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Dalwyn R. Davidson
VICE PRESIDENT
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January 20, 1982

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

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Request for
Meeting - Instrumentation
and Control Systems

Dear Mr. Schwencer:

This letter and its attachment is submitted to provide draft responses to several of the concerns identified in your letter dated November 17, 1981. A meeting was held with members of the Instrumentation and Control Systems Branch on December 16 and 17 to discuss these concerns and identify action items. Remaining agenda items will be addressed in future correspondence as agreed upon with the Instrumentation and Control Systems Branch reviewers.

Very Truly Yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD: mlb

Attachment

cc: G. Charnoff, Esq.
M. D. Houston
NRC Resident Inspector
J. Mauck

B001
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421.10 Provide an evaluation of the effects of high temperatures in reference legs of water level measuring instruments subsequent to high energy line breaks.

Response

The Perry Nuclear Power Plant is of the BWR6-Mark III containment design which has the reactor vessel water measurement reference leg and sensing legs outside of the drywell as illustrated in the attached sketch.

Cleveland Electric has evaluated the potential for high drywell temperatures to cause errors in level sensing instrumentation. High drywell temperatures could produce errors in two ways:

(1) By causing density changes in the water in the sensing lines due to increased temperature in the drywell.

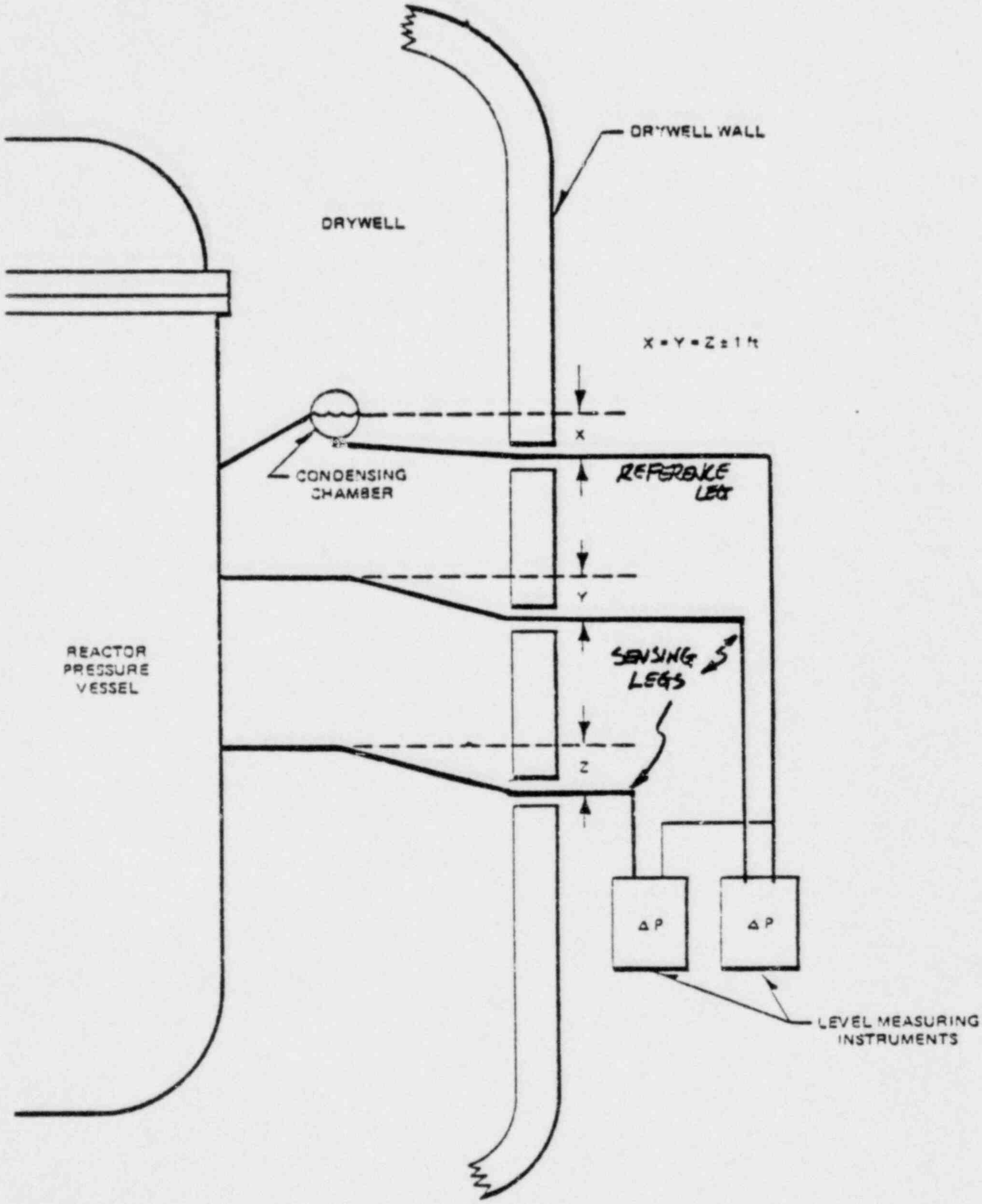
These effects are accommodated in the design by making the vertical drop of the upper and lower sensing lines within the drywell essentially equal so that errors due to density changes resulting from increases in drywell temperature are minimal.

(2) By boiloff of the reference leg when the reactor is depressurized below the saturation pressure of the reference leg temperature (drywell temperature). This effect can only be produced by small steam leaks in the order of 0.01 ft^2 or less for an extended period of the time resulting in heatup of drywell and subsequently the sensing lines within the drywell. Larger breaks do not result in drywell temperature sufficiently high to cause potential problems due to the resulting high drywell pressure signal which will cause scram and depressurization of the RPV in several minutes or less. In the case of the highest drywell temperatures (assuming the operator has ignored), the RPV would have to be depressurized below 118 psia to start boiloff of the reference legs. The water in the sensing legs will always be refilled by the low pressure ECC pumps upon depressurization of the vessel.

Since the maintenance of the reactor level is the primary concern of the operator under the postulated conditions, it is difficult to perceive that depressurization could occur without pumps being on to flood the vessel. In the unlikely event the operators should make such gross procedural errors as to use the SRVs instead of ADS and to neglect the starting of any pumps prior to depressurization to 118 psia, the reference leg initial boiloff would be about 20 percent. Analysis shows it would take about 9 1/2 hours to boiloff the remaining water in the reference leg if no operator action is taken.

Moreover, the emergency procedures will require the operator to monitor the drywell temperature and when that temperature reaches the RPV saturation limit as determined from the figure given in the emergency procedures, to depressurize and flood up the vessel to refill the reference lines. If the operator neglects these emergency procedures as well, and should the level drop under these conditions, a level 2 alarm would occur with the actual level over 12 inches above the top of the active fuel which would allow 10 minutes for operator action to start pumps prior to the level reaching the top of the core.

A similar evaluation was made on the Grand Gulf docket and the NRC Staff has concluded that the reactor vessel level measuring system is acceptable.



Cold Reference Leg Water Level Instrument Piping Routing

TYPICAL FOR MARK III - BWR 6

421.14 A discussion of the Equipment Protection Assembly (EPA) systems is not given in FSAR Section 7.2. Also, the EPA assemblies are not shown in Figure 7.2-1, the reactor protection system instrumentation and control diagram. Discuss the EPA system and how it meets IEEE 279.

Response

A detailed discussion of the Equipment Protection Assembly (EPA) can be found in Perry FSAR Section 8.3.1.1.5.1 on page 8.3-36. Revised section 7.2.2.2 to include this reference is attached.

The RPS is highly reliable and .. provide a reactor scram in the event of anticipated operational occurrence.

7.2.2.2 Conformance to I' .tandards

The following is a discussion of conformance to those IEEE standards which apply specifically to the RPS system. Refer to Section 7.1.2.3 for a discussion of IEEE standards which apply equally to all safety related systems. The non-essential RPS power and its electrical protection assembly (EPA) are discussed in Section 8.3.1.1.5.1.

a. IEEE Standard 279 Criteria for Protection Systems for Nuclear Power Generating Stations - The RPS design complies with the requirements of IEEE-279. The following is a discussion of specific conformance.

   1. General Functional Requirement (IEEE Standard 279, Paragraph 4.1)

      The RPS automatically initiates the appropriate protective actions, whenever the conditions described in Section 7.2.1.1.b reach predetermined limits, with precision and reliability assuming the full range of conditions and performance discussed in Section 7.2.1.2.

   2. Single Failure Criterion (IEEE Standard 279, Paragraph 4.2)

      Each of the conditions (variables) described in Section 7.2.1.1.b is monitored by redundant sensors supplying input signals to redundant trip logics. Independence of redundant RPS equipment, cables, instrument tubing, etc. is maintained and single failure criteria preserved through the application of the PNPP separation criteria as described in Section 8.3.1 to assure that no single credible event can prevent the RPS from accomplishing its safety function.

   3. Quality of Components and Modules (IEEE Standard 279, Paragraph 4.3)

      For a discussion of the quality of RPS components and modules, refer to Section 3.11.

421.14

421.15 Discuss the differential pressure transmitters that are used to monitor reactor vessel high water level (trip level 8) and the low water level trips. Discuss the bypassing of the reactor vessel high water level trip in all operating modes but run. Does this bypass the low level trip at all operating modes but run?

<u>Response</u>

Based on discussion and review with NRC on December 16, 1981, the staff required no further information on this question.

421.23 From the discussions provided in Sections 6.3.2.2.3, 6.3.2.2.4, 7.3.1.1.1.3, and 7.3.1.1.1.4, it is not clear whether or not the LPCS and LPCI injection valves are interlocked to prevent them from opening unless reactor pressure is low enough for injection to be possible. Provide more information concerning the operation of these valves. Also, there are discrepancies in the FSAR as to whether differential or gage (absolute) pressure transmitters are used for the interlocks. For example, Section 7.3.1.1.1.3 and 7.3.1.1.1.4 imply differential pressure transmitters are used. However, the P&I diagram for the LPCS system, Figure 6.3-8, does not show a differential pressure transmitter near the injection valve.

Response

The present Perry design of the high pressure/low pressure interface of the low pressure ECC lines is illustrated in Figure 5.4-13, Sheet 2. The pressure interlock (a pressure transmitter/trip unit between the testable check valve and the motorized injection valve) is designed to be functional during test opening of the MOV. Automatic initiation of the low pressure ECCS by the LOCA signal will bypass the interlock and immediately open the MOV'S. The differential pressure transmitters mentioned on pages 7.3-8 and 7.3-10 were not used in the Perry design and the FSAR. Revised pages are attached to reflect current design.

Reactor vessel water level (Trip Level 1) is monitored by two redundant level transmitters. Drywell pressure is monitored by two redundant pressure transmitters. The vessel level trip unit relay contacts and the drywell pressure trip unit relay contacts are connected in a one-out-of-two twice logic arrangement so that no single instrument failure can prevent initiation of LPCS.

The LPCS components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows:

1. The Division 1 diesel generator is signaled to start.

2. The normally closed test return line to the suppression pool valve MOF012 is signaled closed.

3. When power (offsite or onsite) is available at the LPCS pump motor bus, the LPCS pump is signaled to start.

4. The LPCS pump discharge flow is monitored by a differential pressure transmitter. When the pump is running and discharge flow is low enough to cause pump overheating to occur, the minimum flow return line valve MOF011 is opened. The valve is automatically closed if flow is normal.

   The LPCS pump suction from the suppression pool valve MOF001 is normally open, the control switch is keylocked in the open position, and thus requires no automatic open signal for system initiation.

   The LPCS pump and injection valve are provided with manual override controls. These controls permit the operator to manually control the system subsequent to automatic initiation.

The Division 1 LPCI (Loop A) receives its initiation signal from the LPCS logic.

The LPCI system components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows (the loop A components are controlled from the Division 1 logic; the loop B and C components are controlled from the Division 2 logic):

1. The Division 2 diesel generator is signaled to start from the loop B and C initiation logic.

2. When the offsite power or the diesel generators are providing power to the pump motor buses, sequential loading is provided. This is accomplished by delaying the start of LPCI pumps A and B by 5 seconds while allowing the LPCS and LPCI pumps to start immediately.

3. The following normally closed valves are signaled closed to ensure proper system lineup: 421.3

   (a) The RHR heat exchanger discharge to RCIC valves MOF026 A, B, and AOF065 AB.

   (b) The RHR heat exchanger flush to suppression pool valves MOF011 A, B.

421.27 The initiating conditions for automatic operation of the RHRS-Containment Spray Cooling Mode, discussed in Section 7.3.3.1 of the Construction Permit SER, were not acceptable to the staff. Discuss how the present design complies with the recommendations listed in the Construction Permit SER.

Response

The Perry containment spray system loop A will be automatically initiated from the LOCA signal with a 10 minute delay and with a concurrent high containment pressure signal and high drywell pressure. An additional delay of 90 seconds is imposed on the B loop initiation with the above signals being present. This is consistent with the NRC Staff position given in the Perry Construction Permit SER. Revised Section 7.3.1.1.4 is attached to include this feature.

7.3.1.1.4 RHRS-Containment Spray Cooling Mode (RCSCM) - Instrumentation and Controls

a. Containment Spray Cooling Mode Function

The containment spray cooling mode is an operating mode of the RHR system. It is designed to provide the capability of condensing steam in the suppression pool air volume and/or the containment atmosphere and removing heat from the suppression pool water volume. The system is automatically or manually initiated when necessary.

b. Containment Spray Cooling Mode Operation

Schematic arrangements of system mechanical equipment is shown in Figure 5.4-13. RHR system component control logic is shown in Figure 7.3-5. Elementary diagrams are listed in Section 1.7.1. Plant layout drawings are shown in Section 1.2. Operator information displays are shown in Figures 5.4-13 and 7.3-5.

The Containment Spray Cooling Mode is initiated automatically or manually. LPCI flow is diverted to either the drywell or the suppression pool by opening valves MOF028A, B or MOF024A, B and closing MOF048A, B.

The following conditions must exist before containment spray can be initiated:

1. The LOCA signal which automatically initiated LPCI must still exist.

2. Drywell high pressure is monitored by two redundant pressure transmitters. One of the two transmitters must indicate high pressure.

3. The containment pressure must exceed 9 psig or higher. X2127

4. A 10-minute delay after LOCA is detected.

Initiation of the containment spray automatically closes the LPCI injection valve MOF042 A, B.

421.32 As stated in Section 6.2.5.5. of the FSAR, proper recombiner operation after an accident is determined by monitoring a watt-meter in the control complex. However, it is important that the recombiner temperature be maintained above 1150°F for proper operation. Discuss why temperature measurements are not used to indicate proper operation. IEEE 279 criteria require direct measurement of the desired variable, when practical.

Response

The response to this question was discussed and the NRC staff had no further questions.

421.33 Have hydrogen analyzers, qualified to IEEE 323 and 344, been procured?

Response

The hydrogen analyzers are being procured to be qualified to IEEE 323 and 344.

421.35 Demonstrate that the Safety Relief Valve (SRV) low-low setpoint function is adequate assuming a single failure.

<u>Response</u>

This item will be addressed through LRG-II.

421.41 The statement in Section 7.3.1.1.1.1 that the HPCS provides makeup water to the reactor until the vessel water level reaches the high level (trip level 8) conflicts with the statement in Section 6.3.2.2.1 regarding the HPCS system. Please indicate which discussion is correct.

Response

Current design includes a high drywell pressure interlock as discussed in Section 6.3.2.2.1. There is a design change in progress to remove the high drywell pressure interlock for the HPCS trip. The FSAR will be modified to reflect this change when the design has been finalized.

421.42 Discuss the statement that the HPCS pump motor and injection valves are provided with manual override controls. Does this violate the concern expressed in IE Bulletin 80-06? The statement is made in Section 7.3.1.1.1.4 that once initiated, the LPCI logic seals-in and can be reset by the control room operator only when initial conditions return to normal. This seems to conflict with a statement in the same section that indicates that the operator can manually control the system subsequent to automatic initiation. Explain this conflict.

Response

The IE Bulletin 80-06 will be resolved through LRG II efforts.

The LPCI logic seal-in can be reset by control room operator only when initial conditions return to normal. LPCI pump motors and injection valves are provided with manual override control which permits the operator to manually override the LPCI seal-in logic and control the system subsequent to automatic initiation.

421.43 Can the manual control switches provided for the ADS safety/relief valves initiate the system without the low pressure pumps operating?

Response

System level actuation of the ADS valves can occur (manually or automatic) only when a low pressure pump is operating. Individual ADS valves can be manually operated without low pressure pumps operating.

421.44 Can the LPCS suction valve be closed upon containment isolation if keylocked open?

Response

The LPCS suction valve is not closed upon automatic containment isolation if keylocked open. If this valve would be inadvertently closed, this condition will be annunciated on the Bypass and Inoperation Panel.

421.46 In the discussions concerning the design of the Standby Liquid Control System in FSAR Sections 9.3.5 and 7.4.1.2, very little information is provided concerning the design of the heating system required to prevent precipitation of the sodium pentaborate from the solution during storage. Provide a more detailed discussion on the design of the heating system, including associated instrumentation and controls. Include information concerning the power sources used for the instrumentation and heaters and any alarms used to indicate failure of the heating system.

Response

The following supplements discussions of SLCS data found in Section 9.3.5 and 7.4.1.2 of the FSAR:

The Standby Liquid Control System (SLCS) storage tank has a 10KW in-tank operating heater (B-208-030, Sh. 7) which is initiated automatically by an in-tank temperature switch (C41-N006) when solution temperature falls below 75°F. The heater shuts off when the solution temperature reaches 85°F.

Power source for heater and control is non-1E 480V AC as shown on B-208-030, Sh. 7.

A separate in-tank temperature switch (C41-N003) is provided for SLC storage tank temperature high/low alarm on the ECCS benchboard control room annunciator (B-208-222 Sh. 355).

The sodium pentaborate line between the storage tank and the injection valves is heat traced to prevent precipitation. Low line temperature is alarmed in the control room. Power source for the heat tracing is non-1E 120 V AC.

421.53 This section indicates that the suppression pool water level and temperature, as well as the drywell and containment pressure and temperature, recorders and indicators will not perform during and after a seismic event even though these instruments are required for post accident monitoring. Provide a justification for the design basis of this instrumentation.

<u>Response</u>

The response to this question is provided in revised Section 7.5.2.4.2 attached.

be operable during and after a LOCA in conjunction with a SSE. Power is from independent instrument buses supplied from the two divisional a-c buses. This instrumentation complies with the independence and redundancy requirements of IEEE Standard 279 and provides recorded outputs.

The reactor water level and pressure sensors are mounted on four independent local panels. The sensors and recorders are designed to operate during normal operation and/or post-accident environmental conditions. The design criteria that the instruments must meet are discussed in Section 7.1.2. There are two complete and independent channels of wide range reactor water level and reactor vessel pressure with each channel having its readout on a separate two-pen recorder or water level indicator.

The design, considering the accuracy, range and quality of the instrumentation, is adequate to provide the operator with accurate reactor water level and reactor pressure information during normal operation, abnormal, transient, and accident conditions.

b. Suppression Pool Water Level and Temperature

This instrumentation complies with the requirements of IEEE Standard 279 and provides recorded outputs. All equipment except the recorders and indicators will perform its required function during and after the seismic event. Recorders and instrumentation perform their required function after the seismic event; however, pen or pointer flutter is expected to occur during the event.

c. Drywell and Containment Pressure and Temperature

This instrumentation is redundant, electrically independent, and is qualified to be operable during and after a LOCA. Power is from independent buses and the instrumentation complies with the requirements of IEEE Standard 279 and provides recorded outputs. All equipment except the recorders and indicators will perform their required function during and after a seismic event. Recorders and instrumentation perform their required function after the seismic event; however, pen or pointer flutter is expected to occur during the event.

421.55 Are the positions of the check valves used to interface between the low pressure and high pressure portions of the residual heat removal system annunciated in the control room? According to Figure 5.4-13 (Sheet 2), check valves F019 and F050 do not have their position annunciated. How does the design conform with Position 4 of Branch Technical Position ICSB No. 3.

Response

The interface valve at the high pressure/low pressure boundary for the shutdown cooling return line is F053 and for the vessel head spray line is F023. These valves have open/close indications in the control room. The two lines can only be brought into operation manually after the operator has shutdown the reactor,and the pressure interlock determines the vessel pressure is low enough. The check valves F019 and 050 are not manually operated. These check valves will be tested for leakage periodically.

421.56 Describe the periodic self-test mode of the Rod Pattern Control System.

<u>Response</u>

Upon discussion and review with NRC staff on Dec. 16, 1981, the staff required no further information on this question.

421.58 BWR operating experience has shown that the RHR and RCIC systems have been rendered inoperable because of inadvertent leak detection isolations caused by equipment room area high differential temperature signals. The events occurred when there was a relatively sharp drop in outside temperature. As noted in Section 7.6.1.3, the Perry Nuclear Power Plant incorporates this type of RCIC and RHR (steam) isolation. Provide a discussion of any modifications that have been or will be made to prevent inadvertent isolations of this type which affect the availability and reliability of the RCIC and the RHR systems.

Response

The response to this question has been provided by referring to 440.11 (submitted October 30, 1981).

421.60 Provide a comparison of the systems discussed in Section 7.7 with those of similar approved plants. Identify systems of new design and differences in systems with similar designs.

<u>Response</u>

The response to this question was reviewed and the NRC staff had no further questions.

421.61 Has the Perry design for increasing control and rod insertion rate incorporated the prompt relief trip concept or the fast scram concept?

<u>Response</u>

Perry design does not have the Prompt Relief Trip. The Perry design incorporates the Fast Scram concept.