879

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 20

PROJECT MANAGEMENT CORPORATION

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

Question 130.1 (3.3.2.2)

Provide a description of the method used to combine tornado missile, wind and vacuum pressure structural effects.

Response:

The response to this question is contained in PSAR Sections 3.3 and 3.5.4.5. 25

Question 130.2 (3.5.4)

Describe the procedures by which missile effective structural loads are determined.

Response:

In providing an answer to the above question, Section 3.5.4 has been amended by adding a new Section 3.5.4.5.

Question 130.3 (3.7.2.1)

In 3.7.2.1, describe the method used to determine the significant dynamic response modes for seismic system analysis.

Response:

The response to question 130.3 is provided in the revised PSAR page 3.7-8.

Question 130.4 (3.7.2.3)

In Section 3.7.2.3, describe the procedure used to select the number of masses for lumped mass seismic system analysis.

Response:

A description of this procedure can be found in revised Section 3.7.2.3. 25

Question 130.5 (3.7.3)

In 3.7.3, describe the procedure used to select the number of masses for lumped mass seismic subsystem analysis.

Response:

The procedure used to select the number of masses for lumped mass seismic subsystem analysis is the same as that given in revised Section 3.7.2.3 on seismic system analysis. 25

Question 130.6 (3.7.3)

In 3.7.3, describe the procedure for determining damping of seismic subsystems.

Response:

The damping values and procedure to be used in the analysis of seismic subsystems are similar to those given for seismic systems in the revised Section 3.7.2.14 utilizing Table 3.7-2 "Damping Values" of the PSAR. This table is applicable to both systems and subsystems.

Question 130.7 (3.7.3.9)

In 3.7.3.9, describe the criteria used to assure that the fundamental frequency of subsystems is outside the range of the dominant frequency of the support system.

Response:

The response to question 130.7 is provided in the revised PSAR page 3.7-13.

Question 130.8 (3.7.3.14)

In 3.7.3.14, describe the procedure used to analyze multiply supported subsystems.

Response:

The response to question 130.8 is provided in the revised Section 3.7.3.14.

Question 130.9 (3.8.2.4)

Provide a description of the steel containment design and analysis procedures.

Response:

The information requested is incorporated in the changed PSAR page 3.8-7 and added pages 3.8-7a and 3.8-7b.

Question 130.10 (3.8.2)

In 3.8.2, identify the materials, quality control procedures and special construction techniques (if any) for the steel containment.

Response:

The information requested has been incorporated in the changed PSAR pages 3.8-1, 3.8-2 and 3.8-3.

Question 130.11 (5.3.3.1.4)

In PSAR Section 5.3.3.1.4, 5.6.1.3.1.2 and 5.6.2.3.1.4, provide diagrams showing the limits of Category 1 components for the Primary Heat Transport System, the Steam Generator Auxiliary Heat Removal System and the Overflow Heat Removal Service.

Response:

As indicated on Figure 5.1-2, all components of the Primary Heat Transport System shown on the figure are Seismic Category 1. As explained in revised Section 5.6.1.3.1.2 of the PSAR, all Steam Generator Auxiliary Heat Removal System components are Category 1. Revised Section 5.6.2.3.1.4 and revised Figure 5.1-7 clarify the extent of Category 1 design for the Overflow Heat Removal Service.

Question 130.12 (5.6.1.3.1.2)

In PSAR Sections 5.6.1.3.1.2 and 5.6.2.3.1.4, provide lists of components and supporting structures required to comply with Category I requirements for the Steam Generator Auxiliary Heat Removal System, and the Overflow Heat Removal System.

Response:

PSAR Sections 5.6.1.3.1.2 and 5.6.2.3.1.4 have been revised to include references to pertinent PSAR sections providing the requested lists.

Question 130.13 (5.5.3.1.4)

In PSAR sections 5.5.3.1.4, 5.6.1.3.1.3, and 5.6.2.3.1.4 provide diagrams of the seismic models, indicating points of large changes in flexibility, for components of the Primary Heat Transport System, the Steam Generation System, the Steam Generator Auxiliary Heat Removal System, and the Overflow Heat Removal System.

Response:

The diagrams for the Steam Generator System preliminary seismic models are provided in the following new figures:

Steam generator -Fig. 5.5-6
Sodium dump tank - - - - - - - - - - - - - - - - - - - Fig. 5.5-7
Water dump tank -Fig. 5.5-8
Reaction products storage tank - - - - - - - - - - - Fig. 5.5-9

Seismic models for the remaining components of the heat transport systems components are currently being developed and are not available at this time. Seismic models for all major components will be provided in the FSAR.

Question 130.14 (3.3.2.2)

In 3.3.2.2, describe the procedure by which venting, if considered, is used to reduce the tornado differential pressure loading.

Response:

In the design of all Seismic Category I structures, venting is not considered. Therefore, the full effect of tornado differential pressure of 3 psi will be taken into account in the structural design in accordance with Regulatory Guide 1.76.

Question 130.15 (3.4.2)

In 3.4.2, describe the procedure by which sliding and overturning effects due to dynamic water forces are considered.

Response:

In answering the above question, Section 3.4.2 has been amended.

Question 130.16 (3.5.4)

In 3.5.4, describe the procedure for predicting local damage in composite sections (if utilized).

Response:

Multiple barrier missile shields are not utilized in the design of Seismic Category I structures. Wherever missile barriers are required for protecting safety related equipment or components which are vulnerable to damage, protective measures will be included in the design in accordance with procedures described in Section 3.5.4.3.

Composite sections, utilized for the economic design of structural steel beams for supporting concrete floor or roof slabs in some of the Seismic Category I buildings, are not considered in the assessment of local damage due to either external or internal missiles.

Question 130.17 (3.7.1.1)

In 3.7.1.1 indicate the elevation at which the seismic design response spectra are applied.

Response:

In the lumped-mass type of analysis to be used in the seismic calculations for CRBRP, the seismic design response spectra will be applied at the foundation level as indicated in the revised Section 3.7.1.1.

Question 130.18 (3.7.1.1)

In 3.7.1.1 describe the deconvolution analysis procedures and provide the response spectra at the foundation level as compared to the finished grade design response spectra.

Response:

The design response spectra are applied directly at the foundation level; therefore, there is no deconvolution from the finished grade to the foundation level.

Question 130.19 (3.7.2)

In 3.7.2, specify which Category I structures are designated as seismic systems.

Response:

Section 3.2.1 defines the Category I structures, systems and components of the Clinch River Breeder Reactor Plant. Table 3.2-1 lists the Category I structures. No Category I structure is designated as a seismic system.

Question 130.20 (3.7.3)

In 3.7.3, specify which Category I components are designated as seismic subsystems.

Response:

For the purpose of seismic analysis, the procedures given in the PSAR are applicable, as appropriate,to all Category I components. In accordance with the SFAC, the procedures in Section 3.7.2, "Seismic System Analysis," of the CRBRP Safety Analysis Reports apply to a large extent to Category I buildings and structures; while those in 3.7.3 "Seismic Subsystem Analysis" mostly relate to Category I systems (such as piping) and components (such as valves and reactor internals).

Due to the reasons stated above, no specific differentiation between seismic systems and subsystems is to be attempted for Category I components.

Question 130.21 (3.8.3.1)

In 3.8.3.1, provide a detailed description with appropriate diagrams of the reactor cavity cell liner and other cell liners. Furthermore, provide a discussion of the design and design criteria of the following: sacrificial layers in the steel liner and backup concrete; strength and stiffness considerations for the sacrificial layers; stiffness reduction due to possible sodium corrosion; protection of dampers and snubbers from sodium spills; design variations for the cell liners of the primary sodium loops, compared with the cell liners of the secondary sodium loops; thermal buckling of the cell liner, thermal shears in the concrete; post thermal cracking dynamic resistance; and, liner anchorage in sacrificial or thermal cracked concrete.

Response:

The appropriate diagrams of the reactor cavity cell liner and other cell liners are provided in Figures 3A.8-4, 3A.8-5, and 3A.8-6. Design variations for the cell liners of the primary sodium loops, compared with the cell liners of the secondarv sodium loops will not be considered since there are no lined cells in the Heat Transport System outside containment.

Cell liner design criteria are discussed in the PSAR Appendix 3.8-B. A discussion of strength and stiffness considerations for the sacrificial concrete layers are discussed in a new PSAR Section 3A.8.2. Also discussed in PSAR Section 3A.8.2 are stiffness reductions due to possible sodium corrosion and liner anchorage in sacrificial or thermally cracked concrete. For a discussion of snubbers, dampers and other components in relation to sodium spills, see PSAR Section 3.8.3.7. Paragraph 3.3 of the PSAR Appendix 3.8-B lists the Loading Combinations C and D and thermal effects are factored into these two cases. 37

For the extreme accident condition of the sodium spill, preliminary analyses indicate that the effective Von Mises strain will be below 6% which is less than the specified limit of 75% of the ultimate strain of the material, approximately 22%. This maximum strain level is based on analysis performed to date of liner panels and bi-planar corners. Analysis of a tri-planar corner is presently in progress.

For a discussion of protection of dampers, snubbers and other components from sodium spills, see Section 3.8.3.7. 24

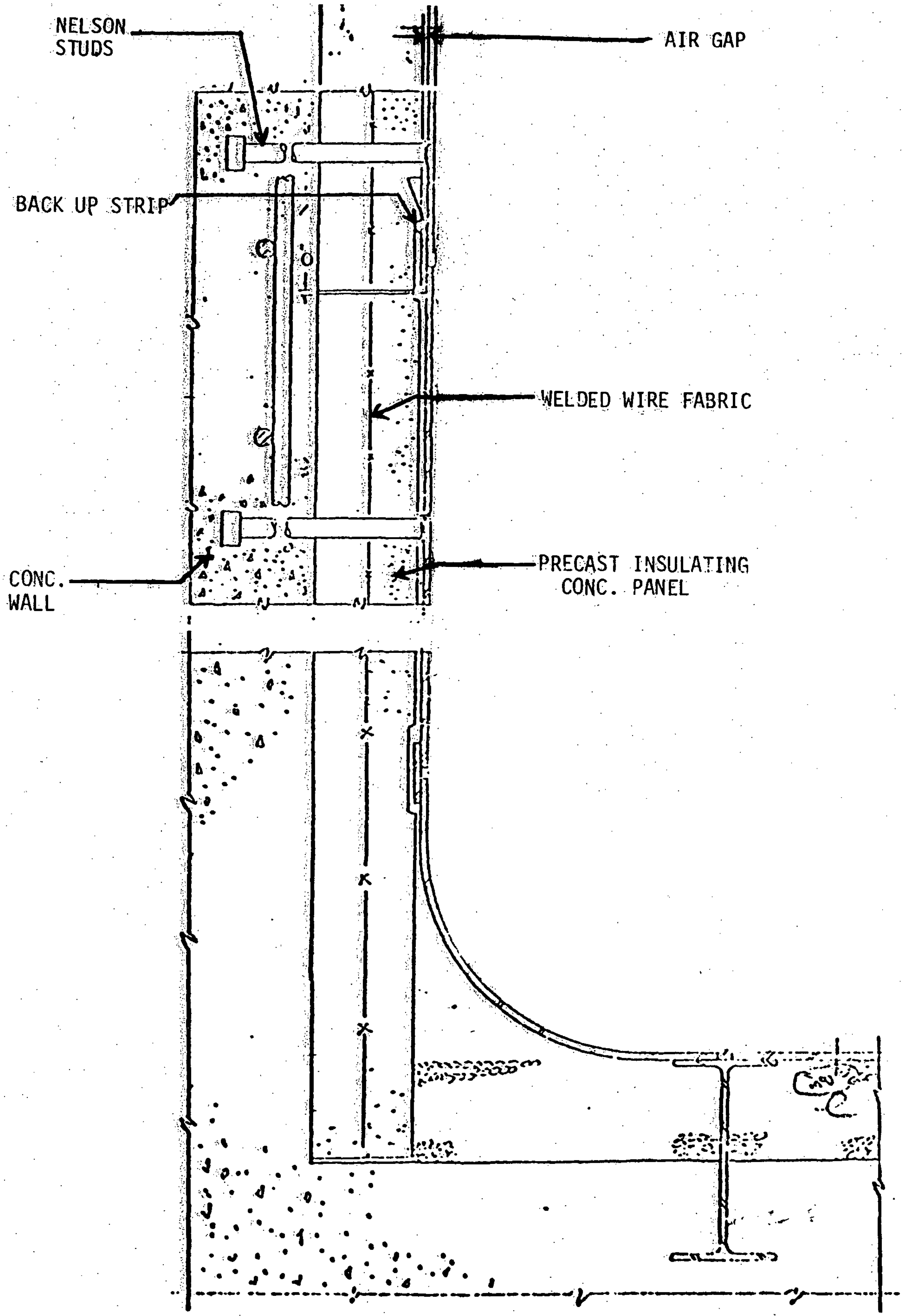


FIGURE Q130.21-1 CELL LINER SYSTEM

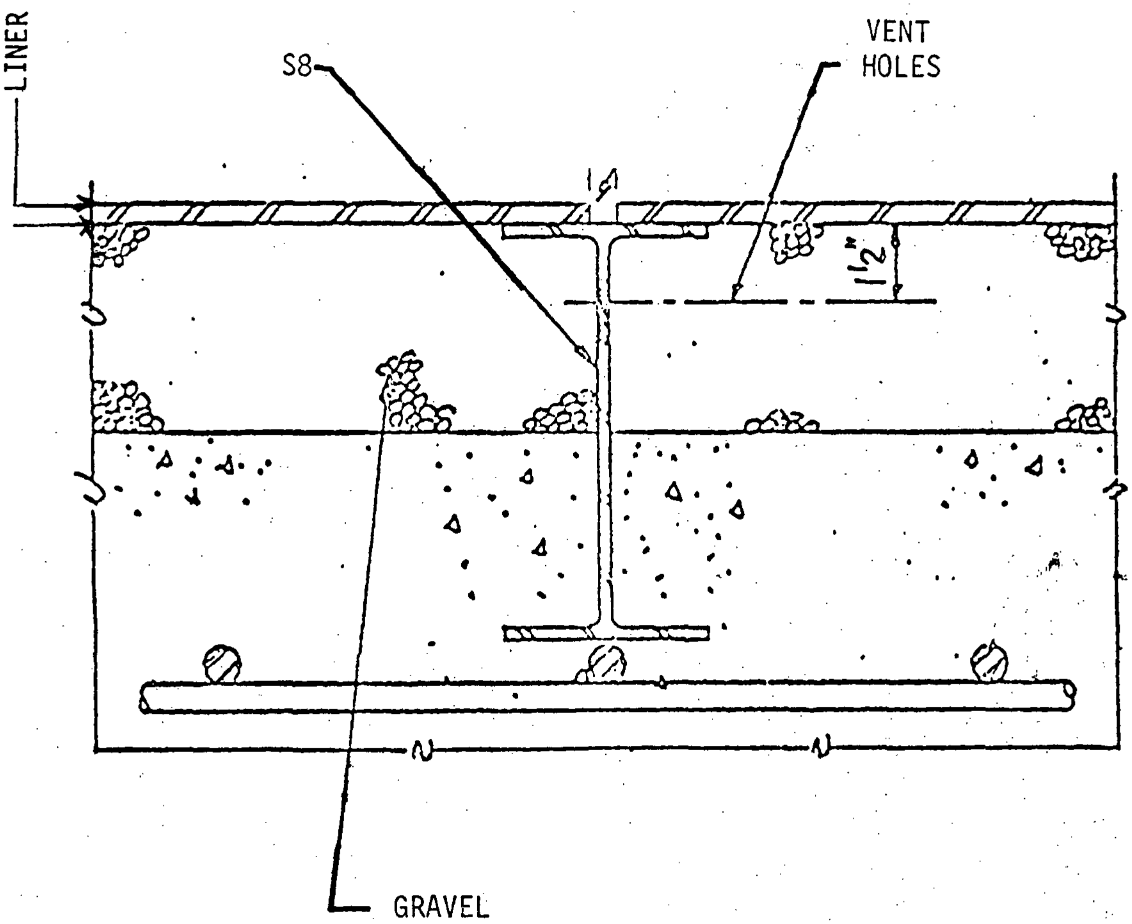


FIGURE Q130.21-2   TYPICAL FLOOR LINER

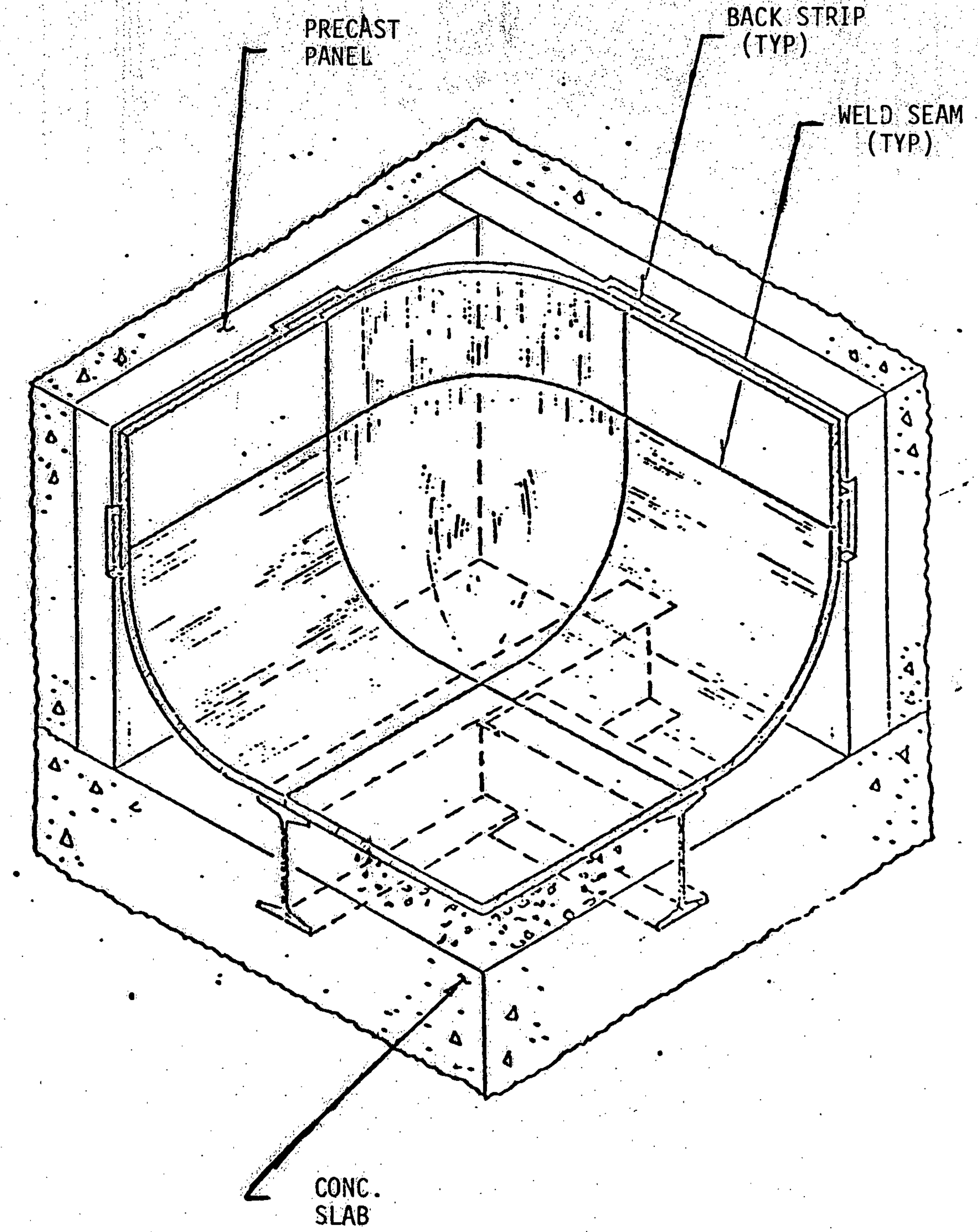


FIGURE Q130.21-3 TRI-PLANAR DETAIL

Amend. 24
July 1976

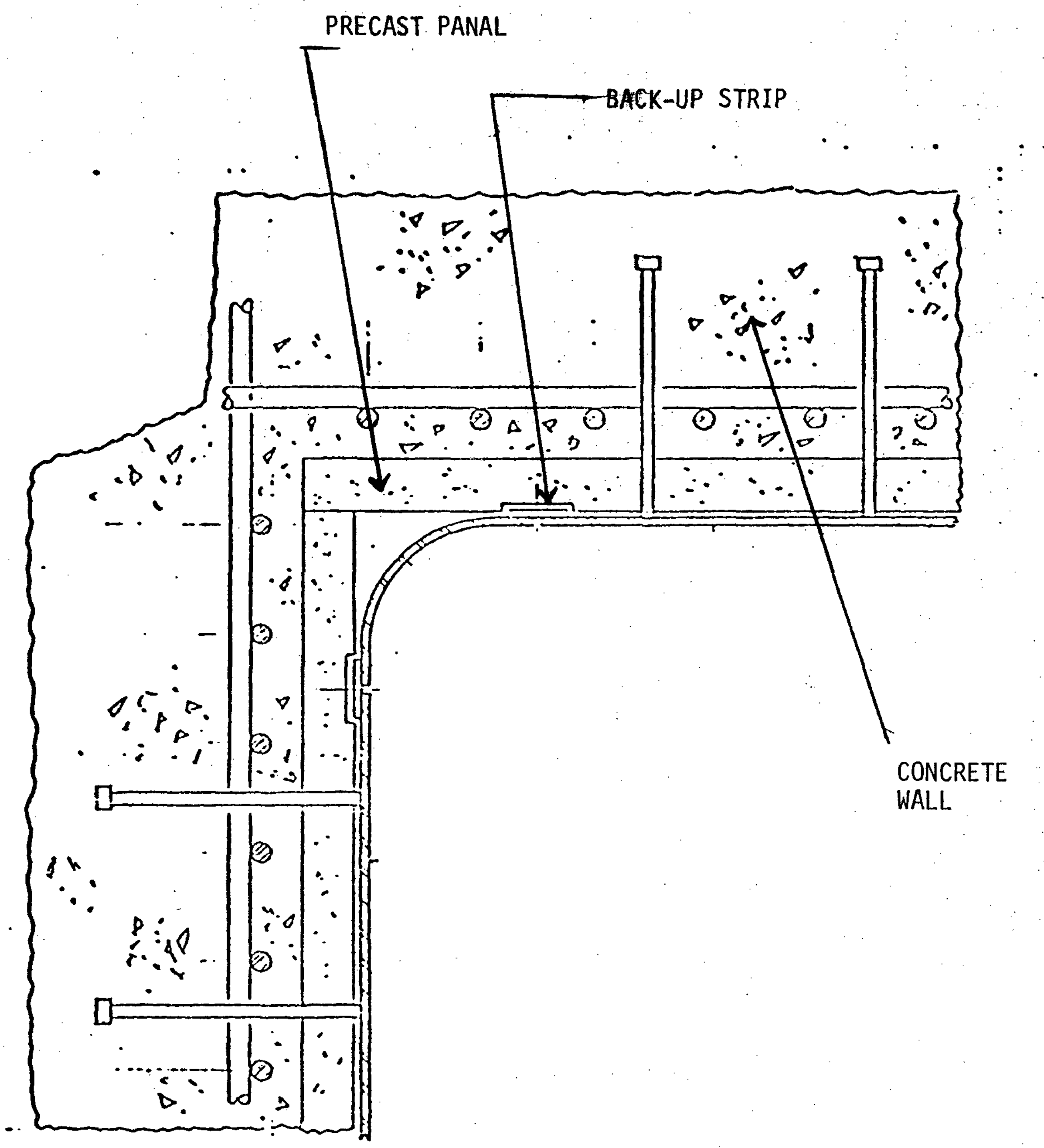


FIGURE Q130.21-4 BI-PLANAR DETAIL

Question 130.22 (3.8.3.3.10.1)

In 3.8.3.3.10.1A.1b, describe the basis for specifying a load factor of 1.4 on "To", the normal operation thermal effects.

Response:

A load factor of 1.4 on "To" is considered appropriate on the basis of the following:

(1) The proposed ACI Standard, "Code Requirements for Nuclear Safety Related Concrete Structures" (Report by ACI Committee 349), as published in February 1975 "Journal of the American Concrete Institute" recommends a load factor of 1.4 on "To". See Paragraph 9.3.1 of this Code for load combinations 9, 10 and 11.

(2) Paragraph 9.3.7 of ACI-318 states that where effects of temperature change may be significant, they shall be included with the dead load D. Since a load factor of 1.4 on D is approved by ACI-318, the same factor should be applied on "To".

(3) Table CC-3230-1 of ASME Code, Section III, Division 2, recommends a load factor of 1.0 on "To" for "service" and "factored" load combinations in contrast to loads like "Pa", "Eo", "L", etc. wherein appropriate load factors in excess of 1.0 are applied. This implies that Division 2 considers "To" the same as "D" in which a factor of 1.0 is also applied.

(4) Some effects of "To" - e.g. temperature gradient through structural sections - unlike "L", are self-limiting and self-equilibrating in plant structures. Since significant stresses are induced in structures due to such effects, a load factor of 1.4 on "To" in view of above is not considered unreasonable.

Question 130.23 (3.8.3.7)

In 3.8.3.7, justify the lack of necessity for testing requirements or in-service surveillance programs for the internal structures, particularly those components with potential contact with sodium, such as cell liners, equipment supports, anchors and anchor bolts, dampers and snubbers, etc.

Response:

The information requested has been incorporated in revised Section 3.8.3.7.

Question 130.24 (3.8.5.3)

In 3.8.5.3, specify the load combinations and safety factors for consideration of sliding, overturning and floatation.

Response:

In providing answer to the above question, Section 3.8.5.3 has been amended by adding a new Section 3.8.5.3.4.

Question 130.25 (3.5.4)

In the design of missile barriers, or other structures which resist impulsive or impactive loads, specify whether or not a dynamic increase in strength of the resisting elements will be utilized. If so, provide the bases for such an increase, and the corresponding dynamic ductility ratios.

Response:

The response to this question is contained in revised section 3.5.4.5. 25

Question 130.26 (3.5.4.4)

In determining the overall structural response due to missile impact load, specify the ductility ratios which are considered to be acceptable for the various structural components.

Response:

The response to this question is contained in revised Section 3.5.4.6. 25

Question 130.27 (3.7.1.6)

Your procedure for soil-structure interaction is not complete. Provide justification for utilizing a static analysis to determine the soil spring constants and the damping coefficients, i.e., show that this yields conservative results when compared with frequency dependent foundation material properties. In addition, describe your method of modeling the dynamic behavior and lateral resistance of the backfill material and surrounding soil adjacent to your deeply embedded foundation walls.

Response:

The response to this question is contained in revised Section 3.7.1.6. 25

Question 130.28 (RSP) (3.7.2, 3.7.3)

Your responses to Questions 130.4 and 130.5 are not adequate. An acceptable criterion to determine an adequate number of masses for lumped mass seismic system is that the number of masses should be such that additional masses, and thus degrees of freedom, do not result in more than a 10% increase in responses. Alternately, the number of masses or degrees of freedom may be taken equal to twice the number of modes with frequencies less than 33 cps.

Response:

The response to Question 130.4 has been revised to incorporate the requested criterion.

Question 130.29 (RSP) (3.7.2.1)

Your response to Question 130.3 is not complete. An acceptable criterion to determine a sufficient number of modes to assure participation of all significant modes is that the number of modes should be such that inclusion of additional modes does not result in more than a 10% increase in responses.

Response:

This criterion is discussed in the revised Section 3.7.2.3.

Question 130.30 (RSP) (3.7.3.9)

Your response to Question 130.7 is not complete. To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure. Alternately, use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads including the effects of dynamic amplification.

Response:

Question 130.7 dealt with Section 3.7.3.9 of the PSAR on the "Use of Simplified Dynamic Analysis". As explained in Sections 3.7.3.5, 3.7.3.9, and revised Section 3.7.2.1.2,, the simplified analysis assumes that the frequency of the subsystem is within the range of, or in resonance with, the predominant frequency of the supporting system. The applicable equipment would thus be designed for the highest dynamic amplification - that given by the maximum peak response on the response spectrum.

Question 130.31 (3.8.1)

Provide the criteria which will be employed in the design, fabrication, and nondestructive examination of containment penetration bellows expansion joints, and specify the code (or code case interpretations), the design loading combinations, the associated operating condition categories that apply (i.e., normal, upset, and emergency), and the corresponding design stress limits that are to be specified.

Response:

The information requested is provided in revised PSAR Section 3.8.2.2.2.

Question 130.32 (3.8.2.2)

Discuss the corrosion potential and protective and/or inspection measures to be taken for that portion of the steel containment that is sandwiched between the foundation concrete and the interior concrete.

Response:

Corrosion potential with respect to the containment vessel is discussed in revised PSAR Section 3.8.2.2.2.

Question 130.33 (3.8.2.3, 3.8.2.4)

Indicate the non-axisymmetric pressure and temperature distributions which could arise and generate the most unfavorable stress and strain distribution in the containment. Describe the analysis procedures for such loads.

Response:

The response to this question is included in revised PSAR Section 3.8.2.4.

Question 130.34 (3.8.2.4)

Describe the analysis procedures and the design features that account for the interaction of the containment shell and the supporting concrete at the level of the operating floor, and for composite action, if any, below that floor.

Response:

Because of the stiffness of the concrete structure at and below the operating floor, the containment shell will be analyzed as fixed at the embedment (Elevation 816 ft.). If the thermal stresses resulting from this condition are excessive, alternate design solutions will be implemented to reduce the stresses to the desired level. For example, it is feasible to include a transition section, consisting of a compressible material between the steel shell and the concrete from the top of the operating floor to the depth necessary to achieve the desired stress levels. If this option were adopted, the flexibility of the transition section would be included in the analysis of the shell.

The reactions obtained from the shell analysis will be applied to the supporting concrete structure. The supporting concrete walls will be analyzed as a composite concrete and steel structure considering the interaction of the steel plate between the exterior and interior walls; where shear transfer is required, shear connectors will be welded to the embedded steel plate.

Question 130.35 (3.8.3)

State whether any heavy aggregate concrete will be used, and if so, where it will be placed, and describe any characteristics and physical constants that will be significantly different from the rest of the concrete, and how compatibility of the different concrete mixes will be assured.

Response:

A very limited amount of heavy-aggregate concrete will be used for mainly shielding purposes on the west side wall of the fuel handling cell. This side wall is located below the operating floor and facing the operating gallery (See Figure 9.1-7 of the PSAR). This location is identified in the revised Section 3.8.4.1.1.

Information pertaining to the characteristics and physical constants of such high-density concrete are described in revised Section 3.8.4.6.1.

To assure the compatibility of the different concrete mixes, the ACI Standard - Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete (ACI 211.1-74) will be followed in the selection of coarse and fine aggregates, proportioning of materials for mix design, and quality control over production and placement. Particular attention will be directed to the details of Appendix 4 - Heavy Weight Concrete Mix Proportioning, attached to the above referenced ACI Standard. 33

Question 130.36 (RSP) (3.8.3.3)

Your response to Question 130.22 is not acceptable. The operating temperature, T_o, is a much more significant load on nuclear safety related structures than on conventional structures. In addition, in structures which are constructed with a complex geometry and subjected to thermal gradients, due to heat convection and radiation, the accurate prediction of thermal loads is much more difficult than the prediction of dead loads. The NRC staff position is that for the strength design method, a factor of 1.7 should be applied to "T_o".

Response:

A load factor of 1.7 will be applied to "T_o" in lieu of 1.4 in the load combinations for concrete structures. Section 3.8.3.3.10.1 has been revised to reflect this change.

Question 130.37 (3.8.3.1)

Additional information is requested in support of your response to Question 130.21 regarding reactor cavity and equipment cell liner response to sodium spills. Provide information regarding your research effort, research schedules or alternate technical basis in lieu of research on this subject. Include a description of the mechanism or scenario for the occurrence of sodium spills and the related quantities, velocities, pressures and temperatures of postulated design bases spills. Also provide the technical bases for establishment of design criteria, design methods, failure modes, failure criteria, consequences of failure, and proof-of-concept or testing requirements for cell liners. When providing this information, consideration should be given to the following items:

A. Temperature Considerations - physical properties of concrete as a function of temperature, short term concrete strength as a function of temperature, degradation of concrete due to temperature, methods of modeling and measuring heat transport in heavily reinforced concrete, design temperature limits on concrete, temperature criteria for establishing boundaries between hot and cold liners.

B. Failure Considerations - potential for and consequences of sodium concrete reactions following liner failure, performance due to a combined sodium spill and seismic event, sodium sprays and jets, rupture criteria for a multiaxial state of stress, brittle failure potential of the liner in irradiated areas.

C. Design Considerations - acceptable strain limits or ductility ratios for normal and accident conditions especially at discontinuities and welds, temperature stress reversals at high and low strain, axial and shear force interaction at embedments, effects of concrete shrinkage, creep and resulting prestressing upon the liner, three dimensional analysis techniques in the corner regions, and pipes and penetrations in the hot liner areas.

Response:

A description of the cell liner design criteria is contained in the new PSAR Appendix 3.8-B. Discussions of liner failure and the research and testing programs are included in the new PSAR Sections 3A.8.3 and 3A.8.4 respectively. 37

D. Consequences of Liner Failure

The consequence of a liner failure is a sodium-concrete reaction and/or the release of radioactivity. Planned testing will verify the extent of the sodium-concrete reaction, however, it is expected that the reaction will be self-limiting and the structure will be able to withstand this reaction. Any radioactive releases will not exceed 10CFR100 limits.

E. Temperature Considerations

The liner will have insulating material between the liner and the structural concrete where necessary to protect the structure from the heat effects of a sodium spill.

F. Performance due to combined sodium spill and seismic event

Stresses during a combined sodium spill and seismic event will be used as the basis for liner design. Stresses for the OBE and SSE will be computed using the response spectrum method. Natural frequencies for the conditions with and without sodium pool will be calculated; it is expected that for both conditions the fundamental frequencies will be very high and that there will be no amplifications of the appropriate floor accelerations.

G. Sodium spray and jets

Sodium sprays and jets will be considered as a design condition. This localized condition will be accounted for in the liner design.

H. Rupture Criteria for a multi-axial state of stress

At embedments where axial, bending and shear stresses are interacting, the criteria for multi-axial state of stress will be applied.

I. Brittle failure potential of the liner in irradiated areas

The increase in ductile-brittle transition temperature due to neutron damage is estimated to be less than 10^{0}F for the reactor cavity liner. This is based on damage function analysis, which indicates that the damage level for the neutron spectrum in the reactor cavity will be approximately 100 times lower than that for LWR reactor vessels.

The maximum neutron fluence on the reactor cavity liner of 7.8×10^{18} n/cm^2 has a neutron spectrum which is predomonately neutrons of less than 0.1 MeV. It is estimated that only 1.2% of the neutrons have energies which exceed 0.1 MeV and only 7.8×10^{-4}% are above 1.0 MeV. This spectrum is significantly "softer" than the typical neutron fluence at LWR ferritic reactor vessels. This difference accounts for the small change in transition temperature in the cavity liner.

Any potential liner brittle fracture will be further minimized by controlling the initial NDT and the concentration of key trace elements in the liner steel.

J. Design considerations

The following Design Criteria will be used for cell liners:

I. Liner Requirements

1. The liner shall be designed so that the accumulated cell average leak rate shall not exceed 1% volume per day under a 2.5 inch H_2O pressure differential to minimize nitrogen usage.

2. The liner shall be designed for maximum long term operating conditions of 180°F.

3. The duty cycle for the liners shall be ten times from 70°F to 140°F, 100 times from 100°F to 140°F and 100 times from 140°F to 180°F for the 30 year plant life.

4. The liner shall be designed to withstand radiation fluence commensurate with its location within the plant, without radiation degradation which would impair its function.

5. The liner shall be designed for corrosion allowances commensurate with environmental conditions for a 30 year plant design life.

6. Floor liner shall be designed to withstand the anticipated floor loads during plant operation and maintenance.

7. The liner shall be designed to insure an essentially elastic response under the normal operating conditions.

8. Material surface condition shall be selected to facilitate decontamination after a sodium spill.

9. The liner shall be designed to contain large sodium spills (faulted condition) with Na spill temperatures up to 1015°F consistent with their application.

10. The external pressure (behind the liner) shall be limited by venting, during a sodium spill.

11. The cell liner shall be constructed, and inspected in accordance with the requirements of Section VIII of this response "Cell Liner Materials, Welding, and Non-Descructive Examination Requirements".

12. Cell liners shall be designed in accordance with Section VII of this response.

II. Liner Anchors

1. The liner anchorage system shall be designed to accommodate tangential shear and normal loads or deformations exerted by the liner. The anchorage system shall be so designed that progressive failure of the entire anchorage system is precluded in the event of a defective anchor.

2. The liner and its anchor system shall be designed to resist the pressure specified in accordance with I-10 above.

III. Penetration Assemblies

Penetration assemblies, including nozzles, reinforcing plates, and penetration anchors shall be designed to accommodate all design loads and deformation without loss of structural integrity in accordance with the leakage reqjirements specified in I-1.

IV. Brackets and Attachments

Brackets and attachments shall be designed to resist all imposed loads without subsequent loss of the liner integrity due to excessive deformation caused by the bracket or attachment loads. The design of bracket and attachment assemblies shall be such that the liner plate is not loaded in the through-thickness direction. Brackets and attachments shall, wherever possible, be placed so that they are backed by an anchor or group of anchors.

V. Seismic Equipment Support

Seismic Category I equipment will not be directly attached to the liners. Structural supports shall be used for this purpose.

VI. Structural Supports

Structural supports penetrating the liner shall be designed to accommodate all design loads and deformations without loss of structural integrity or leakage requirements specified in I-1. Structural supports shall be such that the liner plate is not loaded in the through-thickness direction and that the load in the transverse direction is included in the load combinations.

VII. Load Categories and Load Combinations

Load Categories

The liner shall be designed for the conditions that are specified to occur during the service life of the liner.
These are:

a. Construction loads
b. Normal loads
c. Severe environmental loads
d. Extreme Environmental loads
e. Off normal loads (unlikely fault)
f. Extremely unlikely fault

Construction Loads

Loads resulting from fabrication, construction or preoperational testing are designed as Construction Loads. Loads to be specified in this category are:

a. Dead weight during construction
b. Construction equipment loads
c. Hydrostatic pressure of wet concrete if liner is used as a form.

Normal Loads

Normal loads include loads resulting from system startup, power range operation, shutdown and servicing. Loads that are to be specified in this category are:

a. Dead load, including hydrostatic and permanent equipment load. (D)
b. Live loads, including any movable equipment loads. (L)
c. Pressure differential across cell wall. (Po)
d. Thermal effects due to fluctuations in plant power, loss of cell cooling systems, startup from ambient conditions. (To)
e. Static reactions and loads from piping and support restraints. (Ro)

Severe Environmental Loads

Severe environmental loads are those that could infrequently be encountered during the plant life. This is:

a. Operating Basis Earthquake (E)

Extreme Environmental Loads

Extreme environmental loads are those that are credible but highly improbable. This load is:

a. Safe Shutdown Earthquake (E')

Off Normal Loads (Unlikely Fault)

These are loads which were determined to be prudent to provide a capability for accommodation. These loads are:

a. Sodium or NaK leaks less than 25 kg.
b. Sodium or NaK fires while the plant is shutdown, the system is drained and the cells deinerted.

The effects to be considered under off normal (unlikely fault) loads include:

a. Differential pressures on cell walls due to the events within this category. (Pa)
b. Thermal effects (Ta)

c. Pipe reactions from thermal conditions generated by the events within this category (Ra')

Load Combinations

The following load combinations will be used for design analysis:

Normal/Severe Environmental

1. D + L + To + Ro + Po + E

Extreme Environmental

2. D + L + To + Ro + Po + E'

Off Normal (Unlikely Faulted)

3. D + L + Ta + Ra + Pa + E'

Extremely Unlikely Fault

4. D + L + Ta' + Ra' + Pa' + E'

The load combinations given above are based on the load combinations of the NRC Standard Review Plan, Section 3.8.4 for Category I Concrete Structures and on the ASME Code Section III, Division 2, (the latter specifies unit load factors for liners).

Stress and Strain Allowables

See Table Q130.37-1

VIII. Cell Liner Materials, Welding and Non-Destructive Testing Requirements

Scope

This section covers the materials, welding and non-destructive testing requirements for the cell liner materials.

Applicable Documents

When the specifications for the liners are prepared they shall be evaluated to determine the appropriate codes and standards to be used.

Applicable portions of ASME Code Sections II, III, VIII, IX, the FFTF welding specifications, and RDT Standards F6-5T, F3-6T and F15-2T will be used as appropriate.

Materials Requirements

Materials to be used for the construction of cell liners and welded attachments have been tentatively identified as those in Table Q130.37-2 or equivalent. The material used shall conform to all of the requirements of the code and standards selected. The selected requirements shall be shown to satisfy the following supplemental requirements:

"Certified materials test reports for liner and attachment materials, including welding materials, shall be furnished to the Purchaser for review and approval. Materials Certifications showing actual mechanical and chemical properties, shall be supplied directly from the original materials manufacturer. Results of all required tests and examinations, and records of base metal repairs and heat treatments shall be included in the Certified Mill Test Report. If any of the required tests or processes are conducted by a subsequent manufacturer, an additional material certification shall be issued and attached to the first".

Fabrication

Liner and Attachments

The selected requirements for the liner fabrication shall conform to the following supplemental requirements:

> All liner seam welds shall be full penetration welds. Where field weld joints cannot be welded from both sides of the liner plates, because of inaccessibility, backing strips are permitted. The applicable weld joints shall conform to the requirements of Section III, Division 2, Subarticle CC-3840.

Welding Processes

The following limitations regarding the welding methods permitted by the ASME Code Section III, Division 2, may be selected:

a. Low hydrogen electrodes shall be used for shielded metal arc welding.

b. Alloy-fortified fluxes may not be used for submerged-arc welding.

c. Filler metal addition must be made if the inert-gas tungsten arc process is used.

d. Gas shielded-arc welding by short-circuiting transfer may be used only to deposit the root pass and additional weld passes in the root region of butt joints; however, deposited weld metal thickness produced by multiple pass technique cannot be more than 1/4-inch.

e. Flux-cored wire designed for operation without the use of externally-supplied gas is not allowed.

Welding Procedures and Performance Qualifications

A written welding procudure specification, containing information detailed in Form QW 482 of ASME, Section IX, and the qualification test data containing the information detailed in Form QW483 of ASME Section IX shall be submitted to the Purchaser for review and approval four (4) weeks before initiation or contract work.

A certificate of welder or welding operator performance qualification test shall contain the information as detailed in Form QW484 of ASME Section IX.

All welding repairs shall be made in accordance with a written welding procedures.

Backing Strips

Where Backing Strips are used they shall be flat machine type. Backing strips shall be analytically compatible with the materials being joined. Backing strips shall be tack welded in the joint groove of the weld preparation. All backing strips must fit closely to the underside of the weld preparation.

Studs and Anchors

The welding of the studs or anchors to the liner plates shall conform to the requirements of the ASME Code Division 2, Subarticle CC-4543.5 and ASME Section VIII Division 2, Subarticle AF-210.4C or 210.4D as appropriate.

Storage and Handling of Welding Materials

This shall be done in accordance with ASME Code Section III Division 1, Subsection NF2440.

Nondestructive Examination Requirements

Butt Welds

The entire length of all liner seam welds shall be examined by the vacuum-box method using either a bubble solution or a gas detector technique. Corner welds shall be examined by magnetic particle or liquid penetrant examination.

Leak testing by the vacuum box method shall be performed in accordance with Article 1 of Section V of the ASME Code. Leak testing by halogen diode method shall be performed in accordance with Articles 1 and 10 of Section V of the ASME Code.

Leak testing by the helium mass spectrometer method shall be performed in accordance with Articles 1 and 10 of Section V of the ASME Code. Welds showing through-thickness indications by leak testing shall be considered unacceptable. Leaktested welds are acceptable if they meet the acceptance standards of Article 10 of Section V of the ASME Code for the particular leak test method used.

Magnetic-particle examination shall be performed in accordance with Articles 1 & 7 of Section V of the ASME Code. Welds examined by the magnetic-particle method are acceptable if they meet the acceptance standards of CC-5545 of the ASME Code Section III, Division 2, except that the following relevant indications shall be considered unacceptable:

(1) Rounded defects with dimension greater than 3/16-inch.

(2) Four or more rounded defects in any six square inches of surface where this area is taken in the most unfavorable location relative to the defects being evaluated.

(3) Ten or more rounded defects in any six square inches of surface where this area is taken in the most unfavorable location relative to the defects being evaluated.

Liquid-penetrant examination shall be performed in accordance with Article 1 of Section V of the ASME Code. Welds examined by liquid-penetrant method are acceptable if they meet the acceptance standards of CC-5544 of the ASME Code Section III, Division 2.

Non-butt welds shall be examined by either the magnetic particle method or the liquid penetrant method.

Stud welds shall be visually examined for lack of fusion around the periphery of the weld.

Table Q130.37-1

Liner And Anchor Allowables

Category	Liner-Stress/Strain Allowable: Membrane	Liner-Stress/Strain Allowable: Combined Membrane plus Bending	Anchors-Force/Displacement Allowable: Mechanical Loads	Anchors-Force/Displacement Allowable: Displacement Limited Loads
Construction	fst=fsc= 2/3 fpy	fst=fsc=2/3 fpy	-	-
Normal, Severe and Extreme Environmental(2)	ε_{sc}=0.002 ε_{st}=0.001	ε_{sc}=0.004 inch/inch ε_{st}=0.002 inch/inch	Lesser of: Fa=0.67Fy Fa=0.33Fu	0.25 δ_u
Unlikely Fault Loads	ε_{sc}=0.005	ε_{sc}=0.014 inch/inch	Fa=0.9Fy	0.50 δ_u
	ε_{st}=0.003	ε_{st}=0.010 inch/inch	Fa=0.5Fu	
Extreme Unlikely Fault Loads	SEE NOTE 1			

NOTES:

(1) For load combinations which include Extreme Unlikely Fault loads the Von Mises effective strain shall not exceed 0.75 ε_u for both, liner plates and anchors.

(2) For local conditions like jet impingement or minor spills the limits given for Extreme Unlikely Fault Loads shall apply.

where:

- fst = allowable liner plate tensile stress
- fsc = allowable liner plate compressive stress
- fpy = specified tensile yield strength of liner steel
- ε_{sc} = allowable liner plate compressive strain
- ε_{st} = allowable liner plate tensile strain
- Fa = allowable liner anchor force capacity
- Fu = liner anchor ultimate force capacity
- Fy = liner anchor yield force capacity
- δ_u = ultimate displacement capacity for liner anchors
- ε_u = ultimate strain of liner material under the environmental conditions of interest. ε_u shall be separately evaluated for weld metal and base metal; potential aging and hardening effects shall be considered.

TABLE Q130.37-2

Materials For Construction Of Cell Liners And Welded Attachments

1.	Reactor Cavity Floor and Lower Cavity Wall Liner Plates	ASME SA-387 GR.11 or ASME SA-397 GR.22
2.	Liner Plates, Cell Floors, Walls & Ceilings	ASME SA-515 GR.55 or ASME SA-516 GR.55
3.	Studs	ASTM A-108 GR.1022 (a)
4.	Structural Shapes	ASME SA-36

Notes: (a) Silicon content of 0.15 to 0.30 percent is required.

Question 130.38 (E.3.1.4)

Pipe ruptures could produce high velocity sodium streams on structural surfaces. Indicate whether or not such streams could impact upon cold cell liners. Describe your analysis and design procedures to account for such jet forces on hot liners and, if applicable, on cold liners. If structural or leaktightness damage is anticipated, indicate the capability of the damaged liner to isolate subsequent sodium sprays or spray fires from the concrete.

Response:

The response to this question is contained in a new PSAR Section 3A.8.3. 37

Question 130.39 (15.6.1.6) (Green)

The gas temperature transient in the PHTS and RC cells is estimated to remain above 400^{o}F for approximately 15 minutes if no venting takes place. Due to this transient and due to a lesser transient if venting occurs, it is stated that some spalling of the outer portion of the concrete may occur. Indicate where such spalling is expected, and describe the analysis procedures and results which lead to such a conclusion and the effects which such spalling may have upon safety-related systems.

Response:

Structural concrete will be protected via the use of insulating concrete between the liner and the structural concrete as discussed in the response to Question 130.37 and revised Appendix E Supplement section 15.6.1.6.2.

Question 130.40 (F6.3.4.2)

In Figures 6.3-22, 25 and 26 the loads on the concrete ledge and on the anchor bolts do not return to zero. Indicate the significance of this observation and whether or not there are sufficient additional loading cycles to cause low cycle fatigue problems in either the reinforcement, or localized portions of the bolts, anchor plates, cover plates, etc.

Response

This question requests clarification of information which is no longer a part of the current documentation. The Project has since consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. 60

Question 130.41 (F6.3.4.2)

In Figure F6.3-21, a schematic of the ANSYS structural model is presented. It is not clear from this figure how the concrete ledge and haunch are modelled. Describe the structural model for the ledge and anchor bolt analyses. Include in your description, a discussion of the non-linear geometric and material capabilities to account for concrete cracking, elasto-plastic steel behavior, gaps due to lift off of the head assembly, etc.

Response

This question requests clarification of information which is no longer a part of the current documentation. The Project has since consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. An assessment of Structural Margins Beyond the Design Base for the reactor vessel support ledge is provided in Section 5 of Reference 10a, PSAR Section 1.6.

60

Question 130.42 (F6.4.2.5 Yellow)

Figure F6.4-13 depicts the temperature for the cavity liner as well as the coolant temperature. Indicate the corresponding concrete temperatures behind the liner, the extent of structural damage expected, if any, and provide the resulting margins of safety against loss of required function.

Response:

In Amendment 24 to the PSAR, the Project withdrew the Parallel Design from further consideration by the NRC staff. This question, however, requests analytical information pertinent to the TMBDB Design.

An assessment of the concrete temperatures in and the structural evaluation of the reactor cavity following reactor vessel penetration can be found in Section 3.2 of CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 130.43 (F7.2.4)

In Table F7.2-2, EVALUATION OF THE INNER AND OUTER REACTOR VESSEL SUPPORT BOLTS, there is a footnote which states: "Key Assumption: 1. Thread shear area is adequate to develop tensile strength." In Section 15.1.1.3.4.1 of the reference design there is the following statement: "Where analysis has indicated threaded fasteners inadequate, as in the case of the plug/riser assemblies, a structural restrain is accomplished by a load resisting shear ring." These statements infer that there may be some question regarding the adequacy of threaded fasteners to resist the CDA loads. If other types of fasteners are employed indicate their locations and provide a description of their attachment. With CDA load rise times on the order of several milliseconds and response rise times in tens of milliseconds indicate the bases for concluding that threaded fasteners can adequately resist such loads. Consider the effects of high strain rates upon material toughness, mobilization of ductility (particularly in shear), notch sensitivity and material defects. Provide a description of experimental programs, if any, to assess the adequacy of such fasteners at load magnitudes and strain rates equivalent to the CDA.

Response:*

The bolting thread shear area was not designed at the time of writing the PSAR, thus the reason for "Key Assumption 1". Thread shear area is adequate to develop tensile strength. This does not imply any question regarding the 60 adequacy of these bolts as it has been established that the design of threaded fasteners requires separate evaluations of the shank, thread, nut and head to shank transition and that the thread engagement is sufficient to fully develop the strength of the shank.

60

Bolted plug risers were found to be inadequate in restraining SMBDB loads applied to the vessel closure. This resulted in a revision in the plug/riser 60 design incorporating a shear ring as a means or restraining the rotating plugs of the vessel closure. There is no doubt that with the rotating plug restraint requirement removed from the risers, that threaded fasteners could be utilized with confidence.

The above statements do not imply that there is any question regarding the adequacy of the design with threaded fasteners to resist the SMBDB loads. 60 There are no other types of fasteners (besides welds) employed on the reactor vessel or head used to hold a major component in position.

*Note that Appendix F has been withdrawn. The text, upon which the question was based, can now be found in Section 5 of Reference 10a, PSAR Section 1.6. 60

The general adequacy of threaded fasteners in resisting SMBDB loads is implicit 60
in combination with the relatively insignificant increase in yield stress of common hardened steel fastener materials with moderate dynamic strain rate effects. System analysis reflecting SMBDB loads and fastener bolt rise 60
times shows typical strain rate range in fastner bolts from 1 to 6 in/in/sec. The concrete ledge bolts are 4140 steel while the plug riser bolts are 4340 steel. The relatively insensitive increase in yield stress of 4340 steel (representative of 4140) with strain rate over the range of strain rate expected in the fastener bolts during CDA is illustrated in Figures Q130.43-1 and 2 which are reprinted from References Q130.43-1 and 2 respectively. The experimental data indicates fastener bolt material characteristics including toughness, mobilization of ductility (strain hardening propagation of failure from the bolt shank to thread areas), notch sensitivity, and material defects would be no more significant during SMBDB loading than under static loading. Accordingly, 60
experimental test programs directed toward establishing the validity of threaded fasteners formed from hardened steels in relation to the adequacy of such fasterners at dynamic load magnitudes and SMBDB strain rates are not 60
considered to be required and none are planned based on the relative insensitivity of common hardened steel materials to moderate strain rates (1 to 6 in/in/sec).

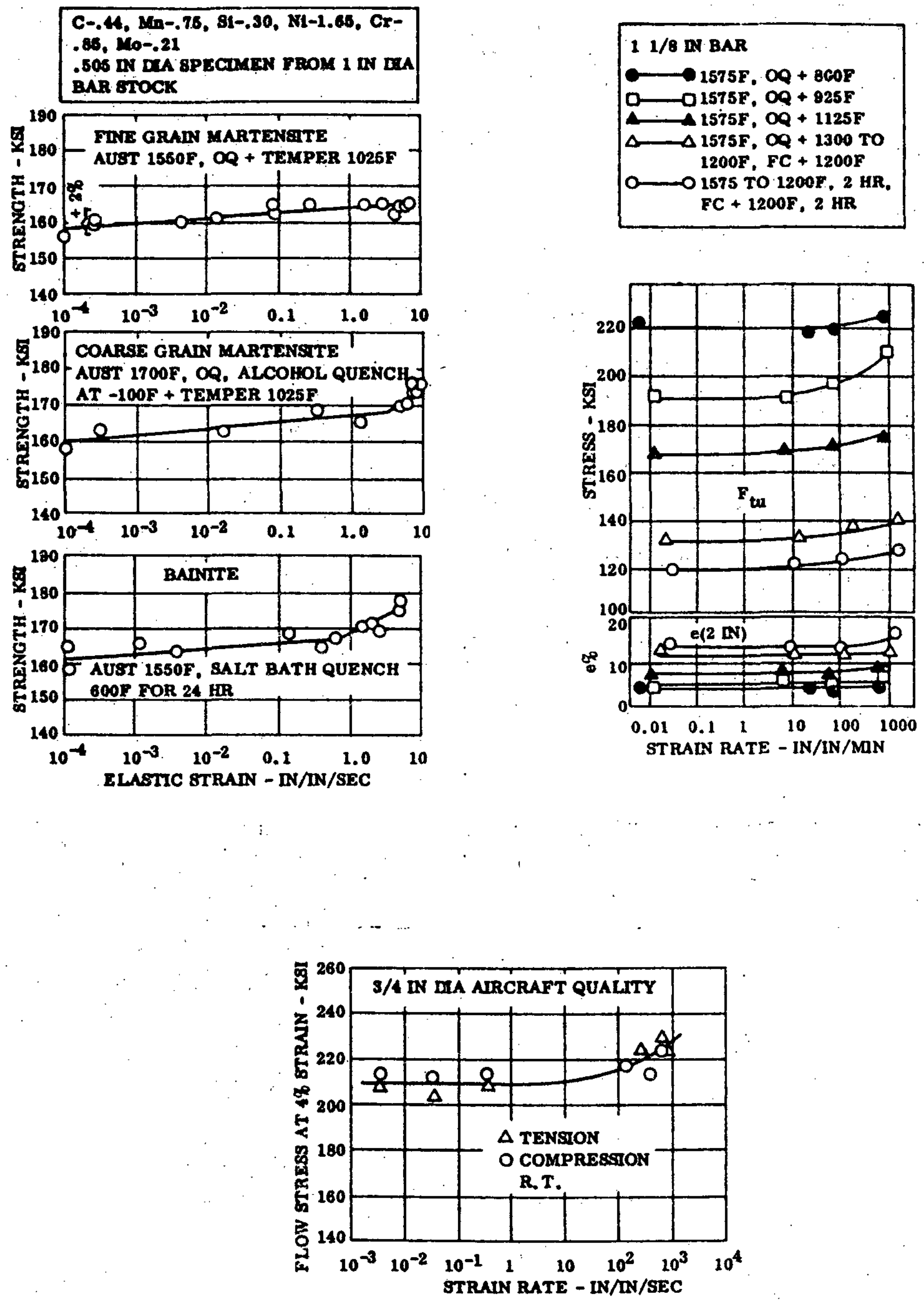
As indicated in PSAR Section 5.2, the vessel support design has been modified to eliminate one row support bolts. The analyses discussed above were performed prior to that modification. However, the modifications are not such that they would alter the conclusions of the analyses with respect to the integrity of the bolts and support ledge. WARD-D-0178 "Closure Head Capability for Structural Margin Beyond Design Base Loading", (Reference 11, PSAR Section 1.6) presents a number of analysis and test results which clearly show that the reactor enclosure has sufficient capability to withstand SMBDB loadings. 60

<u>References</u>:

Q130.43-1 Mechanical Properties Data Center, <u>Structural Alloys Handbook</u>, p. 128, MPDC Belfour Stulen, Inc., Traverse City, Michigan, Vol. 1, 1975 Edition. Includes 1974 Supplement.

Q130.43-2 Air Force Materials Laboratory, <u>Aerospace Structural Metals Handbook</u> p. 8, Code 1206, MPDC Belfour Stulen, Inc.

4340
STEEL



128

Figure Q130.43-1. Reprint from Reference Q130.43-1

Fe
0.4 C
1.8 Ni
0.8 Cr
0.25 Mo

4340,4337

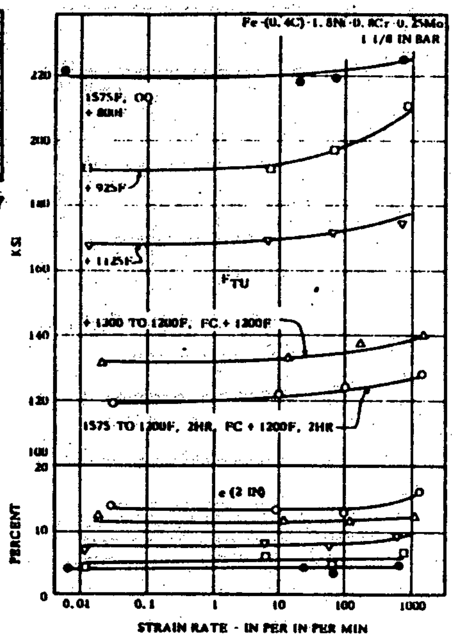


FIG. 3.0213 EFFECT OF STRAIN RATE ON TENSILE STRENGTH AND ELONGATION OF BAR (16, p. 7-25)

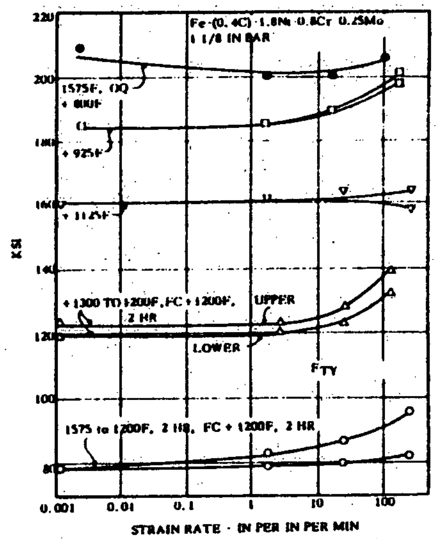


FIG. 3.0214 EFFECT OF STRAIN RATE ON UPPER AND LOWER YIELD STRENGTH OF BAR (16, p. 7-12)

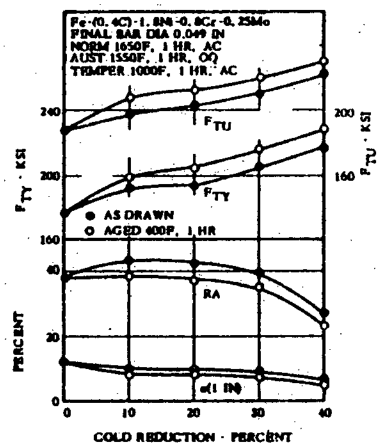


FIG. 3.0215 EFFECT OF COLD REDUCTION ON TENSILE PROPERTIES OF BARS TEMPERED AT 1000 F (53)

CODE	1206
PAGE	8

Figure Q130.43-2. Reprint from Reference Q130.43-2

Question 130.44 (F7.2.7)

Describe the considerations given to a pure shear failure mode (not to be confused with diagonal tension) in the design of the reactor head concrete ledge in the plane of the anchor bolts including an indication of the expected stresses and corresponding allowables.

Response:*

The question relates specifically to information that is no longer a part of the current documentation and is not included in considerations for accidents that are beyond the design base. An assessment of the structural margin of the reactor vessel support system is provided in Section 5.2 of Reference 10a, PSAR Section 1.6. 60

*Note that Appendix F has been withdrawn. 60

Question 130.45 (3.7.2.1.2 Yellow)

Provide the bases for the statement that the Structural Design Basis loading in combination with seismic loadings should not meet the requirements of appropriate codes. Indicate what design requirements will be met and the bases for these requirements. Also indicate what loads will be combined with the CDA loads.

Response:

The probability of occurrence of a CDA is so small that CDA induced loads should not be analyzed in conjunction with loads generated by other low probability events such as an SSE.

The design requirements on the components and systems resulting from the CDA are discussed at length and in detail in Section 5.0 of CRBRP-3, Volume 1 (Ref. 10a, PSAR Section 1.6).

60

Question 130.46 (6.2.7.2 Yellow)

Indicate the extent of the sacrificial concrete around the reactor cavity circular steel shell. Describe the method of analysis used to determine the extent of the concrete degradation or cracking.

Response:

In Amendment 24 to the PSAR, the Project withdrew the Parallel Design from further consideration by the NRC staff. This question, however, requests additional information pertinent to the TMBDB Design with its additional thermal margins as discussed in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6) and will be responded to in that context.

There is no sacrificial concrete around the reactor cavity circular steel shell, as discussed in CRBRP-3, Volume 2. However, a layer of insulating (light weight) material is added between the cell liners and the structural concrete as shown in Figure 3-34 of CRBRP-3, Volume 2. The method of analysis used to determine concrete degradation is discussed in Appendix C.3 of CRBRP-3, Volume 2.

Question 130.47 (Appendices E & F)

Question 130.37 requested additional information with regard to the reactor cavity and equipment cell liner response to sodium spills. Expand the response to this question to encompass the idiosyncrasies of the parallel design, i.e., EVCC and sealed HAA enclosure cells, rapid pressure building on the RC cell, restriction of hot liner expansion in the EVCC, steam buildup behind the RC liner, etc.

Response:

In Amendment 24 to the PSAR, the Project withdrew the Parallel Design from further consideration by the NRC staff. This question requests additional design information on a specific feature of the Parallel Design Accordingly, the question is no longer applicable.

Question 310.48 (15.7.2)

We have performed an evaluation of the radiological consequences of the failure of a RAPS surge vessel assuming credit for the cell being a controlled release environment and using a leak rate of 50% per day (assuming a satisfactory leakage test program in compliance with Appendix J of 10 CFR 50). We calculate that the two-hour dose at the exclusion boundary would be well in excess of the 0.5 rem acceptance criterion given in Standard Review Plan 15.7.1. Since we believe that the radiological consequences of a radioactive waste gas system failure for the CRBRP should not be greater than that for an LWR, it is our position that either the design leak rate of the tank cell should be reduced or else the activity placed in more than one tank such that a failure of any one component will not result in a whole body dose at the nearest exclusion boundary in excess of 0.5 rem. In this regard, indicate what action you plan to take.

Response:

Standard Review Plan Section 15.7.1 "Waste Gas System Failure" was written to guide the review of accidents involving failure of LWR waste gas systems. Specifically, the Plan specifies that offsite doses greater than 0.5 rem, whole body, should not result from single failure of a component of a system "that (complies) with the current staff position on seismic and quality group design requirements". Standard Review Plan Section 11.3 "Gaseous Waste Management Systems", subsection II.3 states that "the seismic and quality group classification of . . . the gaseous waste treatment system . . . should conform to the guidelines of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1)". BTP ETSB 11-1 (Rev. 1) Section II.a states that the system should be designed to industry codes for Quality Group D (non-Safety Class); and Section V.a.(1) requires the system be designed to withstand loadings from the OBE (Seismic Category II). The requirement that a single failure in a non-Safety Class Seismic Category II system result in offsite doses no greater than 0.5 rem is consistent with Regulatory Guide 1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste- containing components of Nuclear Power Plants". That Reg. Guide (Section C.2.e) requires that components "whose postulated failure would result in conservatively calculated offsite doses . . . that exceed 0.5 rem to the whole body . . ." should be constructed to Qualify Group C (Safety Class 3) standards.

The CRBRP Radioactive Argon Processing System (RAPS) has been designated a Seismic Catagory I Safety Class 3 (Quality Group C) system. The PSAR Section 15.7.2.4 provides a conservative analysis of the Safety Class system which shows that accidental offsite doses will be maintained below (2.5 rem wholebody) for failure of such a system.

The RAPS circuit has been relocated with the RAPS surge vessel relocated in the Reactor Containment Building (RCB). Two interconnected vessels are used, but with a total effective volume equal to the original. The RCB has a leak tightness specification of 0.1% per day at a pressure differential of 10 psid, so that the rupture of either of these RAPS surge vessels within the RCB is estimated to result in a maximum 0-2 hr site boundary dose of 0.001 rem. Therefore, this postulated accident is no longer the most serious one in the inert gas receiving and processing system.

52

The revised RAPS circuit has the cold box located in a Reactor Service Building (RSB) cell adjacent to the RCB. A postulated failure of the process gas boundary within the cold box could result in not only the release of absorbed gases from the cryogenic charcoal beds; but also, part of the surge vessel contents could be released into the cold box cell. As a consequence, this cell must have a leak tightness specification with a test program in compliance with Appendix J of 10 CFR 50. For the postulated accident in the cold box cell, the cell tightness required to limit the site boundary dose to 2.5 rem is 40% per day at 11.0 psid.

Because the postulated rupture of the surge vessel in the RCB does not constitute a site boundary dose hazard, and because the postulated cold box rupture in the RSB is potentially the most serious accident in the inert gas receiving and processing system, Paragraph 15.7.2 is retitled and replaced by a description of the cold box accident.

4) Axial tension and Bending

Load combinations will be made for possible maximum axial tension and maximum bending moment as described above. Member should satisfy first for axial tension alone as described above, then it will be checked for combined tension and bending using equation (4) mentioned above with the following exceptions:

f_b = Computed bending tensile stress at the point under consideration as described below:

For load combinations 4, 5 & 6 of PSAR Section 3.8.3.3.10.2.B using elastic section modulus. For load combinations 7 & 8 of PSAR Section 3.8.3.3.10.2.B using plastic section modulus.

Question 130.49 (RSP) (3.5.4.1, 3.5.4.2, 3.5.4.6)

In the response of item 130.26, ductility ratio for concrete elements as high as 25 is indicated. It is the Staff's position that a ductility ratio greater than 10 should not be used unless the bases for using such a high ductility ratio is provided and its use justified.

Response:

The ductility ratios for reinforced concrete members under flexure loading condition have been reduced to a maximum value of 10 for both beams and slabs. This reduction in ductility ratios is indicated in revised Section 3.5.4.6 to comply with the regulatory staff position.

Question 130.50 (3.4.2.1, 3.5.4.2)

From the results of Calspan/Bechtel and EPRI/Sandia missile tests, the staff has established the following minimum concrete wall and roof thickness requirements for Region 1 in which CRBRP is located:

Concrete Strength (psi)	Wall Thickness (inches)	Roof Thickness (inches)
3000	27	24
4000	24	21
5000	21	18

These thicknesses are for protection against local effects only. Designers must establish independently the thickness requirements for overall structural response.

Response:

Actual roof and wall thicknesses will be established based on overall structural response during the final stages of design, however, using a concrete strength of 4000 psi, as specified for CRBRP, and the information provided above the design will meet or exceed the minimum NRC requirements for local effects of missiles.

Question 130.51 (3.7.1.2)

It was stated in Section 3.7.1.2 that "three statistically-independent artificial earthquake records, two for orthogonal horizontal directions and another for the vertical, have been developed". Two of them are shown in Figs. 3.7A-5 and 3.7A-6. The third one should also be included.

Response

A plot of the third ground acceleration-time history (horizontal) is provided in Fig. 3.7A-5A.

Question 130.52 (RSP) (3.7.1.6)

The response to item 130.27 is not acceptable. It is the staff's position that strain-dependent soil properties should be considered in developing the input motions at the foundation evaluation of soil-supported structures. The lumped mass-spring model is acceptable for soil-supported structures that are not deeply embedded or rock supported structures. However, the same idealization shall not be applied to completely buried structures.

Response:

In compliance with the staff position concerning soil structure interaction, in particular, consideration of strain-dependent soil properties and the idealization of completely buried structures, deeply embedded Category I structures founded on soil will be analyzed by finite element techniques. The strain dependent properties of the soil will be considered.

PSAR Section 3.7.1.6 has been revised to reflect this method of analysis.

Question 130.53 (3.7.1.6)

For rock-structure interaction effects, the spring constants determined by the static plane strain finite element analysis should be compared with those using the half space theory. Reasonable correlation between the two sets of results should be demonstrated.

Response:

As stated in section 3.7.1.6 the site conditions are such that half-space theory is not directly applicable to calculate the foundation springs for the seismic analysis. The static finite element analysis considered the actual rock strata and embedment affects. To find a correlation with half-space theory a simplified analysis based on this theory was conducted.

For this analysis the material between grade and the foundation level was assumed homogeneous with a shear modulus equal to the weighted average of the shear modulus of the actual materials in that zone.

$$G_s = \frac{\Sigma G_i h_i}{\Sigma h_i}$$

Where: Gs = shear modulus of the equivalent homogeneous material overlying the foundation

G_i = shear modulus of layer "i" above the foundation

h_i = thickness of layer i

The material underlying the foundation was assumed homogeneous with the shear modulus (G) equal to that of the first layer, siltstone. This assumption will give spring stiffnesses lower than the actual since the limestone layers which are stiffer than the siltstone were not included.

The values used are G_s = 50539.0 k/ft^2 and G = 83077.0 k/ft^2.

The vertical, horizontal and rocking springs were calculated based on the equations of References Q130.53-1 and Q130.53-2 which considered embedded foundations on an elastic half-space. The solutions in these references are based on the approximate analytical approach formulated by Baranov who assumed that the soil underlying the foundation base is an elastic half-space and that the overlying soil is an independent elastic layer composed of a series of infinitesimally thin independent elastic layers. The solutions for the embedment and for the soil underlying the foundation are calculated independently and the results added to obtain the overall spring stiffness. For the torsional spring, the solution of Reference Q130.53-3 was used. Reference Q130.53-3 considered the effects of embedment on the torsional spring of a circular footing on an elastic half-space using a static finite element approach. 40

The torsional stiffness of a circular footing on an elastic half-space is given by:

$$K'_\Theta = \frac{16Gr_o^3}{3} \quad (1)$$

r_o = radius of circular footing.

For use in equation (1) an equivalent radius, based on an equal polar moment of inertia, was found for the actual foundation.

r_o = 218.0 ft.

$G = 8.3077 \times 10^4$ k/ft^2

$K_\theta' = 4.59 \times 10^{12}$ k ft/rad.

For the effect of embedment, Figure 6 of Ref. Q130.53-3 was used.

$$\text{For } \frac{H}{r_o} = \frac{\text{Depth of embedment}}{\text{radius of foundation}} = \frac{100}{218} = 0.46$$

Figure 6 of Reference Q130.53-3 gives $\frac{K_\Theta}{K'_\theta} = 2.3$

Where K_Θ is the total torsional stiffness including embedment. The contribution of the embedment to the total stiffness is then:

$$K''_\Theta = K_\Theta - K_\Theta' = 2.3K_\Theta' - K'_\Theta = 1.3K_\Theta'$$

The ratio of the shear moduli of the overlying and underlying materials is

$$\frac{G_s}{G} = \frac{5.0539 \times 10^4}{8.3077 \times 10^4} = 0.61$$

Since Reference Q130.53-3 assumed the same material for the overlying and underlying soil, K''_Θ needs to be corrected by the ration G_s/G.

$$K''_\Theta = 1.3 \times 0.61 \times 4.59 \times 10^{12} = 3.6402 \times 10^{12} \text{ k ft/rad.}$$

And the total stiffness

$$K_\Theta = (4.59 + 3.64)10^{12} = 8.23 \times 10^{12} \text{ k ft/rad.}$$

The value calculated by the finite element approach with the actual site strata is $K_\Theta = 9.101 \times 10^{12}$ k ft/rad. 40

The ratio between the two solutions is

$$\frac{K_{\theta_o}}{K_\theta} = \frac{9.101 \times 10^{12}}{8.231 \times 10^{12}} = 1.105$$

For the horizontal translational spring, the applicable equation 12 of Reference Q130.53-1 was used:

$$K_H = Gr_o \left(\bar{C}_{u1} + \frac{G_s}{G} \delta \bar{S}_{u1}\right) \quad (1)$$

Frequency independent coefficients were used.

G = Shear modulus of half space = 83077.0 k/ft^2
G_s = Shear modulus of overlying material = 50539.0 k/ft^2
r_o = Equivalent radius of foundation

$$r_o = \left(\frac{A}{\pi}\right)^{1/2} = 208.4 \text{ ft.}$$

Where A = area of foundation

S = Embedment ratio = $\frac{H}{r} = \frac{100}{208.4} = 0.48$

H = Embedment depth = 100 ft.

From table 1 of Reference Q130.53-1 by interpolation, for

$$v = 0.30, \bar{C}_{u1} = 4.78$$

From table 2 of Reference Q130.53-1 by interpolation, for

$$v = 0.30 \qquad \bar{S}_{u1} = 4.03$$

Inserting the corresponding values in equation (1)

$$K_H = 1.032 \times 10^8 \text{ k/ft}$$

From the finite element analysis:

$$K_{H_o} = 1.106 \times 10^8 \text{ k/ft}$$

The correlation is $\frac{K_{H_o}}{K_H} = \frac{1.106}{1.032} = 1.07$

Similarly, for rocking, but using the actual moment of inertia of the base about the East-West centroidal axis, to calculate the equivalent radius of the correlation obtained is: 40

$$\frac{K_{\phi o}}{K_{\phi}} = \frac{5.28 \times 10^{12}}{4.71 \times 10^{12}} = 1.12$$

For vertical translation, based on Reference Q130.53-2.

$$\frac{K_{v_o}}{K_v} = \frac{1.5 \times 10^{8}}{1.12 \times 10^{8}} = 1.38$$

The simplified half-space calculations give values about 10% lower than the finite element except for the vertical spring which is almost 38% lower. This was expected since the effect of the lower layers (limestone) is more significant in the vertical stiffness. The correlation between the finite element and half-space values is therefore considered satisfactory.

References:

Q130.53-1 Beredugo, V.O., Novak, M., "Coupled Horizontal and rocking Vibration of Embedded Footings", Canadian Geotechnical Journal, 9, 477 (1972).

Q130.53-2 Novak, M., Beredugo, V.O., "Vertical Vibration of Embedded Footings". ASCE, Journal of the Soil Mechanics and Foundation Division, SM12, Dec. 1972.

Q130.53-3 Kaldjian, M., "Torsional Stiffness of Embedded Footings", ASCE, Journal of the Soil Mechanics and Foundation Division, SM7, July 1971. 40

Question 130.54 (3.7.2.1)

It is stated on Page 3.7-5 that "the foundation mat will be treated as a rigid member". Clarify if the foundation mat is considered rigid or flexible in your finite element analysis described in Section 3.7.1.6.

Response:

PSAR Section 3.7.1.6 has been clarified to indicate that the foundation mat is considered rigid.

Question 130.55 (3.7.2.1)

The mathematical models shown in Figs. 3.7-16, 3.7-16A, and 3.7-16B do not appear to adequately represent the actual structures. It should be revised to incorporate the addition of the concrete confinement building. Indicate how the buildings are tied together below El. 816. Provide justification of treating these ties as rigid members.

Response:

Figures 3.7-16, 3.7-16A and 3.7-16B have been revised and annotated to reflect the inclusion of the concrete confinement building; Section 3.7.2.1.1 has also been revised to indicate the assumptions made concerning building ties and the justification for these assumptions.

Question 130.56 (3.7.2.3)

Provide in the PSAR a description of the procedures used to lump masses.

Response:

This information has been provided in PSAR Section 3.7.2.3. (Amendment 25 - August, 1976).

Question 130.57 (RSP) (3.7.2.9)

It is the staff's position that the relative displacements have to be imposed in the most unfavorable manner and that the effects caused by input motions, which may be out of phase from one another, may not be negligible and should be considered. Indicate your compliance with these positions.

Response:

A statement confirming compliance with the stated NRC staff position has been included in amended Section 3.7.2.7.

Question 130.58 (3.7.2.13)

It is not clear in Section 3.7.2.13 how the structural stability will be checked. Provide detailed procedure for evaluating the structural stability.

Response:

Section 3.7.2.13 has been revised to include the procedure for evaluating the structural stability.

Question 130.59 (RSP) (3.7.3.6, 3.7.3.14)

The response to item 130.8 is not satisfactory. It is the staff's position that the effects due to differential displacements and due to inertial loadings should be combined by the absolute sum method.

Response:

A statement confirming compliance with the stated NRC staff position has been included in amended Section 3.7.2.7.

Question 130.60 (3.7.3.15)

The hydrodynamic effect of the liquid sodium should be considered in the reactor system mathematical model. If not, justification should be provided.

Response

The hydrodynamic effects of the sodium will be considered in the mathematical model as addressed in updated Section 3.7.3.15.2.

Question 130.61 (3.7.3.12)

The Category I buried piping should be analyzed in accordance with Section 3.7.3.12 of the Standard Review Plan (SRP). The proposed method is not acceptable unless adequate justification is provided.

Response:

Section 3.7.3.12 of the PSAR has been revised to comply with the Acceptance Criteria specified in SRP 3.7.3.12.

Question 130.62 (App. 3.7-a)

Figs. 3.7A-1, 2, 3, 4, 5 and 6 and Tables 3.7A-1 and -2 should be revised to comply with the 0.25 design ground acceleration value.

Response

The project has previously informed the NRC staff that decisions which could impact the overall project schedule are being made on the basis of 0.25g SSE ground acceleration value (Ref: P. O. letter S:L:1253, A. R. Buhl to R. S. Boyd of July 7, 1976). The requested figures and tables are provided as figures Q130.62-1, 2, 3, 4, 5, 5A and 6 and tables Q130.62-1 and 2.

TABLE Q130.62-1

SSE AND OBE HORIZONTAL GROUND RESPONSE SPECTRA CONTROL POINTS VALUE

Percent of Critical Damping	Values for Control Points: Acceleration (g) A (33 cps)	B (9 cps)	C (2.5 cps)	Displacement (in.) D (0.25 cps)
		Safe Shutdown Earthquake		
0.5	0.250	1.240	1.488	28.800
2.0	0.250	0.885	0.063	22.500
5.0	0.250	0.653	0.782	18.450
7.0	0.250	0.568	0.681	16.919
10.0	0.250	0.475	0.569	15.300
		Operating Basis Earthquake		
0.5	0.125	0.619	0.744	14.400
2.0	0.125	0.443	0.532	11.250
5.0	0.125	0.326	0.392	9.225
7.0	0.125	0.283	0.340	[illegible] 460
10.0	0.125	0.238	0.285	7.650

Amend. 31
Nov. 1976

TABLE Q130.62-2

SSE AND OBE VERTICAL GROUND RESPONSE SPECTRA
CONTROL POINTS VALUES

Percent of Critical Damping	Values for Control Points			
	Acceleration (g)			Displacement (in.)
	A (33 cps)	B (9 cps)	C (3.5 cps)	D (0.25 cps)
		Safe Shutdown Earthquake		
0.5	0.250	1.240	1.418	19.169
2.0	0.250	0.885	1.013	15.031
5.0	0.250	0.653	0.744	12.331
7.0	0.250	0.568	0.647	11.250
10.0	0.250	0.475	0.543	10.169
		Operating Basis Earthquake		
0.5	0.125	0.619	0.708	9.585
2.0	0.125	0.443	0.507	7.515
5.0	0.125	0.326	0.372	6.165
7.0	0.125	0.283	0.324	5.625
10.0	0.125	0.238	0.271	5.085

Amend. 31
Nov. 1976

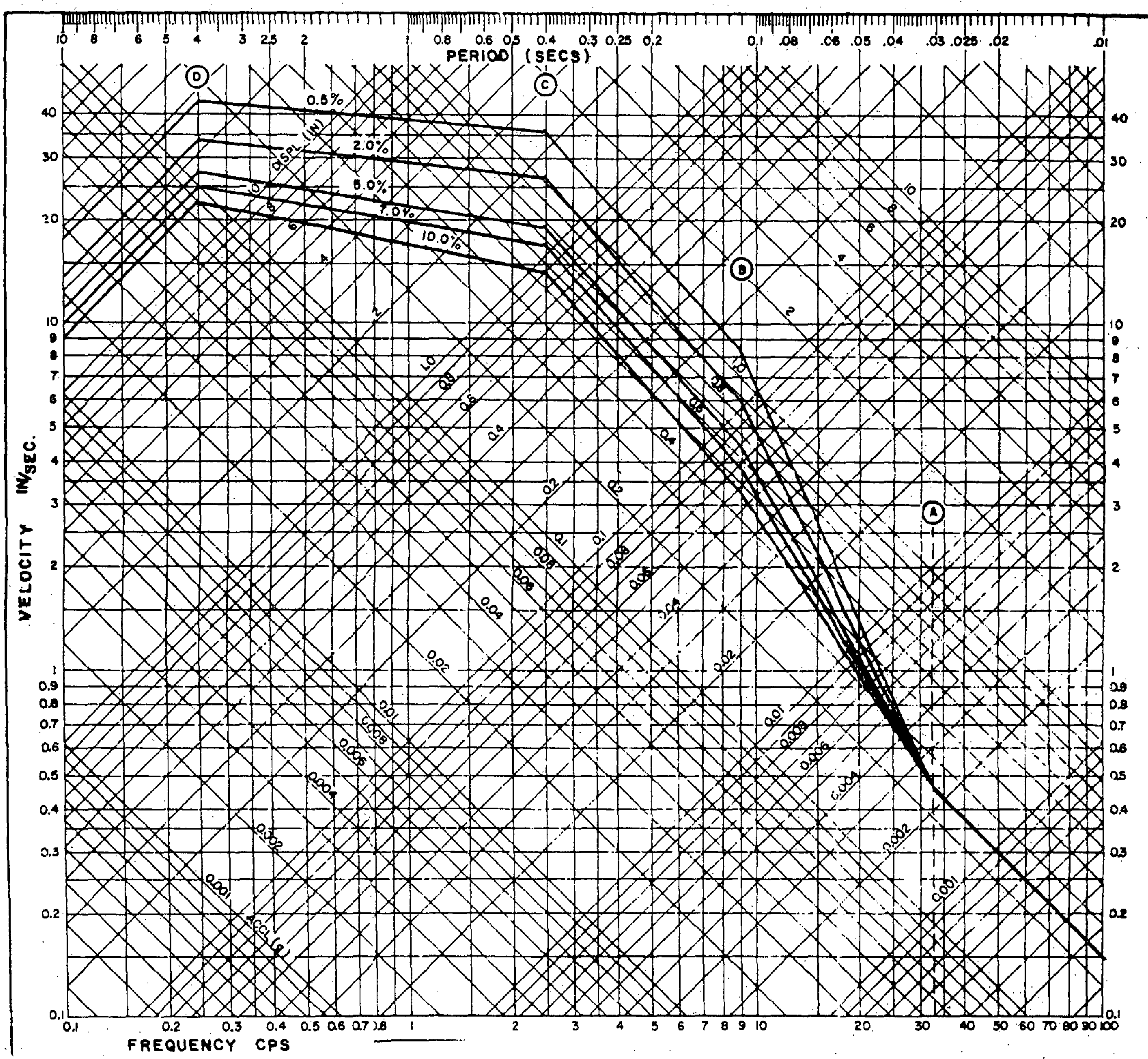


Figure Q130.62-1 HORIZONTAL SSE GROUND REPONSE SPECTRA

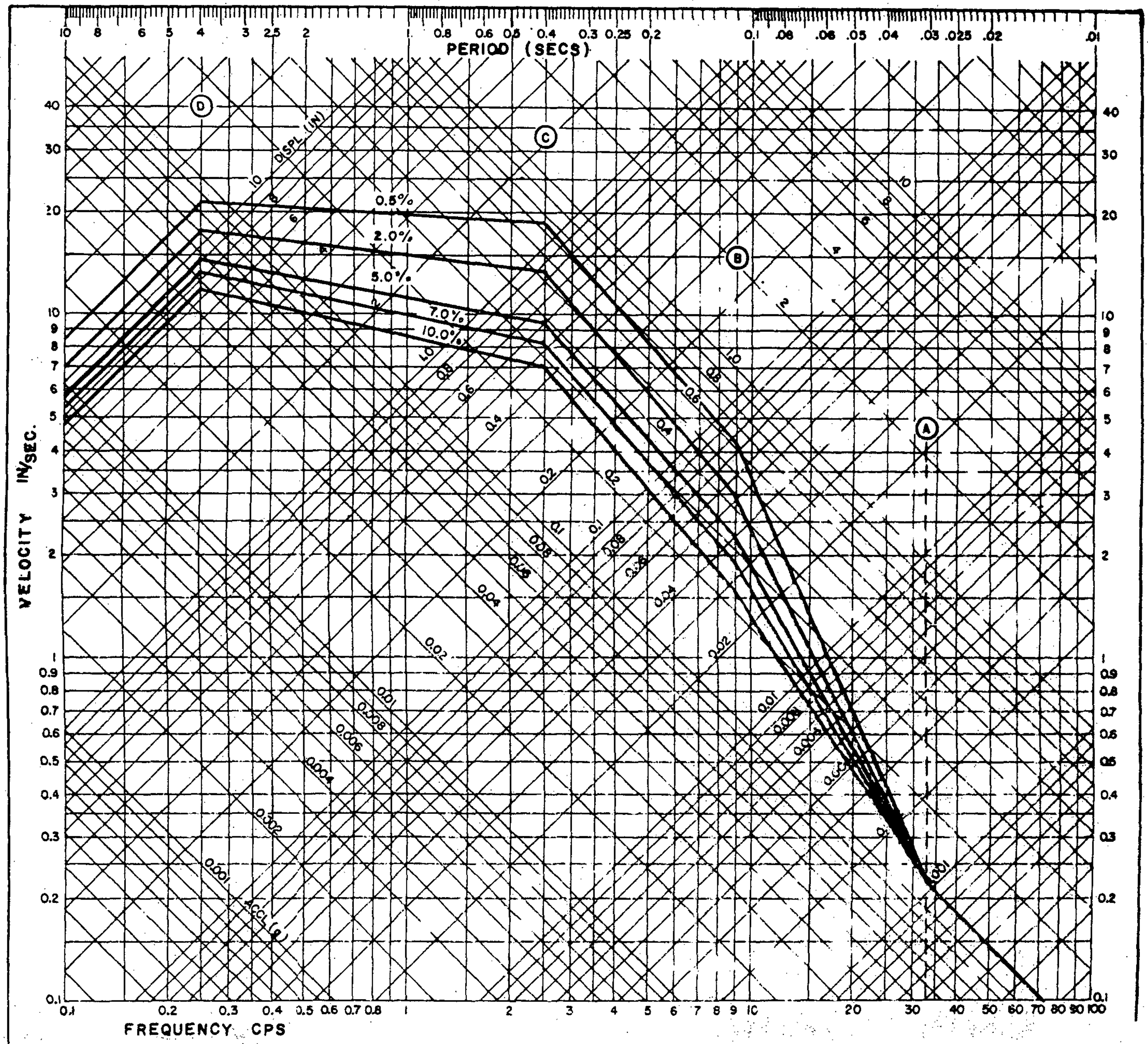


Figure Q130.62-2 Horizontal OBE Ground Response Spectra

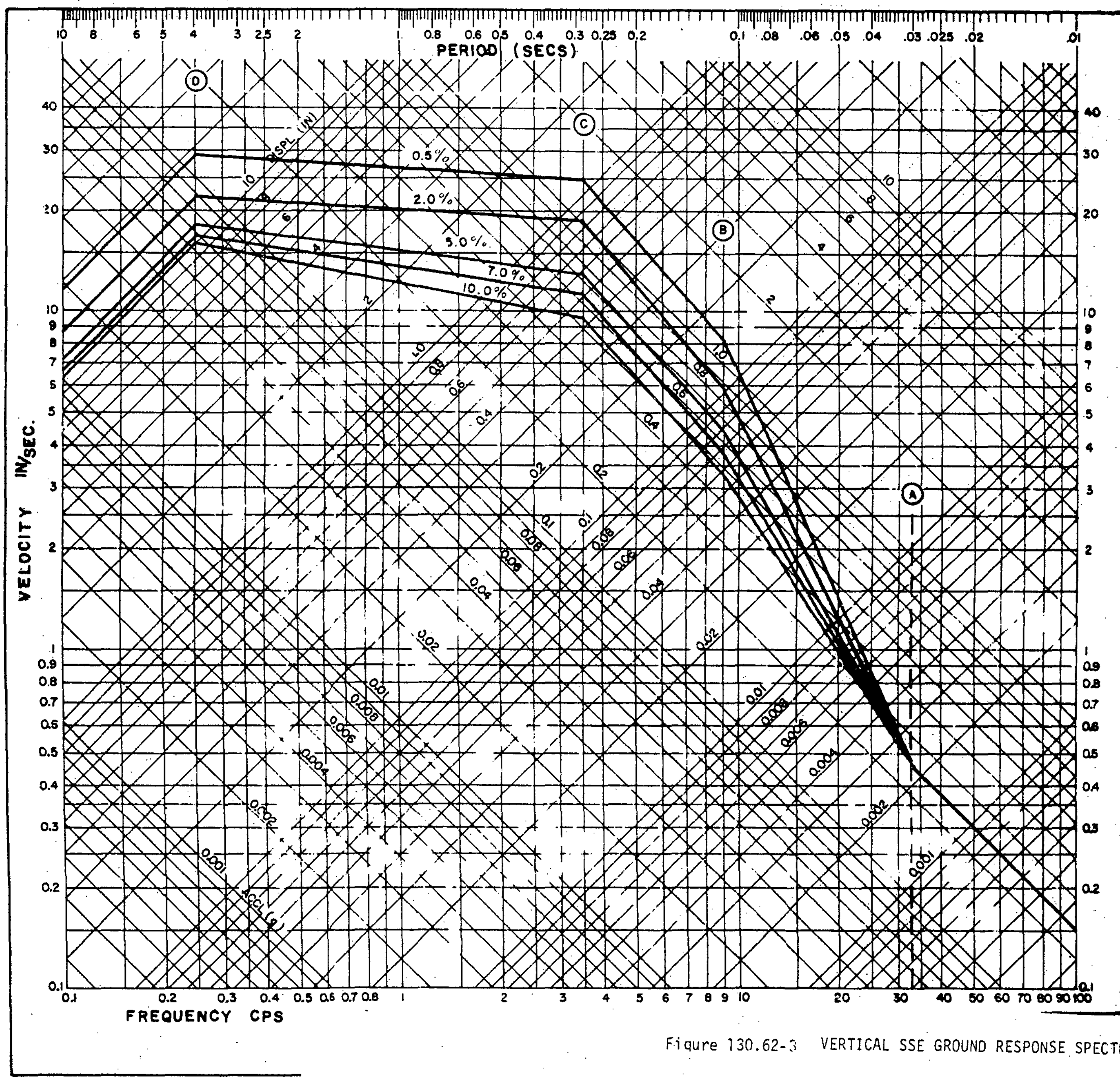


Figure 130.62-3 VERTICAL SSE GROUND RESPONSE SPECTRA

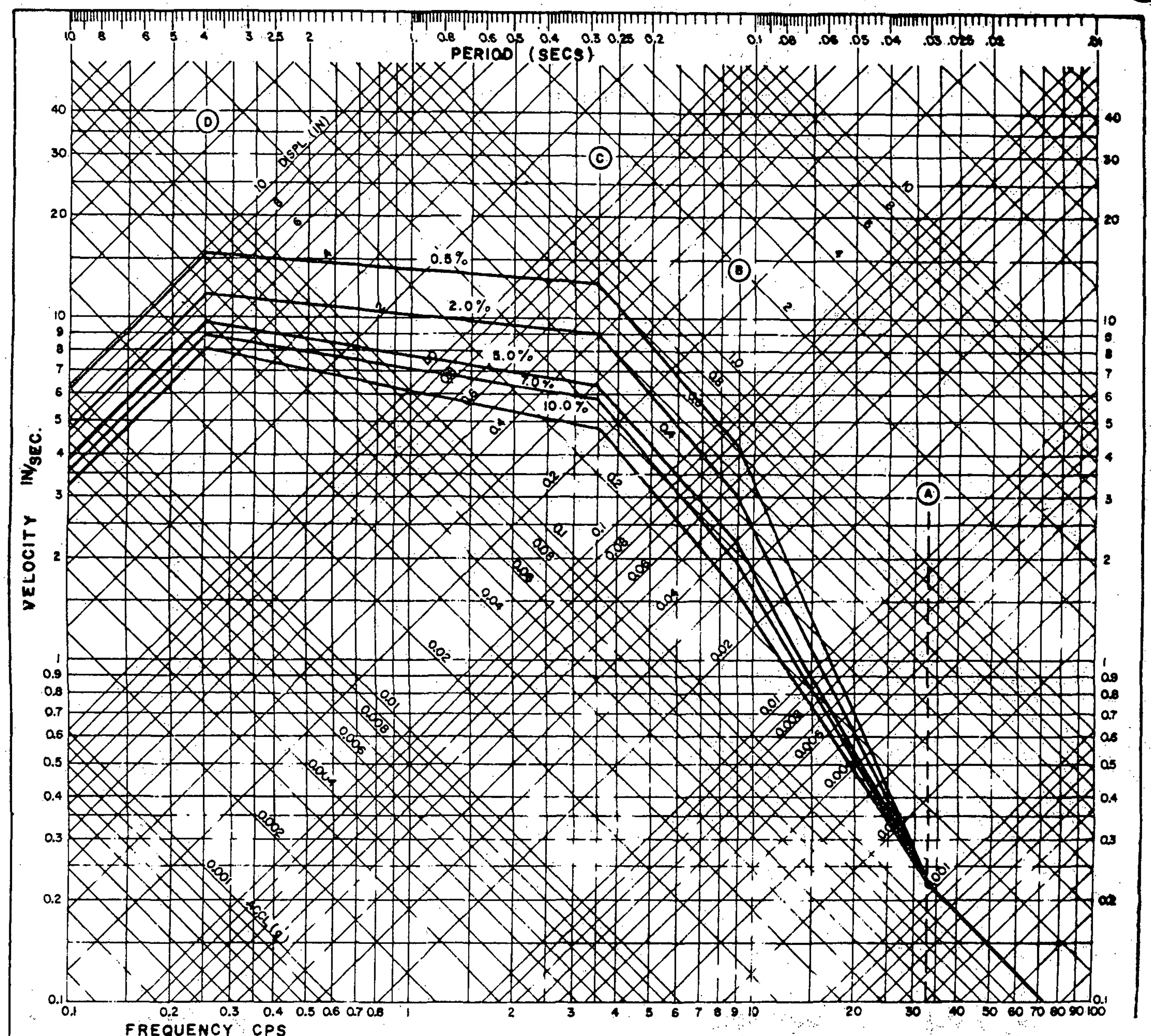
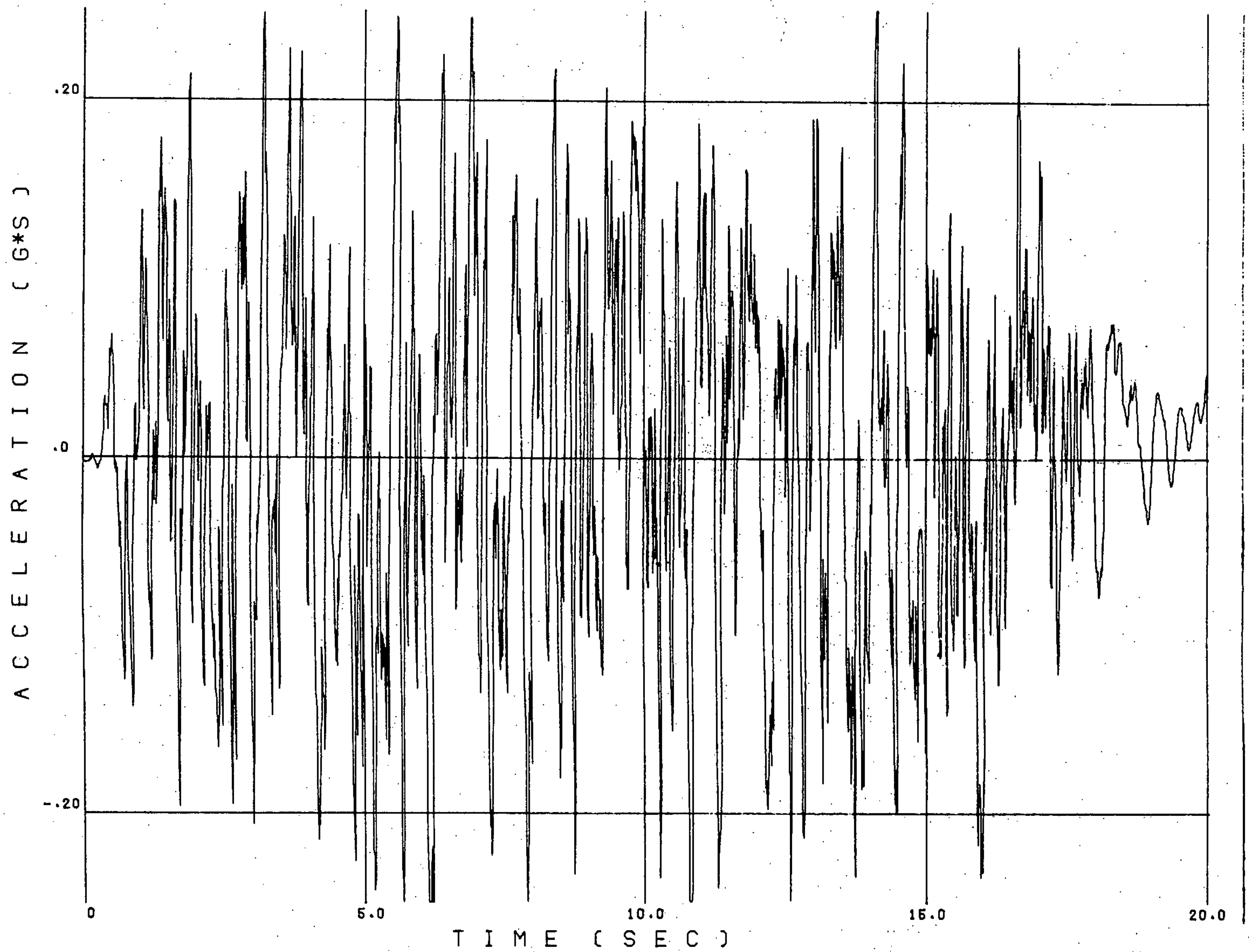


Figure 130.62-4 Vertical OBE Ground Response Spectra



31

Figure Q130.62-5 Horizontal Motion A SSE Ground Acceleration Time-History

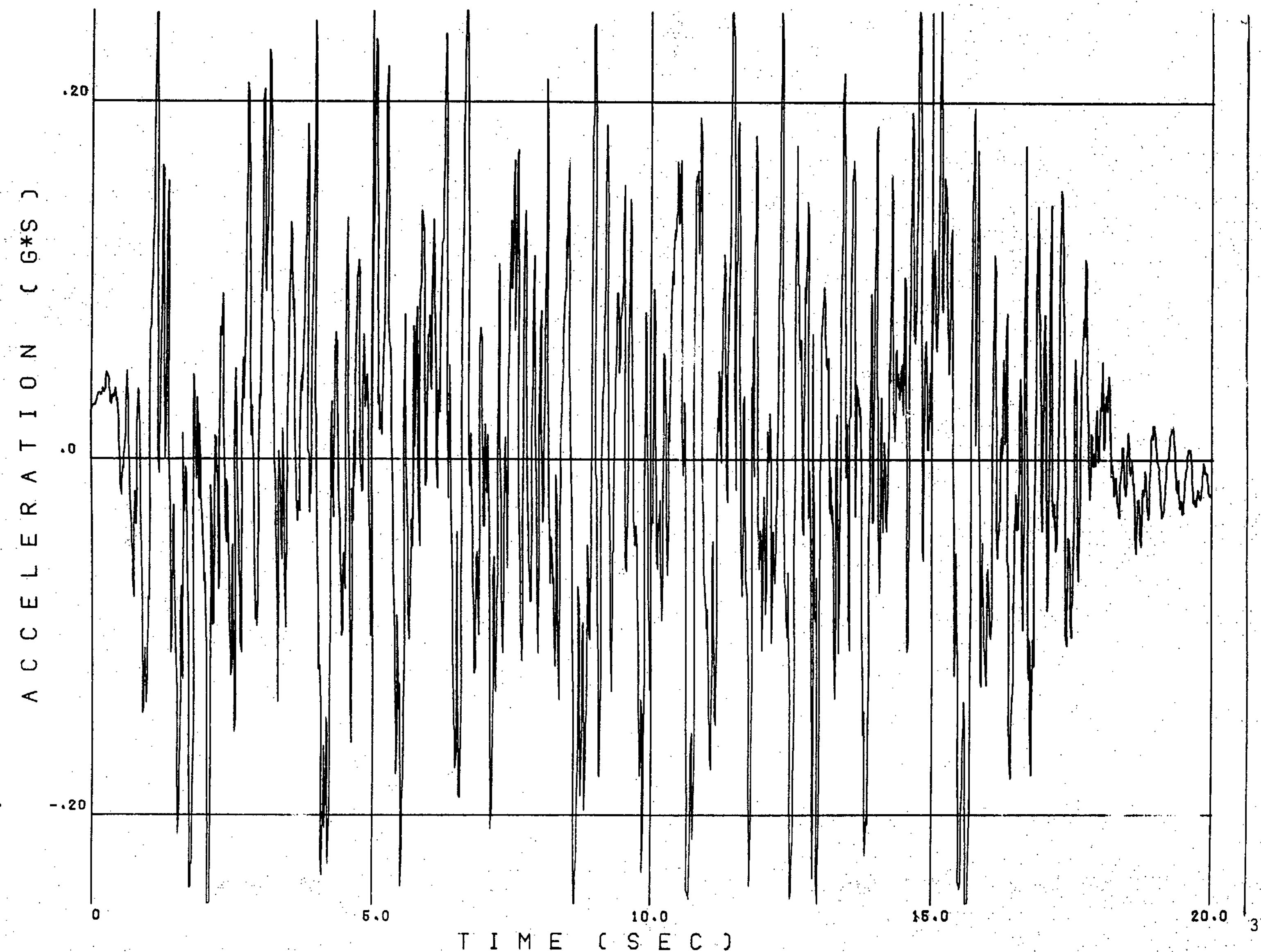


Figure Q130.62-5A Horizontal Motion B SSE Ground Acceleration Time-History

Q130.62-9

Amend. 31
Nov. 1976

Q130.62-10

Amend. 31
Nov. 1976

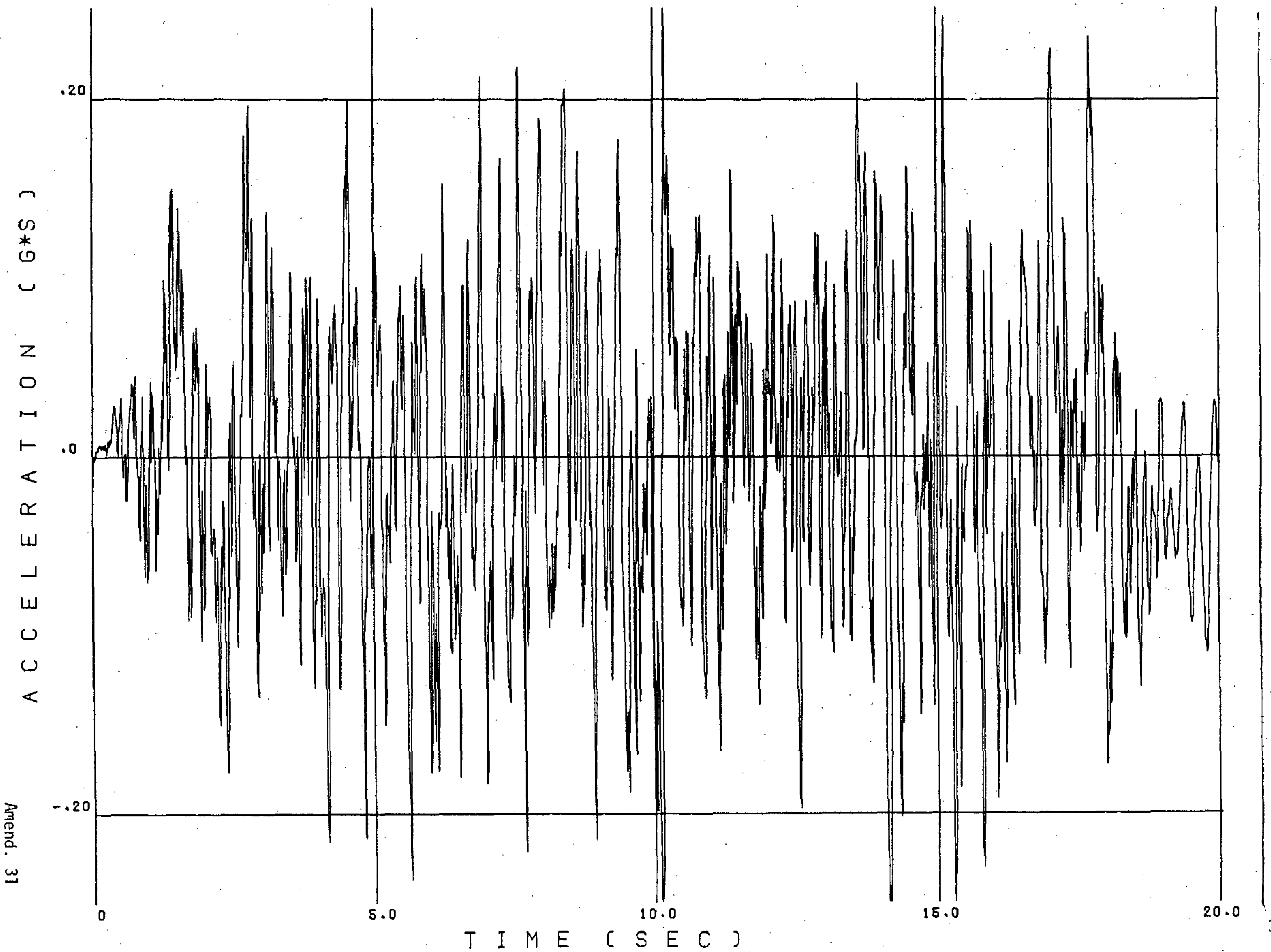


Figure Q130.62-6 Vertical SSE Ground Acceleration Time History

Question 130.63 (App. 3.7-A)

The damping values shown in Table 3.7A-3 are not consistent with Equations (1) and (2) on Page 3.7-A-C3. Provide justification for the damping values of the foundation materials shown in Table 3.7A-3.

Response:

Equations (1) and (2) in Section C.3 of Attachment C to Section 3.7A, and Table 3.7A-3 have been revised. Damping ratios (internal damping) for foundation materials have been provided in new Table 3.7A-3A.

Justification for the damping ratios for backfill material in Table 3.7A-3A was previously provided in the response to NRC question 324.7.

Question 130.64 (3.8.2)

It is stated that the lower portion of the steel cylindrical shell will be embedded in concrete, and concrete does not form a part of the pressure vessel. Indicate:

(a) how the effect of such concrete embedment is taken into consideration in the structural analysis of the steel containment, and how the leaktightness of the embedded portion can be assured.

(b) if reinforcing steel is provided in the concrete, and the quality of concrete to be used,

(c) with a sketch, the design features at the point where concrete terminates and the design consideration for water seepage at that point.

Response:

(a) The effect of the concrete embedment is taken into consideration in the structural analysis of the steel containment as indicated by the revised Section 3.8.2.4.

The leaktightness of the steel containment in the embedded portion, is assured by design and construction activities discussed in revised Section 3.8.2.4.

(b) The internal concrete structure as well as the external circumferential wall will be built of reinforced concrete. The concrete will be of a quality to insure that it will have a compressive strength of 4000 psi @ 28 days. Detailed information is provided in Section 3.8.3.6.

(c) With the containment vessel completely enclosed on the outside by the concrete confinement structure which is designed to be watertight below grade, there will be no potential danger of water seepage at the point where concrete terminates.

Question 130.65 (3.8.2.4)

In Section 3.8.2.4 it is indicated that the containment vessel will be designed for all the loads and interface conditions specified in Westinghouse Specification No. 953014 Rev. 0, June 1974 and ECP No. W-0005, R-2. Since the staff is not aware of such specification, provide in the PSAR the pertinent information with respect to the loads, interface conditions and other structural information contained therein.

Response:

The loads and interface conditions specified for the containment vessel are specified in PSAR Sections 3.8.2.3, 3.8.2.4 and 3.8.2.5. Additional information has been provided to address buckling and penetration loadings. Reference to the Equipment Specification has been removed.

Question 130.66 (3.8.2.5)

The load combinations to be used in the design of the CRBRP steel containment are indicated in Table 3.8-1 which states that the allowable stress limits should be as defined in Subsection NE-3000 of the ASME-III Code. In view of the fact that in Sub-section NE-3000 of the ASME-III Code, stress limits for each load combination are not clearly specified, and are to be established through your interpretation of the code, provide in the PSAR the stress limits together with the buckling criteria to be used for the load.

Response:

The buckling criteria are provided in Appendix 3.8A which has been added to Section 3.8 of the PSAR. The stress criteria taken from the ASME Code have been provided in Table 3.8-3 as part of response 130.65.

Question 130.67 (3.8.3.1)

The staff's concerns expressed in item 130.37 on cell liners have not been resolved. Provide responses to these concerns in the PSAR.

Response:

Your August 17, 1976 request for additional information, which included Question 130.67, was based upon your evaluation of information through PSAR Amendment 23. A response to Question 130.37 was provided in PSAR Amendment 24.

Question 130.68 (3.8.3.1)

In Section 3.8.3.1.1 it is indicated that a concrete corbel integral with the cavity wall is provided to support the reactor vessel. Provide a discussion on the design of the corbel, including the criteria and method of analysis used, specifically with regard to the shear design of the corbel. Indicate the temperature to which the corbel concrete will be subjected under normal operation of the plant.

Response:

The response to Question 130.68 is contained in revised PSAR Section 3.8.3.1.1.

Question 130.69 (3.8.3.3.9)

In section 3.8.3.3.9 it is indicated that for hot spots the temperature of structural concrete will not exceed the recommendation of ACI-ASME (ASME III, Division 2) Code. Provide a list indicating the condition under which the hot spots occur and the corresponding concrete temperature allowed.

Response:

Section 3.8.3.3.9 has been revised to provide the required information.

Question 130.70 (RSP) (3.8.3.3)

The statements contained in the bottom portion on Page 3.8-17 and its continuation on top of Page 3.8-18 are not acceptable. It is staff's position that no load factors shall be less than one.

Response:

The referenced statements in Section 3.8.3.3.10.2 concerning the load factors used in loading combinations have been revised to comply with the staff position.

Question 130.71 (3.8.3.4)

In Section 3.8.3.4.1 it is indicated that the structural analysis for each cell will be performed by considering one-foot wide vertical and horizontal strips. Provide an example to show how the analysis is actually carried out for dead load, live load, thermal load and seismic load.

Response:

In response to this question, Section 3.8.3.4.1 has been revised to provide additional information on structural analysis of cells.

Question 130.72 (3.8.3.4)

In Section 3.8.3.4.3 it is stated that redistribution of stresses in the vicinity of openings will be taken into account in the design of slabs and walls. Provide an explanation of the term "redistribution of stresses", since it is common knowledge that holes and openings in a structural element will give rise to stress concentration.

Response:

The response to this question is contained in revised PSAR Section 3.8.3.4.3.

Question 130.73 (3.8.3, 3.8.4)

In Section 3.8.3.2 and 3.8.4.2, Regulatory Guide 1.10 is not mentioned. Indicate the type of mechanical splices which will be used in the construction of concrete structures.

Response:

During the early stage of the preliminary design of Seismic Category I concrete structures, reinforcing bars larger than No. 11 were not anticipated. Therefore, lapped splices, wherever necessary, will be provided to meet the technical requirements of ACI Code-318.

Following some recent changes in structural arrangement and design loads, it is possible that No. 14 or No. 18 reinforcing bars may be used in certain concrete structures. If No. 14 or No. 18 reinforcing bars are used, mechanical splices will be specified in design and construction documents to meet the requirements of Regulatory Guide 1.10 - Mechanical (Cadwell) Splices in Reinforcing Bars of Category I Concrete Structures. 44

These requirements have been included in revised PSAR Sections 3.8.3 and 3.8.4. 44

Question 130.74 (3.8.4)

It is understood that a concrete confinement building will enclose the steel containment. Therefore it is required that a complete description of design and construction of the concrete confinement building should be provided in this section. The information requested should include all that required for seismic Category I structures. Furthermore, other sections of the PSAR should be modified in order to comply with such a change in containment concept.

Response:

The modifications and additions associated with the confinement structure are included in revised PSAR Section 3.8.4.

Question 130.75 (3.8.4)

In Section 3.8.4 only the design of the diesel fuel storage tank foundation is discussed. Provide:

(a) the material that will be used in the construction of the tanks,
(b) the codes to which the tanks are designed and constructed, and
(c) the loads, loading combinations and criteria considered in the design.

Response:

a) The material used in the construction of the diesel fuel storage tanks is included in revised Section 9.14.1.2.

b & c) The codes to which the tanks are designed and constructed are included in Section 9.14.1.1. The load, loading combinations, and criteria considered for these codes are included in Section 3.9.2.

Amend. 28
Oct. 1976

Question 130.76 (3.8.5.1)

In Section 3.8.5.1.1 it is stated that RCB, RSB, CB, DGB and SGB will be built on a combined reinforced concrete mat. Provide a description on the design to such a mat for dead load, live load, thermal load, and seismic load.

Response:

Section 3.8.5.4.1 (Combined Mat) has been revised to indicate the current design and analysis procedures.

Question 130.77 (3.8.5.1)

For the steel liner on the foundation mat, describe the method of welding the steel plates to steel members embedded in concrete. If concrete is to be placed after the welding operation, indicate the placing methods and the procedure followed to avoid voids forming between stud and concrete.

Response:

The steel liner on the foundation mat will be constructed following the requirements of ASME Boiler and Pressure Vessel Code Section III, Division 2, 1975 edition. Seams will be welded using Category F welds, in accordance with article CC-3841(F), Figure CC-3840-4(a).

The sequence of embedments installation is as follows:

Concrete will be poured up to the bottom elevation of the structural steel that supports the liner. The steel members are grouted for leveling and are anchored into the foundation mat to maintain alignment. With the supports in place and leveled, concrete will be poured up to the top level of the steel.

The steel liner will then be welded on top of the steel members as described above.

Question 130.78 (RSP) (3.8.5.3)

In Section 3.8.5.3.4, the minimum factors of safety acceptable to the staff against floatation for load combination (g) should be 1.10 not 1.0.

Response:

Section 3.8.5.3.4 has been revised to indicate the correct factor of safety against floatation due to design basis flood.

Question 130.79

In Section II.B, Design Requirements, the requirements for various components of the liner system are listed. However, the criteria used in the design to satisfy these requirements are not all identifiable in the report. Therefore indicate for each design requirement the corresponding design criteria (use of a table indicating the sections in which the related criteria are discussed is acceptable).

Response:

The criteria used in the design to satisfy the design requirements have been incorporated in the revised paragraph 3.1 of Appendix B of PSAR Section 3.8.

Question 130.80

In Section II.C, Load Categories and Load Combinations, it is indicated that load combinations are established on the same basis as in ASME Section III, Division 2 Code, specifically, the stress and strains as indicated in Table II-1 for load categories other than the extremely unlikely fault loads. The ASME Section III Division 2 Code is for concrete containments of light water reactors (LWR's). The cell liners used in CRBRP are dissimilar in many respects from the steel liners in concrete for LWR's, specifically in geometrical configuration, construction (use of insulating concrete and air gap in cell liners), pressure and temperature to be subjected to, and leakage requirements. Provide an assessment of the adequacy of using Section III, Division 2 Code criteria for CRBRP cell liners, taking the dissimilarities into consideration.

Response:

The load combinations used for cell liners design are provided in new PSAR Appendix 3.8-B.

Although Division 2 applies to liners of LWR Concrete Containments and the configuration of the cell liners in CRBRP is different, in both cases the liners are not used as strength elements; the main structural elements are the reinforced concrete structures to which the liners are anchored.

The purpose of the cell liners is to maintain the inert atmosphere (Nitrogen) under normal operating conditions, to facilitate decontamination and decrease plant downtime following an accidental sodium spill, and to protect the structural integrity of the cell for the preservation of the capital investment. 37

Question 130.81

In Table II-1 for the category of extremely unlikely fault loads combination, it is indicated that the Von Mises effective strain shall not exceed 0.75 of ultimate strain (ϵu) of liner material. This value appears to be very high. Since the determination of the Von Mises effective strain is based on a series of assumptions and is, therefore, rather uncertain, justify the use of 0.75 ϵu.

Response:

The response to Question 130.81 is discussed in attachment D to PSAR Appendix 3.8-B.

The design limits in Table II-1 for the category of Load Combination D has been modified and more conservative values have been adopted and listed in PSAR Table 3.8-B-1. The criteria for these limits are discussed in Attachment D to PSAR Appendix 3.8-B.

Question 130.82

Whether the insulating concrete serves the only purpose of protecting the concrete structure or as a portion of concrete structure, it has to resist the stress or strain imposed on it. Indicate the strength of the insulating concrete, and the stress and strain criteria used under various loading combinations to which the insulating concrete may be subjected.

Response:

The insulating concrete provides a non-load bearing insulating barrier between the steel liner plate and the structural concrete. Hence, no strength capability criteria have been allocated to the insulating concrete under accident conditions. Accordingly, the insulating concrete will be designed only for construction loads in accordance with the applicable codes, standards, and specifications presented in PSAR Section 3.8.3.2 and the Cell Liner Design Criteria in PSAR Appendix 3.8-B. The insulating concrete will consist of perlite/sand lightweight concrete with a dry density of not less than 65 lbs/cu.ft and a compressive strength of not less than 900 psi at 28 days.

Question 130.83

Indicate how the liner anchor allowable force capacity, yield force capacity, ultimate force capacity and ultimate displacement capacity are determined for Nelson stud anchors. State if the strength of the insulating concrete in which a portion of the stud anchor is embedded affects the force or the displacement capacities of such anchors.

Response:

59 The response to this question is included in revised PSAR Section 3A.8.3.3.

Question 130.84

The two concepts of providing insulating concrete barriers discussed in Section III.A.4 need further clarification, noting that whether the two components of a structural member act integrally or not depends on the manner in which they are put together. For instance, two concrete members which are poured separately may act integrally if sufficient shear keys are provided to transform the horizontal shear.

Response:

The response to this question is contained in the response to Question 130.85.

Question 130.85

The report does not state clearly whether the insulating concrete will act integrally with the wall concrete or not. If it acts integrally, provide the shear transfer detail. If not, indicate how the insulating panel and the steel liner will behave.

Response:

The insulating concrete barrier will not act integrally with the concrete wall. The 4" thick non-load bearing insulating concrete will be prefabricated at the site by the constructor by casting onto the prefabricated liner plates, including anchors and the 1/4" air gap (ethafoam). The prefabricated panels will then be cured and erected as a unit. A bond breaker will be provided at the interface joint between the structural and insulating concrete to preclude the transfer of shear onto the insulating concrete.

The steel liner plate and the insulating concrete panel will not act integrally due to the presence of the air gap. The insulating concrete will provide a temporary lateral restraint to the liner anchors during the early stages of heat exposure as a result of a sodium spill. In the fixed corner liner concept, this initial restraint has been shown to have no significant effect on the overall liner plate or anchor strains prior to the degradation of the insulating concrete in the vicinity of the studs.

Question 130.86

Explain the method by which the air gap of 1/4" will be provided and maintained during the construction and during the life of the plant. In case this air gap is closed or obstructed, indicate what will be the effects on the liner and liner anchor.

Response:

The response to this question is provided in PSAR Section 3A.8.2.

Question 130.87

The first paragraph on top of page III-4 indicates that the integrity of the liner system is maintained even in the event of deterioration of the concrete under accident conditions. Indicate the proportion of concrete that can deteriorate under accident conditions, without affecting the integrity of the liner system.

Response:

59 As stated in the new PSAR Section 3A.8.3.3, the integrity of the liner system is maintained even in the event of the deterioration of the insu-
59 lating concrete under accident conditions. Section 3A.8.3.3 of the PSAR discusses the effect on the structural concrete of a sodium spill in a lined cell and the use of the insulating concrete layer. Preliminary analysis of the integrity of the cell liner system have also been in-
59 cluded in Section 3A.8.3.3 of the PSAR which indicates that the integrity of the liner system is unaffected under accident conditions.

Question 130.88

In Figure III-2, the space between the floor slab and the floor liner plate is filled with aggregate. If the floor liner plate is so designed that the gravel will support any load applied to the floor, indicate the method of construction used to assure an adequate support for the liner by gravel during construction and in service, and the criteria of acceptance for the layer of gravel. Also indicate how the floor liner plate is analyzed.

Response:

The response to this question is included in the revised PSAR Section 3A.8.2 and paragraph 3.1.1.6 of section 3.8-B. The method of analysis of the wall and floor liner system is included in the revised PSAR
59 Section 3A.8.3.3.

Question 130.89

In case of a primary sodium spill which most likely will cover the floor liner, discuss, in detail, the effect of a sodium leak through the floor liner, on the layer of gravel, on concrete beneath and on the floor liner itself.

Response:

A detailed scenario that could lead to a CRBRP cell liner failure and the effects of such a failure is provided. Many other failure modes were considered but were found to be incredible. For example, liner buckling was considered. It was determined, however, that buckling would lead to a less severe test of liner integrity. This is because of the much greater residual elasticity in a buckled panel than in a panel which is fully restrained. Penetrations were considered as a possible failure point due to their complex geometry. However, penetrations will not be covered by a sodium spill and spray on a penetration would result in little or no sodium leakage behind the liner. The effect of the transient through the liner plate is discussed, as it is possible to obtain a tensile stress during this event. However, the net membrane stress in the panel is always compressive because of the fact that the average temperature of the liner is increasing during the thermal transient.

The effect of the failed liner on cell pressures and temperatures is discussed in the response to NRC Question 001.581.

FAILURE MODE ASSESSMENT FOR CRBRP CELL LINER DESIGN

I. DEVELOPMENT OF SCENARIO

From the outset, it should be stated that cell liner failure is not expected. The high level of design analysis and quality assurance employed in the design of the liner system will preclude such occurrences. The spill sizes considered for the liner loading are also extremely improbable and may actually be termed incredible. Finally, even in developing the crack scenario, many pessimistic assumptions are required to establish conditions that would result in a liner failure. These will be recounted in the development of the scenario.

The only significant loading on the liner, i.e.,a loading which could result in a failure, is caused by the differential thermal expansion of the liner relative to the cell if a sodium spill occurs. Seismic loadings, for example, do not cause appreciable loads on the liner or its anchors. The liner is designed to serve as a leak-tight containment for a sodium spill. As such, it is welded to embedments which will receive the thermal expansion loads. The liner itself is comprised of large panels welded togehter with full penetration welds at the joints. A backing strip is used to ensure a full penetration weld and a joint that is actually stronger than the plate itself. Nevertheless, because of the human factor involved in the welding and inspection, a weld will be taken as the most likely point of a failure initiation.

In the very improbable event of a sodium spill, the liner will be suddenly heated by the sensible heat of the spilled sodium. It will be conservatively assumed that the sodium spill volume is large such as that resulting from a MEFS (i.e., Moderate Energy Fluid System) leak. The liner system and spilled sodium will then come to an "equilibrium" temperature which is less than the initial temperature of the spilled sodium. The equilibrium temperature is actually a quasi-steady state point which represents the maximum liner temperature during the thermal transient. This temperature is determined from a heat balance which considers the heat capacity of the liner and equipment wetted by the sodium and the heat capacity of the spilled sodium. The CACECO computer program is generally used to perform these calculations. A conservative heat balance which considers only the heat capacity of the liner is used for scoping calculations.

As stated previously, the liner is designed to withstand the thermal expansion loads caused by heating the liner with respect to the concrete walls. The square corner design backed by I-beam embedments insures that the liner will be in a state of compression at all points. This is illustrated in Figures 1 and 2 which show a floor to wall joint and a wall to wall joint as well as the structural embedments. The stresses in the liner can be characterized as bi-axial compression in the plane of the plate and essentially zero stress out-of-plane. It is conservative to assume that the liner is fully restrained in the in-plane direction.

The effect of the high compressive loading in the plane of the liner will be to close any pre-existing cracks. During this phase of the spill scenario, the spray and/or pool fire will be occurring. The highest cell pressures and atmospheric temperatures will occur at this time. These events do not tend to decrease the pool temperature and the high in-plane compressive loading will tend to seal the liner against leaks.

The pool of sodium will eventually begin to cool down due to the transfer of heat into the cell structure. Water vapor behind the liner liberated by this heating will be vented through the behind-the-liner venting ducts. The structural concrete will tend to lose water from the surface layer; additional water vapor will be driven from the aggregate on the floor and insulating concrete in the wall liner.

As the sodium pool and liner begin to cool down, the high in-plane compressive stress in the liner will be reduced. Further cool down will result in a tensile stress in the liner. When the liner cools down to the point that the in-plane stresses are at the tensile yield point, then a tear in the liner at a weld joint is assumed.

The above approach results in a crack which extends the entire length of the cell and is opened by thermal contraction of the liner. In reality, the anchors will not provide a rigid boundary. In fact, the elasticity of the structural concrete and high local stresses in the concrete will probably result in anchor motion and should allow the liner to contract without ever reaching its tensile yield point. Buckling of the liner, if present, would also reduce the average membrane tensile stresses during cool down and thereby reduce the tendency to crack. It is noted also that even in the event of liner buckling, the membrane stresses are always compressive. The crack is assumed to occur in the floor of the cell for conservatism. It is assumed that the thermal shock stresses caused by the initially high thermal gradient through the liner do not have a pronounced effect upon the gross behavior of the liner. This is verified in Reference (1), in which an elastic-plastic large deformation finite element thermal transient analysis was performed for a similar liner design. It was shown that the through-thickness gradient did not appreciably affect the gross behavior of the liner plate, nor did it result in large strains that would lead to a premature failure at a local point in the liner.

Finally, the effect of local spray or stream impingement on the floor of the liner was considered. It was determined that while the local area would be more highly stressed than surrounding areas, the in-plane compressive loading would predominate and the spreading sodium pool would very quickly produce the assumed conditions. No areas were found that would be subjected to tensile loading during the spill.

II. EFFECTS OF CRACKED LINER ON SODIUM SPILLS

Figure 3 illustrates the location of the assumed crack at (A) the wall to floor junction and (B) an intermediate joint in the floor of the cell. In both cases, the sodium pool is assumed to be deep enough to cover the entire floor and stand a significant height up the wall. If the sodium pool were very shallow (i.e., a small volume spill compared to the cell floor area), the equilibrium temperature of the liner-sodium pool would be much less than the spill temperature. Thus it is assumed that a large sodium spill is of the most interest and the pool will completely cover the floor to a significant depth. This will result in complete immersion of the cracked portion of the liner in the sodium.

Reference (1) contains an analytical evaluation of a flawed liner covered with sodium. In this work a small circular hole was assumed to exist in the liner. An air gap was assumed between the liner and the structural concrete. Both parameters were varied to determine the sensitivity. The results of this analysis indicated that the hydrogen and other gases which are liberated from the concrete due to the sodium concrete/sodium-water reactions tend to limit the ingress of sodium into the gap area. The differential pressure behind the liner was found to be very low, in no case exceeding 0.5 psi. The temperature of the sodium in the gap never exceeded the pool temperature even though an exothermic sodium/concrete reaction was assumed.

The results of this study have the following implications to the above assumed crack. First, while the liner is vented, one could postulate a situation in which the sodium reaction products caused some of the venting paths to become partially clogged. The aggregate, for example, is approximately 40% voids. If significant reaction products are formed then they could serve to reduce the venting efficiency. The results of this study indicate that even in the event of a complete loss of the venting path, no significant pressure buildup is expected behind the liner. (The liners are designed to withstand a 5 psi pressure differential). Secondly, the loss of venting may actually help to limit the sodium from reaching the concrete and therefore limit hydrogen production.

Reference (2) is probably the most important source of information concerning the effect of a flawed liner subjected to a sodium spill. Two tests (S9 and S10) were performed in which 11 lbs of sodium at approximately 400°F were spilled into an inerted atmosphere reservoir. One wall of the reservoir was a vertical 10-inch diameter liner plate separating the sodium pool from a second vessel containing high density (magnetite) concrete. The liner touched the surface of the concrete in one test; there was a 0.25-inch gas-filled gap in the other. In each test the liner was intentionally defected by drilling a hole and a horizontal split both below and above the sodium liquid level. Stainless steel type 304 liner plate was used in one test; carbon steel ASTM A-36 in the other.

The sodium pool was heated rapidly to the specified maximum temperature, held at that temperature for a specified period of time, then allowed to cool. In test S9 the temperature was held at 1380°F ± 20°F for 18.7 hr. In test S10 the temperature was 1600°F ± 20°F for 2.6 hr. After the test articles had cooled to ambient temperatures, they were destructively disassembled for examination and sampled for chemical analysis. The results of the test support the following observations and conclusions:

- The defected liners provided significant protection to the concrete against direct chemical attack by the sodium. The concrete did not crack nor was the concrete deeply penetrated. The depths of sodium penetration into concrete were 5/8 inch in test S9 and 1.25 inches in test S10.

- Most of the water content of the concrete was driven out of the concrete through the liner defects, the only available points of egress.

- The water from the concrete reacted with sodium in the vapor space and on the vessel interior surfaces. Approximately 75% of the released water formed hydrogen on a mole per mole basis. The remaining 25% of the water was swept out of the sodium reservoir before having an opportunity to react.

This test indicates that the presence of a liner, though failed, provides considerable protection to the concrete. It is noted that the test temperatures were much higher than temperatures at which sodium/concrete reactions would occur for the CRBRP concrete. The maximum temperatures at which sodium is expected to contact concrete in the CRBRP design is approximately 800°F. (The effect of lower temperatures on sodium/concrete reactions is discussed later). A very significant observation that can be drawn from this test is that there was little or no corrosive attack tending to enlarge the flaws that were completely covered by the sodium pool whereas the liner above the pool was severely corroded. In fact, one of the liner defects plugged during the test. Another important observation is that the water released from the concrete did not tend to combine with sodium in the pool to the extent that was observed in tests in which no liner was present. This effect should be even more pronounced in the CRBRP liner design because of the large sodium pool area compared to the failed area.

At temperatures below 1040°F, there is essentially no sodium/concrete reaction for limestone concrete (Ref. 3). There can be, however, an exothermic reaction due to sodium/water reaction when free water is released by heating at the surface of the concrete. Since the liner is assumed to remain leaktight until the sodium pool begins to cool down, a large proportion of the free water in the structural concrete will be vented from behind the liner before sodium contacts the structural concrete. Figure 4 indicates the water release as a function of time for a sodium spill in a PHTS cell from a spill of 21660 gallons of sodium. The spill temperature was assumed to be 1015°F. Figure 5 shows the thermal response of the floor liner and the structural concrete for this spill case. It is noted that the maximum temperature of the liner for this spill was calculated to be 910°F. The temperature drop necessary to cause a tear would be 125°F evaluated using minimum yield strength properties. From Figure 5, it can be determined that the liner temperature drops by this amount in approximately 1.3 hours. From the detailed data used to construct Figure 4, it was determined that 60% of the total water that would be released (if the liner did not fail) has already been vented from behind the liner. Also, the surface of the structural concrete will contain very little remaining free water. As can be seen from Figure 6, the wall liners cool down much slower than the floor liner and therefore remain in compression longer. This supports the assumption that the floor liner would be more likely to fail first.

III. REFERENCES

(1) ORNL-TM-5145, "Evaluation of the Structural Integrity of LMFBR Equipment Cell Liners - Results of Preliminary Investigation," W. J. McAfee and W. K. Sartory, January, 1976.

(2) HEDL-TME-75-75, "Concrete Protection from Sodium Spills to Intentionally Defected Liners - Small Scale Tests S9 and S10," R. K. Hillard and W. D. Bochmer, July, 1975.

(3) WARD-D-0141, "The Interactions of Tennessee Limestone Aggregate Concrete with Liquid Sodium," S. A. Meacham, December, 1976.

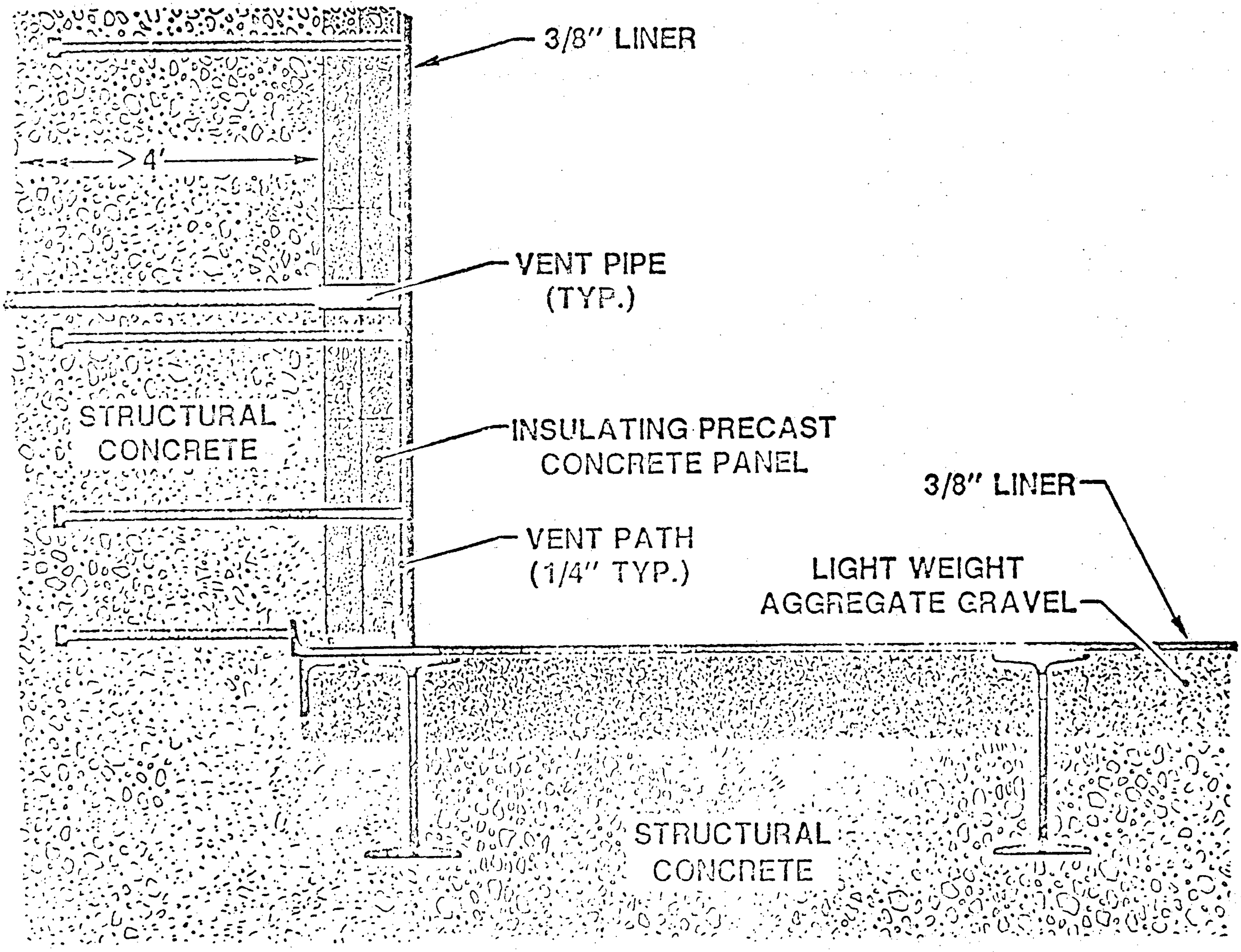


Figure 0130.89-1. Floor-Wall Design

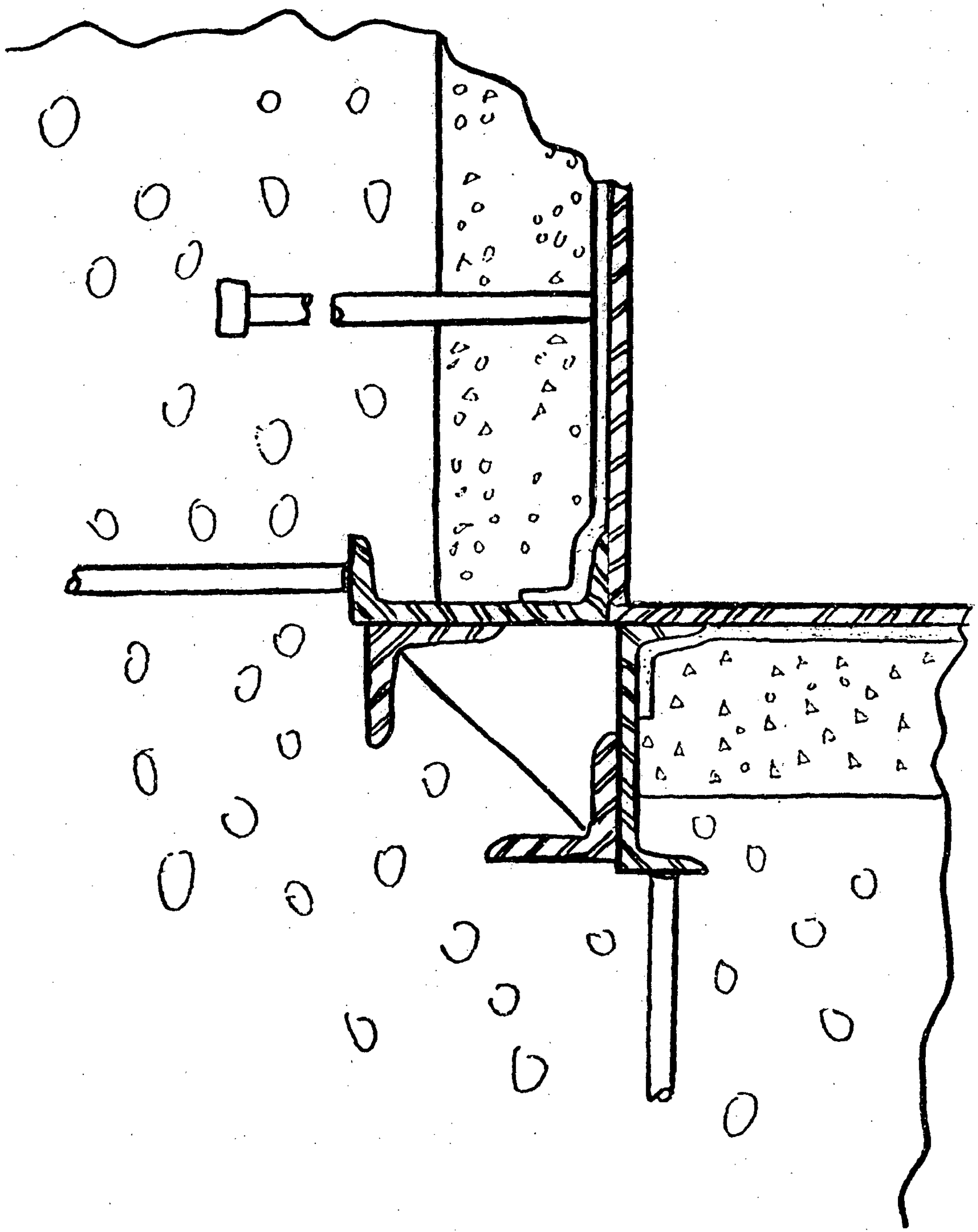
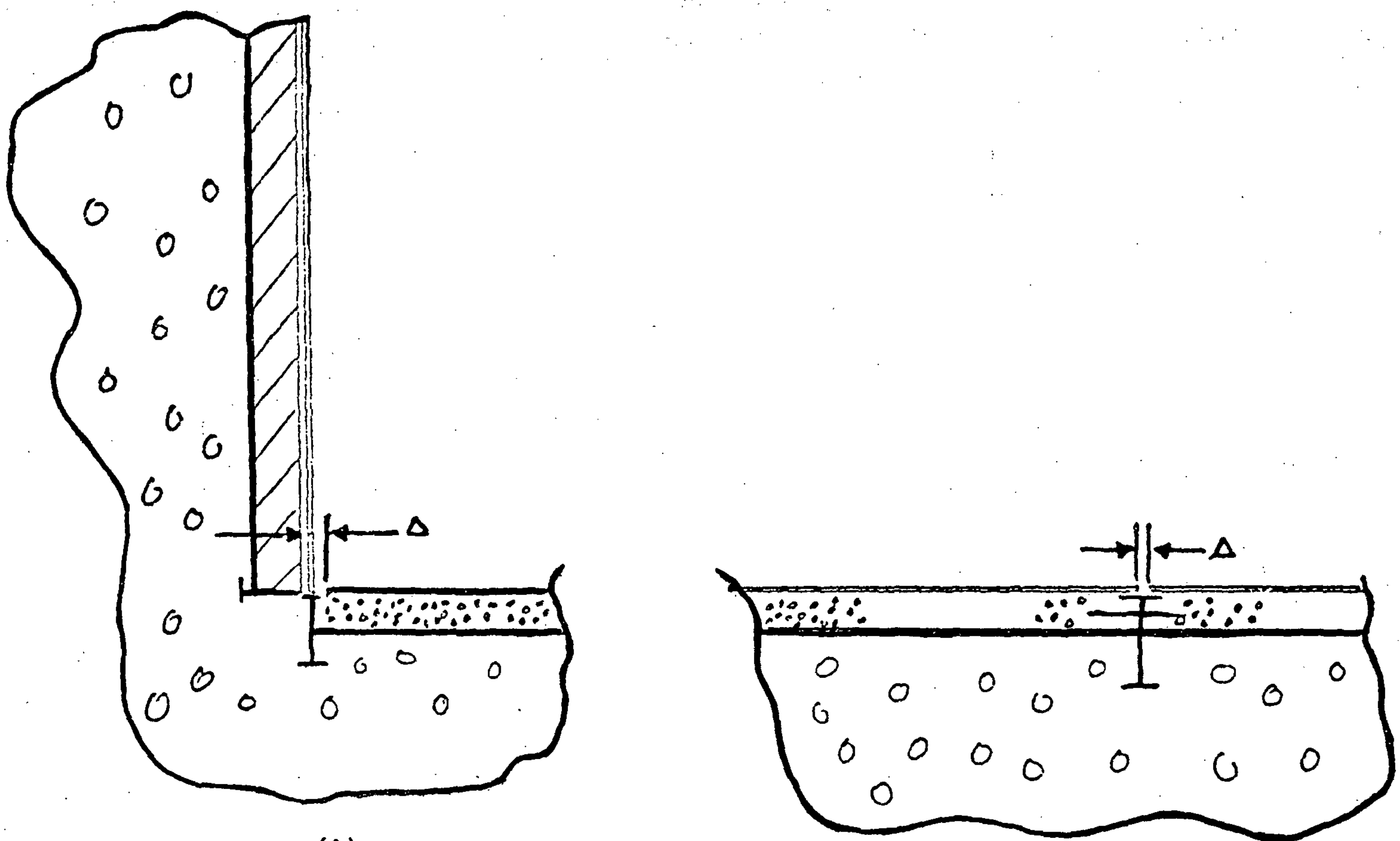


Figure Q130.89-2. Wall-Wall Joint Design



(A) (B)

Figure 130.89-3. Liner Failure at (A) Corner Joint and (B) Intermediate Joint

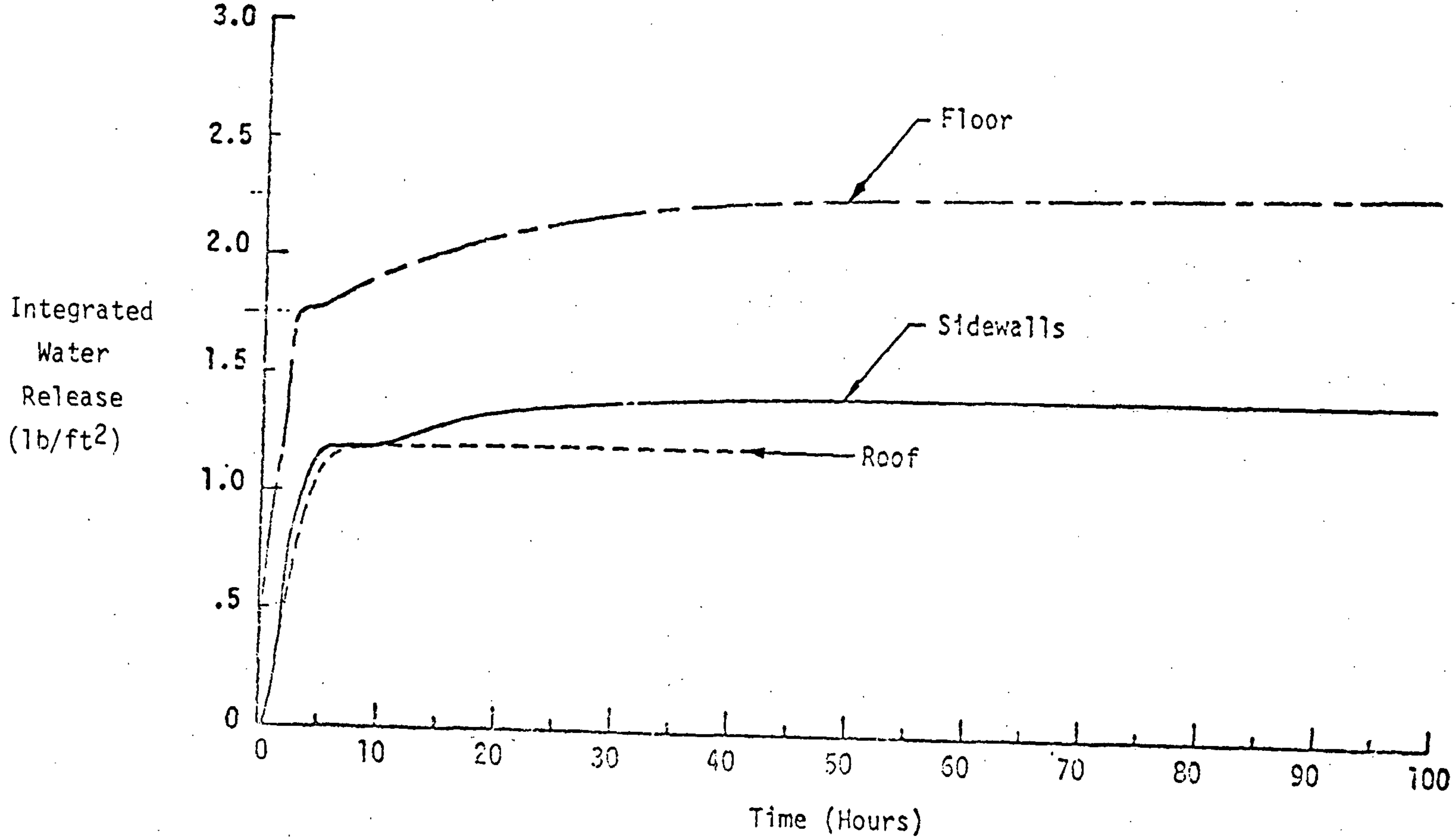


Figure Q130.89-4. PHTS Cell Integrated Water Release, 21660 Gallon Spill, Liner Intact, Updated Thermophysical Property Data, and Updated Cell Surface Areas

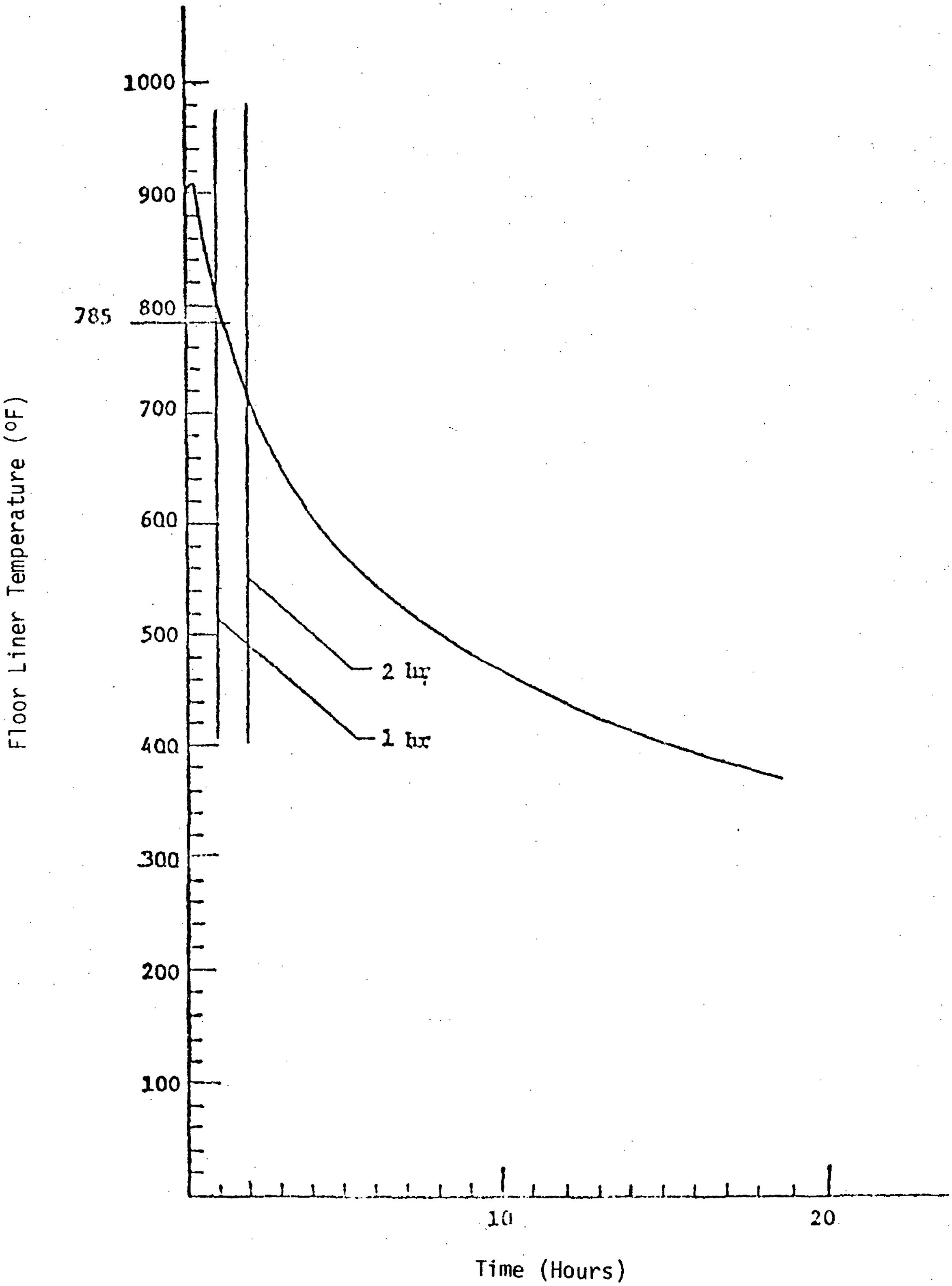


Figure Q130.89-5. PHTS Cell Floor Liner Temperature - MEFS Leak

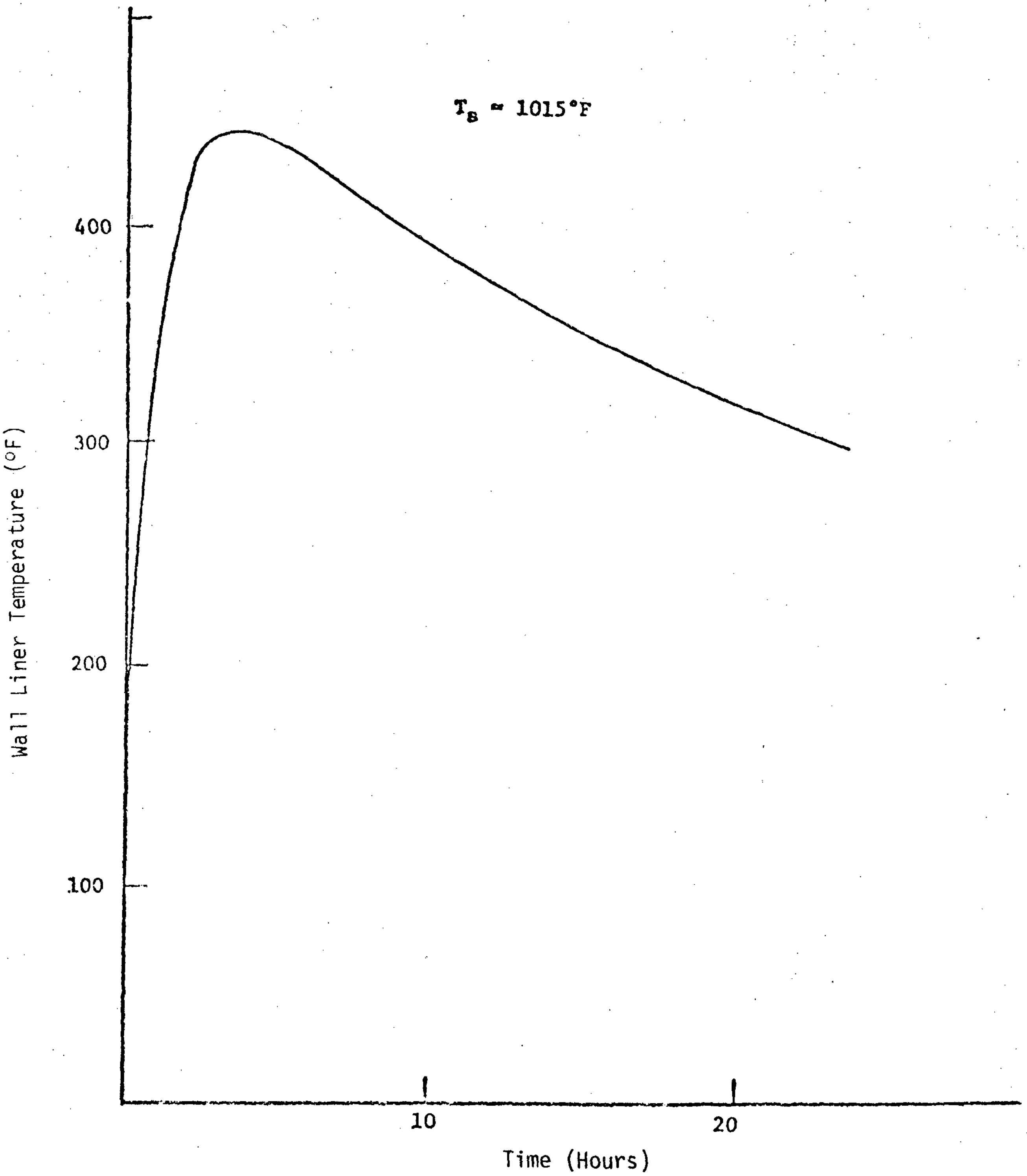


Figure Q130.89-6. PHTS Non-Wetted Wall Liner Temperature - MEFS Leak

Question 130.90

Indicate how the flow of steam or other gas through the gravel, and the pressure acting on the floor liner are determined.

Response:

To determine the pressure drop of the steam and gas flow through the gravel below the inert-cell floor liner, the similarity of a gas flow through a stationary pebble bed will be used. The latter process is developed in Reference Q130.90-1.

To determine the pressure drop of steam and/or gas flow through the gravel below the inert-cell floor liner, the following assumptions were made:

1. Similarity with gas flowing through a stationary packed pebble bed and our steam and/or other gas flow through the gravel to the liner vent pipes.

2. Flowing fluid is compressible and behaves as an ideal gas.

3. Gravel consists of regular 3/4 in. cubes based on actual gravel grading.

Applying Carman-Kozeny fixed bed correlation, the pressure drop may be calculated from equation 22.86 of Ref. Q130.90-1.

$$\frac{(-\Delta P) g_c}{L} \frac{Dp}{\rho \sigma_{sm}^2} \frac{\varepsilon^3}{1-\varepsilon} = 150 \frac{(1-\varepsilon)}{N_{RE}} + 1.75$$

where

$(-\Delta P)$ = Pressure drop through the packed bed, LBf/sq ft

L = Bed length, ft

N_{RE} = Average Reynolds No. based on superficial velocity, $\frac{D_p \sigma_{sm} \rho}{\mu}$

D_p = Particle diameter, $= 6\frac{Vp}{Ap}$, ft.

V_p = Volume of particle ft^3

A_p = Area of Particle ft^2

ρ = Fluid density lb/ft^3

Amend. 37
March 1977

σ_{sm} = Superficial velocity at a density averaged between inlet and outlet conditions, ft/sec

$$= \frac{\frac{G}{\rho_1} + \frac{G}{\rho_2}}{2}$$

$\rho_1 = \frac{PM}{RT_1}$ T_1 = Temperature inlet to gravel °R

T_2 = Temperature outlet of gravel °R

$\rho_2 = \frac{PM}{RT_2}$ P = Inlet pressure lb/ft^2

M = Mol. weight of flowing fluid

R = Universal gas const.

G = Mass flow rate, LBm/HR FT^2

g_c = Conversion Factor, (FT/sec^2)

μ = Absolute viscosity, LBM/HRsec

ε = Bed porosity, dimensionless

Since the porosity is a function of spaerity Ψ which can be calculated from Eq: B-26 page 535 of Ref. Q130.90-1

$$\Psi_v^1 = \frac{\pi\left(\frac{6Vp}{\pi}\right)^{2/3}}{Ap}$$

and using Fig. B-12 of Ref. Q130.90-1 a corresponding porosity can therefore be found. All the parameters are therefore known and the pressure drop can be calculated accordingly.

References:

Q130.90-1 Foust, A. S., "Principles of Unit Operations", 60-6454, Library of Congress, Third Edition, July, 1964, pp. 472-476

Question 130.91

Indicate the criteria which were used in adopting the corner detail design as shown in Figure 3. If the design has been changed describe the new one and state the rationale for the change.

Response:

The corner detail of Figures 1 and 3 of the June, 1976 Report has been
changed to a straight corner detail as a result of stress analyses performed. 37
The original curved corner detail resulted in excessive shear strains in the
wall panel stud anchors. By using the detail shown in PSAR Figure 3A.8-4, 37
the thermal expansion of the plate at the corner has been limited which
results in smaller displacements and lower strains and stresses at the
anchors. PSAR Figure 3A.8-6 shows a typical cell liner isometric-corner 37
detail.

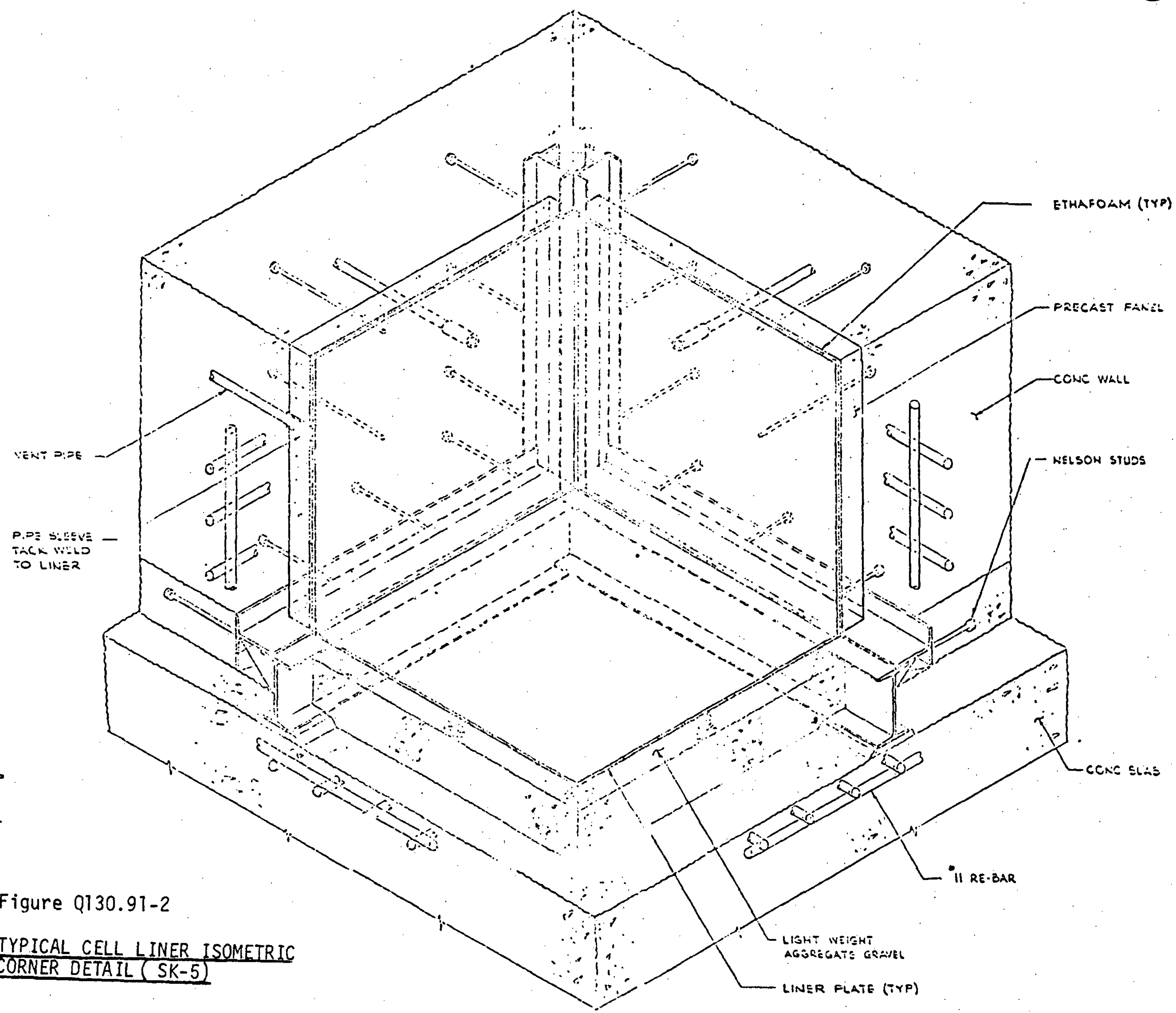


Figure Q130.91-2

TYPICAL CELL LINER ISOMETRIC CORNER DETAIL (SK-5)

Amend. 34
Feb. 1977

Question 130.92

For the floor liner system, the floor liner is welded to a grillage of steel beams. The welds which connect the floor liner to the steel beams are subjected to longitudinal and transverse loads. Describe (1) the criteria used for the design and fabrication of these welds; (2) the method of analysis of these welds; and (3) the acceptance criteria and the in-service inspection requirements for these welds. Indicate if the welds will be X-rayed.

Response:

(1) The criteria used for the design and fabrication of floor liner to steel beam welds

Criteria for the fabrication of the floor liner to steel beam welds are discussed in paragraph 4.0 of Attachment B of PSAR Appendix 3.8-B.

Criteria for the design of the floor liner to steel beam welds are discussed in paragraph 3.8.6 of PSAR Appendix 3.8-B.

(2) The method of analysis of the floor liner to steel beam welds

Finite element, elastic-plastic analysis by computer will be provided to verify the design of floor liner welds.

(3) The acceptance criteria and in-service inspection requirements for the floor liner to steel beam welds

Acceptance criteria for the welding of the floor liner to the steel beams will be in accordance with the ASME codes and is discussed in Section 5.0 of Attachment B of Appendix B to PSAR Section 3.8.

Requirements for in-service inspection of cell liners are paragraph 6.0 of Attachment E of PSAR Appendix 3.8-B. Radiography is a code requirement only if the liner welds are accessible. Radiography, however, is not feasible due to the method of construction and, in addition, the welds will not be accessible after construction. The code allows alternate methods of examination under these conditions which are detailed in the PSAR acceptance criteria described in paragraph 5 of Attachment B of PSAR Appendix 3.8-B.

Question 130.93

In the 3rd paragraph on page IV-10, it is indicated that analyses have been made for bi-planar and tri-planar corners. Indicate the mathematical models and assumptions made for such analyses.

Response:

The response to Question 130.91 provided a description of the straight corner detail which has been adopted to replace the curved corner detail presented in the June 1976 report.

An elasto-plastic finite element analysis using the computer program ANSYS was used in the preliminary analysis to find the stresses and strains in the bi-planar straight corner. The mathematical model is shown in PSAR Figure 3A.8-3. Since the tri-planar corner is restrained in the same fashion as the bi-planar corner, the analysis and results of the bi-planar corner and the tri-planar corner will be similar. Consequently, no tri-planar analysis will be done.

59 Section 3A.8.3.3 presents the mathematical model used and the assumptions made in the analysis.

Question 130.94

In the last paragraph on page IV-10, it is indicated that as a result of the assumption of symmetry conditions, the analysis shows no load imposed on anchors. Indicate how the anchors are designed and how the spacing of 15 inches for the anchors is determined.

Response:

59 This information is discussed in Section 3A.8.3.3.

Question 130.95

In the final liner analysis, the following topics should be taken into consideration. Address how each of these points will be considered:

(a) Indicate the method used to evaluate membrane and bending strains/stresses beyond the yield point of the material.

(b) Evaluate the effect of geometric restraints which may limit the ductility of the material.

(c) Consider the possibility of thermal shock.

(d) Evaluate the influence of boundary and local conditions which are not symmetrical, such as failure of some anchors, effects of penetrations, transition zones between hot and cold portions of liners and at free edges, effect of local hot spots and effect of fixed equipment supports.

(e) Evaluate the effect of welding carbon steels to steels with different thermal expansion coefficients, such as stainless steel, etc., if any.

(f) Evaluate the influence of local fabrication imperfections.

(g) Evaluate the effect of pre-existing cracks and of cracks generated during the life of the plant; discuss possible propagation of these cracks in the liner.

(h) Describe acceptance and in-service surveillance tests of the liner. Indicate if the welding is to be x-rayed.

(i) Describe the effects on the liner of sliding or rolling equipment supports.

(j) Discuss the need for the liner to be electrically grounded.

(k) State whether liner and penetration subassemblies will be stress-relieved, or heat-treated. Explain whether pre-heating during welding will be done and in what measure its uniformity will be achieved.

(l) Indicate with precision by what method the corrosion allowance has been determined.

(m) Indicate the arrangement where two back-strips cross each other and how a gap at this point is avoided. This gap may generate stress-concentrations in the liner.

(n) Evaluate the effect of all possible local stress risers such as tack welds, plug welds, gaps, etc. (see page A-3).

(o) Indicate how the stresses or strains due to loads such as dead, live, seismic, etc., other than those resulting from primary sodium spill, are obtained and combined with those due to sodium spill.

(p) Indicate the effect on concrete of heat transmitted from the cell liner to the cell liner anchors.

Response:

(a) The liner analysis will be conducted with the finite element computer program ANSYS. Beyond the yield point of the material, an incremental technique is used in "ANSYS". The loading is applied in increments and at each loading level an elastic solution is done, with a correction applied to the next loading step to account for the plasticity occuring during this loading step. The von Mises yield surface is used, along with the Prandle-Reuss flow relations. Since the strains/stresses are claculated at the top, middle, and the bottom surfaces of the plate, the membrane and bending components can be easily separated from the total effects. The membrane strain/stress will be calculated as the average through the section; the bending strain/stress will be calculated as the total strain/stress minus the membrane.

(b) In the areas with geometric restraints or with stress/strain concentrations appropriate mathematical models will be constructed to calculate the effects on the liner; detailed models with regular fine mesh will be used as required.

(c) Those portions of the liner surface subject to thermal shock effects will be reinvestigated. A thermal analysis will be conducted to
59 calculate the temperature distributions through the liner as a function of time, until the thermal soak condition is reached, i.e., the liner temperature through its thickness is uniform and equal to the sodium temperature. Incremental elastic-plastic structural analysis will be conducted for the same time span as the thermal transient analysis.

59 (d) The influence of boundary and local conditions which are not symmetrical will be considered by using mathematical models simulating conditions such as effect of penetrations, transition zones between the heated and unheated portion of liners and at free edges, effect of hot spot and effect of fixed equipment supports.

Limited local failure of the stud anchors will be examined and its effect on the overall integrity of the liner will be investigated.

The problem of hot spots was considered in the preliminary liner analyses by heating a local area of the floor liner in the finite element model of a bi-planar liner corner. It is observed from the results of the analysis that this effect does not govern the design of the liner system.

59 (e) The welding of the austenitic stainless steel to the ferritic steel material, such as in the Fuel Handling liner will not pose a problem because of the different thermal expansion coefficients of the materials. The cell liner and hence the dissimiliar weld point
59 will be exposed to a normal operating temperature of 150°F maximum. The dissimiliar weld joint will not be subject, under normal operating temperature, to any thermal cycling. As a result, the weld joint between stainless steel and the carbon steel will not develop differential expansion strains. Failure is not expected to occur due to the absence of long-term cyclic temperature service.

(f) The influence of the following local imperfections will be considered
59 in the final analysis:

(1) <u>Initial bow in the liner plate</u>
In the mathematical model an initial bow will be assumed on a liner panel with adjacent panels assumed straight. This will induce unbalanced forces on the anchors and begin strains
59 on the bent panel.

(2) <u>Liner thicker than nominal due to rolling tolerances</u>
If the bowed panel is thinner than adjacent panels, the membrane force imposed on the buckled panel and the shear forces on the anchors will be larger than for uniform thickness. The CRBRP analysis accounts for variation in thickness by utilizing the actual maximum strength of the liner plate in determining
59 the thermally generated in plane load. This combined with the corrosion allowance assessment bracket the response.

(3) <u>Yield stress higher than normal</u>
Higher yield stress in the straight panels will have an effect similar to an increased thickness on the bowed panel. The analysis described in section 3A.8.3.3 have utilized the maximum actual yield strength of the liner plant and not the
59 code minimum values, in order to maximize liner strains.

(4) <u>Variation of anchor spacing</u>
59 Small variations in spacing as can be expected during construction are not considered to have appreciable effect on the anchor-liner system.

(5) <u>Local concrete crushing on the anchor zone</u>
59 The analysis considered two cases: (a) that the insulating concrete and insulating gravel on the floor provides full lateral support to the anchors; (b) that the insulating concrete
59 provides no lateral support to the anchors and are free to bend. These two cases are considered as envelopes to intermediate situations in which there is partial yield in the supporting material.

(g) The fabrication requirements for the liners includes an extensive quality assurance program which will detect pre-service cracks. The liner welds will be liquid penetrant inspected or magnetic particle inspected. In addition, all liner welds will be vacuum box tested. The welds are designed to provide a full strength joint. Because of the quality assurance procedures adopted, initial through-wall cracks will be precluded.

In order to propagate a pre-existing crack, a significant cyclic stress must be experienced by the liner. The thermal cycling of the liner due to normal temperature fluctuations within the cell, has been calculated and the cyclic stresses have been determined. Based upon the number of heat-up and cool-down events estimated for the plant, the fatigue usage factor for the liner is very small, and appears to be well below the endurance limit for the liner material.

The liner has been analyzed for sodium spill conditions which greatly exceed the design basis spill event. These analyses indicate that the liner will not develop cracks or tears.

(h) See Paragraph 3 of the response to Question 130.92.

(i) Refer to Section 3A.8.2 of the PSAR and, also, the response to Question 130.88.

(j) The cell liner system will be electrically grounded to ensure personnel safety in the event of electrical equipment ground faults within the cell cavities. The grounding of the steel liner is deemed necessary to preclude any shock hazard that might otherwise be present in cell cavities consisting of steel-lined floors and walls, and containing electrical equipment. Grounding shall be in accordance with Article 250-44 of the National Electrical Code (1975 Edition), and the method for grounding shall conform to the requirements of Article 250-51 of the National Electrical Code.

(k) Pre-heating during welding of liner and penetration assemblies will be in accordance with the applicable ASME Code, Section III, Division 2, sub-sub article CC-4551 requirements.

The nominal material thickness of the liner and penetration assemblies will be controlled to the requirements of ASME Code Section III, Division 2, CC-4552.2.7 so that mandatory postweld heat treatment is not required. 59

(l) Owing to the presence of inert atmosphere in the cells, corrosion of the carbon steel or stainless steel liners, under normal operating conditions, is entirely absent.

The back side of the carbon steel of stainless steel wall liner material is separted by a ¼-inch air gap followed by a 4-inch insulating precast concrete panel. The back side of the carbon steel or stainless steel floor liner is also separated by a 1/8-inch air gap and a 4-inch thick precast panel of insulating concrete followed by structural concrete. The surface of the insulating/structural concrete interface and the joints between adjacent panels will be sealed to minimize the migration of water toward the liner plate during construction and normal operating conditions. The amount of moisture that might be present in these materials will cause negligible corrosion of the
59 liner material. Accordingly, a 1/16 inch corrosion allowance has been used as referenced in PSAR Appendix 3.8-B.

(m) Continuous backup strips will be provided by welding abutting ends prior to covering with the liner plates.

(n) Tack welds will be used to secure alignment of the liner plates. Their stopping and starting ends will be properly prepared by grinding or other suitable means so that they may be satisfactorily incorporated into the final weld. Tack welds will be visually examined and any defective tack welds will be removed and repaired.

59 No plug welds are anticipated in the fabrication of cell liners.

(o) Since the liner stiffness is negligible compared to the stiffness of floors and walls on which the liner is supported, the liner will deflect with the supporting system. The displacements of supporting points of the liner due to loads such as dead, live, seismic, etc., will be determined considering the concrete floors and walls without liner. Using these displacements at the supporting points of the liner and the loads directly imposed on the liner, the complete liner analysis will conducted. The liner stiffness is such that under seismic conditions it will not vibrate indepentently of the supporting structures.

(p) A three-dimensional transient thermal analysis of the liner with stud anchors has been performed and reported in ORNL-TM-5145 (January, 1976). The analysis indicates that, despite rapid heat-up of the liner plate and direct thermal coupling of the liner and stud anchor, the anchor is ineffectual to transmit heat into the structural concrete. This is particularly true over the short term, when the liner is imposing its maximum loads on the stud anchors. Over the long term, the concrete temperature will rise. The analytical results indicate that heat conduction into the stud anchors will neither accelerate nor localize the process. Consequently, no degradation of concrete capabilities directly attributable to heat conduction into the liner anchors is expected to occur.

Question 130.96

In Section V.C., the planned testing program should also include:

(a) Testing of the insulation concrete in order to assess its behavior under various conditions including the direct contact with hot sodium, and

(b) Testing of the gravel under the floor liner, when in contact with hot sodium.

Response:

(a) The insulating concrete as used in the design is assumed to have no load carrying capacity. Analyses have been performed that show no deleterious effects to the studs or anchors will be caused by thermal effects on the insulating concrete. Accordingly, no credit is taken for the insulating concrete in the structural analysis of the cell liner system.

The physical and thermal properties of the insulating concrete are included in the planned testing program titled "Comprehensive Testing Program for Concrete at Elevated Temperatures."

(b) The gravel used under the floor liner is composed of magnesia (MgO). Tests have been run which indicate the MgO will not react with sodium at temperatures up to the boiling point of sodium. These tests are reported in Reference 130.96-1.

Reference:

130.96-1 - WARD-D-0097, "Sodium Compatability of Refractory Materials Considered for CRBRP Parallel Design Ex-Vessel Retainers", April, 1975.

Question 130.97

In order to investigate the behavior of concrete at elevated temperatures, the concrete should be tested as follows:

(a) It should be reinforced as in the actual structure.

(b) It should be tested for bi-axial and tri-axial stress/strain distribution.

(c) The influence of cracks in concrete on heating up of the reinforcing bars should be studied.

(d) The influence of cracks in concrete on liner strains should be evaluated.

(e) The interaction of concrete and embedded steel should be investigated.

Response:

Data deemed sufficient to investigate the behavior of concrete at elevated temperatures has been specified in the proposed testing program in PSAR Section 3A.7.4.

The responses to items (a) thru (e) are as follows:

(a) Discussed in PSAR Section 3A.8.4.1.

(b) Discussed in PSAR Section 3A.8.4.1.

(c) The reinforced structural concrete is protected from thermal degradation by the insulating concrete layer as described in Section 3A.8.3.5 of the PSAR. Accordingly the reinforcing bars will not be heated as a result of the cracking of the structural concrete.

(d) The influence of concrete degradation (cracks) on liner strains is discussed in PSAR Section 3A.8.3.5.

(e) The interaction of concrete and embedded steel is discussed in Section 3A.8.3.5. of the PSAR and will be considered in the cell liner analysis.

Question 130.98

For Sodium-Concrete Reaction Tests, the following should be included:

(a) Investigate the effects of cracks in concrete;

(b) Investigate the effects of sodium on reinforcing; and

(c) Establish the characteristics of the design crack in the liner.

Response:

Points (a) and (b) have been considered by the following. The phenomenon of concrete cracking has been addressed via other sodium-concrete test programs (HEDL-TME-74-36), as well as some small scale tests conducted at ARD. The results of the ARD test program indicate that exposure to 1600°F sodium will cause localized cracking of concrete, but than sodium penetration of the cracks does not cause sodium-concrete reactions. The HEDL tests, conducted on a slightly larger scale, showed significant concrete cracking only for the case where only part of the concrete surface was in contact with sodium; i.e., the case for a failed liner. In those instances where sodium completely covered the concrete surface, no cracking was observed. In any event, cracking appears to be a very localized phenomenon, in keeping with the very high thermal gradients in the concrete near the surface. Cracking appears to be restricted to the region of the sodium-concrete reaction in the HEDL tests which, by virtue of size and test temperature, appear to be more structurally representative than the ARD tests. The unreacted concrete appears to be structurally sound but will be subject to thermal degradation and cracking commensurate with its proximitity to the sodium concrete reaction front.

Carbon steel does not significantly react chemically with sodium. As was noted in the response to Question 130.95 (p), embedded steel is a very ineffectual heat path, so differential thermal expansion between reinforcement and concrete is not anticipated to be a problem.

There is no design basis crack for the cell liner. The cell liner material, welding and non-destructive testing requirements discussed in response to NRC Question 130.37 will insure no cracks exist. Furthermore, a failure modes and effects analysis for the cell liners will be submitted in response to Question 130.89.

Question 130.99 (130.37)

Under Item H on page Q130.37-3, it is stated that criteria for multi-axial state of stress will be applied at locations where axial, bending and shearing stresses are interacting. A clarification of this statement is required. Indicate if a beam which is under the action of axial, bending and shearing stresses is considered to be in a multi-axial state of stress.

Response:

A beam under the action of axial, bending and shearing stress is considered to be in a multi-axial state of stresses (Biaxial). Based on those stresses, a set of two principal stresses and a principal shear are calculated.

Question 130.100

In a letter to the NRC staff dated October 15, 1976, the Project states that the PHTS cells are designed to a pressure of 30 psig. Revise the PSAR to demonstrate conformance with this statement, including the following:

(a) Describe in detail the calculations relied upon to demonstrate the adequacy of the 30 psig over-pressurization. Include in the discussion a concise description of the chemical reactions considered (e.g., sodium-concrete, sodium-water, etc.) and, list and explain each assumption for satisfactory operation of other features (e.g., vents), describe such features.

(b) Indicate the maximum sizes of the PHTS cells, the cell wall thickness and describe the analyses of such cells stating whether the cell is considered as a unit with walls as two-way slabs, or the cell is represented by a unit width of rectangular rigid frame.

Response:

The response to part (a) is contained in the response to NRC Question 001.581.

The response to part (b) is contained in PSAR Sections 1.2 and 3.8.3.4.1.

Question 130.101

On Page II-10 under Item B in Section 2.3.1 of the TLTM report, it is stated that the pressure behind the reactor cavity liners should be limited to 5 psig. Provide the basis on which this limit is established.

Response:

The response to this question is provided in Section 2.2.5 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 130.102

In Section 2.3.3.1 it is indicated that in order to vent the steam and CO_2, the steel liner and concrete surface are separated by approximately 1/4 inch. Provide the assurance that this 1/4 inch space will meet its intended purpose, specifically under pressure and high temperature conditions. (You may wish to coordinate this response with Item 001.629).

Response:

The response to this question is provided in section 2.2.5 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Further analytical work will be performed based on the data from the development programs described in appendices A.5 and A.6 of CRBRP-3, Volume 2.

Question 130.103

On page II-22 it is stated that the RCB annulus is partitioned to provide a spiral flow path around the RCB. Provide the structural details showing the manner in which the partitions are supported. Also indicate how the pressure in the annulus and the extent of variation of the annulus pressure are determined. Relate this to potential structural design impacts.

Response:

Figure 130.103-1 provides the details showing the manner in which the partitions are supported. Annulus pressure is determined using probes as indicated on PSAR Figure 9.6-5 (Amendment 49). The variation of the pressure in the annulus during the operation of the annulus cooling system has been calculated on the basis of American Society of Heating Refrigeration and Air Conditioning Engineers (ASHRAE) method and the calculation results indicated that the differential pressure on the annulus partitioning is inconsequential to the structural design.

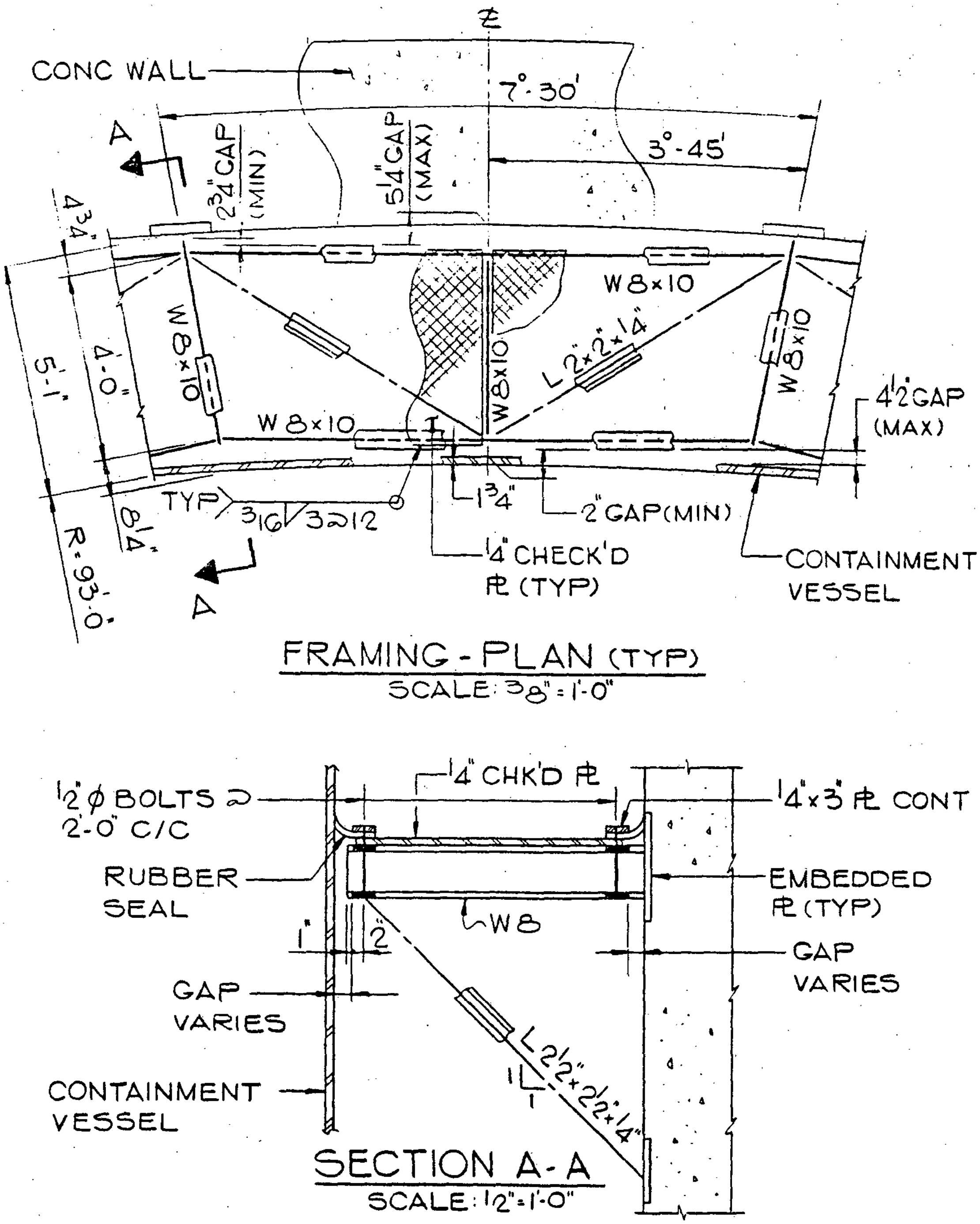


FIGURE 130.103-1

Framing Plan (TYPICAL)

Question 130.104

On Page III-2, in Items C(1) to C(4), various non-structural related phenomena resulting from sodium-concrete reaction are assumed. In the scenario postulated, the possibility of hydrogen or steam explosions (due to water collecting in local pockets and voids in the concrete and then interacting with core debris; refer to Item 001.648) are not included. A justification for not considering explosions should be provided.

Response:

The information requested will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Sections 2.1.2.5, 2.2.5 and 3.2.1.

Section 2.1.2.5 describes the reactor cavity and pipeway cell liners vent functional requirements, Section 2.2.5 describes the liner vent design and Section 3.2.1 indicates how the liner vent design prevents steam explosions.

Question 130.105

On Page D-4 it is stated that a substantial part of the lower reactor cavity wall would be melted or heated excessively and it would be expected to fail. Based on these statements the concrete support ledge above the reactor may be assumed to fall down. Indicate if the consequences of such a failure are considered consistent with the staff requirement for 24 hour containment system integrity. (You may wish to coordinate this response with Item 001.631.)

Response:

No total failure of the Reactor Cavity is expected before sodium boil-dry. A description of the response of the cavity wall to the TMBDB accident is given in Section 3.2.2.5.1.2 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 130.106

On Page F-2, it is indicated that a simplified mathematical model of infinite length is used to represent the reactor cavity cylinder. The cylinder is relatively short and its radial expansion is restrained by other adjacent structures, and furthermore, the temperature is not uniform throughout the height of the cylinder under the postulated accident condition (Figure F-6). In view of the above observations, provide the justifications for the mathematical model adopted.

Response:

The response to this question is provided in Section 3.2.2.5.1.2 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 130.107

In Item B on Page F-3, a maximum strain of 0.004 is indicated. However, according to the ACI 318-71 Code, the maximum usable compressive strain is 0.003 (see 10.2.3 of the code). Justify the use of a larger maximum concrete strain.

Response:

The response to this question is provided in Section C.3.2.3 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 130.108

On Page F-3, in Item D, it is stated that the liner is adequately anchored to non-degraded concrete. However, in the next paragraph it is indicated that the innermost 21 - 30 inches of concrete adjacent to the hot side will be degraded and crushed. It appears that these two statements are contradictory and a clarification should be provided. In addition, provide a description of the R&D program which will confirm the assumption that the liners anchors will keep degraded concrete in place and prevent spalling.

Response:

The response to this question is provided in Section 2.2.5 and 3.2.2.5.1.1.2 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 130.109

The statement in the first paragraph on Page F-4 regarding failure of the reactor cavity contradicts that in the paragraph near the middle of Page D-4, a clarification should be provided.

Response:

This question is no longer applicable because it refers to statements in the TLTM documentation that have been updated and replaced by CRBRP-3, Volume 2 (Reference 10b, PSAR Section 1.6).

Question 130.110

Justify the use of only 2 psi (refer to Item F, Page F-3) internal pressure in your structural analysis of the RC.

Response:

A more detailed structural evaluation of the Reactor Cavity concrete wall and steel liner is included in Section 3.2.2.5 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6). The internal pressures for the Reactor Cavity during TMBDB conditions are given in Section 2.0 of the above reference.

Question 130.111

The material properties presented in Appendix G will be affected by various factors. All material properties used in the structural design should be established on a realistic basis; therefore, provide a discussion of how you will account for the effects of the following:

(1) biaxial and tri-axial stress/strain distribution

(2) stress/strain gradients

(3) non-uniform transient temperature gradients

(4) occluded gases/fluids

(5) sodium penetration

(6) insert and reinforcing bars subjected to temperature changes and reactions with sodium

Response:

(1) The response to this part of the question is provided in Section C.3.2.9 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

(2) The response to this question is provided in Section C.3.2.2 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

(3) In the evaluation of the various structural components for TMBDB, consideration was given to the non-uniform transient temperature gradients. Thermal stress analyses were carried out, using finite element models suitable for non-uniform temperature gradients (Sections 3.2.2.5 and 3.2.3.3 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6)). The analytical models were properly discretized at regions of steep thermal gradients.

(4) From the test program of concrete at elevated temperatures and the literature review, the properties of the concrete to be used in the TMBDB analysis were selected as explained in Appendix C.3 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6). The effects of occluded gases and fluids are included in these properties since the tests were conducted with material representative of the concrete to be used in CRBRP.

(5) The response to this part of the question is provided in Section G.2.1 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

(6) The temperature effects on reinforcing steel and steel inserts were accounted for by using temperature dependent properties as discussed in Section C.3.3 of the CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6). The concrete cover of the reinforcing steel is such that no contact with liquid sodium is expected. Even if there is contact of steel with sodium, test results reported in References Q130.111-1 and Q130.111-2 show that there is no significant chemical reaction of carbon steel with sodium.

References:

Q130.111-1 - Boehmer, W. D., and Hilliard, R. K., "Sodium Fire Protection by Space Isolation, Open Catch Pan and Nitrogen Flooding - FFTF Proof Test F5" HEDL-TME 75-98, UC-791, October 1975.

Q130.111-2 - Hilliard, R. K., "Sodium Concrete Reactions, Liner Response, and Sodium Fire Extinguishment", Hanford Engineering Development Laboratory, Paper 2-5, HEDL-SA-983.

Question 130.112

In Section 3, Item 2 of the TLTM update report, it is stated that the effect of the 4 inches of insulating concrete has been incorporated in the current analysis. Indicate how the effect of the insulating concrete (including the loose gravel aggregate on the cell floor) is considered in your analysis.

Response:

The information requested will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Appendix C.1.

Appendix C.1 discusses the concrete wall design including the insulating concrete thickness employed and the representation of the floor gravel aggregate. The insulating concrete thermophysical properties are provided in Appendix C.1.4.

Question 130.113

The ASME Code does not specify a relationship between temperature and yield stress. Indicate how the relationship shown in Table 7 of the TLTM update report is established for the specific type of steel being used for the RCB.

Response:

The information requested will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Section 3.2.2.5 and Appendix C.3. The structural material properties and their bases are provided in Appendix C.3. The application of those properties is discussed in Section 3.2.2.5.

Question 130.114

In Section 4.4.1 of TLTM update report, it is stated that the flow resistance in concrete will lead to reduction of hydrogen generated in the reactor cavity. Indicate if such flow resistance in concrete will give rise to a pressure higher than design behind the steel liner, thus resulting in the failure of portion of the liner not in contact with the hot sodium.

Response:

The information requested will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Sections 2.1.2.5 and 2.2.5 which discuss the requirements and design of the reactor cavity and liner vent system. The system is designed to limit the pressure behind the liner, to preclude failure.

Question 222.1 (1.1.3)

Complete Table I (Section 1.1.3) to include the applicability of all Regulatory Guides issued to date. Note that Regulatory Guide 1.75, as revised, will be applicable to the Clinch River application. Complete the applicability information for this Regulatory Guide.

Response:

The response to this question is provided in the expanded Table I, Section 1.1.3.

Question 222.2 (1.2)

Figures 1.2-4 thru 1.2-33, in large part, are not legible. Submit a set of larger scale and legible copies to be used as a working set during our review.

Response:

Two sets of larger scale General Arrangement Drawings (Figures 1.2-4 thru 1.2-9 and 1.2-11 thru 1.2-33) have been provided, along with a cross-reference index of Burns and Roe drawing numbers and PSAR figure numbers. Additional General Arrangement drawings (Figures 1.2-34 thru 1.2-46) have been added to the PSAR as part of Amendment 1. Large scale prints of these additional drawings have been provided and are identified on the cross-reference index.

Question 222.3 (1.5)

On page 1.5-10 you discuss the details of Primary and Secondary Shutdown Systems. There is an appreciable difference in the testability provided for the primary and secondary systems. Provide and describe the testability provided for the secondary system as required for compliance with IEEE Std 279-1971 and Regulatory Guide 1.22.

Response:

The test program described on pg. 1.5-7 through pg. 1.5-11 will provide relevant data necessary to confirm the overall CRBRP shutdown systems reliability. This test program is not related to the shutdown system testability required by IEEE 279-1971 and Regulatory Guide 1.22. On line testing, shutdown testing and calibration are performed on the shutdown systems in accordance with IEEE 279-1971 and Regulatory Guide 1.22.

Both the Primary and Secondary Reactor Shutdown Systems can be tested during shutdown by inserting test signals at the sensors and observing the final shutdown logic actuation. Equipment is calibrated at intervals to assure that required system availability is maintained. Both shutdown systems can also be tested on line. Further information about System Testability can be found in Section 7.2.1.1 of the CRBRP PSAR. Conformance with Regulatory Guide 1.22 "Periodic Testing of Protection System Actuation Functions" is discussed in Section 7.1.2.8 of the CRBRP PSAR.

Question 222.4 (1.5.2 and 7.5.1.1.1)

Provide the design bases and design criteria for the Source Range Flux Monitoring (SRFM) System discussed in Section 1.5.2.6 and 7.5.1.1.1.

Response:

Section 7.5.1.1.1 has been expanded and reference to another section added to provide the requested design bases and design criteria.

Question 222.5 (3.10)

Section 3.10 does not address the safety related equipment discussed in Chapter 8.0. Furthermore, we will require that seismic qualification be performed and/or documentation be provided according to the requirements of IEEE Std. 344-1975 (emphasis added).

Identify specifically, and provide a listing of all safety-related equipment and structures that will be seismically qualified as Seismic Category I.

Response:

The information requested is provided in revised Section 3.10.1 and revised Table 3.2-3.

Question 222.6 (3.11)

Section 3.11 is not adequate in informational content for us to perform an evaluation of your environmental qualification program and its scope. Provide additional information as specified by the Standard Format to include definition of the hostile environment in various locations of the plant housing safety-related equipment and cabling, by specifying the levels of temperature, pressure, humidity, chemical, and radiation exposure and "time in hostile environment." Discuss compliance with and applicability of IEEE Stds 317-1974 (emphasis added), 382-1972, 334-1971, and Regulatory Guides 1.69 and 1.89.

Response:

The environmental extremes that will be used for qualification of safety related equipment will be provided in August 1976. Recent design changes and the definition of design basis leaks must be factored into the definition of these environmental conditions. The response to Q222.54 identifies the equipment, types of environment and the applicable qualification standards. 24

Question 222.7 (7.1)

Supplement the list of safety-related systems in Table 7.1-1 to include all such systems identifiable on the basis of your preliminary design.

Response:

A revised Table 7.1-1 has been provided.

Question 222.8 (7.1)

Provide two copies of each RDT Standard listed in Table 7.1-4. Discuss how each will be used and show that there would be no conflict between these standards and applicable GDCs, IEEE Standards, and Regulatory Guides.

Response:

For the Plant Protection System, the C and E series of RDT Standards listed in Table 7.1-4 will be used to form the basis for preparation of equipment specifications. In order to ensure their correct use in the procurement of equipment, specific sections or paragraphs will be included in the specifications, as appropriate. For other safety related instrumentation and controls, the C and E series of RDT Standards listed in this Table will be used as required to supplement existing commercial and nuclear codes already in existence. When adequate commercial or nuclear codes do not exist, selective use will be made of these RDT Standards to upgrade the equipment to the level required by its intended function.

RDT F2-2 is applicable to all of the safety related instrumentation and controls and will be applied by including appropriate sections into the equipment specifications. The rest of the F series of RDT Standards listed in this Table will be applied on a selective basis.

These RDT Standards, applicable GDCs, IEEE Standards, and Regulatory Guides will be reviewed by the appropriate organizations at the time of application and Regulatory will be notified of any identified conflicts as well as proposed solutions to these conflicts.

Two copies of each RDT Standard listed in Table 7.1-4 are provided herewith.

Question 222.9 (7.2.1.2, 7.3.1.2)

Supplement the information submitted in Section 7.2.1.2 and 7.3.1.2 to include ranges and setpoints of the protective channels of the PPS and ESF.

Response:

A. Section 15.1.3 provides the information on the values of plant parameters which will result in the trip of a particular protective subsystem. Section 15.1.3 presents the trip levels and equations which are used in the performance analysis of the protective subsystems. These expressions represent the estimated worst case performance of the protective subsystems. Nominal setpoints will be provided as part of Chapter 16 in the FSAR.

B. Specific setpoints for the Containment Isolation Functions have not yet been specified. However, as stated in Section 7.3.1.2.1, the setpoints will be so chosen so as to limit any activity release to the appropriate levels.

Question 222.10 (7.2-7.6)

Your discussion of the PPS and ESF general functional requirements is incomplete. Expand Sections 7.2.2, 7.3.2, 7.4.2, 7.5.2 and 7.6.2 to conform to the content specified in the corresponding sections of the Standard Format. Specifically, discuss how the requirements of GDCS and IEEE Standard 279-1971 are met on an individual and section by section basis. Also, show compliance with the recommendations of applicable IEEE Standards, Regulatory Guides and Branch Positions as given in Appendix 7A of the Standard Review Plan.

Response:

The discussion of the means of meeting the GDC applicable to the instrumentation systems is included in Section 3.1. Discussion of the compliance with Regulatory Guides 1.11, 1.22, 1.47, 1.53, 1.62 and 1.63 is included in Section 7.1 as is the compliance with IEEE Standard 323-1974, 336-1971, and 338-1971. Tables 7.1-2, 7.1-3 and 7.1-4 list the Regulatory Guides, IEEE Standards and RDT Standards, respectively, which the project has stated that the design will meet.

Analyses of the performance of the shutdown and decay heat removal systems are presented in Chapter 15 to demonstrate the necessary performance capability of the systems. Evaluation of the capabilities of the equipment's designs to perform the necessary function in the environment that these equipments are subjected to is included in Section 7.2.1.2.3.

A summary of the requirements of IEEE 279-1971 on the Plant Protection System and the specific sections of Chapter 7 in which these requirements are discussed, follows:

IEEE 279-1971, Section 4.1 General Functional Requirement for the Plant Protection System to automatically initiate appropriate protective action when necessary is discussed in Section 7.1.2.1 (Design Basis) and 7.2.2 (Analysis - General Functional Requirement). The General Functional Requirement for Engineered Safety Features is discussed in Section 7.3.1 (Containment Isolation System Description). The General Functional Requirements for the Steam Generator Auxiliary Heat Removal Instrumentation and Control System (SGAHRS) is discussed in Section 7.4.1.1 (Design Description). The General Functional Requirement of the Outlet Steam Isolation Instrumentation and Control System (OSIS) is discussed in Section 7.4.2.1. The General Functional Requirement of the Sodium-Water Reaction Pressure Relief System (SWRPRS) Instrumentation and Control System is discussed in Section 7.5.6.1 (Design Description). The General Functional Requirements for the Treated Water Instrumentation and Control System, Fuel Handling and Storage Safety Interlock System and the Overflow Heat Removal Service Instrumentation and Control System are discussed in Sections 7.6.1.1, 7.6.2.1, and 7.6.3.1.

4.2 Single Failure Criterion requiring that any single failure not prevent protective action is discussed in Section 7.1.2.10 (Conformance with Regulatory Guide 1.53, "Application of Single Failure Criterion to Nuclear Power Plant Protection Systems") and in Section 7.2-2 (Analysis-Single Failure Criterion). Design features to prevent random failure degradation of the two diverse Reactor Shutdown Systems, each with three independent channels are discussed in Section 7.2.1.1 (Reactor Shutdown System Description). IEEE 379-1972, "IEEE Trial Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems", is applicable as listed in Table 7.1-3. Design features of the Containment Isolation System to prevent random failure degradation including the arrangement of the redundant channels are discussed in Section 7.3.1.1. Section 7.3.2.2 (Design Features - Single Failure) discusses the effect of Single Failures on the Containment Isolation System. Redundancy and Diversity features of SGAHRS to assure that it performs its safety function following a single failure is discussed in Section 7.4.1.1.5 (Redundancy/Diversity) and in Section 7.4.1.2 (Analysis). Redundancy and diversity in OSIS is discussed in Section 7.4.2.1.5 (Redundancy/Diversity) and in Section 7.4.2.2 (Design Analysis). Design Analysis on a single failure within SWRPRS and its effect on SGAHRS is discussed in Section 7.5.6.2 (Design Analysis). Redundant monitoring necessary to meet the Single Failure Criterion for the Treated Water Instrumentation and Control System is discussed in Section 7.6.1.2 (Analysis).

4.3 Quality of Components and Modules

Conformance to IEEE 336-1971, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations", is discussed in Section 7.1.2.6. Regulatory Guide 1.30, "Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment", is applicable as listed in Table 7.1-2. RDT F2-2, "Quality Assurance Program Requirements", is applicable as listed in Table 7.1-4. Refer to Chapter 17 of the CRBRP PSAR for more information on the Quality Assurance Program.

4.4 Equipment Qualification to verify that Plant Protection System equipment meets the necessary performance requirements is discussed in Section 7.1.2.5 (Conformance to IEEE 323-1974, "Qualifying Class IE Electric Equipment for Nuclear Power Generating Stations").

4.5 Channel Integrity to maintain the necessary functional capability of Plant Protection System Channels under applicable extremes of conditions is discussed in Section 7.2.1.2.3 (Environmental Changes).

4.6 Channel Independence of redundant safety related equipment is assured through specific physical separation criteria listed in Section 7.1.2.2. Regulatory Guide 1.75, "Physical Indpendence of Electric System", and IEEE 384-1974, "IEEE Trial Use Standard Criteria for Separation of Class IE Equipment and Circuits", are applicable to safety related Instrumentation and Control Systems as shown in Tables 7.1-2 and 7.1-3.

4.7 Control and Protection System Interaction

The capability of the Plant Protection System to protect the plant regardless of worst case control system action or lack of action is discussed in Section 7.2.2. (Control and Protection System Interaction) as are the isolation devices used between control and protection functions (RDT Standard C16-3T on PPS Buffers is applicable). Section 7.7.2 discusses cases where instrumentation signals are provided to the control system by the protection system. Analysis shows that the consequences of potential multiple failures caused by a credible control system action are mitigated by diverse instrumentation in the other independent (and thus unaffected) shutdown systems. As stated in Section 7.3.2.2, there are no shared components between the Control System and the Containment Isolation System.

4.8 Derivation of System Inputs

Protection system inputs are derived from variables that are direct measurements of desired variables. The Reactor Shutdown System inputs are discussed in Section 7.2.1.1 (System Instrumentation). The Containment Isolation System Subsystems are discussed in Section 7.3.1.2.1. The SGAHRS initiating circuits are discussed in Section 7.4.1.1.3. The OSIS initiating circuits are discussed in Section 7.4.2.1.3. The SWRPRS initiating circuits are discussed in Section 7.5.6.1.2. The OHRS initiating circuits are mentioned in Section 7.6.3.1.3 (which refers to Section 9.1.3 which has been revised to include the requisite information).

4.9 Capability for Sensor Check is provided to assure the operational availability of each system input sensor during reactor operation. Cross-checking sensors with redundant sensors and related measurements is discussed in Section 7.2.1.1 (System Testability) as is Channel Output Monitoring.

4.10 Capability for Test and Calibration during reactor operation for instrument channels, logic trains, final actuation logic, and HTS shutdown is discussed in Section 7.2.1.1 (System Testability) and in Section 7.2.2 (Analysis - Periodic Testing). Conformance with Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions", is discussed in Section 7.1.2.8. Conformance with IEEE 338-1971, "Periodic Testing of Nuclear Power Generating Station Protection System", is discussed in Section 7.1.2.7. Channel output monitoring is also referred to in Section 7.2.1.1. The Containment Isolation System Testability is referred to in Section 7.3.2.2 and stated to be the same as those specified for the Reactor Shutdown System (Section 7.2.2). SGAHRS testability is discussed in Sect. 7.4.1.1.7.

4.11 Channel Bypass or Removal from Operation to permit one channel to be maintained, tested or calibrated during power operation is discussed in Section 7.2.1.1 (System Testability) and in Section 7.2.2 (Analysis - Bypass).

4.12 Operating Bypasses which are automatically removed when permissive conditions are not met are discussed in general in Section 7.2.1.1 (System Testability). Specific functional performance of each bypass is discussed with the subsystem descriptions in Sections 7.2.1.2.1 (Primary Shutdown System Subsystems), and 7.2.1.2.2 (Secondary Shutdown System Subsystems). No bypasses are provided for the Containment Isolation System (as stated in Section 7.3.2.2). There are no bypasses for SGAHRS as discussed in Section 7.4.1.1.4. Bypasses for OSIS are discussed in Section 7.4.2.1.4. For SWRPRS as mentioned in Section 7.5.6.1.4 and for OHRS as mentioned in Section 7.6.3.1.4, the design has not yet progressed to the point at which bypasses can be fully described.

4.13 Indication of Bypasses to continuously show that some part of the Plant Protection System has been bypassed or deliberately rendered inoperative is discussed in Section 7.2.2 (Analysis - Bypasses). Conformance with Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems", is discussed in Section 7.1.2.9.

4.14 Access to Means of Bypasses

Administrative controls are used to assure correct use of bypasses as stated in Section 7.2.2 (Analysis - Bypasses).

4.15 Multiple Set Points where necessary to provide protection for a particular normal mode of operation are discussed in Section 7.2.2 (Analysis - Multiple Set Points).

4.16 Completion of Protective Action Once It is Initiated is discussed for the Reactor Shutdown System in Section 7.2.2 (Analysis - Completion of Protective Action) and for the Containment Isolation System in Section 7.3.2.2 (Design Features - Completion of Protective Action). As stated in those sections, both systems are designed so that, once initiated, a protective action at the system level must go to completion. Return to operation requires manual reset by the operator.

4.17 Manual Initiation of each protective action at the system level is discussed in Section 7.2.2 (Analysis - Manual Initiation). Conformance with Regulatory Guide 1.62, "Manual Initiation of Protective Functions", is discussed in Section 7.1.2.11. Manual Initiation of the Containment Isolation System is addressed in Section 7.3.2.2. Manual Initiation of SGAHRS, OSIS, and SWRPRS is mentioned in Section 7.2.2. Manual initiation of OHRS is discussed in Section 7.6.3.1.3.

4.18 Access to Set Point Adjustments, Calibration, and Test Points using administrative controls is discussed in Section 7.2-2 (Analysis - Access).

4.19 Identification of Protective Actions is discussed in Section 7.2.2 (Analysis - Information Read-Out and Annunciator for PPS Alarm Trips).

4.20 Information Read-Out providing the operator with protection system information pertinent to its status and the plant's safety is discussed in Section 7.2-2 (Analysis - Information Read-Out) and Annunciator for PPS Alarm Trips). Indication of Bypasses is discussed in Section 4.13 above. Information provided to the operator for the status of SGAHRS is listed in Section 7.4.1.1.9. Operator information provided for OSIS is addressed in Section 7.4.2.1.8. Monitoring instrumentation for SWRPRS is discussed in Section 7.5.6.1.6. Instrumentation for OHRS is mentioned in Section 7.6.3.1.2 (which refers to Sections 9.1.3 and 9.3), which have been revised to include the requisite information.

4.21 System Repair

To facilitate the recognition and location of malfunctioning components, the capability for test and calibration is provided (see Section 4.10 above for appropriate references to the CRBRP PSAR).

4.22 Identification distinctly denoting Plant Protection System equipment is described in Section 7.1.2.3 (Physical Identification of Safety Related Equipment).

The above paragraphs detail the conformance of the PPS and ESF with the requirements of the GDCS and IEEE 279-1971.

Question 222.11 (7.2-7.4)

Submit a set of Functional Logic Control Diagrams for PPS, ESF systems and other systems required for safety, showing the flow of signals and the logic structures associated with each subsystem including interlocks, permissives, bypasses, etc.

Response:

The requested text material and diagrams have been added to revised Sections 7.2.11 and 7.3.11. Figure 7.3-2 has been added and figure 7.2-2 has been replaced with new figures 7.2-2A through 7.2-2E.

Question 222.12 (15.0)

For each postulated event analyzed in Chapter 15, submit a list of all systems which are assumed to be operable and required to mitigate the consequences of the event analyzed. List separately the primary and secondary shutdown systems.

Response:

A list of all systems which are assumed to be operable and required to mitigate the consequences of the accident events analyzed in Chapter 15 is provided in revised Section 15.1.3 through the addition of Table 15.1.3-3.

Question 222.13 (7.2.1.1)

In Section 7.2.1.1 you state that the Primary Shutdown System acting alone can terminate all extremely unlikely events without exceeding specified limits even if the most reactive control rod in the (primary) system system cannot be inserted. In the same paragraph you indicate that the Secondary Shutdown System (acting alone) can terminate all anticipated and unlikely events without exceeding the specified limits even if the most reactive control rod in the (secondary) system cannot be inserted. To assist us in understanding the exact meaning of these statements, please indicate if the words added above, in parentheses, reflect your intended meaning.

Furthermore, discuss your reasons for designing the Secondary Shutdown System to be non-redundant to the Primary Shutdown System.

Response:

Yes, the words added above do reflect the intended meaning. The secondary shutdown system, acting alone, can terminate all anticipated and unlikely events without exceeding the specified limits even if the most reactive control rod in the secondary system cannot be inserted.

The objective of providing separate primary and secondary shutdown systems is protection against common mode failure degradation of the reactor trip function, not redundancy. Diversity of trip initiation parameters is considerably more effective, and therefore more important, than redundancy in preventing common mode failure degradation of both systems.

Question 222.14 (7.5)

The Sodium-Water Reaction Pressure Relief System (SWRPS) Instrumentation and Controls are improperly classified as part of the "Instrumentation and Monitoring System." Discuss your reasons for not classifying this system as an Engineered Safety Features System.

Response:

Functions of the SWRPRS Instrumentation and Control System that result in scram of the reactor are safety related and are part of the Plant Protection System.

The sodium-water reaction detection instrumentation for monitoring rupture disk failure falls into this category. This equipment consists of three redundant, independent sensors which are classified Class IE and will be qualified using all applicable US NRC Reg. Guides and Industry Standards.

Upon receipt of these sodium-water reaction signals, indicating rupture disk failure, the secondary plant protection system initiates scram and trips the PHTS and IHTS sodium pumps in the affected loop.

Indication of rupture disk failure also initiates SWRPRS functions that are not classified as engineered safety features systems. This includes tripping of the SGS recirculation water pump, closure of the IHTS argon cover gas supply valve and normal feedwater isolation valve, actuation of equipment necessary to isolate and vent the steam generator of the affected loop, and operation of the evaporator water dump valve and nitrogen purge gas control valve to isolate and control a sodium-water reaction.

Failure of these non-safety feature systems does not affect the integrity of the reactor coolant boundary, or the ability of the decay heat removal system to perform its function. Failure of these systems will result in prolonged steam venting through the SWRPRS and the generation of larger quantities of hydrogen which will result in prolonged hydrogen venting.

Since the failure of these non-safety feature systems is not expected to result in the release of radioactive material from the primary system or the loss of decay heat removal capability, the current classification is justified.

Question 222.15 (7.6)

The information submitted in Sections 7.6.3.1.4 and 7.6.3.2 on the Overflow Heat Removal Service Instrumentation and Control System (OHRS) is not adequate. Referenced Sections 9.1.3 and 9.3 do not contain the required information. Submit the information required for the above sections.

Response:

The response to this question is provided in revised PSAR Section 7.6.3. 26

Question 222.16 (7.7)

On page 7.7-1 you state that "The Plant Control Systems and other auxiliary instrumentation and control systems are not safety-related since a worst case postulated failure of these systems will not affect the operation of the Plant Protection System". This is not a necessary nor a sufficient reason for these systems to be so classified. Review the classification of these systems on the basis of whether or not they are required to terminate an event or mitigate the consequences of an event that may lead to unacceptable consequences without the function of these systems, or if by a failure in these systems, a condition resulting in unacceptable consequences is created. Reclassify these systems as necessary.

Response:

The original review of these systems was on the basis of whether or not the systems are required to terminate an event or mitigate the consequences of an event that may lead to unacceptable consequences without the function of these systems, or if by a failure in these systems, a condition resulting in unacceptable consequences is created. See revised Section 7.7.

Question 222.17 (7.0)

Most of the information submitted in Section 7.5 should be part of Section 7.2 and 7.3. Similarily, the information submitted in Sections 7.8 and 7.9 (not provided by the Standard Format) should be in Section 7.5. It is suggested that Chapter 7.0 be rearranged to be in accordance with the Standard Format.

Response:

The Standard Format ("Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", February 1974) states on page ii that compliance with the format is not required. A revised format for Chapter 7 was prepared and discussed with the Regulatory Staff in a meeting on March 28, 1974. In recognition of the fact that there were departures from the Standard Format, a special document was produced showing the relationship between the PSAR and the Standard Format and transmitted to regulatory January 30, 1975. It is believed that no format changes need be made.

Question 222.18 (1.0, 3.0, 5.0, 7.0)

Expand all references to IEEE Standards to include the year of the particular standard edition you are referring to in the PSAR.

Response:

It was an objective in writing the PSAR to include the year with the references to IEEE Standards. Pages 3.7-A.19, 3.7-A.21, 7.1-4, and 7.1-5 inadvertently did not include the year. These pages have been revised to include the year.

Question 222.19 (8.2.2.1)

Discuss in detail how your design complies to Regulatory Guide 1.32 and GDC-17.

Response:

Revised Sections 8.2.2.1 and 3.1.3.2 provide the requested discussions.

Question 222.20 (8.2)

Discuss your design criteria of both switchyards. Describe your design provisions for fault clearing.

Response:

The information requested is discussed throughout Section 8.2. Specific information on fault clearing is provided in Section 8.2.2.3. 25

Question 222.21 (8.0)

Discuss compliance of your electric power system to GDC-18.

Response:

The requested discussions are provided in revised Section 3.1.3.2.

Question 222.22 (8.3)

State the consequences of loss of ventilation in the electrical equipment room.

Response:

The Electrical Equipment Rooms (Control Building CRDM Equipment Rooms, Battery Rooms, and the Diesel Generator Building Class 1E Switchgear Rooms) ventilation systems are discussed in Section 9.6.1.2B and the effect of loss of components in this system is addressed in Section 9.6.1.3.4A and Table 9.6-3. 25

Question 222.23 (8.3)

Provide a load, voltage, and frequency vs. time set of curves for the diesel-generator sets. Describe in detail a qualification test program to establish the reliability of the diesel-generator sets to start and accept load within the minimum required time assumed in Chapter 15.0. Relate this reliability to the assumptions made in setting and achieving the goals of the reliability programs of Appendix C.

Response:

Curves indicating variation of load at different time periods for each Diesel Generator are provided in Figures Q222.23-1 and Q222.23-2. Since the Diesel Generators have not been purchased, the variation of voltage and frequency vs. time cannot be furnished at this time. However, these curves will be furnished subsequent to purchase of the diesels.

It has been assumed in Chapter 15.0 that in the event of loss of all offsite AC power sources, the standby Diesel Generators will provide power for the Pony motors of the Primary and Intermediate Sodium Pumps and the SGAHRS Auxiliary Feedwater Pumps for decay heat removal; the power supply is required within a period of 20 seconds after loss of all offsite AC power sources. No reliability requirement for start of the diesel generators is imposed. To assure the capability of the Diesel Generators to start and provide power to the Pony motors and the SGAHRS Auxiliary Feedwater Pumps for decay heat removal within the time period indicated above, the Diesel Generators are tested periodically during plant operation as indicated in Sections 8.3.1.1.1 and 16.4.5.3.

In the event the Diesel Generator used is of a type or size that has not been previously used as a standby emergency power source in nuclear power plant service, the following test program will apply:

a. At least two tests are performed on each Diesel Generator to demonstrate the start and load capability of these units with some margin in excess of the design requirements.

b. Prior to initial fuel loading, at least 300 valid start and load tests are performed. This includes all valid tests performed offsite. (a valid start and load test is defined as a start from design cold ambient conditions with loading to at least 50 percent of the continuous rating within the required interval, and continued operation until temperature equilibrium is attained).

c. A failure rate in excess of one per hundred requires further testing as well as a review of the system design adequacy.

   (1) If failures to start are found to be caused by failures of a generic nature in a single component, it may be possible to correct the problem by use of a different kind of component or to correct the deficiency in the component. If it is possible to independently test the component after its deficiencies

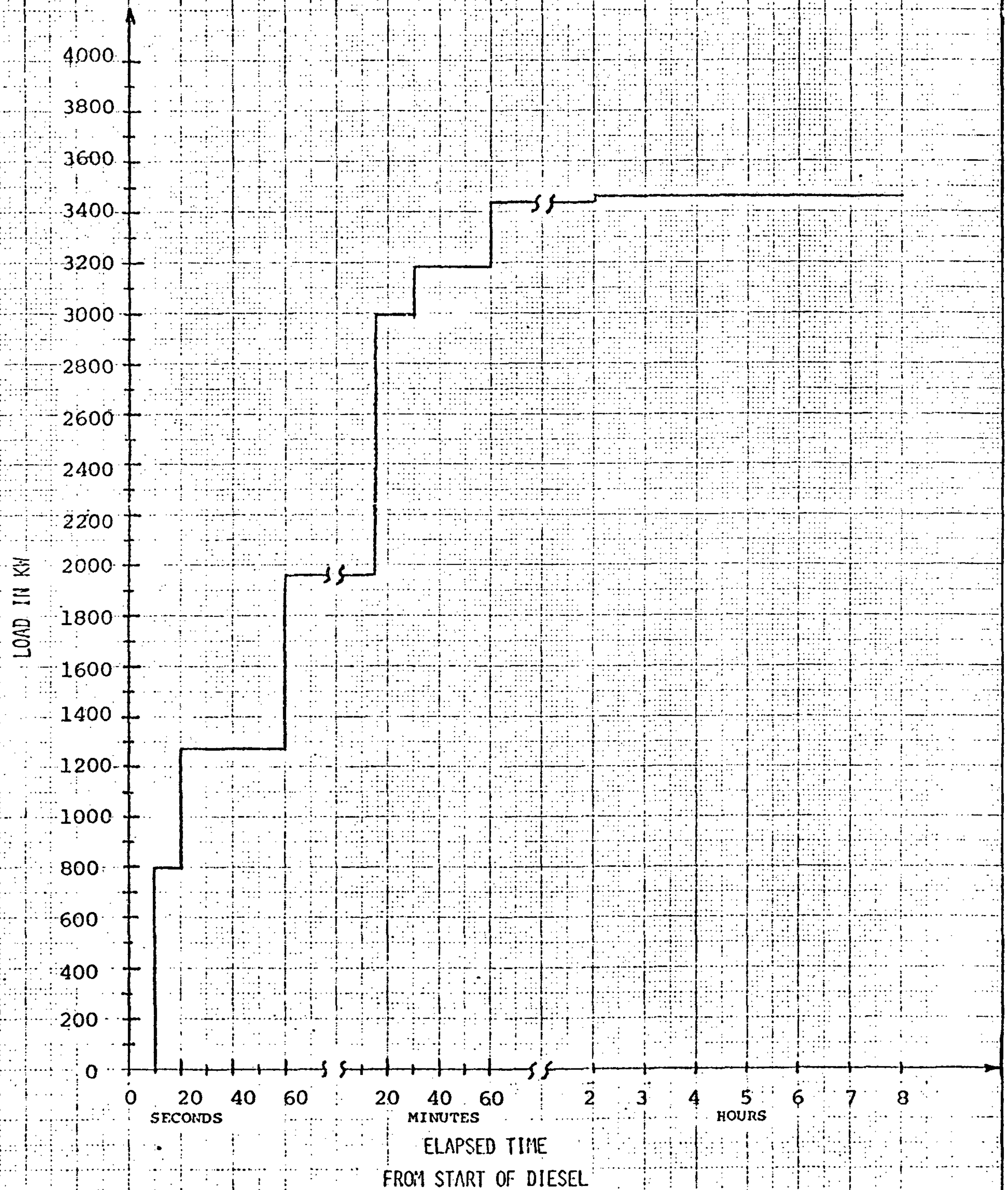
have been corrected, it is not necessary to repeat the 300 starting tests of the complete Diesel Generator unit. If the component is successfully tested 300 times or more under acceptable simulated starting conditions, it will only be necesary to continue and complete the original required 300 unit tests with the replacement component.

(2) If starting failures are of a random nature or cannot be readily identified as being generic component failures, additional starting tests of the complete unit are performed after each starting problem has been corrected. The additional tests are of a sufficient number to verify the required starting reliability.

The Shutdown Heat Removal System Reliability program does not explicitly use the diesel generator failure to start rate but the effect of these failures was included in the failure to start rate (1/1000) which is used for the SGAHRS auxiliary feedwater pump, based on judgemental use of historical data. As currently evaluated by the shutdown heat removal reliability model, variation in failures to start rate of the SGAHRS auxiliary feedwater pump of even a factor of 10 will not significantly affect the relative total unreliability of the heat removal system.

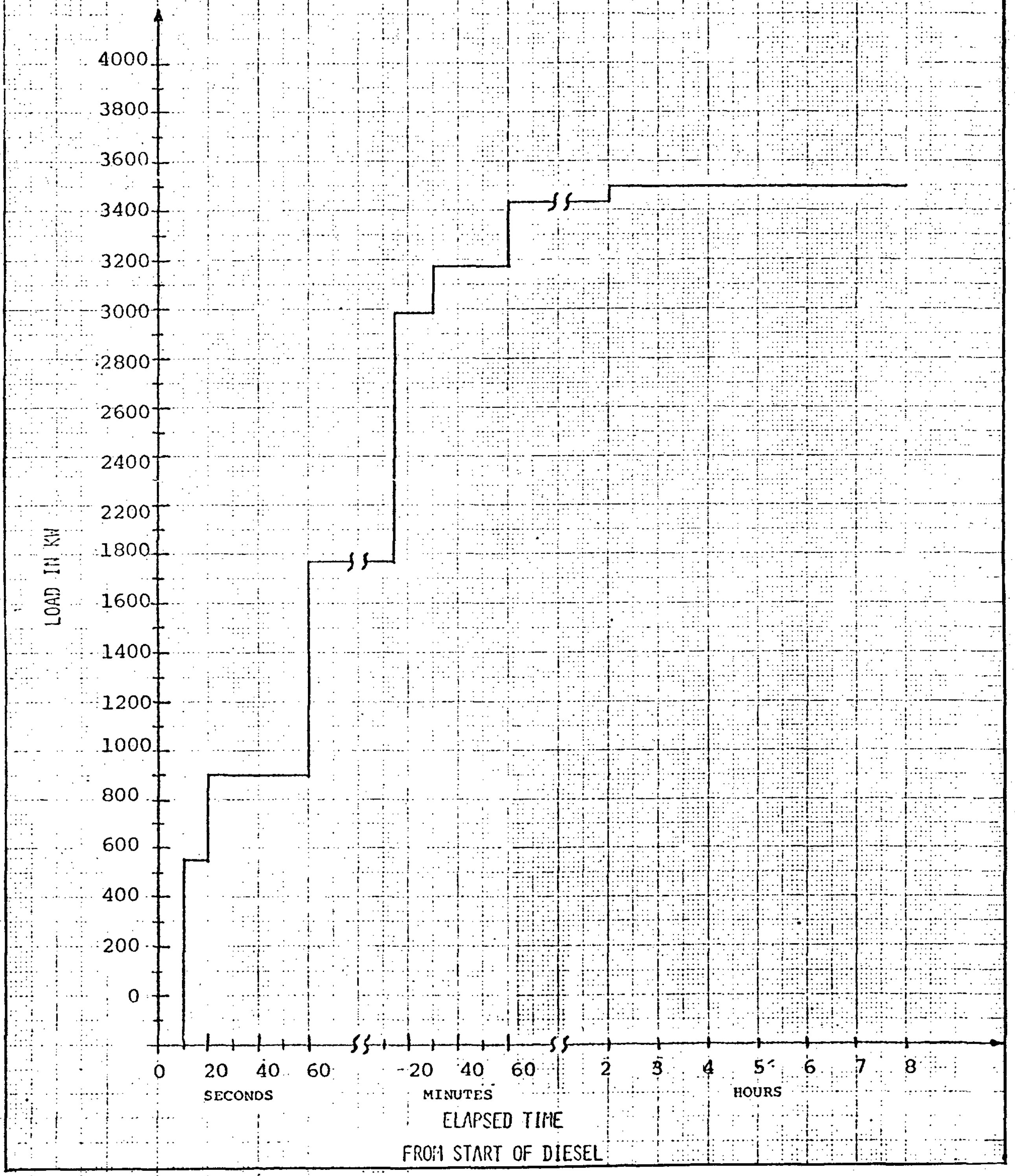
Diesel Generator unreliability will be a continuing consideration during the evolutionary process of refined shutdown heat removal overall modelling and assessment. Negligable effect on the conclusions from the preliminary assessment is expected.

Figure Q222.23-1
DIESEL GENERATOR "A"
LOADING SEQUENCE



Amend. 3
Aug. 1975

Figure Q222.23-2
DIESEL GENERATOR "B"
LOADING SEQUENCE



Amend. 3
Aug. 1975

Question 222.24(8.2, 8.3)

Submit your quantitative evaluation of the unreliability of the off-site power sources to supply electric power to the safety-related ac distribution system during an "extremely unlikely event" (DBA). Identify the location of a sodium fire that would represent the worst case of such a DBA for the CRBRP reference design. Do the same for the onsite power system, estimate the overall unreliability of electric power and relate these estimates to the assumptions made in setting and achieving the goals of the reliability programs of Appendix C.

Response:

The off-site and on-site power systems have been designed to meet the requirements of the CRBRP General Design Criteria and the IEEE Standards listed in PSAR Section 8.1.3. Based on precedents established in LWR licensing and equally applicable to the CRBRP since the reactor type is not germane; the loss of both on-site and both off-site sources is not included in the Design Basis.

Quantitative reliability evaluations were performed as a part of the Shutdown Heat Removal System reliability prediction in Reference (1). Loss of off-site power was estimated to occur at a rate of 3×10^{-5} per seven-day period. This number represents the random independent failure probability of the two 161 KV lines connection independent reserve transformers to the TVA grid and assumes the preferred power source is unavailable. It was an initial estimate reflecting experience data on the Fort Loudon -K31 161 KV line which will supply power to the reserve yard. Over a three-year period this line averaged 0.046 outages/yr. mile. The Fort Loudon -K31 line will be divided into two sections.

(A)	Fort Loudon - CRBRP	11.0 miles
(B)	K-31 - CRBRP	3.5 miles

Using the experience data from above, the probability of line (A) and (B) being out during a seven-day period was computed

$$^{(P)}(A_7) = 9.7 \times 10^{-3} \ \frac{\text{outage}}{\text{7 day period}}, \qquad ^{(P)}(B_7) = 3.1 \times 10^{-3} \ \frac{\text{outage}}{\text{7 day period}}$$

The probability of both lines being out during a seven day period is then

$$^{(P)}(A_7) \times {}^{(P)}(B_7) = 3.0 \times 10^{-5}$$

This is the estimate used for loss-of-off-site power in Reference (1). Since the issuance of Reference (1), additional data has been made available from TVA on over 8300 miles of 161 KV lines. The average outage over a five year period (1971-1975) for this network is shown to be 0.049 outages/yr. miles.

This shows a close correlation to the experience data on the Fort Loudon -K31 line. Operational experience from this broad data base provides confidence that the estimates for individual line outages should be representative of CRBRP power sources.

The numbers noted above represent the probability that in a given one-week period that both lines to the reserve yard would be lost due to random independent failures. Calculation in this manner assumes that if a failure of one line occurs during a seven day period, it will remain in the failed state for the remainder of the period. It implies a maximum repair time of 168 hours for a faulted line. This is a conservative assumption.

The above estimates do not specifically include common cause events which could result in loss of all off-site power lines. Accurate quantification of these types of fault events is difficult but best estimates from available data will be made as a part of the continuing reliability evaluations of the SHRS. However, it should be noted that the above estimate for loss-of off-site power sources does not take any credit for the two physically separate 161 KV lines which constitute the preferred CRBRP off-site power source. Additionally, the plant is designed to provide unit station service in the event of loss-of-off-site power. These design aspects tend to minimize the susceptibility of the CRBRP to common cause events.

The on-site power system consists of two redundant diesel generators. Reference (1) provides an estimate of 1.0×10^{-4} failures per challenge. This estimate was based primarily on a reliability requirement for each diesel to start of 1.0×10^{-2} per challenge. This number will be confirmed through acceptance testing. Additional reliability analyses of nuclear power plant emergency power sources tend to support the estimates given above. These analyses which represent an extensive effort to collect and interpret nuclear power plant diesel generator experience consider generator starting, loading and running phases, with and without repair. When repair considerations are included, the analyses estimate a probability of 1.0×10^{-4} that neither of two diesels will start and run for one hour.

Loss of all AC power as presented in Reference 1 then was estimated to be:

$$P_{(LOSP)} \times P_{(DG)} =$$

$$1.6 \times 10^{-3} \text{ event/yr.} \times 1.0 \times 10^{-4} = 1.6 \times 10^{-7} \text{ event/yr.}$$

This number for loss of all A.C. power is the current estimate based on the available data.

The goal of $.8 \times 10^{-7}$ for the Shutdown Heat Removal System (Appendix C, PSAR) includes unreliability contributions from loss of A.C. power. The relationship of loss of A.C. power to various plant shutdown conditions is discussed in Reference 1.

In identifying the location of a worst case sodium fire, it should be noted 161 KV transmission lines and the switchyard are located outside the plant and are not subjected to sodium fires. The on-site power is supplied by diesel generators located in a Seismic Category 1 building. Wiring for redundant components is separated to prevent a single incident from disabling redundant safety related equipments. The criteria for separation are specified in PSAR Section 8.3.1.4. The separation of the loops and safety related equipment in individual cells or buildings provides the basis for maintaining this separation. Postulated fires involving sodium used in the Ex-Vessel Storage Tank (EVST) were judged to be the worst case in terms in radiological consequences and temperature/pressure transients and are discussed in Section 15.6 of the PSAR.

Reference (1): "An Update of the Preliminary Reliability Prediction for CRBRP Shutdown Heat Removal System" by MEPM-14082, dated January, 1976.

Question 222.25 (8.3)

Revise Tables 8.3-1 and 8.3-2 listing the diesel-generator and d-c system loads in a consistent set of units.

Response

Tables 8.3-1A and 8.3-1B have been updated and revised to provide the requested information for Diesel-Generators A and B respectively. Tables 8.3-2A, 8.3-2B and 8.3-2C provide the requested information for the three(3) Class IE DC Systems.

Question 222.26 (9.3.1.5)

Provide the description, the design bases, and design criteria for sodium and NaK Receiving System Instrumentation discussed in Section 9.3.1.5.

Response:

The revised Section 9.3.1.5, given in response to question 001.203, also provides this information.

Question 222.27 (9.7.5)

Provide the description, design bases, and design criteria for the Auxiliary Coolant Fluid System Instrumentation discussed in Section 9.7.5.

Response:

See revised text of Section 9.7.5.

Question 222.28 (5.6, 7.4)

Provide the design description, design bases, and design criteria for the Steam Generator Auxiliary Heat Removal System (SGAHRS) Instrumentation discussed in Section 5.6.1.15 and 7.4.

Response:

The design description and the design basis of the SGAHRS are presented in PSAR Sections 5.6.1 and 5.6.1.1 respectively. The functional requirements for the SGAHRS instrumentation are identified in Section 5.6.1.1.6.

SGAHRS Instrumentation and Control Design Criteria

IEEE Standard criteria for Class IE engineered safety features systems as identified in Section 7.1 of the PSAR, will be used to demonstrate the ability of SGAHRS I&C to meet their functional requirements under conditions produced by the Design Basis Events.

It is a requirement for the SGAHRS I&C system that it be designed to accommodate an initiating event with an additional single active failure.

Independent power sources will assure continuous SGAHRS I&C operation in the event of Design Basis Events as specified for the SGAHRS Mechanical design.

SGAHRS I&C shall provide sufficient surveillance instrumentation in both the control room and at local panels, to assure availability of operator information of all critical SGAHRS parameters necessary to operate this system under all postulated plant conditions, and during all SGAHRS operational modes.

Where immediate operator information is needed, individual annunciators and indicators and the plant data handling and display systems will supply that information to meet the intent of Regulatory Guide 1.97.

Instrumentation and Control for the redundant flow paths in each Heat Removal loop shall be designed such that no single failure will impair its operation. Cabinets field wiring and sensor location selection shall meet all applicable separation criteria, namely IEEE 384-1974 and USNRC Regulatory Guide 1.75.

Three divisional separation will be adhered to, to obtain maximum availability of SGAHRS I&C consistent with the requirements for the SGAHRS mechanical system design. The intent of BTP APCSB 10-1 as applicable to instrumentation and Control is met by assuring diverse systems powered by different energy sources. This design includes availability of off-site and standby diesel generator AC power, as well as three divisional DC battery power. In conjunction with the Steam Turbine Driven Auxiliary Feedwater Pump, the battery system design will permit auxiliary feedwater instrumentation and control operation with loss of all AC power.

In order to meet GDC 19, SGAHRS Instrumentation and Control shall be available from a location remote to the control room. Transfer from the control room to local control shall be accomplished at local SGAHRS I&C panels.

Shorts, open ground or hot wires in the control room shall not disable SGAHRS remote control from its local panels. Three divisional separate power supplies are used for remote control of SGAHRS I&C. Loss of one division of power will not effect SGAHRS Operation.

Instrumentation and Control equipment qualification will be performed as outlined in Chapter 3.11 of the PSAR, ENVIRONMENTAL DESIGN OF ELECTRICAL EQUIPMENT, and the environmental conditions specified for the CRBRP using applicable standards, Regulatory Guides, BTPs, Federal Regulations, etc., identified in Sections 1.1.3, 3.1 and 7.1 of the PSAR. The method and type of qualification used for SGAHRS I&C will be submitted as part of the FSAR.

Seismic qualification will be performed as outlined in Chapter 3.10 of the PSAR, SEISMIC DESIGN OF CATEGORY I ELECTRICAL EQUIPMENT.

I&C equipment shall be qualified to OBE and SSE conditions as specified for the CRBRP and identified in Section 3.7 of the PSAR.

Plant safety criteria as applicable to the SGAHRS Instrumentation and Control are identified in Section 7.1.2 of the PSAR.

The design criteria for the SGAHRS Instrumentation and Control System are compatible with the Plant Protection System, which initiates SGAHRS I&C by employing a 2/3/ logic of LOW STEAM DRUM LEVEL or HIGH STEAM TO FEEDWATER FLOW RATIO.

Question 222.29 (5.6.2.1.6)

Provide the design descriptions, design bases and design criteria for Overflow Heat Removal Service (OHRS) System Instrumentation discussed in Section 5.6.2.1.6.

Response:

The response to this question is provided in revised PSAR Section 5.6.2.1.6. 26

Question 222.30 (C.1.0)

This Section states that the CRBRP Reliability Program provides the means for assisting in the determination of which events should be included or excluded as CRBRP design basis events. The identification of events which should be included in the CRBRP design basis and those which should be excluded has been made without the full benefit of confirmation through design assessment, confirmatory analysis and testing. Discuss the provisions for and the impact on the overall design safety approach for the later incorporation of events for which exclusion from the CRBRP design basis cannot be justified as a result of ongoing reliability confirmatory analysis/testing.

Response:

As discussed in the response to question 001.1, part d, an updating of the evaluation of events having potential to cause 10CFR100 guidelines to be exceeded is being performed. However, based on the preliminary evaluation which considers a large amount of experience with, and historical information from operating reactors, discovery of events in this category not previously identified, is believed to be unlikely. Should such an event be uncovered, the first step would be to evaluate the reference designs ability to cope with it within the defined reliability requirement. If it is determined that the reference design is inadequate to meet reliability objectives for any event, then a redesign of relevant components would be developed with a trade-off of the schedule, cost, technology, etc., conducted to ascertain project impact. An example of the project approach for such concerns is the provision for incorporation of parallel design features as discussed in Appendices E and F.

Question 222.31 (C.1.3.3)

Section C.1.3.3 presents an initial reliability allocation for the shutdown system (SDS), shutdown heat removal system (SHRS), and for other faults leading to loss of in-place coolable geometry. Provide a table which sub-allocates the SDS, SHRS, and other systems whose faults could lead to loss of inplace coolable geometry down to the major component level such as is shown in Figures C2.2-1 and C2.2-2.

Response:

Reliability goals allocations are no longer applied to reactor shutdown and shutdown heat removal systems and components. The major thrust of the CRBRP Quantitative Analysis Program (Appendix C) is to identify the major contributors to plant unrealibility and to evaluate approaches to appropriately minimize the impacts of these contributors.

Question 222.32 (C.1.3.4)

The reliability program plan presented in Section C.1.3.4 is inadequate. It is evident that many of the key reliability program plan elements are scattered throughout the three sections of Appendix C. Provide a separate section which describes the proposed reliability program plan in its entirety. It is recommended that the format of MIL-STD-785A be used as a guide in the preparation of the reliability program plan.

Response;

Although Section C.1.3.4 describes several key elements found in the program, it was not intended to serve the purpose of a program plan and the title of this section is, perhaps, inappropriate. The attached revised page includes a change to the title.

The safety related reliability effort for the CRBR was originally defined to address two key areas; the Shutdown System and Shutdown Heat Removal System/Structural Reliability. Separate Reliability Program Plans have been written for each of these tasks, the essence of which is contained in Section C.3. As the reliability program evolved, it became clear that additional areas had to be considered to completely address the criteria discussed in Sections 1.1 and C.1 of the PSAR. An effort is currently underway to document in the form of an integrated program plan, the overall reliability program. This integrated program plan will address all important items of MIL-STD-785A.

The essential elements of the reliability program as related to the safety aspects of components procurement are in place. These include reliability involvement in design reviews, reliability review and approval of key design documents, incorporation of reliability requirements into SDD's and E-Spec, and preliminary FMEA's for reliability critical components. Most items identified in MIL-STD-785A have received considerable attention in C.3 including reliability design and evaluation, reliability testing and demonstration and failure data, along with a chart of key milestones in support of these activities.

The integrated plan will be provided upon its completion.

Question 222.33 (C.1.5.4.6)

Figure C.1-2, Reliability Interface with Design, illustrates how the probability program interfaces with the design activities for the primary shutdown system only. Modify Figure C.1-2 to include the reliability interface with design for the entire CRBRP reliability program.

Response:

Figure C.1-2 was included in the PSAR for illustrative purposes only. A schedule containing this level of detail is subject to constant revision, and for this reason comparable schedules were not included for other portions of the reliability program. Figure C.1-3, however, does give the major milestones associated with the shutdown and shutdown heat removal system programs. Further details of the interrelationship between the reliability program and the design are included in the Reliability Program Plan, made available to NRC on January 19, 1976.

Question 222.34 (C2.2.2.2.2)

Provide a failure mode and effects analysis (FMEA) for the CRBRP shutdown system (SDS) using the format outlined in Table C.2.2-2.

Response:

A failure Modes and Effects Analysis for the Shutdown System has been completed, and is in New Supplement I of Appendix C. 25

Question 222.35 (C.3.1.1.1.2)

In addition to the specialized areas covered in the reliability manual, it is recommended that a section be added entitled "Designing for Reliability". As a minimum, such items as preferred circuits, redundancy, protective techniques, derating, part tolerance and drift analysis, statistical design analysis, worst case analysis, marginal testing techniques and environmental considerations should be included to aid designers in the principles of sound reliability design techniques.

Response:

The suggestion made by the question is a good one and such a section will be considered for a later issue of the Manual in which a condensation of design practices and guides referred to above may be assembled in one location.

Currently, the project utilizes a wide variety of published Design Reliability Practices type documents to complement the specific topics selected for treatment in the Reliability Manual. These include the many NASA, NAVORD, IEEE, SAE, etc. publications on this subject as well as the many known published texts treating the many reliability design practice topics.

Question 222.36 (C.2.0)

Describe the degree of conformance to IEEE Standard 352-1975, "General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems" for the CRBRP assessment of reliability.

Response:

The CRBRP Reliability Program is structured along the guidelines provided in IEEE 352-75 and conforms to the intent of that standard.

The stated objective of IEEE-352-75 is to present the general principles that may be used to evaluate the qualitative and quantitative reliability and availability of safety related nuclear power plant systems. Thus, while general principles may be followed, utilization or non-utilization of specific methods described in the standard does not indicate a degree of conformance. The standard discusses four basic areas of Reliability evaluation;

1. Qualitative Analysis Principles
2. Quantitative Analysis Principles
3. Data Acquisition and Use
4. Establishment of Test Intervals

Any well defined reliability program must address each of these topics and the CRBRP program is not an exception.

The in-depth qualitative reliability assessments of the safety systems that are being performed are described in Appendix C of this PSAR.

The reliability program has been designed to provide additional assurance beyond the normal design process that the probability of a loss of core coolable geometry is as low as reasonably achievable. This program addresses the Plant Protection System, and any other equipment whose failure could prevent or degrade the reactor shutdown or shutdown heat removal functions. The analytical principles and intent of IEEE Standard 352-1975 are in evidence in the program's elements, described in Appendix C of the PSAR and discussed as follows:

1. A listing of the equipment critical to the qualitative reliability program objectives is maintained. This safety-related equipment is subjected to Failure Mode and Effects Analysis (FMEA) at the component and system level. The FMEAs are extended to address the effects of faulted interfaces from auxiliary and supporting systems. The FMEAs are used in conjunction with tailored checklists to search out the susceptibility of components and system functional redundancies to common causative factors. These qualitative analyses are conducted on a schedule to permit integration of the program and its findings into the design development process. The reliability assessment is a formal part of the design review for components and systems. The qualitative reliability program assessments are documented at the system level and selected major lower equipment levels in Reliability Design Support Documents (RDSDs).

2. The quantitative analysis principles of the standard are employed in block diagram success path modeling of the reactor shutdown system including the plant protection system. Failure path block diagram modeling of the shutdown heat removal system is employed. The Project uses these tools as aids to the design decision-making process. They provide valuable insights into the identification of the dominant contributors to unreliability. Sensitivity studies of the dominant contributors are conducted to more intensively evaluate the uncertainties associated with failure rate assignments, operating assumptions, and to examine the benefit of potential design modifications. The overall objective is a balanced design through identification and minimization of the effects of major risk contributors.

3. To accomplish this objective, the model is quantified with point estimates of failure rates derived from all available data sources. A failure rate data book is maintained for the SHRS model that documents the derivation of the failure rate of each failure mode assignment in the model. The basis of failure rate assignments in the RSS model are an integral section of the assessment document. A statistically meaningful data base for many first-of-a-kind equipment in CRBRP does not exist. For this reason, the Project can establish only limited statistical confidence in the calculated overall RSS or SHRS failure predictions in these assessments. Although this lack of empirical data deters from the Project's statistical confidence in the overall system failure predictions, it does not negate the models' value as a design decision aid. With emphasis on even application of realistic failure rate assignments (not overly conservative, nor overly optimistic) comparative insights into the overall balance of the RSS and SHRS designs can be gained. These comparative insights can be refined through sensitivity studies that vary the significant modeling parameters to evaluate the system effects of data uncertainties.

4. The last major topic addressed in IEEE 352-75 is that of test interval definition. Implications of test interval on CRBR safety system availability are addressed in C.5.3. Test intervals defined by test and analysis will be incorporated into the proper plant operating specifications.

Question 222.37 (C.3.0)

Describe the qualification procedures and methods as outlined in IEEE Std. 323-1974, Section 6.0, for all Class IE equipment associated with the shutdown systems and shutdown heat removal systems. List and justify any exceptions to IEEE Std. 323-1974. Provide a list of equipment to be qualified.

Response:

Reference 13 of PSAR Section 1.6, "CRBRP Requirements for Environmental Qualification of IE Equipment", establishes the qualification program for qualifying all Class IE equipment to perform its required function under normal, abnormal, design basis event and post design basis event conditions. The entire program is designed to conform to IEEE Std. 323-1974 as clarified by the forward issued by NPEC on July 24, 1975 as IEEE Std. 323A-1975.

Class IE equipment includes the essential safety related electrical equipment of the Reactor Shutdown System, the Containment Isolation System, the Steam Generator Auxiliary Heat Removal System, and other safety related systems. A specific list of Class IE equipment to be qualified is listed in Reference 13 of PSAR Section 1.6.

Type testing, operating experience, analysis or combinations thereof will be used to document that Class IE equipment will meet or exceed its performance requirements throughout the equipment's qualified life.

The qualification will be based upon the most severe environment predicted to occur prior to and during those portions of the specific accident transients for which the component is required to perform its safety function.

Much of the safety related instrumentation in the plant has been successfully employed by the nuclear industry and has already been qualified for nuclear power plant service. Where new instrumentation has been developed (e.g., Reactor Vessel Sodium Level), development test experience will be used for qualification, supplemented by analysis where necessary. FFTF production, acceptance and environmental cycling test results are available for use in qualifying some Class IE equipment (e.g., PPS comparators and logic).

To qualify Class IE equipment by type test, IEEE 323-1974 requires that the equipment to be qualified be aged to simulate its end of qualified life condition and then subjected to the qualifying type tests. After development of industry standards for methods to adequately "age" electric and electronic equipment, the Project will incorporate the appropriate requirements into Reference 13 of PSAR Section 1.6.

In summary, the CRBRP design will comply with the requirements of IEEE-323-1974. The detailed qualification test result summaries for each item of equipment will be submitted upon completion of the necessary analysis and planning as part of the FSAR.

Question 222.38 (Q222.37)

In reference to Acceptance Review Question 222.37, provide: (1) a schedule which identifies the Class IE equipment (including dates) to be qualified, (2) verification that all Class IE items will be qualified, and (3) identification of the method of qualification for each item, i.e., type testing, operating experience or qualification by analysis.

Response:

(1) Class IE equipment to be qualified is identified in Reference 13 of PSAR Section 1.6, "CRBRP Requirements for Environmental Qualification of IE Equipment". Class IE instrumentation and control qualification is scheduled to start in late 1977 and to be concluded prior to the expected issuance of the Operating License Safety Evaluation Report.

(2) All Class IE items will be qualified to the CRBRP design basis event environmental and seismic conditions, using applicable industry standards, Regulatory Guides, Federal Regulations, etc., as discussed in Reference 13 of PSAR Section 1.6.

(3) The methods for qualification will comply with the requirements of IEEE-323-1974 as discussed in Reference 13 of PSAR Section 1.6. Identification of the method of qualification will be provided in the FSAR. A documentation data base will be established which will contain environmental qualification data about each item of IE equipment. This information will be available when the FSAR is submitted.

Question 222.39 (C.1.3.4)

Describe the methods of computing and combining confidence limits for the experimental and analytical results for the Shutdown System and Shutdown Heat Removal System. Provide a sample calculation which illustrates how confidence limits will be used in the CRBRP reliability analysis.

Response:

Confidence statements will be made in the presentation of experimental and analytical results to provide additional assurance of component or system reliability. The approach that is being used is to determine confidence limits for individual tests or analysis. In order to determine the confidence limits of a parameter of interest, the distribution of the data must be known. These distributions will be derived based on test program results, past experience, and/or analysis. All assumptions made during the derivation will be documented, and arguments will be presented as to the reasoning behind the assumptions. In this manner, system estimates can be derived based on inputs with confidence from lower level analysis or testing. In addition to the above procedure, numerous sensitivity studies will be performed to provide insight into the criticality of the assumptions used in any analysis.

There are several potential methods for combining the probability density functions (PDF's) of variables to determine the PDF for a function of the variables. Some of these methods are:

1) Closed form solution of the combination integral.
2) Numerical integration of the combination integral.
3) Monte Carlo simulation.
4) Evaluation of moments.

However, the viability of each method depends on the complexity of the function itself. Also each method requires that a PDF be defined for each input. The input PDF's may be derived directly from confidence statements based on either experiment or judgement. The selection of which method depends on the complexity of the function, the type of PDF's, and the required accuracy. Further comment on generation of PDF's will be provided in the response to Question 222.40.

An example, for a series model using the evaluation of moments, is presented below.

Consider a 3 element series system for which the system failure rate is the sum of the failure rates of the serial elements or:

$$\lambda_S = \lambda_1 + \lambda_2 + \lambda_3$$

where λ_i is a random variable having mean $\bar{\lambda}_i$ and variance σ_i^2.

For a linear combination of random variable, the mean, or expectation, is

$$\bar{\lambda}_S = E(\lambda_S) = E(\lambda_1 + \lambda_2 + \lambda_3) = E(\lambda_1) + E(\lambda_2) + E(\lambda_3) = \bar{\lambda}_1 + \bar{\lambda}_2 + \bar{\lambda}_3 \quad (1)$$

while, if the variables are independent, the variance is:

$$\sigma_S^2 = \mathrm{Var}(\lambda_S) = \mathrm{Var}(\lambda_1 + \lambda_2 + \lambda_3) = \mathrm{Var}(\lambda_1) + \mathrm{Var}(\lambda_2) + \mathrm{Var}(\lambda_3) = \sigma_1^2 + \sigma_2^2 + \sigma_3^2 \quad (2)$$

If the input PDF's are Gaussian (normal), then the PDF for λ_S is also Gaussian with mean and variance as determined by equations (1) and (2). Thus confidence limits may be determined.

Generally, the PDF's for failure rates will not be Gaussian. However, for a linear combination of any PDF, the results still tend toward Gaussian. Sensitivity analyses will be performed on the PDF's of the parameters to explore impact of uncertainty.

The example above was of a generic nature. Current evaluations, employing only point estimates, provide conservative results by using conservative data in the most critical areas. Analyses have not yet been applied to generate and combine distribution functions of the input variables in the models to obtain confidence limits on the results. These analyses are being developed.

Question 222.40 (C.1.3.3)

In reference to the initial reliability goal allocation of $<10^{-7}$ for the Shutdown System, $<8 \times 10^{-7}$ for the Shutdown Heat Removal System, and $<10^{-7}$ for other systems whose faults lead to loss of in-place coolable geometry, provide your method for incorporating a margin of error to account for errors in prediction, measurement, and test conditions.

Response:

Reliability goals allocations are no longer applied to reactor shutdown and shutdown heat removal systems and components. The major thrust of the CRBRP Quantitative Analysis Program (Appendix C) is to identify the major contributors to plant unrealibility and to evaluate approaches to appropriately minimize the impacts of these contributors.

Question 222.41 (C.1.4.1)

Discuss the rationale of the expected frequency of occurrence of the upset events listed in Table C.1-1 with regard to why the individual "event frequencies" are considered to be conservative and how they were derived.

Response:

The frequencies given for Anticipated (Upset) Events in Appendices B and C are in the process of being revised to reflect recent minor changes to the Duty Cycle. The rationale given below will be equally applicable to the new duty cycle, though there may be some minor changes in specific numbers. Table C.1-1 reflects the list of events categorized as "Upset" events from Appendix B, Section B.1.2. This list of upset (anticipated) events includes those events which could have significance to the structural integrity of the NSSS components and piping. Each of these events was assigned a value for the number of occurrences based principally on engineering judgement supplemented by past experience where applicable. It is to be noted that the list includes 180 scram occurrences which have no specifically defined origin but were included to add conservatism to the total number of transients used for structural evaluation of individual components.

The rationale presented in Section C.1.4.1 was to use this list of upset events to derive a list of anticipated events (defining a "reliability duty cycle") which could potentially require PPS action to prevent a loss of coolable core geometry. Seven events were identified as events which could result in hot channel sodium temperature in excess of 1700°F within 5 minutes if no scram occurred. These are:

U-2b	- 10
U-2c	- 10
U-3b	- 15
U-5b	- 5
U-6	- 10
U-16	- 5
U-18	- 6
	61

To this number of anticipated occurrences, 5 additional occurrences were added to account for Emergency (unlikely faults) events, giving a total of 66 over the 30 year life of the plant. The basis for the assignment of frequencies for these specific upset or anticipated events in the design duty cycle is summarized below.

Event U-2b and U-2c, Uncontrolled Rod Withdrawals

These events consider uncontrolled rod withdrawls (from full power and during startup). The design duty cycle includes 20 occurrences of these events. This is judged to be a conservative value since the occurrence of even a small number of such events would result in specific action being taken to correct the cause of such undesirable events. It is also conservative from the standpoint that the event could only be significant when the reactor was not in automatic control (which would correct, by itself, the over-power condition).

Event U-3b, Loss of Power to One Primary Pump

While the title implies a loss of power to a pump (or its M-G set), from a structural (and reliability) standpoint, it includes not only loss of power but other failures resulting in the same effect: a rapid coastdown of one pump in the PHTS. Based on data from Table III 2-1 of WASH-1400 (Reference Q222.41-1), a frequency of 4 would be anticipated considering "pump failure" and motor/generator failures. Failures of the fluid coupling itself together with a fault in the breaker connecting the bus and the pump drive system would not be expected to more than double this frequency i.e., yielding a total anticipated frequency of ∿8), therefore the total frequency of 15 is considered conservative.

Event U-5b, Loss of Normal Feedwater to All Steam Generators

From a structural standpoint this is included to provide a set of cold leg down transients by the injection of SGAHRS water to the steam generators. This event we selected to cover a set of failures such as loss of feedwater pump with failure of its outlet check valve and loss of feedpump suction. Considering pump failures (coincident with check valve failures), pipe breaks in the feed system and coincident failures of valves in the feed system results in an anticipated frequency of <1. A frequency of 5 is therefore very conservative.

Event U-6, Loss of Flow in Two Sodium Loops

Recognizing that the electrical distribution system within the CRBRP place the main pumps in two of the three loops on a single bus, an event was specified to cover this potential common mode failure. Considering failure rate data for transformers, the electrical conductors, and premature transfer for breakers, the anticipated frequency would be <2.5. Ten occurrences is considered conservative.

Event U-16, Operating Basis Earthquake

The frequency specified on the design duty cycle (5) is consistent with WARD-D-0037, Rev.1, "Seismic Design Criteria for the CRBRP."

Event U-18, Loss of Preferred and Alternate Preferred Power

The specified frequency in Appendix B for this event is six (6). This frequency would be consistent with a probability of 0.2 per year (which is consistent with values found in Reference Q222.41-1).

The relationship between the "design duty cycle" and the "reliability duty cycle" is discussed in Appendix 9.2 of Reference Q222.41-2.

This more recent update of the reliability assessment, (Reference Q222.41-2) then, uses a value of approximately 1.2 as the expected number of challenges per year requiring scram action to avoid hot channel temperature in excess of 1700°F within 10 minutes. This level of challenges is consistent with that estimated in Reference Q222.41-3. Even if one takes the position that the frequencies assigned to those events which could lead to hot channel temperatures >1700°F in ten minutes are not conservative enough, one could increase the number without altering the conclusion that the assessment of the CRBRP reactor shutdown system reliability meets its goal. This conclusion is supported by the summary given in Table 1 of Reference Q222.41-2.

To further support the conclusions presented herein, a task is underway to evaluate operating experience in LWR's and other pertinent facilities. This information will be used to provide additional confidence that the choice of events and their related frequencies is appropriate for the type of equipment involved.

References

Q222.41-1. WASH-1400, Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", USNRC, October, 1975.

Q222.41-2. WARD-D-0118, Rev.1, "Reliability Assessment of CRBRP Reactor Shutdown System," November,1975.

Q222.41-3. WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors", September,1973.

Question 222.42 (C.1.5.1.1)

The results of the unavailability assessment for the shutdown systems presented in Table C.1-2 show that for the primary electrical system, the currently assessed unavailability is 5 x 10^{-5} (point estimate) compared to the allocated unavailability of 2.5 x 10^{-5}. What provisions, if any, are planned to upgrade the primary system electrical assessed unavailability to the allocated unavailability goal?

Response:

A revised assessment was completed and reported in WARD-D-0118, Reliability Assessment of CRBRP Reactor Shutdown System, Rev. 1, dated November 10, 1975. The average unavailability of the primary electrical subsystem was predicted to be 5 x 10^{-6} (see Table 6.1-1 in WARD-D-0118, Rev. 1). This improvement results from using additional data sources and improved evaluation and analysis of the data, and improved modeling of the electrical subsystems.

As indicated in the letter which submitted WARD-D-0118, Rev. 1 (S:R:250, P. S. Van Nort to R. Boyd), PSAR material on the reliability program will be updated in a future amendment. Table C.1-2 and the related text will be revised at that time.

Question 222.43 (C.1.5.4.6)

Figure C.1-2, Reliability Interface with Design, illustrates how the reliability program interfaces with the design activities for the Primary Shutdown System only. Provide additional figures which illustrate how the reliability program interfaces with the design activities for the entire CRBRP reliability program.

Response:

This question has been answered in the response to 222.33.

Question 222.44 (C.2.1) (C.2.2.2.1.5)

Provide an analysis which justifies selection of the exponential distribution for the Shutdown System and Shutdown Heat Removal System reliability models. Describe (1) the burn-in program that is required to eliminate infantile failures, and (2) the maintenance-restoration program that is planned to ensure that the failure rate remains constant throughout the 30-year plant life.

Response:

Use of the Exponential Distribution in CRBRP Reliability Models

The exponential distribution and its associated assumption of constant failure rate/probability are utilized in the reliability models for the CRBRP Shutdown System and the Shutdown Heat Removal System. The rationale for the selection and application of this distribution in the reliability models is discussed in the following paragraphs:

Mechanical Shutdown System

The basic distributions used in the reliability confirmation (binomial and poisson) imply a constant failure probability; however, the failure probability of mechanical equipment is known to increase with time and the effects of use. Methods used to resolve this apparent anomaly are: 1) In the instances where the probability of a failure mechanism is known to increase significantly with time, a worst case condition will be used to establish the associated failure probability. This resultant confirmed failure probability will represent an upper bound, and hence be conservative. 2) Units will be tested over their entire cycle life and beyond to envelope all operating conditions and to accumulate wear data for both the PCRS and the SCRS. The large number of scrams accumulated at and beyond anticipated lifetimes will allow results to be initially analyzed based upon the binomial distribution, or a combination of the binomial and poisson distributions. The test will provide an additional verification that the assumption of constant cyclic and time related failure probability is reasonable throughout the design life of the PCRS and SCRS by the following actions.

1. The CRDM will be operated at least one life time during the test. It will be inspected for general indications of wear and for abnormalities after this lifetime. Acoustic signatures will be monitored in an effort to detect potential failures. The signature analysis as well as wear trend data will be utilized to aid in justifying the constant failure rate assumption.

2. The drivelines used during the subsystem test phase will be subjected to at least two lifetimes of operation. Drivelines will be inspected for wear as part of a general inspection for abnormalities which will be performed at points within the test period (roughly equivalent to one lifetime) and again at the end of the second lifetime.

3. The control assemblies will be operated at least five equivalent one-year lifetimes during the Reliability Confirmation Tests. The ducts and wear pads will be replaced and the control assembly will be inspected for wear and abnormalities at the end of each test phase.

Shutdown Heat Removal System

Precriticality plant checkout, including hot functiontal testing, is analogous to a burn-in period for the shutdown heat removal system (SHRS) components. The plant checkout process will be based upon prior power station experience to provide reasonable assurance that heat removal system components will have been adequately screened against infant mortality type failures.

Details of the CRBRP maintenance procedure preparation are not available at this state in the plant design schedule. However, appropriate periodic inspection and repair will be scheduled as suggested by meantime before failure data and the importance of a given component.

Primary and Secondary Electrical System

The assumption of an exponential distribution of time to failure (constant failure rate) is usually made in calculations of electronic equipment reliability. Most text books on reliability usually introduce the exponential distribution as applicable to electronic equipment. MIL-HDBK-217B uses experience data to develop models for the base failure rate of many generic electronic piece part types. With the exception of some rotary devices, these models presented in 217B yield a constant failure rate.

In the CRBRP program, manufacturers of PPS electronic equipment will be required to provide a 100 hour or greater burn-in test in addition to the other production tests which will be performed on all the plant equipment. The plant equipment will receive additional burn-in as part of the pre-startup testing the plant will undergo. By the time that the plant is started, the likelihood of infantile failures will have been significantly reduced and the equipment will be operating near the level portion of the bathtub curve.

A surveillance and replacement program will be conducted and items reaching the end of their design life will be replaced or renewed to assure that all items operate within the constant failure rate portion of their lifetime.

Question 222.45 (C2.2.1) (Q222.31)

Some of the Shutdown Heat Removal System component failure rates presented in Table C.2.2-1 and C.2.2-2 are optimistic. Provide additional information which substantiates these low failure rates. Identify the source for each failure rate. When will the initial report showing the lower level failure rate allocations used for test planning purposes be available for NRC review?

Response:

Our analyses of the source data and the current design indicate that the failure rates assigned are realistic. The sources for failure rates are described in Section 5.2 of NEDM-14082, "An Update of the Preliminary Reliability Prediction for CRBRP Shutdown Heat Removal System," dated January, 1976, in which the probabilistic data sheets are provided. These data sheets represent an update of the PSAR information based upon extensive additional data seraches and evaluation of that data application to the SHRS.

Summary Tables Q222.31-1 and Q222.31-2 show the SHRS reliability allocations* to the subsystem/component levels. 34

* These allocations are provided as guidance to design engineers for use in their engineering assessments of and design modifications for improved reliability. 34

Question 222.46 (C.1.3.4, Q222.32)

The integrated reliability program plan is considered to be a key element in the overall evaluation of the CRBRP reliability program. When will the integrated reliability program plan be submitted to NRC?

Response:

The integrated reliability program plan providing an overview of the Project reliability activities and their relationship to other Project activities was submitted to NRC on January 13, 1976.

This version of the program plan should be considered the first submittal of a working document which may be revised as reliability program development requires. A revision to the plan is currently in work and will be submitted to NRC in the third quarter of CY-76.

Question 222.47 (C.2.2.2.2) (Q222.34)

Modify the Shutdown System FMEA to include the failure rate, repair time, and actual probability of occurrence (from zero-to-one) for each item listed in Table A, Q222.34, Amendment 3, of the PSAR. Include system symbolic logic block diagrams of the Shutdown System FMEA which are used in the derivation of Table A.

Response:

The items analyzed in the Electrical Protective System FMEA are listed in the attached table with their associated failure rate estimates as described in attached Table 222.47-1. As described in the reference, when an item is found failed (by interchannel comparison which is annunciated or scheduled tests), the procedure will be to trip the channel in which the failed item is located. The item can then be repaired. After the repair is completed, the trip will then be removed. Thus, during the period in which the item is repaired, the 2 of 3 system is effectively operating as a 1 of 2 system. Since a 1 of 2 system has a smaller probability of failure to scram when required than a 2 of 3 system, the Electrical Protective System has a pointwise availability during the period when an item is being repaired higher than it has normally. For these reasons, the mean time to repair for all the items was assumed to be zero.

The mean time to trip or the mean time between the discovery of a failure and the tripping of the failed channel is a significant parameter in the evaluation of the Electrical Protective System probability of failure. Operating procedures will dictate that when a failure occurs, the failed channel is to be tripped immediately. As stated in Reference Q222.47-1 in the evaluation of the Electrical Protective System, the mean time to trip had a predicted value of two hours for all items. Analysis has shown however, that the average probability of Electrical Protective System response is not degraded significantly for mean time to trips approaching one shift or eight hours.

Since the Preliminary Failure Modes and Effects Analysis was performed at the system level (as noted in Section 7.2.2 of the PSAR), system level diagrams were used as a basis. Details about the instrument channels analyzed appear in PSAR Section 7.2.1.1, the Reactor Shutdown System Configuration for Primary and Secondary Instrument Channels and Logic are shown in Figures 7.2-3 and 7.2-4, the HTS Coolant Pump Shutdown Logic is shown in Figure 7.2-2E and the Control Rod Drive Mechanism System arrangement is shown in Figure 7.7-4. Logic Block Diagrams were not used for the Reactor Shutdown System Electrical Failure Modes and Effects Analysis and are not used as a part of the CRBRP Shutdown System design process. Detailed instrument channel schematic block diagrams will be submitted in the FSAR with updated Failure Modes and Effects Analysis which will reflect the Reactor Shutdown System design as it is finalized in the FSAR.

Reference:

Q222.47-1 Reliability Assessment of CRBRP Reactor Shutdown System, WARD-D-0118, Revision 1, submitted to NRC by CRBRP Project Office, November, 1975.

TABLE Q222.47-1

Failure Rate Predictions

Item No.	Item Name	Item Unsafe Failure Rate
1.	Reactor Vessel Sodium Level PPS Input	6.4×10^{-6} fail./hr.
2.	PPS Sodium Flow Input	5.0×10^{-6} fail./hr.
3.	Primary Pump Electric Power Sensor	4.0×10^{-6} fail./hr.
4.	Compensated Ion Chamber Nuclear Input	18.4×10^{-6} fail./hr.
5.	Fission Chamber Nuclear Input	18.4×10^{-6} fail./hr.
6.	Primary Loop Inlet Plenum Pressure Input	7.0×10^{-6} fail./hr.
7.	Sodium Pump Speed Input	4.0×10^{-6} fail./hr.
8.	Steam Mass Flow Rate Input	9.0×10^{-6} fail./hr.
9.	Feedwater Mass Flow Rate Input	8.0×10^{-6} fail./hr.
10.	Steam Drum Level Input	8.0×10^{-6} fail./hr.
11.	Primary Comparator	1.9×10^{-6} fail./hr.
12.	Secondary Comparator	1.7×10^{-6} fail./hr.
13.	Primary Logic Train	1.0×10^{-6} fail./hr.
14.	Secondary Logic Train	1.5×10^{-6} fail./hr.
15.	Primary Calculation Unit	1.9×10^{-6} fail./hr.
16.	Secondary Calculation Unit	1.9×10^{-6} fail./hr.
17.	Scram Actuation Logic	See Note 1
18.	Heat Transport (HTS) Shutdown Logic	See Note 2
19.	Control Rod Drive Mechanism (CRDM) Power Train	5.0×10^{-4} fail./demand
20.	PPS Voltage Signal Buffer	3.1×10^{-6} fail./hr.
21.	PPS Current Signal Buffer	3.1×10^{-6} fail./hr.

NOTE 1:

Credit for manual scram initiation is taken only upon the occurrence of a slow acting transient which would not lead to Loss of Coolable Geometry (LCG) within ten minutes if unmitigated. When credit for manual scram initiation is taken, it is assumed that the scram actuation logic has an unsafe failure rate of zero. This is assumed to represent a realistic condition for the following reasons:

(1) It is expected that multiple protective functions would trip upon the occurrence of a slow transient resulting in an automatic electrical response in time to prevent LCG.

(2) Even if the scram actuation logic was in a failed condition, there would be time for the initiation of other means to activate the mechanical subsystem, such as manually tripping breakers which would cut off power to the control rods or using the control system to drive the rods back into the core.

NOTE 2:

There are presently no failure modes identified which could prevent automatic scram initiation.

Question 222.48 (C.3.1.1.1 Q222.35, Q222.36)

The methods and procedures for reliability assessment of LMFBR safety related components will be compiled in the Reliability Manual (SRD-74-113). Describe the indoctrination and training program planned for designers and reliability engineers in the use of this manual, and state the official relationship of this manual to the CRBRP reliability program. When will Revision 1 of the manual be available to NRC, and will it contain the staff recommended section, "Designing for Reliability" as described in Acceptance Review Question 222.35?

Response:

The following steps have been taken to implement the methodology described in the Reliability Manual.

1. Project wide distribution to assure that all personnel having a reliability interface have access to the Manual.

2. A continuing training program, which includes seminars addressing individual chapters of the Manual. These seminars provide background concerning applicability of the method, description of how to use the method and examples relevant to reactor systems.

3. Close interfacing between Reliability and Engineering personnel in the actual application of the methodology.

The following seminars have been given to reliability staff personnel and those design groups directly interfacing with aspects of the reliability program. These seminars were presented at General Electric and Westinghouse, the two organizations having direct responsibility for conduct of the reliability program.

- Failure Modes and Effects Analysis
- General Statistical Test Planning
- Application of Bayesian Techniques to Reliability Test Planning
- CRBRP Reliability Confirmation Test Methods (2 sessions)
- Basic Structural Reliability Methods
- Practical Considerations in Structural Reliability Analysis
- Fault Tree Analysis
- Accelerated Test Methods
- Reliability Prediction for Electrical Circuits
- Design of Experiments

- Common Mode Failure Analysis
- Renewal Theory
- Markov Chains

Relationship of Reliability Manual to the Program

The purpose of the Reliability Manual is to provide a central source of reliability methodology and guidance for applying that methodology in the reliability program. The appropriate procedures to be used are described in the Manual to assure consistency of approach. Although the Manual covers a broad specturm of methods applicable to the program, it would be impractical (and unnecessary) to include all reliability methodology that might be used in the program within the Manual. Additional sources exist and are referenced. An example is Mil Handbook 217B, which provides not only a source of data, but accepted methods for electrical system reliability predictions using the data. Further, to be useful to all elements of a broad based program, the manual must accommodate variations in the application of methods.It, therefore, was not written in the form of a procedure requiring rigid conformance to a specific approach.

Revision 1 of the LMFBR Reliability Manual was provided to the NRC staff on February 5, 1976. The Project does not consider a general section on 'Designing for Reliability' is appropriate for inclusion in the Reliability Manual. The Manual provides a compendium of reliability analysis methods for assistance in design evaluation. However, it is not intended to provide design criteria and guidelines.

The Reliability Manual, in its current form, does contain a broad spectrum of design reliability practices and procedures (e.g., Failure Modes and Effects Analysis, Common Mode Failure Analysis, Fault Tree Analysis). The effort put forth to develop this Manual, and the current use of the Manual by various Project personnel, does indeed satisfy the suggestion that a "Designing for Reliability" type document be available and used on the CRBRP. In addition, the Project utilizes a wide variety of published Design Reliability Practices type documents to complement the specific topics selected for treatment in the Reliability Manual. Appropriate documents for such use are referenced in the Reliability Manual.

More importantly, the Reliability Manual and the many related documents mentioned above are made an integral part of the equipment design effort via the personal working relationships between reliability specialists and designers that occur on a daily basis. No degree of documentation is considered a viable alternative to this process, and the approach to design reliability in CRBRP recognizes this as the key to achieving the effective use and implementation of proper design reliability practices.

Question 222.49 (C.3.0) (C.3.2.2.1)

In addition to the design verification and reliability test program information contained in Section C.3.2.2.1 of Appendix C, provide the following for all eleven tests: (1) the analytical success criteria for each proposed test item; (2) the expected test results; (3) the risk of failure associated with each test, i.e., high, moderate, or low; (4) the components considered to be critical items; and (5) the impact on the program schedule due to failures of major components of the SDS and SHRS to meet the required goals.

Response:

The section of PSAR Appendix C referred to by this question has been deleted. However, test evaluation is still germaine to the Reliability Program. The Reliability Program is discussed in Section C.1.3, C.1.3.1, C.1.3.2, C.1.3.3, and C.1.3.4 of Appendix C. Our current Reliability Program emphasizes qualitative analyses and numerical assessments based upon system functional models and failure parameters gleaned from historical experience and carefully considered engineering judgments. However, the established and ongoing test programs are continuously monitored from the standpoint of reliability. Although a statistical data base is not being sought, any failures during testing that impact reliability will lead to corrective design actions. Various testing programs are discussed in the following sections of Appendix C:

C.5.1.1	C.6.1.2	C.6.5.2
C.5.2.2	C.6.2.2	C.6.6.2
C.5.3.2	C.6.3.2	
C.5.4.2	C.6.4.2	

Question 222.50

Identify and justify all exceptions taken to the detailed provisions of Regulatory Guide 1.75, as amended. Your discussions of compliance to this guide given in the indicated sections are not adequate and they imply that conformance may be a problem. Your response should also address the consideration given to potential fire and other hazard areas because of sodium leaks and spills.

Response:

As stated in PSAR Section 7.1.2.2, the CRBRP instrumentation and control systems including the Plant Protection System will meet the requirements of Regulatory Guide 1.75 (1/75). The electrical system will meet the intent of the Regulatory Guide requirements as noted in Section 8.3.1.2.

Question 222.51 (7.1.2.2) (RSP)

The physical separation criteria listed in Section 7.1.2.2 provide that wiring for the containment isolation and other safety related systems may be run in conduits containing either primary shutdown system wiring, or conduits containing secondary shutdown system wiring.

Furthermore, your criteria provide that this wiring may be brought through the penetrations of either primary shutdown or secondary shutdown systems, provided that no degradation of the separation between primary and secondary shutdown systems results.

These provisions of your separation criteria would not be acceptable because:

(1) There would be an inherent degradation of independence between the primary and secondary shutdown systems by potential sneak paths through the containment isolation and other safety related systems; and,

(2) The potential interaction between the primary and secondary shutdown systems (de-energize to actuate) on one hand and the containment isolation and other safety related systems on the other (energize to actuate) due to a single event such as fire or overheating may put in jeopardy the initiation and/or completion of safety functions.

Amend your criteria to alleviate these concerns, or provide an analysis to show that your present criteria will not make your design subject to these concerns.

Response:

Amended and expanded physical separation criteria for the Containment Isolation System has been included in Sections 7.1.2.2 and 7.3.2.2.

The Containment Isolation System is similar to the Primary and Secondary Shutdown Systems in that the signals from the Control Room to the isolation valve operators function as de-energize to actuate.

Question 222.52 (7.1.1.2)

You state in Section 7.1.1.2 that "Instrumentation equipment associated with redundant channels shall be mounted in separate racks (or completely, metallically enclosed compartments)...". State and justify the criteria for the design of these compartments. You also state that "All interrack PPS wiring shall be run in conduits (or equivalent) with...". Specify the type of conduit to be used, the criteria for its selection, what is equivalent, and how equivalency is established.

Response:

As required by Section 5.7 of IEEE 384-1974, "Redundant Class IE instruments shall be located in separate cabinets or compartments of a cabinet". Racks or completely metallically enclosed compartments will meet the definition in Section 3 of IEEE 384-1974 for Safety Class Structures, and will "protect Class IE equipment against the effects of the design basis events". While specific design criteria for PPS racks have not been developed, the design will assure that no credible single failure can cause the failure of redundant safety functions. Wherever practical, separate racks will be utilized to house redundant safety related equipment.

In general, rigid metal conduit or enclosed raceways will be used to carry PPS wiring. Specific design criteria will meet the requirements of IEEE 384-1974, and Regulatory Guide 1.75.

Question 222.53 (7.1.2.2, 8.3.1.2)

Provide your design criteria and procedures for fire stops and seals. Your response should address but not be limited to the following:

(1) Interval at which the fire stops are installed for vertical and horizontal cable trays;

(2) List of materials used and their characteristics with regard to flammability;

(3) The QA and test procedures used to verify that penetration fire stops and seals have been properly installed;

(4) The administrative procedures and controls that will be followed when it becomes necessary to breach a completed fire stop and to add or remove cables;

(5) The qualification testing of the fire stops and seals to demonstrate adequacy over the life of the plant; and,

(6) The maintenance procedures provided to identify open or deteriorated fire stops and seals.

Response:

Revised PSAR Section 8.3.1.4.D, "Sealing Raceway Blockouts and Wall and Floor Penetrations" discusses fire stop intervals, fire stop materials, materials testing, and QA program requirements. The revised Section 8.3.1.4.D completes the response to items (1) and (2) above. The complete response to items (3) and (5) will be included in the FSAR as this information will not be developed until the detailed plant design is finalized.

Administrative and maintenance procedures (Items 4 and 6) for fire stops and seals will be developed as the design progresses and will be included in the FSAR.

Question 222.54 (7.1.2.5)(RSP)

We require that your design comply to the recommendations of IEEE Std-323-1974 and Regulatory Guide 1.89. The source term for establishing the radiation environment for the reference design should be accepted prior to the calculation of the applicable radiation doses.

Supplement the information given in Sections 3.11.1 and 7.1 by submitting in tabular form a listing of all safety related equipment and components (e.g., motors, cables, sensing, and control devices, etc.), located in the primary containment and elsewhere that are required to function during and subsequent to any of the design basis accidents: provide their location, and for each location, define the worst case design basis environmental conditions in terms of temperature, pressure, and humidity, chemical contaminants, and radiation.

We will require that the following qualification test program information be provided for all Class IE equipment as part of the FSAR:

(1) equipment design specification requirements;

(2) test plan;

(3) test setup;

(4) test procedures; and,

(5) acceptability goals and requirements.

This information will have to be provided for at least one item in each of the following groups of Class IE equipment:

(1) switchgear;

(2) motor control centers;

(3) valve operators;

(4) motors;

(5) logic equipment;

(6) cable; and,

(7) diesel generator control equipment

Response:

CRBRP design will comply with the requirements of IEEE-323-1974 and Regulatory Guide 1.89 as discussed in Reference 13 of PSAR Section 1.6.

The detailed qualification test programs for each item of equipment will be submitted upon completion of the necessary analysis and planning as part of the FSAR.

Question 222.55 (7.1.2.8) (RSP)

Discuss the capability for test and calibration of each protection system channel (both reactor shutdown systems and engineered safety features) listed in Chapter 7.0 of the PSAR, from sensor to final channel output signal. Since these channels must be tested more frequently than the normal time interval between plant shutdown, describe the testing features provided for each type channel, and discuss any restrictions on access to or use of these test features during operation. Identify each channel which is not tested during operation, and show how its design satisfies the requirements of Section 4.10 of IEEE Std 279-1971 and the recommendations of Regulatory Guide 1.22.

Branch Technical Position EICSB 22 of Appendix 7A of the Standard Review Plan provides guidance for the application of Regulatory Guide 1.22. Submit all information specified in the subject Branch Technical Position.

Response:

Additional details on the test and calibration capability of the protection system are given in revised Section 7.1.2.8.

Question 222.56 (7.1.2.9) (RSP)

Branch Technical Position EICSB 21 of Appendix 7A of the Standard Review Plan provides guidance and certain clarifications of the application of Regulatory Guide 1.47. Define the degree of conformance of your design to this Branch Technical Position.

Response:

As stated in Section 7.1.2.9 of the PSAR, the design of the system to automatically indicate at the system level, the bypass (or deliberately induced inoperability) of the Protection System, systems actuated or controlled by the Protection System, or supporting systems that must be operable for the Protection and related Systems to perform their safety related functions will be in accordance with Regulatory Guide 1.47. In addition, the design will fully incorporate the supplemental guidance provided by Branch Technical Position EICSB 21 for the implementation of Regulatory Guide 1.47.

Question 222.57 (7.5.1.1) (4.3.2.1)

Provide an analysis showing the derivation of the relationship between the flux level readings obtained by the ex-core BF_3 filled proportional counters and reactivity of the core in a subcritical state. Your analysis should include a determination of the accuracy and precision of this system under conditions as close as practicable to those in actual installation of CRBRP.

State all automatically initiated protective functions and alarms by the SRFM and provide a description and design bases and design criteria for the associated electrical and instrumentation subsystems.

At the end of the first paragraph on page 7.5-3 of the PSAR, you state that, "In order to achieve this lifetime without retracting the counters, the operating voltage will be removed and the anode of the counter shorted to ground...". Describe how this will be accomplished, and your design feature that will prevent inadvertent removal or inadvertent failure to restore the operating voltage when needed.

Response:

The information requested is provided in revised PSAR Sections 4.3 and 7.5.

Question 222.58 (7.1-1)

Provide a complete listing of the instrumentation necessary to assure the plant is maintained in a safe shutdown status referred to in Table 7.1-1.

Response:

The preliminary list of this instrumentation and of subsystems containing safety related instrumentation is provided in Table F.1-6 in response to Question 222.54. Instrumentation specifically identified for post accident monitoring (PAM) is so marked. Some of these channels will also be required for a safe shutdown from a normal operating status (no accident involved).

Question 222.59

Update Tables 7.1-2 and 7.1-3 to include all Regulatory Guides and IEEE Standards listed in Table 7.1 of the Standard Review Plan.

Response:

The applicable Regulatory Guides and IEEE Standards identified in SRP 7.1 are addressed in Section 7.1. The following Regulatory Guides were not addressed in Section 7.1 which applies to safety related instrumentation and control systems for the reasons given.

Regulatory Guide 1.6, "Independence Between Redundant Power Sources and their Distribution Systems" is complied with as discussed in Sections 8.3.1.2 and 8.3.2.2 of the PSAR with other discussions on power supplies.

Regulatory Guide 1.7 "Control of Combustible Gas following a Loss of Coolant Accident" is noted in Section 1.1.3 to be not applicable to CRBRP.

Regulatory Guide 1.11 "Instrument Lines Penetrating Containment" is not applicable to CRBRP since there are no instrument lines penetrating containment as discussed in Section 7.1.

Regulatory Guide 1.29 "Seismic Design Classification" is discussed in PSAR Section 3.2 "Classification of Components" where it is stated "structures, systems, and components are classified in full conformance with Regulatory Guide 1.29." A list identifying instrumentation and control equipment classified as Seismic Category I is provided in response to question 222.54. 24

Regulatory Guide 1.68 "Test Programs for Water-Cooled Power Reactors" details guidelines for all plant preoperational testing. As noted in PSAR Section 1.1.3 and Chapter 14.0, Regulatory Guide 1.68 will be considered in preparing test plans, though specific plans other than those for LWR's in the guide must be developed for CRBRP.

Regulatory Guide 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" applies to the entire PSAR and provides a suggested format for LWR SAR's. In general, the CRBRP is consistent with the Draft SFAC for LMFBRs, but the guide is not considered appropriate for inclusion in a list of guides applicable to the design of safety related instrumentation and control.

Regulatory Guide 1.78 "Control Room Habitability During Chemical Release" applies to design of LWR control room habitability systems. PSAR Section 6.3 discusses the CRBRP Control Room Habitability System. It is not considered appropriate for a list of guides applicable to safety related instrumentation and control systems.

Regulatory Guide 1.89 "Qualification of Class IE Equipment for Nuclear Power Plants" specifies the applicability of IEEE Standard 323-1974 which will be applied as discussed in Section 7.1.2.5. The regulatory guide also identifies a source term based on an event which would result from failure of all the equipment being qualified. As indicated in new PSAR Section 7.1.2.12, that guidance will be followed in intent, but not detail.

Regulatory Guide 1.96 "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants" applies specifically to a boiling water reactor system and, as indicated in Section 1.1.3, will not be applied to CRBRP safety-related instrumentation and control systems.

Question 222.60 (7.2.1.1)

It is not clear from your discussion of operating bypasses on page 7.2-1 whether or not your design is intended to meet Sections 4.12, 4.13 and 4.14 of IEEE Std 279-1971. Provide sufficient preliminary design information to demonstrate that it is.

Response:

The design is intended to meet Sections 4.12, 4.13, and 4.14 of IEEE 279-1971, as demonstrated by the information in revised Section 7.2.1.1.

Question 222.61 (7.2.1.1)

For each parameter providing a trip function in the primary and secondary shutdown systems provide a logic diagram such as in Figures 7.2-2A, C, showing the associated permissive parameter. Justify the permissive bypass in each case.

Response:

Instrument channel diagrams showing each PPS function, associated instrumentation and other interfaces, including the operating bypass permissive parameter (for PPS functions which are bypassed) will be submitted to NRC in the FSAR. Design information necessary to prepare the instrument channel diagrams will be completed for FSAR submittal. The instrument channel diagrams will be in a form suitable to determine that each operating bypass of a protective function is in full compliance with paragraphs 4.12, 4.13 and 4.14 of IEEE 279-1971 and Regulatory Guide 1.47.

General design information about operating bypasses is provided in PSAR Section 7.2.1.1 in response to Q222.60. The use of operating bypasses is discussed for each Primary and Secondary protective function in Section 7.2.1.2 of the PSAR. For your convenience, it is summarized below.

Primary Reactor Shutdown System

The following summarizes the use of bypass permissives in the Primary Reactor Shutdown System:

Functions which are never bypassed:

- Flux - Delayed Flux
- Flux - $\sqrt{\text{Pressure}}$
- High Flux
- Reactor Vessel Level
- IHX Primary Outlet Temperature

Functions which must be bypassed to start the main sodium coolant pumps:

- Pump Electrics
- Primary to Intermediate Speed Ratio
- Steam - Feedwater Flow Mismatch

Secondary Reactor Shutdown System

The following summarizes the use of bypass permissives in the Secondary Reactor Shutdown System.

Functions which are never bypassed:

- Modified Nuclear Rate
- Flux - Total Flow
- Steam Drum Level
- Evaporator Outlet Sodium Temperature
- Sodium - Water Reaction

Functions which must be bypassed to start the main sodium coolant pumps:

- Primary to Intermediate Flow Ratio
- Loss of Condenser Vacuum

Functions which must be bypassed in ascent to power:

- Startup nuclear

Question 222.62 (7.2-2C)

It is shown in Figure 7.2-2C that a permissive condition is required before a manual trip may be initiated. Discuss your rationale for this design feature. Define the term "Flux Electronics" appearing on the same Figure and specify its relationship to the secondary PPS safety function.

Response:

Figure 7.2-2C has been amended for clarity. Note that before a trip comparator can be bypassed (i.e., trip comparator outputs a reset signal regardless of the relationship between the analog signal input and the setpoint), a permissive condition must exist as determined by the permissive comparator and manual actuation must be performed by the operator. The trip comparator is automatically returned to the operating state as soon as the permissive condition no longer exists, regardless of manual actuation. However, manual actuation, as shown in Figure 7.2-2C, relates only to the bypass of a trip comparator when permissive conditions exist, and is not related to the manual trip comparator (i.e., placing the output of a trip comparator in trip, regardless of the relationship between the analog signal input and the setpoint), or the manual trip of the Reactor Shutdown System (i.e., Scram). Manual Trip of the Comparator and Manual Trip of the Reactor Shutdown System are effective at all times.

As noted on revised Figure 7.2-2C,flux electronics means electronic signal conditioning equipment needed to convert the output of the Fission Detector to a standard PPS input signal.

Question 222.63 (7.2-2E)

It appears from Figure 7.2-2E that a failure in a test switch with permissive outputs may prevent tripping of the corresponding primary or intermediate pump. Provide an analysis of such an event.

Response:

The bypass test functions will be designed to conform to IEEE 279-1971 Criteria for Protection Systems for Nuclear Power Generating Stations. The system design approach is to preclude undetectable failures. Should undetectable failures be unavoidable they will be combined with a single active failure and the combination will not result in both Plant Protection System breakers being bypassed, assuring a trip of the corresponding pump. The analysis to provide this assurance has not been performed to date. During vendor design of the bypass test function the vendor will complete the analysis required to assure meeting the criterion as stated above.

Question 222.64 (Q 222.5)

Your response to Question Q222.5-1 provided in revised Section 3.10.1 and revised Table 3.2-3 is not complete. We requested that you "Identify specifically and provide a listing of all safety related equipment and structures that will be seismically qualified as Seismic Category I." To the extent possible at this stage of the design, provide a complete response to this request for additional documentation.

Response:

A listing of all safety-related electrical equipment which is Seismic Category I is provided in Reference 13 of PSAR Section 1.6. All Seismic Category I mechanical equipment is listed in Table 3.2-2.

Question 222.65(Q222.9:15.1.3-1)

For each trip equation presented in Table 15.1.3-1, provide its derivation with a discussion of the assumptions made in such derivation. Furthermore, for each primary and secondary PPS trip and alarm function, and for each ESF trip and alarm function we consider important that the trip setpoint is within a practical and safe range of the associated instrumentation span so that the effects of instrument inaccuracies and drift are kept within an acceptable margin. Discuss how your criteria for the selection of instrument designs and settings which initiate automatic protective actions and alarms conform to the following recommended criteria:

(1) The range selection for instrumentation shall be such as to exceed the expected range of the process variable being monitored.

(2) The accuracy of all the safety trip points will not be numerically larger than the accuracy that was assumed in the accident analysis.

(3) The trip setpoints should be located in that portion of the instrument's range which is most accurate and must be located in a region with the required accuracy.

(4) All safety trip points will be chosen to allow for the normal expected instrument system setpoint drift such that the Technical Specification limit will not be exceeded.

(5) Verification of the above criteria shall be demonstrated as a part of the qualification test program required by IEEE Std 323-1974.

Response:

The Primary and Secondary Reactor Shutdown System protective functions are listed in PSAR Table 7.2-1 and are described and shown in block diagram form in the Figures supporting Section 7.2.1.2. These block diagrams form the basis for modeling the performance of the RSS protective subsystems. Conservative errors for instrumentation repeatability and response time are assumed as listed in Table 7.2-3 (Essential Performance Requirements for PPS Instrumentation). Additional comparator and logic time delays are also assumed. The conservative model of RSS performance is based on assuming that all repeatability errors and time response delays accumulate in the direction to delay a trip when required. The purpose of the analysis of Section 15 is to show that the conservative analytical model assumed for PPS performance acceptably limits the consequences of potential challenges to the core or to radioactive release limits. Information about PPS trip function accuracy, repeatability, time response, instrument accuracy, safety trip points and technical specification limits will be included in the FSAR to show that the performance model used for analysis in Section 15 is indeed conservative.

The trip levels and equations shown in Table 15.1.3-1 represent a conservative analytical model of Plant Protection System performance in response to the accidents analyzed in Section 15. However, Table 15.1.3-1 should not be used to infer hardware details about the real Plant Protection System (which is described in Sections 7.1 and 7.2) as Table 15.1.3-1 only represents a conservative mathematical model of Plant Protection System Performance. To facilitate proper interpretation of Table 15.1.3-1, a more explanatory introduction to Section 15.1.3 has been provided and the table title clarified.

For each of the trip equations presented in Table 15.1.3-1, the derivation was based on using the worst case performance of the equipment comprising the subsystem. An example of this derivation process is presented below for the flux-flow subsystem.

FLUX-TOTAL FLOW SUBSYSTEM TRIP SETPOINT ANALYSIS

Based on the block diagram in figure Q222.65-1, the nominal steady state equation for the performance of this subsystem is

$KF_T + \alpha - \phi \leq 0$ $\qquad \alpha$ = set point

i.e. when the quantity on the left hand side of the inequality is less than or equal to zero, a trip is initiated.

To ensure appropriate conservatism in the analysis of the performance of the subsystem, the variations in the performance of the components must be accounted for. This is done by specifying errors representing worst case performance of the component in question. The resulting equation is

$$K[F_T(1 \pm \Sigma_1)(1 \pm \Sigma_2) \pm \Sigma_6](1 \pm \Sigma_3) - [\phi(1 \pm \Sigma_4) \pm \Sigma_5] \pm \Sigma_c + \alpha \leq 0$$

where Σ_1 = 2% Flow Error

Σ_2 = 1% Flow Summation Error and Error in Gain

Σ_3 = 0.75% Summation Error

Σ_4 = 1% Flux Error

Σ_5 = 0.5% Flux Offset Error

Σ_6 = 1% Flow Offset Error

Σ_c = 0.75% Comparator Error

Note that errors which are not linearly related to the signal level are included. These errors are taken from Table 7.2-3 of the PSAR and represent the worst case performance of the instruments. (The comparator and summation errors were chosen based on FFTF experience.)

Values of K and α were determined to assure that trip occurred at the necessary value for terminating events covered by this subsystem and to provide adequate operating margin. The derivation involved definition of constraining inequalities which included the equipment errors for both safety and operational objectives of the system. These inequalities were solved yielding nominal values of

$$\alpha = 0.064; \quad K = 1.153$$

The constraints were based on assuring trip at a flux to flow ratio of 1.3 for 100% initial conditions which preliminary analyses showed would acceptably limit the results of the design basis events.

Using these values of K and α, the nominal trip equation is

$$1.153F_T + 0.064 - \phi \leq 0$$

and the trip equation which conservatively envelopes the equipment performance errors is

$$K\left[F_T(1+0.02)\ (1+0.01)+0.01\right](1+0.0075)-\left[\phi(1-0.01)-0.005\right]+0.0075+\alpha \leq 0$$

$$1.038KF_T+0.01-0.99\phi+0.005+0.0075+\alpha \leq 0$$

$$1.038KF_T-0.99\phi+\alpha+0.0225\leq 0$$

$$(1.038)\ (1.153) - 0.99\phi+0.064+0.0225\leq 0$$

$$1.2F_T-0.99\phi+0.087\leq 0 \quad \text{(Table 15.1.3-1)}$$

This ensures that the accuracy of the performance of the equipment will be better than that assumed in the safety analysis. Further, the safety analysis trip point does account for setpoint drift. The errors chosen account for the entire range which the instrumentation must provide protection. Note that the instrumentation is required to have the requisite accuracy over range which exceeds the trip value and further that no foldover is allowed for a range of input signals beyond this.

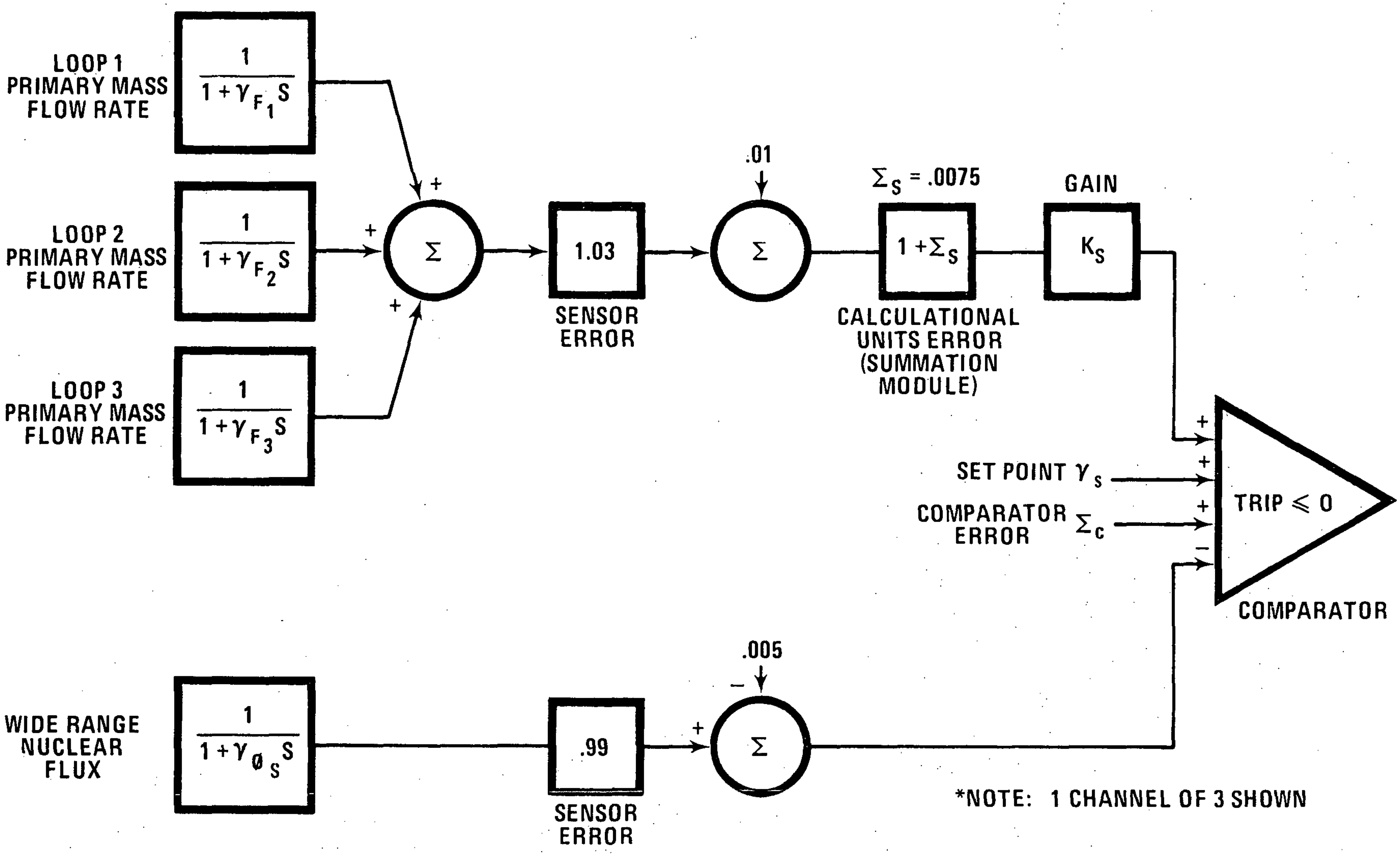
A similar derivation was performed for each of the subsystems using appropriate constraints. The nominal values for K and α are presented in Table Q222.65-1 for each subsystem. Using these values and the maximum errors presented in Table 7.2-3, the trip equation of each of the subsystems used for safety analysis was derived.

TABLE 222.65-1

PPS SUBSYSTEM NOMINAL CONDITIONS

High Flux	α = 1.1312
Flux-Delayed Flux	Positive: time constant = 28 sec K = 1.02 $\alpha = 1.05(0.1625N_p+0.0275)$ Negative: time constant = 28 sec K = 1.0 $\alpha = 1.05(0.1875N_p+0.0325)$
Flux-Total Flow	K = 1.153 α = 0.064
Primary Pump Electrics	α = 75% of Rated Bus Voltage
Primary to Intermediate Speed Ratio	α = 0.042 K = 0.147
Primary to Intermediate Flow Ratio	α = 0.042 K = 0.147
Steam-Feedwater Flow Ratio	at 100% power, trip at 30% mismatch
Flux - $\sqrt{\text{Pressure}}$	α = 0.0425 K = 1.3175
Reactor Vessel Level	8.1" above supressor plate
IHX Primary Outlet Temperature	$\alpha = 786^{o}F$
Steam Drum Level	± 8" from full power steady state level
High Evaporator Outlet Temperature	$\alpha = 704^{o}F$
Sodium-Water Reaction	3 seconds
Startup Nuclear	10% Power

FLUX - TOTAL FLOW SUBSYSTEM*



Q222.65-5

Amend. 24
July 1976

6-76-P0639-1

FIGURE Q222.65-1

Question 222.66 (7.2.1.2, RSP)

With regard to the HTS pump motors disengagement from the offsite power system on underfrequency to assure adequate flow coastdown, provide additional information substantiating that Branch Technical Position EICSB 15, Appendix 7A of Standard Review Plan is complied with.

Response:

Branch Technical Position EICSB 15, Appendix 7A, of Standard Review Plan will be complied with in the following manner following receipt from the Plant Protection System (PPS) of the undervoltage sensing and trip signal generation. To insure that the sodium pump/drive kinetic energy will fall within the range required to satisfy the flow coastdown requirements after a PPS trip, the PPS undervoltage sensors, instrument channels, and HTS pump breakers will be qualified to Class 1E IEEE Standards 279 and 379 as supplemented by Regulatory Guide 1.53, and IEEE Standards 323 and 344 as supplemented by Regulatory Guide 1.89. Separation of the sensors, instrument channel wiring, and breakers will be in accordance with the requirements of Class 1E IEEE Standard 384 and Regulatory Guide 1.75. The PPS breakers will be installed between the motor generators and HTS pump motors and will trip on PPS scram signals. The breakers will be located in the Seismic Category I Diesel Generator Building.

As stated in Section 7.2.1.2 of the PSAR, these undervoltage relays are used to detect loss of power on each HTS pump bus. Undervoltage relays are used since large changes in voltage result from power system perturbations; whereas, frequency on a 60Hz power system may remain essentially at 60Hz throughout the perturbation. Readily detectable changes on HTS Pump Bus voltage always occur immediately upon loss of HTS Pump Bus Power as the HTS Pump Bus voltage goes to zero.

Question 222.67 (7.2.1.1, Q222.13)

(a) Your statements in Section 7.2.1.1 on the primary and secondary shutdown systems, and your response to our Question 222.13 indicate the philosophy adopted in the design of these two systems did not intend to achieve full redundancy between the two systems. The resulting design, as you state, is comprised of a Primary Shutdown System and a Secondary Shutdown System, the latter covering only a part of the spectrum of fault events for which protective action is required. We consider redundancy as one of the most important and basic design features for achieving the high functional reliability required by NRC Criterion 19. The design of the Reactor Shutdown System as presently proposed appears to be lacking in this respect. Explain in detail your rationale in the design of the Reactor Shutdown System. More specifically, explain your basis for designing the Secondary Shutdown System so that it does not provide a protective action for the entire spectrum of fault events for which such action is required, and which action is provided only by the Primary Shutdown System.

(b) Justify your assumptions that (1) the only CDA mechanisms of any import are those that may be initiated by an ATWS, (2) all such ATWS are precluded by RSS protective actions, (3) all possible anticipated and unlikely fault events that may result in a CDA have been considered, and (4) the "Extremely Unlikely Events" which could result in a CDA need only be protected against by the Primary Shutdown System. Moreover, for each parameter listed in Table 7.2-1 for the Primary Shutdown System, provide your justification of the claimed functional diversity for its counterpart listed in the same table for the Secondary Shutdown System.

Response:

Part (a)

The importance of redundancy as a basic design feature in the context of CRBRP Criterion 21 (identical with NRC Criterion 29) is fully recognized and addressed in the response to that Criterion in Section 3.1 of the PSAR.

The degree of internal redundancy within the Primary System, as discussed in PSAR Section 7.2, is such that it meets the requirements of CRBRP Criterion 21 without invoking a need for the Secondary System. This redundancy includes the following features, as illustrated in Figures 7.2.2 and the figures immediately following it:

- Figure 7.2-2A shows that any two out of the three sensors in any channel can provide a trip signal to each of three logic trains.
- Figure 7.2-2B shows how the logic train outputs are combined, such that holding power is cut off from the primary rods. Specifically, the combination of one trip signal from each of two logic trains will cause a trip.
- The same Figure shows that logic train #1 and #3 each supply two signals to the scram breaker arrangement. The scram circuitry, as shown in Figure 7.2-2B, is such that only one such signal is required.

- Figure 7.2-3 shows the channel test arrangements. It should be noted that on-line test capability is provided, with the channel on test being placed in the trip condition. This includes the capability to test channels independently.

The degree of diversity between the primary and secondary systems is discussed in Section 7.2 of the PSAR. Briefly summarized, this comprises:

- Diverse trip parameters, where practicable. Diverse sensors where the same parameters are used.
- Diverse logic (as seen from Figure 7.2-2B and 7.2-2D).
- Use of optical coupling in Primary versus magnetic amplifiers in Secondary.
- Use of different equipment to initiate permissive bypasses.
- Use of different means of testing.
- Use of different hardware in the various circuit elements.

The Secondary System is thus seen in its true perspective as an additional feature to provide yet further enhanced redundancy, and also provides a measure of diversity to reduce the potential for common mode failure. It is totally redundant to the Primary System for all Anticipated and Unlikely Faults. It is a design requirement that the Secondary System operate to terminate a postulated reactivity transient assumed to result from the SSE. This requirement is shown in Section 4.2, Figure 4.2-93. The results of analysis for this event are provided in response to question 001.313. While this is the only Extremely Unlikely Event which is included in the design basis of the Secondary Shutdown System, capability of the Secondary System to successfully protect against each of the other Reactivity Insertion Design Events and Undercooling Events in the Extremely Unlikely category has been analyzed and is described in Chapter 15, Table 15.2-1 and Table 15.3-1.

Part (b)

(1) There is no assumption that the only mechanisms of any import are those that may be initiated by an ATWS other CDA initiators are discussed in the PSAR, Appendix C, Section C.1.3.2 and C.1.4 and Section F.3.

(2) A discussion of all ATWS events is presently in the PSAR. Chapter 15 provides a discussion of anticipated events considered in combination with hypothesized total failure of both scram systems. The consequences of such events, for the expected condition in which the Plant Protection System operates, are shown in Chapter 15, Section 15.2 and 15.3. The discussions in those sections include consideration of Plant Protection System action and demonstrate the efficacy of such action. From the results of those analyses, it can be concluded that successful RSS protective actions preclude an anticipated transient leading to loss of coolable core geometry, assuming successful operation of the decay heat removal systems following reactor shutdown. Section F.3.2 (Parallel Design) provides a discussion of candidate Design Basis Accidents for the Parallel Design and describes those events which are beyond the Plant Protection System design basis, but merit consideration as potential initiators.

(3) Unlikely Faults will not result in a CDA, since the action of the Plant Protection System terminates such events (see Sections 15.2 and 15.3 of the PSAR).

Anticipated Faults have been addressed under (2) above. The only known mechanisms for causing core overheating from a steady state operating condition are those which can result in an imbalance between power production rate and heat removal rate. These are:

Overpower at constant flow
Flow reduction at constant power
Combination of overpower and flow reduction
Fuel assembly coolant channel voiding
Local assembly failure events

Each of these conditions has been examined, and is discussed in Chapter 15 of the PSAR.

(4) The rationale for requiring only the Primary Shutdown System to provide protection against Extremely Unlikely Faults is that the combination of the low probability of the Fault itself and the very low probability of Primary Shutdown System Failure is such that this postulate is not considered appropriate for consideration in the Design Base. The response to Q222.67 (a) provides further discussion of the aspects of the primary and secondary shutdown systems related to extremely unlikely events.

(5) There is no direct correspondence in Table 7.2-1 of Primary to Secondary subsystem diversity. Subsystem diversity must be considered for the specific design basis fault events. Where Primary-Intermediate Speed Ratio and Primary-Intermediate Flow Ratio are the diverse protective functions for loss of a single pump, Primary Pump Electrics and Primary-Intermediate Flow ratio sense loss of 1 or 2 HTS loops for the Primary and Secondary RSS. For loss of 3 HTS loops (which is also loss of preferred and alternate preferred power), Primary Pump Electrics and Flux-Total Flow are the diverse Primary and Secondary protective functions. For some fault events, the diversity between primary and secondary is very obvious. For example, many steam side events are sensed by Steam-Feedwater Flow Mismatch in the Primary RSS and by Steam Drum Level in the Secondary RSS. For other events, the subsystem diversity is not as obvious. For example, on first inspection, IHX Primary Outlet Temperature and Evaporator Outlet Sodium Temperature might seem non-diverse. However, the first measures primary coolant temperature, and the second measures intermediate coolant temperature, which are clearly independent measures of different coolant temperatures.

For many reactivity disturbances, Flux-Pressure and Flux-Total Flow are the diverse subsystems used as are Flux-Delayed Flux and Modified Nuclear Rate. While Reactor Inlet Plenum Pressure and Total Sodium Flow are clearly diverse, Flux does not seem to be. Recognize, however, that flux is the only rapidly responding plant parameter indicative of reactor power, and that for reactivity disturbances, the speed of response of the Plant Protective System is critical.

Therefore, the base protection of the Primary and Secondary RSS against reactivity disturbances are measurements of nuclear flux. In this case, equipment diversity is utilized by employing compensated ion chambers to measure nuclear flux in the Primary RSS and fission chambers in the Secondary RSS as described in PSAR Section 7.5-2.

Question 222.68 (9.3-3)

On Figure 9.3-3 you indicate that certain instrumentation systems are supplied by "System XX." Provide an identifying list of all systems numbered in this fashion for the entire plant. On this list provide the following information for each system:

(a) scope of supply;

(b) safety classification; and,

(c) seismic classification.

Response:

"System XX" notation used internally by the Project to simplify documentation was used in the drawings for Section 9.3. A listing of the numbered systems is provided in Table 9.5-2. All plant systems of the CRBRP are discussed in the PSAR to the depth necessary to demonstrate that the plant design (including safety classifications and seismic categories) satisfactorily assures that there is no undue risk to the health and safety of the public.

Question 222.69 (9.9)

On page 9.9-7 you state that one of the design bases for the Emergency Plant Service Water System is to provide sufficient cooling water to permit safe shutdown, and the maintenance of the safe shutdown condition in the event of an accident resulting in the loss of the Normal Plant Service Water System or the loss of both plant a-c power supplies and all offsite a-c power supplies.

List all safety related systems for which the total loss of a-c power is a design basis event. For each safety related system for which the total loss of a-c power is not a design basis event, provide your justification for the lack of such a design basis.

Response:

There is no safety related system for which the total loss of a-c power is a design basis event. The loss of plant AC power supply (plant AC power supply is defined in Section 8.1.2 (1)) and the loss of all offsite AC power supplies (offsite AC power supply is defined in Section 8.1.2 (2 & 3)), results in the automatic actuation to the Standby (onsite) AC Power Supply which consists of two redundant and independent diesel generators as defined in Section 8.1.2 (4). The Emergency Plant Service Water System is supplied with Standby (onsite) AC Power supply in the event of the loss of both plant AC power supply and all offsite AC power supplies.

Question 222.70 (7.2.1.2, 7.3.2.1)

With regard to the environmental qualification requirements of safety related equipment, you state in Section 7.2.1.2.3 that the environmental extremes are being established and will be supplied at a later date. Provide a specific schedule for the submittal of this information.

Response:

A complete discussion of the CRBRP environmental qualification program is provided in Reference 13 of PSAR Section 1.6.

Question 222.71 (7.3.1.1) | 27

Provide the following additional information on the Containment Isolation System (CIS):

(1) A list of valves that are not required to close in less than ten minutes with a justification of the closure time requirement for each case;

(2) A clarification as to whether or not manual initiation is provided for all isolation valves, and whether or not manual initiation only is provided for those valves not required to close in less than ten minutes. State your rationale for providing automatic and/or manual initiation of the CIS. It should be noted that manual initiation at the equipment level is not acceptable. We will require manual initiation at the system level per Section 4.17 or IEEE std 279-1971. Specify the point in time at which you begin counting time for containment isolation purposes.

(3) Identification of the power sources used to energize the CIS.

(4) A table showing the type and range of the radiation detectors and related instrumentation channels used in the CIS.

(5) A P&ID for the CIS showing all isolation valves.

Response:

(1) Each Normal and Emergency Chilled Water line penetrating the containment is provided with a containment isolation valve. Since the Chilled Water Systems are closed systems, their valves are not required to automatically close upon a CIS signal. If a line rupture within containment can not be isolated by closing other remotely operable system valves inside containment, then the affected Chilled Water Supply containment isolation valves would be closed.

For the Normal and Emergency Chilled Water lines, each containment isolation valve is provided with remote manual operation from either the Control Room or the appropriate local control panel. In addition, each valve may be operated manually at the valve. Leakage from any Chilled Water System pipe inside containment will result in leak detectors alarming the Control Room. Upon confirmation of a leak and determination that the leak cannot be remotely isolated inside containment, manual or remote manual action is taken to initiate closure of the containment isolation valves affected. The Chilled Water Containment isolation valves will close in less than 30 seconds after initiation.

The DHRS Nak piping and heat exchanger inside containment comprise a closed system. In addition, the EVST Nak piping and ABHXs comprise an essentially closed system outside containment. The two lines penetrating containment (Table 6.2-5, items 20 and 21), are split at tees outside containment, and each of the four resulting lines is provided with a remote manually operated containment isolation valve. These valves are not required to automatically close because the valves are normally closed (open only for inservice testing and maintenance) and when opened, must remain so to provide the DHRS safety function. This satisfies CRBRP Design Criteria 45 and 48.

The Sodium Transfer Lines (Table 6.2-5, items 6 and 7) are safety class lines separated from the reactor coolant boundary by two locked closed isolation valves and thus comprise closed systems inside containment. Each line has a locked closed manually operated isolation valve outside containment. The valves will remain locked closed except for very infrequent maintenance operations requiring drainage of an EVST sodium loop or multiple PHTS loops. This satisfies CRBRP Design Criteria 45 and 48.

Each Containment Vacuum Breaker Line (Table 6.2-5, item 3) would be provided two isolation valves (one inside, one outside) which would react directly to a negative pressure across the containment boundary. Otherwise, the valves would be opened (remote manually) only for periodic operability testing. Due to the extremely infrequent operation of the valves, remote manual operation of the valves is considered adequate to satisfy the intent of CRBRP Design Criteria 45 and 47.

(2) Remote manual initiation is provided in the control room for all containment isolation valves except the sodium transfer lines which are provided with (locked closed) manually operated valves (see Table 6.2-5). Automatic isolation capability is provided vis-a-vis remote manual or manual isolation capability in accordance with CRBRP Design Criteria 45 through 48. In accord with IEEE Standard 279-1971, capability for manual initiation of the CIS from the control is provided. For analyses of the Containment Isolation System (detection and initiation equipment), time zero will be initiation of the event under consideration. For analysis of the individual valves (see Table 6.2-5), time zero will be when the signal to close is received by the valve operator.

(3) The three instrument channels of the Containment Isolation System, which include radiation detectors, signal conditioning and comparators are energized from the three independent 120 V AC vital busses. The Containment Isolation System valve actuators are energized from 2 of the 3 vital DC busses (from two of three station batteries), one bus for the inside containment valve actuators, and one bus for the outside containment valve actuators.

(4) A table showing the type and range of radiation detectors and associated instrument channels used in the CIS will be provided in September 1976.

(5) The containment isolation valves for each line penetrating containment will be shown on the P&ID for the system in which the containment isolation valve is included. Note that in the response to Question 001.182, the second column of Table 6.2-5 identifies the section of the PSAR where specific information about the systems which penetrate containment may be found. Where the design has evolved sufficiently to the hardware state, the containment isolation valves are shown in figures in the referenced PSAR sections. For example, the isolation valves for the Containment Ventilation Air Supply and Exhaust Lines
49| (the first two items in Table 6.2-5) are shown in Figure 9.6-4. Likewise, the isolation valves for the Dual Compressed Gas Supplies to Containment (items 4 and 5 in Table 6.2-5) are shown in Figure 9.10-1.

Question 222.72 (7.2)

Describe your design provisions for two-loop operation with respect to more restrictive PPS trip settings. We refer you to Branch Technical Position EICSB 12 of Appendix 7A of the Standard Review Plan. We will require compliance to this position as means for meeting the requirements of Section 4.15 of IEEE Std 279-1971.

Response:

Compliance with BTP EICSB 12 is discussed in revised Section 7.2.2. 31

Question 222.73 (7.4.1.1, RSP)

You state in Section 7.4.1.1.4 that "Control interlocks associated with the operation of active components have not been completely defined" for the Steam Generator Auxiliary Heat Removal System (SGAHRS). Specify your time schedule for completing the design of this system.

It should be noted that your design should provide a diversity of motive and control power so that both trains of the SGAHRS will not depend on one type of power. An example of an acceptable auxiliary feedwater system would be to have one train relying on DC and steam and another train relying on AC only, either one of which can provide the required flow.

Identify the power sources used to provide motive and control power to SGAHRS.

Response:

SGAHRS Instrumentation and Control System design of interlocks associated with the operation of active components is complete. PSAR Section 7.4.1.1.4 includes a functional description of required SGAHRS control interlocks.

With regard to the request to identify the power sources used to provide motive and control power to SGAHRS, refer to NRC Question 020.11 and its response.

Compliance to the applicable portions of BTP APCSB No. 10-1 is met with the following design:

In order to meet the requirement of an auxiliary feedwater system to consist of at least two full capacity independent systems, including diverse power sources, SGAHRS is designed with two 50% capacity motor-driven pumps, each connected to a separate diesel, and one 100% steam turbine-driven pump. Thus, there is 200% total capacity with half of it electrically powered and half steam powered. There are also two parallel auxiliary feedwater paths to each drum, one from the turbine-driven pump through a control valve and one from the headered motor-driven pumps through another control valve. The drum level setpoints are set such that flow normally is supplied from the motor-driven pumps only with the turbine pump recirculating. If the drum level falls below the turbine pump setpoint, the control valve downstream of that pump opens and begins supplying flow.

System "A" and System "B" switchgears provide power to the respective System "A" and System "B" motor driven SGAHRS pumps. The control power for the System "A" and System "B" switchgears is provided from the System "A" and System "B" battery supply as identified on Tables 8.3-2A and 8.3-2B. The control power required for the 100% turbine driven SGAHRS pump is supplied by Battery System "C" as identified in D.C. battery loading table 8.3-2C. This battery system design will permit auxiliary feedwater subsystem operation with loss of all AC power.

Question 222.74 (7.4.2)

Provide a Functional Control Diagram and a P&ID for the Outlet Steam Isolation Subsystem (OSIS). Identify the power sources supplying control and motive power to the OSIS.

Response:

The OSIS functional control is discussed in Section 7.4.2 and is illustrated as part of PSAR Figure 7.5-6 (Sh. 5). The isolation valve location in the Steam Generator System is shown on PSAR Figure 5.1-4, "Steam Generator Schematic Flow Diagram". The power source for this system is Class IE instrument power, both for control and motive power.

Question 222.75 (7.5.5.1)

Provide the design bases and design criteria for the Sodium to Gas Leak Detection System even though its preliminary design is not expected until 1977.

Response:

The information requested is provided in revised Section 7.5.5.1.

Question 222.76 (7.6.1.1, 7.6.3.1, 7.5.6.1, Q222.14)

Provide the design bases, design criteria, P&ID's and Functional Control Diagrams for the Treated Water Instrumentation and Control System, the Overflow Heat Removal Service (OHRS) System, and the Sodium Water Reaction Pressure Relief System (SWRPRS). Provide the same information for the systems listed in Table 7.1-1 under "Other Safety-Related Instrumentation and Control."

Response:

The design bases for the chilled water systems and service water systems are provided in PSAR Sections 9.7 and 9.9, respectively. Those bases apply to the instrumentation and control portions of the systems as well as the other portions. The Design Bases of the Instrumentation and Control for the emergency chilled water and Emergency Plant Service Water Systems, are discussed in revised PSAR Sections 7.6.1 and 7.6.2.

The SWRPRS design basis events are identified in section 15.3.3.3, "Large Sodium Water Reaction." Detail on the instrumentation and control functions to meet these design basis events is provided in PSAR Section 7.5.6. The design basis for the SWRPRS, including the instrumentation and control equipment, is to protect the plant by tripping the reactor and mitigating the pressure pulse in the IHTS for reactions enveloped by a three tube leak in any steam generator module.

PSAR Figures 5.1-4 and 7.5-6 provide P&ID and Functional Control information for the SWRPRS.

The design criteria for the SWRPRS I&C equipment is as follows:

To monitor the SWRPRS reaction products vent line and initiate plant shutdown upon detection of reaction products.

Initiate activation of and monitor the SWRPRS equipment required to mitigate the consequences of a large sodium/water reaction.

The SWRPRS I&C equipment necessary to initiate plant shutdown and initiate activation of SWRPRS equipment upon detection of a significant sodium/water reaction shall be designed and qualified in accordance with the appropriate requirements of Section 7.1. This includes the monitoring equipment necessary to detect events requiring plant shutdown and SWRPRS operation.

As discussed in PSAR Section 5.6.2, the design basis for the DHRS (OHRS) is to provide a single safety class decay heat removal train, capable of being initiated manually from the Control Room and from a local control station.

The bases for the design of the DHRS instrumentation and control is to monitor the operation of the DHRS and provide the control functions as required to meet the following DHRS design criteria and requirements.

a. Provide the capability for manual initiation from the Control Room and a local control station.
b. Provide sufficient information to enable the operator to verify that the DHRS is performing its function of removing heat from the reactor.
c. Provide sufficient instrumentation to allow the operator(s) to monitor and control (manually) the temperatures of the sodium and NaK loops to maintain the system within operating limits.
d. The I&C equipment necessary to assure that DHRS will carry out its decay heat removal function shall be designed and qualified in accordance with the appropriate requirements of Section 7.1.

Sufficient DHRS instrumentation will be provided to enable the operator(s) to verify that the Auxiliary Liquid Metal System valves are appropriately aligned, pumps and fans operating and heat is being transported to and removed from the ABHX's. Instrumentation and control capability will be provided to allow the operator(s) to monitor and control (manually) the temperatures of the sodium and NaK loops to maintain the system within the operating limits. The control for the containment isolation valves in the NaK subsystem will meet the design bases for the CIS provided in Section 7.3.1.2.

The design bases for the Heating, Ventilating and Air Conditioning (HVAC) System, instrumentation and control, are discussed in revised PSAR Section 7.6.4.

The Design Bases of the Instrumentation and Control for the Recirculating Gas Cooling System, are discussed in revised PSAR Section 7.6.6.

Question 222.77 (7.8.1, 7.5-1) (RSP)

Provide a Table similar to Table 7.5-1 listing all instrumentation intended for accident and post-accident monitoring of plant parameters necessary to follow the course of an accident and to achieve a safe shutdown. Branch Technical Position EICSB 23 of Appendix 7A of the Standard Review Plan provides guidance and requirements for the qualification of this type of instrumentation. We will require compliance to these requirements. Moreover, you should provide a discussion outlining and justifying your rationale for the selection of parameters for this group of monitoring instrumentation.

Response:

Section 7.5.10 provides a discussion of instrumentation provided to enable
the plant operator to assure that the plant is maintained in a safe
shutdown status. Table 7.5-4 lists the parameters monitored to perform
this function. Compliance to the qualification requirements of EICSB 23
50 will be carried out as outlined in the response to Question 222.54.

Question 222.78 (7.9.1.5)

From your description and design evaluation of your provisions for a safe shutdown outside the control room, it appears that your design does not meet the requirements of NRC Criterion 17. Discuss in some detail how your present design, and any additions or changes you may propose, meet the requirements of NRC Criterion 17.

Response:

The equipment provided at appropriate locations outside of the control room to shut the plant down and maintain it in a safe shutdown condition is in compliance with CRBRP General Design Criterion 19 (or alternately 10CFR50 Appendix A Criterion 19-Control Room) as identified in Section 3.1.3. This is discussed in Section 7.9.5 of the PSAR which has been amplified in response to this question.

More detail concerning specific locations of equipment outside the control room, and the specific instrumentation necessary to assure the plant is maintained in a safe shutdown condition will be provided in the FSAR.

Question 222.79

You state in Section 4.4.5 that "a document presenting the rationale for the instrumentation selection is being developed. This document should be available by October, 1975". We have not received this document yet. We will need this information to perform our review of the reactor in-vessel instrumentation.

Response:

The document presenting the rationale for the instrumentation selection has been prepared and has been incorporated into new Section 4.4.5 of the PSAR.

Question 222.80 (7.2-7.6)

Your response to our Question 222.10 is not adequate. Although your response provides a detailed reference of pertinent subsections where our concerns are discussed, mentioned or alluded to, we need a discussion for each system described in Sections 7.2-7.6 of how it meets the requirements of the applicable NRC criteria, Regulatory Guides, and, on a section-by-section basis, of IEEE Std. 279-1971.

Response:

Chapter 7 of the PSAR conforms to the Standard Format and Content Document and contains available information on how the instrumentation and control systems meet the requirements from the CRBRP General Design Criteria, NRC Criteria, Regulatory Guides and IEEE Standards. In addition, the response to Question 222.10 details the requirements of IEEE 279-1971 and references pertinent subsections of the PSAR where these requirements are discussed. The information currently in the PSAR is sufficient to show compliance with design requirements.

Question 222.81 (8.2.1)

Revise Figures 8.2-2 through 8.2-6 to show interconnections of the two switchyards to the onsite 13.8-kV buses showing all breakers and any interconnections between the buses.

Response:

Figures 8.2-2 through 8.2-6 have been revised to show all breakers and methods of supplying 13.8-kV auxiliary buses 1N through 9N and ESF buses 1E and 2E. As indicated on these drawings, the auxiliary buses are served simultaneously from one source at a time. The source supply is either the unit station service transformer or one of the reserve station service transformers. The selections available for supplying these auxiliary boards are described in Section 8.3.1.1.4.

Question 222.82 (8.2.1)

You state in Section 8.2.1 that "results of transient stability studies show that the Reserve AC power supply remains a reliable source. . ." However, you do not comment on the results of the analysis on the Preferred AC Power Supply. Provide the results of your analysis pertaining to the Preferred AC Power Supply.

Response:

The Reserve AC power supply provides the two physically separate and independent circuits of the IEEE Standard 308-1974 (preferred power supply) and is in compliance with General Design Criterion 17 as discussed in Section 3.1. The auxiliary buses are connected to the Reserve switchyard through two independent transformers, and the Reserve switchyard is connected to the TVA grid with two physically separate and independent source lines as required by General Design Criterion 17.

The Preferred AC Power Supply (two transmission lines and the CRBRP main generator step-up transformer connected to the CRBRP generating switchyard) is not required to comply with NRC and IEEE criteria for electrical faults on the generator side of the load break switch.

Results of steady state and transient stability studies show that one of the two physically separate and independent sources provided by the Reserve AC power supply remains a reliable source to supply the onsite electric power system for single contingency cases including loss of the CRBRP unit, Preferred AC Power Supply, loss of a critical 161-kV transmission line, or loss of the largest generating unit on the system.

Question 222.83 (8.2.2)

Discuss in Section 8.2.2 the consequences of a turbine trip with the presence of an electrical fault in the generator side of the step-up transformer, or an electrical fault in the offsite (161 KV) lines.

Response:

The response to this question is contained in revised Section 8.2.2.

Question 222.84 (8.2.2.1, Q222.19)

Your response to our Question 222.19 needs clarification with respect to the immediate vs. delayed access provisions of the Preferred and Reserve AC Power Systems. You state on page 8.2-5 under "Safety Guide 1.32" that each supply is connected automatically when required. On the next paragraph you state that if there is a loss of the immediate access circuit, then the delayed access circuit will be available in sufficient time. The two statements appear to be in conflict in that the first one seems to imply that the Reserve AC Power Supply can provide power on an immediate access basis. Provide a clarified discussion of your compliance to Regulatory Guide 1.32. Specify what "sufficient time" is for the availability of a delayed access circuit, and what this circuit may be.

Response:

The Reserve AC Power Supply is available automatically in the event of the failure of the immediate-access circuit. The discussion for Regulatory Guide 1.32 in Section 8.2.2.1 is modified to explain how the Reserve AC Power Supply can provide power essentially on an immediate access basis.

Question 222.85 (Q222.23, 8.3)

In your response to our Question 222.23 you address our concern relating to reliability of the diesel generator sets to start and accept load only with respect to the effect of this reliability on the SGAHRS auxiliary feedwater pumps. Address the same concern related to the pony motors.

Response:

The Shutdown Heat Removal System Reliability Assessment (Ref. 6 of Section 1.6) indicates that the random failure probability of a single pump (independant of power supply)is 2.4×10^{-5} per hour; the probability of failure of the offsite power supply and one diesel generator (probability of a single motor power supply) is 1.8×10^{-9} per hour; and the probability of loss of offsite power and both diesel generators is 1.8×10^{-11} per hour.

Many CRBRP design basis events result in decay heat removal through all three HTS loops. In such a case, the three PHTS pony motors redundantly perform the same safety function. In this case, when the (3) pony motors are viewed as redundant components, the combined probability of failure of all 3 motors is dominated by the probability of loss of all AC power.

Question 222.86 (8.3-2C)

In Table 8.3-2C you list the 250V DC system loads. Identify the redundant counterparts of these loads and the sources from which they are powered.

Response:

In order to implement the recommendations of Regulatory Guide 1.75, three (3) - 125 volt DC batteries will be installed to provide power supply to Class IE loads and two (2) additional nonIE 125 volt D.C. batteries will be installed to provide power supply to non-Class IE D.C. loads. PSAR Section 8.3.2 has been revised to reflect these changes. The redundant counterparts of the loads listed in Table 8.3-2C are listed in Tables 8.3-2A & 8.3-2B, except for load #8. Load #8 (SGAHRS Turbine Pump) does not have a redundant counterpart, as there is only one SGAHRS (Steam Generator Auxiliary Heat Removal System) Turbine Driven Pump. The other two SGAHRS pumps are electric driven, and are listed on Tables 8.3-1A & 8.3-1B. 16

Question 222.87 (8.3.1.1)

Provide single line diagrams showing the 13.8 kv, 480/120 V, 120 volt vital and 250/125 V DC safety related distribution systems and buses indicating to the extent possible during this stage of design all loads on each bus.

Response:

The Electrical System has been revised to provide four 4.16 kv buses. The 4.16 kv buses are indicated in revised Figure 8.3-1.

PSAR Section 8.3 has been updated to include the one line diagrams showing the loads connected to 4.16 kv buses, and the 4160/480 volt unit sub-stations (Figures 8.3-3, 8.3-4 and 8.3-5). The loads connected to the Class IE Motor Control Centers have not yet been fully developed and will be provided in the FSAR.

A single line diagram showing the 125 volt DC and 120 V AC Class IE Power System is provided in revised Figure 8.3-2.

The source of supply of the 125 volt DC Class IE batteries and 120 volt vital AC distribution buses are identified in Figure 8.3-2.

Each Nuclear Island Building is provided with two Class IE DC panels, one supplying power to one load group within that building and the second panel to the other redundant load group. The details of the loads connected to each panel will be provided in the FSAR.

Control Building and the Reactor Containment Building are provided with three vital AC panels. Each panel will be used for supplying power to equipment in that building associated with one PPS channel. The details of the loads connected to the individual panel will be provided in the FSAR. 15

Question 222.88 (8.3.1.4)

In your discussion of the physical separation criteria for cables of Class IE systems on page 8.3-21 you use the terms physical and electrical separation interchangeably. Although we believe this to be an inadvertent oversight, nevertheless it should be corrected to avoid unnecessary confusion. Review Section 8.3.1.4 and correct as necessary. See Question 222.52 for more specific requests for additional information on separation criteria.

Response:

Section 8.3.1.4 has been revised to avoid confusion between electrical and physical separation. This section has also been revised to reflect a change in cable tray fill from 30% to 40%.

Question 222.89 (Q322.5-1 Reference)

In response to our Question 322.5, you have calculated that lightning will strike the plant area once in every 2.5 years. Describe the features of your design that will provide lightning protection for the electrical, instrumentation and solid state logic systems required for safety, including computer systems.

Response:

Each of the four (4) transmission lines connected to the Generating and the Reserve Switchyards are protected from direct lightning strikes by overhead ground wires installed over the transmission towers.

For protection of the Plant Electrical System against propagation of lightning surges, lightning arrestors have been provided at the primary side of the main transformer and the reserve transformers. These lightning arrestors are coordinated with the insulation of the main transformer and the reserve transformers and the auxiliary electrical equipment downstream to reduce lightning surges to below the rated impulse level of the equipment. The margin of protection is in excess of that required by IEEE Standard 28 of 1974.

The main generator is provided with a surge protector to protect the generator against lightning surges or voltage impulses transmitted through the main transformer. The surge protector provides protection for the main Generator winding insulation by limiting the amplitude of any applied impulse waves or reflections within the machine winding. The capacitors in the surge protector provide protection against insulation failure by reducing the steepness of the wave fronts.

In addition, the plant buildings and cooling towers are provided with lightning rods mounted strategically on the top of the structures and connected with separate grounding cables to the buried plant ground grid. This arrangement provides a low resistance path to ground for lightning strikes on buildings.

The lightning protection described above considerably reduces the effects of lightning strikes to the electrical system. The safety related instrumentation and control systems are supplied by the Vital 120 VAC power supply. This 120 VAC supply is located inside a lightning protected building and is isolated from the switchyard power supply by breakers, transformers, cabling, buses, reactors, battery chargers, and inverters. The impedance and filtering provided by this equipment will essentially eliminate surge voltages resulting from lightning strikes in the switchyards.

Lightning protection requirements for the non safety-related data handling and display system will be determined by the system designers and equipment vendors.

Question 222.90 (7.1.2.5 F1-2)

Provide the information requested in our RSP 222.54 and Question 222.70 to reflect environmental conditions applicable to the parallel design. Include in your response an identification of these components required for safety which may be affected by the design basis loading referred to in Section 3.7.2.1.2 (yellow).

Response:

In Amendment 24 to the PSAR, the Project withdrew the Parallel Design from further consideration by the NRC Staff. This question requests additional design information on specific features of the Parallel Design. Therefore, the question is no longer applicable.

The environmental conditions associated with operation of the TMBDB features are included in Section 2.1.2 of CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 222.91 (F5.2.1)

Provide the information requested in our Question 222.77 regarding the accident and post-accident monitoring of plant parameters applicable to the parallel design.

Response:

This question is on design details of Parallel Design features. In Amendment 24, the Project withdrew the Parallel Design; hence the question is no longer directly applicable. However, CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6) identifies the instrumentation provided for post accident monitoring in the case of events which challenge containment integrity.

Question 222.92 (F5.3.7.3.6.2.4.3.7.3.1.1)

Provide the additional information on the CIS requested in RSP 222.71 as it applies to the parallel design. Describe the head access area instrumentation, including the sensors,that will be used to isolate the sealed head access areas from the main containment area.

Response:

In Amendment 24 to the PSAR, the Project withdrew the Parallel Design from further consideration by the NRC staff. This question requests additional design information on a specific feature of the Parallel Design Accordingly, the question is no longer applicable.

Question 222.93 (1.2.6 Yellow)

In Section 1.2.6 (yellow) you state that the Plant Control System receives signals from the Ex-Vessel Core Catcher System. Identify all such signals and describe their intended function.

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer applicable as the features upon which the question is based are no longer a part of the design.

Question 222.94

On page 3.1-15a you state for "for the design basis core disruptive accidents...the electric power systems provide power as required to assure adequate cooling of the fuel materials within the containment". Provide specific information as to what power systems support what safety related systems and associated safety functions within the scope of your above statement.

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer applicable as the features upon which the question is based are no longer a part of the design.

Question 222.95 (3.8.4.1.8)

In Section 3.8.4.1.8 you state that the electrical manholes of your onsite distribution system are Seismic Category I. Specify what systems this provision covers, and describe your flood protection features for these electrical manholes.

Response:

The Seismic Category I Electrical Manholes, as described in Section 3.8.4.1.7, are provided for supplying standby onsite power to the Emergency Cooling Towers in the yard. Flood protection features for these electrical manholes are discussed in revised Section 3.8.4.1.7.

Question 222.96 (6.2.7, 7.9.1.5, Q222.78)

Provide the information requested in our Question 222.78 with respect to the compliance of your parallel design to the General Design Criteria entitled "Control Room".

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer applicable. However, the Control Room habitability requirements for TMBDB conditions are provided in Section 2.1.2.15 of CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 222.97 (8.3-1A Yellow)

In the diesel-generator load listing of Table 8.3-1A you list a number of 4.16 and 13.8 KV loads. For the reference design, the 4.16 KV loads were listed as 13.8 KV loads. Explain the reasons for these changes and describe how you will derive a 4.16 KV supply. Such a supply is not identified in Figure 8.3-1. Identify all non-Class IE loads listed in Tables 8.3-1A and 8.3-1B.

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer directly applicable. However, the electrical power system requirements for TMBDB features are identified in Section 2.1.2.13 of CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6).

Question 222.98 (7.7.1.3)

The information provided to describe the primary and secondary control rod systems is not sufficient to assure that failures of these systems would not impair the protection system capability in any significant manner, or cause plant conditions more severe than those for which plant safety systems are designed. Provide a more detailed description, including one-line schematics and functional control diagrams showing the CRDM pulser, rod block interlocks, and all automatic and manual control functions.

Response:

PSAR Section 7.7.1.3.1 has been expanded to provide the requested information on the primary control rod system.

The secondary control rod system is discussed completely in PSAR Section 4.2.3. To summarize briefly, the secondary control rod system is required for plant shutdown only if the primary system fails. For plant operation
51 | all secondary rods must be withdrawn to their full-out position before any of the primary rods are withdrawn. During plant operation the secondary rods will remain in this full-out position in readiness for a scram demand. Proper positioning of the readiness secondary rods is assured by having two independent position indication systems. Upon receipt of a scram demand from the PPS the latch mechanism which supports the rods in place is actuated by the venting of a pneumatic cylinder, thus releasing the rod. The insertion rate of the control rod is controlled by gravity and the hydraulic pressure forces which assist scram action.

Failures of the Primary CRDM Power Train have been identified in the FMEA submitted in answer in Question 222.34. The consequence of these failures are discussed in the PSAR in Sections 15.2.2.3, 15.2.3.4 and 15.2.3.5. No significant consequences result from these failures. The same FMEA submitted with Question 222.34 identified failures associated with the secondary control rod system.

More detailed information on the final design of the Primary CRDM Controller and Power Train including one line schematics and functional control diagrams will be included in the FSAR. Detailed design information on the secondary control rod drive mechanism controller will be included in the FSAR.

Question 222.99

Provide the criteria, bases and preliminary design of the instrumentation needed to follow the course of the TLTM accident; include the rationale for not including radiation monitors inside containment.

Response:

The information requested will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Sections 2.1.2.12 and 2.2.12.

Section 2.1.2.12 gives instrumentation requirements and Section 2.2.12 describes the instrumentation. Monitors for measuring the radiation level inside containment are being added to the design.

Question 222.100

Describe the proposed environmental qualification program for those selected electrical systems and components necessary for the course of the TLTM accident.

Response:

The information requested will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Sections 2.1.2.12, 2.2.12 and 2.2.13.

Section 2.1.2.12 identifies that TMBDB instrumentation will be designed, manufactured, and qualified to all standards applied to Class 1E instrumentation and specifies the environmental design conditions for each instrument. Sections 2.2.12 and 2.2.13 further discuss features of the instruments and electrical power systems which meet 1E standards.

Question 222.101

Describe how the transfer from emergency operation of the containment systems (provided to mitigate the consequences associated with the site suitability source term) to the TLTM function is accomplished in the event of a core melt accident. Include the following items:

(a) indicate what plant information is necessary to initiate the transfer;

(b) indicate what plant information will be provided to the operators in the control room;

(c) indicate how inadvertant initiation of the TLTM features will be precluded.

Response:

Pertinent information will be found in CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6), Sections 2.1.2.12 (instrumentation requirements), 2.2.12 (instrumentation design) and 2.3 (operator actions).

Amend. 60
Feb. 1981