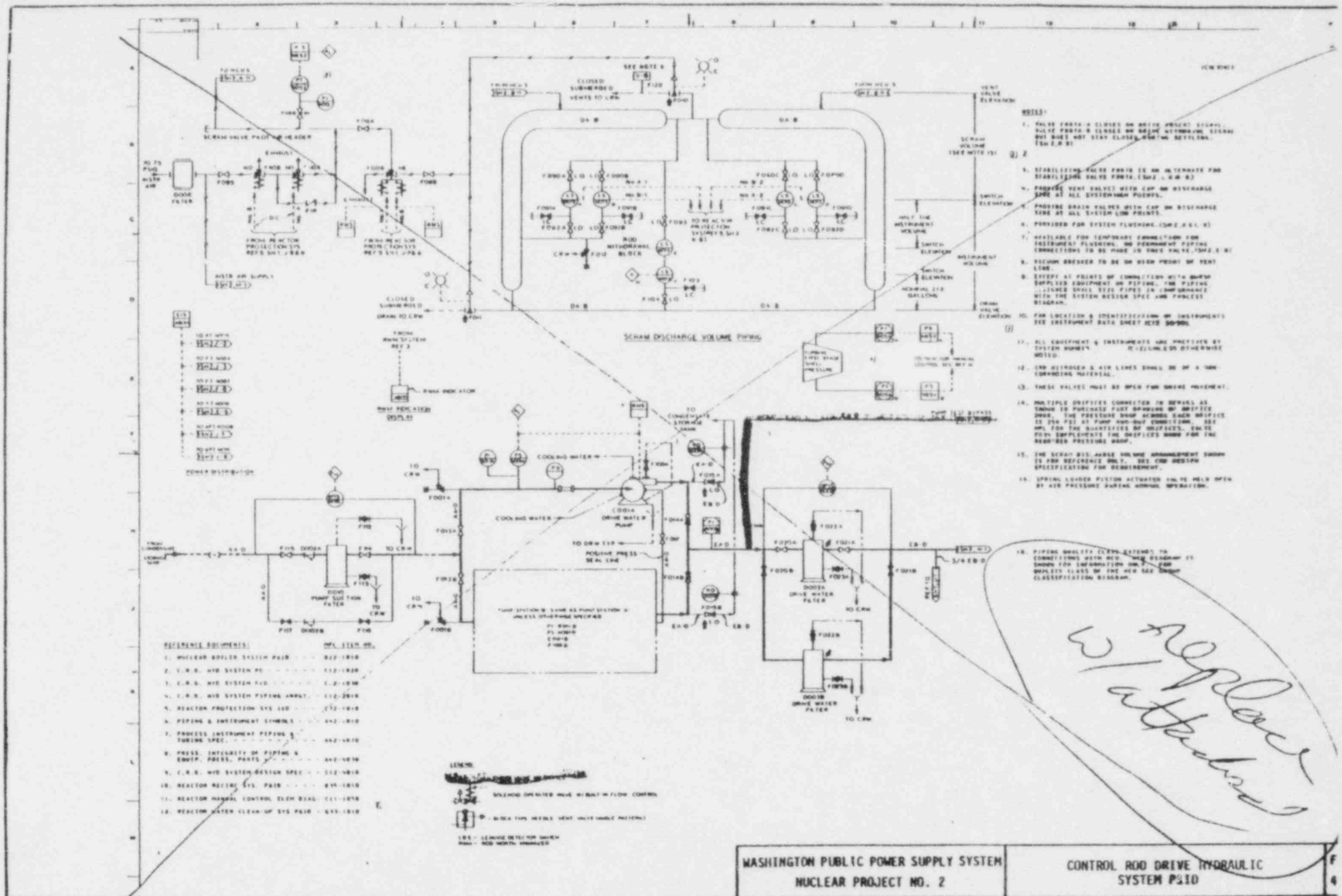
If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the control room.

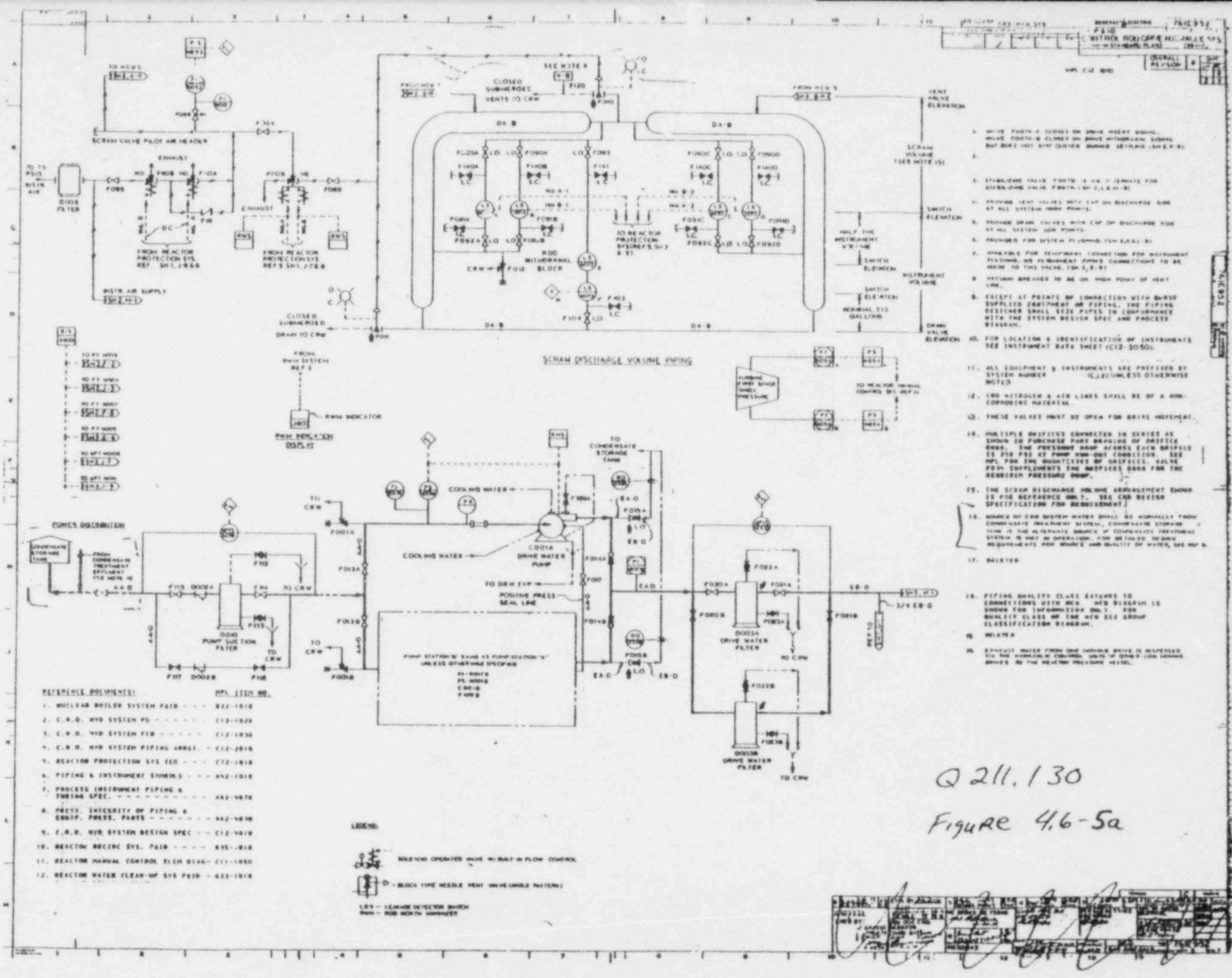
If the plug failure were to occur during control rod withdrawal, (it would not be possible to unlatch the drive <u>after</u> such a failure) the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 feet per second. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.3.2.2.8 ~~Drive~~ Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive/cooling water pressure control valve. This valve is either a motor operated valve or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

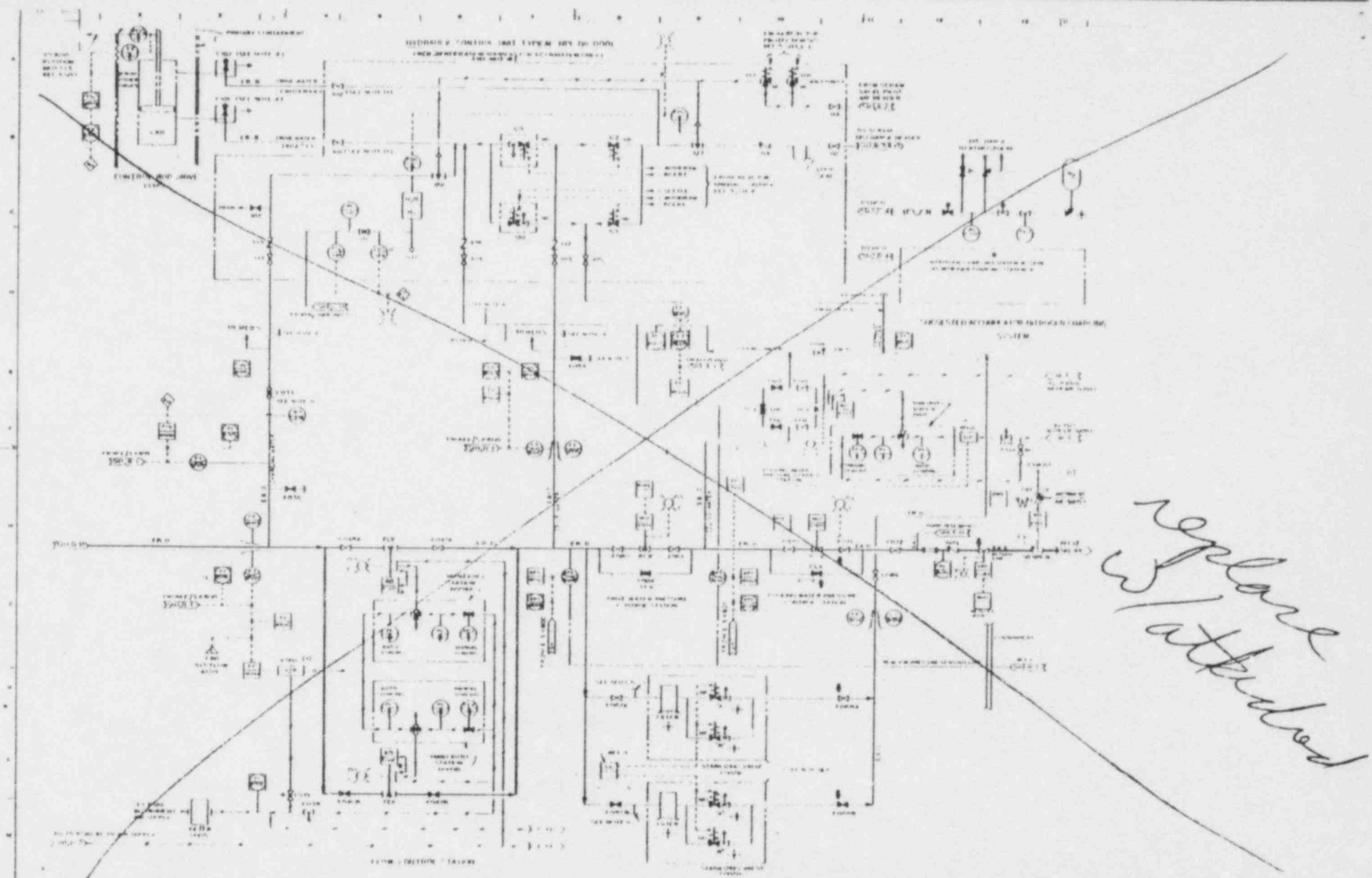
If the flow through the ~~drive~~ drive/cooling water pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this conditon during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations indicate that the drive would accelerate from 3 in./sec to approximately 6.5 in./sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.


SCRAM DISCHARGE VOLUME PIPING
SCRAM VALVE PILOT AIR HEADER
CLOSED SUBMERGED VENTS TO LRW
CLOSED SUBMERGED DRAIN TO CRW
ROD WITHDRAWAL BLOCK
COOLING WATER
DRIVE WATER PUMP
PUMP SUCTION FILTER
DRIVE WATER FILTER
TO CRW
POWER DISTRIBUTION
REFERENCE DOCUMENTS:
LEGEND:
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2
CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID
replace w/attached



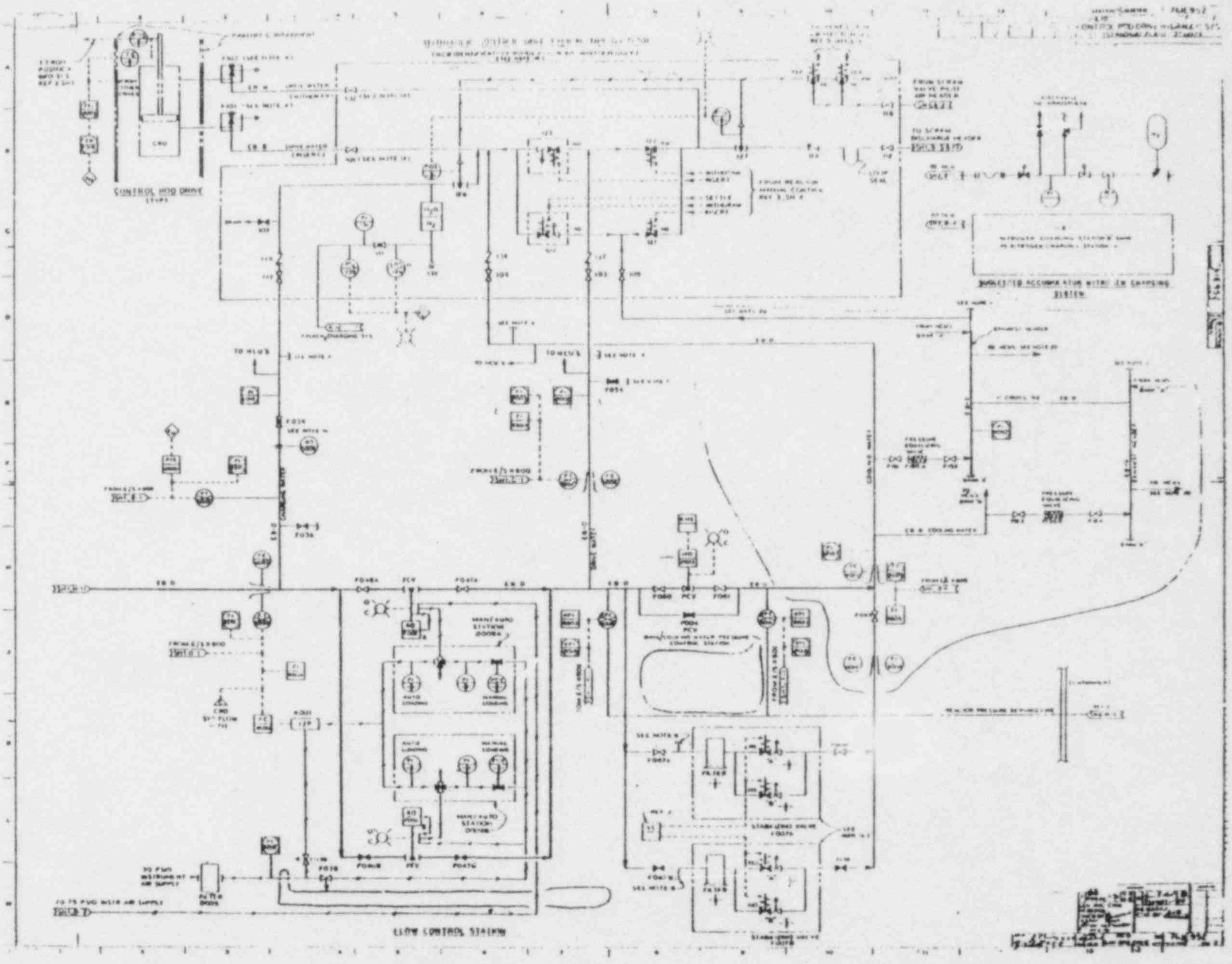
Q 211.130

Figure 4.6-5a

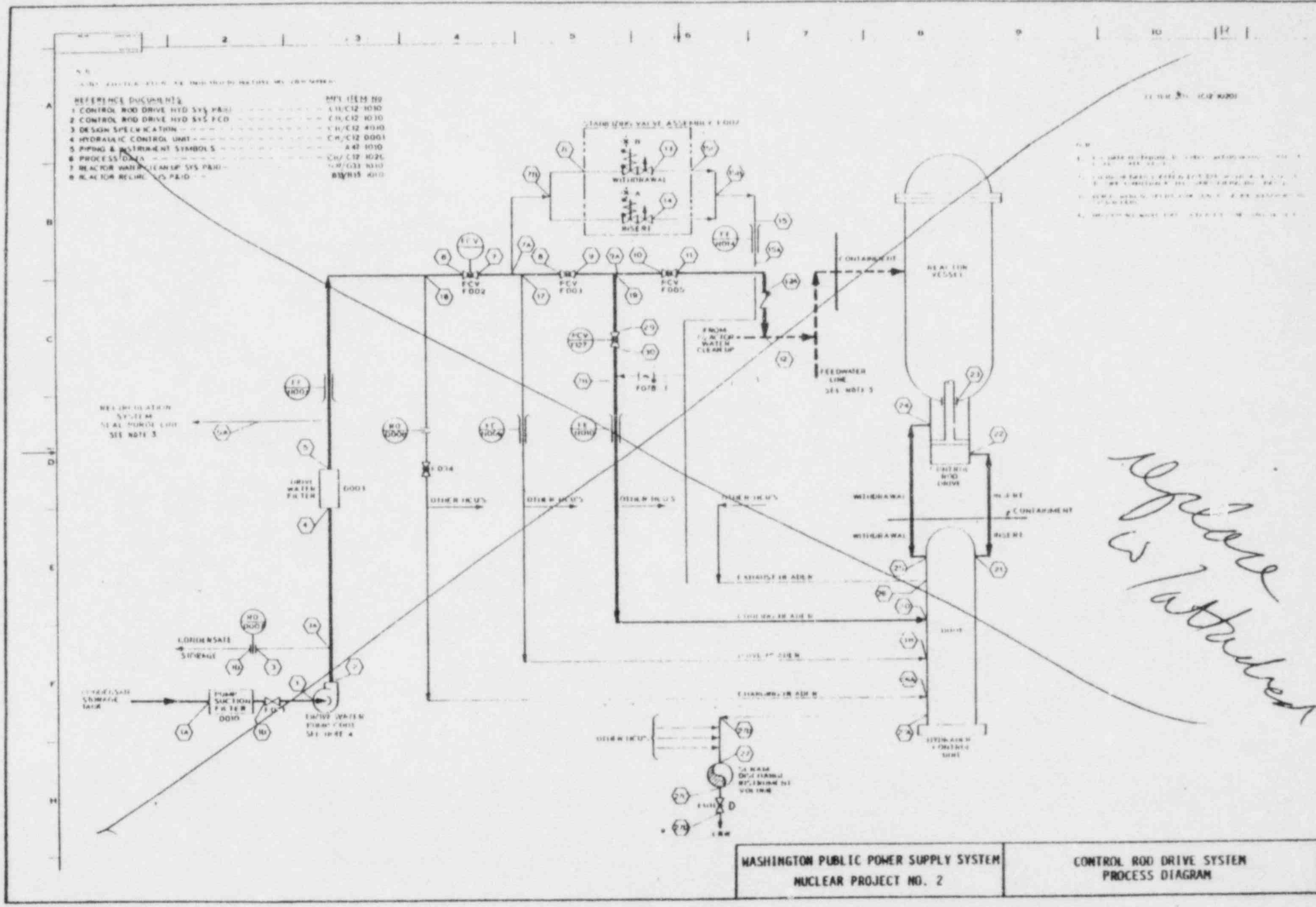


replace
w/attached

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID



Replaces FSAR
Figure 4.6-5b



Replace w/attached

GENERAL ELECTRIC

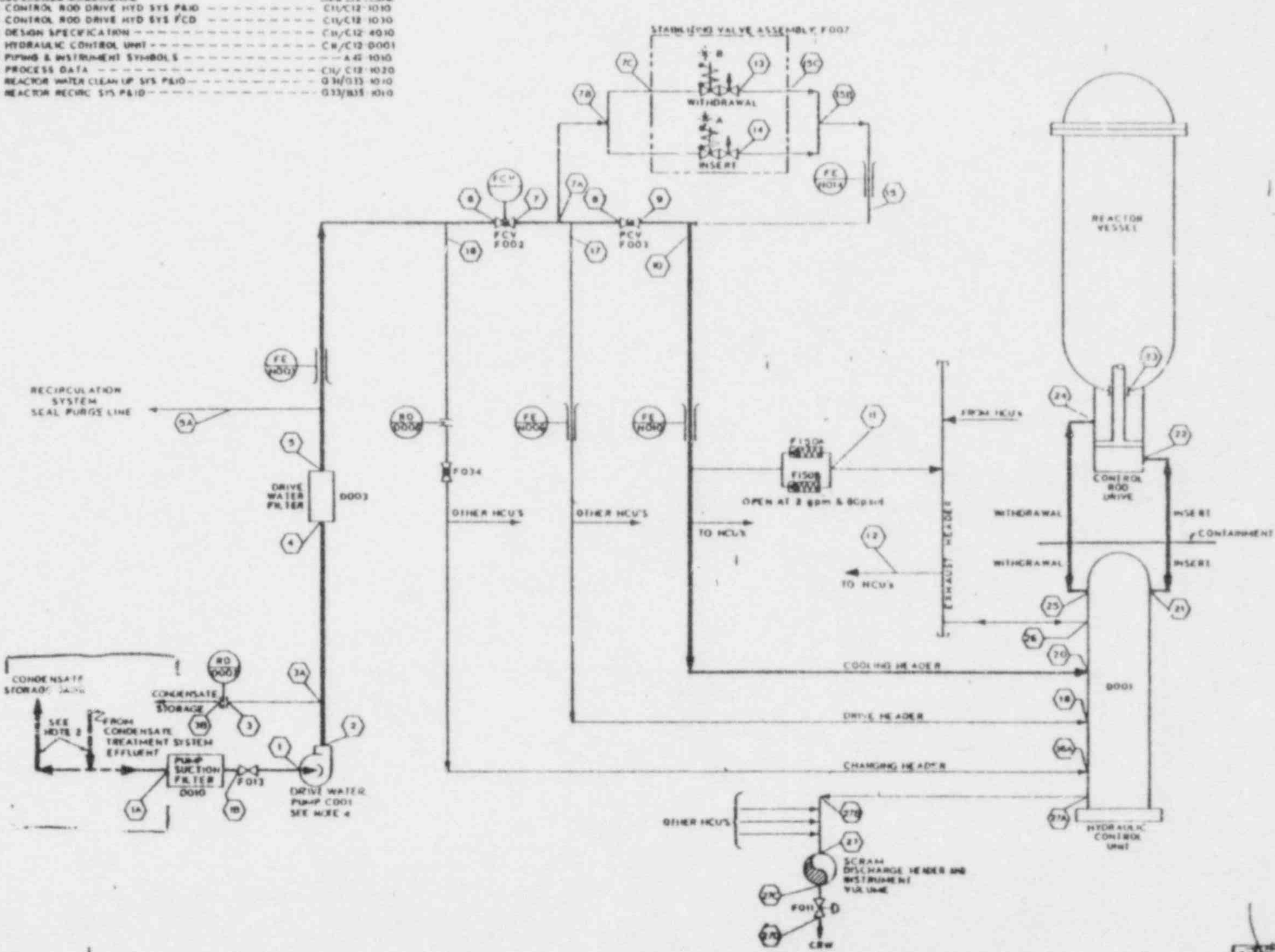
PROCESS DIAGRAM

112D144B

CONTROL ROD DRIVE HYD SYS

REFERENCE DOCUMENTS

	MPL ITEM No
1 CONTROL ROD DRIVE HYD SYS P&ID	C11/C12-1010
2 CONTROL ROD DRIVE HYD SYS PCD	C11/C12-1010
3 DESIGN SPECIFICATION	C11/C12-4010
4 HYDRAULIC CONTROL UNIT	C11/C12-D001
5 PIPING & INSTRUMENT SYMBOLS	A41-1010
6 PROCESS DATA	C11/C12-1020
7 REACTOR WATER CLEAN UP SYS P&ID	G31/G33-1010
8 REACTOR RECIRC SYS P&ID	G33/B35-1010



NOTES:

1. FOR DATA PERTAINING TO NUMBERS WITHIN HEXAGONS REFER TO PROCESS DATA REF 6.
2. SOURCE OF CRD SYSTEM WATER SHALL BE NORMALLY FROM CONDENSATE TREATMENT SYSTEM. CONDENSATE STORAGE TANK IS THE ALTERNATE SOURCE IF CONDENSATE TREATMENT SYSTEM IS NOT IN OPERATION. FOR DETAILED DESIGN REQUIREMENTS FOR SOURCE AND QUALITY OF WATER, SEE REF 3.
3. DELETED.
4. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.

112D144B

Replaces FSAR Figure 4.6-6a

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	106.9	106.9	106.9	19.0	87.9	87.9	10.0	77.9	77.9	71.9	71.9
PRESS. PSIG	21.	19.	1429.	1429.	1401.	1388.	1388.	1383.	PR+260.	PR+260.	PR+60.

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	15.0	15.0	15.0	2.0	4.0	6.0	0.	/	/	56.9	.34MAX
PRESS. PSIG	PR+60.	PR+50.	PR+50.MAX	PR+20.MAX	PR+20.MAX	PR+20.MAX	[illegible]			PR+60.	PR+15.

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	.34MAX	.34MAX	.34MAX	0	/	0	0	62.9	56.9	56.9	
PRESS. PSIG	PR+14.	PR+14.	PR	PR		PR+20.MAX	0	PR+20.MAX	PR+60.	PR+20.MAX	

CONDITIONS

1. DRIVES LATCHED.
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES.

MODE A SIZES THE COOLING WATER HEADERS
MINIMUM REQUIRED PRESSURE AT POSITION 1A IS SHOWN.
PRESSURE AT LOCATION 16 SHALL NOT EXCEED 1510 PSIG
PRESSURE AT LOCATION 28 SHALL NOT EXCEED PR+20.
LINE LOSS FROM LOCATION 30 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	"	"	"	"	"	"	"	"	"	"	"
PRESS. PSIG	"	"	"	"	"	"	"	"	"	"	"

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	"	"	15	"	0	2.0	"	4.0	4.0	"	/
PRESS. PSIG	"	"	"	"	"	"	"	PR+260.	PR+250.	"	"

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	4.0	4.0	1.3	.7	.7	.7	"	56.9	"	"	
PRESS. PSIG	PR+81.	PR+90.	"	PR+20.MAX	PR+20.MAX	PR+20.MAX	"	PR+20.MAX	"	"	

CONDITIONS:

1. DRIVES INSERTING.
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS

FOR REVISIONS SEE SHEET 1

THIS SHEET TO BE USED WITH PROCESS DIAGRAM 112D1448

(replace w/ attached)

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6b

GENERAL ELECTRIC

PROCESS DIAG DATA

CONTROL ROD DRIVE HYD SYSTEM

112D1448AD

OVERALL REVISION 2

MPL NO. C12-1020

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	63	0	0	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	PR	PR	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	4	6	0	0	0		.34 MAX	.34 MAX	.34 MAX	.34 MAX	0	0	0	0	
PRESSURE PSIG	PR + 30	PR + 30	1455				PR + 15	PR + 14	PR + 14	PR	PR		PR	0	

CONDITIONS:

1. DRIVES LATCHED
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES. MINIMUM REQUIRED PRESSURE AT POSITION 1A IS 20 FEET OF WATER AT 200 GPM.

MODE A SIZES THE COOLING WATER HEADERS.

LINE LOSS FROM LOCATION 10 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

(FOR NOTES SEE SHEET 2.)

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	62	57	57	58	0	.7	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	<PR + 8	<PR + 8	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	2	0	4	4		0	4	4	1.3	.7	.7	.7	0	
PRESSURE PSIG	PR + 30	PR + 30	1455	PR + 260	PR + 250		PR + 15	PR + 51	PR + 50	PR	PR + 20 MAX	PR + 20 MAX	PR + 8 MAX	0	

CONDITIONS:

1. DRIVE INSERTING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS.

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	45	45	45	20	25	25	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	21	1550	1550									SEE NOTE 9	SEE NOTE 9	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	0	0	0		0	90	90	3.5	30	30	0.1 + SEE NOTE 9	APPROX 5565	
PRESSURE PSIG								1167 MIN	731 MIN	PR	256 MAX	54		65 MAX	

SEE NOTE 10

CONDITIONS:

1. DRIVES SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES.

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	200	200	200	20	180	180	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	19	1210						>PR	>PR	>PR	>PR	>PR	>PR	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	155	0	0		0	0.92	0.92	0.92	SEE NOTE 9	SEE NOTE 9	0.1	0	
PRESSURE PSIG			909					76	76	PR	65 MAX	65 MAX		65 MAX	

SEE NOTE 10

CONDITIONS:

1. SCRAMMING OF DRIVES COMPLETED
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE.

NOTE: MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565 PSIG.

112D1448AD

Replaces FSAR
Figure
4.6-6b

replace w/attachment

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	*	*	*	*	*	*	*	*	*	*	*
PRESS. PSIG	*	*	*	*	*	*	*	*	*	*	*

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	*	*	*	*	*	*	*	*	*	/	0
PRESS. PSIG	*	*	*	*	*	*	*	*	*		*

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	90.	90.	-3.6	29.6	29.6	*	APPROX.5476	*	*	*	
PRESS. PSIG	1167.MIN	731.MIN	*	256. MAX	94.	*	7.7				

CONDITIONS:

1. DRIVE SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000. PSIG
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9
FLOW, GPM	200.	200.	200.	18.	181.	181.	10.	5.	5.	5.	5.
PRESS. PSIG	6.	-4.	1141.	1141.	1064.	1010.	1010	988.	PR	PR	PR

LOCATION	10	11	12	13	14	15	16	17	18	19	20
FLOW, GPM	5.	5.	5.	/	/	/	166.	/	/	/	/
PRESS. PSIG	PR	PR	PR				988.				

LOCATION	21	22	23	24	25	26	27	28	29	30	
FLOW, GPM	0.95	0.95	1.48	-0.53	-0.53	/	0	/	/	/	
PRESS. PSIG	76.	76.	*	65.MAX	65.MAX		65.MAX				

CONDITIONS:

1. SCRAMMING OF DRIVES COMPLETED.
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE

MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565. PSIG.

SEE SHEET 1 FOR REVISIONS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6c

GENERAL ELECTRIC

PROCESS DIAG DATA

112D1448AD

CONTROL ROD DRIVE HYD SYSTEM

TABLE 1

LOCATION	1A-1B	1B--1	2---6	3A-3B	8--9	7A-7B	7B-7C
DESIGN PRESS. (PSIG)	150	150	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	4	4	2	1	1.5	1	0.75

LOCATION	10-20	11-12	15B-15C	15-15B	16-16A	17-18	12-26
DESIGN PRESS. (PSIG)	1750	1750	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	2**	1	0.75	1	2	1	1

LOCATION	21-22	24-25	27A-27B	27B-27	27-27C	27C-27D	5A
DESIGN PRESS. (PSIG)	1750	1750	1250	1250	1250	1250	1750
DESIGN TEMP. (DEG F)	150	550	280	280	280	280	150
ESTIMATED LINE SIZE (INCHES)	1	0.75	0.75	*	10	2	.75

* SEE CRD SYSTEM DESIGN SPECIFICATION.

** 2 INCH HEADER TO EACH HALF OF THE TOTAL QUANTITY OF HCU'S.

NOTES:

1. DEFINITION OF SYMBOLS
   PR - INDICATES PRESSURE OF THE REACTOR
2. MAXIMUM OPERATING TEMPERATURES
   THE MAXIMUM SYSTEM OPERATING TEMPERATURE WILL NOT EXCEED 150 DEG. F. FROM LOCATION 1 THROUGH 27 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAXIMUM TEMP. (DEG. F.)
MODE A -	23	200
MODE C -	23	546
	24	546
	25	280
	27	280
MODE D -	23	200
	24	280
	25	280
	27	280

3. MODE A -
   A. MAXIMUM CHARGING WATER PRESSURE SHALL BE 1600 PSIG NOMINAL, ACCUMULATOR PRECHARGE PRESSURE SHALL BE 575 PSIG NOMINAL, 580 PSIG MAXIMUM AT 70° F.
   B. DELETED
   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR + 15 PSIG. FOR THE CONDITIONS INDICATED.
   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.
4. MODE B -
   A. LOCATION 13 AND 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSES ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 18 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR + 250 PSIG. FOR THE CONDITIONS INDICATED.
5. MODE C -
   A. DELETED
   B. THE 546 DEG. F. TEMPERATURE LISTED IN NOTE 2 FOR MODE C POSITIONS 23 AND 24 SHALL BE USED ONLY IN DETERMINING THE MINIMUM PIPE WALL THICKNESS IN VICINITY OF THE DRIVE HOUSING AND NOT IN DETERMINING STRESSES DUE TO THERMAL EXPANSION. IN DETERMINING MINIMUM WALL THICKNESS IT MAY BE ASSUMED THAT THIS TEMPERATURE OCCURS LESS THAN 1 PERCENT OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYD. SYSTEM DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESSES DUE TO THERMAL EXPANSION.
   C. LOCATION 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 435 PSI AT 50 GPM FOR ANY CRD.
   D. LOCATION 23 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL, INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE THE REACTOR CAN REACH 10 GPM PER DRIVE.
   E. LOCATION 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 30 GPM FOR ANY CRD.
   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.
   G. SCRAM DRAIN VALVE F011 AND VENT VALVE F010 CLOSE WITH A SCRAM SIGNAL.
6. MODE D -
   A. DELETED
   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 50 PERCENT STROKE IS LESS THAN 65 PSIG.
7. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.
8. PROCESS DIAGRAM 112D1448 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.
9. DURING SCRAM, THIS FLOW WILL BE DIRECTED INTO THE SCRAM DISCHARGE VOLUME. FOLLOWING SCRAM, THIS FLOW WILL DECLINE AS VALVE F002 CLOSES AND AS THE SCRAM DISCHARGE VOLUME PRESSURIZES TO EQUAL THE REACTOR PRESSURE. AFTER THE SCRAM DISCHARGE VOLUME AND THE REACTOR VESSEL PRESSURE HAVE EQUALIZED, FLOW WILL BE DIVERTED TO THE REACTOR VESSEL VIA THE CRD WITHDRAW LINES AT A FLOW RATE DEPENDENT ON THE REACTOR PRESSURE:
   I.E. (A.) APPROX. 15 GPM AT "0" PSIG. REACTOR PRESSURE.
   (B.) APPROX. 6 GPM AT "1000" PSIG. REACTOR PRESSURE.
10. THIS VALUE APPLIES IMMEDIATELY FOLLOWING COMPLETION OF SCRAM. PRESSURE WILL SUBSEQUENTLY EQUALIZE WITH REACTOR PRESSURE.
11. DESIGN PRESSURE AND TEMPERATURE SHOWN IN "TABLE I" IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.
12. ALL VALUES SHOWN IN MODES A, B, C, AND D ARE NOMINAL UNLESS OTHERWISE NOTED.

112D1448AD

Replaces FSAR
Figure 4.6-6C

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE SYSTEM

FIGURE 4.6-6d

NOTES:

1. DEFINITION OF SYMBOLS

/ INDICATES CONDITIONS FOR 0 FLOW RATE.

" INDICATES THE SAME CONDITION AS LISTED UNDER MODE A.

PR INDICATES PRESSURE OF THE REACTOR.

2. MAXIMUM OPERATING TEMPERATURES

THE MAXIMUM SYSTEM OPERATING TEMPERATURES WILL NOT EXCEED 150 DEGREES F. FROM LOCATION 1 THROUGH 30 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAX TEMPERATURE (DEGREES F.)
MODE A -	23	200
MODE C -	23	546
	24	546
	25	280
	27	280
MODE D -	23	200
	24	280
	25	280
	27	280

3. MODE A -
   A. LOCATION 12 - THE RETURN LINE PRESSURE SHALL NOT EXCEED PR+50. PSIG WITH THE CRD COOLING WATER FLOW RATE AT 0.20 GPM/DRIVE. PRESSURE IN EXCESS OF PR+50 PSIG UNDER THE ABOVE CONDITIONS WILL ADVERSELY AFFECT CRD OPERATION.
   B. LOCATION 16 - THE MAXIMUM ACCUMULATOR CHARGING PRESSURE SHALL NOT EXCEED 1510 PSIG. ACCUMULATOR PRESSURE IN EXCESS OF 1510 PSIG WILL CAUSE CRD DAMAGE DURING A SCRAM.
   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR+15 PSIG FOR THE CONDITIONS INDICATED.
   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED, MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.

4. MODE B -
   A. LOCATIONS 13 & 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSED ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 16 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR+250 PSIG FOR THE CONDITIONS INDICATED.

FOR REVISIONS SEE SHEET 1

5. MODE C -
   A. CONDITIONS LISTED FOR MODE C REPRESENT THOSE CONDITIONS WHICH EXIST AT 10 PERCENT OF THE FULL STROKE INSERTION.
   B. THE 546° F. TEMPERATURE LISTED IN NOTE 2 FOR MODE C POSITIONS 23 & 24 SHALL BE USED ONLY IN DETERMINING THE MINIMUM PIPE WALL THICKNESS IN VICINITY OF THE DRIVE HOUSING AND NOT IN DETERMINING STRESSES DUE TO THERMAL EXPANSION. IN DETERMINING MINIMUM WALL THICKNESS IT MAY BE ASSUMED THAT THIS TEMPERATURE OCCURS LESS THAN 1% OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYD. SYS. DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESSES DUE TO THERMAL EXPANSION.
   C. LOCATIONS 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 436 PSI AT 90 GPM FOR ANY CRD.
   D. LOCATION 23 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE REACTOR CAN REACH 10 GPM PER DRIVE.
   E. LOCATIONS 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 29.6 GPM FOR ANY CRD.
   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.
   G. SCRAM VENT VALVE F010 AND DRAIN VALVE F011 CLOSE WITH A SCRAM SIGNAL.

6. MODE D -
   A. LOCATIONS 24 AND 25 - A NEGATIVE FLOW RATE HERE INDICATES A TRANSIENT CONDITION IN WHICH FLOW FROM THE WITHDRAW LINE PASSES THROUGH THE CRD AND INTO THE REACTOR. DURING SCRAM THE DRIVE ACTS AS A PUMP TO CHARGE THE SCRAM DISCHARGE VOLUME TO A PRESSURE ABOVE THAT OF THE REACTOR. IMMEDIATELY FOLLOWING SCRAM, THE WITHDRAW LINE WILL REJECT WATER TO THE VESSEL UNTIL THE LOSS OF THIS WATER REDUCES THE WITHDRAW LINE PRESSURE TO APPROXIMATELY THAT OF THE REACTOR.
   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 100% STROKE IS LESS THAN 65 PSIG.

7. PROCESS DIAGRAM 112D1448 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.

-lete

Delete

8. DESIGN PRESSURE & TEMPERATURE GIVEN BELOW IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.

LOCATION	1A-1B	1B--1	2---6	3A-3B	6---9A	9A-11	7A-7B	7B-7C	11-12A	15A-15B	15B-15C
DESIGN PRESS. (PSIG)	150.	150.	1750.	1750.	1750.	1750.	1750.	1750.	1750.	1750.	1750.
DESIGN TEMP. (DEG F)	150.	150.	150.	150.	150.	150.	150.	150.	150.	150.	150.
ESTIMATED LINE SIZE (INCHES)	4.0	4.0	2.0	1.0	2.0	1.5	1.0	0.75	3.0	1.0	0.75

LOCATION	12A-12	16-16A	17-18	19-20	26-26A	21-22	24-25	27A-27B	27B-27	27-27C	27C-27D
DESIGN PRESS. (PSIG)	**	1750.	1750.	1750.	1750.	1750.	1750.	1250.	1250.	1250.	1250.
DESIGN TEMP. (DEG F.)	**	150.	150.	150.	150.	150.	150.	280.	280.	280.	280.
ESTIMATED LINE SIZE (INCHES)	3.0	2.0	1.0	***	1.0	1.0	0.75	0.75	*	10.0	2.0

*SEE CRD SYSTEM DESIGN SPECIFICATION.

**TO SAME CONDITIONS AS THE FEEDWATER PIPING (BY OTHERS)

***2" HEADER TO EACH OF THE TOTAL QUANTITY OF HCU'S.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE SYSTEM	FIGURE 4.6-6e

Q. 010.045
(4.6.1.1.2.4.1)

The scram discharge volume header piping is sized to receive and contain all water discharged by the control rod drives during a scram, independent of the instrument volume. Show quantitatively how a minimum volume of 3.34 gallons per drive is required since approximately 4 gpm is required to insert the rods with up to an additional 0.34 gpm required for cooling.

Response:

The 3.34 gallon minimum volume requirement is in no way related to the 4 gpm drive insert flow rate or the 0.34 gpm cooling water flow rate. The nominal 4 gpm drive insert flow is the volumetric flow rate delivered to the underside of the control rod drive (CRD) piston to displace the piston in the upward direction and achieve normal rod insertion at about 3 inches per second. The 0.34 gpm cooling water is also delivered to the underside of the drive piston and from here it is discharged to the reactor pressure vessel (RPV) via an engineered flow path up through the annular thermal sleeve region between the CRD piston mechanism and the CRD housing. On the other hand, the scram discharge volume is sized to provide a low pressure discharge point for the volume of water above the drive piston displaced during the period of scram insertion and an additional, conservatively defined, maximum leakage from the RPV to the top of the drive piston during the scram. The volume of water over the drive piston of a fully withdrawn CRD is 0.76 gallons. To this is added a conservative 10 gpm leakage flow from the RPV for an extended period of 10 seconds. (Normal full rod insertion is complete in less than 3 seconds.) Finally, some more volume is added to accommodate the air potentially trapped in the SDV so as to assure that the SDV pressure at 10 seconds after the time of scram initiation is ≤ 65 psig The sum of the 0.76 gallons displaced from the top of the drive piston, the 10 seconds of 10 gpm post-scram leakage flow from the RPV and the free volume required for the air trapped in the SDV adds up to the specified minimum value of 3.34 gallons per CRD.

Q. 010.046
(4.6.1.1.2.4.2.2)

In Figure 4.6-5b and Drawing M528, pressure transmitter (N005) transmits a signal to a pressure switch (N600) in the process instrumentation panel in the control room, which energizes an annunciator in the control room at any time pressure in the charging header falls below the setpoint. Explain why an alarm on high is indicated for the pressure switch (N600) instead of an alarm on low which would provide protection against charging header pressure falling below the setpoint.

Response:

The charging water header of the Control Rod Drive (CRD) is monitored for high pressure since high charging water header pressure indicates the existence of an abnormal condition in the CRD hydraulic system (e.g., such as a failed close flow control valve). The pressure indicating switch on the charging water header (C11-N600) is set to actuate the control room annunciator if the charging water header pressure exceeds a nominal 1510 psig setpoint, (the alarm is actuated on an increasing pressure). Neither sustained high charging water pressure nor CRD drive water pump operation is required to successfully scram the plant. Each of the control rod drives has its own hydraulic control unit (HCU) which operates independently of any others. Scram is achieved on either HCU accumulator pressure or a combination of accumulator pressure and reactor pressure. Each HCU is safety grade and has its own accumulator. The condition of the accumulators is continuously monitored by the Reactor Manual Control System. Loss of pressure and/or leakage from any of the accumulators is detected by PSL-130 and LDS-129, respectively, for each accumulator, as shown in Figure 4.6-5b. Both occurrences are annunciated and a light signal identifies the particular scram accumulator. This instrumentation, existing locally at each HCU, provides the necessary indication of accumulator charge pressure irrespective of the pressure in the nonessential charging water header.

Q. 010.047
(4.6.1.1.2.4.2.4)

Revised FSAR Figure 4.6-5b (Amendment 16) does not appear to show valves F129, F130, F131, and F132. The applicant should explain if this is what is meant in the response by "removing the discrepancy identified above". In addition, the revised FSAR page was not provided as stated in the response. The applicant should provide the revised FSAR page and describe the revision.

Response:

FSAR Figure 4.6-5b (Amendment 16) does not show valves F129, F130, F131, and F132 because these valves were removed from the CRD system when the CRD return line to the RPV was eliminated.

What is meant by "removing the discrepancy identified above" in the response to Question 211.134 (later changed to 010.047) is that by submitting revised FSAR Figure 4.6-5b the discrepancy identified in Question 010.047 between the FSAR text, M528 (FSAR 3.2-4) and FSAR Figure 4.6-5b was resolved.

The words "revised FSAR page change attached" referred to FSAR Figure 4.6-5b.

Q. 010.C48
(4.6.1.1.2.4.3.9)

The text description of the scram accumulator indicates that a check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost. The symbol for valve 111 in Figure 4.6-5b and Drawing M528 appears to be that of a normally open globe valve instead of a stop-check globe valve. Explain this apparent discrepancy.

Response:

There is no disagreement between 4.6.1.1.2.4.3.9 of the FSAR text and Figure 4.6-5b and Drawing M528 concerning valve 111. "A check valve in the accumulator charging line prevents loss of water pressure in the event the supply pressure is lost" refers to valve 115 and to the "charging water". Valve 111 is closed only when the pressure instrumentation is being serviced and when the nitrogen charging station is being connected and disconnected.*

*Draft FSAR page change attached.

scram valve operators. This prevents the inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

4.6.1.1.2.4.3.7 Scram Inlet Valve

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the control room as soon as the valve starts to open.

4.6.1.1.2.4.3.8 Scram Exhaust Valve

The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of the higher air pressure spring setting in the valve operator.

4.6.1.1.2.4.3.9 Scram Accumulator

The scram accumulator stores sufficient energy to fully insert a control rod at lower vessel pressures. At higher vessel pressures the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging water line prevents loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the control room.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

010.049
Q. ~~211.006~~
(5.2.5)
(7.6.2)
(12.3.4)

Provide a detailed discussion of the sensitivity and response times of the containment airborne radiation monitoring systems for a number of containment background activity levels. The background activity levels which should be considered are those levels in the containment that would result from leakage through the RCPB assuming: (1) relatively clean water in the reactor coolant system at the initial operation of the WNP-2 facility at power; and (2) the maximum level of activity in the reactor coolant permitted by the WNP-2 Technical Specifications. In responding to this item, assume both the normal and the maximum leakage rates identified in your response to Question 211.115. Indicate your assumptions in estimating the response times of the containment airborne radiation monitoring systems (e.g., the preset alarm level for higher background leakage and the plateout factor).

Response:

The two types of radiation monitors used in the WPPSS Nuclear Project No. 2, for monitoring the drywell atmosphere, are the particulate monitor and the noble gas monitor. The sensitivity of these two types of monitors is given in Figure 010.049 ~~211.006~~-1 and ~~211.006~~-2. The same detector is used in both monitors. The detector's noise level is about 25 cpm. The minimum detectable concentration is based on doubling the background count rate. The count ratemeter range and the minimum sensitivity of both types of monitors are:

Noble Gas

Count Ratemeter Range:

1.4×10^{-7} μci/cc - 1.4×10^{-1} μci/cc for Kr-85

Minimum Detectable Concentration:

3.6×10^{-7} μci/cc for Kr-85

Particulate

Ratemeter Range:

2.9×10^{-12} μci/cc - 2.9×10^{-6} μci/cc for Sr/Y-90

Minimum Detectable Concentration:

7.4×10^{-12} μci/cc for Sr/Y-90

The quantity of drywell atmosphere that flows through the filter of the particulate monitor and then through the 2.2 liters chamber of the noble gas monitor is 3 cfm. The drywell atmosphere sample is returned to the drywell. (The free volume of the drywell is 200,540 ft^3). There is a charcoal filter after the particulate and before the noble gas detector. The filter efficiency of the particulate filter is assumed to be 100 percent. Similarly, the efficiency of the charcoal filter is assumed to be 100 percent for all halogens.

The 2-inch diameter scintillation crystal, of the particulate detector, is 1/4-inch away from the face of the filter tape. The filter tape is 2.5 inches wide and moves at 1"/hr.

As per the response to Question 211.005, the total minimum identifiable and unidentifiable leakage rate is taken to be 2.1 gpm. The total maximum identifiable and unidentifiable leakage rate is taken to be 5.5 gpm. The leakage is measured at drywell environmental conditions and thus the water density is assumed to be 1g/cc. Based upon potential sources of leakage (such as number and nominal size of valves), it is estimated that 38.5 percent of the leakage is due to steam leakage. Collection of the identifiable leakage is not seal-tight and thus, volatile radioisotopes can escape into the drywell atmosphere. Water leakage is assumed to be flashing, and that there is an instantaneous mixing of the volatile radioisotopes with the drywell atmosphere. It is assumed that the noble gases are in the steam phase and the small quantities of noble gases in the liquid phase are neglected. On the other hand, it is assumed that the quantities of particulate radioisotopes in the steam phase are small and thus, are neglected.

Decay was considered for the radioisotopes in the drywell and for those radioisotopes accumulated on the particulate filter. No decay was considered, however, for the radioisotopes while in transit to the detectors and while in the noble gas detector chamber (for the atmosphere in the chamber, exchange rate is 1.55 seconds). It is assumed that plating, settling,

impingement, etc., reduce the specific concentration of particulate isotopes in the drywell atmosphere by a factor of 1000 before the air flow reaches the filter of the particulate monitor.

In order to be responsive to the question, the source concentrations used in this analysis were taken to be those in Table 5 of ANS/ANSI N237-1976, reduced by 1/100, as representative of "relative clean water in the reactor coolant". The design basis concentrations, as per General Electric specification document No. 22A2703F, Revision 3, were used as representative of the maximum expected level of radioactivity within the reactor coolant.

The criterion used, as an indication of leakage increase, is the doubling of the background count rate within one hour for 1 gpm (additional) leakage. Each detector was evaluated with respect to responding to the criterion. See Table 2~~11.006~~-1 010.049 for the results of the analysis.

In summary, the particulate monitor will meet the requirements stated in Regulatory Guide 1.45 for the minimum activity concentration of radioisotopes in the reactor coolant. For the cases where it is assumed that the design basis activity exists in the reactor coolant, the background activity exceeds the particulate monitor range. However, the detector can be desensitized accordingly. It can be shown that if a reactor coolant activity is selected based upon the guidance contained in Regulatory Guide 1.45; i.e., if "a realistic primary coolant radioactivity concentration" is used, e.g., equal to that given in Table 5 in ANS/ANSI N237-1976, and expanding the criterion of double background count rate to the desensitized monitor, the requirements stated in the regulatory guide will be met.

The noble gas monitor, however, even though its sensitivity is consistent with Regulatory Guide 1.45 requirements, would not be capable of detecting the additional 1 gpm leakage within one hour utilizing the criterion of double background count as positive indication. The noble gas monitor does, however, provide the most reliable and fastest means of ascertaining increased activity within containment with unidentified leakages higher than 1 gpm.

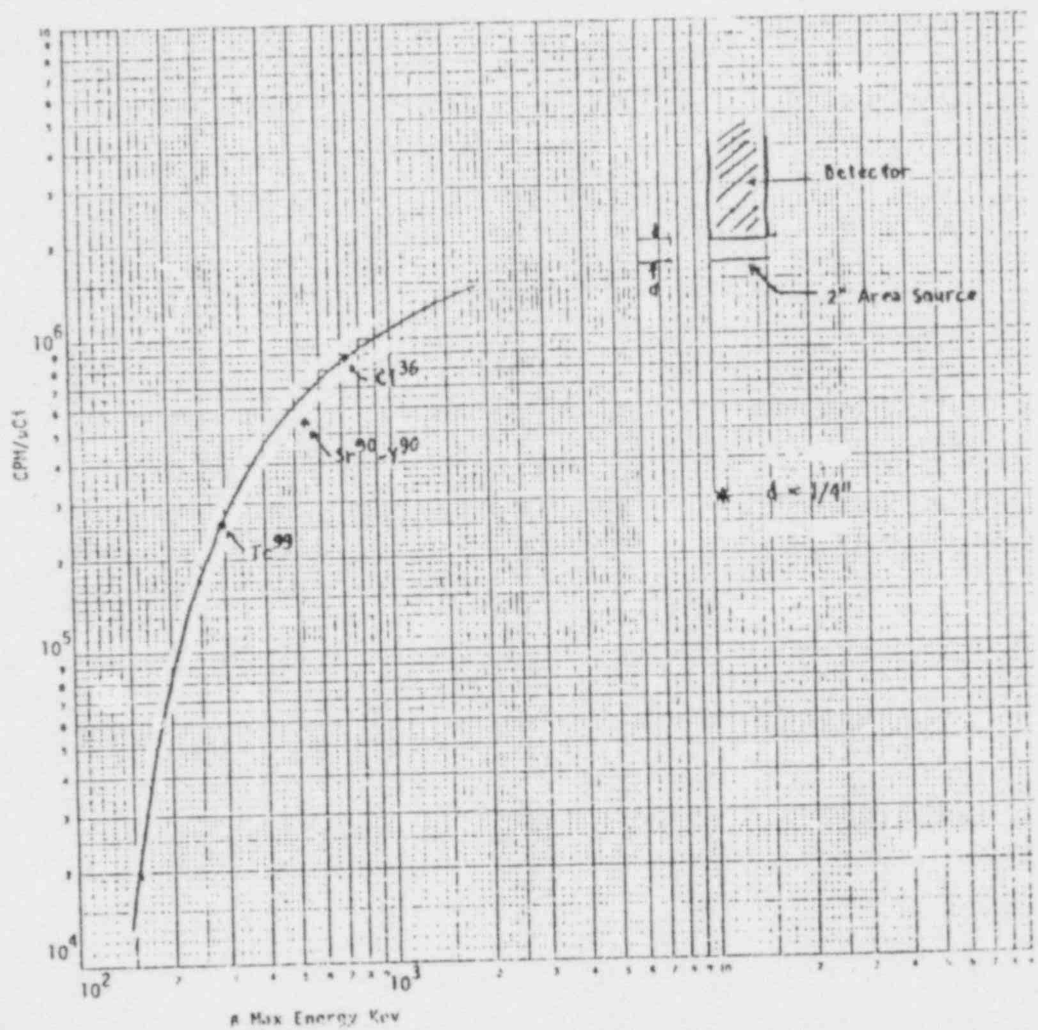
010.049
TABLE ~~211.006~~-1

RESULTS OF MONITOR ANALYSIS

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 5 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector before the event is about 30. With the increase of 1 gpm leakage, the count rate would increase one hour after the event to only about 32 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 46 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 12 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector before the event is about 37. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only 39 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 49 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 81 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 106. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 136 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 51 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 213 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 238. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 269 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 53 minutes for the background count rate to double.

TABLE ~~211.006-1~~ 010.049 (Continued)

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 10 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 35. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 36 cpm.	The background count rate will double as a result of 1 gpm unidentified leakage, within abut 51 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 25 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 50. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 52 cpm.	The background count rate would double as a result of the 1 gpm unidentified leakage, within about 53 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duraton of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 165 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 190. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 219 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 57 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 431 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 456. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 485 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 58 minutes for the background count rate to double.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PARTICULATE SYSTEM EFFICIENCY

FIGU
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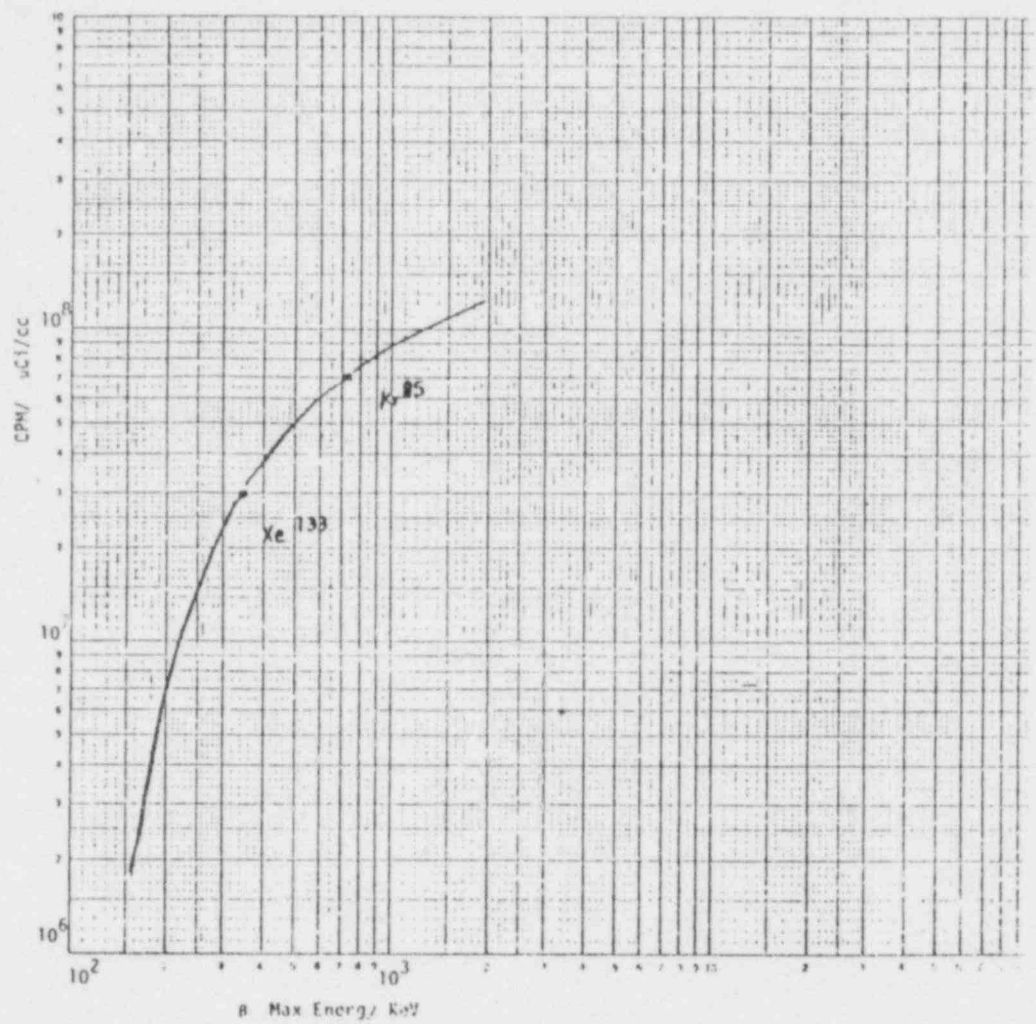
010.0



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GASEOUS EFFLUENT SYSTEM EFFICIENCY	FIGURE 211. 006-2

010.05

Q. 010.050
(5.2.5)

The response to Question 211.007 requires additional information. It is unclear how the comparison will be made between the radioactivity monitoring and the sump level monitoring.

Describe briefly the mechanics of making these data comparisons. What calibration and operability verification tests will be performed for each independent leakage detection system? Which leakage detection system is to be used as the reference for comparison with the other systems? Do the radiation monitoring systems have radioactive sources (check sources) built into the systems?

Response:

Radiation monitors are useful as leak detection devices because of their sensitivity and rapid response to leaks. After several weeks of full power operation, a set level of background radiation will be established. Any sudden or unexplained increase in background radiation would indicate a possible primary coolant leak within the primary containment. If an increase is noted, a comparison with other leak detection devices having a relationship to each other will be made, particularly the equipment and floor drain flow rate monitors, and the reactor building sump pumps activation on high sump level. Using the flow rate monitors as a reference, the comparisons provide independent indications of a leak within the primary containment.

The radiation monitoring channels are redundant, allowing cross-checks between channels. Each channel is equipped with test sources and purge systems so that proper operation of each individual channel can be verified from the detector through to the indicator.

A discussion of calibration and operability verification tests for the leak detection system is discussed in 7.6.2.4.a.

Q. 010.051
(5.2.5)

Identified leakage is determined during preoperational testing or is measurable during reactor operation. Will the identified leakage be measured regularly and recorded? If so, provide the frequency that these data will be recorded and indicate what procedural guidelines are to be used to change the magnitude of the base identified leakage rate?

Response:

The identified leakage is measured continuously and the leakage rate will be calculated and recorded on a frequency of at least once per day in accordance with the plant technical specifications. The procedures describing how the identified leakage rate is determined will include provisions for showing the identified leakage rate numbers not exceeded the maximum allowable value of 25 gpm.

Each equipment leak-off connection has been provided with a temperature element which will identify to the operator that a higher than normal temperature exists at that particular location.

Q. 010.052
(5.2.5)

Standard Review Plan 5.2.5 specifies that unidentified leakage should be collected separately from the identified leakage so that a small unacceptable unidentified leak is not masked by larger acceptable identified leakage. Section 5.2.5 of the FSAR does not clearly indicate that separate collection of identified and unidentified leakage is provided.

Provide assurances that identified and unidentified leakage will be collected separately. If separate collection is not to be provided, provide justification for use of a common collection reservoir and show that a small unidentified leak of about 1 gpm would be recognized within one hour.

Response:

Identified and unidentified leakage are collected separately. Identified leakage is collected, monitored, and indicated by the Equipment Drain System (see FSAR Figure 3.2-9) while the unidentified leakage is collected, monitored, and indicated by the Floor Drain System (see FSAR Figure 3.2-10). Section 5.2.5.6 refers, in part, to 7.6.1.3 for further explanation. In subsections 7.6.1.3.4, 7.6.1.3.5 and 7.6.1.3.6 of this section, the two separate collection systems are described.

Q. 010.053
(5.2.5)

Provide a list of all indications available to the control room operator for evaluating and detecting unidentified leakage. Show haw the operator will determine the amount of leakage by observing the indications that are available to him, including the need for unit conversion (count rate [illegible] gpm, etc). If the monitoring is computerized, discuss the backup procedures available should the computer become inoperative.

Response:

The following indications are available to the Control Room Operator for evaluating and detecting unidentified leakage:

1.	Drywell Pressure Recorders	-3 to +3 psig 0 to 25 psig 0 to 180 psig
2.	Drywell Temperature Recorders	50° - 400° F
3.	Drywell Floor Drain Total Flow Recorder	0 - 30 GPM
4.	Reactor Building Floor Drain Sump Fillup Rate Timer	0 - 30 Min.
5.	Drywell Cooler Cooling Water Differential Temperature Recorder	0 - 150° F
6.	Reactor Vessel Water Level Recorders	-353.2" to - 153.2" -185" to +25" -35 to +85"
7.	Drywell Atmosphere Radiation Monitors	1 - 10^6 CPS

The indications listed above have no definitive correlation between their engineering units. For example, a specific count rate indicated by the Drywell Atmosphere Radiation Monitors cannot be directly converted to a leak rate in GPM, nor can a high drywell temperature be converted to an equivalent GPM leak rate. The indications listed are provided as an early warning to the operator os a potential leak. The actual unidentified leak rate is determined by observing the drywell floor drain system flow rate on flow recorders provided in the Control Room. The monitoring is not computerized.

Q. 010.056

Your response to question 010.021 is completely inadequate. your design should be modified to provide one of the following alternatives:

1. A Seismic Category I, Quality Group C, tornado missile protected Spent Fuel Pool Cooling System including the Secondary Fuel Pool Heat Exchanger Cooling System.

2. A Seismic Category I, Quality Group C, tornado misprotected, make-up water supply to the Spent Fuel Pool and HVAC (the HVAC design environment should be 212°F and 100% humidity). The structure above the refueling floor should be seismic Category I and tornado missile protected.

3. A Seismic Category I, Quality Group C, make-up water supply to the Spent Fuel Pool and the results of an analysis which verifies that with the loss of the structure above the refueling floor, cooling with only the Seismic Category I make-up, and the most unfavorable atmospheric diffusion conditions (X/Q) that the site boundary does will not exceed 25% of the limits specified in Title 10, Code of Federal Regulations, Part 100.

Response:

A Seismic Category I, Quality Group C, tornado missile protected Spent Fuel Pool Cooling System has been provided which satisfies Alternative I. See revised Section 9.1.3.*

The valves piping, and components of the cooling portion of the Fuel Pool Cooling and Cleanup System are located within the Reactor Building and are protected form tornado generated missiles. The cooling portion of the system is Seismic Category I, and will automatically isolate from the non-seismic clean-up portion of the system on low fuel pool water level. Remote-manual startup from the main control room of redundant active components of the cooling portion of the system is provided. Safety grade cooling water to the heat exchangers and fuel pool makeup water is available from the standby service water (SW) system and can be initiated by remote-manual operation from the main control room. The active components of the cooling portion of the system are powered from Class IE sources.

The response to to question 010.021 has been revised to refer to this response.*

*FSAR draft page change attached.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The fuel pool cooling and cleanup system has been designed to comply with the objectives set forth in Regulatory Guides 1.13, Revision 1 and 1.26 Revision 3 to the extent specified in the following subsections. The system and equipment are designed to the classifications given in Tables 3.2-1 and 9.1-2.

During normal reactor operation

the system is designed to remove the decay heat released from the spent fuel elements and maintain a specified fuel pool water temperature, water clarity, and water level by accomplishing the following:

a. Minimizing corrosion product buildup and controlling fuel pool water clarity so that fuel assemblies can be efficiently handled under water.

b. Minimizing fission product concentration in the fuel pool water thereby minimizing the release of fission products from the pool to the reactor building environment.

c. Monitoring surge tank water level to thereby maintain a pool water level above the fuel sufficient to provide shielding for normal building occupancy and to control make-up flow rate from the condensate transfer system.

d. Maintaining the fuel pool water temperature below 125°F under normal operating conditions. The maximum heat load in the fuel pool under normal operating conditions occurs at the end of the 12th refueling cycle at which time there are 2068 fuel assemblies in the high density fuel racks. The estimated refueling data is given in Table 9.1-3.

Insert

9.1.3.2 System Description

9.1.3.2.1 Normal Operation

The fuel pool cooling and cleanup system flow diagram is shown on Figure 9.1-4. System performance data are summarized in Table 9.1-1. Major components of the system are summarized in Table 9.1-2. The system is designed to dissipate the fuel pool heat load during equilibrium or non-equilibrim fuel cycle conditions.

Page 9.1-23 insert:

Following any seismic event or major plant disturbance, i.e., during abnormal operation, the system is designed to prevent fuel pool boiling and maintain adequate water level in the spent fuel pool by means of the following:

a. Automatic isolation ~~on~~ low fuel pool water level of the Seismic ~~cooling~~ portion of the system from the non-seismic, cleanup portion of the system.

b. Remote-manual startup from the control room of redundant, active components of the fuel pool cooling portion of the system and initiation of safety grade cooling water, i.e., standby service water (SW), to the fuel pool cooling heat exchangers.

c. Remote-manual, redundant SW system make-up to the fuel pool and fuel pool level monitoring from the control room.

If required, heat removal capacity is available for the full core removal load during either of these periods, in addition to the spent fuel load already stored. The system design heat load is based on the data given in Table 9.1-3.

The system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the reactor building closed cooling water system. Water purity and clarity in the storage pool, reactor well, and dryer-separator pit are maintained by filtering and demineralizing the pool water through a filter demineralizer. In addition to fuel pool water demineralization, the system will be used on occasion to demineralize suppression pool water.

The pool cooling and cleanup system consists of two 50% capacity circulating pumps, two 50% capacity heat exchangers, two 100% filter demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through scuppers and skimmer weirs to the surge tanks. Make-up water for the system is transferred from the condensate storage tank to a skimmer surge tank to make up evaporative losses. The fuel pool pumps and heat exchangers are located in the reactor building
an enclosed room on the 548 foot level of
beneath the fuel pool.

Fuel pool water is continually recirculated except when draining the reactor well and dryer-separator pit. The operating temperature of 125°F is permitted to rise to 150°F when the circulating flow is interrupted for draining the reactor well and dryer-separator pit, or when larger than normal batches of fuel are stored. The fuel pool cooling and cleanup system is interconnected with the residual heat removal system to supplement the pool cooling during refueling in the event that a larger than normal batch of fuel is stored.

To establish a circulating pattern of flow in the reactor well and storage pool, the diffusers and skimmer drains are placed to sweep particles dislodged during refueling operations away from the work area and out of the pool.

Fuel pool water clarity and purity is maintained by a combination of filtering and ion exchange. The filter demineralizer maintains a total heavy element content (Fe, Cu, Hg, Ni, etc.) of 0.1 ppm or less with a pH range of 6.0 to 7.5.

Particulate material is removed from the water by the pressure precoat filter demineralizer units. The finely divided disposable filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted. The spent filter medium is backwashed to the waste sludge phase separator tank for processing in the solid radwaste handling system. New filter medium is mixed in a precoat tank and is transferred as a slurry by a precoat pump where the solids deposit on the filter elements. The holding pump connected to each filter demineralizer maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation. A strainer is provided in the effluent stream of the filter demineralizers to limit the migration of the filter material.

The two filter demineralizer units are located separately in shielded cells in the radwaste building. Sufficient clearance is provided in the cells to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and its associated piping. All valves are located on the outside of one shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls.

Instrumentation is provided for both automatic and remote manual operation. Indication is provided in the control room and pump room. Surge tank high and low water level switches are provided. A local level indicator is provided to monitor reactor well water level. Control of flow to or from the reactor well can be accomplished during refueling. A fuel pool high/low water level switch operates a local indicator light and sounds an alarm in the control room whenever the level is either too high or too low. The trip point is adjustable over the range of the skimmer weir adjustment.

The pumps are controlled from ~~either~~ the pump room, the control room, or the vicinity of the fuel pool filters. ~~Pump low suction pressure automatically turns off the pumps~~. A pump low discharge pressure alarm annunciates in the control room and in the pump room. The controls for the remote controlled fuel pool discharge valves are located on a rack in the pump room and in the control room. The open or closed condition of each of these valves is indicated by a light in the pump room and in the control room.

The flow rate through the filter demineralizers is indicated by a flow indicator on the pump room panel.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the operating floor instrument racks and is alarmed in the control room.

The filter demineralizers are controlled from a local panel. Differential pressure and conductivity instrumentation is provided for each unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicators are provided to monitor the condition of the filter demineralizers.

* insert ①

9.1.3.3 Safety Evaluation

The maximum possible heat load will be the decay heat of one full core load of the fuel due to an emergency dump into the pool plus the remaining decay heat of previously discharged batches of fuel. The residual heat removal system (RHR) can be operated in parallel with the fuel pool cooling and cleanup system during this condition when the pool has a greater than normal load and when its temperature exceeds 125°F. The RHR system can be used in parallel with the fuel pool cooling system to remove abnormal heat loads, as well as during the normal refueling mode. The RHR system will not be initiated unless the reactor is in a cold shutdown condition. The operator must insert spool pieces in supply and discharge piping and open normally closed valves to permit the use of this system for supplementary cooling.

The fuel pool heat exchangers are normally cooled by the reactor building closed cooling water system to prevent contamination outside the reactor building in the event of a fuel pool heat exchanger tube failure. The system can maintain the fuel pool water temperature below 125°F when removing the nominal heat load from the pool with the reactor building closed cooling water temperature at its maximum of 95°F. The fuel pool water temperature is permitted to rise to approximately 150°F while the system water flow is diverted from the pool to drain the reactor well and dryer-separator pit, or when larger than normal batches of spent fuel are stored in the pool.

* insert ②

There are no connections to the fuel storage pool which could allow the fuel pool to be drained below the pool gate between the reactor well and the fuel pool. Two diffusers are placed in both the reactor well and the fuel pool to distribute the return water as efficiently and with as little turbulance as possible. Diffusers are placed to minimize stratification of

Page 9.1-26 inserts:

(1) 9.1.3.2.2 Abnormal Operation

The portion of the FPC system which is required for cooling the fuel pool is located within the reactor building and is designed to Seismic Category I criteria. The portion of the FPC system which is used for fuel pool cleanup is located within the radwaste building and is isolable from the reactor building by means of two Seismic Category I isolation valves per line located within the reactor building. The isolation valves are either check valves or motor operated gate valves. The motor operated valves close automatically on a fuel pool low water level condition.

The redundant, active components required for fuel pool cooling are powered from division 1 and 2 power sources. Following a loss of off-site power, these components can be energized from on-site emergency power. On a loss of reactor building closed cooling (RCC) to the fuel pool heat exchangers, the RCC lines to the heat exchangers can be isolated by redundant, Seismic Category I, motor operated gate valves powered from separate Class 1E power supplies. Standby service water (SW) can be supplied to the heat exchangers through motor operated valves which are normally key locked closed. Radiation detectors are located on the SW return lines. Operation and monitoring of fuel pool cooling portion of the FPC system can be done entirely from the control room.

All components required for fuel pool cooling are qualified to the reactor building accident environment or else, they are located in enclosed rooms meeting the criteria set in 3.11.4.2. The fuel pool equipment room located on the 548 foot location is provided with both division 1 and 2 reactor building emergency cooling system.

(2) During abnormal operation the SW system is available to cool the fuel pool heat exchangers preventing any boiling of the fuel pool. The SW pressure is higher than the fuel pool pressure; thus, any leakage will be into the fuel pool system. In addition, radiation monitors on the SW return line will detect any gross tube or tube sheet failure.

either temperature or contamination. A check valve is connected to each pipe outside the pool to prevent the pool water from being siphoned out of the pool and uncovering the spent fuel. Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.

* Insert I

A make-up water valve controlled by tank level switches supplies condensate from the condensate transfer system to the pool to replace evaporative and leakage losses. The backup source of make-up water is from the Seismic Category I, Safety Class 3 standby service water system. This connection supplies makeup for long-term evaporative losses. ~~enough water to prevent the uncovering of the spent fuel. By use of the standby service water as make-up, the fuel pool will be cooled by evaporation of the pool water.~~

add * notes as shown on attachment II

~~Each filter demineralizer is capable of continual performance~~ during normal operations, operating ~~at a fuel pool water flow rate of 100% of rated flow~~ at a maximum fuel pool water flow rate of 1000 gpm, and will ~~maintain water conditions as specified in 9.1.3.2.~~

~~The following components of the fuel pool cooling and cleanup system (FPC) are designed to ASME Section III, Class 3: fuel pool cooling pumps, filter demineralizers, pumps, valves, and piping, FPC piping, and fuel pool heat exchanger. The system heat exchangers are also designed to the standards of the Tubular Exchangers Manufacturers Association, Class R. Piping in the reactor building is controlled and supported to Seismic Category I requirements. The water lines between the fuel pool and RHR systems are designed to ASME Section III, Class 3, Seismic Category I requirements . The FPC pumps are not designed to Seismic Category I requirements. Condensate piping in the reactor building is controlled and supported to Seismic Category I requirements.~~

A radiological evaluation of the cleanup system is presented in Chapter 12.

From the foregoing analysis, it is concluded that the fuel pool cooling and cleanup system meets its design basis and satisfies the requirements of Regulatory Guide 1.13, Revision 1 with exceptions as noted in this section.

Page 9.1-27 insert I:

All piping connecting to the fuel pool, reactor well, and dryer separator pool and their respective liner drains are Seismic Category I up to and including either the normally closed, manually operated drain valve or the normally open, redundant isolation valves which can isolate the non-seismic portion of the system. Since the fuel pool system is at low temperature and pressure (moderate energy system) postulated breaks in the Seismic Category I portion are limited to cracks.

Fuel pool cooling can be established and monitored from the control room following a design basis LOCA. Entry to the reactor building is not required. One of the two redundant trains is adequate to prevent fuel pool boiling by a large margin. Due to the large thermal capacity of the fuel pool, sufficient operator time is available after a LOCA or any other event for the operator to take action.

Attachment II for page 9.1-27.

Each filter demineralizer is capable of continual performance at a normal fuel pool water flow rate of 575 gpm, or a maximum fuel pool water flow rate of 1000 gpm, and will maintain water conditions as specified in 9.1.3.2.

9.1.3.4 Testing and Inspection Requirements

No special tests except as noted below are required because at least one pump, heat exchanger, and filter demineralizer are continuously in operation while fuel is stored in the pool. Duplicate components are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

SW flow to the fuel pool heat exchangers and operability of the valves which intertie the SW and RCC systems are tested in conjunction with testing of the SW system.

TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM EQUIPMENT DATA

Fuel Pool Heat Exchangers

Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/heat exchanger	4.0×10^6
Cooling Water Flow, gpm/heat exchanger	575
Code and Standards	ASME/III-Class 3 and TEMA-Class R
Seismic Category	~~II~~ I

Fuel Pool Circulation Pumps

Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	575
Head, Ft of H_2O	160
Motor Size hp	40
Seismic Category	~~II~~ I
Code	ASME/III-Class 3

Fuel Pool Filter Demineralizer

Number	2
Design Flow Rate, gpm	1000
Design Pressure, psig	150
Design Temperature, °F	150
Material	CS-Plastic Lined
Code	ASME/III-Class 3
Seismic ~~Class~~ Category	II

Piping and Valves

Design pressure, psig	150/300
Design Temperature, °F	220
Material	CS
Code	ASME/III-Class 3
Seismic Category	
Fuel pool cooling portion:	I
Cleanup portion:	II

Q. 010.021
(9.1.3)

Provide a cooling system and a source of makeup water for the spent fuel pool which are both designed to Seismic Category I criteria in accordance with the staff positions contained in Regulatory Guide 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis," December 1975.

Response:

~~WNP-2 has a Seismic Category I source of makeup water for the spent fuel pool from the Seismic Category I standby service water service system. This is shown on Figure 9.1-4 and stated in 9.1.3.3. Cooling under emergency conditions for the fuel pool is supplied by evaporation of pool water. Regulatory Guide 1.13, Rev. 1, makes no specific statements about requiring a Seismic Category I spent fuel pool cooling system. As a result, WNP-2 meets the applicable criteria of the Regulatory Guide and the intent of the question. However, further evaluation of the design in this area is ongoing due to the interaction of fuel pool cooling and post-LOCA secondary containment pressure-temperature response. (See the response to Question 312.018).~~

See the response to question 10.56

Q. 010.057
(9.1.3)

Verify that your use of the phrase "... controlled and supported to Seismic Category I requirements" means that it meets all requirements for Seismic Category I qualification.

Response:

The cooling portion of the Fuel Pool Cooling and Cleanup System, including valves, piping, and components, meets all Seismic Category I requirements. See revised Section 9.1.3.* Non-Seismic Category I piping systems in the Reactor Building are nevertheless supported to the same Seismic Category I requirements. (See Notes 10 and 32 to Table 3.2-1.)

*Draft FSAR page change attached.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The fuel pool cooling and cleanup system has been designed to comply with the objectives set forth in Regulatory Guides 1.13, Revision 1 and 1.26 Revision 3 to the extent specified in the following subsections. The system and equipment are designed to the classifications given in Tables 3.2-1 and 9.1-2.

During normal reactor operation the system is designed to remove the decay heat released from the spent fuel elements and maintain a specified fuel pool water temperature, water clarity, and water level by accomplishing the following:

a. Minimizing corrosion product buildup and controlling fuel pool water clarity so that fuel assemblies can be efficiently handled under water.

b. Minimizing fission product concentration in the fuel pool water thereby minimizing the release of fission products from the pool to the reactor building environment.

c. Monitoring surge tank water level to thereby maintain a pool water level above the fuel sufficient to provide shielding for normal building occupancy and to control make-up flow rate from the condensate transfer system.

d. Maintaining the fuel pool water temperature below 125°F under normal operating conditions. The maximum heat load in the fuel pool under normal operating conditions occurs at the end of the 12th refueling cycle at which time there are 2068 fuel assemblies in the high density fuel racks. The estimated refueling data is given in Table 9.1-3.

Insert

9.1.3.2 System Description

9.1.3.2.1 Normal Operation

The fuel pool cooling and cleanup system flow diagram is shown on Figure 9.1-4. System performance data are summarized in Table 9.1-1. Major components of the system are summarized in Table 9.1-2. The system is designed to dissipate the fuel pool heat load during equilibrium or non-equilibrim fuel cycle conditions.

Page 9.1-23 insert:

Following any seismic event or major plant disturbance, i.e., during abnormal operation, the system is designed to prevent fuel pool boiling and maintain adequate water level in the spent fuel pool by means of the following:

a. Automatic isolation on low fuel pool water level of the seismic, cooling portion of the system from the non-seismic, cleanup portion of the system.

b. Remote-manual startup from the control room of redundant, active components of the fuel pool cooling portion of the system and initiation of safety grade cooling water, i.e., standby service water (SW), to the fuel pool cooling heat exchangers.

c. Remote-manual, redundant SW system make-up to the fuel pool and fuel pool level monitoring from the control room.

If required, heat removal capacity is available for the full core removal load during either of these periods, in addition to the spent fuel load already stored. The system design heat load is based on the data given in Table 9.1-3.

The system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the reactor building closed cooling water system. Water purity and clarity in the storage pool, reactor well, and dryer-separator pit are maintained by filtering and demineralizing the pool water through a filter demineralizer. In addition to fuel pool water demineralization, the system will be used on occasion to demineralize suppression pool water.

The pool cooling and cleanup system consists of two 50% capacity circulating pumps, two 50% capacity heat exchangers, two 100% filter demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through scuppers and skimmer weirs to the surge tanks. Make-up water for the system is transferred from the condensate storage tank to a skimmer surge tank to make up evaporative losses. The fuel pool pumps and heat exchangers are located in ^ the reactor building beneath the fuel pool.
an enclosed room on the 548 foot level of

Fuel pool water is continually recirculated except when draining the reactor well and dryer-separator pit. The operating temperature of 125°F is permitted to rise to 150°F when the circulating flow is interrupted for draining the reactor well and dryer-separator pit, or when larger than normal batches of fuel are stored. The fuel pool cooling and cleanup system is interconnected with the residual heat removal system to supplement the pool cooling during refueling in the event that a larger than normal batch of fuel is stored.

To establish a circulating pattern of flow in the reactor well and storage pool, the diffusers and skimmer drains are placed to sweep particles dislodged during refueling operations away from the work area and out of the pool.

Fuel pool water clarity and purity is maintained by a combination of filtering and ion exchange. The filter demineralizer maintains a total heavy element content (Fe, Cu, Hg, Ni, etc.) of 0.1 ppm or less with a pH range of 6.0 to 7.5.

Particulate material is removed from the water by the pressure precoat filter demineralizer units. The finely divided disposable filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted. The spent filter medium is backwashed to the waste sludge phase separator tank for processing in the solid radwaste handling system. New filter medium is mixed in a precoat tank and is transferred as a slurry by a precoat pump where the solids deposit on the filter elements. The holding pump connected to each filter demineralizer maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation. A strainer is provided in the effluent stream of the filter demineralizers to limit the migration of the filter material.

The two filter demineralizer units are located separately in shielded cells in the radwaste building. Sufficient clearance is provided in the cells to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and its associated piping. All valves are located on the outside of one shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls.

Instrumentation is provided for both automatic and remote manual operation. Indication is provided in the control room and pump room. Surge tank high and low water level switches are provided. A local level indicator is provided to monitor reactor well water level. Control of flow to or from the reactor well can be accomplished during refueling. A fuel pool high/low water level switch operates a local indicator light and sounds an alarm in the control room whenever the level is either too high or too low. The trip point is adjustable over the range of the skimmer weir adjustment.

The pumps are controlled from ~~either~~ the pump room, the control room, or the vicinity of the fuel pool filters. ~~Pump low suction pressure automatically turns off the pumps.~~ A pump low discharge pressure alarm annunciates in the control room and in the pump room. The controls for the remote controlled fuel pool discharge valves are located on a rack in the pump room and in the control room. The open or closed condition of each of these valves is indicated by a light in the pump room and in the control room.

The flow rate through the filter demineralizers is indicated by a flow indicator on the pump room panel.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the operating floor instrument racks and is alarmed in the control room.

The filter demineralizers are controlled from a local panel. Differential pressure and conductivity instrumentation is provided for each unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicators are provided to moni-- the condition of the filter demineralizers.

* insert ①

9.1.3.3 Safety Evaluation

The maximum possible heat load will be the decay heat of one full core load of the fuel due to an emergency dump into the pool plus the remaining decay heat of previously discharged batches of fuel. The residual heat removal system (RHR) can be operated in parallel with the fuel pool cooling and cleanup system during this condition when the pool has a greater than normal load and when its temperature exceeds 125°F. The RHR system can be used in parallel with the fuel pool cooling system to remove abnormal heat loads, as well as during the normal refueling mode. The RHR system will not be initiated unless the reactor is in a cold shutdown condition. The operator must insert spool pieces in supply and discharge piping and open normally closed valves to permit the use of this system for supplementary cooling.

The fuel pool heat exchangers are normally cooled by the reactor building closed cooling water system to prevent contamination outside the reactor building in the event of a fuel pool heat exchanger tube failure. The system can maintain the fuel pool water temperature below 125°F when removing the nominal heat load from the pool with the reactor building closed cooling water temperature at its maximum of 95°F. The fuel pool water temperature is permitted to rise to approximately 150°F while the system water flow is diverted from the pool to drain the reactor well and dryer-separator pit, or when larger than normal batches of spent fuel are stored in the pool.

* insert ②

There are no connections to the fuel storage pool which could allow the fuel pool to be drained below the pool gate between the reactor well and the fuel pool. Two diffusers are placed in both the reactor well and the fuel pool to distribute the return water as efficiently and with as little turbulance as possible. Diffusers are placed to minimize stratification of

Page 9.1-26 inserts:

(1) 9.1.3.2.2 Abnormal Operation

The portion of the FPC system which is required for cooling the fuel pool is located within the reactor building and is designed to Seismic Category I criteria. The portion of the FPC system which is used for fuel pool cleanup is located within the radwaste building and is isolable from the reactor building by means of two Seismic Category I isolation valves per line located within the reactor building. The isolation valves are either check valves or motor operated gate valves. The motor operated valves close automatically on a fuel pool low water level condition.

The redundant, active components required for fuel pool cooling are powered from division 1 and 2 power sources. Following a loss of off-site power, these components can be energized from on-site emergency power. On a loss of reactor building closed cooling (RCC) to the fuel pool heat exchangers, the RCC lines to the heat exchangers can be isolated by redundant, Seismic Category I, motor operated gate valves powered from separate Class 1E power supplies. Standby service water (SW) can be supplied to the heat exchangers through motor operated valves which are normally key locked closed. Radiation detectors are located on the SW return lines. Operation and monitoring of fuel pool cooling portion of the FPC system can be done entirely from the control room.

All components required for fuel pool cooling are qualified to the reactor building accident environment or else, they are located in enclosed rooms meeting the criteria set in 3.11.4.2. The fuel pool equipment room located on the 548 foot location is provided with both division 1 and 2 reactor building emergency cooling system.

(2) During abnormal operation the SW system is available to cool the fuel pool heat exchangers preventing any boiling of the fuel pool. The SW pressure is higher than the fuel pool pressure; thus, any leakage will be into the fuel pool system. In addition, radiation monitors on the SW return line will detect any gross tube or tube sheet failure.

either temperature or contamination. A check valve is connected to each pipe outside the pool to prevent the pool water from being siphoned out of the pool and uncovering the spent fuel. Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.

* Insert I

A make-up water valve controlled by tank level switches supplies condensate from the condensate transfer system to the pool to replace evaporative and leakage losses. The backup source of make-up water is from the Seismic Category I, Safety Class 3 standby service water system. This connection supplies makeup for long-term evaporative losses. ~~enough water to prevent the uncovering of the spent fuel. By use of the standby service water as make-up, the fuel pool will be cooled by evaporation of the pool water.~~

add* notes as shown on attachment II

~~Each filter demineralizer is capable of continual performance, at a fuel pool water flow rate of 100% of rated flow~~ and will ~~maintain water conditions as specified in 9.1.3.2.~~

during normal operations, operating at a maximum fuel pool water flow rate of 1000 gpm

~~The following components of the fuel pool cooling and cleanup system (FPC) are designed to ASME Section III, Class 3: fuel pool cooling pumps, filter demineralizers, pumps, valves, and piping, FPC piping, and fuel pool heat exchanger. The system heat exchangers are also designed to the standards of the Tubular Exchangers Manufacturers Association, Class R. Piping in the reactor building is controlled and supported to Seismic Category I requirements. The water lines between the fuel pool and RHR systems are designed to ASME Section III, Class 3, Seismic Category I requirements . The FPC pumps are not designed to Seismic Category I requirements. Condensate piping in the reactor building is controlled and supported to Seismic Category I requirements.~~

A radiological evaluation of the cleanup system is presented in Chapter 12.

From the foregoing analysis, it is concluded that the fuel pool cooling and cleanup system meets its design basis and satisfies the requirements of Regulatory Guide 1.13, Revision 1 with exceptions as noted in this section.

Page 9.1-27 insert I:

All piping connecting to the fuel pool, reactor well, and dryer separator pool and their respective liner drains are Seismic Category I up to and including either the normally closed, manually operated drain valve or the normally open, redundant isolation valves which can isolate the non-seismic portion of the system. Since the fuel pool system is at low temperature and pressure (moderate energy system) postulated breaks in the Seismic Category I portion are limited to cracks.

Fuel pool cooling can be established and monitored from the control room following a design basis LOCA. Entry to the reactor building is not required. One of the two redundant trains is adequate to prevent fuel pool boiling by a large margin. Due to the large thermal capacity of the fuel pool, sufficient operator time is available after a LOCA or any other event for the operator to take action.

Attachment II for page 9.1-27.

Each filter demineralizer is capable of continual performance at a normal fuel pool water flow rate of 575 gpm, or a maximum fuel pool water flow rate of 1000 gpm, and will maintain water conditions as specified in 9.1.3.2.

9.1.3.4 Testing and Inspection Requirements

No special tests except as noted below are required because at least one pump, heat exchanger, and filter demineralizer are continuously in operation while fuel is stored in the pool. Duplicate components are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

SW flow to the fuel pool heat exchangers and operability of the valves which intertie the SW and RCC systems are tested in conjunction with testing of the SW system.

TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM EQUIPMENT DATA

Fuel Pool Heat Exchangers

Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/heat exchanger	4.0×10^6
Cooling Water Flow, gpm/heat exchanger	575
Code and Standards	ASME/III-Class 3 and TEMA-Class R
Seismic Category	~~II~~ I

Fuel Pool Circulation Pumps

Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	575
Head, Ft of H_2O	160
Motor Size hp	40
Seismic Category	~~II~~ I
Code	ASME/III-Class 3

Fuel Pool Filter Demineralizer

Number	2
Design Flow Rate, gpm	1000
Design Pressure, psig	150
Design Temperature, °F	150
Material	CS-Plastic Lined
Code	ASME/III-Class 3
Seismic ~~Class~~ Category	II

Piping and Valves

Design pressure, psig	150/300
Design Temperature, °F	220
Material	CS
Code	ASME/III-Class 3
Seismic Category	
Fuel pool cooling portion:	I
Cleanup portion:	II

Q. 010.058
(9.2.2)

Since the non-safety-related reactor building component cooling water system provides cooling for the reactor recirculation pumps, state the length of time that the pumps can be left without component cooling water flow before significant seal damage can occur, with consequent potential primary coolant leakage:

a) if pumps are kept running; and
b) if pumps are turned off.

Response:

Recirculation pump seal cooling is provided by both closed cooling water to the pump seal heat exchanger and control rod drive seal purge flow. If an event occurs where both pump seal cooling sources are lost, the pump seals will heat up, causing pump seal deterioration when temperatures exceed 250°F. Vendor test data, taken while operating at approximately 530°F and 1040psia, indicate that the seals will reach 250°F approximately 7 minutes after a total loss of cooling. This will occur whether or not the pump is running.

Similar test data indicate that if one of the two seal cooling sources is operating, the pump seal temperatures will remain below 250°F and no seal deterioration should occur.

If both pump seal cooling sources fail, resulting in extreme degradation of the pump seals, the primary coolant loss has been analyzed to be less than 70 gpm. Refer to NEDO-24083, "Recirculation Pump Shaft Seal Leakage Analysis", November 1978 (Licensing Topical Report). This small amount of primary coolant leakage will be compensated for by normal or emergency water level controls. Consequently, no hazard to the health and safety of the public will result from total loss of recirculation pump seal cooling.

The position discussed above has been presented to NRC in FSAR Appendix B, response to NUREG 0737, item II.K.3.25.

Q. 010.059

Regulatory Guide 1.27 requires that there be sufficient water in the spray ponds for 30 days of cooling without make-up. Discuss how you will monitor the build-up of sediment on the floor of the ponds so as to assure available of the 30 day water supply. Describe how you will clean the spray ponds without losing redundancy or degradation of the system.

Response:

Sediment build-up on the floor of the spray ponds will be monitored once every 3 months from fuel load to the first refueling outage, when the frequency of monitoring will be adjusted based on operating experience. Sediment depth will be limited to an average of 0.5 feet based on the assumption made in Section 9.2.5.3a for spray pond thermal analysis.

Sediment will be removed by sludge pumps utilizing hand held suction lines. Make-up water will be supplied by normal pond make-up. This method will allow cleaning of the spray ponds without losing redundancy or degradation of the system.

Q. 010.060

The FSAR states there is "...a suction head of at least 20 feet during RCIC operation" from the condensate storage tank at elevation 443'0" and the RCIC impeller elevation 427'3". Discuss how the 15'9" elevational difference between the condensate storage tank and the RCIC impeller satisfies the 20' requirement.

Response:

The 15'9" elevation difference stated in the question is the distance between the bottom of the condensate storage tank (CST) (el. 443'0") and the RCIC pump impeller (el. 427'3"). The RCIC pump does not take suction from the bottom of the CST, but rather from the side of the tank at elevation 445'4" (centerline of the suction pipe). The RCIC suction automatically transfers to the suppression pool when the water level in CST reaches an elevation of 447'4" (low-low-level). Therefore, the minimum static head when the RCIC pump is taking suction from the CST is at least 20 feet (el. 447.4" minus 427'3").

FSAR Section 9.2.6.5 states that at least 20 feet of "suction" head is available between the low-low setting and the RCIC pump impeller. Although this statement is correct, it has been revised to say "static head" instead of "suction head" to avoid confusion.* The suction head available to the RCIC pump when the suction transfers at low-low-level in the CST is calculated using the equation for net positive suction head (NPSH) as shown in our response to FSAR Question 022.038. Assuming a water temperature of 100°F, there is about 48 feet of NPSH available at the centerline of the pump suction which more than satisfies the 21 feet of NPSH required by the RCIC pump.

* Draft revised FSAR page change attached.

(Revise for SCN ~~81-204~~)

The ~~E~~elevation differential ~~s~~ between the low-low level setting and ~~the HPCS pump impeller (Elev. 420'-5 1/2") and~~ the RCIC pump impeller (Elev. 427'-3") provides a static ~~suction~~ head of at least 20 feet for ~~during those~~ RCIC operation. ~~(Tank bottom elevation is at 441 feet)~~. The calculated NPSH available for RCIC operation is ~~at~~ ~~and~~ 48 feet of which 21 feet is required.

Thermostatically controlled tank heaters are provided to maintain water temperature in the tanks at or above a nominal 40°F at all times. All above ground piping that contains water is heat traced to prevent freezing.

PHM 6/25/81

System logic diagrams are given in Chapter 7.

9.2.7 STANDBY SERVICE WATER SYSTEM

9.2.7.1 Design Bases

a. The standby service water system (SW) is designed to remove heat from plant systems which are required for a safe reactor shutdown following a LOCA.

b. The system is designed to remove reactor decay heat from the residual heat removal system during normal plant shutdown.

c. The system is designed to perform its required cooling water function following a LOCA, assuming a single active failure.

d. The system is designed to provide a means of flooding the vessel and containment, if required during the post-LOCA period.

e. The system is designed to provide makeup source of water for ensuring fuel pool evaporative cooling following a LOCA in conjunction with a design basis earthquake.

f. The system is designed to Seismic Category I and ASME Code, Section III, Class 3 requirements with the exception of that portion to and from the plant cooling towers, which is designed to ANSI B31.1 and Seismic Category II requirements.

9.2.7.2 System Description

The standby service water system includes vertical service water pumps located adjacent to the two spray ponds in two

Q. 010.061

The nitrogen bottles with its associated equipment and containment instrument air system shall be a minimum of Quality Group C.

Response:

All components of the containment instrument air system from and including the outboard containment isolation valves to the main steam isolation, safety relief, and automatic depressurization (ADS) valve operators in containment are Quality Group B. The portion of the system associated with the backup supply of nitrogen to the ADS valve operators is Quality Group C from the outboard containment isolation valves to and including the solenoid pilot valves (CIA-SPV-1A through 15A and -1B through -19B) which are mounted adjacent to the pressure reducing valves for the nitrogen bottles. See Figure 9.3.2.

The nitrogen bottles and associated control valves are standard, commercially available units that meet the requirements of Department of Transportation (COT) and Compressed Gas Association (CGA) standards. The nitrogen bottle units are mounted per Siesmic Category I, Quality Class I requirements. This type of nitrogen bottle assembly has proven highly reliable based on years of reactor operating experience.

The remaining components of the containment instrument air system, which are Quality Group D, are not essential for safe operation of the plant.

Q. 010.062
(9.4.1)

In your response to question 010.29 there seems to be a contradiction between the thickness of the air intake roof slab and the height of the top of the roof slab above grade. Please clarify your numbers and provide physical drawing(s) of the air handling system with details of the remote air intake structures.

Response:

The remote air intake structure is a buried structure. Only a portion (15") of the 24 inch thick roof slab projects above grade. Please refer to sections 4298 and 4299 of figure 3.5.52.

Q. 010.063

Discuss the control room environment which will result from the most extreme ambient and accident conditions (including the worst single failure for the HVAC). Note: The temperature/humidity for all operating/accident conditions shall be maintained within the comfort zone as defined by ASHRAE. This requirement applies to all areas which require operating personnel.

Response:

To maintain the control room at an ambient conditions which is compatible to the comfort zone defined in ASHRAE redundant seismic and environmentally qualified liquid chillers are being incorporated into the control room HVAC design. Paragraph 9.4.1 and 6.4 will be updated when this design is completed. The control room is the only area with essential equipment where personnel are routinely required during accidents.

Q. 010.064

Discuss the effects of a potential failure of the non-seismic Category I heaters in the standby service water pumphouse under the most adverse environmental conditions on the operability of the pumps.

Response:

The standby service water pumphouse electric unit heaters are designed to maintain the building above freezing during extreme environmental condition. Failure of the unit heaters during a seismic event would not degrade the operability of the pumps. If the pumps are required to operate following a seismic event, the heat generated from the pump motor is sufficient to maintain the building above freezing. Annunciation is provided in the main control room when temperature in the pump room drops to 35°F. Appropriate action can be taken from the control room such as starting of the standby service water pump to prevent pipe freezing.

Q. 010.065

Your responses to question 010.034 regarding the potential flooding of safety related equipment due to a circulating water failure are inadequate. An analysis shall be conducted in accordance with Standard Review Plant 10.4.5, "Circulating Water System", which assumes:

1) An expansion joint break (note: an incident of this type occurred at an operating BWR).

2) No credit shall be taken for isolation valve closure unless these valves are designed to safety grade requirements.

Response:

Please see revised section 10.4.5.3.*

* Draft FSAR page changes attached.

10.4.5.3 Safety Evaluation

The circulating water system is a non-safety related system. Consequently, the circulating water system is not designed to Seismic Category I requirements. Refer to 9.2.5 for a description of the ultimate heat sink which is designed to perform safety-related functions.

The condenser design assures that the pressure on the tube side is always maintained higher than the pressure on the shell side, thus eliminating leakage into the circulating water system should tube failure occur. Consequently, the design of the circulating water system precludes radioactive leakage into the system.

Periodic injection of chlorine is performed for biocide treatment, and sulfuric acid is added for scale-corrosion control within the circulating water system. An analysis of the transportation, handling, storage, and utilization of chlorine is presented in 6.4.

Two ~~A detailed~~ evaluations were ~~was~~ performed to determine the effects of a postulated failure in the circulating water system inside the turbine building; an "realistic" evaluation and a boundary evaluation. For ~~this analysis~~ the "realistic" evaluation, a moderate energy crack was postulated to occur in the circulating water system barrier, (e.g., the rubber expansion joints) at the inlet to the main condenser. The inlet side was selected because it yields the severest results. For the bounding evaluation, a complete circumferential expansion joint break was assumed.

The entire condenser area is drained by means of sumps (see Figure 9.3-9), each equipped with duplex pumps. Sumps T-2 and T-3, servicing the inlet and outlet of the condenser, each have 50 gpm pumps. Each of these sumps is equipped with a level alarm and is therefore capable of detecting a circulating water system barrier failure. The level alarm will annunciate in the main control room upon reaching high level, providing a means of detecting the postulated failure within 5 minutes.

"Realistic" Break

The crack area for this postulated failure was assumed to be equal to 1/2 the pipe diameter times 1/2 the pipe wall thickness.

$$A = \frac{d}{2} \times \frac{t}{2} \quad \text{(see 3.6.2.1.4.2.b)}$$

In the first 5 minutes after a crack, 8,435 gallons of water will spill into the inlet basin. The capacity of each basin and its capability to store excess flow were calculated to be as follows:

a. Inlet basin: 22,500 gallons from El. 436 to El. 441

b. Outlet basin: 27,500 gallons from El. 436 to El. 441

c. Net volume under condenser: 180,500 gallons from El. 433 to El. 441.

The time required to fill the inlet basin, after a postulated crack occurs, is computed to be 13.3 minutes. This includes the 50 gpm outflow from the sump pump. The circulating water leakage flow will continue for 6.7 minutes after filling the inlet basin, until reaching the total estimated shutoff time of 20 minutes. It can be assumed that 10% of this water will flow out over the floor at El. 441, and the remainder, about 10,170 gallons, will flow into the condenser basin area. During this same time period, 4 sump pumps in the condenser basin area will have alternately pumped out 670 gallons, leaving 9500 gallons or 0.42 feet of water in the condenser basin. The rate of rise of water, therefore, is 0.021 ft/min during the first 20 minutes after the postulated crack occurs. Note that on high sump level, both pumps run simultaneously rather than alternately, thus doubling the calculated outflow capacity.

After the valves are closed, the water contained in the condenser unit water box will continue to discharge to the area. The quantity of water remaining is estimated to be 87,000 gallons. The flow will vary with a diminishing head, the head going from about 25 feet to zero feet. Using a 20 ft head and the same orifice flow criteria, the rate of flow will be approximately 819 gpm, discharging the remaining water in about 106 minutes. There will be an outflow from all the sump pumps of 150 gpm, with 10% of the flow from the crack again assumed to flow out over the floor. The water will accumulate in the condenser basin at about 590 gpm. After 106 minutes, the water level in this basin will rise an additional 2.77 feet, ~~or~~ at 0.0261 ft/min. The total height of water when the discharge has stopped is therefore 3.19 feet to El. 436.19. This elevation is 5 ft. below the ~~[illegible]~~ floor level of the Turbine Bldg (el. 441), thus there will be no ~~safety~~ impacts on safety-related equipment from this event.

→ [INSERT from attached page]

~~There are no safety-ralated system components that could be affected by the flood elevation established above. Additionally, there are no safety-related electrical systems or system components that could be potentially submerged. In addition, the circulating water piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, spray effects are of no consequence. The pipes exit the rooms below grade in their routing to and from the cooling towers. Also, as stated above the floor onto which water would spill in event of a break is grade level. As a result, excess water would accumulate either in drainage basins or leak outside the building.~~

Discharge operation of water accumulated under the condenser shall be performed in accordance with radioactivity checking requirements for sump discharges.

10.4.5.4 Tests and Inspections

All system components, except the condenser, are accessible during operation and may be inspected visually. The circulating water pumps are tested in accordance with the Hydraulic Institute Standards.

Q 010.65

Insert to 10.4.5.3

Boundary Evaluation

A complete circumferential expansion joint break in the circulating water system would result in the release of large amounts of water into the turbine-generator building. The water would fill the net volume under the condensor, tripping the sump high level alarms which annunciate in the main control room. Remote-manual operation of the circulating water pumps and butterfly valves is provided in the main control room to mitigate the accident.

Disregarding operator action; however, the following evaluation is provided: Water would spill across the grade level floor of the turbine-generator building at elevation 441 ft., exiting through the railroad bay and access doors. Water could flow into the reactor building stairwells and elevator shafts from the 441 ft. elevation down to the 422 ft elevation, eventually filling the stairwells and elevator shafts with water. There is no safety-related equipment located in the stairwells or elevator shafts. The access doors to the ECCS pump rooms at elevation 422 ft. are sealed watertight and designed to withstand a static head of 44 ft. of water. All penetrations into the reactor building below the 471 ft. elevation are sealed watertight. Water would not affect any safety related equipment in the reactor building.

Water could also spill across the grade level floor into the Radwaste /control building. The basement level of this building is 437 ft. It is thus possible to flood this level with four feet of water before the water would exit at grade level (441 ft) through access doors. There are no safety-related components which would be affected by this flooding.

The railroad bay and access doors of the turbine-generator building are not watertight and are not designed to withstand any static head of water, therefore no significant depth of water could accumulate in the turbine-generator building. All safety-related equipment in the turbine-generator building is located above the 471 ft. elevation and would not be affected.

In conclusion, a complete circumferential expansion joint break in the circulating water system inside the turbine-generator building would have no effect on safety-related equipment.

Q. 010.34
(10.4.5)

Your response to Item 010.09 is unacceptable. Specifically, your analysis of flooding due to failure of the circulating water system is based on a crack whose area is equal to one-quarter of the pipe diameter times the pipe thickness (.5t X .5d). Provide an analysis of flooding due to a postulated failure of the expansion joint in the circulating water system assuming a double-ended guillotine break at this location.

Response:

~~The original response to Item 010.09 has been rewritten for clarity (see 10.4.5).~~

~~The double-ended guillotine break referred to above was not considered. The circulating water system is a moderate energy system by definition. Therefore, in accordance with NRC Standard Review Plan Section 3.6.1, 3.6.2, and 10.4.5, and the associated Branch Technical Position MEB 3-1, the criteria for a postulated failure shall be a through - wall leakage crack of the type addressed in the written response (10.4.5). In any case, as stated at the end of 10.4.5, circulating water piping is located remote from any safety-related equipment. The piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, safety-related equipment is not vulnerable to environmental effects of a circulating water pipe rupture. The pipe exits the room below grade in its routing to and from the cooling towers. It should be also noted that the condenser is located on grade level. Therefore, water above the floor elevation will drain outside and not collect other than in collection basins.~~

See revised FSAR section 10.4.5-3.