APP/3

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Docket Nos. 50-387/388

Mr. Norman W. Curtis
Vice President - Engineering
and Construction
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

MAR 21 1980

Dear Mr. Curtis:

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS NOS. 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION

As a result of our review of your application for operating licenses for the Susquehanna Steam Electric Plant, we find that we need additional information in the area of Reactor Systems. The specific information required is listed in the Enclosure.

Some of this review has been performed by the Savannah River Plant (SRP). The questions in the Enclosure were originated by SRP.

Please contact us if you desire any discussion or clarification of the information requested.

Sincerely,

Original Signed by
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosure:
As stated

cc w/enclosure:
See next page

8004210410

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ENCLOSURE

211.0 REACTOR SYSTEMS BRANCH

211.193 (3.5.1.2) In 3.5.1.2.3 you state that "Equipment which is not necessary for operation or safety is removed from containment or secured in place prior to operation of the reactor to ensure that it will not become a missile." Are all the supports for the above equipment capable of surviving during an SSE?

211.194 (3.5.1.2) Discuss the possibility of the CRD mechanism becoming a missile inside containment.

211.195 (3.5.1.2) Your response to Question 211.39 in reference to bonnet ejection of ANSI 900 trip rated valves in revised Subsection 3.5 and 3.5.1.2 is qualitative. Supply a mathematical analysis supporting your contention that bonnet ejection of ASME, Section III, pressure seal bonnet type valves is improbable.

211.196 (3.5.1.2) You assign low probabilities (although no numerical values are given) to pressurized components or rotating equipment parts becoming primary missiles. In the event such a missile does occur, however:

a) What specific barricades are provided to prevent failure of safety equipment within containment due to the impact of the missile?

b) What secondary missiles might be generated by the primary missile and how will their effects be mitigated?

211.197 (3.5.1.2) Estimate the damage or failure caused in safety related equipment within containment due to impact by credible primary or secondary missiles.

211.198 (5.2.2) In Subsection 5.2.2.2.3.1 of the FSAR, you state that the required safety valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. Figure 5.2-1 shows curves produced by this analysis.

a) In Figure 5.2-1 the curves for vessel pressure rise and steam line pressure rise exceed the scale of the graph so that it is unclear what the maximum pressure is and when it occurs. Provide a plot with appropriate scales so that the maximum pressures are clearly shown.

b) In your response to Q211.4 you state that analyses show that adequate margin exists in the design of the S/R valve system, so even if the flux scram signal failed and the event was terminated by a pressure scram, the peak vessel pressure would be less than the ASME code limit. Provide the results of these analyses and indicate the % relief capacity needed to keep peak vessel bottom pressure less than the ASME code limit.

211.199 (5.2.2) What is the pressure safety margin calculated for the MSIV closures with flux trip?

211.200 On page 5.2-14, it is stated that it is not feasible to test the safety/relief valve setpoints while the valves are installed. It would appear that improper setpoints (due to such faults as erroneous setpoint calculation) would be credible common failure mode which can result in degradation of the pressure relief systems. Provide assurance that a credible common failure mode in the failure-to-opendirection has been properly considered. Provide the results of a data search of operating reactors indicating the frequency with which this type of failure has occurred (improper setpoint).

211.201 (5.2.2) Provide the results of the hydraulic calculations that show the Mach number, pressure, and temperature at various locations from upstream of the safety/relief valves to the suppression pool at maximum flow conditions. The concern is related to the potential for the development of damaging shock waves to the discharge piping. Include the effects of suppression pool swell variations on the operation of the safety relief valves.

211.202 (5.2.2) Referring to Subsection 5.2.2.4.1, on page 5.2-9 in the FSAR, you state that the pneumatic accumulator provided for each safety/relief valve has sufficient capacity to provide one safety/relief valve actuation. It appears from Figure 5.1-2, Nuclear Boiler, that the air supply line upstream of the ball inlet check valve for non-ADS safety/relief valves is not safety grade. If an airline break occurred upstream of the check valve, would there be indication in the control room of this break and the status of the accumulator? If indication is given, what operator action would be required?

211.203 (5.2.2) In the Susquehanna analyses, what capacity is assumed for each group of valves that are actuated at their power-operated relief setpoint?

211.204 (5.2.2) Submit an overpressure report as required by the ASME Boiler and Pressure Vessel Code, Section III which is referenced in Section 5.2.2 of the Standard Review Plan.

211.205 (5.2.2) Were the curves in Figure 5.2-5 which shows the pressure at the vessel bottom versus time for the MISIV transients based on 105% of rated steam flow? If not, provide these curves.

211.206 (5.2.2) In your response to Question 211.4 in Table 1 on page 211.4-5, you state:

a) Safety/relief Valve Setpoint - psig 1091 to 1111

b) Typical Valve Capacity - % NBR Steam Flow - 5-10 per valve

c) Typical Total Relief Valve Capacity (% NBR Steam Flow) 75-85

211.206 (5.2.2) In Chapter 15 and in your response to Question 211.76, you give the power-operated relief setpoints used in your transient analysis as 1091 - 1131 psig.

In your response to Question 211.76, you state the total capacity of the valves at the first relief setpoint of 1091 psig to be 99% NBR steam flow.

At 1091 psig, two (2) valves open according to the groups defined in Table 5.2-2. If each valve has 5-10% NBR steam flow capacity, how can the total capacity at 1091 psig be 99% of NBR steam flow? Clarify all the above inconsistencies involving setpoints and capacities.

211.207 (5.2.2) Nominal spring mode safety/relief valve setpoints are given in Table 5.2-2 and again in Table 1 of Figure 5.1-3a. The lowest setpoint in Table 5.2-2 is given as 1146 psig whereas in Figure 5.1-3a it is shown to be 1130 psig. Resolve this inconsistency. Also, what is the basis for the pressure setpoint increments between groups?

211.208 (5.2.2.2) (15.0) In Table 5.2-2, five safety/relief valve groups are identified with the nominal spring mode pressure setpoints given for each group. Table 15.0-2 identifies five power actuated relief mode setpoints. Do the latter correspond to the same groupings given in Table 5.2-2? That is, are there two valves set at 1091 psig, four at 1101 psig, etc.? If not, provide the proper power actuated mode groups.

211.209 (5.2.2) (15.0) In Table 15.0-1, the number of relief valves involved in the first blowdown following various transients is given.

In the case of pressure regulator fail-open, the maximum steam line pressure is indicated as 1092 psig. With the first group of safety/relief valves set at 1091 psig, two valves should blowdown, not zero as indicated by Table 15.0-1.

In the case of the loss of auxiliary power transformer, the maximum steam line pressure is indicated at 1105 psig. If the valve groupings are the same as in Table 5.2-2, this should cause blowdown of the first two groups of valves (6), not the 10 indicated in Table 15.0-1.

Resolve these apparent discrepancies.

211.210 (6.3) Expand the discussion in Section 6.3 to describe the design provisions that are incorporated to facilitate maintenance (including draining and flushing) and continuous operation of the ECCS pumps, seals, valves, heat exchangers, and piping runs in the long-term LOCA mode of operation considering that the water being recirculated is potentially very radioactive.

211.211 (6.3) Severe water hammer occurrence in the ECCS discharge piping during startup of the ECCS pumps is avoided by ensuring that the discharge pipes are maintained full of water. The condensate transfer system is used to achieve this function for all ECCS piping. Since the condensate transfer system also supplies water to numerous other systems, the following areas require clarification:

a) Justify the use of a common filling system for all ECCS discharge piping versus independent jockey pumps.

b) Identify the expected demands on the condensate transfer system and what effects, if any, would be expected on the makeup required to keep the discharge pipes full of water?

c) Can individual "fill lines" be isolated to permit maintenance on one ECCS system without affecting the other system?

d) The discharge piping "fill system" is apparently considered to be an auxiliary system. Are any priority interlocks provided to ensure that the "filling system" will be given priority over the other uses of the condensate transfer system water?

e) The individual fill lines apparently do not have instrumentation to monitor low pressure. Provide assurance that when the condensate transfer pumps are operating that the individual ECCS discharge lines are full of water.

f) What is the history of water hammer events at other plants employing this design?

211.212 (6.3) The description of the filling system for ECCS discharge piping in Section 6.3.2.2.5 of the FSAR addressses system operation for the RHR and core spray piping. The HPCI discharge piping is not discussed in this Section or Section 6.3.2.2.1 but Figure 6.3-1a shows a "filling line" for the HPCI discharge piping. Resolve this apparent discrepancy.

211.213 (6.3) The results presented in the FSAR for Section 6.3.3.7.5, 6.3.3.7.6, and 6.3.3.7.7 are supposedly taken from "typical" or the "lead plant" analysis for this product line. Identify the typical and/or lead plant and justify the selection in view of the criteria specified in Topical Report NEDO-20566, Vo. II, page III-33.

211.214 (6.3) NPSH considerations require clarification in the following areas:

a) Provide calculations or other evidence to show how the ECCS suction lines in the suppression pool are designed to prevent formation of vortices and air ingestion when the ECCS is in operation. Section 6.3.6 states that NPSH calculations, assuming the worst case passive failure in an ECCS pump

211.214 (6.3) and the subsequent drop in the suppression pool level, show adequate margin to assure proper pump operation. Justify the use of a minimum suppression pool level to prevent vortice formation versus providing mechanical vortex barriers for the ECCS suction lines in the suppression pool.

b) Section 6.3.2.8 of your FSAR states that "10 minutes following the accident, the operator is required to throttle the CS and LPCI pumps to rated CS and LPCI flow rate in order to ensure that adequate NPSH is available to the pumps". Evaluate the consequences of delaying the throttling action until 20 minutes after the accident. Provide manufacturer's pump test data which demonstrates the required NPSH for each ECCS pump.

c) Provide new Figures 6.3-3a and 6.3-6a referenced in the response to Question 211.77.

211.215 (6.3) Provide the minimum required capacity of the condensate storage tank and the suppression pool. Assuming no makeup to the CST or to the suppression pool and considerating NPSH requirements, provide the calculations which show how long the ECCS could operate under the worst conditions.

211.216 (6.3) Valves in the Safeguards systems are interlocked to minimize the potential for operational malfunctions (e.g., to ensure valving changes are performed in a proper sequence, and to ensure that two separate modes of equipment operation cannot occur simultaneously).

Present a tabulation of all electrical interlocks for all electrically-controlled pneumatically or hydraulically operated or motor operated valves of the systems that are shown on the FSAR figures listed below. The tabulation should:

1. Identify all other valves (by valve number and electrical division) that are interlocked with each valve shown in the listed figures.

2. List the required position (open, closed, or intermediate position) of these other valves that will permit motion of the valves shown on the listed figures.

3. List any permissives (interlocks) that each valve shown provides to any other valve(s) and to control circuits for pumps.

FSAR Figure	System Description
6.3-1	HPCI System
6.3-4	Core Spray System
5.4-13	RHR System

211.217 (6.3) Discuss the design provisions that permit manual override on the ECCS subsystems once they have received an ECCS initiation signal. Also include a discussion of any lockout devices or timers that prevent the operator from prematurely terminating ECCS functions. For example, if offsite power is not available, the operator must wait until the core is flooded, and then secure several of the ECCS pumps, to permit the manual starting of the RHR service water pumps without overloading the diesel generators. Discuss the design provision that permits the operator to shutdown these ECCS pumps after they have been automatically started.

211.218 (6.3) Provide piping isometric drawings that show the relative elevations and physical locations of the valves, suppression pool, primary containment, pumps, heat exchangers, and the lengths of piping for the entire ECCS. The locations and valve numbers of all valves should be shown on the isometric drawings. The valve nomenclature should be identical to that used on the P & ID's presented in the FSAR.

211.219 (6.3) Section 6.3, III, 24 of the Standard Review Plan recommends that periodically ECCS pumps and valves are to be operated (on normal and emergency power) to demonstrate that the system can respond to a LOCA. These tests are to be completed during plant operation. During refueling outages, the ECCS systems are tested to verify proper coolant flow to the reactor vessel. The FSAR indicates that "flow test" lines are provided for the CS, LPCI, and the HPCI systems, but the type, the duration, and the frequency of the testing is not clear. Provide additional information to specify the "periodic system surveillance" programs for each of the ECCS systems.

211.220 (6.3, 5.2.2) Your response to Question 211.70 requires additional clarification. Parameters such as environmental temperature, pressure ramp rates, operating pressure, solenoid voltage, and backpressure were varied consistent with test facility capabilities to establish safety relief valve service life. Provide assurance that the worst anticipated operating conditions were simulated in this test program.

In response to Question 211.70, you state that the accumulator capacity will provide air for one actuation while Section 5.2.2.4.1 states that the accumulator capacity is adequate for two actuations.

In 7.3.1.12.1.4.2, you state that a dual solenoid-operated pilot valve controls the pneumatic pressure applied to the "bellows actuator" which controls the safety/relief valve directly. In Figure 211.70-2, you show a cross-section of a Crosby valve which has a piston type pneumatic actuator. Also, in Table 211.70-1, you state that the SSES safety/relief valves have no pilot valves but in Section 7.3.1.1a.1.4.2, you state that the air accumulator is sized to provide air for five actuations of the pilot valve following a failure of the pneumatic supply. Resolve these discrepancies.

211.221 (6.3, 5.2.2) Recent event reports from operating BWRs have shown that multiple relief valve failures may occur from a common failure mode. Provide assurance that your relief valve design is qualified (including testing after being subjected to an environment representative of an extended time period at normal operating conditions) to support your assumption that 5 of the 6 ADS valves will operate. A history of safety/relief valve operation, including similar valves in other plants, should be included in this evaluation. Both satisfactory and unsatisfactory operation should be included, noted as the number of times the valve opened or failed to open; the number of times the valve closed or failed to close.

211.222 (6.3) Section 6.3.3.7.3 of the FSAR states that Figure 6.3-13 is a graphical representation of the break spectrum calculations presented in Table 6.3-3. Figure 6.3-13 is a graphical representation of lower plenum enthalpy versus time. Resolve this discrepancy. The title for Figure 6.3-13 incorrectly identifies the curve as core flow versus time for the 68% DBA recirculation discharge break. Correct the title of the figure.

Figure 6.3-31 appears to be mislabeled as a curve for a DBA recirculation "Discharge" break instead of a suction break. Correct the title of the figure.

211.223 (6.3) A timer is used in each ADS logic. The basis for the time delay before ADS actuation is to ensure that the HPCI system has time to operate, but yet short enough to ensure that the LPCI or the CS systems can adequately cool the fuel should HPCI fail to start. Manual reset circuits are provided for the ADS initiation signal and primary containment high pressure signals.

Discuss in detail any criteria to be given to the operator (e.g., emergency procedures or operator training) that would form the bases for the operator's decision to use the manual reset circuits to delay or prevent ADS actuation.

211.224 (6.3, 5.2.2) Section 5.2.2.4.2.1 states that cyclic testing has demonstrated that the safety/relief valves are capable of at least 60 actuation cycles between required maintenance. Are the actuations of the safety/relief valves recorded? If so, how are these data recorded and reported to the NRC?

211.225 (6.3) Initiation of the HPCI system automatically occurs for a "low water level". Table 6.3-2 of your FSAR indicates that this occurs at or less than 131.6 inches above the top of the active fuel. Figure 5.3-2 indicates that the "low water level" initiation of HPCI occurs at level, L2 or 123.2 inches above the top of the active fuel. Resolve this inconsistency.

211.226 (6.3) Provide data to verify that representative HPCI active components (in particular, the pump) have been "proof-tested" under the most severe operating conditions that are anticipated. The service life and the maximum expected operating time accumulated during the service life of that HPCI pump should be specified.

211.227 (5.4.6) Provide the trip settings and setpoint ranges for the RCIC system isolation instrumentation. Indicate the method of specification of these settings and the provisions for minimizing the potential for inadvertent isolation of RCIC.

211.228 (6.3) The response to Questions 211.13 and 211.105 require additional clarification. Reference was made to another BWR/4 with LPCI modification (Shoreham) and the results of an analysis for LPCI diversion at Shoreham was identified as applicable to Susquehanna. Does Susquehanna have an interlock similar to that at Shoreham which would prevent LPCI diversion prior to reflooding the reactor core to the 2/3 level? If not, justify the use of the Shoreham analysis for LPCI diversion at Susquehanna.

Describe operator requirements to activate LPCI diversion. Can the diverted LPCI loop be returned to provide additional core flooding, if required? What instructions, if any, are provided to the operator to ensure that the operable LPCI loop is not prematurely diverted to containment cooling in the event that one LPCI loop is disabled?

211.229 (6.3) The LPCI head flow characteristics shown in Figure 6.3-7 are incomplete. Provide horsepower, NPSH, and other normal pump characteristics.

211.230 (6.3) Section 7.3.1.1a.1.4.11.2 states that ADS safety/relief valve operability will be monitored by a temperature element installed on the valve discharge piping. Operating experience has shown that a "false" temperature increase may be indicated even though the valve has not operated. Justify use of the temperature element over a direct valve position indication to assure safety/relief valve operability.

211.231 (6.3) The flow rate from each core spray loop is stated to be 6,350 gpm in Figure 6.3-5 and Table 1.3-3. Table 6.3-2 and Table 6.2-2 state that the flow rate per core spray loop is 6,250 gpm. Resolve this discrepancy.

211.232 (6.3) Provide assurances that the pre-operational and initial startup test programs outlined in Section 14.2.12.1 and 14.2.12.2 conform to Regulatory Guide 1.68. The statement that "The system performance characteristics are in accordance with applicable design documents" is not acceptable. Compliance with the criteria outlined in Appendix A of Regulatory Guide 1.68 is not readily apparent. No pre-operational or initial startup test programs for the LPCI (RHR) system were found in the FSAR.

211.233 (6.3) Section 6.3.2.9 of the FSAR refers to Table 6.3-9 for a listing of all manual ECCS valves and the methods for assuring correct valve position. Provide Table 6.3-9.

211.234 (6.3) The low pressure systems of the ECCS are provided with relief valves to prevent the components and piping from inadvertent overpressurization. Provide justification to support the relief valve capabilities and setpoints that are stated in the FSAR for the Core Spray and Low Pressure Coolant Injection system. The isolation pressure for the low pressure systems should be included in this discussion.

211.235 (6.3) Tables 6.3-1 and 6.3-2 do not agree on the time delay between initiation signal to HPCI injection valve opening and the HPCI pump at rated flow. Table 6.3-1 states this delay is 35 seconds while Table 6.3-2 states that the delay is 30 seconds. Resolve this discrepancy. Provide the basis for the time delay before the HPCI pump is at rated flow.

211.236 (6.3) The answers to Questions 211.10 and 211.104 are incomplete. The leak detection system has been described in generalities, but the maximum leak rate and the allowable time for operator action have not been identified. Provide a scenario for the response of the leak detection system and the operator response for the maximum anticipated leak rate. Included in this scenario should be quantitative values for the leak rate and the response times.

211.237 (6.3) Section 6.3.3.2 states that conformance to criterion 3 of 10 CFR 50.46, Maximum Hydrogen Generation, is shown in Table 6.3-4. However, Table 6.3-6 shows oxidation fraction versus PCT and MAPLHGR. Provide the maximum hydrogen generation for these conditions.

211.238 (5.2.5) Discuss what monitors are available to identify the source of leakage between such components as the pump seals, valve stem packing, and the equipment warming drains and all other compartment sources drained to the drywell equipment drain tank.

211.239 (5.2.5) With respect to leak detection, provide the following additional information:

1. What is the quantitative relationship between the drainage flow and sump level to the leakage rate from any source?

2. Provide assurance that all leakage within the drywell and reactor building will flow directly to the sumps and that there are no reservoirs which must be filled before any sump drain flow occurs.

3. Provide a schematic of the drywell and the drywell area showing the locations and elevations of leakage detection instrumentation.

4. Are all the components in the leakage detection system qualified for the post-LOCA environment long-term cooling mode of the ECCS?

211.240 (5.2.5) On page 5.2-49, in Subsection 5.2.5.1.2.4.1, you state "The drywell equipment drain tank is equipped with two (2) 50 gpm tranfer pumps. Either one of these pumps will be capable of preventing the drywell equipment drain tank from overflowing to the drywell floor drain sump during conditions of acceptable identified leakage rates." State quantitatively what constitutes acceptable identified leakage rates and discuss the consequences of exceeding these rates.

211.241 (5.2.5) In Subsections 5.2.5.2, on page 5.2-54, you state:

(a) "The recirculation valve packing leakoff connections are piped to the drywell equipment drain through normally closed isolation valves." Show this on P & I diagram M-143.

(b) "The main steam isolation valve packing leakoff piping is provided with a normally closed isolation valve, and is capped." "Keeping these leakoff's isolated provides two sets of packing for limiting steam leakage." Estimate the increase in unidentified leakage as a result of the above feature.

211.242 (5.2.5) Sections III.7 and III.8 of Standard Review Plan 5.2.5 state that:

(1) The control room operators shall have a chart or graph that permits rapid conversion of count rate into gpm, that the conversion procedures shall take into account the isotope being monitored and the activity of the primary coolant, and that the plant will maintain a running record of background leakage, so that its effect may be subtracted from any sudden increases in leak detection, which may be "unidentified" leakage and require prompt action. If monitoring is computerized, backup procedures should be available to the operator.

(2) The radiation monitoring systems shall have a radioactive source built into the system to permit system test and calibration during operation, and that the flow of "identified" leakage, which may amount to as little as .05 gpm or as much as 0.25 gpm representing a total daily flow of between 72 and 360 gallons, will be used to provide an operability check during operation for the sump monitoring systems and the containment air cooler condensate flow monitors. The directly measured quantity of flow thus obtained from the sump and air cooler monitors can be used to calibrate the radiation monitoring systems.

Provide verification that the leak detection systems comply with the above requirements. Include a list of all indications available to the control room operator for evaluating and detecting unidentified leakage of concern. Show how the operator will determine the amount

of leakage by observing the indications available to him and how he will maintain a record of background leakage. In addition, discuss the procedures used by the operator to convert all leak detection indications in the control room to a common leakage equivalent; e.g., gpm.

211.243 (5.2.5) Section 5.2.5.1.2.3 states that radioactivity monitor alarm setpoints will be set significantly above background to prevent nuisance alarms. Provide an indication of how high above background these alarms will be set and an indication of what size leak these monitor alarms would detect assuming the sump level monitor fails to alarm.

211.244 (5.4.6) Confirm that the RCIC electro-hydraulic system integrated with the turbine governing valve is of safety Class 2, and Seismic Category I design.

211.245 (5.4.6) The ASME Boiler and Pressure Vessel Codes, Section III, Article NB-7000 requires that individual pressure relief devices be installed to protect lines and components that can be isolated from normal system overpressure protection. With reference to the appropriate P &ID, discuss compliance with the above code for the RCIC pump discharge line.

211.246 (7.4.1, 5.4.6) In Subsection 7.4.1.1.3.1, you state that one of the two testable check valves on the pump discharge line is located inside the drywell. According to P & ID Diagrams M-149 and M-141, the RCIC pump discharge line connects to the feedwater line outside the drywell. Please explain the above inconsistency.

211.247 (5.4.7) Some relief valve discharge lines (e.g., for RHR system) penetrate primary containment and have outlets below the surface of the suppresssion pool. Since these lines form part of the primary containment, the concern is that excessive dynamic loads during relief valve actuation may cause line cracking or rupture. Identify these lines penetrating containment and provide information concerning measures taken to prevent line damage. Of particular concern in this regard are water slugs in lines discharging steam (e.g., RHR heat exchangers). Such water slugs would be drawn up from the suppression pool as the result of low pressures with steam condensation or result from inadequate draining of low points.

211.248 (5.4.7) In the evaluation of the turbine trip transients, 0.10 second is assumed for full-stroke closure time of the turbine stop valve. Demonstrate that turbine stop valve closure times smaller than 0.10 second do not result in unacceptable increases in MCPR and reactor peak pressure or provide either (1) justification that smaller closure time cannot occur or (2) a minimim closure time to be incorporated in the Technical Specifications.

211.249 (5.4.7) Provide results of an analysis to demonstrate that no single failure will result in overpressurization of the RHR system. Provide the design basis used to determine the capacity of the relief valves of the RHR system.

211.250 (5.4.7) Complete the RHR process flow diagram in Figure 5.4-14 to include the pressures at various locations in the sytem. For example, the service water outlet and primary coolant inlet pressures at the RHR heat exchanger are required in the assessment of the provisions to monitor heat exchanger tube leakage.

211.251 (5.4.7) In Hatch 2, it is possible to discharge reactor coolant into the suppression pool when performing shutdown cooling with the RHR system. Identify any modes of equipment operation that permit such a flow path in Susquehanna. If such paths exist, what design features or instrumentation will be used to monitor this flow?

211.252 (5.4.7) On previous occasions, leakage of steam past valves in the steam supply lines to RHR heat exchangers has resulted in steam bubble formation and the occurrence of damaging water hammer following startup of the RHR pumps. Describe the provisions (e.g., sensors with alarms) and procedures to be used for Susquehanna to prevent such an occurrence due either to leakage or inadvertent valve opening.

211.253 (5.4.7) Discuss the procedures for minimizing the potential for exceeding the allowable cooldown rate (greater than 100 degrees Fahrenheit/hour) of the RHR and the reactor coolant system when placing the plant in a shutdown cooling mode following planned normal conditions or an emergency.

211.254 (15.0) Specify and justify your selection of the core burnup that yields the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution which was used in the analytical model for all transients analyzed.

211.255 (15.4.5) Explain why the transient resulting from recirculating flow control failure with increasing flow is most severe at the low end of the rated flow control line, specifically at 65% NB rated power and 50% rated core flow.

211.256 (15.4.5) Explain how Event 11 in Recirculation Loop Flow Control Failure to Maximum Demand (in Appendix 15A) is a planned operation in state C and also in state D with mode switch not in run. (Figure 15A.6-11).

211.257 (15.4.5) In reference to Figure 15.4-7:

a) Explain why the level curves do not show identical traces. Are the traces in percent of level instead of in inches? (This also applies to Figure 15.4-6).

b) Explain why diffuser flow #2, decreases between time T = 2 and T = 3 seconds.

c) Explain why the curves do not show an L-8 trip at approximately 36 seconds, when the narrow range level reaches the trip setpoint. Also explain why Table 15.4-4 indicates an L-8 trip does not occur until 50+ seconds.

d) Explain why vessel steam flow, and turbine steam flow, increase between T = 20 seconds and T = 27 seconds.

211.258 (15.1.3.3.2) The assumed pressure regulator failure at 115% NBR steam flow appears low compared to a failure value of 130% NBR steam flow used in other plant safety analyses. Explain the basis for the assumed pressure regulator failure at 115% NBR steam flow.

211.259 (15.0) Table 15.0-2 does not contain all of the input parameters used in the REDY computer code. For the transients analyzed in Chapter 15.0, provide the following:

a) A list of all input parameters for each transient.

b) Justification that these input parameters for each transient are suitable.

211.260 (4.6) Identify the Failure Mode and Effect Analysis for evaluating the control rod drive system which you state is provided in Appendix 15A.

211.261 (15.0) GE calculations performed for decrease in reactor coolant temperature (Section 15.1) and for reactor pressure increase (Section 15.2) events using the proposed ODYN licensing basis model (NEDO-24154) have shown that in some cases a more limiting CPR is predicted than by the current REDY licensing bases model (NEDO-10802). Based on a letter to Glen C. Sherwood dated 1/23/80 from Richard P. Denise, the staff's ODYN licensing position is that GE can proceed with ODYN analysis of transients described in Chapter 15 of licensing application Safety Analysis Reports. Provide an ODYN analysis of the applicable events listed in Tables 2-1 and 2-2 of NEDO-24154-P.

211.262 (15.3.3.2.2) For the "recirculation pump seizure" accident, coincident loss of offsite power is not simulated with the assumed turbine trip and coastdown of the undamaged pump. Reanalyze this transient assuming coincident loss of offsite power and incorporate this reanalysis with that previously requested in Q211.120.

211.263 (6.3) Table 6.3-3 specifies that the limiting break is a 1.5 square foot (0.80 DBA) break in a recirculation discharge pipe with a peak cladding temperature of 1874°F. The same peak cladding temperature (1874°F) is shown for the 0.68 DBA case shown in table 6.3-6. Section 6.3.3.7.4 indicates that the most limiting case is obtained by combining the LAMB/SCAT results for the 0.80 DBA case with the SAFE/REFLOOD results for the 0.68 DBA case.

Explain the bases for selecting 0.8 DBA rather than a larger break for use in the LAMB/SCAT analysis since larger breaks generally descrease the time to boiling transition.

Are the values listed in table 6.3-3 for the 1.5 square foot recirculation break the results of combining the 0.68 and the 0.80 DBA results?

Discuss the reasons for the 0.68 DBA having the longest period for which the hot node is uncovered.

Provide curves to show the results of the 0.80 DBA analyses and curves of the composite analyses used to identify the limiting break.

211.264 (15.6.5) You state that the quantitative analyses of the spectrum of pipe breaks is covered in Section 6.2, 7.1, 7.3, 8.3, and Appendix 15A. However, most of the information provided applies only to the DBA line break.

Provide a list of the pipe size and break locations that were analyzed for LOCA inside containment.

211.265 ((15.0) In several transient and accident analyses (e.g., loss of offsite power, rod drop accident) RCIC is credited as the backup system to HPCI for providing initial core cooling. RCIC normally takes suction on the condensate storage tank (CST) but must be manually switched to the suppression pool should CST water be unavailable. Since the CST and its piping are not qualified structures, consideration must be given to a delay in the cooling function. What is the effect on the consequences for each event of a 20 minute delay in the switch-over of RCIC to the suppression pool assuming HPCI has been incapacitated by a single failure (see Q211.144).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

APP/3

MAR 0 3 1980

DISTRIBUTION:
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LWR #3 File
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M. Rushbrook

Docket No. 50-387 & 50-388

Mr. Norman W. Curtis
Vice President - Engineering
and Construction
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Subject: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

The following documents concerning our review of the subject facility are transmitted for your information:

- [ ] Notice of Receipt of Application.
- [ ] Draft/Final Environmental Statement, dated _____.
- [ ] Safety Evaluation, or Supplement No. _____, dated _____.
- [ ] Notice of Hearing on Application for Construction Permit.
- [ ] Notice of Consideration of Issuance of Facility Operating License.
- [ ] Application and Safety Analysis Report, Vol. _____.
- [ ] Amendment No. _____ to Application/SAR, dated _____.
- [ ] Construction Permit No. CPPR-_____, dated _____.
- [ ] Facility Operating License No. DPR-_____, NPF-_____, dated _____.
- [ ] Amendment No. _____ to CPPR-_____ or DRR-_____, dated _____.
- [X] Other: Order Setting Prehearing Conference
- [ ] _____

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

OFFICE ▶	LWR #3:LA	LWR #3:BC				
SURNAME ▶	MRushbrook/LLM	ODParr				
DATE ▶	3/ /80	3/ /80				

NRC-21 (6-76)

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Charles Bechhoefer, Chairman
Dr. Oscar H. Paris
Glenn O. Bright

In the Matter of	)	
	)	
PENNSYLVANIA POWER & LIGHT COMPANY	)	Docket Nos. 50-387
and	)	50-388
ALLEGHENY ELECTRIC COOPERATIVE, INC.	)	
	)	
(Susquehanna Steam Electric Station,	)	
Units 1 and 2)	)	

ORDER SETTING PREHEARING CONFERENCE
(February 22, 1980)

On February 4, 1980, the Applicants in this operating license proceeding filed three motions which, if granted, would result in the dismissal of Citizens Against Nuclear Dangers (CAND) from this proceeding and would limit the participation of the Environmental Coalition on Nuclear Power (ECNP) and Susquehanna Environmental Advocates (SEA) with respect to certain contentions. Given the severity of the sanctions requested and the effect on the proceeding which would ensue should we grant the motions in their entirety, we desire to hear oral argument on these motions.

PLEASE TAKE NOTICE that a prehearing conference will be held for this purpose, as well as to consider other matters (such as future scheduling) which may be pertinent to the course of this proceeding. (Parties that wish to raise other specific matters

are requested to advise us and other parties; such advice must be received by Monday, March 17, 1980.) All parties except Ms. Marsh are directed to appear; Ms. Marsh, as well as the Commonwealth of Pennsylvania, are welcome to do so if they wish. The conference will commence at 9:30 a.m. on March 20, 1980, in Courtroom No. 2, U. S. Federal Building and Courthouse, 197 South Main Street, Wilkes-Barre, PA 18701.

Following the conclusion of the formal business of the conference, the Board will hear oral limited appearance statements pursuant to 10 CFR §2.715(a). The Board will give preference to those who have heretofore requested the opportunity to make such a statement but, to the extent that time is available, will hear others who are present and wish to make statements. It is expected that statements will be received the afternoon of March 20. (Further opportunity to make statements will be offered at later sessions of this proceeding.) Those who wish to make oral statements are requested (if they have not already done so) to inform the Secretary of the Commission, ATTN: Docketing and Service Branch, U. S. Nuclear Regulatory Commission, Washington, DC 20555.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Charles Bechhoefer

Charles Bechhoefer, Chairman

Dated at Bethesda, Maryland,
this 22nd day of February, 1980.