repro 3-22
Ltr sent

Distribution w/enclosure:
Docket Files
NRC PDR
Local PDR
LWR #3 File
NRR Reading
RSBoyd
RCDeYoung
DBVassallo (2)

Docket Nos. 50-438
and 50-439

FJWilliams
HSmith
ODParr
WPike
MRushbrook
JRutberg, OELD
JSaltzman, AIG
[illegible]les, OPA
NDube, MIPC
GWilliams, DSE
KSinger, SS
KGoller
IE (3)

MAR 22 1978

HDenton, DSE
VAMoore, DSE
RHVollmer, DSE
MErnst, DSE
WGammill, DSE
OELD
WRoss
WHaass
RClark
JRBuchanan, NSIC
TBAbernathy, TIC
JYore, ASLBP
ACRS (16)

Tennessee Valley Authority
ATTN: Mr. W. B. Hughes
Manager of Power
830 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

SUBJECT: ACCEPTANCE REVIEW FOR BELLEFONTE NUCLEAR PLANT

On February 3, 1978, you tendered an application for operating licenses for the Bellefonte Nuclear Plant Unit Nos. 1 and 2. Your application included a Final Safety Analysis Report (FSAR), Environmental Report (ER) and general information.

I regret that we have not been able to make a determination as to acceptability within 30 days, but the completeness of the application vis-a-vis information relating to the reactor protection system has required additional study. Bellefonte plans to use a reactor protection system (RPS-II) of an advanced design that we have not reviewed and approved. In a February 21, 1978 letter, you informed us that final design information will not be submitted until October 15, 1978, and that verification test results will not be submitted until April 1979. As such, you are proposing to submit essential final design information more than a year after tendering of the application, and only eight months prior to TVA's presently anticipated fuel load date of December 1, 1979. On the one hand, we are reluctant to accept an incomplete application. On the other, it would be difficult, if not impossible, to complete the required review based on the present application by the end of next year.

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SURNAME ➤						
DATE ➤						

Considering this delemma, I think there is a need for two distinct meetings to discuss these matters. First, it is essential for us to establish as accurate a construction completion date for Unit 1 as possible. Using a rather generic model, and assuming that Unit 1 is about 52% complete (as reported in February 1978 Construction Status Report) we can estimate construction completion in the Spring of 1981. If in fact this is a reasonable estimate, then TVA would have time to develop the required information on its present schedule and the application could be docketed in April 1979 with sufficient time for the staff to complete its review prior to the completion of construction of Unit 1. To this end, we will contact you shortly to set up a meeting to review in detail the plant-specific aspects of Unit 1 construction to see if we can arrive at a mutually acceptable estimate for a construction completion date.

In the event that it appears that construction of Unit 1 would be complete within the next 21-30 months, then we should schedule a meeting promptly with TVA and B&W to discuss the required reactor protection system information and explore means to obtain sufficient information to permit accepting the application and starting the review on a more accelerated schedule.

During the course of our acceptance review of your FSAR we have discovered some areas where we will need additional information for our detailed review. These are identified in the attached requests for additional information (Enclosure 1), and includes the information related to the reactor protection system. Some of the information requests are of the type normally resulting from the first round of our detailed review of an application and are provided now to enable you to amend the application with the required information at the earliest practical time. We are prepared to meet with you in the near future to discuss the requests.

Further, we have identified two problem areas which could result in a delay of the issuance of the Bellefonte operating licenses unless appropriate action is taken during the early stages of our review. These problem areas are as follows:

(1) In a letter dated December 27, 1977, we informed you of seismic design questions that must be addressed in the Bellefonte FSAR. You responded in a letter dated February 6, 1978, stating that the information will be supplied on or about July 3, 1978. This schedule must be met to avoid any subsequent delays in our review of Bellefonte. Our technical staff is available to discuss this matter further.

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(2) Section 11 of the Bellefonte FSAR does not adequatly address the requirements of Appendix I to 10 CFR Part 50 (see Request 321.2 of Enclosure 1). I understand this information can be available within the next two months.

I do not believe these matters will present scheduling problems unless it turns out that construction completion is must sooner than we now anticipate.

Sincerely,

/S/

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional
Information for Final
Safety Analysis Report

cc w/enclosure:

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Tennessee Valley Authority
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Knoxville, Tennessee 37902

Mr. E. G. Beasley
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OFFICE ➤	LWR #3:LPM	LWR #3:BC	LWR:AD	D:DPM		
SURNAME ➤	WJPike/LLM	ODParr	DBVassallo	RSBoyd		
DATE ➤	3/ /78	3/ /78	3/ /78	3/ /78		

ENCLOSURE
REQUESTS FOR ADDITIONAL
INFORMATION FOR FINAL
SAFETY ANALYSIS REPORT
BELLEFONTE NUCLEAR PLANT
DOCKET NOS. 50-438 AND 50-439

010.0 AUXILIARY SYSTEMS

010.1 (3.6) With regard to postulated pipe breaks outside containment, we require that you demonstrate that safety related equipment will be adequately protected in the event of a high energy fluid system pipe break. We also require that you show compliance with Regulatory Position C.2 of Regulatory Guide 1.29 to assure that nonseismic high-energy lines outside containment will not fail in the event of an SSE and thereby reduce the functioning of other safety related equipment. We request that you submit sufficient analyses to confirm the adequacy of your design in this regard.

010.2 (9.0) Provide a tabulation of all valves in the reactor pressure boundary and in other seismic Category I systems (per Regulatory Guide 1.29) e.g., safety valves, relief valves, stop valves, stop-check valves, and control valves whose operation is relied upon either to assure safe plant cold shutdown or to mitigate the consequences of an accident. The tabulation should identify the system in which it is installed, the type and size of valves, the actuation type(s), and the environment of conditions to which the valves are qualified.

010.3 (9.1) You state that the safety provisions that are designed into the reactor building polar crane and the primary auxiliary building crane are discussed in Section 3.8. We find no evidence of these safety provisions as stated. Provide a discussion of the safety provisions for these cranes.

010.4 (9.5) Your fire hazard analysis does not include an evaluation of the effects or consequences of postulated fires including exposure fires. The effects or consequences should be evaluated with and without actuation of an automatic suppression system, including consideration of transient combustible exposure fires. Additional guidance for preparation of a fire hazard analysis is provided in proposed Regulatory Guide 1.120, Revision 1.

In addition to your fire hazard analysis, provide drawings which show the site related systems and components, fire protection system P&I drawings, and yard fire loop arrangement drawings including fire pumps and storage tanks.

010.5 (9.2) (RSP) We are unable to establish from your component cooling water system drawings that inadvertent closure of any of the valves in this system or a moderate energy line break would not terminate cooling water flow to the reactor coolant pumps which could potentially lead to fuel damage or break of the primary system barrier resulting from locked reactor coolant pump motors. Revise your design to meet our position concerning design of the cooling water system:

(1) A single failure in the component cooling water system shall not result in fuel damage or damage to the reactor coolant system pressure boundary caused by an extended loss of cooling to the reactor coolant pumps. Single failure includes operator error, spurious actuation of motor-operated valves, and loss of component cooling water pumps.

010.5 (2) Moderate energy leakage crack or an accident that is initiated from a failure in the component cooling water system piping shall not result in excessive fuel damage or a breech of the reactor coolant system pressure boundary when an extended loss of cooling to the reactor coolant pumps occurs. A single active failure shall be considered when evaluating the consequences of the accident. Moderate leakage cracks should be determined in accordance with the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment."

To meet the two criteria above, that portion of the component cooling water system which supplies cooling water to the reactor coolant pumps can be designed to non-seismic Category I requirements and Quality Group D if you demonstrate that the reactor coolant pumps are capable to operate with loss of cooling for longer than 30 minutes without loss of function and the need for operator protective action. And, safety grade instrumentation to detect the loss of component cooling water to the reactor coolant pumps and to alarm the operator in the control room is provided. The entire instrumentation system, including audible and visual status indicators for loss of component cooling water should meet the requirements of IEEE Std 279-1971. Alternately, if it cannot be demonstrated that the reactor coolant pumps will operate longer than 30 minutes without loss of function or operator corrective action, then your design must meet the following requirements for the entire component cooling water system:

010.5 (1) Safety grade instrumentation consistent with the criteria for the protection system shall be provided to initiate automatic protection of the plant. For this case, the component cooling water supply to the seals and bearings of the pumps may be designed to non-seismic Category I requirements and Quality Group D.

(2) The component cooling water supply to the pumps shall be capable of withstanding a single active failure or a moderate energy line crack as defined in our Branch Technical Position APCSB 3-1 and be designed to seismic Category I, Quality Group C and ASME Section III, Class 3 requirements. Provide additional assurance that you comply with this position.

010.6 (9.3) Design deficiencies have been identified in other B&W plants that following a loss of offsite power the reactor coolant pump seals cannot withstand the resulting interruption of seal water flow without damage. Expand Section 9.3.6 to address the Bellefonte Plant design relative to the above stated deficiency, if there is any, and confirm that the Bellefonte Plant can withstand a loss of offsite power without seal damage to the reactor coolant pumps.

022.0 CONTAINMENT SYSTEMS

022.1 (6.2.1) With regard to the containment response analysis for a postulated main steam line break, provide the following additional information:

(1) Provide the results of a spectrum of break sizes at various power levels in order to identify the worst case main steam line break for containment pressure and containment atmosphere temperature.

(2) Provide a discussion of the single failure analysis and include the results of the analysis to demonstrate that the design basis active failure has been identified.

(3) Provide justification for the reliance on any equipment or component which is non-safety grade to mitigate the accident.

(4) For the case which results in the maximum containment pressure and for the case which results in the maximum containment atmosphere temperature, graphically show the containment atmosphere pressure and temperature and structural heat transfer coefficient as a function of time. Include a table of accident chronology and containment energy distribution similar to that provided for LOCA analyses.

(5) Discuss the manner in which unisolated inventories in the steam line and feedwater lines are accounted for in the mass and energy release calculation.

(6) Specify and justify the design temperature of the containment structure shell and concrete, the design temperature of the internal structures and the temperature used to qualify the safety-related instrumentation located within the containment.

022.2 Provide the following information regarding the environmental qualification of safety related equipment.

(1) Provide a comprehensive list of equipment required to be operational in the event of a main steam line break (MSLB) accident to mitigate the accident consequences and assure a safe shutdown of the plant. The list should include, but not necessarily be limited to, the following safety related equipment:

(a) Electrical containment penetrations
(b) Pressure transmitters
(c) Containment isolation valves
(d) Electrical power cables
(e) Electrical instrumentation cable
(f) Level transmitters

Describe the qualification testing that was done, including the test environment, namely, the temperature, pressure, moisture content, and chemical spray as a function of time.

(2) It is our position that the thermal analysis of safety related equipment which may be exposed to the containment atmosphere following a main steam line break accident should be provided based on the following:

(a) A condensing heat transfer coefficient based on the recommendations in Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation" should be used.

(b) A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period it is appropriate to use a conservatively evaluated forced convection heat transfer correlation. For example:

$$Nu = C(Re)^h$$

where Nu = Nusselt No.

Re = Reynolds No.

C, h = emperical constants dependent on geometry and Reynolds No.

Since Reynolds number is dependent on velocity, it is necessary to evaluate the forced flow currents which will be generated by the steam generator blowdown. The CVTR experiments provide limited data in this regard. Convective currents of from

10 ft/sec to 30 ft/sec were measured locally. We recommend that the CVTR test results be extrapolated conservatively to obtain forced flow currents to determine the convective heat transfer coefficient during the blowdown period. After the blowdown has ceased or been reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable.

(c) For each component where thermal analysis is done in conjunction with an environmental test at a temperature lower than the peak calculated temperature following a main steam line break accident, compare the test thermal response of the component with the accident thermal analysis of the component. Provide the basis by which the component thermal response was developed from the environmental qualification test program. For instance, graphically show the thermocouple data and discuss the thermocouple locations, method of attachment, and performance characteristics, or provide a detailed discussion of the analytical model used to evaluate the component thermal response during the test. This evaluation should be performed for the potential points of failure such as thin cross-sections and temperature sensitive parts where thermal stressing, temperature-related degradation, steam or chemical interaction at elevated temperatures, or other thermal effects could result in the failure of the component mechanically or electrically. If the component thermal response comparison results in the prediction of a more severe thermal transient for

the accident conditions than for the qualification test, provide justification that the affected component will perform its intended function during a MSLB accident, or provide protection for the component which would appropriately limit the thermal effects.

022.3 (6.2.1) In the unlikely event of a pipe rupture inside a major component subcompartment, the initial blowdown transient would lead to nonuniform pressure loadings on both the structure and the enclosed component(s). To assure the integrity of these design features, we request that you perform a subcompartment, multi-node pressure response analysis, and provide the following information:

(1) Provide the results of analyses of the pressure transient resulting from postulated hot-leg and cold-leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity, pipe penetrations, and steam generator compartments. Provide the results of similar analyses for the pressurizer surge and spray lines, and other high energy lines located in containment compartments that may be subject to pressurization.

(2) Provide and justify the pipe break type, area, and location for each analysis. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design or both.

(3) For each compartment provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component supports evaluation.

(4) Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations.

(5) Provide a tabulation of the nodal net-free volumes and interconnecting flow path areas. For each flow path provide an L/A (ft^{-1}) ratio, where L is the average distance the fluid flows in that flow path and A is the effective cross sectional area. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

(6) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of

spatial pressure variation; e.g., pressure variation circumferentially, axially and radially within the compartment. Describe and justify the nodalization sensitivity study performed for the major component supports evaluation, where transient forces and moments acting on the components are of concern.

(7) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

(8) Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressure on structures and components.

(9) For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed.

(10) Provide the peak and transient loading on the major components used to establish the adequacy of the supports design. This should include the load forcing functions (e.g., $f_x(t)$, $f_y(t)$, $f_z(t)$) and transient moments (e.g., $M_x(t)$, $M_y(t)$, $M_z(t)$) as resolved about a specific, identified coordinate system.

(11) Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

022.4 (6.2.1) Discuss the safety grade instrumentation provided to monitor and record conditions in a post-accident containment environment in accordance with Regulatory Guide 1.97 (Revision 0).

022.5 (6.2.1) Provide detailed drawings showing the radiation shield blocks located around the reactor vessel flange. List the materials that make up the blocks. Discuss the potential for the shield blocks becoming missiles under accident conditions.

022.6 (6.2.2) Identify the types of insulation used within the containment (e.g., reflective metal insulation, mass insulation, and encapsulated (sheathed) mass insulation). Discuss the extent to which insulation in the vicinity of a postulated pipe break could be stripped from piping and components and identify the insulation involved. Discuss the potential for loose insulation and other debris to clog drains leading to the sump (e.g., within the refueling canal), and the sump screening.

022.7 (6.2.2) Discuss the potential for water becoming trapped in the instrumentation tunnel, and being prevented from draining to the containment sump.

022.8 (6.2.2) Describe in detail how the minimum NPSH of the spray pumps (20.377 ft) was determined. Specify if the static head at the centerline of pump suction takes credit for the height of water on the containment floor.

022.9 (6.2.2) There appear to be conflicting statements in Section 6.2.2.3.2.4 regarding credit for additional containment pressure over sump vapor pressure in doing NPSH calculations. It is stated that reactor building pressure and vapor pressure used for NPSH calculations equal 14.7 psia. However, there is also a statement that the sump temperature is 225°F, but no credit is taken for containment pressure over atmospheric conditions. Please clarify this section of the FSAR.

022.10 (6.2.3) Identify those portions of the secondary containment that are not normally maintained at a pressure less than -0.25 in. w.g. and those portions which will have a pressure higher than -0.25 in. w.g. in the event of a postulated LOCA inside containment.

022.11 (6.2.3) Provide an analysis of the pressurization of the auxiliary areas due to equipment heat loads in the event of a postulated LOCA inside containment.

022.12 (6.2) The minimum containment pressure analysis for ECCS performance evaluation should be performed with containment data which is conservative with respect to the corresponding containment data used in the analysis of the maximum containment pressure for postulated high energy line pipe breaks. Therefore, demonstrate that the containment data used for ECCS backpressure analysis is conservative or revise the analysis using conservative containment input data. Also, verify if the containment atmosphere temperature and essential raw cooling water temperatures used in the ECCS backpressure analysis are minimum values as specified in BTP CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

022.13 (6.2) It appears that the material thickness listed for the containment wall and dome listed in Table 6.2.1a is in error. Please clarify this information.

022.14 (6.2) Demonstrate the applicability of the test data used as a basis for the metal corrosion rates in the combustible gas control analysis for the Bellefonte plant.

022.15 (6.2) Discuss the parameters sensed for the initiation of containment isolation. It is our position that there should be diversity in the parameters sensed for the initiation of containment isolation for all containment isolation valves. Discuss your conformance with this position.

022.16 Discuss the operation of the reactor building and instrument room purge system. Include a discussion of when the system is intended for use.

031.0 INSTRUMENTATION AND CONTROL SYSTEMS

031.1 (3.10) Section 3.10 of the Standard Format (R.G. 1.70, Revision 2) required that "the results of tests and analyses . . . that demonstrate adequate seismic qualification should be provided in the FSAR". In this regard, the information in Section 3.10.4 is incomplete. Supplement the FSAR to provide the following information to docketing.

(1) For each type of Class 1E equipment qualified by test provide:

(a) Summary of test data including performance parameters measured and their accuracy before, during and after the tests.

(b) References to final test report and other supporting documentation.

(2) For each type of Class 1E equipment qualified by operating experience provide:

(a) Summary of features to be demonstrated by operating experience

(b) The bases used to determine that the data are suitable and the equipment qualified

(c) References to final test report and other supporting documents

(3) For each type of Class 1E equipment qualified by analyses provide:

(a) Summary of the analytically established performance parameters and their acceptability

(b) References to final test report and other supporting documents.

031.2 (3.11) Section 3.11 of the Standard Format requires that a description be provided of the qualification tests and analyses that have been performed on each type of Class 1E equipment and that the results of the qualification tests for each type of equipment be provided in the FSAR. In this regard, the information in Section 3.11 is incomplete. Supplement the FSAR to provide the following information for both TVA and B&W supplied equipment prior to docketing.

(1) For each type of Class 1E equipment qualified by test provide:

(a) Summary of test data, including performance parameters measured and their accuracy before, during and after the tests.

(b) References to final test report and other supporting documentation.

(2) For each type of Class 1E equipment qualified by operating experience provide:

(a) Summary of features to be demonstrated by operating experience

(b) The bases used to determine that the data are suitable and the equipment qualified.

(c) References to final test report and other supporting documents.

(3) For each type of Class 1E equipment qualified by analyses provide:

(a) Summary of the analytically established performance parameters and their acceptability.

(b) References to final test report and other supporting documents.

031.3 (7.0) (1.4) The identification of safety-related systems in Section 7.1 does not completely correspond to the systems as described in Sections 7.3, 7.4, and 7.6 and identified in Section 1.7 of the report. In this regard, the report requires correction. Provide complete and compatible reference identification among sections of the FSAR.

031.4 (7.0) (1.7) The information in Section 7.0 of the FSAR is incomplete with respect to the final system drawings identified in Section 7.0 of the Standard Format. Furthermore, the list of electrical instrumentation and control drawings in Section 1.7 of the FSAR does not list these drawings. In accordance with the Standard Format, provide the final electrical instrumentation and control systems drawings in Section 7.0 of the FSAR or expand the list of drawings

in Section 1.7 of the FSAR to identify all final electrical, instrumentation and control system drawings and provide the drawings separate from the FSAR.

031.5 (7.2) Section 7.2 of the FSAR references Babcock and Wilcox Topical Report BAW-10085, "Reactor Protection Systems," for the design of the "RPS-II" reactor protection system for the Bellefonte units. The staff has not completed the review of this topical report. In addition, BAW-10085 references Topical Report BAW-10121, "RPS Limits and Setpoints," to support many of the bases for the design of the reactor protection system described in BAW-10085. This topical report has not been submitted for review. BAW-10085 also references Topical Report BAW-10082, "Qualification of Safety-Related Control and Instrumentation Equipment", for the qualification of equipment. This topical report is not applicable to Bellefonte instrumentation qualification. Therefore, we require that you provide information regarding equipment qualification, analyses to support the design of RPS-II and interface design details in your FSAR. In this regard, we have itemized in Enclosure 2 those specific requests by the staff for additional information regarding BAW-10085 to which B&W has responded that the information would be available either in Topical Reports BAW-10082 and BAW-10121 or in the applicable plant safety analysis report.

031.6
(7.5)

Recently (August 1977) the staff has issued Regulatory Guide 1.97, (Rev. 1) "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident." The staff has determined that, in addition to the applications covered in the "implementation" section of the regulatory guide, all other applications are also required to address the recommendations of the guide.

Provide a discussion as to how your design will satisfy the recommendation of Regulatory Guide 1.97 or provide an acceptable alternative design approach. (See also Request 022.4).

040.0 POWER SYSTEMS

040.1 (8.1) Table 8.1.4-1 of the FSAR does not include Branch Technical Positions to which the offsite and onsite power systems are designed. Acceptance Criteria for Electric Power, Table 8.1 of the Standard Review Plan, presents Branch Technical Positions to which the offsite and onsite electrical systems should be designed. Revise table 8.1.4-1 to include this information, and provide a description showing how the existing design complies with our positions and justify any non-conformance.

040.2 (8.1) The following regulatory guides must be addressed in the Bellefonte FSAR:

1.63 Rev. 1 - "Electric penetration assemblies in containment structures for light-water-cooled nuclear power plants"

1.100 - "Seismic qualification of electric equipment for nuclear power plants"

1.108 - "Periodic testing of diesel generators used as onsite electric power systems at nuclear power plants"

1.118 - "Periodic testing of electric power and protection systems"

We require that the recommendations included in these regulatory guides be satisfied in your design or provide a description and justification of the alternate criteria you intend to use.

040.3 (8.2,8.3) Provide physical layout drawings of the circuits that connect the onsite distribution system to the preferred power supply, and plant layout drawings depicting the physical separation between redundant portions of the onsite distribution system, as requested in Sections 8.2 and 8.3 of the Standard Review Plan.

040.4 (8.2) Describe your design provisions for testing the transfer of power to the safety-related distribution system from the main generator supply to the preferred power system, or to any other supply. Include in your design description the capability for testing during plant operation. (SRP Section 8.2, Part III, Item 3).

040.5 (7.3,8.3) With respect to the application of single failure criterion to manually-controlled, electrically-operated valves, list all valves for which Branch Technical Position EICSB #18 may apply and provide a schematic diagram showing the design feature of locking out power to these valves.

040.6 (6.3,8.3) It is not clear from the description provided for the controls for isolation valves as to how the requirements to prevent overpressurization of the DHR system and the requirement of the DHR system to achieve cold shutdown will be accomplished in the design, in the

event of a single electrical power or control failure in the DHR suction line motor-operated valves. It is the staff's position that the design of the DHR isolation valves be made to conform to the requirements of GDC 34, both in the decay heat removal mode function and while preventing overpressurization of the DHR system.

Provide a modified design for the DHR isolation valves to meet the above stated staff requirements.

040.7 (8.3) Provide the details of your design that will protect the emergency onsite power system and normal operational load from deleterious effects of a degraded offsite power system.

040.8 (8.3) State whether or not the undervoltage detection scheme provided to separate the emergency onsite power system from the utility grid on loss of offsite power is retained while the diesel generator is sequencing loads on the emergency busses. Provide the bases and justification for this design.

040.9 (8.3) Include a description of how the voltage levels at the safety-related busses will be optimized by transformer tap settings to allow for full load and minimum load conditions and remain within the voltage ratings of the connected loads.

040.10 (8.3) Diesel generator alarms in the control room: A review of malfunction reports of diesel generators at operating nuclear plants has uncovered that in some cases the information available to the control room operator to indicate the operational status of the diesel generator may not be precise and could lead to misinterpretation. This can be caused by the sharing of a single annunciator station to alarm conditions that render a diesel generator unable to respond to an automatic emergency start signal and to also alarm abnormal, but not disabling, conditions. Another cause can be the use of wording of an annunciator window that does not specifically say that a diesel generator is inoperable (i.e., unable at the time to respond to an automatic emergency start signal) when in fact it is inoperable for that purpose.

Review and evaluate the alarm and control circuitry for the diesel generators at your facility to determine how each condition that renders a diesel generator unable to respond to an automatic emergency start signal is alarmed in the control room. These conditions include not only the trips that lock out the diesel generator start and require manual reset, but also control switch or mode switch positions that block automatic start, loss of control voltage, insufficient starting air pressure or battery voltage, etc. This review should consider all aspects of possible diesel generator operational conditions, for example test conditions and operation from local control stations. One area of particular concern is the unreset condition following a manual stop

at the local station which terminates a diesel generator test and prior to reseting the diesel generator controls for enabling subsequent automatic operation.

Provide the details of your evaluation, the results and conclusions, and a tabulation of the following information:

(1) Discuss all conditions that render the diesel generator incapable of responding to an automatic emergency start signal for each operating mode as discussed above.

(2) Provide the wording on the annunciator window in the control room that is alarmed for each of the conditions identified in (1).

(3) Discuss any other alarm signals not included in (1) above that also cause the same annunciator to alarm.

(4) Discuss any condition that renders the diesel generator incapable of responding to an automatic emergency start signal which is not alarmed in the control room.

(5) Discuss any proposed modifications resulting from this evaluation.

040.11 (8.3) Describe how your electrical penetrations and associated connections to the field cables are qualified to withstand LOCA and Steam Line Break environment. Your response should address: 1) test plan, 2) test set up, 3) test procedures, and 4) acceptability goals and requirements. Also, provide an evaluation of the results that demonstrate electrical penetrations are qualified to maintain containment integrity during normal, abnormal and accident conditions.

040.12 (8.3) Provide a description of the capability of the emergency power system battery chargers to properly function and remain stable upon the disconnection of the battery. Include in the description any foreseen modes of operation that would require battery disconnection such as when applying an equalizing charge.

040.13 (9.5.4) In Section 9.5.4 signify how the emergency diesel engine fuel oil temperature and pressure is indicated, controlled and monitored. (SRP 9.5.4, Part III, Item 1).

040.14 (9.5.4) What provisions are made in the design of the emergency diesel engine fuel oil storage and transfer system to minimize the entrance of deleterious material into the system during recharging, by operator error or natural phenomena? (SRP 9.5.4, Part III, Item 4).

040.15 (9.5.4) Discuss the precautionary measures that will be taken to assure the quality and reliability of the fuel oil supply for emergency diesel generator operation. Include the fuel oil impurity and quality limitations as well as diesel index number or its equivalent, entrained moisture, sulfur, particulates and other deleterious substances, periodic inspection, and periodic testing (including interval between tests) of fuel oil. In your discussion include reference to industry (or other) standards which will be followed to assure a reliable fuel oil supply to the emergency generators. (SRP 9.5.4, Part III.)

040.16 (9.5.4) Provide information that shows that the fuel oil day tank associated with each diesel generator is located at an elevation to assure slight positive pressure at the engine pumps. (SRP 9.5.4, Part III, Item 5c).

040.17 (9.5.4) Discuss the ability of the Diesel Generator Fuel Oil Storage and Transfer System to withstand any internally and externally generated missiles. (SRP 9.5.4, Part III, Item 2).

040.18 (9.5.4) Provide plan and elevation drawings of the diesel generator fuel oil system from the storage tank to the diesel generator; show distances and separation. Include on the drawings the arrangement of equipment in the diesel generator rooms. (SRP 9.5.4, Part III, Item 3).

040.19 (9.5.4) Provide a drawing showing where the auxiliary boiler building and the fuel oil tanks for the auxiliary boiler and diesel fire pumps are located in relation to the diesel generator building.

040.20 (9.5.4) (9.5.5) (9.5.6) (9.5.7) (9.5.8) Identify any high and moderate energy piping system(s) in the diesel engine room areas. Indicate what means are provided to protect the following diesel engine systems from the effects of a failure of a high and moderate, energy piping:

(1) fuel oil systems

(2) cooling water system

(3) air starting system

(4) lubrication system

(5) combustion air intake and exhaust system

(SRP 9.5.4, Part III, Item 8; SRP 9.5.5, Part III, Item 4; SRP 9.5.6, Part III, Item 5; SRP 9.5.7, Part III, Item 3; SRP 9.5.8, Part III, Item 6c).

040.21 (9.5.5) Provide information on the emergency diesel engine cooling water system water chemistry and chemical treatment to minimize corrosion. Discuss how this compares with the manufacturers recommendations. (SRP 9.5.5, Part III, Item 1c).

040.22 (9.5.5) In section 9.5.5 describe the temperature sensors and alarms provided in your design of the diesel generator cooling water system for controlling the diesel engine jacket cooling water temperature during normal operation. Discuss any operator actions required during alarm conditions to prevent harmful effects to the diesel engine. (SRP 9.5.5, Part III, Item 1c).

040.23 (9.5.5) You state in section 9.5.5 that the diesel engine jacket cooling water system is provided with a heating system which is activated when the engine is on stand-by mode. Provide a detailed description of this system including alarms and control instrumentation and interaction (if any) with the diesel engine cooling water system. Indicate where the alarms are annunciated. (SRP 9.5.5, Part III, Item 1d).

040.24 (9.5.5) The diesel engine generator sets should be capable of operation at less than full load for extended periods without degradation of performance or reliability. Provide a discussion of your diesel engine operating parameters, including minimum load requirements, and relate this to anticipated minimum loads under accident

recovery conditions and during accident standby operations when off-site power is available. (SRP 9.5.5, Part III, Item 7).

040.25 (9.5.6) Describe the method of starting the Diesel Generator including the mechanism for turning the engine. (SRP 9.5.6, Part III, Item 1).

040.26 (9.5.6) Provide a discussion of the measures taken in the design of the standby diesel generator air starting system to preclude the fouling of the starting air valve or filter with contaminants such as oil carry over and rust. (SRP 9.5.6, Part III, Item 1).

040.27 (9.5.6) Describe the sensors and alarms provided in your design of the diesel generator air starting system to warn the operator when design parameters are exceeded. Include the range of the design parameters. Discuss the operator actions during alarm conditions. (SRP 9.5.6, Part III, Item 1).

040.28 (9.5.7) What measures have been taken to prevent entry of deliterious materials into the engine lubrication oil system due to operator error during recharging of lubricating oil or normal operation? (SRP 9.5.7, Part III, Item 1c).

040.29 (9.5.8) Indicate system components in the diesel generator intake and exhaust system that are exposed to atmospheric conditions (ice, freezing rain and snow) and discuss how these components are pro-

tected from possible clogging during any operating conditions. (SRP 9.5.8, Part III, Item 5).

040.30 (9.5.8) Describe any sensors and alarms provided in your design of the diesel engine combustion air intake and exhaust system to warn the operator when design parameters are exceeded. Discuss the operator actions during alarm conditions. (SRP 9.5.8, Part III, Item 1 & 4).

D40.31 (10.2) In section 10.2.2, Description, you indicate that the unit generator is cooled by hydrogen at 75 psig pressure. Describe with the aid of drawings, the bulk hydrogen storage facility including its location and distribution system. Include the protection measures considered in the design to prevent fires and explosions during operations such as filling and purging the generator, as well as during normal operation.

040.32 (10.2) Discuss the effects of a high and moderate energy piping failure or failure of the connection from the low pressure turbine to condenser on nearby safety related equipment or systems. Discuss what protection will be provided the turbine overspeed control system equipment, electrical wiring and hydraulic lines from the effects of a high or moderate energy pipe failure so that the turbine

overspeed protection system will not be damaged to preclude its safety function. (SRP 10.2, Part III, Item 8).

040.33 (10.4.1) Discuss the measures taken to prevent corrosion and/or erosion of condenser tubes and components. (SRP 10.4.1, Part III, Item 1).

040.34 (10.4.1) Indicate what design provisions have been made to preclude failures of condenser tubes or components from turbine by-pass blowdown. (SRP 10.4.1, Part III, Item 3).

040.35 (10.4.4) Provide the results of an analysis indicating that the failure of the turbine by-pass system (TBS), a high energy line, will not have an adverse effect or preclude operation of any safety related components or systems located close to the TBS. (SRP 10.4.4; Part III, Item 4).

112.00 MECHANICAL ENGINEERING

112.1 (3.6) In Section 3.6 it is stated that for those pipe failures where portions of the staff design criteria lead to unacceptable consequences, further analyses based on more realistic assumptions will be performed. Provide the pipe failure locations where these unacceptable consequences will occur. Indicate what assumptions will be made, the kind of analysis performed, and the results obtained for each failure location.

112.2 (3.6.2.1) The information presented in Section 3.6.2.1.2.1.4 of the FSAR is not completely acceptable. In order for a section of pipe between containment isolation valves to qualify for exclusion of postulated breaks, a commitment to the following criteria is required in addition to the stress criteria outlined in Section 3.6.2.1.2.1.4:

(1) The criteria in Paragraphs B.1.b(2) through B.1.b(5) in Branch Technical Position MEB 3-1 (Reference NRC Standard Review Plan 3.6.2)

(2) The augmented inservice inspection criteria in Paragraphs B.2.d(1) through B.2.d(4) in Branch Technical Position APCSB 3-1 (Reference NRC Standard Review Plan 3.6.1)

Revise Section 3.6.2.1.2.1.4 to provide the above acceptance criteria.

112.3 (3.6.2) Section 3.6.2 is not completely acceptable. Provide sketches showing the locations of assumed ASME Section III Class 1 pipe breaks relative to the location of adjacent pipe restraints, and pipe whip barriers.

112.4 (3.6.2.3) Sections 3.6.1.1.3 and 3.6.2.3.5 of the FSAR state that an energy absorbing material may be used in some pipe whip restraints to absorb the kinetic energy of the ruptured pipe and to limit the loads on the restraint structure. If this option is selected, verify that the pipe whip restraint structure which contains the energy absorbing material will be designed to withstand the load which results from using a crush depth of 70% of the available depth. In addition, verify that these types of restraints are designed to the same criteria that is outlined in Section 3.6.2.3.4, "Pipe Whip Restraint Design."

112.5 (3.9.3.1) Section 3.9.3.1.2.1a(1) references ASME Section NB-3500 for loading combinations for Class 1 valves. The ASME Section III Code does not contain enough information on loading combinations to be acceptable. Provide, in the FSAR, the detailed loading combinations which have been and will be used in the design of Class 1 active and inactive valves.

112.6 (3.9.3) The loading combinations for the upset condition which appear in Tables 3.9.3-1 through 3.9.3-8 inclusive, Table 3.9.3-10 and 3.9.3-37 require some clarification. For the upset condition, the following loading combination is required for all ASME Class 1, 2 and 3 systems, components and supports: Operating Pressure + Deadweight + Operating Basis Earthquake + All Transient Loads Resulting from the Upset Condition. Verify that all of the above loads were considered in arriving at the loading combinations which appear in the above referenced tables.

112.7 (3.9.3) The primary bending and local membrane stresses (Pb and PL respectively) are defined in Table 3.9.3-3. These definitions state that the stresses include inertia earthquake effects. Define what inertia earthquake effects are being referenced, i.e., OBE or SSE.

112.8 (3.9.3) In Tables 3.9.3-7, 3.9.3-8, 3.9.3-9, 3.9.3-10, it is stated that "faulted conditions of the NSSS are covered by the Upset and Emergency Conditions as stated above." This statement needs clarification. Indicate how the faulted conditions are covered by Upset and Emergency conditions. State the faulted condition, loading combinations and the corresponding allowable stress limits.

122.0 MATERIALS ENGINEERING

122.1 (4.5) (5.2) (5.4) (6.1) (10.3) Indicate the degree of conformance with the following NRC regulatory guides:

1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"

1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"

1.34, "Control of Electroslag Weld Properties"

1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel"

1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants"

1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"

1.44, "Control of the Use of Sensitized Stainless Steel"

1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"

1.66, "Nondestructive Examination of Tubular Products"

1.71, "Welder Qualification for Area of Limited Accessibility"

1.85, "Code Case Acceptability - ASME Section III Materials"

122.2 (5.4) Describe the methods used to monitor the secondary coolant purity and show that these are at least as conservative as the positions given in the Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in Pressurized Water Reactor Steam Generators," referenced in the NRC Standard Review Plan, NUREG 75/087, Section 5.4.2.1.

30.0 STRUCTURAL ENGINEERING

30.1 (3.3.2) State if venting of structures is adopted as a design measure to reduce the effects of transforming the tornado generated differential pressure into effective pressure on the structures.

30.2 (3.3.2) State how the tornado wind, pressure effects and missiles will be combined directly in a manner such as to be conservative for the structural element being considered. Also, state that the resultant load considered above will be combined with other loads specified in 3.8.1 and 3.8.5.

30.3 (3.3.2) Address the provisions used for Category I structures to prevent their failure from tornado induced failure of non-category I structures other than the cooling towers.

30.4 (3.4.1) State in this section of the FSAR the flood level values considered and how these are applied as hydrostatic loadings or properly relieved by appropriate means.

30.5 (3.4.2) Identify in this section of the FSAR the procedures used for transferring the dynamic flood-to-load, and how these procedures compare with those delineated in the U.S. Army Coastal Engineering Research Center Technical Report No. 4, if applicable.

30.6 (3.4.2) Identify in this section of the FSAR the procedures used for transfering static flood effects to load. Except where relieved by drainage and pumping, the hydrostatic head from the highest flood or ground water level is applied as a direct foundation slab and basement wall load for structural and buoyancy computations. For consideration of lateral and overturning effects, total head including wave effects should be considered.

30.7 (3.5) State the minimum thickness of the concrete barriers (both walls and roofs) provided for all Category I structures resisting the effects of the postulated tornado winds and missiles. These barriers should meet the requirements for design against local penetration, scabbing/spalling and overall effects. Also, identify in this section the properties of the concrete (f'_c and age for its determination) and the typical reinforcement details.

30.8 (3.5) State the ductility factor limits if any considered, in your design of the concrete barriers for the loads generated by the postulated tornado missiles.

30.9 (3.7.1) Provide in this section of the FSAR information about the soil properties such as shear wave velocity, shear modulus poisson's ratio and density as a function of depth.

130.10 (3.7.2) Justify the discrepancies between the two models for the secondary containment given in Figures 3.7.2-1 and 3.7.2-3. Also, discuss any potential for structure-to-structure interaction between the primary and secondary containment structures that may results from the grating supports shown in figures 3.7.2-2 and 3.8.1-1.

130.11 (3.7.2) State if rocking is a general consideration for all Seismic Category I structures or only for the reactor building.

130.12 (3.7.2) State your criteria for the determination of adequate selection of number of lumped masses based on their relationship with any change in the response of a system with greater number of masses from that of a system with fewer lumped mass representation.

130.13 (3.7.2) State your criteria for floor response spectra broadening of the Category I structures located on the soil media. The effect of parameter variation in floor response spectrum for the soil sites are normally accounted for by increasing the individual peak widths by ± 15%.

130.14 (3.7.3) State if the fundamental frequency of the subsystems is controlled to be greater than twice or less than one-half dominant frequency of the supporting system.

130.15 (3.8.1.3) State if Post LOCA flooding is a design consideration; if affirmative, provide the load combination considered for such event.

130.16 (3.8.1.1) State if any interaction between the primary and secondary containment is considered. Fig 3.8.1-1 gives the impression that the grating supports at various elevations, and the supports for the spray-headers frame may produce such interaction.

130.17 (3.8.1.2) (3.8.1.5) (3.8.1.6) (3.8.1.7) Justify in detail any exception taken from the requirements identified in the SRP. Note that the SRP accepts the ACI 359 code (1973) with certain exceptions as identified in the applicable section of SRP 3.8.1.

130.18 (3.8.1.4) State in this section of the FSAR, code requirements that apply to the deisgn analysis procedures.

130.19 (3.8.1.4) Identify any limitations in boundary conditions and the basic assumptions applicable to the computer programs identified in this section of the FSAR.

130.20 (3.8.1.4) State any local effect considerations given to thickned penetrations. For steel liners and anchors ACI/ASME code, Section III, Div. 2 CC-3000 applies; for steel penetrations without structural concrete backing ASME code Section III, Subsection NE applies.

30.21 (3.8.3) (3.8.4) The review of the load combination equations and their related acceptance criteria in this section of the FSAR shows deviations from those identified in the SRP 3.8.3. Provide detailed information and discussions related to the deviations to facilitate staff review of the technical bases for the deviations.

30.22 (3.8.3.1) Identify and discuss the various types of reactor, generator, pump and equipment supports considered in the evaluation of the structures subjected to pertinent seismic loads.

30.23 (3.8.3.6) (3.8.4.6) State in this section of the FSAR if applicable requirements of the following cases are met for each type of material:

(1) concrete..... ACI-318/ACI-349
(2) Steel..... AISC
(3) Reactor Coolant System Support.....ASME/NF
(4) General Quality control.....ANSI N45.2.5
(5) Rebar welding..... ACI 359

30.24 (3.8.5.3) State if you have considered additional load combinations for assessing sliding, overturning and floatation potential of all category I structures, as recommended by SRP 3.8.5.3.

212.0 REACTOR SYSTEMS

212.1 (3.5.1.2) The discussion of internally generated missiles does not include the potential for damage to safety systems and/or the generation of secondary missiles inside containment as a result of a falling object. The discussion also does not address the potential for failing of safety systems inside containment by secondary missiles generated by primary missiles impinging on a component or structure inside containment. These concerns should be discussed in the FSAR.

212.2 (5.2.2) The discussion of the design evaluation of the overpressure protection provided for the Bellefonte plant should give a detailed description of the design basis event, including identification of the event, assumptions made in calculating the consequences, results, and a comparison of Bellefonte parameters with those of the subject plant in the referenced topical report. Section 5.2.2 should also address the generic concern of low-temperature overpressurization.

212.3 (5.2.5) The discussion of intersystem leakage should address back-leakage to ECC systems through isolation valves and show compliance in this regard with Regulatory Guide 1.45.

212.4 (6.3) The discussion of ECCS passive failures does not adequately address the following branch position and should be amended.

It is the Reactor Systems Branch position that detection and alarms be provided to alert the operator to passive ECCS failures during long-term cooling which allow sufficient time to identify and isolate the faulted ECCS line. The leak detection system should meet the following requirements:

(6.3)

(1) Identification and justification of maximum leak rate should be provided.

(2) Maximum allowable time for operator action should be provided and justified.

(3) Demonstration should be provided that the leak detection system will be sensitive enough to initiate (by alarm) operator action, permit identification of the faulted line, and isolation of the line prior to the leak creating undesirable consequences such as flooding of redundant equipment. The minimum time to be considered is 30 minutes.

(4) It should be shown that the leak detection system can identify the faulted ECCS train and that the leak is isolable.

(5) The leak detection system must meet the following standards:

(a) Control room alarms must be provided.

(b) The requirements of IEEE-279 must be met, except single failure requirements.

212.5 (6.3) Provide, in tabular form a list of all valves in the ECCS and their positions during normal operation, all ECCS modes of operation, and all shutdown modes of operation. The valve identification should be consistent with the P&ID's.

212.6 (6.3.4) Section 6.3.4.1 should discuss preoperational tests for the ECCS showing how requirements of Regulatory Guide 1.79 will be satisfied.

221.0 REACTOR ANALYSIS

221.1 (4.4.6.8) The applicant must commit to the installation of an adequate loose parts monitoring system (LPMS). Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at a number of plants. As a result of a study conducted on the installation of, and experience with, loose parts monitoring systems in operating plants, we have identified the following aspects for a LPMS which we will use to assess the acceptability of the specific system to be provided for Bellefonte Units 1 and 2.

(1) The description of the loose parts monitoring system shall include the location of all sensors and the method for monitoring them. A minimum of two sensors will be required at each natural collection region. For example, in a pressurized water reactor, two sensors should be included at the top and at the bottom of the reactor vessel and at each steam generator primary coolant inlet.

(2) The description of the monitoring equipment shall include the levels and the basis for the alarm settings. In addition, the manufacturer's sensitivity specifications for the equipment shall be provided. Anticipated major sources of internal and external noise shall be identified along with the plans for minimizing the effects of these sources on the ability of the monitoring equipment to perform its intended function.

(3) The loose parts monitoring system will be required to function after any seismic event for which plant shutdown is not required. The procedures of Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants", are acceptable for demonstrating the seismic qualification of this system. An exception to this seismic qualification is that recorders are not required to function within their specified accuracy during or after seismic events without maintenance. However, monitoring (alarm and/or indication) capability must remain available for that channel at all times during and after the seismic event. A description of the precautions to be taken to assure the operability of the system after an operating basis earthquake shall be provided.

(4) The loose parts monitoring system should also be qualified in accordance with the recommendations of Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants", but the qualification program need not include a post-accident environment.

(5) The loose parts monitoring system must be operational and capable of recording vibration signals for signature analysis at the time of initial startup testing. A detailed discussion shall be provided of the operator training program, planned operating procedures, and record keeping procedures for the operation of the system.

312.0 ACCIDENT ANALYSIS

312.1 (2.1.2) Indicate whether TVA's custody of the land within the exclusion area includes the mineral rights.

312.2 (2.1.2) Identify the agencies which have jurisdiction over the portion of Town Creek which is within the exclusion area. Discuss the arrangements you have made with them to control the movement of people on Town Creek in the event of an emergency, as required by 10 CFR Part 100. It is noted that such information is lacking in the site Radiological Emergency Plan.

312.3 (2.1.3) Provide the basis for the selection of the projected population center distance of four miles (i.e., why not three miles or five miles?).

312.4 (2.1.3) Provide a table showing the distance and direction from the site of the reservoir recreation areas within 10 miles.

312.5 (2.2.2) Are there any natural gas distribution lines serving the plant? If so, discuss the precautions which have been taken to prevent an accident.

312.6 (2.2.3) It is stated that no significant amounts of toxic chemicals are shipped by barge past the site. However, based on the information in the FSAR, it is not possible for us to agree with that statement. Data for only one year (1975) are given and these show that approximately 60,000 tons of unidentified chemicals were shipped past the site. Provide additional information on the identity of these chemicals. For example, does "nitrogenous chemical fertilizers" include anhydrous ammonia, a hazardous chemical listed in Table C-1 of Regulatory Guide 1.78? In addition, discuss whether 1975 was a representative year.

312.7 (2.2.3) Provide an analysis of the possible effects on the Intake Pumping Station of an explosion of a barge carrying chemical fertilizers. State your assumptions and show all calculations.

312.8 (2.2.3) Discuss the possible consequences of the inlet channel to the Intake Pumping Station becoming blocked by a barge coming to rest across the entrance to the channel.

312.9 (2.2.3) Your aircraft probability analysis is based on the assumption that the present airport will be closed down upon completion of a new airport. Indicate whether this new airport is presently under construction or in the planning stage. If the movement of the airport cannot be firmly established, discuss the impact on your probability analysis of assuming the old airport remains in operation.

312.10 (3.5.1) As indicated in Revision 1 to Section 3.5.1.4 of the Standard Review Plan (p. 3.5.1.4-3, item 5), OL applicants who were not required at the CP stage to design to the Revision 0 missile spectrum should show the capability of the existing structures and components to withstand at least Missiles C and F, that is,

C. Steel rod, 1 in. diameter x 3 ft. long, weight 8 lb, horizontal velocity 422 ft/sec.

F. Utility pole, 13-1/2 in. diameter, 35 ft. long, weight 1490 lb, horizontal velocity 211 ft/sec.

In this regard, indicate the provisions for protecting all safety related structures, systems, and components (as outlined in Regulatory Guide 1.117) for the proposed plant, and in particular, describe the protection provisions for the following:

(1) The diesel/generator building doors and intake and exhaust stacks.

(2) The borated water storage tank.

(3) The essential raw cooling water pumps at the pump intake station.

(4) The main steam and feedwater pipes between the turbine building and the safety valve room.

312.11 (3.5) Contrary to the statement in Section 3.5.1.3.3 (p. 3.5-16) of the FSAR, Table 3.5.1-11 does not contain LP rotor disc fragment energy data. Clarify this inconsistency.

312.12 (3.5) The disc fragments that are postulated in Section 3.5.1.3.3 of the FSAR apparently are based on the assumption that each of the LP rotor discs is independent of the adjacent discs. Since the discs are welded together, it is reasonable to expect that a fragment from a cracked rotor would have a tendency to transmit some forces to the adjacent discs through the welding at its rim. In an overspeed condition, the addition stresses induced in the adjacent discs due to this type of disc-to-disc coupling may lead to additional disc failures. This in turn may lead to multiple separate disc fragments or possibly single missiles consisting of several disc fragments in a "conglomerate" configuration. Discuss this aspect and provide a basis for eliminating the possibility of multiple disc failures.

312.13 (3.5.1.6) Provide the basis for your statement that the plant can withstand the impact of any single-engine aircraft. Specify the weight and speed of the aircraft covered by your statement.

312.14 (6.2.3) Provide the elevations of the air intake and exhaust headers of the Secondary Containment Air Cleanup System.

312.15 (6.4) Provide an estimate of the free air space volume of the habitability zone serviced by the control room ventilation system.

312.16 (15.4.8) With respect to the control rod ejection accident, please provide the following information:

(1) Please provide an analysis of the radiological consequences of the control rod ejection accident.

(2) Provide curves showing the primary and secondary system pressure and temperature vs. time for a period of two hours following the accident.

(3) Provide a curve showing primary-to-secondary system leakage vs. time from accident initiation until primary and secondary system pressures equilibrate.

312.17 (15.6.3) Your analysis of the radiological consequences of a steam generator tube rupture (SGTR) accident does not appear to consider the case of the accident occurring at a time when the primary coolant iodine activity concentration is elevated above equilibrium values due to a previous (coincident) iodine spike. Note that the B&W Standard Technical Specifications (NUREG-0103) allow full power operation for a small percentage of the year with as high as 60 μCi/gm I-131 eq. primary system activity. As described in SRP Section 15.6.3 the staff assumes the primary coolant iodine activity concentration at the time of the accident is equal to the value allowed by the Technical Specifications. Our preliminary dose analysis using the primary coolant iodine activity concentrations allowed by the Standard Technical Specifications and assuming iodine spiking following the accident indicates that the doses could exceed the acceptable dose criteria given in the SRP. Please provide a radiological analysis assuming that the primary system and secondary system iodine activity concentrations are at the values allowed by the Standard Technical Specifications or at the values to be incorporated as the Technical Specifications if different than the Standard Technical Specifications.

321.0 EFFLUENT TREATMENT SYSTEMS

321.1 (10.4.3) Provide a P&ID for the Turbine Gland Sealing System.

321.2 (11.2) (11.3) Provide a detailed evaluation in Section 11.2 and 11.3 of the FSAR to show conformance with Section II.A, B, C & D of Appendix I to 10 CFR Part 50. In addition, you should provide the required information noted in Appendix B of Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors," April 1976.

321.3 (11.4) Provide a description of your process control program to implement the recommendations of Branch Technical Position, ETSB 11-3, for the solidification system you propose to use.

321.4 (11.4) Provide a description of the Solid Waste Storage Facility. In addition, provide a summary of the total storage capacity of this facility.

321.5 (15.7.3) Provide Section 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (Outside Containment)."

331.0 RADIOLOGICAL ASSESSMENT

331.1 (12.1.1) (12.5.1) Commit to the use of Regulatory Guide 8.8, Revision 2, Sections C3 and C4, rather than Revision 1, Section C4, as the basis for issuance of procedures and instructions and developing the health physics program.

331.2 (12.1.2) With regard to the review of changes made during the plant design process, as described in 12.1.2, in order to maintain occupational radiation exposures ALARA:

(1) Identify by title the individual who has been responsible for this radiation protection design review, and describe how she or he relates to the individual responsible for the overall design.

(2) Provide a breakdown by title of radiation protection personnel who have been participating in such reviews, tabulating the health physics education and experience required of each.

(3) Describe formal arrangements and procedures for assuring that adequate radiation protection reviews are performed throughout the design and construction processes and adequate records are kept to document the completion of each such review.

(4) Describe specific examples of actual dose-reducing changes in design that have resulted from these radiation protection design reviews.

331.3 (12.1.2) Describe how experience from similar past designs and from operating plants has been used to develop design criteria, improved radiation protection design, and will be used to develop detailed plans and procedures.

331.4 (12.2.1) Describe the features that you have incorporated into your design to permit plant operators and maintenance personnel to maintain occupational radiation exposure as low as is reasonably achievable by minimizing and controlling the buildup, transport and deposition of activated corrosion products in reactor coolant and auxiliary systems. Include, as a minimum, information on the following steps taken to minimize Co-58 and Co-60, including:

(1) The use of reduced nickel content in systems in contact with reactor coolant.

331.4 (continued)

(2) Low cobalt impurity specifications in systems in contact with reactor coolant.

(3) The minimization of high cobalt, hard facing wear materials in the systems in contact with reactor coolant.

(4) The use of high flow rate/high temperature filtration for systems in contact with reactor coolant.

(5) The selection of valves and packing materials to minimize crud buildup and maintenance.

(6) Provisions for decontamination of components and systems contaminated with activated corrosion products.

331.5 (12.3.1) Discuss the features that you have incorporated into your design to assist the decommissioning crews to maintain occupational radiation exposure as low as is reasonably achievable during the eventual decommissioning of your plants.

331.6 (12.3.3) Justify the use of the values in 10 CFR Part 20, Appendix B, Table as criteria for airborne concentrations under normal operation, rather than the levels which delimit airborne activity area as indicated in § 20.103 (b) (1).

331.7 (12.5.2) Clearly identify the health physics facilities on Figure 12.3.1; provide additional layouts as necessary to show the facilities in the Office and Service Buildings. Show in detail personnel traffic patterns for male and female workers from outside the restricted area, through the access control area, the clean locker room and laundry complex to the working area, and return through the decontamination facility.

360.0 GEOLOGY

361.1 (2.5.3.2) Page 2.5-54. The last paragraph refers to the "Detailed Supplementary Information reports on the Intake Pumping Station and the QA Records Storage Vault" for further information regarding shearing in seismic Category I structures. Either include an evaluation of these shears in the FSAR in the appropriate section, or include the referenced document as a part of the docketed FSAR.

361.2 (Appendix 2.5-B PSAR) A minor fault was encountered in a line of borings drilled across the proposed intake channel. Specifically, the fault was observed in borings 85 and 86 at depths of 150' to 160'. Additional borings confirmed its presence. It was concluded at that time, based on the available information, that the fault was not a hazard to the intake canal. However, TVA stated that when the channel is dewatered and rock is exposed, detailed geologic studies will be made. Include a final evaluation of this fault in light of new data obtained since the CP review.

362.0 FOUNDATION ENGINEERING

362.1 (2.5.4.12) Page 2.5-77. The second paragraph mentions shear planes that were exposed during excavation for the ERCW Intake Pumping Station. For more information on the shear zones, reference is made to Detailed Supplementary Information Report. Either include this report as part of the docketed FSAR, with appropriate cross referencing in the text, or expand the discussion in this section to include definition of the shear zones and an evaluation as to their significance to the structure.

362.2 (2.5.4.12) Pages 2.5-77 and 78 reference the "Detailed Supplementary Information Report" for foundation conditions of the ERCW Intake Pumping Station before and after grouting. The adequacy of the foundation after the grouting must be demonstrated either by including this report in the docketed FSAR, or by extracting data from the report and including it in the docketed FSAR.

362.3 (2.5.4.5) Page 2.5-70. The first two paragraphs refer to TVA's "General Construction Specification No. 6-29" as the standard by which earthfill will be placed and tested. The Specifications within this document that apply to construction control of Category 1 earthfill placement at Bellefonte should be included in this section. Page 2.5-69, third paragraph, states that backfill around seismic Category 1 structures will be compacted to 95% of maximum dry density (Standard Proctor) at optimum moisture content. Include the laboratory test data that demonstrate that all of these design requirements were met during placement and compaction in the docketed FSAR.

362.4
(2.5.4.5)

Page 2.5-71. The second paragraph states that granular fill will be compacted to an average relative density of 85% and a minimum of 80%. Provide the data that demonstrate that these values were achieved during construction in the docketed FSAR.

362.5
(2.5.5.2.2)

Pages 2.5-80 and 81 describe slope stability analyses. Reanalysis of the dynamic stability of the canal slopes may be necessary pending the responses to seismic design basis questions.

362.6
(2.5)

Our review of the tendon anchorage to the limestone foundation indicated a possible instability of the updipped rock wedge along the extreme east and west sides of the containment.

Demonstrate, by analyses and test data on weak bedding planes, that this rock wedge is stable and present the results of his tests and analyses in the docketed FSAR.

362.7
(2.5)

The docketed FSAR should address the creep of Post-Tensioned anchors extending through bedded limestone where the limestone bedding planes are composed of compressible soils.

371.0 HYDROLOGIC ENGINEERING

371.01 (2.4.3.4) We note in this section that the assumed erosion rate of over-topped dams is based on the base length of the overflow channel, the hydraulic head and the developed angle of friction of the embankment material. We do not consider this approach to be substantiated as conservative since it does not consider the velocity of water at the toe of dam below the assumed breached section. Since these velocities at the toe would cause erosion, there would be subsequent slumping of embankment above the eroded toe. This could significantly increase the rate at which the embankment fails, thereby increasing the discharge downstream. Accordingly, reanalyze the embankment failures to take toe erosion into account and re-estimate the resultant maximum water surface elevation at the plant site.

371.02 (2.4.3.6) Your discussion indicates that there are no wind data records available for the period 1948-1964. Since this is rather a large gap in the record, it is conceivable that information during that period would significantly affect your estimates. Also, there are apparently records at other stations (Atlanta, Knoxville, Memphis, Nashville, Louisville, and Birmingham) that were used to test statistically for homogeneity. Therefore, re-evaluate wind at the site using nearby stations to "fill" the missing record by correlation. In lieu of this analysis, substantiate that your estimate of 28 mph for 46 minutes is conservative by some other means.

371.03
(2.4.4)

It is not clear why your assumed failures of Norris, Cherokee and Douglas Dams were assumed to occur during an SSE coincident with a 25-year flood. Without substantiation that these dams would not fail due to an OBE, you should have used an OBE coincident with one-half of a PMF. Substantiate that the above dams would not fail due to an OBE or reanalyze the failures assuming an OBE coincident with one-half of a PMF.

372.0 METEOROLOGY

372.01 (2.3.1) The discussion of lightning on page 2.3-2 concerns only lightning flashes to ground. Provide the probability of a lightning strike to safety-related structures utilizing the estimates of lightning flashes to ground per unit area. Identify the bases for the estimates, and the assumptions that were considered.

372.02 (2.3.1) Provide estimates of the wind speeds associated with the tornadoes observed in the vicinity of the site, as discussed on page 2.3-2.

372.03 The tornado criteria identified on page 2.3-2 do not meet the criteria specified in Regulatory Guide 1.76. Justify the deviations from the recommendations of the Regulatory Guide.

372.04 (2.3.1) Provide the bases for the following assumptions discussed on page 2.3-4: (1) the 17.1 inches of snow at Huntsville on December 31, 1963 and January 1, 1964 approximates the 48-hour Probable Maximum Winter Precipitation; (2) the water equivalency of 6 inches of snow is 1 inch of water.

372.05 (2.3.1) Recent operating experience has identified various failures of systems from freezing temperatures. Identify the design basis maximum and minimum air temperatures (including frequency and duration) considered in the designs of systems and components such as heating and air conditioning systems, impulse lines, service water valves, steam isolation valves, etc. Also discuss the designs of systems and components with respect to combinations of phenomena such as moisture buildup coincident with freezing temperature.

372.06 (2.3.2) Compare available wind data (preferably a two year period) from the 10 m level of the permanent meteorological tower with the data collected previously (10/72 - 10/74) from the temporary 10 m tower located at the proposed site of the reactor structures.

372.07 (2.3.2) The discussion of the onsite meteorological measurements program in Section 2.3.3 identifies dry bulb temperature, dew point temperature and precipitation measurements being made at the Bellefonte site. However, the discussion of temperature, humidity, and precipitation presented in Section 2.3.2 does not utilize any onsite data. Provide data summaries of dry bulb temperature dew point temperature (or humidity) and precipitation, for measurements from the permanent meteorological facility, and compare these summaries with data from Chattanooga, Huntsville, and Scottsburg.

372.08 (2.3.2) Explain why no observations of fog have been made in the vicinity of the site, particularly on top of Sand Mountain Plateau, where an increase in localized fog may be expected to occur as a result of cooling tower operation.

372.09 (2.3.3) Explain the rationale for measuring dew point temperature only at 10 m and not also at an elevation representative of the height of effluent release from the natural draft cooling towers.

372.10 (2.3.3) Clarify whether the determination of vertical temperature gradient is made from direct measurement or from subtraction of two ambient temperature measurements.

372.11 (2.3.3) Provide additional information concerning the CRT display of meteorological parameters, particularly the averaging time associated with the digital print-out. Also provide further detail concerning the use of TVA Meteorological Forecast Center in support of the Radiological Emergency Plan for Bellefonte.

372.12 (2.3.3) Compare the accuracies of the complete data collection and recording systems with the accuracies recommended in Regulatory Guide 1.23.

372.13 (2.3.4) Provide the bases for the "Design Basis Relative Concentration Values" presented in Table 2.3.4-8.

372.14 (2.3.4) We have reviewed the results of recent atmospheric dispersion field experiements and have concluded that the procedure for calculating short-term (accident) relative concentration (X/Q) values for the Bellefonte site should be modified to incorporate the following effects: (1) Lateral plume meander, as a function of atmospheric stability, wind speed, and distance from the source, during periods of light winds and neutral and stable atmospheric conditions; (2) Boundary distances as a function of direction from the plant; (3) Atmospheric dispersion conditions when the wind is blowing in a specific direction; and (4) The fraction of the time that the wind is expected to blow into each of the 16 compass directions. Enclosed is a copy of DRAFT Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (9/23/77), which describes the new procedure in detail. We will calculate revised X/Q values for appropriate time periods for design basis accident evaluations using the model described in the enclosed Draft Regulatory Guide.

Provide exclusion boundary distances as a function of direction using the procedure described in the Draft Regulatory Guide. Provide a large scale (for independent measurement) map of the site and vicinity, identifying the exclusion zone, site boundary, major structures, significant topographic features, and the location of the permanent meteorological facility.

372.15 (2.3.4) Identify the value of the containment cross-sectional area, A, used in equations 2.3-1 and 2.3-3. Provide the building dimensions used to compute this value.

372.16 (2.3.4) Explain the bases for the X/Q values presented in Table 2.3.4-7 and clarify the differences between Table 2.3.4-7 and Table 2.3.4-9.

372.17 (2.3.4) It appears that the relative concentration (X/Q) and relative deposition (D/Q) values presented in Tables 5.2-1 and 5.2-2 of the Environmental Report are for assumed continuous releases. Describe the development of diffusion and deposition estimates for non-continuous (intermittent or batch) routine releases.

421.0 Quality Assurance

421.1 (17.2) Identify or reference in Section 17.2 of the Bellefonte FSAR, those safety-related structures, systems, and components under the control of the QA program. (Ref: Page 17.2-16 of TVA Topical Report)

421.2 (17.2) TVA has elected to reference Section 17.2 of TVA Topical Report TR-75-01 entitled, "Quality Assurance Program Description for the Design, Construction, and Operation of TVA Nuclear Power Plants" for the operational phase of Bellefonte Nuclear Plant Unit Nos. 1 and 2. Since the NRC acceptance (September 1, 1977) of your topical report, certain egulatory uides have been revised (1.33 Rev. 1, 1.28 Rev. 2, 1.38 Rev. 2, 1.116 Rev. 0-R, and 1.123 Rev. 1) which would be applicable on or prior to the forthcoming docketing date for the Bellefonte application. Accordingly, TVA may consider revising their topical report to include the above regulatory guide revisions or provide a specific commitment in Section 17.2 of the Bellefonte application to comply with the regulatory position of the following regulatory guides and the requirements of the following ANSI Standard:

1.8, Rev. 1-R; 1.28, 6-7-72; 1.30, 8-11-72; 1.33, Rev. 1; 1.37, 3-16-73; 1.38, Rev. 2; 1.39, Rev. 2; 1.58, 8-73, 1.64, Rev. 2; 1.74, 2-74; 1.88, Rev. 2; 1.94, Rev. 1; 1.116, Rev. 0-R; 1.123, Rev. 1; and to ANSI N45.2.12, Draft 3, Rev. 4, 2-74.

Any exceptions, alternatives, or clarifications TVA believes are warranted should be clearly identified with sufficient supporting detail to allow review and NRC acceptance.

In addition, the Bellefonte FSAR should correct the apparent typographical error of TVA-TR75-01 to TVA-TR75-1A and reference the topical report by specific revision number. The FSAR should also include any plant specific exceptions or alternatives considered appropriate by TVA.

422.0 Conduct of Operations

422.1 (13.1.1) Describe the number of persons assigned to sections of the Power Protection Department that will provide support for the operation of the Bellefonte Nuclear Plant; a summary presentation of their qualifications; and the specific qualification of the principal supervisors of these sections.

Note: See Parts (2) and (3) of questions 422.5 of your Watts Bar application.

422.2 (13.1.3.2) Provide the qualifications of principal supervisory personnel of your plant staff (Figure 13.1.2-1) in resume format.