QUESTION 010.1

The criteria for your high energy and moderate energy line analysis in the FSAR is in accordance with Branch Technical Position APCSB 3-1, "Protection Against Piping Failures in Fluid Systems Outside Containment." However, other than Tables 3.6-2 and 3.6-3, the results of your analyses and the environmental effects regulating from high energy line breaks and leakage cracks have not been provided. Provide these analyses and results for each of the assumed breaks or leakage cracks at their postulated locations.

RESPONSE:

The requested information has been provided in Appendix 3.6A.

QUESTION 010.2

We require that the compartment between the containment and the reactor building which houses the main steam lines and feedwater lines and the isolation valves for those lines, be designed to consider the environmental effects (pressure, temperature, humidity) and potential flooding consequences from an assumed crack, equivalent to the flow area of a single ended pipe rupture in these lines. We require that essential equipment located within the compartment, including the main steam isolation and feedwater valves and their operators be capable of operating in the environment resulting from the above crack. We also will require that if this assumed crack could cause the structural failure of this compartment, then the failure should not jeopardize the safe shutdown of the plant. In addition, we require that the remaining portion of the pipe in the tunnel between the reactor building and the turbine building meet the guidelines of Branch Technical Position APCSB 3-1.

We require that you submit a subcompartment pressure analysis to confirm that the design of both areas conforms to our position as outlined above.

We request that you evaluate the design against this staff position, and advise us as to the outcome of your review, including any design changes which may be required. The evaluation should include a verification that the methods used to calculate the pressure build-up in the subcompartments outside of the containment for postulated breaks are the same as those used for subcompartments inside the containment. Also, the allowance for structural design margins (pressure) should be the same. If different methods are used, justify that your method provides adequate design margins and identify the margins that are available. When you submit the results of your evaluation, identify the computer codes used, the assumptions used for mass and energy release rates, and sufficient design data so that we may perform independent calculations.

RESPONSE:

The requested information is provided in Appendix 3.6A.

QUESTION 010.3

The peak pressures and temperatures resulting from the postulated break of a high energy pipe located in compartments or buildings is dependent on the mass and energy flows during the time of the break. You have not provided the information necessary to determine what terminates the blowdown or to determine the length of time blowdown exists. For each pipe break or leakage crack analyzed, provide the total blowdown time and the mechanism used to terminate or limit the blowdown time of flow so that the environmental effects will not affect safe shutdown of the facility.

RESPONSE:

For those pipe breaks analyzed, termination of blowdown was not a controlling factor in the analysis since the temperature and pressure peaked within the first few seconds after the line break. Short term blowdown in these cases does not result in higher temperatures and pressure. Termination of the blowdown for breaks outside containment is accomplished by an automatic isolation signal from the Leak Detection System described in Subsection 7.6.1a.4.

QUESTION 010.4

The design criteria for the main steam isolation valve leakage control system (MSIVLCS) does not contain provisions to prevent the operation of the MSIVLCS when the inboard MSIV fails to close. We will require that an additional interlock be provided on the main steam isolation valve leakage control system so that the operation of an inboard leakage control system is prevented should an inboard main steam line isolation valve fail to be in its fully closed position.

RESPONSE:

The system design basis considers that there will be appreciable hold up time following the design basis LOCA before fission products from the core are transported down the main steam lines. The leakage control system is designed so that if the inboard system is actuated with one inboard MSIV failed open, the vent line from that steam line will automatically reclose by the time fission products, assuming plug flow, move down the steam line about 1/2 the total pipe run from the reactor vessel to the failed open inboard MSIV. Further, the operator is afforded with information from the control room regarding the status of the MSIV's through the valve position switches. Operating procedure inhibits the operator in activating the system in the event the valve position indicator shows failed open. Therefore, it is concluded that the additional interlock is not warranted.

NOTE:

MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7).

QUESTION 010.5

You state that the Unit 1 facilities reactor building crane is a single failure proof crane and is designed to handle the spent fuel cask, and that the Unit 2 crane is not single failure-proof and is designed to handle all normal plant operation loads except the spent fuel cask. Provide the following information for these fuel handling systems:

(1) Describe the normal plant operation loads that the Unit 2 reactor building crane is capable of carrying in the fuel building area.

(2) Describe the means used to prevent the Unit 2 reactor building crane from handling the spent fuel cask when stored in the spent fuel shipping cask storage pool.

(3) Describe the mechanical stops and/or electrical interlocks that would restrict the path of the 125-tone crane to those areas identified on Figure 9.1-16A and 9.1-16B.

(4) State whether the Unit 1 reactor building crane has been designed to meet the guidelines of Branch Technical Position ASB 9-1, "Overhead Handling Systems for Nuclear Power Plants."

RESPONSE:

1) Please see revised Subsection 9.1.5 for this information.

2) Please see revised Subsection 9.1.5 for this information.

3) Please see revised Subsection 9.1.5.3 for this information.

4) See response to Question 010.25.

QUESTION 010.6

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.

RESPONSE:

The system design basis from the outset has been that there will be appreciable hold up time following the design basis LOCA before fission products from the core are transported down the main steam lines. The leakage control system is designed so that if the inboard system is actuated with one inboard MSIV failed open, the vent line from that steam line will automatically reclose by the time fission products, assuming plug flow, move down the steam line about 1/2 the total pipe run from the reactor vessel to the failed open inboard MSIV. Besides, the operator is afforded with information from the control room regarding the status of the MSIV's through the valve position switches. Operating procedures inhibit the operator in activating the system in the event the valve position indicators show failed open. We conclude that the additional interlock is not warranted.

NOTE:

MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7).

QUESTION 010.7

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.

RESPONSE:

It is stated in Section 6.7.1.3 of the FSAR that the main steam isolation valve leakage control system (MSIV-LCS) is designed to process MSIV leakage rates up to 100 SCFH for each MSIV in each line. This is a design basis for the MSIV-LSC and is not the design basis leakage rate for the MSIV's. The Standard Technical Specification in Chapter 16 of the FSAR specifies the MSIV leakage rate at 11.5 scf per hour.

NOTE:

MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7).

QUESTION 010.8

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.

RESPONSE:

See revised FSAR Subsections 9.1.1.1.1.2, 9.1.1.2 and 9.1.1.3.1.

QUESTION 010.9

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.

RESPONSE:

See revised Subsections 9.1.2.1 and 9.1.2.2.

QUESTION 010.10

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.

RESPONSE:

See revised FSAR Subsection 9.1.2.2.

QUESTION 010.11:

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.

RESPONSE:

A complete analysis showing the amount of radioactive release following a complete loss of fuel pool cooling is provided in Appendix 9-A. As shown in Table 9A-1 the thyroid dose consequences of the boiling pool are well below the guideline values of 10CFR50.67 and the 0.5 REM TEDE thyroid guideline.

Subsection 9.1.2.2 provides the logic which shows that the spent fuel pool will not drain following an SSE.

QUESTION 010.12

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.

RESPONSE:

A spent fuel pool water temperature of 125°F is maintained when the fuel pool cooling and cleanup system (FPCCS) is used in conjunction with the RHR cooling system to cool the emergency heat load. Refer to revised section 9.1.3.1 for FPCCS design basis.

QUESTION 010.13:

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.

RESPONSE:

The storage capacity design basis for the spent fuel pool are provided in revised FSAR Section 9.1.2.1.3. Ambiguities upon this matter have been removed from the text.

QUESTION 010.14

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.

RESPONSE:

The decay heat during normal storage conditions and the maximum spent fuel pool temperature using the fuel pool cooling system only is given in Subsection 9.1.3.1.

For the normal storage conditions, the maximum decay heat load, maximum spent fuel pool temperature and time to boiling are provided in the dose release calculation of Appendix 9-A.

The emergency storage condition is discussed in Subsection 9.1.3.1.

QUESTION 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.

RESPONSE:

Table 9.2-5 has been revised to cover this 30 day time span.

QUESTION 010.16

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.

RESPONSE:

See revised FSAR Section 9.2.

QUESTION 010.17

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.

RESPONSE:

See revised FSAR Section 9.2.

QUESTION 010.18

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.

RESPONSE:

The results of the analysis requested are presented in graphical form in Figure 9.2-21*.

Part 5. The maximum allowable inlet water temperature is discussed in Subsection 9.2.7.3.1, 9.2.7.3.6, Table 9.2-12, and Figure 9.2-21; and the initial pond temperature is identified in Table 9.2-23 and Figure 9.2-21. The initial cooling water temperature is based on the most adverse meteorological conditions.

Part 6. Vertical pump suction requirements are normally measured in submergence. The bottom of the pond is higher than the minimum submergency required by the pumps (refer to Subsections 9.2.5.1, 9.2.6.1, and 2.4.11.5).

* Tabular results of the analysis and graphical and tabular heat load data for uprated conditions were not included in FSAR Revision 49 (power uprate).

QUESTION 010.19

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 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 cubiels for loops A and B.

RESPONSE:

Subsection 9.2.7.3 has been revised to include this information.

QUESTION 010.20

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.

RESPONSE:

Table 6.2-22 and Dwg. M-187, Sh. 2 of the FSAR have been modified to show the required isolation valves inside containment.

QUESTION 010.21

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 for 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.

RESPONSE:

For response see revised Subsection 10.4.1.3.

QUESTION 010.22

The Standard Review Plan, Section 9.1.4, part II provides guidelines for a fuel handling system. The fuel handling platform bridge and its subcomponents such as crane rails, clamps and clips are not excluded from a system's approach. They perform a safety related function, i.e. position fuel and limit the displacement of the fuel handling platform during a seismic event. Your FSAR Volume II, page 9.1-17 identifies the fuel handling platform as a safety Class 2, seismic category 1 piece of equipment and discusses its seismic requirements. There are no departures or exceptions identified in the FSAR.

On the basis of the above, please indicate your intent to upgrade the crane rails, rail clips and clip plates to meet the requirements of 10 CFR Part 50 Appendix B or provide your justification for retaining your present design. Your justification should include the consequences of the failure of these items or show that failure of any of these items will not result in damage to the fuel.

RESPONSE:

The crane rails, rail clips, and clamps have been classified as safety-related (see revised subsection 9.1.4.1).

QUESTION 010.23

Your response to Q010.1 and Q010.2 regarding pressure loadings in compartments within the reactor building does not address differential pressure loadings upon interior walls. While we think it unlikely that the exterior walls of the reactor building will be severely effected we remain concerned with the behavior of the interior walls. The impact of possible structural degradation of interior reactor building walls has implications with regards to safety related equipment. Your response should be expanded to support your conclusions that the differential pressure upon interior walls can be neglected.

RESPONSE:

The interior walls in the reactor building are designed to withstand without structural degradation, differential pressures arising from

1) Water flooding and

2) steam pressures from high energy line breaks as described in our response to Questions 010.1 and 010.2

Table 3.8-8 lists the loading combinational formulas used for Reactor Building interior partitions. The symbol "R" as defined in Table 3.8-2 incorporates the effects of differential pressure loads and is considered an abnormal load.

The room pressures given in the responses to Q010.1 and Q010.2 were incorporated in the interior wall loadings. Hence, interior walls will not fail due to room pressurization.

QUESTION 010.24

We have reviewed your response to Q010.6 and find it unacceptable. We will require an interlock so that the leakage control system actuation valves can be opened only if the associated MSLIV is in a fully closed position. (See Regulatory Guide 1.96.) Please modify your response accordingly.

RESPONSE:

Regulatory Guide 1.96 Rev. 1 has been interpreted to require backfitting only for BWR 6. As such this backfitting does not apply to SSES.

NOTE:

MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7).

QUESTION 010.25

Your response to Q010.5 for the cranes for Unit 1 and Unit 2 crane is unacceptable. The Unit 1 and 2 cranes do not meet Branch Technical Position APCSB 9-1.

Please reevaluate the design of your cranes in line with APCSB 9-1 and modify your response accordingly.

RESPONSE:

Our response to Q010.5 is now Appendix 9B of the FSAR. Four items are indicated as not complying with APCSB 9-1. These four items have been evaluated against NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," dated May 1979 and issued by the Division of Engineering Standards, Office of Standards Development, and are summarized below.

Further comments on our positions may be found in FSAR Section 3.13 under Regulatory Guide 1.104.

A. Fleet Angles

NUREG-0554 Section 4.1 allows fleet angles up to 3 1/2°. The reactor building crane fleet angle is 3° 7'. We agree with the NUREG that this angle has proven to be reliable in service and is an acceptable design.

B. 200% Static Test

NUREG-0554 section 8.2 recommends a 125% load test as does ANSI B.30.2. The RB crane is rated at 125 tons to yield a test weight of 156 tons. The maximum anticipated load is a spent fuel shipping cask weighing 100 tons. This results in a effective 156% static load test.

C. Bridge & Trolley Speeds

NUREG-0554 refers to C.M.A.A. Specification 70, Figure 70-6 for recommended bridge and trolley speeds. As noted above, the maximum lift weight anticipated is 100 tons although the crane's rated capacity is 125 tons.

From the CMAA figure, at 100 tons the suggested trolley slow speed is 50 fpm as is the suggested bridge slow speed. These numbers are identical to our maximum speeds.

D. Two Blocking

We reiterate our position stated in Appendix 9B, item #14; this test is unnecessary due to the protective devices installed, it is hazardous to testing personnel, and it has the potential of causing undetected damage to hoist components. Verification testing of the upper limit switches and the motor overloads will be performed to assure their proper functioning.

Summary

A review of the Branch Technical Position as well as other industry and NRC standards reveals no design feature or test procedure, that in our judgement, requires modification.

QUESTION 010.26

The potential for a partial density moderator (i.e. fire fighting foam) to reach the fresh fuel is a criticality concern. Provide details of the cover that protects the fresh fuel racks which show how effective the seals of the cover plates would be in preventing this unlikely event.

RESPONSE:

Details of the new fuel vault covers are shown in the new Figure 9.1-2. Design features are discussed in the new text added to Subsection 9.1.1.3.2.

Also, no fire fighting foam systems are installed anywhere in the plant.