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Docket Nos. 50-387
and 50-388

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

NOV 22 1978

Dear Mr. Curtis:

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION UNIT 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 Auxiliary Systems. The specific information required is listed in the Enclosure.

Please inform us of the date when this requested additional information will be available for our review.

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

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

SUSQUEHANNA STEAM ELECTRIC STATION

DOCKET NOS. 50-387 AND 50-388

010.0 AUXILIARY SYSTEMS BRANCH

010.6 (RSP) (6.7) A single failure of an inboard MSLIV would allow a continuous blowdown of the containment atmosphere to the reactor building standby gas treatment system for a specified period of time when the MSIVLCS is initially actuated. This violates our containment isolation criteria and the consequences of the blowdown are unacceptable. It is our position that an interlock be provided so that the leakage control system actuation valves can be opened only if the associated inboard MSLIV is in a fully closed position. Revise the FSAR to indicate conformance to our position.

010.7 (RSP) (6.7) The design criteria for the main steam isolation valve leakage control system indicates that you propose to allow a main steam isolation valve (MSIV) leakage rate up to 100 SCFH for each MSIV in each steamline. It is our position that the design basis leak rate of 100 SCFH is not an acceptable MSIV leakage rate for normal operation. Therefore, we will still impose a technical specification limit of 11.5 SCFH for the MSIV leak rate and a leak rate verification testing frequency consistent with the plant Technical Specifications used for other operating BWR's. Revise the FSAR to indicate that the MSIV leak rate for normal operation will be limited to 11.5 SCFH.

010.8 (RSP) (9.1.1) Confirm that a Keff of less than 0.98 will be maintained with fuel of the highest anticipated reactivity in place in the new fuel storage racks and assuming optimum moderation.

010.9 (RSP) (9.1.2) The information contained in the Susquehanna FSAR is not of sufficient detail to support a conclusion that the liner plate for the spent fuel pool is designed to seismic category I. Therefore, we require, that you demonstrate compliance with Regulatory Guides 1.13 and 1.29 by showing that a failure of the liner plate as a result of an SSE will not affect any of the following: significant release of radioactive materials due to mechanical damage to the spent fuel; significant loss of water from the pool which could uncover the fuel and lead to release of radioactivity due to heat-up; loss of ability to cool the fuel due to flow blockage caused by a portion or one complete section of the liner plate falling on top of the fuel racks; damage to safety related equipment as a result of pool leakage; or uncontrolled release of significant quantities of radioactive fluids to the environs.

010.10 (RSP) (9.1.2) Confirm that all portions of the structure (reactor building) which serve as a low leakage barrier to provide atmospheric isolation of the spent fuel storage pool and associated fuel handling area are designed to seismic Category I criteria.

010.11 (RSP) (9.1.3) The spent fuel pool cooling system is a non-seismic system. This does not meet the guidelines set forth in Regulatory Guide 1.13 and 1.29. Analyze the design of the spent fuel pool cooling system to show that the pumps and piping are supported so that they are capable of withstanding an SSE, or provide the results of an analysis to show that for the complete loss of fuel pool cooling that would result in pool boiling, a release of significant quantities of radioactivity to the environment will not result.

010.12 (9.1.3) Confirm that a spent fuel pool water temperature of 125°F is maintained when the fuel pool cooling system is used to cool the emergency heat load.

010.13 (9.1.3) Based on information provided in your FSAR, it appears that either the spent fuel pool is capable of storing over 2 1/2 full cores or has a storage capacity for 4 1/2 full cores. State the design bases storage capacity provided for the spent fuel pool.

010.14 (9.1.3) The decay heat during normal (1/4 full core load, plus previous refueling loads) storage conditions has not been provided. Assuming that fuel assemblies from 1/4 of a full core load are placed in the pool 7 days after reactor shutdown and the remaining storage spaces are filled with spent fuel from previous refuelings, reevaluate the spent fuel pool cooling system's and the residual heat removal system's capability using the heat loads determined by the methods set forth in

Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." Also, reevaluate the systems capability for the emergency (1 full core unloaded from the reactor 7 days after shutdown plus the normal refueling load that has been in the pool for 30 days) storage condition. For both the normal and emergency storage condition stage the maximum decay heat load, the maximum spent fuel pool temperature, and provide the time required to raise the temperature of the pool to boiling assuming the cooling systems are not available.

010.15 Our criteria for safety related cooling systems is that sufficient cooling must be provided for at least 30 days: (1) to permit simultaneous safe shutdown and cooldown of both nuclear reactor units and maintain them is a safe shutdown condition, or (2) to mitigate the consequences of an accident in one unit and a safe shutdown and cooldown in the other unit and maintain it in a safe shutdown condition. Expand table 9.2-5 in the FSAR to cover this 30 days time span.

010.16 (9.2.5) The emergency service water system (ESWS) is designed to take water from the spray pond and provide cooling to safety related components during safe shutdown and accident conditions. During safe shutdown or the loss of offsite power non-safety related components are cooled by the ESWS. Demonstrate that the safety function of the system will not be affected assuming a failure in the non-safety related portion of the system coincident with a single failure in the safety related portion of the system. Also, provide an evaluation of the effects of flooding on safety related components.

010.17 (9.2.5) The reactor building closed cooling water (RBCCW) heat exchanges and turbine building closed cooling water (TBCCW) heat exchanges are not designed to seismic Category I requirements. However, these components are cooled by the safety related emergency service water system and isolated by a single isolation valve. This does not meet the single failure criterion. Revise your design to meet single failure.

010.18 (RSP) (9.2.7) In order to permit an assessment of the Ultimate Heat Sink, provide the results of an analysis of the thirty-day period following a design basis accident in one unit and a normal shutdown and cooldown in the remaining unit, that determines the total heat rejected, the sensible heat rejected, the station auxiliary system heat rejected, and the decay heat release from the reactors.

In submitting the results of the analysis requested, include the following information in both tabular and graphical presentations:

(1) The total integrated decay heat.

(2) The heat rejection rate and integrated heat rejected by the station auxiliary systems, including all operating pumps, ventilation equipment, diesels, spent fuel pool makeup, and other heat sources for both units.

(3) The heat rejection rate and integrated heat rejected due to the sensible heat removed from containment and the primary system.

(4) The total integrated heat rejected due to the above.

(5) The maximum allowable inlet water temperature taking into account the rate at which the heat energy must be removed, cooling water flow rate, and the capabilities of the respective heat exchangers.

(6) The required and available NPSH to the Emergency and RHR service water pumps at the minimum Ultimate Heat Sink water level.

The above analysis, including pertinent backup information, is required to demonstrate the capability to provide adequate water inventory and provide sufficient heat dissipation to limit essential cooling water operating temperatures within the design ranges of system components.

Use the methods set forth in Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," to establish the input due to fission produce decay and heavy element decay. Assume an initial cooling water temperature based on the most adverse conditions for normal operation.

010.19 (9.2.7) Sufficient information is not available for us to evaluate the plant safe shutdown capabilities from internal flooding of the engineered safeguard service water pumphouse. For a moderate energy leakage crack in the residual heat removal service water system piping or the emergency service water system piping, determine the effects of flooding on the safety related pumps located within the pump cubicle assuming 30 min. for any operator action. Also, describe any communication pathways between service water system pumps cubicles for loops A and loop B.

010.20 (9.2.12) The Reactor Building chilled water system is designed seismic Category I from the isolation valve outside containment to piping just inside containment. Figure 9.2-13B in the FSAR does not show any safety related valving inside containment for system isolation. The system and components inside containment and outside the containment penetrations are not seismically designed. The rupture of these non-seismic lines, plus a single active failure of the isolation value outside containment would cause a breech of containment. Provide the required isolation valves inside containment.

010.21 (10.4.4) Your FSAR does not evaluate the effects of an expansion joint failure at the condenser. Expand the information provided to include an evaluation regarding the effects of possible circulating water system failure inside the turbine building. Include the following:

(1) The maximum flow rate through a completely failed expansion joint.

(2) The potential for and the means provided to detect a failure in the circulating water transport system barrier such as the rubber expansion joints. Include the design and operating pressures of the various portions of the transport system barrier and their relation to the pressures which could exist during malfunctions and failures in the system (rapid valve closure).

(3) The time required to stop the circulating water flow (time zero being the instant of failure) including all inherent delays such as operator reaction time, drop out times of the control circuitry and coastdown time.

(4) For each postulated failure in the circulating water transport system barrier give the rate of rise of water in the associated spaces and total height of the water when the circulating water flow has been stopped or overflows to site grade.

(5) For each flooded space provide a discussion, with the aid of drawings if necessary, of the protective barrier provided for all essential systems that could become affected as a result of flooding. Include a discussion of the consideration given to passageways, pipe chases and/or the cableways joining the flooded space to the spaces containing safety related system components. Discuss the effect of the flood water on all submerged essential electrical systems and components.