DCS-016

AUG 25 1982

Docket No. 50-447

General Electric Company
ATTN: Glenn G. Sherwood, Manager
Safety & Licensing Operation
Nuclear Power Systems Division
175 Curtner Avenue, Mail Code 682
San Jose, California 95125

Gentlemen:

Subject: Request for Additional Information Regarding the General Electric Application for an FDA for a Standardized Nuclear Island (GESSAR-II)

In our review of your request for a Final Design Approval (FDA) of your standardized nuclear island, we have identified a need for additional information (Enclosure 1). Our request for information addresses the areas reviewed by the Auxiliary Systems Branch, the Power Systems Branch, the Effluent Treatment Systems Branch, the Radiation Assessment Branch, the Structural Engineering Branch, the Hydrologic and Geotechnical Engineering Branch, the Procedures and Test Review Branch and the Chemical Engineering Branch. We request that you submit your responses to our questions by November 12, 1982. Where applicable, our positions regarding certain aspects of your proposed nuclear island have been identified.

We have noted several occurrences where your application does not reflect the resolution of open issues on facilities similar to your proposed nuclear island (e.g., Grand Gulf, Perry and Clinton) nor does it adequately address our formally published positions such as the Standard Review Plan. This has resulted in the need for additional information. Additionally, there is insufficient content in some areas of your application; e.g., in those portions reviewed by the Power Systems Branch. We recommend that you review the dockets of similar applications for operating licenses where we have recently issued SER's and SSER's and compare the design features of your proposed nuclear island with those of these comparable facilities. To assist you in responding, we have prepared a table (Enclosure 2) which presents comments on those portions of your application we believe need additional attention. As you will observe from this table, many of our questions are related to conformance with our regulatory guides, the Standard Review Plan or various NUREG documents. We further recommend that in responding to our questions, you address current staff positions.

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Recognizing the relatively compressed review schedule for your application, we suggest that you meet with us in about two weeks to discuss the more significant concerns we have identified in our review to date. Also, at that time, we will be prepared to discuss schedules for transmitting questions addressing the reactor systems, the containment systems and the instrumentation and control systems of your proposed nuclear island. We propose to use our questions for these three portions of your proposed design as the basic agenda for specific meetings on these review areas. A similar meeting regarding the mechanical engineering aspects of your proposed design is planned for early October. We expect these meetings to: (1) accelerate your response to our Round 1 questions; (2) identify earlier in the review process, those issues which may tend to remain open; and (3) seek an early resolution of these potentially open issues well in advance of the final SER preparation. This last item is the most significant consideration for these meetings since the present review schedule dictates that we achieve a resolution of most potentially open issues by mid-November of this year.

If you have any questions on these matters, please contact M. D. Lynch, at 301/492-9793.

Sincerely,

/s/

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosures:
As Stated

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Sincerely,

T M Novak

Thomas M. Novak, Assistant Director
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Division of Licensing

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"The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511."

GESSAR II

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AUG 25 1982

ENCLOSURE 1

Round 1 Questions on GESSAR-II

Docket No. STN 50-447

Structural Engineering Branch	220.01 to 220.44
Hydrologic and Geotechnical Engineering Branch	240.01 to 240.05
Chemical Engineering Branch	281.01 to 281.10
Auxiliary Systems Branch	410.01 to 410.42
Power Systems Branch	430.01 to 430.117
Effluent Treatment Systems Branch	460.09 to 460.18
Radiation Assessment Branch	471.04 to 471.20
Procedures and Test Review Branch	640.01 to 640.41

220.0 STRUCTURAL ENGINEERING BRANCH

220.01 (3.3.2) It is not clear in Section 3.3.2.2 of your FSAR how you combine the effects of the wind, the differential pressure and missiles all associated with a tornado. Clearly state the tornado loading combinations which you use in the design of all seismic Category I structures. A method of combining these effects which we find acceptable is given in Section 3.3.2 of the Standard Review Plan (SRP).

220.02 (3.3.2) In Section 3.3.2.1 of your FSAR, you state that you will vent the diesel-generator and auxiliary buildings. State whether the differential pressure associated with a tornado is transformed into an effective reduced pressure. If so, provide your proposed procedure to accomplish this.

220.03 (3.4.3) In Section 3.5.3.1 of your FSAR, you indicate that you use the modified Petry formula for local damage prediction of concrete barriers. You also indicate that your proposed design procedures have been substantiated by full scale impact tests conducted by the Sandia National Laboratory. State whether the thicknesses of the concrete missile barriers which will be established using your proposed design procedures will in no case be less than those listed in Table 1, Section 3.5.3 of the SRP.

220.04 (3.5.3) You state in Section 3.5.3.2 of your FSAR that you use an analysis procedure similar to that in Reference 6 (Williamson & Alvy) to determine an equivalent static load representing the tornado missile. Describe the actual procedure by which tornado generated missiles are transformed into effective loads. Verify that your proposed design procedure produces static loads comparable to those determined using the Williamson & Alvy formula.

220.05 (3.5.3) Submit details of the methods and assumptions which you use in the evaluation of the overall response of concrete and steel barriers subjected to impactive and impulsive loads. If you use the ductility ratio concept, indicate the ductility ratios you assume and verify that you meet the criteria delineated in Appendix A of Section 3.5.3, Revision 1, of the SRP.

220.06 (3.7.1) State in Section 3.7.1.2 of your FSAR, your frequency range and the actual frequency values you use in generating the response spectra from the synthetic records. Compare these with the frequency range and frequency values indicated in Item II.1.b of Section 3.7.1 of the SRP.

220.07 (3.7.1) In our review of Figures 3.7-7, 3.7-8, 3.7-13, 3.7-14, 3.7-19 and 3.7-20 of your FSAR, we note that for higher damping values, the response spectra from your synthetic time history are not in agreement with the enveloping values contained in Item II.1.6 of Section 3.7.1 of the SRP. Discuss in Section 3.7.1.3 of your FSAR, the effect of this apparent deviation from the response spectra contained in the SRP.

220.08 (3.7.1) In Section 3.7.1.3 of your FSAR, you correctly quote our position in Section C.3 of Regulatory Guide 1.61. However, it is not clear whether you have complied with our position on this matter. Accordingly, clearly state whether you comply with this portion of Regulatory Guide 1.61. If so, indicate the mechanism used to assure this compliance. If not, justify your position.

220.09 (3.7.1) Our position regarding the soil-structure interaction is contained in Item II.4 of Section 3.7.2 of the SRP and states that in addition to a finite element method of analysis, the elastic half-space method should also be used. Accordingly, provide in Section 3.7.1.4 and Appendix 3A of your FSAR, your procedure and the results from an analysis using the elastic half-space approach, including a discussion on the effect of variations in soil properties.

220.10 (3.7.2) In Section 3.7.2.1.5.1.1 of your FSAR, you state that a study has been conducted which shows that the interaction between the steel containment vessel and the polar crane can be ignored and that the crane mass can be lumped into the containment model at that level. Provide this study.

220.11 (3.7.2) At the time of this review, Appendix 3H which decribes the effect of the concrete between the containment and the shield building on the seismic analysis, is not available. Indicate when this appendix will be provided. This information should be made available prior to the forthcoming structural audit in December 1982.

220.12 (3.7.2) Your decoupling criteria between systems and subsystems are not clear in the discussion provided in Section 3.7.2.3 of your FSAR. Accordingly, demonstrate that your decoupling criteria are either equivalent to, or more conservative than, those given in Item II.3.b of Section 3.7.2 of the SRP.

220.13 (3.7.2) It is not clear in the discussion provided in Sections 3.7.2.3 and 3.7.2.5 of your FSAR how you have accounted for the vertical flexibility of floors in the generation of the vertical response spectra. Accordingly, provide the procedures you have used to account for this phenomenon.

220.14 (3.7.2) In Section 3.7.2.11 of your FSAR, you indicate a method of analysis for torsional effects in your models. However, our position is that an additional eccentricity of five percent of the maximum building dimension at the level under consideration, be assumed in the design of seismic Category I structures to account for accidental torsion. This extra eccentricity is in addition to that which results from the actual geometry and mass distribution of the building. (Refer to Item II.11 in Section 3.7.2 of the SRP). State whether you comply with our position on this matter or whether you will pursue another method.

220.15 (3.7.3) Indicate in Section 3.7.3.10 of your FSAR whether, in performing a static analysis in lieu of the vertical dynamic analysis, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. (Refer to Items II.1b(3) and II.10 of Section 3.7.2 of the SRP.)

220.16 (3A3.1) For the fixed base cases (i.e., the plant founded on rock), describe the input motion you use at the base of the structure. Indicate whether the motion for the fixed base case was deconvolved from plant grade. Indicate how you account for the effect of embedment in this case.

220.17 (3.7.2) Describe your procedure to compute the dynamic lateral earth pressure and the hydrodynamic groundwater pressure during a seismic event.

220.18 (3.7.2) Describe the procedures used in the seismic analysis of the polar crane. Discuss how you account for the effects of cable jerking.

220.19 (3.7.4) Describe your proposed in-service surveillance program for the seismic instrumentation, including a discussion of your proposed in-service inspection, testing and calibration. A program which we find acceptable is contained in Item II.5 of Section 3.7.4 of the SRP.

220.20 (3.8.2) (3.8.3) (3BA8.4) Provide the following information applicable to pool dynamic loads, their load combinations and the analysis of these loads:

a. The procedures used to generate the in-structure response spectra at critical locations such as the reactor pressure vessel supports. Discuss how the effects of soil-structure interactions are accounted for in this analysis.

b. The extent, if any, to which structures adjacent to the reactor building will experience the effects of these loads.

c. Your procedures for combining static and alternating dynamic loads (Section 3BA.8.4) do not agree with our positions on this matter. (Refer to Sections 3.8.2 and 3.8.3 of the SRP.) Discuss the effect of this deviation. In addition, indicate whether your method of analysis includes the effects of fluid-structure interaction in the manner specified in the last paragraph of Item II.3.a of Section 3.8.3 of the SRP; i.e., whether you comply with the Appendix to Section 3.8.1 of the SRP. (Refer to Question 220.23)

d. Describe the analysis performed to determine the effects of negative pressures in the suppression pool on the containment and drywell lower liner plates, particularly when combined with the effects of high temperatures, seismic loads and cracking of the concrete.

220.21 (3.8.2) In Section 3.8.2.3.15 of your FSAR, you state that the structural design criteria for the steel containment vessel are consistent with our positions in Regulatory Guide 1.57. However, the stress intensity limits for various loading combinations presented in Table 3.8-2 of your FSAR do not clearly depict this. Accordingly, present these limits in a tabular form similar to that of Table 3.8.2-1 in Section 3.8.2 of the SRP. Verify that your stress intensity limits are consistent with our values in the SRP.

220.22 (3.8.2) In Table 3.8-1 of your FSAR, you present the proposed loading combination for the design of the steel containment vessel. However, the contents of this table are not clearly consistent with load combinations which are acceptable to us. Accordingly, provide the loading combinations in a tabular form which is consistent with the load combinations contained in Item II.3.b of Section 3.8.2 of the SRP. Verify that your proposed loading combinations are in agreement with those contained in the SRP.

220.23 (3.8.2) In your proposed design and analysis procedures presented in Section 3.8.2.4 of your FSAR for the steel containment vessel, it is not clear how you have treated the nonaxisymmetric loads and the transient loads. Provided a detailed discussion of your procedures on these matters. (Refer to Part (c) of Question 220.20.)

220.24 (3.8.2) The staff will review the ultimate capacity of the containment vessel with respect to internal pressure build-up due to accidents when we review the GESSAR PRA. However, for our review of your application for an FDA, state in Section 3.8.2.4 of your FSAR whether your proposed design of the steel containment vessel complies with our position on this matter as outlined in Item II.4.d of Section 3.8.2 of the SRP. You should be prepared to discuss this matter in detail at the forthcoming structural audit in December 1982.

220.25 (3.8.2) Provide in Section 3.8.2.4 of your FSAR, a discussion of the localized deformations at penetrations in the steel containment vessel due to the internal pressure build-up resulting from postulated accidents. Discuss the effect of these internal pressure loads resulting from postulated accidents on the leak rates at the penetrations in the containment vessel.

220.26 (3.8.2) In Appendix 3F of your FSAR, you state that you use a value of 2.0 for the factor of safety against buckling which conforms to our position on this matter in Regulatory Guide 1.57. However, our current position differs from that presented in this regulatory guide and is provided in Attachment 1 to this set of questions. The factors of safety against buckling of steel containment vessels which we now find acceptable are:

a. For design conditions and Level A and B services limits, use a factor of safety of 3.0.

b. For Level C service limits, use a factor of safety of 2.5.

c. For Level D service limits, use a factor of safety of 2.0.

The safety factors cited above are independent of the knockdown factor. This factor is used to reduce to experimentally determined values of buckling stress, the calculated buckling stress obtained from the classical theory of buckling based on small displacements of a shell assumed to have no structural imperfections. Verify that your analyses of the steel containment vessel meet our current positions regarding the required factors of safety against buckling.

220.27 (3.8.3) (3.8.4) (3.8.5) In Sections 3.8.3, 3.8.4 and 3.8.5 of your FSAR, you state that the design of concrete internal structures, other seismic Category I structrues and foundations is performed in accordance with the requirements of ACI-318 (1971). Our present position on this matter is that you should use ACI-349, as augmented by Regulatory Guide 1.142. Evaluate and assess the impact of satisfying our position on this matter. Identify specific deviations from our present position. Indicate those areas where use of the ACI-318 (1971) Code produces a less conservative design. Discuss specific means for modifying those portions of your proposed structures which are less conservatively designed. Alternatively, justify their design adequacy.

220.28 (3.8.3) Item (5) in Section 3.8.3.3.1.3.2 of your FSAR is the factored load combination for the abnormal/severe environmental condition and is given as:

$$D + L + F_{ego} + H_a + T_a$$

However, Item II.3.f of Section 3.8.3 of the SRP states our position that you should use Subsection CC-3000 of the ASME, Section III, Division 2 Code, which presents the corresponding load combination as:

$$D + L + F_{ego} + H_a + T_o$$

Explain this discrepancy. Verify that your load combination complies with our position on this matter.

220.29 (3.8.3) In Section 3.8.3.5.1 of your FSAR, you state that a high degree of leak-tightness for the drywell is not a requirement since the drywell is not a fission product barrier and moderate leakage under accident conditions is tolerated by the pressure suppression process. State what degree of leakage is considered tolerable and indicate the leak rates at the drywell head, the equipment hatch and the personnel lock when the internal pressure build-up reaches the ultimate capacity of the drywell pressure boundary.

220.30 (3.8.3) In Section 3.8.3.4.1.4 of your FSAR, you state that tangential shear from the drywell vent plates is transferred to the drywell base plate and in turn is transmitted to the foundation concrete through the shear lugs under the plates. Indicate the allowable values of tangential shear stress you have used. Verify whether your proposed allowable shear stresses comply with our position in Item II.5.a of Section 3.8.1 of the SRP.

220.31 (3.8.2) Discuss, from a consideration of buckling, the effect of a postulated pipe break in the annulus region between the shield building and the containment vessel. Indicate to what elevation this could flood the annulus, thereby causing an external hydrostatic pressure on the steel containment vessel.

220.32 (3.8.3) In Section 3.8.3.3.6.2.1 of your FSAR, you state that the load combination for service load conditions of concrete internal structures are:

$$S = D + L + T_o + R_o \qquad (3.8\text{-}3)$$

$$S = D + L + T_o + R_o + F_{ego} \qquad (3.8\text{-}4)$$

However, our position on this matter is contained in Item II.5 of Section 3.8.3 of the SRP which states that the stress limits for these cases to be 1.3 S. Indicate whether your proposed design of internal concrete structures satisfies our position in the SRP on this matter.

220.33 (3.8.3) In Section 3.8.3.3.6.3.2 of your FSAR, you indicate that you satisfy three out of the four load combinations presented in Item II.3.c (ii)(a) of Section 3.8.3 of the SRP for the factored load conditions for steel structures using the elastic working stress design method. State why you omitted Equation (4) of Item II.3.c(ii)(a) and verify that you satisfy our position on the load combination represented by Equation (4).

220.34 (3.8.3) Describe the analytical and design techniques you use to determine the effect of annulus pressurizaton loads on the shield wall surrounding the reactor vessel. Indicate in this description how these pressurization loads are combined with other coincident loads, including the seismic loads and the LOCA and/or SRV loads assumed to be occurring coincidently in the suppression pool.

220.35 (3.8.3) For materials, quality control and special construction techniques, you state in your FSAR that you satisfy the requirements of the ACI-318 (1971) Code. Indicate in Section 3.8.3.6 of your FSAR how you satisfy the requirements of ACI-349, as augmented by Regulatory Guide 1.142, which is our current position for the design of seismic Category I structures other than containment. Identify specific deviations from our position on this matter and justify the design adequacy for such areas.

220.36 (3.8.3) In Section 3.8.3.7 of your FSAR, although certain of your test requirements are acceptable to us, there are some portions in the description of your proposed testing which differ from our position on the testing of the concrete and steel internal structures of the containment. Our position on testing and in-service surveillance requirements for the drywell in a Mark III containment is presented in Item II.7 of Section 3.8.3 of the SRP. Verify that your proposed test procedures in the FSAR comply with our position on this matter in the SRP.

220.37 (3.8.3) (3.8.4) In Section 3.8.3 and 3.8.4 of your FSAR, revise your list of applicable codes and standards to include Regulatory Guides 1.94, 1.115 and 1.142, as applicable. Identify any exceptions and deviations you have taken and provide justification for them.

220.38 (3.8.4) In Section 3.8.4 of your FSAR, you don't indicate whether masonry construction is utilized in your proposed structures. If seismic Category I masonry walls will not be used in your proposed design, so indicate. If you will use seismic Category I masonry walls, identify any differences between the criteria for safety-related masonry walls which we find acceptable (refer to Appendix A in Section 3.8.4 of the SRP) and your proposed criteria for materials, testing, analysis, design and construction of this type of structure.

220.39 (3.8.4) In Section of 3.8.4.3.2.3 of your FSAR, the load combination in Equation 3.8-40 includes the SSE. We beleive that you actually intend this load combination to include the OBE instead of the SSE, similar to the combination presented in Item II.3.b(i)(a) of Section 3.8.4 of the SRP. If this equation is in error, correct it. If this equation is not, state why you consider this load combination.

220.40 (3.8.4) In Section 3.8.4.1.3 of your FSAR, discuss in detail the design of your proposed spent fuel pool racks. Explain how the racks are attached to the fuel pool and indicate how you ensure that these racks withstand seismic forces. Our positions on this matter are attached for your use (Attachment 2). Modify your analysis and design, if necessary, to comply with our positions.

220.41 (3.8.4) In Section 3.8.4 of your FSAR, you have not furnished information regarding the design and analysis of the cable tray and conduit supports. Describe in detail the methods used in the design and analysis of seismic Category I cable tray and conduit supports, including references to the codes and standards which you propose to use.

220.42 (3.8.5) Our position regarding the foundation design of all seismic Category I structures is presented in Item II.3 and II.5 of Section 3.8.5 of the SRP and states that some additional load combinations should be checked to determine if the factors of safety against sliding, overturning and floatation are within acceptable limits. It is not clear in your FSAR whether you have checked these additional load combinations. Verify that the foundations of all seismic Category I structures are analyzed for these additional loading combinations (i.e., Item II.3) and ensure their design adequacy (Item II.5).

220.43 (3.8.5) Your calculated factors of safety for seismic Category I structures against sliding, overturning and floatation are given in Figure 3.8-75 of your FSAR. We note that you state the factors of safety against sliding for the reactor, the auxiliary and the control buildings are 1.01, 1.02 and 1.04, respectively. Inasmuch as these values are below our minimum acceptance criteria of 1.1, we find them unacceptable. Accordingly, revise your proposed design and demonstrate with calculations, including all your assumptions, that you satisfy our acceptance criteria on this matter. Coordinate your response to this question with your response to the Question 220.42. (This question is similar to and replaces Question 241.11. Accordingly, your response to Question 241.11 should cross-reference your response to this question.)

220.44
(3A.5.2)
(Fig.3A-18)
In Section 3A.5.2(1) of your FSAR, you indicate use of a deconvolution analysis (i.e., FLUSH) to determine the motion which would have to be developed in an underlying bedrock formation to produce the specified control motion at the finished grade in the free field. We consider this approach not sufficiently conservative and, therefore, unacceptable. Our position on this matter is that the control motion should be applied at the foundation level in the free field when performing a deconvolution analysis. Indicate whether your analysis will conform to our position on this matter. (Refer to Item II.4.iii of Section 3.7.2 of the SRP.) In responding to this question, cross-reference to your response to Question 220.09.

AUG 25 1982

INTERIM CRITERIA FOR EVALUATING STEEL CONTAINMENT BUCKLING

I. INTRODUCTION

During the review of the subject buckling issue, it was found that there exists a considerable confusion regarding buckling analysis and associated safety factor. Both industry and NRC criteria are far from satisfactory.

ASME recently proposed buckling criteria for steel containment. In spite of our long effort to resolve what appears to be a fundamental problem in the ASME proposed criteria, we were unable to come to an agreement. We, therefore, recommended Mr. R. Bosnak to cast a negative vote when presented to ASME full committee for an adaption (Ref. letter from S. B. Kim to R. Bosnak). It was later adapted as a code case for a trial basis (N 284).

SEB, with the help of RES, is working on a long term project to resolve the issue of containment buckling. However, in view of current status of confusion and because of the fact that the final position will take some time, we have provided an interim position.

II. REVIEW AREAS

The following areas relating to steel containments are reviewed.

1. Description of the Containment
   In addition to the description of the steel containment provided in Section 3.8.2, further information concerning buckling such as shell imperfection, stiffeners, opening reinforcement (cut out and penetration) and stress strain curves are reviewed.
2. Loads and Load Combinations
   Information pertaining to the applicable loads and load combination is reviewed. The basic loads and load combinations are found in SRP's 3.8.2. Latest version of the documents will be used.
3. Area of Evaluation
   For containment vessel with large cut out such as equipment hatch, the effect of such penetration should be considered in the buckling analysis. The analysis should take into consideration of the effect of ring stiffeners as well as stringer stiffeners. The evaluation should include local as well as over all structural buckling.

III. ACCEPTANCE CRITERIA

The acceptance criteria for each of the review areas are given below. Alternate approaches to satisfying the review areas may be proposed and will be reviewed.

1. Description of Containment

The description should contain sufficient information so that one can perform an independent analysis. In particular, a detail of stiffeners and imperfection as well as stress and strain curves of the materials used is needed.

2. Loads and Loading Combinations

The loads and load combinations are reviewed for conformance to the staff positions. An acceptable list of load combinations is contained in Standard Review Plan, Section 3.8.2.

3. Evaluation

a. When an evaluation assumes that the geometry of the containment is axisymmetric even though there is a large cut out for equipment hatch, the acceptance of the evaluation is contingent upon the following two provisions.

i). locally the cut out should be reinforced to the original (without cut out) buckling strength;

ii). 25% of the reduction from the calculated strength be provided to account for the local effect on overall structural symmetry.

b. When an evaluation account for the cut out effect by means of an asymmetric geometric model, no reduction of buckling capacity*/ is needed. However, the adequacy of such three dimensional method should be demonstrated to the satisfaction of the staff. A confirmatory verification to the ongoing NRC sponsored Los Alamos Lab buckling test results by the 3D computer code is one acceptable way for fulfilling this requirement.

*/ The reduction of the buckling capacity refers to the item described in Section III, 3, a, ii above and it is independent of knockdown factor (a definition of the knockdown factor is given in conjunction with safety factor next).

c. Factor of Safety

The following factor of safety will be acceptable.

(a) For Design Conditions and Level A and B Services Limits use FS = 3.0

(b) For Level C Service Limits use FS = 2.5

(c) For Level D Service Limits use FS = 2.0

The above safety factors are independent of knockdown factor. The knockdown factor here is determined as a factor that is used to reduce calculated results from classical shell buckling theory based on small displacement and perfect shell to experimentally determined values.

TECHNICAL POSITION ON SPENT FUEL POOL RACKS

Introduction

The purpose of this appendix is to provide minimum requirements and criteria for review of spent fuel pool racks and the associated structures which would meet the design standards specified in subsection II of this SRP section.

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plants and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool, the spent pool liner fuel, and the racks. The main safety function of the spent fuel pool, including the liner, and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed by the Accident Evaluation Branch (AEB); Structural Engineering Branch accepts the findings of the AEB review for the purpose of review of the integrity of the racks and the fuel pool including the fuel pool lines due to a postulated fuel handling accident. Sketches and sufficient details of the fuel handling system should be provided to facilitate this review.

(2) Applicable Codes, Standards, and Specifications

Construction materials should conform to Section III, Subsection NF of Ref. 3.1. All materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon Subsection NF requirements of Ref. 3.1 for Class 3 component supports.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor responses spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in SRP Section 3.7. The ground response spectra and damping values should correspond to Regulatory Guides 1.60 and 1.61, respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guides 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

The peak response from each direction should be combined by square root of the sum of the squares in accordance with Regulatory Guide 1.92. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

Submergence in water may be taken into account. The effects of submergence are considered on case-by-case basis.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to the applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and base slab. Temperature gradient across the rack structure due to differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The fuel pool racks, the fuel pool structure including the pool slab and fuel pool liner, should be evaluated for accident load combinations which include the impact of the spent fuel cask, the heaviest postulated load drop, and/or accidental drop of fuel assembly from maximim height.

The acceptable limits (strain or stress limits) in this case will be reviewed on a case-by-case basis but in general the applicant is required to demonstrate that the functional capability and/or the structural integrity of each component is maintained.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of SRP Section 3.8.4, subsection II.3, and Table 1.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: The methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with SRP Section 3.8.4, subsection II.4 are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach, two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and

(b) A static analysis of the rack system by subjecting it to the maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the applicant to use equivalent mass and stiffness properties obtained from calculations similar to those described in Ref. 4.1. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

The structural acceptance criteria are those given in the Table 1. When buckling loads are considered in the design, the structural acceptance criteria shall be limited by the requirements of Appendix XVII to Reference 3.1.

For impact loading, the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modulus under all probable service conditions shall be in accordance with SRP Section 3.8.5, subsection II.5. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by SRP Section 3.8.5, subsection II.5.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

The fuel pool structure should be designed for the increased loads due to the new and/or expanded high density racks. The fuel pool liner leak tight integrity should be maintained or the functional capability of the fuel pool should be demonstrated.

(7) Materials, Quality Control, and Special Construction Techniques

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during the construction phase should be provided.

If connections between the rack and the pool liner are made by welding, the welder as well as the welding procedure for the welding assembly shall be qualified in accordance with the applicable code.

TABLE 1

LOAD COMBINATION	ACCEPTANCE LIMIT
$D + L$	Normal limits of NF-3231-1a, ASME Code
$D + L + T_o$	Normal limits of NF-3231-1a, ASME Code
$D + L + T_o + E$	S_y
$D + L + T_a + E$	S_y
$D + L + T_o + P_f$	S_y
$D + L + T_a + E'$	Faulted condition limits of NF 3231-1c
$D + L + F_d$	The functional capability of the fuel racks should be demonstrated
Limit Analysis:	
$1.7\ (D + L)$	XVII 4000 of ASME ASME Code Section III
$1.3\ (D + L + T_o)$	
$1.7\ (D + L + E)$	
$1.3\ (D + L + E + T_o)$	
$1.3\ (D + L + E + T_a)$	
$1.3\ (D + L + T_o + P_f)$	
$1.1\ (D + L + T_a + E')$	

Notes:

1. The abbreviations in the table above are those used in subsection II.3.a of this SRP section where each term is defined except:
   T_a is defined here as the highest temperature associated with the postulated abnormal design conditions.
   F_d is the force caused by the accidental drop of the heaviest load from the maximum possible height
   P_f is upward force on the racks caused by postulated stuck fuel assembly.
   S_y is the yield stress for the material as tabulated in Appendix I, ASME Code.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 of Reference 3.1 shall be amended by the requirements of paragraphs c.2,3 and 4 of Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

VI. REFERENCES

1. Regulatory Guides

1.29 - Seismic Design Classification

1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

1.61 - Damping Values for Seismic Design of Nuclear Power Plants

1.76 - Design Basis Tornado for Nuclear Power Plants

1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan Section

3.7 - Seismic Design

3.8.4 - Other Category I Structures

3. Industry Codes and Standards

1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1

2. American National Standards Institute, N210-76

3. American Society of Civil Engineers, Suggested Specification for Structures of Aluminum Alloys 6061-T6 and 6067-T6

4. The Aluminium Association, Specification for Aluminum Structures

4. Other

1. Briggs, John M., "Introduction to Structural Dyanmics," McGraw-Hill Book Co., New York, 1964.

240.0 HYDROLOGIC AND GEOTECHNICAL ENGINEERING BRANCH

240.01 (2.4.1) You state in Section 2.4.1.1 of your FSAR that the total design of safety-related structures is compatible with plant sites having groundwater levels up to two feet below grade. Indicate the actual design basis groundwater level. In this regard, some plants select plant grade for the design basis groundwater level to conservatively bound groundwater fluctuations or to account for nearby flooding effects even though the ambient groundwater level may be somewhat lower. State whether your proposed design will be modified on a site specific basis to accomodate the plant under these circumstances.

240.02 State whether the groundwater level which will be used as the design basis for subsurface hydrostatic loading will also be used in combination with other extreme environmental loadings such as an earthquake or a tornado or whether a lower groundwater level will be used. If a lower groundwater level is to be used in your proposed standardized design as the design basis for extreme environmental loadings, indicate what this level will be. Alternatively, indicate whether this level will be site specific. If so, state the interface requirement for this site specific requirement. If the combined loadings are site specific, state the purpose of having a standardized design basis groundwater level which is two feet below plant grade for hydrostatic loading only.

240.03 (2.4.1) State in Section 2.4.1.1 of your FSAR whether the design basis flood level established at one foot below plant grade includes coincident wind-generated wave activity. If not, indicate how this will be accomodated in your proposed design. Indicate the wave runup your proposed design can withstand.

240.04 (2.4.2) State in Section 2.4.2.3 of your FSAR whether you plan to have parapets on the roofs of the safety-related structures. If so, indicate whether the parapets will have scuppers or openings to limit the depth of water buildup resulting from a local Probable Maximum Precipitation (PMP). State the design basis load on roofs. Indicate the maximum short duration rainfall intensity that the scuppers or openings can handle. State what credit is taken for roof drains in determining this rainfall intensity. You should note that we assume roof drains are blocked with debris during the design basis event.

240.05 (2.4.11) State whether the ultimate heat sink will be a site specific item with regard to the source of emergency cooling water or whether there will be some standard components such as mechanical draft cooling towers, cooling ponds or spray ponds.

281.0 CHEMICAL ENGINEERING BRANCH

281.01 (5.4.8) Recognizing that resins may enter the reactor recirculation system in the event of a failure of a filter-demineralizer resin support septum, we established a design criterion (Item II.2.f) in Section 5.4.8 of the Standard Review Plan (SRP) that a strainer should be provided on the outlet of each filter-demineralizer unit. In addition, we established a design criterion in the SRP that the reactor water cleanup system (RWCS) should have provisions for monitorin differential pressures to assure that the design limits on filter-demineralizer septums and resin strainers are not exceeded. Describe how your design is consistent with these requirements.

281.02 (5.4.9) Your description of the RWCS does not indicate that you will use a holding pump to maintain flow through each filter-demineralizer in the event of low flow or loss of flow in the system. Indicate whether you propose to use a holding pump in the system or plan to achieve this function in some other manner. (Refer to Item II.2.c of Section 5.4.8 of the SRP).

281.03 Verify that provisions have been made for draining and venting the components of the RWCS through a closed system in accordance with the requirements of General Design Criteria (GDC) 60 and 61 of Appendix A to 10 CFR Part 50.

281.04 (6.1.1) Demineralized water from the condensate storage tank or the suppression pool, with no additives, is used in the containment sprays and to inject core cooling water. Indicate the limits you will place on the conductivity, the chlorides and the pH of this water to minimize stress corrosion cracking of unstabilized austenitic stainless steel components.

281.05 (6.1.2) Indicate the total amounts of protective coatings and organic materials inside containment which do not meet the requirements of ANSI N101.2 (1972) and which do not comply with our position in Regulatory Guide 1.54. Evaluate the generation rates and total quantity of combustible gases that can be formed from these unqualified organic materials in the event of a design basis accident (DBA). Evaluate the volume of solid debris which can be formed from these unqualified organic materials under DBA conditions and which can reach the containment sump. Provide the technical basis and the assumptions you use for this evaluation.

281.06 (9.1.3) Describe the samples to be taken and the instrument readings, including their frequency of measurement, which will be used to monitor the water purity in the spent fuel pool (SFP) and to determine when the SFP cleanup system demineralizer resin and filter will need replacement. State the chemical and radiochemical limits of the SFP water which will initiate corrective actions, including the basis for establishing these limits. Your response should consider such variables as: boron concentration; gross gamma and iodine activity; demineralizer and/or filter differential pressure; demineralizer decontamination factor, pH; and crud level.

281.07 In Item II.1.a of Section 9.3.2 of the SRP, we state in part that the atmosphere and sumps inside containment should be sampled in order to satisfy the requirements of the relevant GDC. Accordingly, describe the provisions to sample inside containment in accordance with the requirements of GDC 64 of Appendix A to 10 CFR Part 50. Indicate how your design is consistent with the provisions of Regulatory Guide 1.97, Revision 2.

281.08 (9.3.2) In Item 3.f of Section 9.3.2 of the SRP, we state that there should be passive flow restrictions to limit reactor coolant loss in the event of a rupture of the sample line. However, this criterion is not addressed in your FSAR. Accordingly, describe how your design is consistent with the design philosophy of maintaing exposures to "as low as is reasonable achievable" (ALARA) in the event of a rupture of the sample line containing contaminated primary coolant. The staff's position on this matter is also contained in Section C.2.1 (C) of Regulatory Guide 8.8, Revision 3 (June 1979).

281.09 (9.3.2) Provide information demonstrating that you satisfy the requirements of Item II.B.3," Post Accident Sampling Capability," of NUREG-0737. Specifically, demonstrate the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples, without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during, and following, an accident in which there is no core degradation. Additionally, you should: (1) review and modify, as necessary, your sampling, chemical analysis and radionuclide determination capabilities to comply with NUREG-0737, II.B.3; (2) provide us with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the requirements have been met. Materials to be analyzed and qualified include certain radionuclides that are indicators of the severity of core damage (e.g., noble gases, iodines, cesium and non-volatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron and chlorides in reactor coolant samples in accordance with the requirements of NUREG-0737.

In your detailed response, address the following ten matters:

a. Your compliance with all requirements of NUREG-0737, II.B.3, for sampling, chemical and radionuclide analysis capability, under accident conditions.

b. Shielding to meet the requirements of GDC-19, assuming Regulatory Guide 1.4 source terms.

c. Your compliance with the sampling and analysis requirements of Regulatory Guide 1.97, Revision 2.

d. Verify that all electrically powered components associated with post-accident sampling are capable of being supplied with power and operated within thirty minutes of an accident in which there is core degradation, assuming a loss of off-site power.

e. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

f. Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

g. State the design and/or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from on-line gas analysis equipment if it is used.

h. Provide a method for verifying that reactor coolant dissolved oxygen is less than 0.1 ppm if reactor coolant chlorides are determined to be greater than 0.15 ppm.

i. Provide information on: (1) testing frequency and type of testing to ensure long term operability of the post-accident sampling system; and (2) operator training requirements for post-accident sampling.

j. Demonstrate that your proposed sample locations in the reactor coolant system and suppression pool will yield results which are representative of core conditions.

Your response should contain sufficient documentation to demonstrate compliance with our requirements on this matter. In addition to the information requested above, we request that you submit data supporting the applicability of each selected analytical chemistry procedure or on-line instrument. In the event our generic review determines a specific procedure is unacceptable, we will require you to make modifications as determined by our generic review.

281.10 (9.1.2) Provide the following information about your high density neutron absorber racks which you proposed to use for spent fuel storage:

a. Indicate the nature of the neutron absorber materials to be incorporated into these racks.

b. State whether the compartments in the racks containing the neutron absorber materials are vented or are exposed to the spent fuel pool environment.

c. Provide additional information on the frequency of inspection and the type of sampling used in monitoring this system.

410.0 AUXILIARY SYSTEMS BRANCH

410.01 (3.4.1) In Section 3.4.1.1.2 of your FSAR, you state that in "(flooding) cases involving visual inspection of the affected areas followed by a remote or local operator action, a minimum of 30 minutes is allowed for the operator to take action." This implies that some areas of the plant may be protected against internal flooding sources only by visual operator inspection. If any of these areas are required for safe cold shutdown, revise your design so that positive means of flood detection are provided. Identify which areas of the plant rely on visual detection and verify that failure to discover the flooding condition will not result in flooding of safety-related equipment.

410.02 (3.4.1) All of your flooding analyses in Section 3.4.1.1.2 of your FSAR are based on either high-energy line breaks or leakage cracks in moderate-energy piping systems. Verify that flooding due to complete failure of a non-seismic Category I tank or piping system cannot result in conditions worse than those which you have analyzed. Note that complete piping system failures should be postulated in non-seismic moderate-energy piping systems rather than leakage cracks if the complete failure represents the worst case. As an example, your analysis of flooding in the control building assumes that the largest possible pipe break is from a crack in the six inch fire protection line. Verify that the fire protection piping in question is seismic Category I or analyze the consequences of a complete pipe break.

410.03 (3.4.1) In your flooding analyses of the steam tunnel, safe shutdown of the plant depends upon water level detection and normally closed isolation valves in the floor drainage system. With respect to these analyses, provide the following information:

a. Verify that your proposed detection system is designed to safety-grade requirements.

b. Verify that your proposed drainage system up to, and including the normally closed isolation valves, is designed to seismic Category I requirements.

c. Provide a Technical Specification or an interface requirement for a Technical Specification that the drainage system valves be locked in the closed position and verified closed as part of a monthly surveillance program.

410.04 (3.4.1) In your flooding analysis of the fuel building, you state that a crack postulated in the eight inch fuel pool cooling system line between the shutoff valve and the fuel storage pool can result in leakage of a large quantity of water from the pool with a potential for an unacceptable long-term loss of cooling. You further state that operator action (e.g., removal of a screen and installation of an inflatable plug) will be relied upon to correct this condition and that the dose rate calculated at the surface for plug installation is less than 10 mrem/hr.

a. Since you indicate that the fuel pool level will be maintained at its normal level, explain how the operator will install the inflatable plug.

b. Describe how the leak is detected and identify the time available for the operator to secure the leak, thereby limiting the total leakage to about 6800 cubic feet as you have indicated.

c. Verify that this leakage water will not damage any safety-related equipment. Describe where the water accumulates and how it is drained.

d. Verify that the calculated dose rate is based on your new high density spent fuel storage configuration.

410.05 (3.5.1) Provide in Section 3.5.1.1 of your FSAR, the results of your analysis to verify that the turbine drive of the reactor core isolation cooling (RCIC) system is not a missile source. Alternatively, verify that missiles from the turbine cannot damage safety-related equipment.

410.06 (3.5.1) With respect to internally generated missiles inside containment, evaluate the effects of gravitational missiles such as fuel handling equipment which may be generated by a seismic event. Provide in Section 3.5.1.2 of your FSAR, the results of your evaluation and verify that both safety-related equipment and stored fuel are protected in an acceptable manner.

410.07 (3.5.1) In addition to the possible missile sources you have identified, verify in Section 3.5.1.2 of your FSAR, that your analyses inside containment have included the reactor vessel head bolts and the automatic depressurization system (ADS) accumulators.

410.08 (3.5.2) Verify that the seismic Category I charcoal delay tanks are protected against tornado missiles. Alternatively, provide justification for the tanks not being protected.

410.09 (3.6.1) (RSP) Demonstrate your compliance with the design criteria contained in Branch Technical Position ASB 3-1, attached to Section 3.6.1 of the Standard Review Plan (SRP), in accordance with the implementation section of ASB 3-1. Alternatively, demonstrate your compliance with Appendix C to ASB 3-1. Identify where your criteria differ from the criteria contained in the documents cited above. Provide justification for any deviations.

410.10 (3.6.1) In Section 3.6.1.1.3 of your FSAR, you state that where a pipe break event occurs in one of two or more redundant divisions or trains of an essential system, a single failure in the other trains or divisions of that system is not assumed, provided certain criteria are met. It is our position that the above single failure exclusion following a pipe break may only be used for a postulated crack in dual-purpose moderate-energy systems as defined in Branch Technical Position ASB 3-1. Verify that for all other systems, a single active failure can be assumed following a pipe break or crack and that safe shutdown will not be precluded.

410.11 (3.6.1) Your assumption in Section 3.6.1.1 of your FSAR that only seismic Category I piping systems can be used to mitigate the consequences of a postulated pipe break may be unduly restrictive when used in conjunction with Branch Technical Position ASB 3-1. Your assumption is necessary when considering breaks in non-seismic Category I systems but it is not necessary for breaks in seismic Category I systems. Any non-seismic Category I system which will be available following a break in a seismic Category I system, may be relied upon to mitigate the consequences of that break.

410.12 (3.6.1) (RSP) Your separation analyses in Section 3.6.1.3 of your FSAR is based on consequences which you find acceptable as a result of damage to only one division of a redundant system. These analyses are unacceptable since you did not consider a single active failure. Accordingly, revise your analyses to include protection against postulated high-energy system pipe breaks coincident with a single active failure.

410.13 (3.6.1) (RSP) Revise Appendix 3G of your FSAR to consider single active failures coincident with postulated pipe breaks in all the high-energy systems analyzed. For all instances where a redundant system is relied upon in the event of a pipe break, verify that the single failure criterion is met. For example, in Section 3G.2 you state that Division 2 reactor heat removal (RHR) system piping and Division 1 ADS piping could be damaged due to a high-energy pipe break but, since each has a redundant system, no protection is required. It is our position that you must provide protection or demonstrate that a single active failure of one of the redundant systems is acceptable.

410.14 (3.6.1) Appendix 3G to your FSAR does not include a pipe failure analysis of the main steam and feedwater lines inside the main steam tunnel. Accordingly, revise your FSAR to include these analyses. Identify the equipment in the main steam tunnel which must be environmentally qualified for these postulated pipe breaks.

410.15 (3.6.1) In Section 6.2 of your FSAR, you provide the results of subcompartment pressure analyses for some areas outside containment which are considered part of the secondary containment. In order that we may evaluate the adequacy of the environmental qualification of the equipment in these subcompartments, provide the temperature profiles resulting from these postulated pipe breaks. Verify that the equipment necessary to mitigate the consequences of a postulated pipe break, including a single active failure, will be environmentally qualified. Perform additional analyses for any safety-related areas outside containment which are not considered part of the secondary containment.

410.16 (4.6) In your letter dated February 12, 1982, you state that the review base for Section 4.6 of your FSAR is the Clinton plant. Revise your FSAR to include the additional information provided on the Clinton docket in the course of the Clinton review, including that additional information which was submitted to close the open items in this portion of the Clinton SER.

410.17 (5.2.5) Verify that there are no differences between your reactor coolant pressure boundary and your proposed ECCS leakage detection system and those which we have reviewed and accepted on the Clinton docket. Revise your FSAR, as necessary, to be consistent with Clinton.

410.18 (6.7) Revise Section 6.7 of your FSAR to reference Regulatory Guide 1.96 instead of Branch Technical Position APCSB 6-1 since this regulatory guide has replaced APCSB 6-1. Address all of the acceptance criteria contained in Section 6.7 of the SRP.

410.19 (9.1.1) (9.1.2) In your letter of February 12, 1982, you state that the new and spent fuel storage facilities which you propose for your nuclear island are the same as those for the Perry Nuclear Power Plant. However, your FSAR describes high density new and spent fuel storage facilities which were not evaluated during the Perry review. Correct this apparent discrepancy.

410.20 (9.1.1) In accordance with Section 9.1.1 of the SRP, identify any deviations in your new fuel storage facility design from the criteria specified in ANS 57.1, "Design Requirements for LWR Fuel Handling Systems," and ANS 57.3, "Design Requirements for New LWR Fuel Storage Facilities," as they relate to the prevention of criticality and to the aspects of radiological control.

410.21 (9.1.2) For your proposed spent fuel storage facilities, identify deviations from the acceptance criteria of Section 9.1.2 of the SRP including the appropriate portions of Standard ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

410.22 (9.1.2) Add the spent fuel pool and the pool liner to Table 3.2-1 of your FSAR. If the liner will not be designed to seismic Category I requirements, verify that a failure of the liner plate resulting from a seismic event will not result in unacceptable damage as discussed in the review procedures of Section 9.1.2 of the SRP.

410.23 (9.1.2) Your FSAR does not contain sufficient information regarding the design of your high density storage racks nor does it reference any report where the information can be found. It appears that the design of the spent fuel racks may be the same as the design which was reviewed and accepted for Hatch, Units 1 and 2. Provide either a reference to an appropriate docket or provide a report where the detailed design information may be found. Alternatively, verify that the proposed design of your high density storage racks is identical to that of the Hatch facility.

410.24 (9.1.3) Verify that the information provided in Section 9.1.3 of your FSAR is based on the new high density spent fuel pool storage capacity. Provide additional information regarding the spent fuel decay heat load for the maximum, normal and abnormal heat loads as discussed in Items 1.d and 1.h of the review procedures in Section 9.1.3 of the SRP.

410.25 (9.1.3) In Section 9.1.3.2 of your FSAR, you describe the chemistry of the water with regard to its compatibility with the aluminum storage racks. Revise this section of your FSAR to be consistent with your new high density stainless steel racks described in Section 9.1.2 of your FSAR.

410.26 (9.1.3) (RSP) In Section 9.1.3.3 of your FSAR, you state that the reactor heat removal (RHR) system will be used only to supplement the fuel pool cooling system when the reactor is shutdown. It is our position that the reactor should be in a cold shutdown condition prior to using the RHR system for supplemental fuel pool cooling.

410.27 (9.1.3) Provide the design parameters for the spent fuel pool cooling system including the cooling water temperature at which the heat exchangers are rated at 8.8×10^6 btu/hr. Verify that this heat removal rate is sufficient to maintain the pool water temperature at 125°F as stated in your FSAR for the high density storage conditions described in Section 9.1.2 of your FSAR.

410.28 (9.1.4) Verify that in the event any of the light loads (i.e., those which weigh less than a fuel assembly and its handling tool) were to be dropped over the fuel pool from their maximum normal elevation, they would cause less damage than a dropped fuel assembly. (We assume damage to be in proportion to the kinetic energy on impact.)

410.29 (9.1.4) Provide the same information for the fuel handling system as is requested in Question 410.17 for the leak detection system since your FSAR is not consistent with the Perry FSAR.

410.30 (9.1.5) With regards to the overall heavy load handling systems within the scope of your proposed nuclear island, verify that your design meets the guidelines of NUREG-0612. In your response, provide sufficient information so that we can make an independent evaluation of whether you meet the guidelines of NUREG-0612.

410.31 (9.1.5) For the fuel servicing equipment and cranes listed in Table 3.2-1 of your FSAR which are characterized as non-seismic Category I, verify that they are designed not to be a missile source as a result of a safe shutdown earthquake.

410.32 (9.1.5) For your fuel handling and heavy load handling systems, address each of the acceptance criteria identified in Sections 9.1.4 and 9.1.5 of the SRP for the equipment within the scope of your proposed nuclear island.

410.33 (9.1.5) In Section 9.1.4 of your FSAR, you state that the cask and containment polar cranes will be supplied by the applicant. However, you also list these cranes in your equipment classification in Table 3.2-1 of your FSAR. State who is responsible for these cranes. If supplied by the applicant, provide the interface requirements for these cranes with respect to any assumptions you make such as the maximum lift heights, the rail travel limitations and other interlocks. Identify the specific portions of the system within your scope of design such as the crane rails or the load blocks.

410.34 (9.2.1) In Section 9.2.1.2 of your FSAR, you state that a differential flow switch is used to detect leakage in the nonsafety-related portion of the service water system. Verify that this detection device and the associated isolation capability will be designed to safety-grade requirements.

410.35 (9.2.6) Provide the results of an analysis to show that a postulated failure of the 7000 gallon condensate supply surge tank, located in the auxiliary building, does not result in damage by flooding to any safety-related equipment. Verify that the level instrumentation on the surge volume which initiates alarms and automatic switchover of the HPCS and RCIC suction to the suppression pool, will be designed to safety-grade requirements.

410.36 (9.2.8) Verify that you have performed analyses of postulated failures of the heated water distribution system and that its failure will not damage any safety-related equipment due to the resulting environmental conditions.

410.37 (6.8) (9.3.1) In Section 9.3.1.2 of your FSAR, you state that the instrument air supplied to the main steam safety relief valves and isolation valves is filtered to remove all particles larger than 50 microns. To be consistent with Section 9.3.1 of the SRP and ANSI MC11.1-1976, this air should be filtered to 3 microns or less. Revise your design to meet this criterion. Address, as an interface requirement if necessary, the maximum total oil content of the air supply to these valves and their accumulators in accordance with Section 9.3.1 of the SRP. These same requirements should also be addressed for the pneumatic supply system.

410.38 (6.8) (9.3.1) Identify the testing requirements and frequency of tests for the safety-related accumulators and check valves provided in the compressed air system and pneumatic supply system. To assure continuous reliable functioning of the instrument air system and the pneumatic supply system, provide a procedure or an interface requirement for a procedure which requires periodic testing of the air quality for both the instrument air system and the pneumatic supply system.

410.39 (9.4.1) You indicate in Figure 9.4-1b of your FSAR that there are many single fire dampers which could fail closed resulting in a loss of direct ventilation flow to either the control room, the cable rooms, the computer room, the electrical equipment rooms or the control equipment room. Verify that adequate cooling would still exist for these various rooms following a loss of direct ventilation. Alternatively, verify that there will be adequate time and capability to manually reopen these dampers. Adequate accessibility should be assured if you take credit for manual reopening of these dampers.

410.40 (9.4.1) In addition to the scenario described in Question 410.39, consider the consequences of an actual fire closing the damper. Demonstrate that the safety-related areas downstream of the closed fire damper can receive adequate ventilation to allow safe reactor shutdown. Describe how such ventilation is accomplished. Note that to maintain adequate ventilation, it may be necessary to eliminate some fire dampers and use three-hour rated ductwork for some areas. It may also be necessary to rely on your remote shutdown capability. In this case, you must ensure credit is not taken for equipment downstream of the closed damper.

410.41 (9.5.1) Provide the details of your proposed design to demonstrate that you satisfy the criteria of Sections III.G and III.L of Appendix R to 10 CFR Part 50. In your response, provide the following information:

a. Describe the methodology used to verify that proper separation is provided for the safe shutdown capability in accordance with the requirements of Section III.G.2 of Appendix R. Provide the area arrangement drawings showing the safe shutdown system, including the cable routing.

b. Address the means you will provide for assuring the proper functioning of your safe shutdown capability, assuming fire induced failures in the associated circuits. Attachment 1 provides our concerns with associated circuits. This attachment also provides guidance for reviewing the associated circuits of concern and the additional information we need. Your response should specifically address Part II.C of this attachment.

c. Confirm that your proposed design will have the capability to achieve cold shutdown conditions within 72 hours and maintain cold shutdown thereafter, as defined in Section III.L of Appendix R to 10 CFR Part 50 and Section 5.C of Branch Technical Position CMEB 9.5-1, assuming that offsite power is not available.

d. Commit to develop and implement alternate shutdown procedures. These procedures should address the manpower requirements and the manual actions required to accomplish shutdown. Submit a summary of these procedures.

e. With respect to those repairs required to achieve safe shutdown, it is our position that systems and components used to achieve and maintain hot shutdown conditions must be free of fire damage with no credit taken for repairs. Systems and components used to achieve and maintain cold shutdown should be either free of fire damage or the fire damage should be limited so that repairs can be made and cold shutdown achieved within 72 hours. Develop repair procedures for cold shutdown systems. Material for repair should be maintained onsite. Electrical or pneumatic jumpers are not a suitable method of repair to achieve cold shutdown.

410.42 (10.4.7) Revise Section 10.4.7 of your FSAR to describe and evaluate only those portions of the main feedwater system within the scope of your design. All other information in this section of your FSAR which you consider necessary for the condensate and feedwater system design (e.g., chemistry, temperature, capacity and pressure) should be specifically identified as interface requirements.

ATTACHMENT 1

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ASSOCIATED CIRCUIT GUIDANCE

I. INTRODUCTION

The following discusses the requirements for protecting redundant and/or alternative equipment needed for safe shutdown in the event of a fire. The requirements of Appendix R address hot shutdown equipment which must be free of fire damage. The following requirements also apply to cold shutdown equipment if the applicant/licensee elects to demonstrate that the equipment is to be free of fire damage. Appendix R does allow repairable damage to cold shutdown equipment.

Using the requirements of Sections III.G and III.L of Appendix R, the capability to achieve hot shutdown must exist given a fire in any area of the plant in conjunction with a loss of offsite power for 72 hours. Section III.G of Appendix R provides four methods for ensuring that the hot shutdown capability is protected from fires. The first three options as defined in Section III.G.2 provides methods for protection from fires of equipment needed for hot shutdown:

1. Redundant systems including cables, equipment, and associated circuits may be separated by a three-hour fire rated barrier; or,

2. Redundant systems including cables, equipment and associated circuits may be separated by a horizontal distance of more than 20 feet with no intervening combustibles. In addition, fire detection and an automatic fire suppression system are required; or,

3. Redundant systems including cables, equipment and associated circuits may be enclosed by a one-hour fire rated barrier. In addition, fire detectors and an automatic fire suppression system are required.

The last option as defined by Section III.G.3 provides an alternative shutdown capability to the redundant trains damaged by a fire.

4. Alternative shutdown equipment must be independent of the cables, equipment and associated circuits of the redundant systems damaged by the fire.

II. Associated Circuits of Concern

The following discussion provides A) a definition of associated circuits for Appendix R consideration, B) the guidelines for protecting the safe shutdown capability from the fire-induced failures of associated circuits and C) the information required by the staff to review associated circuits. It is important to note that our interest is only with those circuits (cables) whose fire-induced failure could affect shutdown. Guidelines for protecting the safe shutdown capability from the fire-induced failures of associated circuits are provided. These guidelines do not limit the alternatives available to the licensee for protecting the shutdown capability. All proposed methods for protection of the shutdown capability from fire-induced failures will be evaluated by the staff for acceptability.

A. Our concern is that circuits within the fire area will receive fire damage which can affect shutdown capability and thereby prevent post-fire safe shutdown. Associated Circuits* of Concern are defined as those

*The definition for associated circuits is not exactly the same as the definition presented in IEEE-384-1977.

cables (safety related, non-safety related Class 1E, and non-Class 1E) that:

1. Have a physical separation less than that required by Section III.G.2 of Appendix R, and;

2. Have one of the following:

   a. a common power source with the shutdown equipment (redundant or alternative) and the power source is not electrically protected from the circuit of concern by coordinated breakers, fuses, or similar devices (see diagram 2a), or

   b. a connection to circuits of equipment whose spurious operation would adversely affect the shutdown capability (e.g., RHR/RCS isolation valves, ADS valves, PORVs, steam generator atmospheric dump valves, instrumentation, steam bypass, etc.) (see diagram 2b), or

   c. a common enclosure (e.g., raceway, panel, junction) with the shutdown cables (redundant and alternative) and,

      (1) are not electrically protected by circuit breakers, fuses or similar devices, or

      (2) will allow propagation of the fire into the common enclosure (see diagram 2c).

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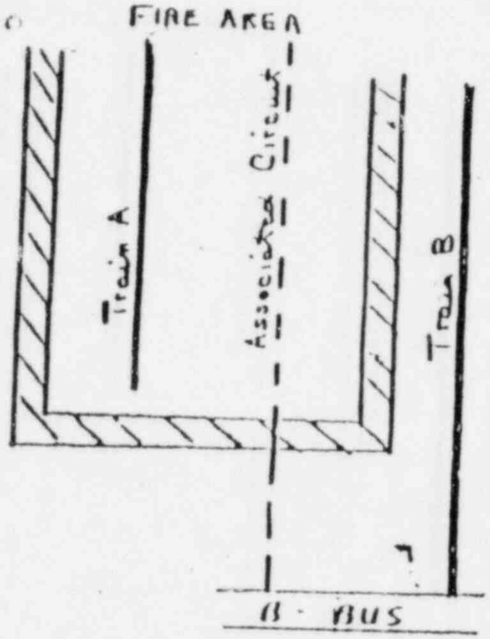
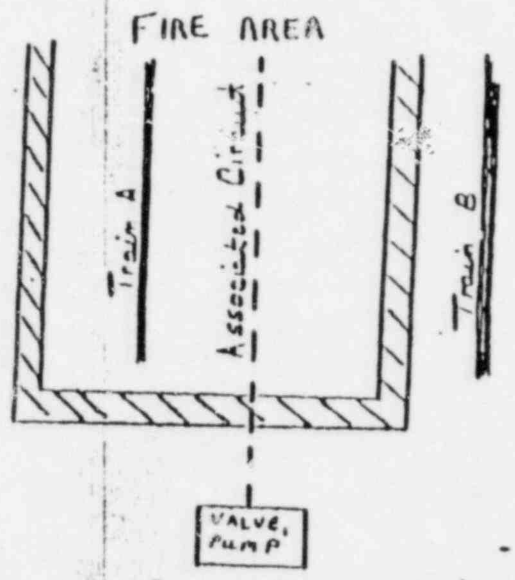
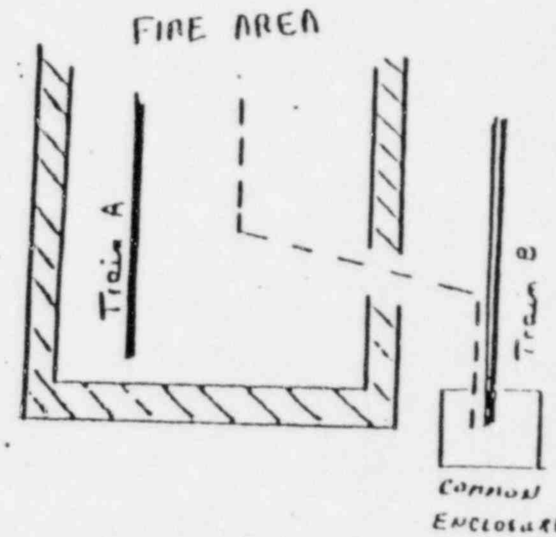
EXAMPLES OF ASSOCIATED CIRCUITS OF CONCERN





Equipment where spurious operations could affect shutdown



The area barriers shown above meet the appropriate sub-paragraphs (a-f) of section III.G-2 of Appendix R.

Diagram 2A

Diagram 2B

Diagram 2C

410-11

B. The following guidelines are for protecting the shutdown capability from fire induced failrues of circuits (cables) in the fire area. The shutdown capability may be protected from the adverse effect of damage to associated circuits of concern by the following methods:

1. Provide protection between the associated circuits of concern and the shutdown circuits as per Section III.G.2 of Appendix R, or

2. a. For a common power source case of associated circuits:
Provide load fuse/breaker (interrupting devices) to feeder with fuse/breaker coordination to prevent loss of the redundant or alternative shutdown power source. To ensure that the coordination criteria are met the following should apply:

(1) The associated circuits of concern interrupting devices (breakers of fuses) time-overcurrent trip characteristic for all circuit faults should cause the interrupting device to interrupt the fault current prior to initiation of a trip of any upstream interrupting device which will cause a loss of the common power source,

(2) The power source shall supply the necessary fault current for sufficient time to ensure the proper interruption without loss of function of the shutdown loads.

The acceptability of a particular interrupting device is considered demonstrated if the following cirteria are met:

(i) The interrupting device design shall be factory tested to verify overcurrent protection as designed in accordance with the applicable UL, ANSI, or NEMA standards.

(ii) For low and medium voltage switchgear (480 V and above) circuit breaker/protective relay periodic testing shall demonstrate that the overall coordination scheme remains within the limits specified in the design criteria. This testing may be performed as a series of overlapping tests.

(iii) Molded case circuit breakers shall periodically be manually exercised and inspected to insure ease of operation. On a rotating [illegible]ling outage basis a sample of these breakers shall be [illegible]ed to determine that breaker drift is within that allowed by the design criteria. Breakers should be tested in accordance with an accepted QC testing methodology such as MIL STD 10 5 D.

(iv) Fuses when used as interrupting devices do not require periodic testing. Administrative controls must insure that replacement fuses with ratings other than those selected for proper coordination are not accidently used.

b. For circuits of equipment and/or components whose spurious operation would affect the capability to safely shutdown:

(1) provide a means to isolate the equipment and/or components from the fire area prior to the fire (i.e., remove power cables open circuit breakers); or

(2) provide electrical isolation that prevents spurious operation. Potential isolation devices include breakers, fuses, amplifiers, control switches, current XFRS, fiber optic couplers, relays and transducers; or

(3) provide a means to detect spurious operations and then procedures to defeat the maloperation of equipment (i.e., closure of the block valve if PORV spuriously operates, opening of the berakers to stop spurious operation of safety injection);

c. For common enclosure cases of associated circuits:

(1) provide appropriate measures to prevent propagation of the fire and

(2) provide electrical protection (i.e., breakers, fuses or similar devices)

C. INFORMATION REQUIRED

. The following information is required to demonstrate that associated circuits will not prevent operation or cause maloperation of the shutdown method:

a. Describe the methodology used to assess the potential of associated circuits adversely affecting the shutdown capability. The description of the methodology should include the methods used to identify the circuits which share a common power supply or a common enclosure with the shutdown system and the circuits whose spurious operation would affect shutdown. Additionally, the description should include the methods used to identify if these circuits are associated circuits of concern due to their location in the fire area.

b. Show that fire-induced failures (hot shorts, open circuits or shorts to ground) of each of the associated circuits of concern will not prevent operation or cause maloperation of the shutdown method.

2. The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, the interface most likely consists of two redundant and independent motor operated valves. These two motor operated valves and their associated cables may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire initiated LOCA through the high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

   a. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant.

   b. For each set of redundant valves identified in a., verify the redundant cabling (power and control) have adequate physical separation as required by Section III.G.2 of Appendix R.

c. For each case where adequate separation is not provided show that fire induced failures (hot short, open circuits or short to ground) of the cables will not cause maloperation and result in a LOCA.

430.0 POWER SYSTEMS BRANCH

430.01 (8.3.1) Describe in Section 8.3.1.1.2 of your FSAR, the interlocking scheme provided on the crosstie circuit breakers between Division 1, bus F1 and Division 2, bus E1. State whether these circuit breakers are interlocked with the bus supply breakers. It is our position that bus ties compromise the independence and redundancy of the onsite electrical power supplies required by General Design Criterion 17 of Appendix A to 10 CFR Part 50. Accordingly, justify why Divisions 1 and 2 ac power supplies cannot be made completely independent by eliminating this crosstie.

430.02 (8.3.1) You state in Section 8.3.1.1.5.1, part (4) of your FSAR that Class 1E indicating light circuits do not require any special analysis or test since they do not extend past the Class 1E equipment and raceways. Explain this statement.

430.03 (8.3.1) Provide the minimum starting voltage of the Class 1E, Division 1 and 2 motors. Indicate the minimum difference between the motor torque and pump torque of the Class 1E, Division 1 and 2 motors, during acceleration. Explain the sentence in section 8.3.1.1.5.3, part (2) of your FSAR in which you state: "In some cases, motor sizing torque and load requirements are accomodated to limitations imposed by the circumstances of the system or specific functional requirements."

430.04 (8.3.1) The undervoltage relaying described in Section 8.3.1.1.7 of your FSAR, by itself, will not protect the Class 1E equipment against a degraded voltage condition. Branch Technical Position PSB-1 contained in Chapter 8 of the Standard Review Plan (SRP) requires that a second level of undervoltage protection be provided to protect Class 1E equipment against degraded voltage conditions. Describe your compliance with this position for Class 1E, Divisions 1, 2 and 3.

430.05 (8.3.1) Provide the following information regarding the load shedding and sequencing discussed in Section 8.3.1.1.7 of your FSAR:

a. Indicate what sequence of events occurs if the alternate preferred power source is lost when it is powering the Class 1E buses and the diesel is running in standby. State whether the residual bus voltage is allowed to decay to less than 30 percent as is done when transferring from the primary preferred source.

b. For the loss of preferred power during the diesel-generator parallel testing event, indicate what will automatically trip the diesel-generator circuit breaker. You state that if the alternate preferred source is used for load testing the diesel-generator and it is lost, the diesel-generator circuit breaker will be tripped and the bus will be re-energized by local manual control only. This results in a loss of the Class 1E bus. Explain why this bus is not automatically re-energized.

c. If the diesel-generator is powering the safety buses and offsite power is subsequently restored, indicate whether the safety buses automatically transfer back to the offsite source.

d. Describe the load sequencer logic, circuitry and components. Since the emergency loads are sequenced on both the offsite and onsite power sources, we require that you either provide a separate sequencer for offsite and onsite power for each electrical division. Alternatively, provide a detailed analysis to demonstrate that there are no credible sneak circuits or common failure modes in the sequencer design which could render both onsite and offsite power sources unavailable. In addition, provide additional information concerning the reliability of your sequencer and reference the design detailed drawings.

430.06 (8.3.1) In Section 8.3.1.1.8.1.1 of your FSAR, you state that separate unit station service transformers and separate reserve station service transformers are used for the normal and alternate preferred power supplies for each division. Indicate whether this arrangement is specified by the interface requirements. State whether there are other arrangements permissible under the interface specifications. Indicate why there is only one feeder from the preferred power sources provided for Division 3 while two are provided for Divisions 1 and 2.

430.07 (8.3.1) Provide the following information regarding the Divisions 1 and 2 diesel-generator qualification testing discussed in Section 8.3.1.1.8.5 of your FSAR:

a. You state in Section 8.3.1.1.8.5 that the 300 start tests have been run on similar units. If the tests were not performed on identical units, the Divisions 1 and 2 diesel-generators must be requalified in accordance with the requirements of Sections 5.4.2, 5.4.3 and 5.4.4 of IEEE Std. 387-1977.

b. The load capability test was conducted in reverse order from our position stated in Item C.14 of Regulatory Guide 1.9, Revision 2. Provide justification for this difference.

c. Provide the test results for our review.

430.08 (8.3.1) In Section 8.3.1.1.6.4 of your FSAR, you state that the diesel-generator overcurrent relay protection has a voltage restraint so that disturbances in the plant auxiliary power system which result in excessive voltage drops, will not damage the diesel-generator. Indicate how far into the plant distribution system from the diesel-generator the relays will sense a disturbance. State whether these relays are sensitive to voltage transients created by normal power system evolutions such as motor starting.

430.09 (8.3) A review of malfunction reports of diesel-generators at operating nuclear plants has disclosed that in some cases, the information available to the control room operator to indicate the operational status of the diesel-generator may be imprecise and could lead to misinterpretation. This can be caused by the sharing of a single annunciator station to: (1) alarm conditions that render a diesel-generator unable to respond to an automatic emergency start signal; and (2) alarm abnormal, but not disabling, conditions. Another cause can be the use of wording in an annunciator window which does not specifically indicate 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 this purpose.

Accordingly, review and evaluate the alarm and control circuitry for the diesel-generators in your proposed nuclear island to determine how each condition which 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 which block automatic start. Other conditions in this category are loss of control voltage, insufficient starting air pressure or low battery voltage. Your review should consider all aspects of possible diesel-generator operational conditions (e.g., test conditions and operation from a local control station). 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 resetting of the diesel-generator controls to permit subsequent automatic operation.

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

a. All conditions which render the diesel-generator incapable of responding to an automatic emergency start signal for each operating mode as discussed above.

b. The wording on the annunciator window in the control room which is alarmed for each of the conditions identified in your response to Item (a) above.

c. Any other alarm signals which are not included in Item (a) above and which also cause the same annunciator to alarm.

d. Any condition which renders the diesel-generator incapable of responding to an automatic emergency start singal and which is not alarmed in the control room.

e. Any modifications you propose following your evaluation of these matters.

430.10 (8.3.1) The prototype qualification test discussed in Amendment 3 to NEDO-10905 and referenced in Section 8.3.1.1.9.5.6 of your FSAR was conducted on a 4160 volt diesel-generator and a high pressure core spray (HPCS) pump combination. However, you indicate in Section 8.3.1.1.9.5 and Figures 8.3-1, 8.3-3a and 8.3-3b of your FSAR that you propose to use a 6900 volt diesel-generator and a HPCS pump combination. Since these are not the same units reported on in Amendment 3 to NEDO-10905, it is our position that the qualification test must be conducted on the actual diesel-generator and pump combination you propose for your nuclear island. Figure 8.3-14a of your FSAR indicates use of a 4160 volt HPCS diesel-generator and switchgear. Correct this error.

430.11 (8.3.1) Provide the following additional information regarding the loading of the HPCS diesel-generator:

a. If the HPCS is operating on the preferred power source with the diesel-generator in standby, indicate the sequence of events following a loss of the preferred power sources. State whether the residual bus voltage is allowed to decay or whether a synchronizing scheme is utilized.

b. State whether the diesel-generator will automatically separate from the test mode if an accident signal is received. Indicate the sequence of events.

c. Indicate the sequence of events if the diesel-generator is on test in parallel with the offsite source and the offsite source is lost. Indicate whether the HPCS bus will require re-energization by local manual control in a manner similar to the Divisions 1 and 2 buses.

d. If the diesel-generator is powering the HPCS bus and offsite power is subsequently restored, state whether the safety buses automatically transfer back to the offsite source.

430.12 (8.3.1) The separation you describe in Sections 8.3.1.4.2.3.1 and 8.3.1.4.2.3.2 of your FSAR for the scram solenoid circuits and the main steam line (MSL) isolation valve circuits must be justified by analysis, based on tests, to show that there is no detrimental effect on Class 1E circuits with which these circuits are run. Additionally, demonstrate that the function of the scram solenoid circuits and MSL isolation circuits will not be impaired by this arrangement. Explain how isolation is maintained between the Class 1E power supply feeding the "A" solenoids and the non-Class 1E power supply feeding the "B" solenoids since these circuits are run in a common conduit.

Explain the use of the D1 through D4 inputs shown in Figure 8.3-23 of your FSAR, coming via isolators into the load drivers of the "B" scram solenoid circuits.

430.14 State in Section 8.3.1.1 of your FSAR, whether the nuclear system protection system (NSPS) non-Class 1E power supplies which feed the "B" scram solenoids have a separate and redundant Class 1E protective package installed between the power supply and bus consisting of overvoltage, undervoltage and underfrequency protection. If not, this package should be installed to protect the solenoids against a condition which could fail them in the unsafe direction. Discuss the susceptibility of the load drivers to power supply anomalies such as over/undervoltage, over/underfrequency, voltage transients, voltage spikes, EMI and harmonics. The protective package must provide protection against any conditions which would fail the load drivers in the unsafe (i.e., shorted or closed) direction.

430.15 (8.3.1) State whether the penetrations described in part (6) of Section 8.3.1.4.2.2.3 of your FSAR, carry an electrical cable or wire. If so, explain how the pentration seal can prevent a fire being initiated in both divisions assuming a fault of the wire which induces a short circuit current to flow in the wire on both sides of the penetration.

430.16 (8.3.1) The penetration layout shown in Figure 8.3-12 of your FSAR shows that the vertical separation between some Class 1E and non-Class 1E circuits is less than four feet rather than the five feed required by IEEE Std. 384-1974. According, it is our position that an analysis, based on tests, is required to verify that the smaller separation which you propose, is acceptable.

430.17 (1.8) Provide the following additional information regarding the exceptions you take in Section 1.8 of your FSAR, to Regulatory Guide 1.75:

a. You state with respect to Position C.1 in this regulatory guide that interrupting devices actuated only by a fault current are not considered to be isolation devices unless acceptable coordination can be verified by tests. However, you should first provide justification why the non-Class 1E load must be connected to the Class 1E system and cannot be tripped on an accident signal. If suitably justified, such a design must provide two isolation devices in series, each coordinated with the upstream bus feeder circuit breaker, and periodic testing of the coordination of these devices must be performed. Provide a complete list of the non-Class 1E loads connected to Class 1E systems and identify those loads which are not tripped on a signal indicating a loss-of-coolant accident (LOCA).

b. You state with respect to Position C.4 of this regulatory guide that associated circuits will be subject to the same requirements as Class 1E circuits unless it can be demonstrated that the Class 1E curcuits are not degraded below an acceptable level by the absence of such requirements. Identify each area where this exception is taken and provide an analysis showing that the absence of Class 1E requirements will not significantly reduce the availability of the Class 1E circuits.

c. The exception you take to Position C.6 of this regulatory guide is unacceptable. Specifically, identify all areas where independence or separation is less than that required by IEEE Std. 384-1974. Provide an analysis based on tests.

d. Justify the exception you take to Position C.7 of this regulatory guide by an analysis demonstrating that Class 1E circuits are not degraded below an acceptable level. Provide this analysis.

e. Explain the exceptions taken to Positions C.8 and C.11 of this regulatory guide since they appear to be only a slightly reworded statement of the criteria in the guide.

430.18 (8.3.1) Describe in Section 8.3.1.4 of your FSAR, the cable spreading area and the separation of cables in this area with respect to the requirements contained in Section 5.1.3 of IEEE Std. 384-1974 as modified by Regulatory Guide 1.75. State whether: (1) this area contains high-energy equipment such as switchgear, transformers and rotating equipment or piping (both high and moderate-energy) which could be a potential source of missiles or pipe whip; (2) flammable materials are stored in this area; (3) power cables are routed through this area; and (4) redundant cable spreading areas are utilized. Provide the cable tray plan for this area and the electrical equipment room areas.

430.19 (8.3.1) In Section 8.3.1.3.2 of your FSAR, you state that associated cables are uniquely identified by a longitudinal stripe and/or the data on the cable. This cable should be marked, preferrably color coded at least every five feet, in accordance with our position on this matter in Regulatory Guide 1.75. We hold the same position for the cables installed in the power generation control center (PGCC) floor sections discussed in Section 8.3.1.3.2.1(6) of your FSAR.

430.20 (1.8) You have provided insufficient detail in your discussion of Regulatory Guide 1.128 in Section 1.8 of your FSAR to permit us to evaluate your compliance with this guide. Accordingly, provide a response which specifically addresses compliance with each position of this guide.

430.21 (8.3.2) State in Section 8.3.2.2 of your FSAR, whether the alternate chargers provided for the Class 1E dc systems were intended to be used to avoid a limiting condition of operation (LCO) on loss of the normal charger. Since the alternate chargers are powered from the non-Class 1E ac system, we allow no credit for their use. Accordingly, the plant will have to enter the limiting conditioning of operation status when the normal charger is lost even though the alternate charger is available.

430.22 (8.3.2.2) Both the conclusion contained in NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plant" and operating experience indicate that bus ties between redundant dc divisions are a prime contributor to dc system unreliability. As a result, we recommend in NUREG-0666 eliminating the use of a bus tie breaker between redundant buses. Based on the findings in NUREG-0666 and the fact that bus ties compromise the independence and redundancy of the onsite electric power supplies required by Criterion 17 of the GDC it is our position to prohibit the use of bus ties between redundant dc divisions in new plant designs. Accordingly, justify in Section 8.3.2.2 of your FSAR why dc Divisions 1 and 2 cannot be made completely independent by eliminating the interconnecting bus tie shown in your proposed design.

430.23 (8.3.2.1) The specific requirements for monitoring the dc power system derive from the generic requirements embodied in Section 5.3.2(4), 5.3.4(5) and 5.3.3(5) of IEEE Std. 308-1974 and the guidance we provide in Regulatory Guide 1.47. In summary, these general requirements state that the dc system composed of batteries, distribution systems and chargers shall be monitored to the extent that it can be shown to be ready to perform its intended function. Accordingly, the guidelines used in our review of the dc power system designs are that the following indications and alarms of the Class 1E dc power system should be provided in the control room:

- Battery current (ammeter-charge/discharge)
- Battery charger output current (ammeter)
- DC bus voltage (voltmeter)
- Battery charger output voltage (voltmeter)
- Battery discharge
- DC bus undervoltage and overvoltage alarm
- DC bus ground alarm (for ungrounded systems)
- Battery breaker(s) or fuse(s) open alarm
- Battery charger output breaker(s) or fuse(s) open alarm
- Battery charger trouble alarm (one alarm for a number of abnormal conditions which are usually indicated locally)

We conclude that the monitoring cited above, augmented by the periodic test and surveillance requirements included in the Technical Specifications, provide reasonable assurance that the Class 1E dc power system is ready to perform its intended safety function. Indicate your compliance with these provisions for monitoring the Class 1E prower system. Alternatively, justify any deviation.

430.24 (8.3.2) Explain the statement in Section 8.3.2.1.3.1 of your FSAR that: "The normal dc supply is from the battery two nondivisional buses."

430.25 (8.3.2) Verify that the periodic testing of the ac and dc electrical distribution system will be in accordance with the Standard Technical Specifications applicable to your proposed design.

430.26 (8.3.2) Since the feeder from the Class 1E dc systems to the balance of plant test equipment could compromise the independence of the Class 1E dc systems, provide a feeder circuit breaker which is locked open during plant operation and annunciates in the control room when the circuit breaker is closed. Revise section 8.3.2.2 of your FSAR accordingly.

430.27 (8.3.2) Provide the specified operating voltage range of the Class 1E dc loads. Provide the maximum equalizing charge voltages for the Class 1E batteries and the dc system minimum discharge voltage at the end of the two hour design discharge. Provide the rating of the Division 3 battery charger and indicate the number of cells in each Class 1E battery. State whether the Division 3 battery charger will be affected by the voltage sag which occurs when the HPCS pump is started on the diesel-generator.

430.28 (8.3) Provide the one-line diagrams for the motor control centers and buses fed from the 480 volt load centers and the 125V dc distribution panels.

430.29 (8.3.1) Provide the following additional information regarding diesel-generator load sequencing:

a. The method for determining the loading of motor-operated valves in Tables 8.3-1 and 8.3-2 of your FSAR is not consistent between Divisions 1 and 2. Indicate the total loading in these tables and in Table 8.3-3. Revise these three tables.

b. The actual load sequencing times should be given in Table 8.3-4 of your FSAR rather than the maximum allowable time. Indicate the totals and subtotals for each load sequencing step. Provide a revised Table 8.3-4 incorporating the above comments.

c. Table 8.3-5 of your FSAR seems to imply that all the safety loads except RHR pumps A and B and one ESW pump are block loaded on the diesel-generators at time zero. Explain this matter in the text of your FSAR.

430.30 (8.3.1) Explain note 8 of Figure 8.3-2 in your FSAR, particularly the phrase "for BUS E Normal Feeder Backfeed."

430.31 (1.8) Provide the following additional information regarding the protection regarding the protection of containment electrical penetrations:

a. You indicate in Part I.2.13 of Section 1.8 of your FSAR that an analysis is required for circuits normally protected by small fuses or breakers such as control circuits, alarms and solenoids. Provide this analysis.

b. In this same portion of the FSAR, you also indicate that where very low currents are involved such as in instrumentation circuits, thermocouples and annunciators, no action is required and that conformance with the provisions of Regulatory Guide 1.63 is accomplished by inspection. Explain what is meant by the phrases "no action required" and "conformance by inspection." It is our position that if the fault current available from these circuits is greater than the continuous current rating of the penetrators, the penetrations must be protected by at least two fault current interrupting devices.

c. Provide the fault current clearing-time curves of the primary and secondary current interrupting devices for the penetrations plotted against the thermal capability (I^2t) curve of the penetration. Our concern in this matter is the maintenance of mechanical integrity. Provide a simplified one-line diagram showing the location of the protective devices in the penetration circuit and indicate the maximum available fault current of the circuit. If the overcurrent protection is not fault current actuated, identify the power source to the trip circuits. It is our position that the power source for the primary protection device should be from a division different from that supplying the secondary protection device.

430.32 (1.8) In Part I.2.27 of Section 1.8 of your FSAR, you state that your design thermal overload devices are active only when the equipment is in the test mode and are bypassed when the equipment is in the normal mode. Provide details of the means used to bypass the overloads. State whether indication is provided in the control room that the bypass is removed. Provide a schematic of the bypassing and indication scheme.

430.33 (8.3.3) In Section 8.3.3.2 of your FSAR, you state that cable tunnels in the control building are divisionalized. Describe how they are "divisionalized" and explain how this complies with Position C.8 of Regulatory Guide 1.75.

430.34 (8.3.1) Recent experience with protective relays for Class 1E electrical system equipment in nuclear power plants has established that the relay trip setpoint of conventional relays drifts from its initial setting. This in turn, has resulted in premature trips of redundant safety-related system pump motors when the safety system was required to be operative. While the basic need for proper protection for feeders/equipment against permanent faults is recognized, it is our position that total non-availability of redundant safety systems due to spurious trips in protective relays, is not acceptable. Accordingly, provide a description of your circuit protection criteria for safety systems/equipment to avoid: (1) an incorrect selection of the initial setpoint; and (2) the drifting of the trip setpoint of protective relays

430.35 (8.3) We have noted during our reviews of other applications that pressure switches or other devices were incorporated into the final actuation control circuitry for large horsepower safety-related motors used to drive pumps. These switches or devices preclude automatic (i.e., upon receipt of a safety signal) and manual operation of the affected motor/pump combination unless permissive conditions such as lube oil pressure are satisfied. Accordingly, identify all safety-related motor/pump combinations which you propose to incorporate in your nuclear island and which operate as noted above. Describe the redundancy and diversity which is provided for the pressure switches or permissible devices used in this manner.

430.36 (6.3) (8.3) Identify all electrical equipment, both safety and non-safety, that may become submerged as a result of a LOCA. For all such equipment that is not qualified for service in this environment, provide an analysis to determine the following:

a. The safety significance of the failure of this electrical equipment (e.g., spurious actuation or loss of actuation function) as a result of flooding.

b. The effects on Class 1E electrical power sources serving this equipment as a result of such submergence.

c. Any proposed design changes resulting from this analysis.

430.37 (8.3) Provide the results of a review of your operating, maintenance, and testing procedures to determine the extent of usage of jumpers or other temporary means of bypassing functions for operating, testing, or maintenance of safety-related systems. Identify and justify any cases where the use of temporary bypasses cannot be avoided. Provide criteria for any use of jumpers when testing.

430.38 (8.1.) Incidents have occurred at operating nuclear power plants which indicate a deficiency in the design of the electrical control circuitry. These incidents include the inadvertent disabling of a component by racking out the circuit breakers for a different component. Accordingly, review the electrical control circuits of all safety-related equipment in your proposed nuclear island to assure that disabling of one component does not, through incorporation in other inter-locking or sequencing controls, render other components inoperable. All modes of test. operation and failure should be considered. Provide the results of your review.

430.39 (8.3) Provide a listing of all motor-operated valves in your proposed nuclear island which require power lock-out in order to satisfy our single failure criterion. Indicate the features of your design which permit you to satisfy this requirement.

430.40 (8.3) Certain nuclear power plants which have two-cycle turbocharged diesel engines manufactured by the Electromotive Division (EMD) of General Motors driving emergency generators, have experienced a significant number of turbocharger mechanical gear drive failures. These failures have occurred as a result of running the emergency diesel-generators at no-load or light-load operation for extended periods. This type of operation could occur during periodic equipment testing or during accident conditions when offsite power is available. When this equipment is operated under no-load conditions, the volume of exhaust gas is insufficient to operate the turbocharger. As a result, the turbocharger is driven mechanically from a gear drive in order to supply enough combustion air to the engine to maintain its rated speed. However, the turbocharger and mechanical drive gear normally supplied with these engines are not designed for the standby service encountered in nuclear power plants where the equipment may be called upon to operate at no-load or light-load conditions at full-rated speed for a prolonged period. (The EMD equipment was originally designed for locomotive service where no-load speeds for the engine and generator are much lower than full-load speeds. The locomotive turbocharged diesel hardly even runs at full speed except at full load.) Accordingly, the EMD has strongly recommended that this particular diesel engine not be operated at no-load or light-load conditions at full-rated speed for extended periods due to the short life expectancy of the turbocharger mechanical gear drive unit normally furnished. No-load or light-load operation also causes a general deterioration in any diesel engine. To cope with the severe service to which the equipment is normally subjected when installed in nuclear power plants and in the interest of reducing failures and increasing the availability of its equipment, EMD has developed a heavy-duty turbocharger drive gear unit which can replace existing equipment. This is available as a replacement kit; engines can also be ordered with the heavy-duty turbocharger drive gear assembly.

To assure optimum availability of the emergency diesel-generators on demand, it is our position that you should only supply the heavy duty turbocharger mechanical drive gear assembly if you intend to order emergency generators driven by two-cycle diesel engines manufactured by EMD. This position is consistent with the recommendation by EMD for the class of service encountered in nuclear power plants. Confirm your compliance with this requirement.

430.41 (8.3) Diesel-generators with a high degree of reliability are an essential part of the safety systems for nuclear power plants. Accordingly, provide a discussion of the level of training which will be required for the applicant's personnel to ensure that diesel-generator reliability levels inherent in your nuclear island will be maintained. As applicable, state your recommendations for the types of personnel to be trained; i.e., operators, maintenance crew, quality assurance personnel and supervisors. In your discussion, identify the amount and kind of training you recommend for each of the above categories and the type of ongoing training program you recommend to assure optimum availability of the diesel-generators. Discuss the level of education and minimum experience requirements you recommend be met for the various categories of operations and maintenance personnel associated with the emergency diesel-generators.

430.42 (8.3) The availability on demand of an emergency diesel-generator is dependent upon, among other things, the proper functioning of its controls and monitoring instrumentation. This equipment is generally mounted on panels and in some instances, the panels are mounted directly on the diesel-generator skid. Major diesel engine damage has occurred at some operating plants from vibration induced wear on skid-mounted control and monitoring instrumentation. This sensitive instrumentation is not made to withstand and function accurately for prolonged periods under the continuous vibrational stresses normally encountered with internal combustion engines. Operation of sensitive instrumentation under this environment rapidly deteriorates the calibration, the accuracy and the control signal output.

Accordingly, except for sensors and other equipment which must be directly mounted on the engine or associated piping, it is our position that the controls and monitoring instrumentation should be installed on a free-standing floor-mounted panel separate from the engine skids and located on a floor area free from vibration.

If the floor is not free of vibration, the panel shall be equipped with vibration mounts. Confirm your compliance with this requirement. Alternatively, provide justification for noncompliance.

430.43 (9.5.2) Identify all working stations in your proposed nuclear island where it may be necessary for plant personnel to communicate with the control room or the emergency shutdown panel during and/or following transients or accidents in order to mitigate the consequences of the event or attain safe plant shutdown. Provide a tabulation of these working stations.

430.44 (9.5.2) In Section 9.5.2.2.2.3 of your FSAR, you state that sound-powered phones are used for intraplant fixed-type emergency communications. The arrangement for the sound-powered phones is presented in Figures 9.5.4 through 9.5.9 of your FSAR. Based on our review of these drawings, we conclude that there is no master station in the control building nor are any of the numerous jack stations equipped with ringing devices. Considering these two facts, explain how communications are established between the control room and any specific jack station serving a working station identified in your response to Question 430.43 during and/or following transients or accidents.

430.45 (9.5.2) Provide a diagram showing the locations of the loud speakers associated with the coded-call automatic paging (CAP) system. Identify the source of power for the CAP system. State what, if any, function the system serves in establishing intraplant communications during and/or following transients or accidents. (Intraplant communications beyond the nuclear island, interplant communications, and plant to offsite communications will be evaluated as plant specific items.)

430.46 (9.5.2) Provide a diagram showing the location of the private automatic exchange (PAX) system phones and phone jacks. State what, if any, function the PAX system serves in establishing intraplant communications during and/or following transients or accidents. State whether the PAX system is designed to seismic Category I requirements. Alternatively, describe the device(s) which will isolate the PAX system from its Class 1E power source following a design basis seismic event.

430.47 (9.5.2) Provide a discussion of the communications between the emergency or remote shutdown panel and the remainder of the plant. Show how communications between this area and working stations throughout the plant will be established during and/or following transients or accidents.

430.48 (9.5.2) Provide a tabulation of the communication system(s) extensions to the balance of plant which will be required in order to provide adequate communications under all operating conditions, including transients and accidents. Identify the nuclear island/balance of plant interfaces of these communication system(s) extensions.

430.49 (9.5.3) Provide in Section 9.5.3.1.2(1) of your FSAR, a numerical value for the term "approximating" as used in connection with IES recommended illumination levels. Provide justification for not conforming with IES recommedations.

430.50 (9.5.3) Provide a tabulation of the vital areas where emergency lighting is needed for: (1) safe shutdown of the reactor; (2) to maintain it in a safe shutdown condition; and (3) for evacuation of personnel in the event of an accident. In this tabulation, indicate the access routes to and from safety-related areas.

430.51 (9.5.3) Provide the following information regarding the standby lighting system:

a. State whether all transformers, panels, and cable trays associated with the system are designed to seismic Category I requirements.

b. State whether all standby lighting system light fixtures are seismically supported.

c. If the standby lighting system components are not seismically qualified, provide a discussion of the isolation devices which will be used to disconnect the standby lighting system from its Class 1E power source following a design basis seismic event.

430.52 (9.5.3) Provide a following information regarding the emergency lighting system:

a. The seismic qualification of the self-contained emergency lighting sets.

b. The seismic qualification of the panels, cable trays, breakers, and other components of the emergency lighting system(s) connected to the Division E and F, Class 1E, 125 V dc station battery.

c. State whether the emergency lighting system light fixtures are seismically supported.

d. If the emergency lighting system components are not seismically qualified, provide a discussion of the isolation devices which will be used to disconnect the emergency lighting system from its Class 1E power source following a design basis seismic event.

430.53 (9.5.3) If the standby and emergency lighting systems are not seismically qualified, provide a discussion of how adequate lighting will be provided for safe plant shutdown after an elapsed time of 8 hours following a design basis seismic event.

430.54 (9.5.3) Demonstrate that the control room and the remote shutdown panel illumination levels under emergency conditions are in conformance with the applicable sections of NUREG-0700.

430.55 (9.5.3) In order that we may understand Table 9.5-1 of your FSAR, provide the following additional information:

a. Indicate the percentage of plant lighting which is connected to the normal ac lighting system, and the percentage which is connected to the standby ac lighting system.

b. Indicate how many main circuits for normal lighting are included in your plant design and their source of power.

c. Indicate how many main circuits for standby lighting are included in your plant design and their source of power.

d. Indicate the minimum number of different normal and standby lighting circuits that will be utilized in providing lighting for any safety-related area.

e. Indicate the source of "auxiliary" power for normal lighting in the event of loss of standby lighting power.

f. Indicate the electrical separation criteria which has been used in the design of the normal, standby and emergency plant lighting system. State whether the safety-related lighting systems are treated the same way as plant Class 1E circuits. Indicate in which trays the safety-related and nonsafety-related systems are installed.

430.56 (9.5.3) (Lighting systems for the balance of plant beyond the nuclear island will be reviewed as plant specific items.) Provide the interface data for continuation of normal, standby and/or emergency lighting to the balance of plant.

430.57 (9.5.4) In Section 9.5.4.2 of your FSAR, you state that the diesel-generator fuel oil booster pumps operate with a flooded suction and that the fuel oil day tanks have a minimum capacity sufficient for two hours of diesel-generator operation at full load. However, you show in Figures 1.2-21 and 1.2-22 of your FSAR that the bottom of the Divisions 1 and 2 fuel oil day tanks are below the diesel engine base. Accordingly, provide the following information for the Divisions 1 and 2 diesel-generator fuel oil system:

a. The overall capacity of the day tanks.

b. The capacity of the day tanks at the level at which the diesel engine fuel oil booster pump would no longer be flooded.

c. The positive suction head requirements for the diesel engine fuel oil booster pump.

d. The diesel engine fuel oil consumption rate at maximum load.

e. The day tank capacity at the low-level alarm point.

Provide the day tank capacity, the diesel engine consumption rate at maximum load and the day tank capacity at the low-level alarm point for Division 3.

430.58 (9.5.4) Provide the quality group classification for the diesel fuel oil day tanks.

430.59 Provide the following additional information:

a. Revise Figure 9.5-10 of your FSAR to show the interface between the fuel oil system piping and the diesel engine mounted piping/components. Provide quality group classifications for all system piping and components and, if applicable, identify all changes in piping/component quality group classifications at the interface.

b. Explain the purpose of the duplex strainer, the blind flanges, the relief valve and the instrumentation in the line parallel to the engine driven fuel oil booster pump.

c. The duplex strainer in the two inch diesel fuel oil supply line from the balance of plant is monitored with a switch indicating pressure differential. Indicate where the differential pressure indication appears and where the associated high differential pressure alarm annunciates. If this alarm does not annunciate in the control room, provide the rationale for your proposed design. (This paragraph is applicable to the Division 1, 2 and 3 diesel-generator fuel systems as shown on Figures 9.5-10 and 9.5-11 of your FSAR.)

d. The duplex strainer in the fuel oil booster pump suction line is monitored with a differential pressure switch. Indicate whether this switch activates an alarm and, if so, where the alarm annunciates. If the alarm does not annunciate in the control room, or no alarm is provided, provide justification for your proposed design.

e. The duplex filter on the fuel oil booster pump discharge is monitored with a differential pressure indicator. State where the high differential pressure indication appears. Provide your rationale for not using audible alarms as part of the filter differential pressure monitoring.

430.60 Provide the following additional information:

a. Revise Figure 9.5-11 of your FSAR, to show the interface between the fuel oil system piping and the diesel engine mounted piping/components. Provide the quality group classifications for all system piping and components and, if applicable, identify all changes in piping/component quality group classifications at the interface.

b. We note that there are significant differences between the Divisions 1 and 2 diesel fuel oil system instrumentation and controls and that of the Division 3 diesel fuel oil system as shown in Figures 9.5-10 and 9.5-11 of your FSAR. These differences are in the areas of the day tank high and low level switches, the day tank level indicators/transmitters, the booster pump suction strainer differential pressure monitoring and the fuel filter differential pressure monitoring. Moreover, the Division 3 diesel-generator is equipped with an electric fuel oil-booster pump in addition to the engine-driven booster pump and both of these pumps are fitted with simplex suction strainers. Conversely, the Divisions 1 and 2 diesel-generators have only the engine-driven fuel oil booster pump but are fitted with duplex suction strainers. Provide your rationale behind this design approach, with particular attention as to why monitoring and alarms are not required on the Division 3 diesel-generator fuel oil booster pump suction strainers and duplex fuel oil filters. State why the instrumentation, controls, and components cannot be identical for all 3 divisions. (Refer to Question 430.110.)

430.61
(9.5.4) You show on Figures 9.5-10 and 9.5-11 of your FSAR, the day tank vents terminating somewhat outside the diesel-generator room. However, it is not clear from Figures 9.5-10 and 9.5-11 nor from Figures 1.2-18 through 1.2-22 of your FSAR, exactly where the Divisions 1, 2, and 3 day tank vents terminate. Accordingly, provide additional information on these vents. Show vent the terminations on appropriate views in Figures 1.2-18 through 1.22 of your FSAR and provide details of the terminations which show that they are protected from tornados, floods and the effects of severe weather conditions.

430.62 Identify all high and moderate-energy lines and systems which will
(9.5.4) be installed in the diesel-generator room. Discuss the measures which
(9.5.5) will be taken in the design of the diesel-generators to protect
(9.5.6) the safety-related systems, piping and components from a postulated
(9.5.7 failure of either a high or moderate-energy line. Our concern is
(9.5.8) the availability of the diesel-generators when needed.

430.63 Discuss what precautions have been taken in the design of the fuel
(9.5.4) oil system when selecting the location of the fuel oil day tank and the connecting fuel oil piping in the diesel-generator room. Our concern is the possible exposure of these components to ignition sources such as open flames and hot surfaces.

430.64 You state in the text and in Table 3.2-1 of your FSAR that the
(9.5.4) components and piping systems for the diesel-generator auxiliaries
(9.5.5) (e.g., the fuel oil cooling water, lubrication , air starting,
(9.5.6) and intake and combustion systems) are mounted on auxiliary skids which
(9.5.7) are designed to seismic Category I requirements and are built to ASME
(9.5.8) Section III, Class 3 quality standards. You also state that engine-mounted components and piping are designed and manufactured to DEMA standards and are designed to seismic Category I requirements. However, this is not in accordance with our position in Regulatory Guide 1.26 in which we state that all the diesel-generator auxiliary systems should be designed to ASME Section III, Class 3 or Quality Group D standards. Provide the industry standards which you will use in the design, manufacture, and inspection of the engine mounted piping and components. Show on the appropriate P&I diagrams where the Quality Group Classification changes from Quality Group C.

430.65 In your description of the emergency diesel engine fuel oil storage
(9.5.4) and transfer system (EDEFSS) in Section 9.5.4.1 of the FSAR, you do not specifically reference ANSI Standard N195, "Fuel Oil Systems for Standby Diesel Generators." Indicate if you intend to comply with this standard in your design of the EDEFSS. Alternatively, provide justification for noncompliance.

430.66 The Division 3 diesel-generator fuel system includes an electrically
(9.5.4) driven, backup booster pump. Discuss the purpose and operation of this pump. State why an electrically driven backup booster pump is provided for the Divisions 1 and 2 diesel-generators. Indicate the source of power for the Division 3 backup pump.

430.67 Add a section to your FSAR which describes the instruments, controls,
(9.5.4) sensors, and alarms provided for monitoring the diesel engine storage and transfer system and discuss their function. Discuss the testing necessary to maintain and assure highly reliable instruments, controls, sensors and alarms. Indicate where the alarms are annunicated. Identify the temperature, pressure, and level sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer. Describe what operator actions are required during alarm conditions to prevent harmful effects to the diesel engine. Discuss the system interlocks provided in your proposed design.

430.68 (9.5.4) Provide the following balance of plant (BOP) interface data:

a. The piping requirements for the BOP section of the fuel oil storage and transfer system, including pipe sizes, materials, quality group classifications and the location of the interface.

b. The source of power for the BOP fuel oil transfer pumps including the bus, voltage, number of phases and MCC location.

c. The BOP fuel oil transfer pump minimum capacity in gallons per minute (gpm) and the discharge head requirements for those portions of the system associated with the nuclear island.

d. The minimum quantity of fuel to be stored for each diesel-generator and your basis for calculating the minimum quantity.

e. The diesel fuel oil quality standards which must be met in accordance with the standards of the diesel engine manufacturer and to comply with Item C.2 of Regulatory Guide 1.137.

430.69 (9.5.5) In Section 9.5.5 of your FSAR, you indicate that the function of the diesel-generator cooling water system is to dissipate the heat transferred through: (1) the engine water jacket; (2) the lube oil cooler; (3) the engine air water coolers; and (4) the governor lube oil cooler. Provide information on the individual component heat removal rates (btu/hr), flow (lbs/hr) and temperature differential (°F) and the total heat removal rate required. Provide the design margin (i.e., the excess heat removal capacity) provided in the design of major components and subsystems. The design margin should be stated either as a percentage or as btu per hour.

430.70 (9.5.5) Indicate the measures you have taken to preclude long-term corrosion and organic fouling in the diesel engine cooling water system since these would degrade the system cooling performance and affect the compatability of the system. State whether the water chemistry is in conformance with the engine manufacturer's recommendations.

430.71 (9.5.5) Recent licensee event reports (LER's) have shown that tube leaks are occurring in the heat exchangers of diesel engine jacket cooling water systems resulting in failures of the engines to start on demand. Provide a discussion of the measures you propose to detect tube leakage and the corrective measures that will be taken. Include a consideration of jacket water leakage into the lube oil system (standby mode), lube oil leakage into the jacket water (operating mode) and jacket water leakage into the engine air intake and governor systems (operating or standby mode). Provide the permissable inleakage or outleakage in each of the above conditions which can be tolerated without degrading engine performance or causing engine jacket water/service water systems leakage.

430.72 (9.5.5) Describe the provisions you have made in the design of the diesel engine cooling water system to assure that all components and piping are filled with water.

430.73 (9.5.5) For the Division 1 and 2 diesel-generators, you show an atmospheric vent at the top of the standpipe in Figures 9.5-12 and 9.5-13 of your FSAR. This indicates that the top of the standpipe is the highest point in the diesel engine cooling water system. For the Division 3 diesel-generator, however, no atmospheric vent is shown in any part of the system. This indicates that the jacket water expansion tank is not the high point in the cooling water system as shown on Figure 9.5-13. Clarify this matter. If the expansion tank is not the highest point in the system, then: (1) revise Figure 9.5-13 to show the proper elevation of the tank relative to other piping and components in the cooling water system; and (2) refer to Question 430.72 and show how air is vented from the system. Demonstrate that air in the piping at the system high point will not be forced to another part of the system such as the jacket water cooler where it could cause a partial or total blockage. Describe how air is purged from the system piping once the diesel engine is running. Indicate the time required to accomplish this purging following startup.

430.74 (9.5.5) If the Division 3 diesel generator expansion tank is not at the cooling water system high point, then provide a discussion of how you will prevent corrosion in the piping which is exposed to air when the engine is not operating (standby) and in the remainder of the system due to entrapped air in the system cooling water.

430.75 (9.5.5) The diesel-generators are required to start automatically on loss of all offsite power and in the event of a LOCA. The diesel-generator sets should be capable of operation at less than full load for extended periods without degradation of performance or reliability. Should a LOCA occur and offsite power is available, discuss the design provisions and other parameters which you have considered in the selection of the diesel-generators to enable them to run unloaded (on standby) for extended periods without degradation of engine performance or reliability. Explicitly define the capability of your design with regard to this required characteristic.

Describe the make and type of engine and the design features which enables the engine to operate at no load and full speed for seven days without degradation of performance and reliability. Provide the manufacturer's test results which verify the above cited capability.

430.76 (9.5.5) The Divisions 1 and 2 and the Division 3 diesel-generator cooling water system standpipes and expansion tank, respectively, provide for expansion of the cooling systems inventory when the diesel-generators are operating. In addition, the standpipes and the expansion tank provide makeup to the systems inventory to compensate for minor leaks at pump shaft seals, valve stems, and other components. Provide the size (i.e., the capacity) of the standpipes and the expansion tank for the Divisions 1 and 2 and the Division 3 diesel-generators, respectively. Demonstrate by analysis that the standpipe and expansion tank sizes will be adequate to provide makeup water for seven days of continuous diesel-generator operation at full rated load without requiring any makeup water supply to the standpipes and to the expansion tank. (Refer to Item (a) of Question 430.110.)

The Divisions 1 and 2 diesel-generator standpipes are mounted vertically on the floors of their rooms. When determining the adequacy of the standpipe inventory with respect to the required seven days of makeup, you should consider only that volume of coolant which can be lost from the standpipe and yet still maintain a net positive suction head (NPSH) to both the engine-driven and motor-driven cooling system circulating pumps.

430.77 (9.5.5) For the Division 3 diesel-generator, demonstrate that the expansion tank does, in fact, provide a NPSH for the jacket water pumps at both the normal and the lowest permissible operating water level in the expansion tank.

430.78 (9.5.5) Provide a detailed discussion of how the diesel-generator cooling water systems function in the standby mode to maintain jacket water temperatures above ambient temperatures to enhance the diesel engine start capability. Your discussion should address how the jacket water is heated, how heated water is circulated through the diesel engines and the design jacket water temperature at the anticipated ambient temperatures of the diesel-generator rooms. Identify any excess capacity in the jacket water heating system.

The operation of the Division 3 diesel-generator cooling water system during standby requires additional discussion since there is an apparent lack of heated jacket water under forced circulation in this mode.

430.79 (9.5.5) Describe the instrumentation, controls, sensors and alarms provided for monitoring the diesel engine cooling water system and describe their functions. Discuss the testing necessary to maintain and assure highly reliable instruments, controls, sensors and alarms. Indicate where the alarms are annunciated. Identify the temperature, pressure, level and flow sensors, where applicable, which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer. Describe what operator actions are required during alarm conditions to prevent harmful effects to the diesel engine. Discuss the systems interlocks you will provide.

430.80 (9.5.6) Describe the instrumentation, controls, sensors and alarms provided for monitoring the diesel engine air starting system. Describe their function. Describe the testing necessary to maintain highly reliable instruments, controls, sensors and alarms. Indicate where the alarms are annunciated. Identify the temperature, pressure and level sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer. Describe any operator actions required during alarm conditions to preclude degradation of diesel-generator starting capability. Provide the setpoints at which these alarms function. Discuss system interlocks you will provide.

430.81 (9.5.6) Provide a detailed description of the diesel engine starting system which is shown on Figures 9.5-14 and 9.5-15 of your FSAR. Additionally, describe: (1) the components and their function; (2) the instrumentation, controls, sensors and alarms; and (3) a diesel engine starting sequence. In describing the diesel engine starting sequence, include the number of air start valves used and whether one or both air start systems are used.

430.82 (9.5.6) For the Divisions 1 and 2 diesel-generators, provide a discussion of the air starting system downstream of the left and right bank air distributors. Revise Figure 9.5-14 of your FSAR to show the additional system components.

430.83 (9.5.6) Expand your discussions of the air starting systems for the Divisions 1 and 2, and the Division 3 diesel-generators. Identify the differences between the two types of systems. Your description of these differences should cover both the systems components and the instrumentation and controls. (Refer to Item (b) of Question 430.110.)

430.84 (9.5.6) In Section 9.5.6.1 of your FSAR, you state that the storage tanks, valves, and piping up to the air start motors are designed to seismic Category I requirements and ASME Section III, Class 3 standards. Review your design and indicate if there are any non-ASME items or sections in the system. If so, identify these and indicate their locations on Figures 9.5-14 and 9.5-15 of your FSAR. In any case, revise Figures 9.5-14 and 9.5-15 to reflect their seismic and quality group classifications of system piping and components. Indicate where changes in classification occur.

430.85 (9.5.6) In Section 9.5.6.3 of your FSAR, you briefly discuss the air dryers in the Divisions 1 and 2 diesel-generators air start system. However, there is no mention of an air dryer for the Division 3 diesel-generator nor is one shown on Figure 9.5-15 of your FSAR. Provide a discussion of why air dryers are used with the Divisions 1 and 2 diesel-generator air start system but not with the Division 3 diesel-generator air start system.

430.86 (9.5.6) In Section 9.5.6.2 of your FSAR, you describe the compressed air and air start systems. However, this description appears to cover only the Divisions 1 and 2 diesel-generators. Revise this section to include a detailed description of the Division 3 diesel-generator compressed air and air start systems. State whether all four air start motors are used in every engine start. For the diesel engine driven compressor, describe how this unit cycles on and off, what inputs are used to stop and start the engine and/or compressor, whether the diesel engine operates continuously and any other pertinent information. Show how the Division 3 diesel-generator air start system is, operationally, completely redundant. (Refer to Item (b) of Question 430.110.)

430.87 (9.5.6) In Section 9.5.6.2 of the FSAR, you state that each redundant air start system has sufficient capacity for five automatic or manual starts without recharging the air receivers. There are two different types of systems for the Divisions 1 and 2, and Division 3 diesel generators, respectively. For both types of systems, provide the following information:

a. Describe what constitutes a completed "start cycle."

b. Indicate the design working pressure for the air start motors for Division 3 and the direct cylinder injection for Divisions 1 and 2.

c. Indicate how much air, measured as either a pressure drop or standard cubic feet per minute (SCFM), is consumed for each starting cycle. Indicate the resulting air receiver pressures; i.e., at the beginning of the start cycle and on its completion for each of the other five starts. Provide the time required for the diesel-generator to reach full speed, voltage, and frequency and be ready to accept load for each of the five starts.

d. State the pressure at which the five start capacity is determined; i.e., compressor cut-in, compressor cut-out or mid-point.

e. Indicate the capacity of the air receivers.

430.88 (9.5.6) Indicate the source of power to the solenoid valves in the diesel-generators air start systems.

430.89 (9.5.6) You incorporate in Figures 9.5-14 and 9.5-15 of your FSAR, symbols and abbreviations for which no explanation is included on Figure 1.7-4 or any other drawing showing symbols or legends. Accordingly, revise these drawings, as required, to ensure there is an explanation for all symbols and abbreviations. Explain the purpose of the heavy black arrows shown at various locations on Figures 9.5-14 and 9.5-15.

430.90 (9.5.6) In NUREG/CR-0660, air dryers in diesel generator air start systems are described as being safety significant. In Section 9.5.6.2 of your FSAR, you briefly discuss air dryers in the Division 1 and 2 diesel-generator air start systems. Provide details of these air dryers, including the type (desiccant or refrigerant), manufacturer and model number, capacity, special features, principal of operation and other pertinent details. Show that the dew point in the air system will be maintained below the recommended minimum value in accordance with our position on this matter in Section 9.5.6 of the SRP. Since the air dryers are safety significant, provide details of the system operation and/or system maintenance procedures which, when implemented, will ensure proper functioning of the air dryers at all times.

Provide a comparable discussion for the air dryer to be installed in the Division 3 system, if you do not provide justification for the lack of an air dryer.

430.91 (9.5.6) In Figure 9.5-14 of your FSAR, you show the air dryers for the Divisions 1 and 2 diesel-generator starting air system mounted on the air receivers. Since the air receivers are safety-related, provide the seismic qualification for the air dryers. Alternatively, show that failure of the air dryers as a consequence of a design basis event will not impair operation of the diesel-generator air start systems.

430.92 (9.5.6) Provide the pertinent characteristics of the air compressors for the diesel-generator air start systems; i.e., the rated air flow in cfm at design pressure, rated duty, motor HP and duty, motor voltage and number of operating phases and the source of power to the motor-driven compressor.

430.93 (9.5.6) Provide enlarged and more detailed plan and elevation views of the Division 3 diesel-generator air start system air compressors. Show the intake, the exhaust, the cooling system and the fuel supply for the diesel engine-driven compressor. Incorporate these enlarged views into the appropriate drawings in Section 1.2 of your FSAR.

430.94 (9.5.7) The seismic and quality group classification of the diesel-generator's lubrication system piping and components are not clearly identified in Section 9.5.7, in Table 3.2.1 or Figures 9.5-16 and 9.5-17 of your FSAR. This is not acceptable. The lubrication system should conform to the positions we present in Regulatory Guide 1.26; i.e., all the diesel-generator auxiliary systems should be designed to ASME Section III, Class 3 or Quality Group C standards. Provide the industry standards you will follow for the design, manufacture, and inspection of the lubrication system piping and components, including engine-mounted piping and components. Show this information on Figures 9.5-16 and 9.5-17. Indicate where the Quality Group Classification changes from Quality Group C, as applicable. (Refer to Section 9.5.4 of your FSAR.)

430.95 (9.5.7) For the diesel engine lubrication systems described in Section 9.5.7 of your FSAR, provide the following information: (1) define the temperature differentials, flow rate, and heat removal rate of the interface cooling system external to the engine and verify that these are in accordance with the recommendations of the engine manufacturer; (2) discuss the measures that will be taken to maintain the required quality of the oil, including its inspection and replacement when oil quality is degraded; (3) describe the protective features such as blowout panels provided to prevent an unacceptable crankcase explosion and to mitigate the consequences of such an event; and (4) describe the capability to detect and control system leakage. In your response, consider the different types of diesel engines in the design of your nuclear island and any special requirements for lube oil and lube oil analysis which may exist.

430.96 (9.5.7) Indicate what measures you have 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.

430.97 (9.5.7) Under certain emergency conditions, the diesel-generators may be required to operate continuously for an extended period (i.e., 7 days or more). During this time, the diesel engines will consume lube oil. In your FSAR, you do not discuss: (1) provisions for checking or monitoring the lube oil level during engine operation; or (2) the capability to add lube oil to the sump during engine operation. Provide a discussion of these items. If extra lube oil is stored in the diesel-generator buildings, describe the oil storage containers and the area in which they are stored. Show the storage locations on appropriate plan and elevation views in Chapter 1 of your FSAR and show any piping on Figures 9.5-16 and 9.5-17. Provide seismic and quality group classifications. Alternatively, show that there is sufficient inventory in the diesel engine sumps at all times to allow for oil consumption during seven days of continuous engine operation at full load while still maintaining enough lube oil for lubrication, cooling, and adequate suction head to the lube oil pressure pump(s).

430.98 (9.5.7) Describe the instrumentation, controls, sensors and alarms provided for monitoring the diesel engine lubrication oil systems and their function. Indicate where the alarms are annunciated. Identify the temperature, pressure and level sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer. Describe any operator action required during alarm conditions to prevent harmful effects to the diesel engine. If any of the systems, controls and/or alarms are associated with an automatic engine shutdown, discuss the interlocks provided for bypassing the shutdown function under emergency conditions.

430.99 (9.5.7) Describe your program for periodic testing and calibration of sensors, controls, and instrumentation which will be implemented to ensure a highly reliable lubrication system.

430.100 (9.5.7) Expand your description of the diesel engine lube oil system to include a detailed system description of what is shown on Figures 9.5-16 and 9.5-17 of your FSAR. In your response, describe: (1) the components and their function; and (2) a diesel-generator starting sequence for a normal start and an emergency start.

430.101 (9.5.7) In Section 9.5.7.4 of your FSAR, you refer to alarms for low oil pressure, high oil temperature and low oil level. However, none of these alarms are shown on Figure 9.5-16. Further, you show these alarms on Figure 9.5-17 in addition to a low oil temperature alarm, a lube oil high temperature and a high pressure alarm associated with a relief valve and an extra lube oil low pressure alarm. None of these alarms are described in the text of your FSAR. Revise Figures 9.5-16 and 9.5-17 to agree with the text and/or revise the text to agree with Figures 9.5-16 and 9.5-17.

430.102 (9.5.7) On Figure 9.5-16, you show a 12 inch "engine L.O. drain," and a 2 inch "drain." Explain the function of each of these drains.

430.103 (9.5.7) Expand your description of the lube oil keepwarm circuit for the Divisions 1 and 2 diesel-generators to include such specific items as the keepwarm pump capacity, L.O. heater capacity, design L.O. temperature during standby operation, minimum design ambient temperature in the diesel-generator room, and instrumentation and controls for the keepwarm system.

430.104 (9.5.7) Provide the seismic and quality group classifications for the keepwarm pump, heater, and associated piping and components, and for the L.O. sump vent.

430.105 (9.5.7) One of the recommendations in NUREG/CR-0660 is for prelubrication of the diesel engines prior to starting, thereby minimizing wear due to a lack of adequate lubrication at the time of starting. The keepwarm curcuit shown on Figure 9.5-16 provides continuous prelubrication to the Divisions 1 and 2 diesel engines, except for the turbochargers and the upper part of the diesel engine. Show that this lack of prelubrication does not impair diesel engine operation or reliability.

If the Divisions 1 and 2 diesel engines will be manufactured by DeLaval, revise your lubrication system P&I diagrams to show vendor modifications to provide drip lubrication to the turbocharger thrust bearings. State whether vendor modifications to the governor lube oil circuits have been, or will be, incorporated. If the Division 3 diesel-generator is manufactured by EMD, show that the recommendations of MI-9644 have been incorporated. (Refer to Item (c) of Question 430.110.)

430.106 (9.5.7) Describe the function of the pressure pump, piston cooling pump, scavenging pump, and soak back pump for the Division 3 diesel-generator. (Refer to Figure 9.5-17 of your FSAR.) Describe how these pumps are driven; i.e., common shaft or separate shafts.

430.107 (9.5.7) The lube oil filter shown on Figure 9.5-17 of your FSAR has a single inlet line from the scavenging pump discharge and two outlet lines, both of which terminate at the lube oil strainer. Describe the operation of the lube oil filter and the function of each of the outlet lines. Describe the operation of the lube oil filter internal relief valve. Indicate how this relief valve interfaces with the system temperature and pressure alarms.

430.108 (9.5.7) You show on Figure 9.5-17 of your FSAR, a line between the soak back pump discharge and the turbocharger lube oil filter outlet. State the purpose of this line. If the soak back pump operates continuously during standby, describe how a buildup of lubricating oil in the diesel engine exhaust system is prevented. (NUREG/CR-0660 indicates that excess oil in the exhaust system could be a fire hazard.) Describe the function and operation of the spring check valve and the connecting line between the soak back pump discharge and the lube oil filter inlet shown on Figure 9.5-17.

430.109 (9.5.7) Using Figure 9.5-17 of your FSAR as an aid, describe how diesel engine prelubrication is accomplished. State whether the prelube system operates continuously during periods of diesel-generator standby. Describe how the lube oil temperature is maintained during standby. If any parts of the diesel engine do not receive prelubrication, identify the affected parts and explain how engine reliability is not degraded as a consequence. Revise Figure 9.5-17 as required.

430.110 (9.5.5) (9.5.6) (9.5.7) In Chapter 5 of NUREG/CR-0660, personnel training is listed under the category of "Most Significant Corrective Action." This is based on data which show that lack of knowledge of diesel-generators and systems has contributed significantly to diesel-generator failures and an overall lack of reliability. In response to these data, we now review personnel training and training programs as an integral part of our licensing procedure.

Considering the significance of personnel training, provide justification for proposing:

a. Cooling water systems for the diesel engines of Divisions 1 and 2 which have significant design differences from that of Division 3. (Refer to Question 430.76)

b. Compressed air starting systems for the Divisions 1 and 2 diesel-generators which have significant design differences from that of Division 3. (Refer to Question 430.86.)

c. Diesel-generator lubrication systems for Divisions 1 and 2 which have significant design differences from that of Division 3. (Refer to Question 430.105.)

430.111 (9.5.7) Revise Figure 9.5-10 of your FSAR, to show the complete combustion air intake and exhaust systems. Alternatively, provide a new P&I diagram showing these systems, including all three divisions. Show all instrumentation and controls associated with the systems.

430.112 (9.5.8) Describe the instrumentation, controls, sensors and alarms provided in the design of the diesel engine combustion air intake and exhaust system which alert the operator when parameters exceed ranges recommended by the engine manufacturer and describe any operator action required during alarm conditions to prevent harmful effects to the diesel engine. Discuss systems interlocks provided.

430.113 (9.5.8) In Section 9.5.8.3 of your FSAR, you state that all intake and exhaust ducting will be seismic Category 1 and conform to ANSI B31.1 piping code requirements. This is not acceptable. We require the air intake and exhaust system, up to the diesel engine interface, be designed to seismic Category 1 requirements and be built to ASME Section III, Class 3 or Quality Group C standards. Revise your design accordingly. Identify the engine interface for both intake and exhaust systems.

430.114 (9.5.8) In Section 9.5.8.3 of your FSAR, you state that the air intakes for the Divisions 1 and 2 diesel-generators are located 7 feet, 9 inches above grade. This is not acceptable. In NUREG/CR-0660, it is recommended that air intakes be located a minimum of 20 feet above ground to minimize ingestion of dust and debris stirred up at grade level or by the velocity of the air entering the intakes. Revise your design accordingly.

430.115 (9.5.8) In Section 9.5.8.3 of your FSAR, you briefly discuss the effects of decreases in barometric pressure on diesel engine performance. Expand this discussion to be more specific as to the effect of decreasing barometric pressure. State the maximum tornado-induced pressure change, in units of psi per second, the diesel engines can withstand without significantly affecting performance. State the minimum barometric pressures (in. of Hg regulating from a hurricane) at which the diesel engines can operate for: (1) up to one hour; and (2) for extended periods without degrading output or causing engine problems. In your response, discuss the three diesel-generators.

430.116 (9.5.8) Experience at some operating plants has shown that diesel engines have failed to start due to an accumulation of dust and other deliterious material on electrical equipment associated with starting of the diesel-generators (e.g., auxiliary relay contacts and control switches).

Describe the provisions you have made in your diesel-generator building design, electrical starting system, and ventilation air intake design(s) to preclude this condition, thereby assuring the availability of the diesel-generator on demand.

Describe what procedures will be used during normal plant operation to minimize accumulation of dust in the diesel-generator room. Specifically address the control of concrete dust. In your response, consider the condition of one unit in operation with one or more additional units under construction at the same site.

430.117 (9.5.8) Show by analysis that a potential fire in the Division 2 and Division 3 diesel-generator building occurring with a coincident single failure of the fire protection system, will not degrade the quality of the diesel combustion air, thereby permitting the remaining diesel-generator to provide its full rated power.

460.0 EFFLUENT TREATMENT SYSTEMS BRANCH

460.09 (1.8) (11.2) (11.3) (11.4) Provide a table in Section 1.8 of your FSAR comparing the design features of the liquid, gaseous and solid radwaste systems with each position of Regulatory Guide 1.143, Revision 1 (October 1979). Justify each position for which an exception is taken. If information is provided in other sections of the FSAR for the individual items, cross-references to these sections is acceptable. We consider compliance with Section C.5 of Regulatory Guide 1.143 to be essential. Verify whether you satisfy our acceptance criteria for concentrations of radioactive constituents in accordance with Item II of section 15.7.3 of the Standard Review Plan (SRP). Our position is that limiting doses to 0.5 rems, as stated in Section 11.3.2.20 of your FSAR, is not an acceptable alternative.

460.10 (3.2) Add sections for effluent radiation monitors and engineered safety feature (ESF) filters in Table 3.2-1 of your FSAR. Also add to this table, under appropriate sections, the recombiners in the off-gas system and the process radiation monitors themselves.

460.11 (6.5.1) Provide additional information on the following items for the ESF filters of the standby gas treatment system (SGTS) and the control building:

a. State whether instrumentation for measuring flow rates through the ESF filter systems will be provided in accordance with Regulatory Guide 1.52, Revision 2 (March 1978).

b. Indicate the type of recording device which will be provided for recording pertinent pressure drops and flow rates in the control rooms.

c. Since the explanations given in Table 6.5-1 of your FSAR indicating how you satisfy positions C.2.j and C.4.b of the regulatory guide cited in Item (a) above are unclear, explain how replacements of either all or part of the filter train will be accomplished when this is required. Also explain how the filter train components will be maintained by service personnel located outside the housing. Indicate whether the ESF atmosphere cleanup system will be totally enclosed.

d. State whether duct and housing leak tests will be performed in accordance with the provisions of Section 6 of ANSI N 510-1975 and in accordance with position C.2.1 of the regulatory guide cited in Item (a) above.

e. With regard to the position C.3.b of this regulatory guide, state whether the manual overtemperature cutoff switches for the air heaters will be accessible following a postulated loss-of-coolant accident (LOCA). Note that the temperature set point should not exceed 225 F per ANSI N 510-1975.

460.12 (11.1) Provide information on source terms for the following items:

a. Provide the appropriate data for the items listed in Chapter 4 of NUREG-0016, Revision 1 (January 1979). For those items for which information has already been provided elsewhere, cross-references to the applicable sections are acceptable.

b. Release data for tritium from operating BWR's does not support your conclusions regarding release via: (1) the gaseous pathway as compared to the liquid pathway; or (2) the total release. In fact, for a number of operating BWR's, tritium releases are significantly higher than your estimate. Accordingly, verify your estimates for tritum release via the gaseous and liquid pathways using actual release data.

c. Verify and correct the N-16 concentration given in Table 11.1-4 of your FSAR. Additionally, verify and correct, as appropriate, the reactor water concentrations for Na-24, P-32, Cr-51, Mn-54 and Zn-65 since these are significantly lower than the corresponding concentrations given in NUREG-0016, Revision 1.

d. Add Fe-55 to Table 11.1-5 of your FSAR.

460.13 (11.2) Provide additional information on the following items applicable to the liquid waste management system:

a. Provide the liquid waste inputs in gallons per day (GPD), averaged on a yearly basis, of waste generation for low conductivity and high conductivity wastes to be used for evaluating liquid effluent releases and related off-site doses. In addition to the waste streams you have identified as design basis inputs in Table 11.2-4, you should also include the resin rinse and cleanup phase separator decant inputs. State the primary coolant activity fractions for each of the individual streams for these two waste subsystems.

b. Your inputs for chemical laboratory waste, laboratory wash water and laundry drains are low in comparison with the corresponding values given in NUREG-0016, Revision 1, on a per reactor basis. Verify and correct, as appropriate, these inputs.

c. Since you have considered only the deep bed regenerant system for condensate cleanup and you have also stated that the condensate cleanup system is within the applicant's scope, indicate whether usage of the deep bed regenerant system for condensate cleanup is an interface requirement. Additionally, indicate whether ultrasonic resin cleaning is also an interface requirement.

d. Since the filtered detergent wastes may be directly discharged into the circulating water discharge canal, state the fraction of detergent wastes that you expect to be discharged in a year to the circulating water discharge canal.

e. Indicate what you mean by a "waste collector subsystem" to which you refer in Section 11.2.2.2 of your FSAR; we do not find it discussed anywhere.

f. Since the excess water tank collects excess water from both the low and high conductivity subsystems, explain how you can selectively prevent discharge of excess water from the low conductivity subsystem during the time when excess water from the high conductivity subsystem is discharged to the environment. If you cannot prevent discharge of low conductivity wastes to the environment at all times, then include the appropriate fraction of waste discharge from this subsystem to the environment.

g. Since your P&I diagrams for the waste subsystems are for a dual unit radwaste system, indicate whether the equipment that you have listed on page 11.2-30 of your FSAR is for both units or whether it is on a per unit basis.

h. Describe the provisions for preventing uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor tanks if the latter is within your scope. If these provisions will be described in your response to Question 460.09, a cross-reference to the relevant portion of Section 11.2 is acceptable.

i. Provide the concentrations of radionuclides in the excess water storage tank. Verify and correct, as appropriate, the amount of radioactivity, in curies, for I-131 and the total curies in the concentrated waste tank given in Table 12.2-13 of your FSAR.

j. Indicate whether your estimated releases and corresponding doses due to liquid effluents are based on design basis reactor coolant source terms provided in Tables 11.1-2 and 11.1-3 of your FSAR. If not, use reactor coolant source terms consistent with the bases in NUREG-0016.

In responding to the ten items above, revise the appropriate tables throughout your FSAR in a consistent manner and so indicate in your response.

460.14 (11.3) Provide additional information on the following items applicable to the gaseous waste management systems:

a. Since your system description, tables and figures in Chapter 9 of your FSAR do not clearly indicate whether there are provisions for both HEPA and charcoal adsorbers for the reactor building pressure control mode and purge exhaust, provide the appropriate information relating to filter units for the reactor building.

b. Total airborne effluent releases of noble gases, including Ar-41, tritium and C-14 and some of the particulates given in Table 11.3-8 of your FSAR, are not consistent with NUREG-0016, Revision 1, and are lower than corresponding releases for radionuclides cited in this document. We assume that you have not taken any credit for particulate removal by HEPA filters in the building exhaust systems since you state in Section 1.8 of your FSAR that the need for HEPA's and charcoal absorbers will have to be decided on a site specific basis. Accordingly, verify that your estimated releases are conservative. You should note that using an off-gas release rate of 25,000 Ci/sec for noble gases after a 30 minute delay is not consistent with the basis provided in NUREG-0016, Revision 1. A release rate of about 53,000 Ci/sec is appropriate according to this document. You should also note that the caption for Table 12.2-22 is misleading since the annual airborne releases from the various sources for evaluating the environmental impact should be used for total plant release and corresponding off-site gaseous effluent doses. Either correct the caption for Table 12.2-22 or revise the contents of the table so as to reflect expected releases rather than design basis releases. Revisions to Table 11.3-8 should be coordinated with corresponding revisions to gaseous effluent dose estimates given on page 11.3-25.

c. Add flow rate measuring devices for the monitors and samplers for all the airborne effluent release pathways.

d. Since the off-gas system is located in the turbine building which is not within the scope of your design, state whether the design of the off-gas system lies within your scope. If not, state whether the off-gas system you have described is an interface requirement for the balance of plant.

e. State whether the source terms you have used to evaluate off-site doses due to a postulated failure of the off-gas system are consistent with Branch Technical Position ETSB 11-5 (July 1981).

f. State whether the seismic criteria for the proposed off-gas system will conform to Section C.5 of Regulatory Guide 1.143. In responding to this question, a cross-reference to another section of your FSAR is acceptable.

460.15 (11.4) Provide additional information on the following items applicable to the solid radwaste system:

a. Provide the isotopic breakdown of the total curie content of "wet" solid wastes that are expected to be shipped annually to a licensed burial site, accounting for the minimum decay available during storage prior to shipment. The total should include contributions from: (1) evaporator bottoms associated with high conductivity and detergent wastes; (2) spent resins associated with reactor water cleanup, radwaste, regenerant condensate deep bed, fuel pool and suppression pool cleanup demineralizers; and (3) filter sludges. Provide an estimate of the number of containers which will be shipped annually.

b. Experience with operating BWR's indicates that a deep bed condensate polishing system can generate a significantly higher volume of solidified "wet" solid wastes (i.e. about 41,000 cubic feet for a 3400 Mwt plant) than that presented in Table 11.4-2 of your FSAR. Accordingly, verify that your inputs to Table 11.4-2 of your FSAR are correct.

c. Add the suppression pool cleanup wastes in Section 11.4.1 of your FSAR

d. Describe your provisions for complying with Branch Technical Position ETSB 11-3, Revision 2 (July 1981). Your description should include: (1) the curbs and drainage provisions for containing radioactive spills; (2) a reference to the process control program as an interface requirement; (3) heat tracing for evaporator concentrate piping and tanks that are likely to solidify at ambient temperatures; (4) flushing connections, wherever appropriate; (5) the direct venting of equipment which uses compressed gases for the transport of resins or filters sludges; (6) the appropriate waste storage capacities for tanks accumulating spent resins from the reactor water cleanup system and other sources and filters sludges in accordance with our position in the branch technical position cited above; and (7) the volume of the available waste storage area for both the high and low-level wastes.

e. Add an interface requirement to control the release of airborne dusts generated during the compaction process for "dry" solid wastes.

469.16 (11.5) Provide additional information on the following items applicable to the process and effluent and radiological monitoring and sampling systems:

a. Provide in tabular columns, the sampling frequency, the minimum analysis frequency and the sensitivity in Ci/cc for the following airborne effluents and process streams:

1. Grab sampling for the principal gamma emitters and tritium for the plant vent, turbine building vent and radwaste building ventilation system effluents.

2. Grab sampling for the principal noble gas gamma emitters for the off-gas system, the drywell purge system and the fuel building ventilation system effluents.

3. Grab sampling for iodine in process streams for the off-gas treatment system; the drywell purge system; the auxiliary, fuel, radwaste and turbine buildings vent systems; the evaporator vent systems; and the pre-treatment liquid radwaste tank vent gas systems.

4. Continuous sampling of the effluents for iodines, particulates and gross alpha emitters for the plant vent, turbine building vent and radwaste building vents.

Your sampling and analysis frequencies and sensitivities for Items (1) through (4) above should be consistent with the appropriate frequencies and sensitivities in NUREG-0473, Revision 2 (February 1980). State whether the turbine building monitoring and sampling provisions are within the applicant's scope.

b. For liquid effluents and process streams:

1. Add your proposed grab sampling provisions for the service water and the detergent drain tank effluents to Table 11.5-6 of your FSAR.

2. Add your grab sampling provisions in the process liquid streams for the component cooling water system and the laboratory and sample system waste systems in Table 11.5-4 of your FSAR. Clearly indicate whether the fuel pool filter-demineralizer includes both spent fuel and refueling pools.

3. It is our position that your grab sampling and the associated analysis should identify the isotopic composition and determine the concentrations of the principal radionuclides and determine the concentration of the alpha emitters in addition to determining the gross radioactivity for all liquid effluents and process streams.

4. Explain what you mean by the waste sample tanks and the floor drain sample tank to which you refer in Table 11.5-6 of your FSAR. We find these references to be unclear since the discharge to the environment from the liquid radwaste system can only be from either the excess water tank or the detergent drain tank according to your system description.

5. Add the radionuclide Fe-55 to the isotopic analyses of effluent and process streams.

c. State whether the design criteria for the radiological effluent monitors will conform with the manufacturer's standard per ANSI N13.10 (1974) and the staff's position on quality assurance in Sections C.4 and C.6 of Regulatory Guide 1.143, Revision 1. If not, provide justification for any deviations.

460.17 Since the radiological consequences resulting from the release of contaminated liquid to the environs due to a postulated failure of the liquid tank are dependent upon site specific geological and hydrological parameters, provide justification for not leaving the evaluation of the off-site radiological consequences within the applicant's scope. Our understanding of your proposed nuclear island is that your scope of work should be only to supply the source terms. In this regard, your assumption that iodine is the critical isotope which will determine whether radionuclide concentrations at the nearest surface water supply in an unrestricted area will be within the limits of 10 CFR Part 20, is not valid. (In general, the long-lived isotope Cs-137 is the critical isotope.)

460.18 Provide additional information on the following items applicable to Item III.D.1.1 of NUREG-0737:

a. Add the containment and primary coolant sampling and containment spray recirculation systems to those systems requiring periodic leak tests.

b. State whether high pressure injection recirculation is part of the leak test programs.

c. Describe the leak reduction measures which will be incorporated into your design.

471.0 RADIOLOGICAL ASSESSMENT BRANCH

471.04 (12.1.1) Revise Section 12.1.1.3.1 of your FSAR to show compliance with Regulatory Guide 8.8, Revision 3, as you state in Section 1.8.

471.05 (12.1.2) Our position in Section C.1.d(2) of Regulatory Guide 8.8 states that licensees should propose designs which incorporate features to maintain occupational doses to "as low as reasonably achievable" (ALARA) during decommissioning. We state in Section 12.1.2 of the Standard Review Review Plan (SRP) that our determination of the acceptability of the proposed design will be based on our evaluation of your proposed measures for assuring that occupational doses during decommissioning will be ALARA. Accordingly, describe in Section 12.1.2.1 of your FSAR, your proposed design considerations for minimizing radiation doses during decommissioning including, for example, a description of your proposed provisions for major equipment removal from the drywell, process equipment removal through hatches or removable sections of shield walls and knock-out walls.

471.06 (12.2.2) Provide an estimate of the airborne sources of radioactivity in the reactor containment during normal plant operation, including the assumptions you use. Describe in Section 12.2.2.2 of your FSAR, the maximum expected airborne sources in accessible areas of the reactor containment following relief valve venting. Estimate the dose to personnel at the travelling in-core probe (TIP) drives while the operating personnel are leaving the containment following relief valve venting, including the assumptions you use.

471.07 (12.2.2) In Section 12.2.2.3 of the FSAR, you state that other potential airborne radioactivity could occur during vessel head venting and fuel movement. Explain why the entrapped radioactive gases, collected under the vessel head, could not be vented or exhausted via the gas treatment system prior to vessel head removal.

471.08 (12.3.1) In Section C.1.e of Regulatory Guide 8.8, we recommend the use of low cobalt and low nickel bearing materials for primary coolant piping, tubing, vessel internal surfaces and other components in contact with the primary coolant. Indicate in Section 12.3.1 of your FSAR, the cobalt and nickel content of such materials. Describe in this section, the steps you have taken to eliminate cobalt and nickel from such surfaces. State whether the following design features were considered: (1) selection of alternative materials, other than Stellite, for hard facings of wear materials; (2) limiting the cobalt content in stainless steel to a specified maximum such as 0.05 percent for reactor internals; and (3) limiting the cobalt content in stainless steel in contact with the primary coolant to a maximum cobalt content of 0.2 percent for uses other than reactor internals. If these measures were considered, indicate what actions you took in this regard.

471.09 Provide a table of primary system components (e.g., the reactor pressure vessel internals, clad, fuel, the recirculation loop piping and the feedwater piping downstream of the CCS) which are in contact with the reactor coolant showing the corrosion producing areas in units of square feet, a description of the material (e.g., stainless steel, zirconium, Stellite, Inconel or carbon steel) the proposed cobalt content limits expressed as a weight percent of cobalt and the corrosion rate (in mg/dm^2-mo) for each material. Additionally, provide a table of the various materials used in the primary system and indicate their contribution to the cobalt in the primary system, expressed as a percentage; the total contributions should equal 100 percent.

471.10 Provide the results of cost/benefit analyses evaluating the effects of reducing the cobalt content of cobalt contributing materials and components (e.g., the reactor vessel internals at core vicinity, the reactor pressure vessel cladding, the primary recirculation loop and the feedwater piping). This cost/ benefit evaluation should be done for the cobalt content reduced to 0.25, 0.10 and 0.05 weight percent. In addition, correct the radiation survey data for the cobalt housing in Table 12.2-19 of your FSAR, which indicates 3000 mr/hr before cleaning and 4000 mr/hr after cleaning.

471.11 (12.3.2) In Section 12.3.2.3 of your FSAR, you state that the SPCU circulation pumps are located in an open corridor at the minus 32 foot elevation and that during operation, dose rates in the pump area are less than 1 mr/hr. However, you further state that during an isolation transient, dose rates in this area temporarily increase to 700 mr/hr and that due to the nature of the event, egress from the area can be accomplished well before dose rates reach this level. Explain how an individual in this area will know that the dose rate is increasing so that egress can be accomplished in sufficient time.

471.12 (12.3.2) In Section 12.3.2.3 of your FSAR, you state that the dose rate in the control room is much less than 1 mr/hr during normal reactor operating conditions. However, you show radiation levels in the control room and in the control building to be 1 to 5 mr/hr in the control building radiation zone map drawings (Figures 12.3-16, 12.3-17, 12-3-18 and 12.3-19). Correct this discrepancy and revise the zone map drawings as required.

471.13 (12.2.2) In Section 12.2.2.1 of your FSAR, you state that your basis for release is, among others, 24 drywells purges per year, 365 hours between each purge. Explain why this basis for estimating the average I-131 release was chosen recognizing that you state in Section 9.4.5.2.2 of your FSAR that the drywell purge system functions only during plant shutdown.

471.14 (12.3.2) In Section 12.3.2.3 of your FSAR, you state that access to the fuel transfer tube is through a hatch shielded by a stepped composite concrete and lead shield plug. It is our position that all accessible portions of the plant near the spent fuel transfer tube and/or canal must be shielded during fuel transfer. Refer to our position in Section C.2.a of Regulatory Guide 8.8 which states that extraordinary design features are warranted for very high radiation areas. Using removable shielding for this purpose is acceptable. In this regard, the removable shielding shall be such that the resultant contact radiation levels shall be no greater than 100 rads per hour. All accessible portions of the spent fuel transfer tube shall be clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as described above. It is our position that if permanent shielding is not used, local radiation monitors capable of providing audible and visible alarms must be installed to alert personnel when the temporary fuel transfer tube shielding is removed during fuel transfer operations. Accordingly, provide the following additional information:

a. State whether an interlock is provided to prevent spent fuel passage when the shield plugs at the 11 foot and 26 foot elevations are open.

b. State whether unique caution signs (i.e., (1) high radiation area; and (2) potentially lethal radiation fields are possible during fuel transfer) will be provided.

c. Indicate the thickness of the spent fuel transfer tube shielding on Figure 19.3.12.3-6 of your FSAR at the 26.5 foot elevation.

d. Provide a description of your proposed shielding and access controls for access to the fuel transfer tube valve room in the annulus area. (annulus access at elevation 11'-0", Figure 19.3.12.3-6)

471.15 Describe the shielding for protection of personnel on the platform at elevation 47'-2" in the upper drywell area from radiation exposure which could occur during passage of the spent fuel over the reactor vessel flange to the fuel pool gate.

471.16 In Table 1AA-2 of your FSAR, you indicate a source term of zero percent noble gases, 50 percent halogens, and 1 percent all remaining. This mix corresponds to a source representative of depressurized reactor water. State whether a pressurized water source was used for the shielding design of the post-accident sampling station and for estimating personnel exposures for this activity. In this regard, we state in NUREG-0737 that a source mix representative of pressurized water is 100 percent noble gases, 50 percent halogens and 1 percent all remaining. It is our position that this pressurized water source should be used as the basis for establishing the shielding design of the post-accident sampling station and for estimating personnel exposures during the taking, transporting and analyzing of reactor water samples.

471.17 In paragraph (4) of Item II.B.2 of NUREG-0737, we state that you should submit post-accident dose rate maps for potentially occupied areas and indicate the projected doses to individuals who must be in vital areas for certain necessary occupancy times. Accordingly, provide post-accident radiation zone maps and the estimated doses received by individuals assigned to perform the following functions:

a. Operate three manual valves in the auxiliary and fuel building (1AA.2.C).

b. Obtain reactor coolant and containment gas samples in less than 1 hour.

c. Perform radiochemical/chemical analyses of samples in less than 2 hours.

In addition, specify the location of the post-accident sampling and sample analysis areas.

471.18 Provide your response to Item II.F.1.3 of NUREG-0737. (In containment high range radiation monitors). (GE will provide in September 1982)

471.19 In Section 1AA.2 of your FSAR, you state that it is not necessary for operating personnel to have acess to any place other than the control room and three manual valves in the auxiliary and fuel buildings to operate the equipment of interest during the 100 day period. You also state in Section 1AA.3.3 of your FSAR that necessary shutdown and post-accident operations are performed from the control room, expect for the several manual valves cited above. Revise this section of your FSAR to indicate the required personnel access to the post-accident sampling station and the sample analysis area, as stated in NUREG-0737.

471.20 In Section C of Regulatory Guide 8.19, we state that you should provide assessments of the annual occupational doses in man-rems, principally during the design stage. We further state that as a result of the dose assessment process, we expect that various design changes and innovations to reduce radiation doses will be incorporated in your design. We designate certain design features in Section C.2.e of Regulatory Guide 8.8 which should be considered in the crud control effort. Accordingly, state whether the following design features were considered in your proposed design and indicate what actions you took:

a. High temperature filters (i.e., magnetic filters) for crud removal from the primary coolant during reactor operation.

b. Stainless steel piping and heat exchanger tubing downstream of the condensate cleanup system.

c. Reduction of corrosion by minimizing the internal surfaces of the primary system.

d. Reduction of personnel exposure during in-service inspection by reducing the amount of weld footage; e.g., using forged sections as opposed to forged-welded plant sections of pressure system components.

e. Reduction of the iron and cobalt content in the reactor coolant water by increasing the efficiency of the reactor water purification systems and by increasing the cleanup flow rate.

f. Provisions for injecting oxygen into the feedwater line.

640.0 PROCEDURES AND TEST REVIEW BRANCH

640.01 (14.1) Modify Table 14.1-3 and Figure 14.1-1 of your FSAR to either delete the reference to Test Condition 7 or to state why it has been included since no tests are indicated as being conducted at these conditions. Additionally, operation in excess of your rated thermal power is not permitted.

640.02 (14.1) Modify Figure 14.1-1 of your FSAR to show the location of A through F and Test Condition 6 on this figure. In addition, provide a description for those lines and cross-hatched areas which are not described. Alternatively, remove these lines and cross-hatched areas.

640.03 (14.2.7) Most of the exceptions to Regulatory Guide 1.68 listed in Section 14.2.7.2 of your FSAR were presented to us in your letters dated March 18, 1974, and December 17, 1974, as comments to a proposed Revision 1 to this guide. Many of these comments were incorporated into Revision 2 of Regulatory Guide 1.68 and are no longer applicable. Accordingly, modify Section 14.2.7.2 to address those exceptions still applicable to Revision 2 of this regulatory guide.

640.04 (14.2.7) Modify Section 14.2.7.3 of your FSAR to indicate the level of conformance of your intitial test program with the following regulatory guides: (1) Regulatory Guide 1.68.1; (2) Regulatory Guide 1.68.2; (3) Regulatory Guide 1.95, Position C.5; (4) Regulatory Guide 1.108, Position C.2.a; (5) Regulatory Guide 1.128, Position C.4; (6) Regulatory Guide 1.140, Position C.5.

640.05 (14.1.3) State in Section 14.1.3.3 of your FSAR whether the completion of the preoperational testing which is required prior to fuel loading includes the review and approval of the test results. If portions of any preoperational tests are intended to be conducted, or their results approved, after fuel loading, provide the following information: (1) list each test; (2) state which portions of each test will be delayed until after fuel loading; (3) provide technical justification for delaying these portions; and (4) state when each test will be completed (key to test conditions defined in Chapter 14). Adding this type of information into your FSAR will permit facilities built per the GESSAR II FDA to conduct a "phased initial test program" similar to that approved for Unit 1 of the Grand Gulf facility.

640.06 (14.2.12) Describe how acceptance criteria for your proposed tests will be developed. We are concerned about a number of instances in which tests failed to meet established acceptance criteria but upon further review of the test results by the applicant or licensee, the acceptance criteria were changed and the test results then accepted. Identify in the appropriate sections of Chapter 14, the bases for the acceptance criteria for all tests. Examples of such "bases" might include: (1) regulatory guides; (2) Technical Specifications; (3) assumptions used in Chapter 15 analyses; (4) topical reports; (5) references to other GESSAR sections; and (6) codes and standards.

640.07 (14.2.12) You list in Section 14.2.12.1 of your FSAR, 15 preoperational test descriptions which the applicant will supply. However, there are a number of additional tests specified in Regulatory Guide 1.68 which you do not list. State whether the applicant's FSAR will describe the tests listed below or provide descriptions of these tests in the appropriate sections of your FSAR. If complete test descriptions are provided elsewhere in your FSAR, insert a cross-reference in Section 14.2. The additional tests to be added, if necessary, are:

a. Closed cooling water (CCW) system tests. (Refer to Section 9.2.2 of your FSAR.)

b. Combustible gas control system tests, including hydrogen monitors and analyzer. (Refer to Section 6.2.5.4 of your FSAR.)

c. Fuel storage system tests, including:

   1. Spent fuel pit cooling system tests, including the testing and antisiphon devices and low water level alarms.

   2. Operability and leak tests of sectionalizing devices and drains and leak tests of gaskets or bellows in the refueling canal and fuel storage pool.

d. Containment isolation valve function and closure timing tests.

e. Containment penetration leakage tests.

f. Containment airlock leak rate tests.

g. Integrated containment leakage tests.

h. Isolation initiation (CRVICS) logic tests. (See Section 7.3.2.3.3 of your FSAR.)

i. Containment air purification and cleanup system tests. (Refer to Section 6.5.1.4.1 of your FSAR.)

j. Bypass leakage tests.

k. Autodepressurization system tests. Testing should include items such as sensor and logic train operability, accumulator capacity, relief valves and operability using all alternate power and pneumatic supplies.

l. Emergency response information system (ERIS) tests.

m. Reactor water sampling system tests. Verify that the test will be adequate to verify flow paths, holdup times and procedures.

n. Preoperational testing to determine expansion, vibration, and dynamics effects for: (1) ASME Code Class 1, 2, and 3 systems; (2) other high-energy piping systems inside seismic Category I structures; (3) high-energy portions of systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level; and (4) seismic Category I portions of moderate-energy piping systems located outside containment.

640.08 (14.2.12) Modify your acceptance criteria in Section 14.2.12.1.4 of your FSAR for the preoperational test of the reactor water cleanup system to ensure that the system meets the required head and flow values.

640.09 (14.2.12) Modify in Section 14.2.12.1.5 of your FSAR, the general test methods and acceptance criteria for the Standby Liquid Control System Preoperational Test to include:

a. Testing to verify proper mixing of the neutron absorber solution.

b. Test firings of the explosive-actuated injection valves.

c. Demonstration of the design injection rate capability in accordance with Section 9.3.5.3 of your FSAR.

d. Flow testing for all modes listed in Section 9.3 and Table 9.3-8 of your FSAR.

e. Verification that the manual system initiation, both local and remote, operate properly.

640.10 (14.2.12) Expand the following test descriptions to include, either directly or by reference, the applicable features included in Section 5.4.7.4 and 6.3.4.1 of your FSAR. These tests are the Residual Heat Removal System Preoperational Test (Section 14.2.12.1.7); the Low Pressure Core Spray System Preoperational Test (Section 14.2.12.1.12); and the High Pressure Core Spray System Preoperational Test (Section 14.2.12.1.14).

640.11 (14.2.12) Describe in Section 14.2.12.1.12(3) of your FSAR, how the proper operation of the fuel handling and the vessel servicing equipment will be tested prior to handling fuel.

640.12 (14.2.12) Expand the test description of the Liquid and Solid Radwaste Systems Preoperational Tests in Section 14.2.12.1.17 of your FSAR to specify those subsystems and components which will be tested and the particular test to be performed.

640.13 (14.2.12) Explain in Section 14.2.12.1.18 of your FSAR how the Reactor Protection System Preoperational Test will:

a. Account for process-to-sensor hardware (e.g., instrument lines, hydraulic snubbers) delay times.

b. Provide assurance that the response time of each primary sensor is acceptable.

c. Provide assurance that the total reactor protection system response time is consistent with your accident analysis assumptions.

Item (b) above can be accomplished by: (1) measuring the response time of each sensor during the preoperational test; or (2) stating that the response time of each sensor will be measured by the manfacturer's certification process in sufficent detail for us to conclude that the sensor response times are in accordance with the design.

640.14 (14.2.12) The Process Computer Interface System Preoperational Test should not be considered within the scope of the GESSAR II FDA unless the system description is also covered in your FSAR. Accordingly, either delete this test from Section 14.2.12.1.23 of your FSAR or describe the interfaces in Chapter 7.

640.15 (14.2.12) Add in Section 14.2.12.1.26 of your FSAR, verification of alarms and recorders in the Offgas System Preoperational Test.

640.16 (14.2.12) Modify in Section 14.2.12.1.27 of your FSAR, the general test method and acceptance criteria for the Environs Radiation Monitoring System Preoperational Test to include the filter equipment.

640.17 (14.2.12) Modify in Section 14.2.12.1.35 of your FSAR, the test abstract for the Demineralized Water and Condensate Distribution System Preoperational Tests to include testing of the isolation valves and the ability of the system to satisfy the appropriate interface requirements (Section 9.2.3.2).

640.18 (14.2.12) Modify in Section 14.2.12.1.36 of your FSAR, the acceptance criteria for the Clean and Dirty Radwaste Drains Preoperational Tests to ensure that drain flow to proper sumps.

640.19 (14.2.12) Revise the test description of the Heated Water Distribution System Preoperational Test (Section 14.2.12.1.40) to specify testing at design temperatures or justify how testing at lower temperatures will verify the operation and safety of the system at the rated temperatures.

640.20 (14.2.12) Expand the Polar Crane Preoperational Test in Section 14.2.12.1.53 of your FSAR to include a static load test of 125 percent of the maximum critical load.

640.21 (14.2.12) Provide test descriptions of the following tests which will ensure that the systems under test meet the design and construction requirements described in Chapter 8 and 9 of your FSAR. Our position is that the scope of Chapter 14 testing requirements should parallel the requirements for design and construction and the balance of plant (BOP) interfaces specified in other sections of your FSAR. These tests are the Heating, Ventilating, and Air Conditioning (HVAC) Systems Preoperational Test (Section 14.2.12.1.54); the Electric Systems Preoperational Test (Section 14.2.12.1.55); and the RHR Complex Heating and Ventilation System Preoperational Test (Section 14.2.12.1.57).

640.22 (14.2.12) Identify any of the post-fuel loading tests described in Section 14.2.12.3 of your FSAR which are not essential to the demonstration of conformance with design requirements for structures, systems, components, and design features which meet any of the following criteria:

a. Will be relied upon for the safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

b. Will be relied upon for the safe shutdown and cooldown of the reactor under transient (i.e., infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

c. Will be relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility Technical Specifications.

e. Are assumed to function, or for which credit is taken, in the accident analysis of the facility as described in your FSAR.

f. Will be used to process, store, control, or limit the release of radioactive materials.

These tests will be exempt from operating license conditions requiring NRC prior approval for major test changes.

640.23 (14.2.12) Add a test description in Section 14.2.12.3 of your FSAR for a high temperature containment penetration area concrete temperature survey as described in previous applications for an operating license.

640.24 (14.2.12) You do not establish prerequisites in Section 14.2.1.5 of your FSAR for the following test abstracts even though this particular section is referenced in the test abstracts. These test abstracts are the Fuel Loading Test (Section 14.2.12.3.3) and Full Core Shutdown Margin (Section 14.2.12.3.4). Accordingly, modify Sections 14.2.12.3.3 and 14.2.12.3.4, as necessary, to remove this discrepancy.

640.25 (14.2.12) Modify the test abstract for the Control Rod Drive System Test (Section 14.2.12.3.5) to include the following test requirements:

a. Perform full-flow and no-flow scrams to bound the conditions under which the control rods might be required to function to achieve plant shutdown or provide a detailed technical justification which will ensure that your test conditions have, in fact, bracketed the expected operating envelope.

b. Perform tests on the control rod decelerating devices.

c. Modify the table contained in Section 14.2.12.3.5.3 of your FSAR as follows:

1. In the first control rod drive test, change "Indication" to "Position Indication" and add "all" in the "Preop Test" and "0 (psig)" column.

2. In the last control rod test, add "normal" to the "Accumulator Pressure" column and delete "normal" from the Preop Test Column.

d. Include in the acceptance criteria, a scram time versus the RPV pressure envelope for individual control rod drive scram measurements.

640.26 Provide a description of how the first reactor heatup will be accomplished (i.e., pump heat, nuclear or auxiliary steam). If non-nuclear, indicate what tests will be performed. Also indicate if non-nuclear heatups will be performed before or after fuel loading or both.

640.27 (14.2.12) Modify the test abstract for the Reactor Core Isolation Coolant (RCIC) System Test (Section 14.2.12.3.12) to address the following concerns:

a. Our review of licensee events reports (LER's) has disclosed several instances of RCIC pump failure to start on demand and of inadvertent trips. It appears that many of these deficiencies could have been avoided through better testing during the plant's initial test programs. To demonstrate the reliability of the RCIC system, state your plans to demonstrate cold, quick pump starts over a wide range of pressures during your initial test program. Include starts initiated by both manual means and by injection of simulated low water level signals.

b. IE Information Notice No. 82-13, dated May 28, 1982, "HPIC/RCIC High Steam Flow Setpoints," discussed problems pertaining to incorrect setpoint values for the RCIC steam supply line high flow isolation trip. Accordingly, modify the Level 2 criteria to:

1. Ensure that the differential pressure switch setting is accomplished in accordance with the guidance provided in the IE notice cited above.

2. Describe whether there are any time delay devices (e.g., orifice snubbers or electronic timers) used to preclude spurious isolation trips. Include the testing of these time delay devices.

640.28 (14.2.12) Modify Section 14.2.12.3.16.2 of your FSAR to include determination of the minimum critical power ratio in the Core Performance Test (Section 14.2.12.3.16.3) and any other thermal-hydraulic or power distribution limits.

640.29 (14.2.12) Include tests to determine the runout capability and the loss of maximum credible feedwater heating capability in the Feedwater System Test (Section 14.2.12.3.19).

640.30 (14.2.12) Provide a description of how the startup test data will be recorded. Indicate the parameters to be recorded (i.e., the signal list), the equipment to be used (i.e., Startrec, ERIS), and how the portable instrumentation will be isolated from the permanently installed instrumentation. Alternatively, indicate that the information cited above will be included in the OL applicant's FSAR.

640.31 (14.2.12) You state in Section 5.4.5.2 of your FSAR that the valve poppet of the main steam isolation valves (MSIV) is closed at about 90 percent of the valve stem travel and that the last 10 percent of travel closes the pilot valve only. Accordingly, provide technical justification in the description of the Main Steam Isolation Valves Test (Section 14.2.12.3.21) for your linear extrapolation from 90 percent to 100 percent closed.

640.32 (14.2.12) State in the Relief Valves Test description (Section 14.2.12.3.22) whether the temperature return to within 10 F of the initial temperature is a Level 1 or a Level 2 acceptance criterion. Our position is that it should be a Level 2 criterion and not both a Level 1 and a Level 2.

640.33 (14.2.12) Verify in the Turbine Trip and Generator Load Rejection Test description (Section 14.2.12.3.23) that both turbine trips (stop valve closure) and generator trips (fast control valve closure) will be conducted at full rated power (test condition 6), in both the manual and automatic flow control modes. Alternatively, provide technical justification which shows how proper protective actions for the turbine and the reactor can be demonstrated with a reduced number of trips.

640.34 (14.2.12) Modify the test description for the Shutdown From Outside the Main Control Room Test (Section 14.2.12.3.24) to address the following:

a. State that all personnel actions including scram and MSIV closure will be accomplished from outside the control room.

b. Demonstrate that the plant can be maintained at stable hot, standby conditions for a least 30 minutes.

c. Demonstrate operation of the RHR system in the suppression pool cooling mode with change over to shutdown cooling mode. State that the cooldown in the shutdown cooling mode will lower coolant temperature at least 50 F.

640.35 (14.2.12) Modify the test description of the Recirculation System Test (Section 14.2.12.3.26) to include two-pump trips as indicated in Table 14.1-3 and to determine the drive flow coastdown curve. Modify Table 14.1-3 of your FSAR to indicate the correct test condition for the non-cavitation test.

640.36 (14.2.12) Except for the test title, the test description for the Loss of Turbine-Generator and Offsite Power Test (Section 14.2.12.3.27) is essentially identical to the Turbine Trip and Generator Load Rejection Test (Section 14.2.12.3.23). Accordingly, revise this test description to describe the Loss of Turbine-Generator and Offsite Power Test. This test should be initiated from a sufficient power level and, as discussed below, should be maintained for a period of time sufficient to demonstrate that the necessary equipment, controls, and instrumentation are available following a simulated loss of offsite power to remove decay heat from the core using the onsite power systems. It is our position that you should initiate this test from a generator output of at least 10 percent and maintain the simulated loss of offsite power for at least 30 minutes in order to demonstrate this capability.

640.37 (14.2.12) Provide either a test description or a suitable reference for a "confirming test" of the RPV Internals Vibration Test (Section 14.2.12.3.29).

640.38 (14.2.12) Revise the Suppression Pool Makeup System Test description (Section 14.2.12.3.36) so as not to describe "periodic" (i.e., surveillance) testing but, instead, describe the testing to be conducted during the initial startup. Clarify the test condition since Table 14.1-3 of your FSAR specifies heatup while Section 14.2.12.3.36.3 specifies shutdown. Indicate in Section 14.2.12.1.45 of your FSAR, the satisfactory completion of the preoperational test as a prerequisite. This test is for an ESF system and should also verify redundancy and divisional separation.

640.39 Compare all test descriptions in Section 14.2.12.3 of your FSAR with recent General Electric Startup Test Specifications provided to BWR-6 licensees and OL applicants. Describe and explain any differences not due to plant-unique features.

640.40 Review the BWR Owners' Group response to Item I.G.1 of NUREG-0737 in their letter from D. B. Waters to D. G. Eisenhut, dated February 4, 1981. Revise Chapter 14 of your FSAR to include Appendix E (additional tests).

640.41 Rearrange the format of Chapter 14 of your FSAR to conform with the standard format recommended in Regulatory Guide 1.70 (November 1978). This will facilitate our review of the interfaces with the FSAR's of future operating license applicants.

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ENCLOSURE 2

TABLE 1

Round 1 Questions Requiring Additional Attention

Question /1	Comment	
(SEB) 220.05	B	SRP 3.5.3
07	B	SRP 3.7.1
09	B	SRP 3.7.2
14 to 15	B	SRP 3.7.2
19	B	SRP 3.8.3
20	B,C	SRP 3.8.3
21 to 22	B	SRP 3.8.2
24	B	SRP 3.8.2
26	C	Buckling factors of safety
27	C	ACI-349, R.G. 1.142
28	B	SRP 3.8.3
30	B	SRP 3.8.1
32 to 33	B	SRP 3.8.3
35	C	ACI-349, R.G. 1.142
36	B	SRP 3.8.3
39	B	SRP 3.8.4
42 to 43	B,C	SRP 3.8.5
44	B,C	SRP 3.7.2
(CMEB) 281.01 to 02	B	SRP 5.4.8
03	A	GDC 60,61
07 to 08	B	SRP 9.3.2
09	A,C	NUREG-0737
(ASB) 410.01 to 04	A	Flooding
09	B	SRP 3.6.1
10	B,C	SRP 3.6.1
12 to 14	A,C	Pipe Failure
15	A	Subcompartment analysis
16	A	Update to Clinton
19	A	High density fuel storage
21 to 22	B	SRP 9.1.2
26	C	RHR cooling mode
30	A	NUREG-0612
32	B	SRP 9.1.4, 9.1.5
37	B	SRP 9.3.1
41	A	Appendix R

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ENCLOSURE 2
(Cont'd)

TABLE 1

Round 1 Questions Requiring Additional Attention

Question /1		Comment	
(PSB) 430 series			

(Special attention should be paid to this portion of the Round 1 questions due to the extensive need for additional information in this review area.)

(ETSB) 460.09		A,C	R.G. 1.143
	10 to 11	A	ESF filters
	12 to 14	A,C	NUREG-0016
	15	A,C	Solid radwaste
	16	A,C	NUREG-0473
	17	A,C	Nearest potable water
	18	A	NUREG-0737
(RAB) 471.05		B	SRP 12.1.2
	08	A	R.G. 8.8
	11	A,C	High doses
	13	A,C	Purging
	14	A,C	R.G. 8.8; high doses
	15	A	Shielding during refueling
	16	A,C	NUREG-0737
	18 to 19	A	NUREG-0737
	20	A	R.G. 8.19

(PTRB) 640 series

(Special attention should also be paid to this portion of the Round 1 questions since we request extensive modifications to your proposed test procedures.)

Notes

/1 where a question is not listed, no comment was made
A insufficient information
B nonconformance with our positions in the Standard Review Plan
C needs special attention