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ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

CATEGORY DOCUMENT
F/INFO ACRS

Docket No. STN 50-447

APR 11 1974

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General Electric Company
ATTN: Mr. John A. Hinds, Manager
Safety and Licensing
175 Curtner Avenue
San Jose, California 95114

Gentlemen:

In order that we may continue our review of your GESSAR application, additional information on those matters set forth in the enclosure is needed.

The information requests are generally broken down into three areas. These are quality assurance, instrumentation, and reactor systems.

On March 20, 1974, we discussed with you the additional information we needed to complete our review of your quality assurance program. At your request, we are documenting our needs for additional information in the enclosure.

From meetings and discussions with you, it is apparent that the depth of information available on your new instrumentation design is less than that specified in the "Standard Format" for a construction permit. We understand that all the necessary information needed on your preliminary design will not be available for our review before January 1975. Accordingly, our Safety Evaluation Report (SER) will contain a discussion of the additional information needed to complete the review, together with a schedule for submittal of the information and completion of our review. Many of our enclosed questions in this area are, therefore, intended to obtain documentation of the design basis for the features of the design which will reduce the number of open items in our SER. In addition, with detailed responses to these requests, we expect that fewer questions will arise during our review of the preliminary design when it becomes available for review.

Based on the information you have to date, you should propose a schedule for submittal of the preliminary design information of the type discussed with you at our February 27, 1974 meeting. We understand that this schedule will be based on the information you have available at this time and that the proposed dates are somewhat variable.

GESSAR

As a result of many of the instrumentation design changes proposed in GESSAR, many of the transients and accidents analyzed in Chapter 15 and Appendix B will have to be reevaluated. This reevaluation will have to include such things as the impact, if any, of the prompt relief trip on the NSSS and unique analyses to support the new design. An example of the unique analyses is the evaluation of the consequences of the continuous withdrawal of the worst group of rods undergoing ganged motion. This event was not previously applicable to a BWR. A complete list of questions concerning those transients and accidents that need to be reviewed are enclosed.

To maintain our licensing review schedule, we will need a completely adequate response to the quality assurance requests by May 9, 1974, and the remainder of the information discussing instrumentation and reactor systems by May 17, 1974.

Please inform us within 7 days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests, it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the amount of extension will most likely be greater than the extent of delay in your response.

The questions in the enclosure have been grouped by sections that correspond to the relevant sections of GESSAR, and the question numbers continue consecutively from previous letters.

Please contact us if you desire additional discussion or clarification of the material requested.

Sincerely,

John F. Stolz, Chief
Light Water Reactors Project Branch 2-1
Directorate of Licensing

Enclosure:
GESSAR Request for Additional
Information

cc: Mr. W. Gilbert, Manager
Safety and Standards
General Electric Company
175 Curtner Avenue
San Jose, California 95114

Mr. L. Gifford, Manager
Regulatory Operations Unit
General Electric Company
4720 Montgomery Lane
Bethesda, Maryland 20014

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

GENERAL ELECTRIC COMPANY

GESSAR PLANT

DOCKET NO. STN 50-447

3.0 Design of Structures, Components, Equipment and Systems

3.168 In Table 3.2.1, the relief valve discharge piping from the anchor point to the suppression pool and classified as "Quality Group D and non-Seismic Category 1" is unacceptable. The Regulatory staff position is that the relief valve discharge piping from the relief valve to the suppression pool performs a safety function and should be classified Quality Group C, Quality Assurance Requirement B and designed to Seismic Category 1 requirements. Delete the reference to note (u) and delete this note on page 3.2-26.

3.169 Your comments in response to Question 3.130 regarding the classification of those portions of the Service Water System(s) which provide cooling for the reactor recirculation pumps in item XXVI of Table 3.2.1 are unacceptable.

The Regulatory staff position is that the cooling lines to the reactor recirculation pumps should be part of a cooling water system which is classified Safety Class 3, Quality Group Classification C, Quality Assurance Requirement B, and Seismic Category 1 to be in conformance with Regulatory Guides 1.26 and 1.29, unless it can be shown that the consequences of recirculation pump cooling water failure are acceptable (i.e. MCHFR > 1.0) for 2 pump seizure.

3.170 In Table 3.2.2 Quality Group Classification C corresponds to Safety Class 3. Delete the reference to Quality Group Classification D for Safety Class 3.

3.171 The response to Question 3.59 states that the pipe whip protection requirements for GESSAR piping systems inside containment will conform to the intent of AEC Reg Guide 1.46, "Protection Against Pipe Whip Inside Containment". However, the criteria for postulating break locations and types of breaks which is discussed in Section 3.6 on page 3.6-2 of GESSAR does not conform to the intent of Regulatory Guide 1.46. Demonstrate that your criteria is at least as conservative as the criteria outlined in Regulatory Guide 1.46.

5.0 Reactor Coolant System and Connected Systems

5.36 The response to Question 5.20 is not acceptable since it does not address the Staff's primary concern, that of the undesirable performance history of abnormal occurrences involving malfunction of safety and safety relief valves on operation BWR's. It is the Staff's position that significantly better safety and safety/relief valve performance is required of GESSAR plants than that which has been experienced in the past with currently operating plants. Provide a detailed description of differences (improvements) between presently operating plants and proposed GESSAR plants in the following areas, together with full explanation of why the noted differences are expected to provide the required performance improvement.

5.36.1 Valve type and/or design. Compare the valve types to be used on GESSAR plants with those on currently operating plants. If these valves have not been selected, provide the design criteria for these valves that will result in better operating performance.

5.36.2 Specifications - What new methods are employed to ensure that valve specifications include required operation under actual postulated environment?

5.36.3 Quality Assurance - What new programs have been instituted to assure that valves are manufactured to specifications and will operate to specifications?

5.36.4 Valve Operability - What new valve testing and operability programs will be required of GESSAR plants?

5.37 The response to question 5.21 stated in part that, "Destructive motor overspeed can be prevented by use of a decoupling device in the shaft between pump and motor." Specify whether decoupling devices will or will not be installed on GESSAR plants.

5.38 The response to question 5.23 is not acceptable in that it does not address the staff's primary concern, that of the undesirable performance history of many types of motor operated valves on ECC and related systems. Because of abnormal occurrences involving malfunction of motor operated valves on operating BWR's, it is the staff's position that significantly better motor operated valve performance is required of GESSAR plants than that which has been experienced in the past with currently operating plants. For motor operated valves on ECC Systems (and related systems, whose operation is required for successful ECC operation), provide a detailed description of differences (improvements) between presently operating plants and proposed GESSAR plants in the following areas, together with full explanation of why the noted differences are

expected to provide the required performance improvement.

5.38.1 Valve type and/or design. Compare the valve types to be used on GESSAR plants with those on currently operating plants. If these valves have not been selected, provide the design criteria for these valves that will result in better operating performance.

5.38.2 Specifications - What new methods are employed to insure that valve specifications include required operation under actual postulated accident environment? Your design should meet IEEE-382.

5.38.3 Quality Assurance - What new programs have been instituted to assure that valves are manufactured to specifications and will operate to specifications?

5.38.4 Valve Operability - What new valve testing and operability programs will be required of GESSAR plants?

6.0 Engineered Safety Features

6.123 The response to Question 6.25 is unacceptable and is insufficient to allow a complete evaluation of the inadvertent actuation of the ECCS. The inadvertant start of an HPCS pump at power represents an abnormal operational transient resulting in a small moderator temperature decrease. Of additional concern is the inadvertant start of the HPCS or other large capacity emergency core cooling systems while the reactor vessel pressure is low enough to allow injection of large quantities of cold water resulting in a larger moderator temperature decrease and possibly thermal shock to the primary coolant system. Analyze the potential for, and consequences of such an event occurring during a cooldown or heatup. If credit is taken for any safety system to effect a safety action, provide the protection sequence diagram in Appendix B.

6.124 With regard to Question 6.38, please provide the specific requirements provided to the Architect Engineer by GE concerning criteria and guidelines used to insure that ECC systems are "independent within the confines of design basis considerations." Provide any criteria and guidelines provided to AE in addition to those in Section 3.12 with resepct to:

6.124.1 How much physical spacing is required for components of two "independent" systems?

6.124.2 What types of barriers are required to protect against non-independent behavior due to a fire affecting two "independent" systems?

6.124.3 What types of barriers or elevation differences are required to protect against non-independent behavior due to rising water from any source affecting two "independent" systems?

6.125 The response to question 6.41 on manual operator actions following a LOCA is not acceptable. Provide chronological lists of <u>all manual actions</u> that are to be performed by the operators following a LOCA until stable long term cooling is achieved. These lists should be for both large and small LOCA, and should indicate:

a. the actual, physical action taken by the operator, i.e., switch thrown, gauge checked, button pushed, etc., and the operators physical location necessary to perform the required action.

b. effect on the reactor systems of the action, i.e., the system or item of equipment turned on, turned off, or whose operating state is changed (power source changed, water source changed, water destination changed, etc.).

c. the information required by the operator to know when or if he should perform the operation, that is, what parameter must reach what level before the operation is required, and through what instrument does the operator obtain that information, where, in the control room or locally, is that instrument's readout physically located, and how is the information conveyed to the operator? (meter or graph position, audible or visible alarm, etc.).

Also include all of the information requested above for automatic operations which are to be verified by the operator, indicating what manual actions he is required to take if the automatic action is not properly executed.

7.0 Instrumentation and Control

7.12 The discussion of conformance to IEEE Std 317-1972 in Section 7.1.2.4 indicates that the requirements for electrical testing will not be applied to all types of electric penetrations. However, Section 3.8.6.2.2 indicates complete conformance with Regulatory Guide 1.63. It is our position that, even if certain types of penetrations are not required to be energized during a LOCA, the requirements for withstanding the effects of a LOCA without loss of mechanical integrity are applicable. Since electrical failure can lead to loss of mechanical integrity, exemption from electrical testing is not justifiable solely on the basis that the penetration is not required to be energized.

7.12.1 Identify specifically (e.g., by paragraph numbers) the electrical testing requirements from which exemption is requested.

7.12.2 Discuss your reasons for concluding that each such exemption is justifiable. This justification could be based on the technical reasons for concluding that electrical failures do not lead to loss of mechanical integrity or on a description of the means to be used to prevent (even with a single failure) the non-qualified penetrations being energized during a LOCA.

7.12.3 The term "100 Series Electric Penetrations" is undefined in Section 7.1.2.4. If there is some significance intended by the use of this term, define the information intended to be conveyed.

7.13 The response to Question 7.9 is not sufficiently detailed to permit an independent evaluation of the adequacy of the proposed design criteria. Expand the response to include the specific criteria to be followed in implementing the general criterion that the instruments' range and accuracy will be those required.

7.14 The description of the Reactor Trip System in Section 7.2.1.1.3.1 and Prompt Relief Trip System includes a discussion of the use of turbine first stage pressure as an input signal. We note that your analysis of conformance to IEEE Std 279-1971 includes a discussion of how turbine first stage pressure is "equivalent" or "corresponds" to reactor power. Discuss your reasons for concluding that use of a turbine first stage pressure signal meets the requirements of Section 4.3 of IEEE Std 279-1971 that "to the extent feasible and practical, protection system inputs shall be derived from signals that are direct measures of the desired variables." (emphasis added)

7.15 The discussion of the Turbine Stop Valve Closure and Turbine Control Valve Fast Closure scrams in Section 7.2.1.1.5.4 indicates that these scrams are "required to provide a satisfactory margin to core thermal-hydraulic limits."

It is also indicated that these scrams provide additional margin over the high-pressure scram to preclude overpressurizing the nuclear system, although the nuclear steam high-pressure scram, in conjunction with the pressure relief system is adequate.

Since these sensors (for the turbine stop valve closure and turbine control valve fast closure scrams) are located in a non-Category I structure, provide an analysis of the effects of total failure of these two scrams during a turbine/generator trip or an analysis of the adequacy of other protection system signals to provide a satisfactory margin to core thermal-hydraulic limits.

7.16 Section 7.3.1.1.1.1.2 discusses a feature in the HPCS system to automatically stop flow by closing the injection valve based on a measurement of high water level. Provide an analysis of the effects on safety of failure to fulfill this termination function.

7.16.1 If the effects are acceptable, discuss the reasons for including this feature in the design, i.e., what benefit justifies encumbering the design with the added complexity.

7.16.2 If the effects of failure to terminate injection are unacceptable, describe the design changes necessary to make the design of the termination function conform to IEEE Std 279-1971.

7.17 The description of the LPCI and LPCS instrumentation indicates the use of differential pressure transducers measuring the differential pressure across the four low pressure injection valves to fulfill the function of initiating opening of the valves when reactor vessel pressure has dropped to a value at which the pumps are capable of injecting water into the vessel. However, since one of the sensing lines for each transducer is connected to the injection system piping between the injection valve and the downstream check valve, it appears that the sensed differential pressure across the injection valve may not be representative of the actual pressure differential between the pump discharge pressure and reactor vessel pressure.

7.17.1 Discuss the benefits derived from including the function to delay opening the injection valves that justify encumbering the design with this added complexity.

7.17.2 Discuss the design basis on which the decision was made to connect one side of the transducer to the upstream side of the check valve rather than downstream of the check valve where a more direct measurement of the desired variable could be obtained.

7.18 The interface information that is required to complete the review of the instrumentation, control and electric power systems described or referenced in Chapters 7 and 8 is:

A. Scope of Design and/or Supply Responsibility

Identify (list) the specific systems or components which are not within the "nuclear island." For each item, state whether the applicant will be responsible for the design, supply, or both.

B. Criteria and Design Bases.

For each component identified in A. above, list the criteria and design bases which must be met by the applicant in order to support your conclusions regarding the adequacy of the "nuclear island."

The above information is needed for both direct electrical interfaces, e.g., turbine sensor inputs to the protection system, and indirect functional interfaces, e.g., instrumentation to control cooling and ventilation systems for rooms in which protection system equipment is located.

7.19 It is not apparent that the response to Question 7.3 addresses our position on single failures. Enclosure 1 provides background information to that position and restates the position. Discuss the extent to which the design will be in accordance with this position and, if appropriate, discuss your reasons for requesting exemptions from any part of the position.

7.20 Figure 7.3-12L illustrates an ADS logic design where the 120-seconds delay can be automatically reset by reactor vessel water level signal "A" or "E". The staff concludes that this is not in conformance with the requirements of Section 4.16 of IEEE Std 279-1971. Revise the design to eliminate this potential for automatic reset of the initiating signal or discuss your reasons for not conforming to the requirements of IEEE Std 279-1971.

7.21 Figure 7.3-12L also shows that manual (as well as automatic) initiation of ADS is subject to 120-seconds delay. However, on Page 7.3-58, it is stated that: "This delay timer is not provided for manual initiation since the operator will not initiate ADS until he determines it necessary without further delay." The staff's position, which is in agreement with the latter statement, is that this delay in manual initiation is unnecessary and it unduly compromises the independence between automatic and manual initiation. Revise the proposed design to eliminate the 120-seconds delay in manual initiation of ADS.

ENCLOSURE 1 TO SECTION 7

TECHNICAL POSITION ON THE APPLICATION OF THE SINGLE FAILURE CRITERION MANUALLY-OPERATED VALVES

Discussion

In the "Definitions and Explanations" section of Appendix A to 10 CFR Part 50, it is stated:

> "Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development."

Thus, Appendix A to 10 CFR Part 50 defines which system should withstand single failures of active components and which systems should withstand single failures, but not concurrent failures, of both active and passive components without loss of capability to perform their safety functions. The distinction between active components and passive components is made by the following definition in Regulatory Guide 1.48:

> "Active Pumps and Valves. Components that must perform a mechanical motion during the course of accomplishing a system safety function."

The difficulty which arises when both these definitions are considered together results from the lack of explicit direction in evaluating the relationship between the electric systems and the fluid systems. With regard to passive components in fluid systems, Appendix A does not presently set forth the conditions under which the failure of a passive component itself should be considered in designing the system against a single failure. In contrast, it is clearly intended that single failures of both active and passive components should be assumed in designing electric systems against a single failure. Furthermore, no distinction is made between whether such failures in electric systems affect an active component or a passive component in a fluid system. Where a single failure in an electric system can result in loss of capacity to perform a safety function, the effect on public safety must be evaluated. This is necessary regardless of whether the loss of safety function is caused by an active component failing to perform a requisite mechanical motion or by a passive component performing an unnecessary mechanical motion.

This position establishes the acceptability of disconnecting power to the electric systems of a fluid system component as a means of designing against a single failure. These provisions are based on the assumption that the component then is equivalent to a similar component that is not designed

for electric operation, e.g., a valve that can be opened or closed only by direct manual operation of the valve stem. They are also based on the assumption that no single failure can both restore power to the electric system and cause mechanical motion of the components served by the electric system. The validity of these assumptions should be verified when applying this position.

Position

The Staff generally takes the position that single failures of both active and passive components in the electric systems of valves and other fluid system components should be considered in designing against a single failure, even though the fluid system component may not be called upon to function in a given safety system operational sequence. Where it is determined that the effects of such failures are unacceptable, a design that conforms to the following is acceptable to the Staff.

(1) The requirements of General Design Criteria 17, 21, 35, 38, 41, 44, 54, 55, and 56 should be interpreted as requiring the associated system to be designed to perform the system safety function, assuming a single failure. Included in the single failures to be considered in meeting this requirement are failures of both active and passive components in electric systems, regardless of whether such failures affect active mechanical components or passive mechanical components.

(2) Where it is determined that failure of a single active or passive component in an electric system can cause mechanical motion of a passive component in a fluid system and this motion results in total loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the component. The plant technical specifications should include a list of all electrically-operated passive valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

(3) Electrically-operated valves which are classified as active valves, but which are manually-controlled should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless: a) electric power can be restored to the valves from the main control room, b) valve operation is not necessary for at least 10 minutes following occurrence of the event requiring such

operation, and c) it is demonstrated that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

(4) When the single failure criterion is satisfied by removal of electric power from passive valves or from active valves meeting the requirements of 3, above, the associated valves should have redundant position indication in the main control room and the position indication system should itself meet the single failure criterion.

(5) The phrase "electrically-operated valves" includes both valves operated directly by an electric device (e.g., a motor operated valve and a solenoid-operated valve) and those valves operated indirectly by an electric device (e.g., an air operated valve whose air supply is controlled by an electric solenoid valve).

8.0 Electric Power

8.14 Provide the following information with regard to ESF motors and motors in supporting safety systems (reference Section 8.3)

8.14.1 Specify the minimum voltage required at each motor to successfully accelerate its pump load within the required period. Correlate these requirements with the recommendations contained in Regulatory Guide 1.9 for sequencing safety system loads on an onsite power system.

8.14.2 State the minimum margin of motor torque allowed over the pump load torque during the accelerating period of the pump load, and provide criteria for selecting this minimum value.

8.14.3 Describe the temporary or permanent instrumentation design features provided to monitor the temperature in large horsepower motor components.

8.14.4 Provide an outline of your initial and periodic tests that verify the motor's capability to accelerate its pump load within the number of starts prescribed by NEMA-MG-1 without incurring a temperature excursion in excess of design rating.

15.0 Accident Analyses

15.49 The response to Question 15.4 is not acceptable and is insufficient to allow an adequate evaluation. General Electric's bases for the acceptability of the consequences of transient and accidents on their nuclear system are stated in Section 15.1. The classification of events in this response does not agree with the classification of transients and accidents in either Appendix B of GESSAR or earlier product line GE reactors. The concern is that General Electric does not have consistent ground rules upon which they are assessing the adequacy of their system designs to mitigate the consequences of transient and accidents. What unacceptable results have been adopted by General Electric that are not identified in Appendix B for the events listed in Part a.3 of the Response? Why did General Electric not place them in the "Special Events" category of Appendix B and, therefore, perhaps applicable to other ground rules than transients and accidents? Why did General Electric not list the complete "Accident" category in their response?

Our position is that we will evaluate the events in Part a.3 of the response against the unacceptable results for transients listed on page 15.1-2.

15.50 The response to Question 15.5 is not acceptable and is insufficient to allow an adequate evaluation. A revision to GESSAR changing the ground rules upon which the analyses in Appendix B were based (see revised page 15.1-2) indicates that there are some transients for which the containment is required to mitigate their consequences. Identify these transients and explain this change in ground rules from earlier plants (Pilgrim and Browns Ferry).

15.51 The response to Question 15.7.4 is insufficient to allow an adequate evaluation. Explain the change in scram specifications without revising the analysis. Justify that these scram characteristics are conservative, relative to the assumed axial power shape, and discuss the applicability of this curve to the remaining transients and accidents in Chapter 15.0. Also, relate this curve to Figure 4.3-3 (and the 0.8 factor for uncertainties) and to Figure 4.3-18 (which is incorrectly labeled and which shows different scram specifications than the response to Question 15.7.4 has indicated). Correct the inconsistencies and review all transients and accidents to assure that the most conservative scram characteristics were, in fact, utilized.

15.52 The response to Question 15.8 is insufficient to allow an adequate evaluation. Reference is made to Question 4.57 for the reactivity coefficients assumed for the transients in subsections 15.1.1-15.1.10 and 15.1.17-15.1.26. Question 4.57 references Question 4.60 which refers to page 4.3-6 in GESSAR. This page states, "... near the end of cycle the overall moderator temperature coefficient becomes slightly positive."

Our concern is that, where applicable, the proper reactivity coefficients may not have been applied. For example, some transients analyzed in subsections 15.1.1-15.1.10 and 15.1.17-15.1.26 would seem to be more severe with a beginning-of-life reactivity coefficient, rather than end-of-life. Nothing in the response to this question lists the values used in each transient so that confirmation that the most conservative assumptions were used could be made. Provide such a tabulation of values and include the time of core life (BOL vs EOL) at which the reactivity effect would be maximized.

15.53 With regard to Question 15.11, the response does not provide the Summary Table requested. Provide the requested information.

15.54 Section 15.1.28 on ATWS is not consistent with current requirements. Our position is that this section should be changed to address the Regulatory staff position published in the "Technical Reports on Anticipated Transients Without Scram for Water-Cooled Power Reactors," (WASH-1270) dated September 1973.

15.55 The response to Question 15.15.2 regarding inadvertent start of a HPCS pump states that, ..."there are no safety actions required to prevent unacceptable results." With regard to this answer,

a. Section 15.1.10.1.3 on page 15.1-51 states that for this event, "The operator must take the necessary actions to stop the injection of cold water to prevent rapid cooldown and high water level in the reactor."

b. The response to Question 15.15.1 states that for this event, "high level protection trips also will prevent the level from reaching the steam line."

c. Subsection 15.1.10 does not show the worst case which is inadvertent operation of all ECCS (see the partial discussion on page 6.3-25).

In light of the above concerns, our position is that this transient must be reanalyzed, subsection 15.1.10 must be revised and that, if the safety actions in parts a and b (or others) are required, a protection sequence diagram should be provided.

15.56 The response to Question 15.15.4 that this event does not require manual action by the operator to avoid unsafe consequences is in conflict with the analysis on page 15.1-51 which states, The operator must take the necessary actions to stop the injection of cold water to prevent rapid cooldown and high water level in the reactor." Correct this inconsistency.

15.57 With regard to the response to Question 15.16.5, on continuous rod withdrawal during power range operation, our position is that the protection sequence diagrams in Appendix B should include the transients and accidents addressed in Chapter 15. The intent of Appendix B as stated on page B-4, is to show that, "The 'worst cases' analyzed in Chapter 15 correspond to one protection sequence for each event in Appendix B." Since credit is taken for the rod block monitor (RBM) in the "worst case" event analyzed in subsection 15.1.11, either revise Figure B-16 to reflect a proper identification of ALL essential protection sequences, or provide an additional protection sequence diagram to reflect the analysis provided in Chapter 15.0.

15.58 With regard to the response to Question 15.17.1, our position is that the protection sequence diagram for this event should be included in Appendix B. Page B-4 of this Appendix states, "The 'worst cases' analyzed in Chapter 15 correspond to one protection sequence for each event in Appendix B." Since credit is taken for the "control system" in the worst case event analyzed in subsection 15.1.12, either revise Figure B-16 to reflect a proper identification of ALL essential protection sequences, or provide an additional protection sequence diagram in Chapter 15.0.

Also, we have noted that at the time that this GESSAR was submitted, no description of this system existed on the docket. The acceptability of taking credit for this system is pending the outcome of the staff review of the material submitted in Amendment 11 and any further requirements of the staff for information to complete a proper evaluation of these new designs.

15.59 With regard to Question 15.17.2, the response is insufficient to allow an adequate evaluation. The answer provides a completely revised analysis with new assumptions. Justify the new initial assumptions which depart from previous BWR control rod withdrawal event methods.

Also, this new analysis should address all assumptions with regard to the Rod Pattern Control System and Rod Block Monitor. Revise subsection 15.1.12.

15.60 The response to Question 15.21 is not consistent with the intent of Appendix B as clarified in the response to Question 15.15.2; that is, if no safety actions are required to prevent unacceptable results, a protection sequence diagram is not necessary. Therefore, the worst-case Chapter 15 analysis in subsection 15.1.17 and Appendix B should agree with this intent.

15.61 With regard to Question 15.23, the response is insufficient to allow an adequate evaluation. The protection system diagrams still appears to be inconsistent with page 15.1-102 which indicates that core cooling is required. Our concern is that sufficient coordination with analysts has not been conducted on Appendix B. Page 15.1-102 states that, "The operator must regain control of reactor water level through RCIC operation or by restart of a feedwater pump." This indicates that more safety actions are required than are reflected in the protection system diagram submitted in the response to response to Question 15.23. Also, discuss the time delay during which the operator's failure to act properly will have no unsafe consequences, and the consequences if the action is not performed at all.

15.62 With regard to Question 15.25, the response is insufficient to allow an adequate evaluation. Our concern is that the system is not designed to perform its function following a postulated failure. List the exact procedures that an operator would follow should he discover an RHR suction line failed closed. List the single active failures which would negate both the RHR suction from the recirculation loop and the LPCI suction from the suppression pool (such as a diesel failure).

15.63 The response to Question 15.26 is insufficient to allow an adequate evaluation. Whether or not the loss of instrument air is a limiting event does not appear relevant. Page B-3 in Appendix B states,

> "The tendency to be preoccupied with 'worst case' (those that give the most severe consequences) is recognized; however, the protection sequences essential to lesser cases may be different from the worst case sequence. To assure that operational requirements are found for all equipment essential to attaining acceptable consequences, all essential protection sequences must be identified. In this way a comprehensive level of safety is attained. Thus, the NSOA is protection sequence oriented to achieve comprehensiveness."

Therefore, this event should be included in Appendix B, especially if safety actions are required.

15.64 The response to Question 15.27.5 is insufficient to allow an adequate evaluation. Being a limiting accident condition was not a criterion by which events were included in Appendix B. (see the statement in the preceding Question)

Also, not being an abnormal operational transient was not a criterion by which events were excluded from Appendix B. Accidents and Special Events have their place also.

Furthermore, certain safety actions, such as scram, are apparently required to mitigate the consequences of this event (see page 15.1-142). Therefore, our position remains that the protection sequence for this event should be in Appendix B.

15.65 The response to Question 15.28.1 is insufficient to allow an adequate evaluation. Also, the top of page B-4 in Appendix B states, "The 'worst cases' analyzed in Chapter 15.0 correspond to one protection sequence for each event in Appendix B." If credit is allowed for the Rod Pattern Control System in the worst case event analyzed in subsection 15.1.38, we will require either a revision of Figure B-25 to reflect a proper identification of ALL essential protection sequences, or the addition of a protection sequence diagram to reflect the worst case accident submitted in Chapter 15.0

15.66 With regard to Question 15.28.2, the response is not acceptable. Conservatively selected input paramters can be used in the analysis of this event. These values can be based on nominal values of the new core design. Provide the requested analysis.

15.67 With regard to Question 15.28.3, the response is insufficient to allow an adequate evaluation since the revised Figure B-25 was not received and the response conflicts with the discussion on page B-34 of Appendix B. Justify that a proper reassessment and revision of this accident was made by detailing the initiating parameter(s) which would cause containment isolation to be complete and automatic. Specify the isolation components included for consideration.

15.68 With regard to the response to Question 15.29, the top of page B-4 states, "The 'worst cases' analyzed in Chapter 15.0 correspond to one protection sequence for each event in Appendix B." Our position is that we will require either a revision of Figure B-25 to reflect a proper identification of ALL essential protection sequences, or the addition of a protection sequence diagram to reflect the worst case accident submitted in Chapter 15.0.

15.69 An Engineered Safeguard/Engineered Safety Feature is defined in Appendix A as "A safety/system, the actions of which are essential to a safety action required in response to accidents." Figure B-25 in Appendix B clearly shows the RCIC system as necessary to meet the single failure criterion in mitigating the consequences of the design basis control rod drop accident. It is our position the RCIC system should be reclassified and upgraded to meet the design and quality control requirements associated with an Engineered Safety Feature/Engineered Safeguard.

15.70 With regard to Question 6.25, the response is insufficient to allow an adequate evaluation. Provide the safety design basis for the ECCS Discharge Line Fill System with an appropriate safety evaluation showing how the design basis will be met. Estimates of flow rates, design pressures and temperatures, pump initiation signals, and single failure analyses should be provided for this essential system, just as this type of information is provided for the ECCS.

17.0 Quality Assurance

17.34 Provide an organization chart similar to Figure 17.1-1 for each of the GE department (such as APED manufacturing in Wilmington) participating in activities associated with the BWR/6 and identify the major components of each of these departments.

17.35 Provide a complete Quality Assurance and Quality Control Organization chart for each of the GE departments and identify those individuals or groups assigned the responsibility of performing the quality acceptance inspection and non-destructive testing, both at the in-process and final inspection stages. Clearly indicate from whom technical QA direction is received and from whom administrative control (salary review, hire/fire, position assignment) is exercised.

17.36 Identify those individuals or groups on Figure 17.1-1 responsible for performing in-process and final inspection and verification operations at GE's vendors and subcontractors facilities.

17.37 Will GE be responsible for QA and QC functions associated with GE equipment, such as inspection, verification, and the review of quality related documents, at the applicant's nuclear facility during the installation, preoperational and startup phases? If so, identify the individual or groups and describe their authority and responsibility.

17.38 Describe in more detail how the Manager of the Nuclear Energy Division, the Manager of the Atomic Power Equipment Department, and the Manager of the Nuclear Fuels Department assess the scope, implementation, and effectiveness of the QA Program.

17.39 Describe in more detail the receiving inspection activities at the various GE plants. It is our position that a visual inspection of the product at receiving inspection be performed to assure it has not been damaged during shipping and handling.

17.40 Who determines when special processes are needed and at what stage of the design? How are these special processes controlled?

17.41 Identify those individuals or groups responsible for determining when preplanned operations and inspections can be bypassed or performed out of sequence.

17.42 During the design process are there controls requiring that the characteristics of the designs or specifications be classified as to their importance to safety?

17.43 The response to question 17.19 is not complete. Provide the following information with respect to nonconformance reports:

17.43.1 Describe in more detail the procedure for determining what nonconformance reports dispositioned "use as is" or "repair" are submitted to the applicant. It is our position that any safety related nonconformance reports should be sent to the applicant.

17.43.2 Describe if nonconformance reports dispositioned "use as is" or "repair" will be identified on the acceptance documentation that accompanies the hardware to the applicant's nuclear facility.

17.43.3 Describe if nonconformance reports are uniquely identified such that traceability to the associated hardware can be achieved.

17.43.4 Describe if there is a system established by GE whereby nonconformance reports are periodically analyzed by GE QA to determine quality trends.

17.44 Describe in detail the responsibilities of GE QA in determining the need for corrective action and for assuring that corrective action has been properly accomplished.

17.45 Describe those provisions for filing and maintaining nonconformance reports, corrective action reports and audit results at GE. Describe those provisions by which the applicant and AEC personnel have access to these documents.

17.46 Describe QA responsibilities for collecting, storing, maintaining, and controlling quality related records.

17.47 Describe those provisions in GESSAR which require that all QA audits will be conducted using preestablished written procedures.

17.48 Describe in more detail the guidelines used in determining the frequency of the audits conducted by the NED departments and by NED QA Operations.

17.49 Identify those minimum areas of activity that will be audited by each of the NED departments and by NED QA Operations.

17.50 NED QA Operations is required to audit each NED department at least once a year. Describe the depth of these audits and the expected time to conduct one of these audits.

17.51 Identify the management level required to receive those audit results conducted by each NED department and by NED QA Operations.

17.52 Discuss the means by which deficiencies that are revealed by results of audits are resolved.

17.53 Your response to question 17.24 is not clear in that it is not evident whether internal audits of the major departments are performed by the respective QA manager or by the department manager. Identify the specific individuals (by job position not name) or groups that are responsible for conducting internal audits of each NED department.