PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

June 1, 1984

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
DRAFT SAFETY EVALUATION REPORT
OPEN ITEMS

Pursuant to your letter dated March 5, 1984, which transmitted the Hope Creek Draft Safety Evaluation Report (SER), enclosed for your review and approval (see Attachment 2) are the resolutions to those Draft SER open items listed in Attachment 1.

Should you have any questions or require any additional information on these open items, please contact us.

Very truly yours,

R L Mittl/R P Douglas

C D. H. Wagner (w/attach.)
USNRC Licensing Project Manager

W. H. Bateman (w/attach.)
USNRC Senior Resident Inspector

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ATTACHMENT 1

OPEN ITEM	SECTION NUMBER	SUBJECT
5a & d	2.4.5	Wave impact and runup on service water intake structure
7b	2.4.11.2	Thermal aspects of ultimate heat sink
9	2.5.4	Soil damping values
10	2.5.4	Foundation level response spectra
11	2.5.4	Soil shear moduli variation
12	2.5.4	Combination of soil layer properties
13	2.5.4	Lab test shear moduli values
14	2.5.4	Liquefaction analysis of river bottom sands
15	2.5.4	Tabulations of shear moduli
16	2.5.4	Drying and wetting effect on Vincentown
17	2.5.4	Power block settlement monitoring
18	2.5.4	Maximum earth at rest pressure coefficient
19	2.5.4	Liquefaction analysis for service water piping
20	2.5.4	Explanation of observed power block settlement
21	2.5.4	Service water pipe settlement records
22	2.5.4	Cofferdam stability
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14
24	2.5.4	Soil depth models for intake structure
27	2.5.5	Slope stability

ATTACHMENT 1 (Cont'd)

OPEN ITEM	SECTION NUMBER	SUBJECT
30	3.5.1.2	Internally generated missiles (inside containment)
41	3.8.2	Steel containment buckling analysis
42	3.8.2	Steel containment ultimate capacity analysis
43	3.8.2	SRV/LOCA pool dynamic loads
44	3.8.3	ACI 349 deviations for internal structures
45	3.8.4	ACI 349 deviations for Category I structures
46	3.8.5	ACI 349 deviations for foundations
47	3.8.6	Base mat response spectra
48	3.8.6	Rocking time histories
49	3.8.6	Gross concrete section
50	3.8.6	Vertical floor flexibility response spectra
53	3.8.6	Design of seismic Category I tanks
54	3.8.6	Combination of vertical responses
55	3.8.6	Torsional stiffness calculation
56	3.8.6	Drywell stick model development
57	3.8.6	Rotational time history inputs
58	3.8.6	"O" reference point for auxiliary building model
59	3.8.6	Overturning moment of reactor building foundation mat
60	3.8.6	BSAP element size limitations
61	3.8.6	Seismic modeling of drywell shield wall

ATTACHMENT 1 (Cont'd)

OPEN ITEM	SECTION NUMBER	SUBJECT
62	3.8.6	Drywell shield wall boundary conditions
63	3.8.6	Reactor building dome boundary conditions
64	3.8.6	SSI analysis 12 Hz cutoff frequency
65	3.8.6	Intake structure crane heavy load drop
67	3.8.6	Critical loads calculation for reactor building dome
68	3.8.6	Reactor building foundation mat contact pressures
69	3.8.6	Factors of safety against sliding and overturning of drywell shield wall
70	3.8.6	Seismic shear force distribution in cylinder wall
71	3.8.6	Overturning of cylinder wall
72	3.8.6	Deep beam design of fuel pool walls
73	3.8.6	ASHSD dome model load inputs
74	3.8.6	Tornado depressurization
75	3.8.6	Auxiliary building abnormal pressure
76	3.8.6	Tangential shear stresses in drywell shield wall and the cylinder wall
77	3.8.6	Factor of safety against overturning of intake structure
78	3.8.6	Dead load calculations
79	3.8.6	Post-modification seismic loads for the torus
80	3.8.6	Torus fluid-structure interactions
81	3.8.6	Seismic displacement of torus

ATTACHMENT 1 (Cont'd)

OPEN ITEM	SECTION NUMBER	SUBJECT
82	3.8.6	Review of seismic Category I tank design
83	3.8.6	Factors of safety for drywell buckling evaluation
84	3.8.6	Ultimate capacity of containment (materials)
85	3.8.6	Load combination consistency
110b	4.6	Functional design of reactivity control systems
124	6.2.1.5.1	RPV shield annulus analysis
129	6.2.2	Insulation ingestion
152	9.4.4	Radioactivity monitoring elements
154	9.5.1.4.a	Metal roof deck construction classificiation
159	9.5.1.5.a	Primary and secondary power supplies for fire detection system
161	9.5.1.5.b	Fire water valve supervision
162	9.5.1.5.c	Deluge valves
163	9.5.1.5.c	Manual hose station pipe sizing
164	9.5.1.6.e	Remote shutdown panel ventilation
165	9.5.1.6.g	Emergency diesel generator day tank protecton
182	15.9.10	TMI-2 Item II.K.3.18
185	7.2.2.2	Trip system sensors and cabling in turbine building
190	7.2.2.7	Regulatory Guide 1.75
192	7.2.2.9	Reactor mode switch
194	7.3.2.2	Standard review plan deviations

ATTACHMENT 1 (Cont'd)

OPEN ITEM	SECTION NUMBER	SUBJECT
197	7.3.2.5	Microprocessor, multiplexer and computer systems
200	7.4.2.2	Remote shutdown system
205	7.5.2.4	Plant process computer system
209	7.7.2.3	Credit for non-safety related systems in Chapter 15 of the FSAR
210	7.7.2.4	Transient analysis recording system
218	9.5.1.1	Fire hazards analysis
TS-3	4.4.5	Core flow monitoring for crud effects
LC-1	4.2	Fuel rod internal pressure criteria

JS:gs
M P84 80/12 5-gs

ATTACHMENT II

DSER Open Item No. 5a and d (DSER Section 2.4.5)

WAVE IMPACT AND RUNUP ON SERVICE WATER INTAKE STRUCTURE

The applicant has analyzed the wind waves that would traverse plant grade coincident with the PMH surge hydrograph and runup on safety-related facilities. These calculations were based on the assumption that wind waves would be generated in the Delaware Estuary and progress to the site. As the surge level would begin to rise, resulting from the approaching eye of the postulated hurricane, the wind speed would progressively change direction from the southeast clockwise to the west. Waves encroaching on the southern end of the Island would be depth-limited (i.e., the waves would "feel" bottom and thus become shallow water waves) by plant grade elevation on both the Salem and Hope Creek sites. These depth-limited (shallow water) waves will impact and runup on the southern and western faces of the safety-related structures in the power block. The applicant has stated that the southern face of the Reactor Building and the Auxiliary Building are designed for a flood protection level of 38.0 ft msl or 3.2 ft above the maximum calculated wave runup height of 34.8 ft msl and the other exposures of safety-related structures have a flood protection level of 32.0 ft msl or 1 ft above the maximum calculated wave runup height of 31.0 ft msl.
[The staff has requested the applicant to provide additional information on the waves that impact on the river face of service water intake structure. The waves impacting on this face of the structure are not reduced in height (depth-limited) as those that traverse plant grade.]-5a

As indicated in Section 2.4.1, the applicant states that all accesses to safety-related structures (doors and hatches) are provided with water-tight seals designed to withstand the head of water associated with the flood protection levels. [But, the applicant has not indicated whether the water-tight doors are designed to withstand either the combined loading effects of both static water level and the dynamic wave impact]-5b or, [as cited in Sections 3.4.1 and 3.5.1.4 of this report, the impact of a barge propelled by winds and waves associated with a hydrologic event that floods plant grade.]-5c

Item No. 5a and d (Cont'd)

Based upon its analysis according to SRP 2.4.5, the staff concludes that the flood protection level of El. 38.0 ft msl for the southern face of the Reactor Building and Auxiliary Building and El. 32.0 ft msl for the remaining safety-related structures within the power block meets the requirements of Regulatory Guide 1.59. [Until additional information and analysis are available, the staff cannot conclude that the flood protection level of El. 32.0 ft msl for the Service Water Intake Structure meets the requirements of Regulatory Guide 1.59.]-5d Based on its analysis, the staff cannot conclude that the plant meets the requirements of GDC 2 with respect to the hydrologic aspects of Probable Maximum Surges and Seiche Flooding.

RESPONSE

The requested information for the service water intake structure has been provided in the responses to the following NRC questions:

Information Provided	Question No.
Wave runup elevations	240.8
Wave impact loads	240.9
Flood protection	240.8 and 410.69

DSER Open Item No. 7b (DSER Section 2.4.11.2)

THERMAL ASPECTS OF ULTIMATE HEAT SINK

The applicant has analyzed the ability of the service cooling water supply to withstand the effect of such severe natural phenomena as ice blockage, flooding, low water, and thermal aspects of UHS. As indicated in Section 2.4.7, the effects of ice blockage would not obstruct the flow to safety-related pumps. Thus the staff concludes that the intake structure and essential service cooling water flow is adequately protected against ice effects. [As indicated in Section 2.4.5, the ability of the service water intake structure to withstand the effects of PMH surge flooding and associated wave runup and overtopping remains an open item.]-7b

The applicant reported that the minimum historical low water level at the Reedy Point, Delaware tide station is -8.6 ft msl. The applicant's analysis of the maximum setdown considered the PMH wind speed of 85 mph (the overland PMH wind speed for the direction resulting in maximum setdown) to be blowing down the estuary coincident with 10% exceedance low spring astronomical tide of -3.9 ft msl and the associated trough of the 6.0 ft maximum wind wave. The resultant low water level would be -13.0 ft msl. The applicant has stated that -13.0 ft msl is the design basis minimum low water level for service water pumps. Based on its independent analysis, the staff concurs that -13.0 ft msl is an appropriate design basis minimum low water level. [The applicant has not identified the maximum intake temperature that will allow the plant to safely shut down under normal and emergency conditions as discussed in Regulatory Guide 1.27 nor the ability of the Delaware River to supply water below this temperature. Until this information is available, the staff cannot conclude that the plant meets GDC 44 with respect to the thermal aspects of UHS.]-7a

Based upon the evaluation described above, we conclude the hydrologic characteristics of the Ultimate Heat Sink meet the requirements of 10 CFR Part 100 and 10 CFR Part 100, Appendix A. As indicated above, certain aspets related to flooding level for the service water intake structure are

DSER Open Item No. 7b (Cont'd)

unresolved. Therefore, the staff cannot conclude that the Ultimate Heat Sink System meets the requirements of General Design Criterion 2 with respect to hydrologic characteristics. In addition, the staff cannot conclude that the Ultimate Heat Sink meets the requirements of GDC 44 with respect to thermal aspects of the heat transfer system.

RESPONSE

For information on the ability of the service water intake structure to withstand the effects of PMH surge flooding and associated wave runup and overtopping, see the response to DSER Open Item Number 5a and d.

DSER Open Item No. 9 (DSER Section 2.5.4)

SOIL DAMPING VALUES

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.1 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 10 (DSER Section 2.5.4)

FOUNDATION LEVEL RESPONSE SPECTRA

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.2 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 11 (DSER Section 2.5.4)

SOIL SHEAR MODULI VARIATION

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.5 from the NRC Structural/Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 12 (DSER Section 2.5.4)

COMBINATION OF SOIL LAYER PROPERTIES

Cn the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.6 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 13 (DSER Section 2.5.4)

LAB TEST SHEAR MODULI VALUES

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.8 from the NRC Structural/Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 14 (DSER Section 2.5.4)

LIQUEFACTION ANALYSIS OF RIVER BOTTOM SANDS

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.15 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 15 (DSER Section 2.5.4)

TABULATIONS OF SHEAR MODULI

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item B.6 from the NRC Structural/Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 16 (DSER Section 2.5.4)

DRYING AND WETTING EFFECT ON VINCENTOWN

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item B.12 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 17 (DSER Section 2.5.4)

POWER BLOCK SETTLEMENT MONITORING

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item B.13 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 18 (DSER Section 2.5.4)

MAXIMUM EARTH AT REST PRESSURE COEFFICIENT

The below-grade walls of structures were designed to resist both the static and dynamic pressure resulting from the surrounding earth and water. The value of the lateral earth pressure coefficient at rest used in the design was 0.43. The dynamic lateral earth pressure on the below-grade wall was determined from the results of soil-structure interaction analyses. The procedure used to obtain the dynamic lateral earth pressure is in accordance with the state-of-the-art methods required by the Standard Review Plan (NUREG-0800) and is therefore acceptable. Although the lateral earth pressure at rest is low, during the structural and geotechnical engineering audit held in January 1984, the applicant demonstrated that the below-grade walls have the capacity to resist substantially higher lateral earth pressures and will so state this face in a future amendment to the FSAR.

RESPONSE

Section 2.5.4 has been revised to include a statement that the below-grade walls have the capacity to resist lateral earth pressures substantially higher than the actual lateral earth pressures.

Several methods were used to compute the settlement of these structures (References 2.5-115, -116, -117, and -118). The results of these analyses indicate that the total maximum settlement under the net loads is estimated to be about one inch including the recompression of heave. These analyses were performed by assuming that the mats were uniformly loaded. The settlement will be due, for the most part, to elastic deformations of the subsoil, a very small fraction being contributed by the elastic deformation of the lean concrete on structured backfill. Considering the granular soil type and that the total load of the structures consists mainly of dead load, most of the settlement will have occurred during construction. As a result, post-construction differential settlement is expected to be less than 1/2 inch.

The areas around the reactor, auxiliary, and turbine building structures are back-filled to final grade with compacted well-graded granular soils. The walls of these structures are designed to resist the lateral pressures of the soils under static and dynamic loadings. The static earth pressures are based on "at-rest" conditions, whereas the dynamic earth pressures are determined based on soil-structure interaction analysis discussed in Section 3.7.2.5. Figures 2.5-60 and 2.5-61 provide the earth pressures used as design bases.

Add "Insert A"

2.5.4.10.1.2 Service Water Intake Structure

The Service Water Intake Structure, approximately 100 x 120 feet in plan area, is a safety-related structure. It is located at the waterfront and consequently is partially submerged. The structure will be founded on a mat at elevation +65.5. Tremie concrete will be placed between the base of the mat and the bearing level in the Vincentown sands. The unweathered greenish-gray Vincentown sands considered suitable as a bearing stratum occur at approximately elevation +25 feet in the intake area of the site and, borings and initial excavation operations at the location of the Service Water Intake structure encountered the unweathered Vincentown at approximately elevation +23 to +29 feet (Reference 2.5-119). This lower occurrence of the bearing stratum in this area was taken into account in the configuration and calculated contact stresses of the intake structure.

The stress relief due to excavation of approximately 70 feet of submerged soil is expected to be 4000 lbs/ft^2. However, because the total excavation area is only 100 x 120 feet and because sheet piles extend below the excavation level, the elastic rebound is expected to be negligible. About 70% of the removed load will be restored by the time placement of lean concrete is completed at the proposed grade, elevation +65.5 feet. The net load to be imposed by the proposed construction is calculated to be very small because of stress relief and buoyancy effects.

DSER OPEN ITEM 18

INSERT A

Although the static lateral earth pressures given in Figures 2.5-60 are low, the below-grade walls have the capacity to resist substantially higher lateral earth pressures.

DSER Open Item No. 19 (DSER Section 2.5.4)

LIQUEFACTION ANALYSIS FOR SERVICE WATER PIPING

The liquefaction potential was determined by comparing the shear stresses induced in the soil by the SSE with the cyclic shear strength of the soil under field conditions. The maximum shear stresses at various points in the foundation soils were obtained from previous dynamic analyses. The cyclic shear strength number for the Vincentown sands was determined through laboratory tests. The dynamic strength of the soil layers overlying the Vincentown sands (hydraulic fill, river bottom sands, Kirkwood clays, and basal sands) was evaluated on the basis of static strength tests, index properties, field tests, and correlation with data from literature, in addition to limited dynamic triaxial testing. On the basis of these results, the applicant concluded that only the sandy portions of the hydraulic fill may experience SSE-induced liquefaction.

Because the safety-related structures were surrounded by hydraulic fill, the sliding stability under SSE condition were further evaluated by the applicant. The applicant concluded that, because the safety structures were embedded at least 60 ft in soil and only the upper 30 ft could liquefy, this afforded at least 30 ft of stable soil confinement to the power block structure. In addition, the applicant stated that the nonliquefiable backfill surrounding the structure would provide additional resistance to sliding. The staff concurs with the applicant's conclusion that the power block structures will be stable under SSE conditions.

However, the applicant has not provided sufficient information about the sliding stability of the intake structure and the effects of potential liquefaction on the intake structure and the service water pipeline.

RESPONSE

This item corresponds to Item A.2 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 20 (DSER Section 2.5.4)

EXPLANATION OF OBSERVED POWER BLOCK SETTLEMENT

The staff concurs in the applicant's assessment that the factor of safety against bearing capacity failure is adequate. However, the measured settlements presented in FSAR Figures 241.25-1 through 241.25-30 show some erratic movements. The applicant has been requested to assess the observed settlements in the power block area and to determine the settlement characteristics along the pipeline.

RESPONSE

This item corresponds to Item A.3 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 21 (DSER Section 2.5.4)

SERVICE WATER PIPE SETTLEMENT RECORDS

All safety-related structures as well as the turbine building are founded on lean concrete bearing on structural backfill placed on the dense to very dense sands of the Vincentown formation. Foundation levels, dimensions, and static loads for the major facilities of the station are listed in FSAR Table 2.5-18. The applicant has calculated the factor of safety for bearing capacity to be greater then 3. The calculated settlement is about 1 in., including the recompression of heave. The postconstruction differential settlement is expected to be less than 1/2 in. No settlement estimate is presented along the service water pipeline.

The staff concurs in the applicant's assessment that the factor of safety against bearing capacity failure is adequate. However, the measured settlements presented in FSAR Figures 241.25-1 through 241.25-30 show some erratic movements. The applicant has been requested to determine the settlement characteristics along the pipeline.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 22 (DSER Section 2.5.4)

COFFERDAM STABILITY

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.5 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 23 (DSER Section 2.5.4)

CLARIFICATION OF FSAR TABLES 2.5-13 and 2.5-14

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.6 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 24 (DSER Section 2.5.4)

SOIL DEPTH MODELS FOR INTAKE STRUCTURES

On the basis of the applicant's design criteria and construction specifications and the results of the applicant's investigation, laboratory and field tests, and analyses, and the results of the January 1984 audit, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the identified open items can be resolved.

RESPONSE

This item corresponds to Item A.13 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 27 (DSER Section 2.5.5)

SLOPE STABILITY

The applicant stated in the FSAR that there are no natural slopes within the plant boundaries. However, there are slopes along the river bank in the vacinity of the intake structure, and the failure of these slopes could adversely affect the intake structure. Therefore, the stability assessment of these slopes is required. The applicant, during the January 1984 audit, stated that the design analyses for the river bank slope protection will be provided in April 1984 for NRC review. The staff will provide its evaluation in a future supplement to the SER.

RESPONSE

This item corresponds to Item A.5 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 30 (DSER Section 3.5.1.2)

INTERNALLY GENERATED MISSILES (INSIDE CONTAINMENT)

Based on our review, we cannot conclude that the design is in conformance with General Design Criterion 4 as it relates to protection against internally generated missiles. We cannot determine that the design of the facility for providing protection from internally generated missiles meets the acceptance criteria of SRP Section 3.5.1.2.

RESPONSE

This item is not an open item per telephone conversation (see attached) between J. M. Ashley (PSE&G) and John Ridgely (NRC-ASB) on March 22, 1984.

JES:dh

6/1/84

M P84 54/07 1-dh

TELEPHONE NOTES

PSE&G Hope Creek Licensing (Bethesda)

Date: March 22, 1984

From: J.M. Ashley

To: D. Wagner, J. Ridgely (ASB)

Subject: HCGS DSER Open Items

Discussion

Ashley called to find out what NRC concerns existed with respect to FSAR Sections 3.5.1.2 (Item 30), 9.2.2 (Item 145) and 9.4.4 (Item 152).

Ridgely explained that these items were inadvertently listed as open items in the listing of open items at the front of the DSER. The NRC has no outstanding concerns with the sections.

J. M. Ashley

DSER OPEN ITEM 30

DSER Open Item No. 41 (DSER Section 3.8.2)

STEEL CONTAINMENT BUCKLING ANALYSIS

The applicant has been requested to submit information regarding the ultimate capacity analysis of the containment and steel containment buckling analysis. The staff has not received all the required information on these two items. The applicant has committed to provide the required information to the staff for review by February 15, 1984. The staff will review and report its resolution of these two items in the Final SER.

RESPONSE

A description of the drywell buckling evaluation has been provided in FSAR Appendix 3E in response to Question 220.11. Additional information has been requested in Item B.1 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 42 (DSER Section 3.8.2)

STEEL CONTAINMENT ULTIMATE CAPACITY ANALYSIS

The applicant has been requested to submit information regarding the ultimate capacity analysis of the containment and steel containment buckling analysis. The staff has not received all the required information on these two items. The applicant has committed to provide the required information to the staff for review by February 15, 1984. The staff will review and report its resolution of these two items in the Final SER.

RESPONSE

A description of the ultimate capacity analysis of the containment has been provided in FSAR Appendix 3I in response to Question 220.22. Additional information was requested in Item B.2 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item was submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 43 (DSER Section 3.8.2)

SRV/LOCA POOL DYNAMIC LOADS

With respect to SRV/LOCA pool dynamic load considerations, the applicant has performed a reevaluation of containment design adequacy based on staff positions provided in NUREG-0661. The applicant has submitted his reevaluation summary report. However, the staff has not completed its review. It will report on the resolution of these issue in the Final SER.

RESPONSE

The Plant Unique Analysis Report (PUAR), which describes the reevaluation of the containment design adequacy based on staff positions provided in NUREG-0661, was submitted to the NRC by a letter dated February 10, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 44 (DSER Section 3.8.3)

ACI 349 DEVIATIONS FOR INTERNAL STRUCTURES

SRP Section 3.8.3 specifies that the code to be used in the design of concrete internal structures is ACI Std 349 as augmented by RG 1.142. The applicant had been requested to provide information regarding an assessment and justifications for all deviations of his internal structural design and analysis from the applicable staff positions as given in SRP Section 3.8.3. The applicant provided the information on January 23, 1984. However, the staff has not completed its review. It will report on the resolution of this item in the Final SER. Additionally, some of the 39 structural audit action items discussed under Section 3.8.6, as they pertain to this section of the SER, are considered unresolved.

RESPONSE

The requested information is included in the response to NRC Question 220.24.

DSER Open Item No. 45 (DSER Section 3.8.4)

ACI 349 DEVIATIONS FOR CATEGORY I STRUCTURES

Category I structures other than the containment and its interior structures are all of structural steel and concrete. The structural components consist of slabs, walls, beams, and columns. The major code used in the design of concrete Category I structures is ACI Std 381-71. For steel Category I structures, AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" is used. The applicant had been requested to provide an assessment and justifications of all deviations from the applicable requirements of ACI 349 as augmented by RG 1.142. The applicant provided the information on January 23, 1984. However, the staff has not completed its review. It will report on the resolution of this item in the Final SER. Additionally, some of the 39 action items discussed in Section 3.8.6 of this SER pertain to this section, and the items remain to be resolved to the satisfaciton of the staff.

RESPONSE

The requested information is included in the response to NRC Question 220.26.

DSER Open Item No. 46 (DSER Section 3.8.5)

ACI 349 DEVIATIONS FOR FOUNDATIONS

The design and analysis procedures that were used for these Category I foundations are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71. The various Category I foundations were designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on ACI 318-71 and on the AISC specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme. The applicant had been requested to provide an assessment and justifications of all deviations of this design from the applicable requirements of ACI 349 as augmented by RG 1.142. The applicant provided the information on January 23, 1984. However, the staff has not completed its review. It will report its resolution of this issue in the Final SER. In the meantime, this item remains open. Furthermore, some of the action items discussed in Section 3.8.6 of this SER, as they pertain to the foundation design and analysis, should be considered open items and remain to be resolved.

RESPONSE

The requested information is included in the response to NRC Question 220.26.

DSER Open Item No. 47 (DSER Section 3.8.6)

BASE MAT RESPONSE SPECTRA

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.3 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 48 (DSER Section 3.8.6)

ROCKING TIME HISTORIES

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 49 (DSER Section 3.8.6)

GROSS CONCRETE SECTION

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.11 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 50 (DSER Section 3.8.6)

VERTICAL FLOOR FLEXIBILITY RESPONSE SPECTRA

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.12 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 53 (DSER Section 3.8.6)

DESIGN OF SEISMIC CATEGORY I TANKS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/ Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 54 (DSER Section 3.8.6)

COMBINATION OF VERTICAL RESPONSES

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.5 from the NRC Structural/Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 55 (DSER Section 3.8.6)

TORSIONAL STIFFNESS CALCULATION

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.8 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 56 (DSER Section 3.8.6)

DRYWELL STICK MODEL DEVELOPMENT

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.9 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 57 (DSER Section 3.8.6)

ROTATIONAL TIME HISTORY INPUTS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.10 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 58 (DSER Section 3.8.6)

"O" REFERENCE POINT FOR AUXILIARY BUILDING MODEL

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.11 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 59 (DSER Section 3.8.6)

OVERTURNING MOMENT OF REACTOR BUILDING FOUNDATION MAT

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.7 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 60 (DSER Section 3.8.6)

BSAP ELEMENT SIZE LIMITATIONS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.8 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 61 (DSER Section 3.8.6)

SEISMIC MODELING OF DRYWELL SHIELD WALL

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.9 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 62 (DSER Section 3.8.6)

DRYWELL SHIELD WALL BOUNDARY CONDITIONS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.10 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 63 (DSER Section 3.8.6)

REACTOR BUILDING DOME BOUNDARY CONDITIONS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.11 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 64 (DSER Section 3.8.6)

SSI ANALYSIS 12Hz CUTOFF FREQUENCY

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.12 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 65 (DSER Section 3.8.6)

INTAKE STRUCTURE CRANE HEAVY LOAD DROP

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.15 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 67 (DSER Section 3.8.6)

CRITICAL LOADS CALCULATION FOR REACTOR BUILDING DOME

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.17 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 68 (DSER Section 3.8.6)

REACTOR BUILDING FOUNDATION MAT CONTACT PRESSURES

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.1 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 69 (DSER Section 3.8.6)

FACTORS OF SAFETY AGAINST SLIDING AND OVERTURNING OF DRYWELL SHIELD WALL

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.2 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 70 (DSER Section 3.8.6)

SEISMIC SHEAR FORCE DISTRIBUTION IN CYLINDER WALL

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.3 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 71 (DSER Section 3.8.6)

OVERTURNING OF CYLINDER WALL

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.4 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 72 (DSER Section 3.8.6)

DEEP BEAM DESIGN OF FUEL POOL WALLS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.5 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 73 (DSER Section 3.8.6)

ASHSD DOME MODEL LOAD INPUTS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.6 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 74 (DSER Section 3.8.6)

TORNADO DEPRESSURIZATION

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.7 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 75 (DSER Section 3.8.6)

AUXILIARY BUILDING ABNORMAL PRESSURE

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.8 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 76 (DSER Section 3.8.6)

TANGENTIAL SHEAR STRESSES IN DRYWELL SHIELD WALL AND THE CYLINDRICAL WALL

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.9 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 77 (DSER Section 3.8.6)

FACTOR OF SAFETY AGAINST OVERTURNING OF INTAKE STRUCTURE

From January 10 through Januray 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.12 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 78 (DSER Section 3.8.6)

DEAD LOAD CALCULATIONS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.13 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 79 (DSER Section 3.8.6)

POST-MODIFICATION SEISMIC LOADS FOR TORUS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.1 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 80 (DSER Section 3.8.6)

TORUS FLUID-STRUCTURE INTERACTIONS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.2 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 81 (DSER Section 3.8.6)

SEISMIC DISPLACEMENT OF TORUS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.3 from the NRC Structural/ Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 82 (DSER Section 3.8.6)

REVIEW OF SEISMIC CATEGORY I TANK DESIGN

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/ Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984, from R. L. Mittl to A. Schwencer.

DSER Open Item No. 83 (DSER Section 3.8.6)

FACTORS OF SAFETY FOR DRYWELL BUCKLING EVALUATION

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.1 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 84 (DSER Section 3.8.6)

ULTIMATE CAPACITY OF CONTAINMENT (MATERIALS)

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.2 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 85 (DSER Section 3.8.6)

LOAD COMBINATION CONSISTENCY

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.3 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 110b (DSER Section 4.6)

FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The control rod drive system was reviewed in accordance with Section 4.6 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the control rod drive system with respect to the applicable regulations of 10 CFR 50.

[The applicant has not addressed the recommendations of NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR scram System Piping."]-110a

The design does not utilize a CRDS return line to the reactor pressure vessel. In accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drives Return Line Nozzle Cracking," dated November 1980, equalizing valves are installed between the cooling water header and exhaust water header, the flow stabilizer loop is routed to the cooling water header, and both the exhaust header and flow stabilizer loop are stainless steel piping.

We have reviewed the extend of conformance of the Scram Discharge Volume (SDV) design with the NRC generic study, "BWR Scram Discharge System Safety Evaluation," dated December 1, 1980. The design provides two separate SDV headers, with an integral instrumented volume (IV) at the end of each header, thus providing close hydraulic coupling. Each IV has redundant and diverse level instrumentation (float sensing and pressure sensing) for the scram function attached directly to the IV. Vent and drain lines are completely separated and contain redundant vent and drain valves with position indication provided in the main control room. [With respect to Design Criterion 8, the applicant stated that the "SDV Piping is continuously sloped from its high point to its low point." In order to provide a response to Design Criterion 8, the applicant must provide

DSER Open Item No. 110b (DSER Section 4.6) (Cont'd)

a description of the SDV from the beginning of the SDV to the IV drain. The description should include piping geometry (ie., pitch, line size, orientation).]-110b

Except for Design Criterion 8, we conclude that the design of the SDV fully meets the requirements of the above referenced NRC generic SER and is therefore acceptable. Additionally, the above-described design of the SDV satisfies LRG-II, Item 1-ASB, "BWR Scram Discharge Volume Modifications."

Based on our review, we conclude that the functional design of the reactivity control system meets the requirements of General Design Criteria 23, 25, 26, 27, 28, and 29 with respect to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences and accident conditions including single failures, and the guidelines of NUREG-0619 and is, therefore, acceptable. We cannot conclude compliance with the guidelines of NUREG-0803 and the generic document dated December 1, 1980. The functional design of the reactivity control system does not meet the applicable acceptance Criteria of SRP 4.6. We will report resolution of these items in a supplement to this SER.

RESPONSE

FSAR Section 4.6.1.2.4.2(f) has been revised to include a description of the SDV piping.

room. Differential pressure between the reactor vessel and the cooling water header is indicated in the main control room. Although the drives can function without cooling water, seal life is shortened by long-term exposure to reactor temperatures. The temperature of each drive is indicated and recorded, and excessive temperatures are annunciated in the main control room.

e. Exhaust water header - The exhaust water header connects to each HCU and provides a low pressure plenum and discharge path for the fluid expelled from the drives during control rod insert and withdraw operations. The fluid injected into the exhaust water header during rod movements is discharged back up to the RPV via reverse flow through the insert exhaust directional solenoid valves of adjoining HCUs. The pressure in the exhaust water header is, therefore, maintained at essentially reactor pressure. To ensure that the pressure in the exhaust water header is maintained near reactor pressure during the period of vessel pressurization, redundant pressure equalizing valves connect the exhaust water header to the cooling water header.

f. Scram discharge volume - The scram discharge volume (SDV) consists of two sets of 12 inch diameter header piping, each of which connects to one-half of the HCUs and drains into a 12 inch diameter scram discharge instrument volume (SDIV). Each set of header piping is sized to receive and contain all the water discharged by one-half of the drives during a scram, independent of the SDIV.

The header piping slopes to a low point with a minimum pitch of 1/8 inch per foot as shown on Figure 4.6-10. The SDIV for each header set is directly connected to the low point of the header piping. The large-diameter pipe of each SDIV thus serves as a vertical extension of the SDV. A 2 inch piping connection at the bottom of the SDIV provides drainage of the SDIV and SDV via sloped drain lines with a minimum 1/8 inch per foot slope.

During normal plant operation, the SDV is empty and is vented to the atmosphere through its open vent and drain valves. When a scram occurs, upon a signal from the safety circuit, these vent and drain valves are closed to conserve reactor water. Redundant vent and drain valves are provided to ensure against loss of reactor coolant from the SDV following a scram. Lights in the main control room indicate the position of these valves.

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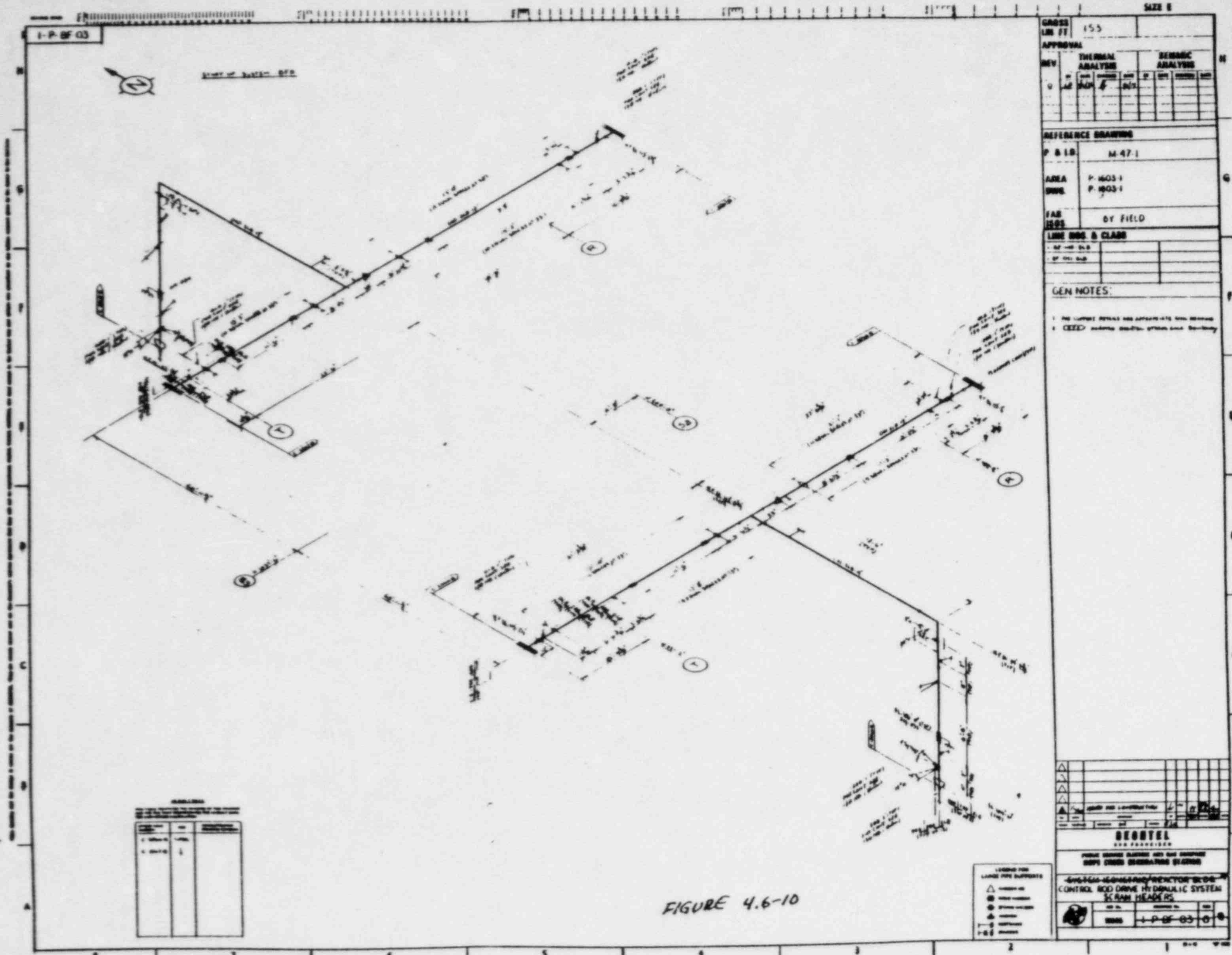

FIGURE 4.6-10

DSER Open Item No. 124 (DSER Section 6.2.1.5.1)

RPV SHIELD ANNULUS ANALYSIS

The applicant's analysis resulted in pressures in the shield annulus that peak at approximately 90 psia in the volumes surrounding the recirculation line break and approximately 100 psia in the volumes surrounding the feedwater line break. [The applicant has not provided a graphical presentation of the differential pressure (psi) responses as a function of time for a selected number of nodes, as requested.]-124a

[In addition, the applicant has not provided the peak and transient loading on the major components used to establish 124b
the adequacy of the supports design. This should include the load forcing functions (e.g., fx(t), fy(t), fz(t)) and transient moments (e.g., Mx(t), My(t), Mz(t)) as resolved about a specific identified coordinate system.] [The applicant also has not provided the projected areas used to 124c
calculate these loads. This information was also previously requested.] The staff intends to perform confirmatory analysis using the COMPARE code upon receipt of this information.

RESPONSE

The graphical presentation of differential pressure is not required per March 30, 1984 conference call between the NRC and Bechtel. Bechtel noted that the initial containment pressure could be considered constant during the transient and thus differential pressure can be determined by subtracting a constant initial pressure from the already provided graphical presentations of absolute pressure.

It should be noted that Part b is being reconsidered by the NRC and will be provided later if necessary.

The requested projected areas for the RPV Shield Annulus Analysis are provided in the attached table.

TABLE 124-1

Projected Areas (in^2)

NODE	VESSEL AX	VESSEL AY	SHIELD AX	SHIELD AY
1	5747	1540	6875	1842
2	4207	4207	5033	5033
3	1540	5747	1842	6875
4	1540	5747	1842	6875
5	4207	4207	5033	5033
6	5747	1540	6875	1842
7	5747	1540	6875	1842
8	4207	4207	5033	5033
9	1540	5747	1842	6875
10	1540	5747	1842	6875
11	4207	4207	5033	5033
12	5747	1540	6875	1842
13	8281	2219	9906	2654
14	6062	6062	7252	7252
15	2219	8281	2654	9906
16	2219	8281	2654	9906
17	6062	6062	7252	7252
18	8281	2219	9906	2654
19	8281	2219	9906	2654
20	6062	6062	7252	7252
21	2219	8281	2654	9906
22	2219	8281	2654	9906
23	6062	6062	7252	7252
24	8281	2219	9906	2654
25	6343	1700	7588	2033
26	4644	4644	5555	5555
27	1700	6343	2033	7588
28	1700	6343	2033	7588
29	4644	4644	5555	5555
30	6343	1700	7588	2033
31	6343	1700	7588	2033
32	4644	4644	5555	5555
33	1700	6343	2033	7588
34	1700	6343	2033	7588
35	4644	4644	5555	5555

TABLE 124-1

	VESSEL		SHIELD	
NODE	AX	AY	AX	AY
36	6343	1700	7588	2033
37	6343	1700	7588	2033
38	4644	4644	5555	5555
39	1700	6343	2033	7588
40	1700	6343	2033	7588
41	4644	4644	5555	5555
42	6343	1700	7588	2033
43	6343	1700	7588	2033
44	4644	4644	5555	5555
45	1700	6343	2033	7588
46	1700	6343	2033	7588
47	4644	4644	5555	5555
48	6343	1700	7588	2033
49	8347	2237	9985	2676
50	6111	6111	7310	7310
51	2237	8347	2676	9985
52	2237	8347	2676	9985
53	6111	6111	7310	7310
54	8347	2237	9985	2676
55	8347	2237	9985	2676
56	6111	6111	7310	7310
57	2237	8347	2676	9985
58	2237	8347	2676	9985
59	6111	6111	7310	7310
60	8347	2237	9985	2676
61	3876	1038	4636	1242
62	2837	2837	3394	3394
63	1038	3876	1242	4636
64	1038	3876	1242	4636
65	2837	2837	3394	3394
66	3876	1038	4636	1242
67	3876	1038	4636	1242
68	2837	2837	3394	3394
69	1038	3876	1242	4636
70	1038	3876	1242	4636
71	2837	2837	3394	3394
72	3876	1038	4636	1242

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DSER Open Item No. 129 (DSER Section 6.2.2)

INSULATION INGESTION

With respect to insulation debris generation and transport, these issues are insulation types and quantity dependent, and also plant design dependent. HCGS plans to use fiberglass blanket sections with 22-gauge, stainless steel jacketing to insulate structures, equipment and piping within the primary containment. LOCA breaks have the capability to locally destroy (or fragment) fibrous insulation. The FSAR does not address the question of LOCA generated debris transport at low velocities (i.e., 0.2 -0.4 ft/sec). The FSAR alleges that it is unlikely that insulation materials would transport to and plug the suction strainers without providing a quantified treatment regarding the minimal amount of insulation destruction and transport thereof. Flow velocities in the vicinity of the RHR suction strainers are sufficiently high to transport shredded fibrous insulation debris to the suction strainers.

RESPONSE

The information requested above has been provided in the report "Evaluation of Drywell Insulation Debris Effects on ECCS Pump Performance" provided under separate cover. Section 6.2.2.2.2 has been revised to reflect this report.

Evaluation of Drywell Insulation Debris Effects on ECCS Pump Performance

Hope Creek Generating Station
Public Service Electric & Gas Company

Bechtel Power Corporation
San Francisco, CA

May 1984
Revision 1

DSER OPEN ITEM 129

1 INTRODUCTION

1.1 PURPOSE

There is a concern that the insulation debris created by a high energy line break in the primary containment will collect on the Emergency Core Cooling System (ECCS) suction strainers and impair the pump performance. These debris considerations are part of Unresolved Safety Issue A-43, Containment Emergency Sump Performance.

The Hope Creek Generating Station (HCGS) was evaluated to determine the maximum quantity of insulation debris that might be generated by a LOCA. The evaluation includes transport of this debris to the ECCS suction strainers and the effect on ECCS pump operation. Only the low pressure coolant injection (LPCI) and core spray (CS) pumps are evaluated because only large pipe breaks can generate significant quantities of insulation debris. Neither the high pressure coolant injection nor the reactor core isolation cooling pumps are able to operate after a large break LOCA.

The only insulation debris in the drywell that might enter the vent pipes, and eventually the suppression pool, are small pieces of shredded fiberglas because of the small openings in the jet deflectors. Whole and torn blankets are assumed to be retained by the drywell internal obstructions and the vent jet deflectors.

1.2 SUMMARY OF RESULTS

It has been determined that the postulated fibrous insulation debris generated by a high energy line break will not jeopardize ECCS pump operation at HCGS. This is based on evaluation of the transport of the worst case shredded debris generation in the drywell. The volume of shredded debris and its transport to the ECCS strainers has been conservatively evaluated. The head loss due to the accumulation of the debris concurrent with the conservative NPSH available conditions does not cause the NPSH available to drop below that required by the ECCS pumps. The NPSH available to the ECCS pumps is at least 10 ft greater than the required 7.5 ft for the LPCI and 3.5 ft for the core spray pumps. The results of the analysis are shown on Table 3-1.

2 BASES

2.1 HCGS is a BWR with a Mark I containment design.

2.2 The ECCS pumps take suction from the suppression pool water inventory following a LOCA. The suppression pool is located in the torus surrounding the base of the drywell. The fluid from a LOCA is released into the drywell and the steam-water-gas mixture is conducted to the suppression chamber by

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the eight vent pipes and the vent/downcomer header system in the torus. When the liquid level on the floor of the drywell reaches the elevation of the vent pipe openings this liquid also flows to the suppression pool through the vent system.

The ECCS suction strainers are located above the bottom of the torus in eight different sections. Figures 2-1 through 2-5 show the general arrangement.

2.3 The only thermal insulation used in the drywell is Owens-Corning "NUKON", stainless steel jacketed fiberglas. The fiberglas is totally enclosed in woven fiberglas covers with glass fiber stitching. The blankets are held on with velcro fasteners and protected with 22 gage stainless steel jackets with seismically qualified mechanical latches.

3 METHOD OF ANALYSIS

3.1 IDENTIFICATION OF PIPE RUPTURE LOCATIONS (PRL)

3.1.1 The PRLs for analysis are based on the locations identified in the pipe break analysis discussed in FSAR Section 3.6 (Reference 4.4).

3.1.2 The break locations analysed were chosen in the large diameter lines with greatest potential for generating significant quantities of insulation debris. The PRLs analysed are in the following lines:

1. Reactor Recirculation Pump Suction
2. Main Steam Line
3. Feedwater Line
4. RHR Supply Line

3.1.3 The pipe break is postulated to be circumferential. The broken pipe is not assumed to shadow the jet cone although pipe movement is restricted by whip restraints. This will result in a conservative estimation of the affected insulation. Slot breaks were not evaluated as the resulting jet would have a smaller zone of influence and would therefore generate less insulation debris.

3.1.4 Extensive use of pipe whip restraints and separation of lines eliminates pipe whip and pipe impact as significant mechanisms for insulation debris generation. The debris generated by these mechanisms would be in the form of the manufactured whole blankets. This form of debris is retained in the drywell and does not affect the ECCS suction strainers.

3.2 DEBRIS GENERATION

3.2.1 The pipe break is postulated to produce a jet from each end of the break. Each jet cone is assumed to expand at

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an angle of 45 degrees from the pipe centerline as recommended in NUREG 0897 (Reference 4.1). The direction of the cone is along the original pipe centerline because the pipe movement is limited by the whip restraint.

3.2.2 Each jet cone is separated into two regions. Region I is the portion from the break to the plane where the jet thrust divided by the jet area is 20 psig. All insulation in Region I is assumed to be shredded into small fragments by the jet impingement forces. Region II is the portion of the jet cone extending from Region I to the plane where the jet thrust divided by the jet area is equal to 0.5 psig. The insulation in the region is assumed to be dislodged in the as-fabricated form.

3.2.3 The pipe whip restraints and large structural steel members inside the jet cone are assumed to cause "shadowing" of the jet (i.e., insulation in the "shadow" of the member is not assumed to be shredded). This is consistent with the criteria for modeling jet impingement forces in FSAR Section 3.6 (Reference 4.4) and SRP 3.6.2 (Reference 4.9).

3.2.4 The stainless steel jacketing on the insulation is assumed to provide no protection of the insulation blankets. This is a conservative assumption because it is expected that the steel jacketing will provide some protection against shredding of the insulation, especially where the jet pressure is between 20 and 60 psig.

3.2.5 A geometric analysis was performed to determine the volume of insulation that would be affected by the selected break locations. The volume of insulation exposed to jet impingement in Region I of the jet cone was quantified. The insulation dislodged in Region II of the jet cone was not quantified because the physical barriers in the drywell discussed in FSAR Section 6.2.2 (Reference 4.4) will prevent the insulation blankets from entering the suppression pool.

3.2.6 The break location generating the largest volume of shredded fibrous insulation is the Main Steam Line. The results of the analysis are provided below:

Main Steam Line Break (Line D)			Insulation
M. S. Line	D	26"Ø	27.0 ft^3
M. S. Line	C	26"Ø	6.75 ft^3
LPCI Line		12"Ø	12.0 ft^3
Recirc Pump Discharge		22"Ø	4.6 ft^3
			50.75 ft^3

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The break location is shown in figures 3-1, -2 and -3. The same geometry applies to Main Steam Line A. Main steam lines B and C are less limiting.

3.3 INSULATION DEBRIS TRANSPORT

3.3.1 Short Term Transport

3.3.1.1 The short term is the period during initial blowdown from the postulated pipe break. The blowdown lasts for about 1.5 minutes for the main steam line break cases. The insulation is transported by the jet force from the break and flow of the spilled fluid to the drywell floor. As indicated earlier, pipe whip and impact are not considered in the analysis.

3.3.1.2 It is conservatively assumed that all of the insulation in Region I of the jet cone is shredded and transported to the drywell floor by the jet forces. In reality, a portion of the shredded insulation would be distributed and retained on the structural steel and on the grating and components in the drywell. Only a portion would reach the floor and be available for transport to the torus. The shredded insulation is assumed to be uniformly mixed in the turbulent liquid collecting on the containment floor. When sufficient liquid has collected on the floor to reach the level of the vent pipes, it overflows and is carried to the suppression pool by the vent header. It is conservatively assumed that all of the shredded insulation in the drywell floor pool, except that in the sumps and inside the cylindrical vessel pedestal, is transported to the suppression pool with the overflowing liquid. The area under the vessel pedestal has only one opening at the elevation of the drywell flooding so this volume of water will become stagnant when the equalibrium flooding level in the drywell is reached. The sumps in the drywell are below the floor so that after they are filled and the drywell flooding level is above the top of the sump they become stagnant pools.

3.3.2 Long Term Transport

3.2.2.1 The long term is the period starting with the end of the initial blowdown from the postulated pipe break. The transport of insulation is caused by the operation of the ECCS pumps. It is assumed that all of the LPCI and core spray pumps are operating at their maximum flow rates. This results in conservatively high flow velocities for the transport analysis.

3.3.2.2 It is further assumed that the shredded insulation is uniformly mixed with the suppression pool water at the end of the short term/beginning of the long term. This assumption is based on the even distribution of the blowdown from the downcomers and the turbulent mixing within the suppression pool and

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is consistent with the uniform distribution of insulation in the drywell during the blowdown.

3.3.2.3 At the beginning of the long term the initial blowdown has ended. The discharge from the downcomers during the long term is due to the overflow of the ECC systems from the drywell. This flow is evenly distributed by the vent header system and results in local turbulence near each downcomer. The containment spray and torus spray modes for the LPCI are used for long term containment cooling following a large break LOCA. The sprays provide an even distribution of fluid returning to the suppression pool causing only shallow surface turbulence. The bulk of the water in the suppression pool is subject to bulk flow velocities due to the removal of water by the ECCS pumps.

3.3.2.4 The suppression pool has ring girders approximately 2 feet deep at each mitered joint and at the mid cylinder of each section. The transport test data in NUREG/ CR-2791, (Reference 4.3) indicates that a flow velocity exceeding 0.3 ft/sec is required to entrain fibrous insulation shreds lying on the bottom of the suppression pool. The maximum flow velocity at the bottom of the suppression pool due to the bulk flow near each strainer is less than 0.3 ft/sec. Any insulation debris that sinks to the bottom outside the sections containing the strainers will not be transported to the strainers. The insulation that settles in the section between ring girders containing a strainer is conservatively assumed to collect on the strainer.

3.3.2.5 The maximum ECCS flow rate is used to determine the flow velocities. This minimizes the time available for settling of the debris and maximizes the flow velocities. The evaluation is based on the simultaneous operation of LPCI and core spray pumps at their runout flow rates.

4 LPCI pumps at 11,000 gpm/pump

4 core spray pumps at 4,015 gpm/pump

The resulting bulk flow velocity in the region near each strainer is 0.037 ft/sec toward each LPCI strainer and 0.014 ft/sec toward each core spray strainer.

3.3.2.6 Based on data in NUREG/CR-2791 (Reference 4.3), there are three types of shredded fibrous debris:

1. Debris that immediately sinks
2. Debris that slowly sinks
3. Debris that floats

The discussion in NUREG/CR-2982 (Reference 4.2) indicates that the debris absorbs water more readily and will sink faster in hot (120°F) water than water at ambient temperature.

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3.3.2.7 Tests performed by Owens-Corning on fibrous "NUKON" fragments discussed in Topical Report OCF-1, Reference 4.5, indicate that the fragments will readily sink after absorbing water and becoming saturated. The shredded debris that enters the suppression pool will be in contact with hot water. This debris is also mixed with the blowdown water and enters the suppression pool below the water surface. The rate of water absorption is also more rapid when the insulation is hot, which is the case for insulation from high temperature lines. Therefore, it is expected that most fragments will rapidly become saturated and settle. Also, because the fragments are thoroughly wetted in transport to the suppression pool, both the floating and slow sinking debris are considered to be slow sinking. Thus, for conservatism no credit is taken for floating debris preventing transport to the strainers.

3.3.2.8 Although there is considerable test data supporting the conclusion made in Section 3.3.2.7, that most "NUKON" fragments will rapidly settle to the bottom of the torus, it is acknowledged there is no specific LOCA test data available describing the post-LOCA buoyancy characteristics of "NUKON". In the absence of specific LOCA test data for "NUKON", the conservative approach is to assume less than all the "NUKON" fragments rapidly settle. In order to arrive at a credible and conservative factor for the lesser amount of debris that rapidly settles, comparable test data contained in Section 4.7.2 of NUREG/CR-2791 (Reference 4.3) describing the post-LOCA buoyancy characteristics of mineral wool insulation was used in the analysis. This data is considered conservative because fragments of as-manufactured fiberglas insulation, and "NUKON" in particular, will become wet and sink faster than mineral wool (i.e., fiberglas has a greater tendency to sink compared to mineral wool). This conclusion is based on data contained in NUREG/CR-2982 and Topical Report OCF-1, (References 4.2 and 4.5). There is no reason to believe LOCA effects would change the buoyancy characteristics of "NUKON" debris so that it would be more buoyant than mineral wool. NUREG/CR-2791 (Reference 4.3) states that 40% to 50% of the fibrous insulation (mineral wool) dislodged by a LOCA can be expected to immediately sink. The calculation, therefore, used the conservative factor of 40% to determine the amount of "NUKON" fragments that rapidly settle. The remaining 60% of the debris in the suppression pool was assumed to be slow settling.

3.3.2.9 The rapid settling debris is assumed to be dispersed uniformly in the suppression pool. The settling rate for this debris is .2 ft/sec based the average sink rate for saturated insulation in Topical Report OCF-1 (Reference 4.5). This settling rate and the bulk velocity of the suppression pool toward each strainer were used to determined the volume of rapid settling debris collecting on each strainer.

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3.3.2.10 The slow settling debris is assumed to be dispersed uniformly in the suppression pool. The settling rate for this debris is .017 ft/sec based on the average sink rate for very small clumps of insulation in Topical Report OCF-1 (Reference 4.5). This settling rate and the bulk velocity of the suppression pool toward each strainer were used to determine the volume of slow settling debris collecting on each strainer.

3.3.2.11 Based on the transport analysis, the maximum expected shredded debris to collect on the RHR strainers is 28.8% of the amount reaching the suppression pool, and that collecting on the core spray strainers is 14.6% of the amount reaching the suppression pool. The total collecting on the strainers is 43.4% of that in the suppression pool or 29% of the total shredded insulation generated.

3.4 ECCS SUCTION STRAINER HEAD LOSSES DUE TO INSULATION DEBRIS ACCUMULATION ON THE STRAINERS

3.4.1 The volume of insulation debris that will accumulate on each strainer is calculated. Each pump has a separate suction line with a strainer located in the torus as shown on figure 3-1. The design and dimensions of the strainers is shown in figure 3-2. The effective surface area for each LPCI strainer is 15.2 ft^2. The effective surface area for each core spray strainer is 5.64 ft^2. The total ECCS strainer area is 83.4 ft^2.

3.4.2 The head loss calculated for accumulation of fibrous insulation debris is based on thickness of a uniform accumulation on the effective surface. Reference 4.8 provides a head loss formula developed for a bed of shredded "NUKON" insulation. It is noted that the maximum approach velocity tested in the reference is 0.5 ft/sec, while the strainer approach velocity for HCGS is near 1.5 ft/sec. Review of the data in the reference indicates that it is reasonable to assume that the straight line logrithmic relationship between head loss and approach velocity can be extended to approach velocities near 1.5 ft/sec. This was confirmed in discussion with the insulation manufacturer. Therefore, the "NUKON" specific head loss formula is used. This formula is provided below:

$$H = 63.8\ (t_i)^{1.07}\ (V)^{1.79}$$

where

H = head loss, ft of water

t_i = equivalent accumulation thickness, ft

v = screen approach velocity, ft/sec

3.4.3 The head loss due to the accumulation of insulation on the strainers is provided in Table 3-1.

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3.5 EFFECT OF ACCUMULATION OF INSULATION DEBRIS ON ECCS PUMP NPSH.

3.5.1 The minimum NPSH is available when the suppression pool temperature is 212°F at 24,000 seconds. It is conservatively assumed that no noncondensables are added to the torus because of blowdown from the drywell and the noncondensables in the torus remain at the same temperature they were at prior to the blowdown. Therefore, the partial pressure exerted by the noncondensables is the same as existed prior to the LOCA.

3.5.2 Table 3-1 shows that the worst case insulation generation and the resulting accumulation on the strainers results in adequate NPSH for the ECCS pumps. The NPSH available for the LPCI pumps is 19.7 ft and the required NPSH is 7.5 ft. The NPSH available for the core spray pumps is 14.12 ft and the required NPSH is 3.5 ft. The NPSH required is taken from manufacturers certified performance data for the pumps. The required NPSH values in the FSAR and the GE Process Flow Diagrams contain a large safety margin above the requirement given by the pump manufacturers.

3.5.3 Non uniform distribution of the insulation in the torus was examined. The ECCS pumps have adequate NPSH available when 70% of the insulation debris reaching the suppression pool is distributed in one half the pool volume. The NPSH available was determined using the same assumptions as previously.

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4 REFERENCES

4.1 Serkiz, A.W., "Containment Emergency Sump Performance," NUREG-0897 (for comment), NRC, April, 1983.

4.2 Brocard, D.N., "Buoyancy, Transport, and Head Loss of Fibrous Reactor Insulation," NUREG/CR-2982, SAND 82-7205, Alden Research Laboratory, November, 1982.

4.3 Wysocki, J.; Kolbe, R., "Methodology of Evaluation of Insulation Debris Effects," NUREG/CR-2791, SAND 82-7067, Burns & Roe, Inc., September, 1982.

4.4 Final Safety Analysis Report (FSAR), Hope Creek Generating Station, Public Service Electric and Gas Company.

4.5 Owens-Corning Fiberglas Corporation, "Topical Report OCF-1, Nuclear Containment Insulation System, NU'K'ON," January, 1979.

4.6 ANSI/ANS-58.2-1980, "Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture".

4.7 ANSI/ANS-58.3-1977 (N182), "Physical Protection for Systems and Components Important to Safety".

4.8 Brocard, D.N., "Transport and Head Loss Tests of Owens-Corning NUKON Fiberglas Insulation, "Alden Research Laboratory, September, 1983.

4.9 NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants", US NRC, July, 1981.

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TABLE 3-1

Summary of Calculated ECCS Strainer Head Loss
Due to Accumulation of Fibrous Debris and
the Effect on ECCS Pumping $NPSH_A$

	ECCS STRAINER	PUMP FLOW gpm	DEBRIS GENERATED ft^3	DEBRIS VOLUME REACHING STRAINER ft^3 (1)	DEBRIS THICKNESS in.	DEBRIS HEAD LOSS ft	MINIMUM $NPSH_A$ WITH CLEAN STRAINERS ft	MINIMUM $NPSH_A$ WITH COVERED STRAINERS ft	$NPSH_R$ FOR PUMP ft
MAIN STEAM	LPCI	11,000	50.75	2.47	1.92	22.9	42.6	19.7	7.5
LINE BREAK	CS	4,015		1.21	2.52	30.13	44.25	14.12	3.5

(1) Value for a single strainer.

(2) Includes piping loss to the pump.

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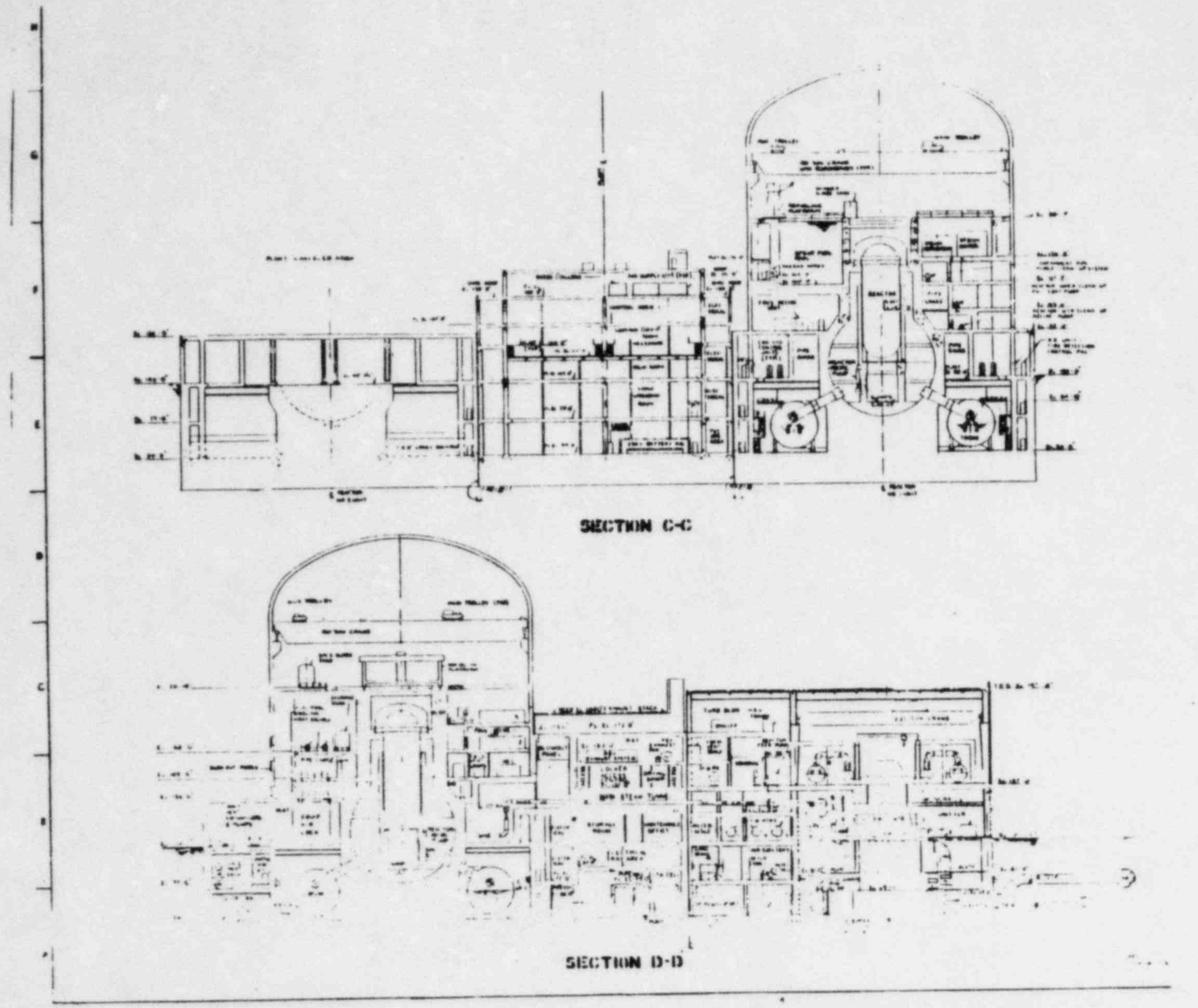

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HOPE CREEK
GENERATING STATION

GENERAL ARRANGEMENT
SECTION C C & D D

FIGURE 21

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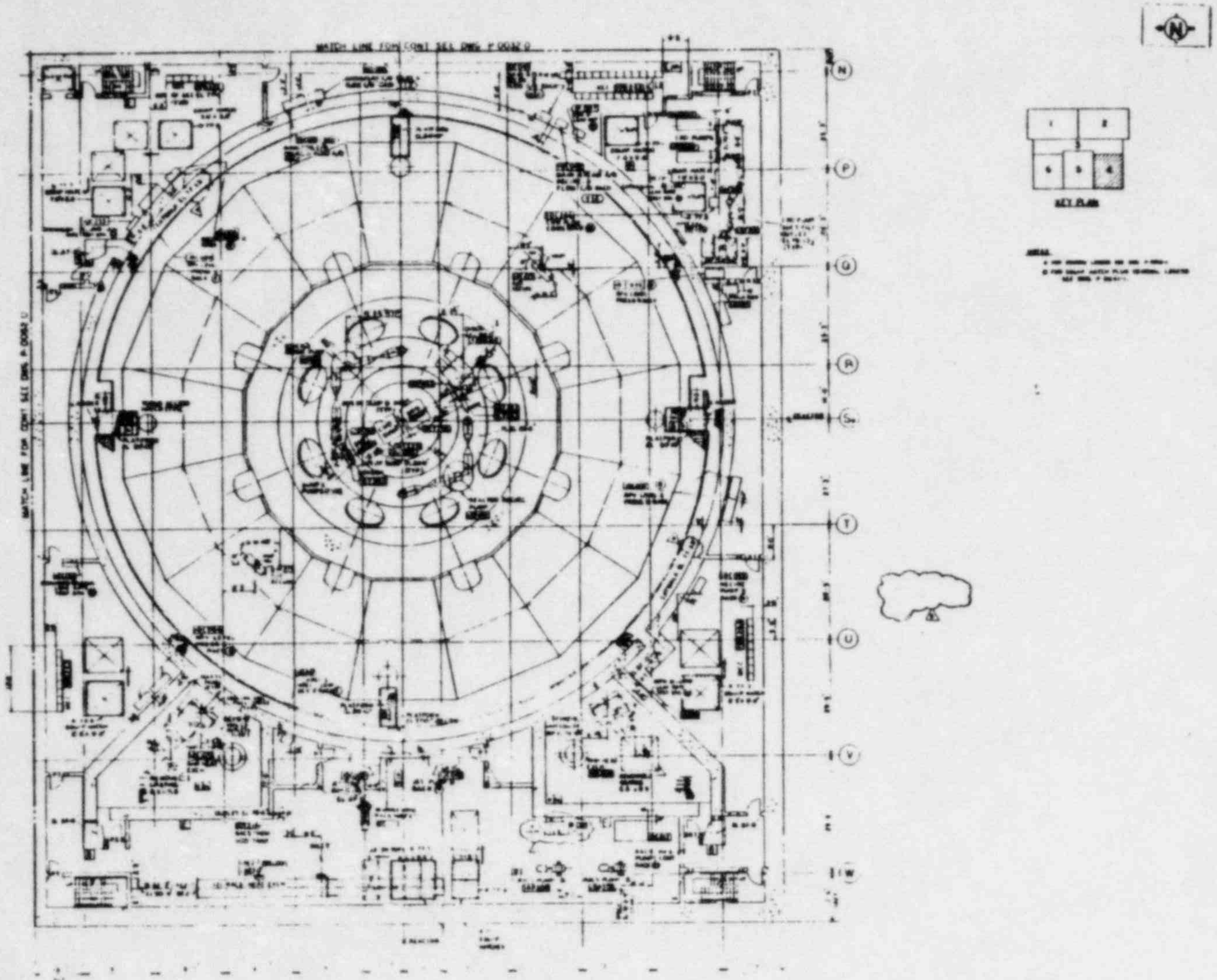
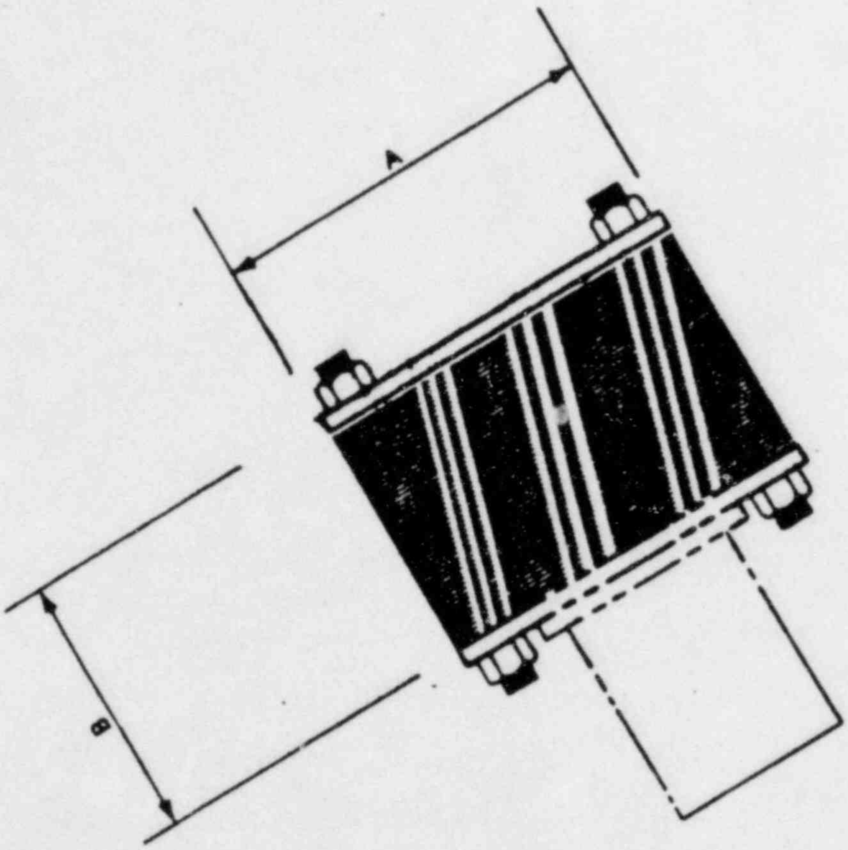


HOPE CREEK
GENERATING STATION

REACTOR BUILDING

FIGURE 2-2



STRAINER DIMENSIONS

	A	B
LPCI	32"	21 13/16"
CS	23 ½"	16 1/8"

HOPE CREEK
GENERATING STATION

SUPPRESSION POOL STRAINERS

FIGURE 2-3

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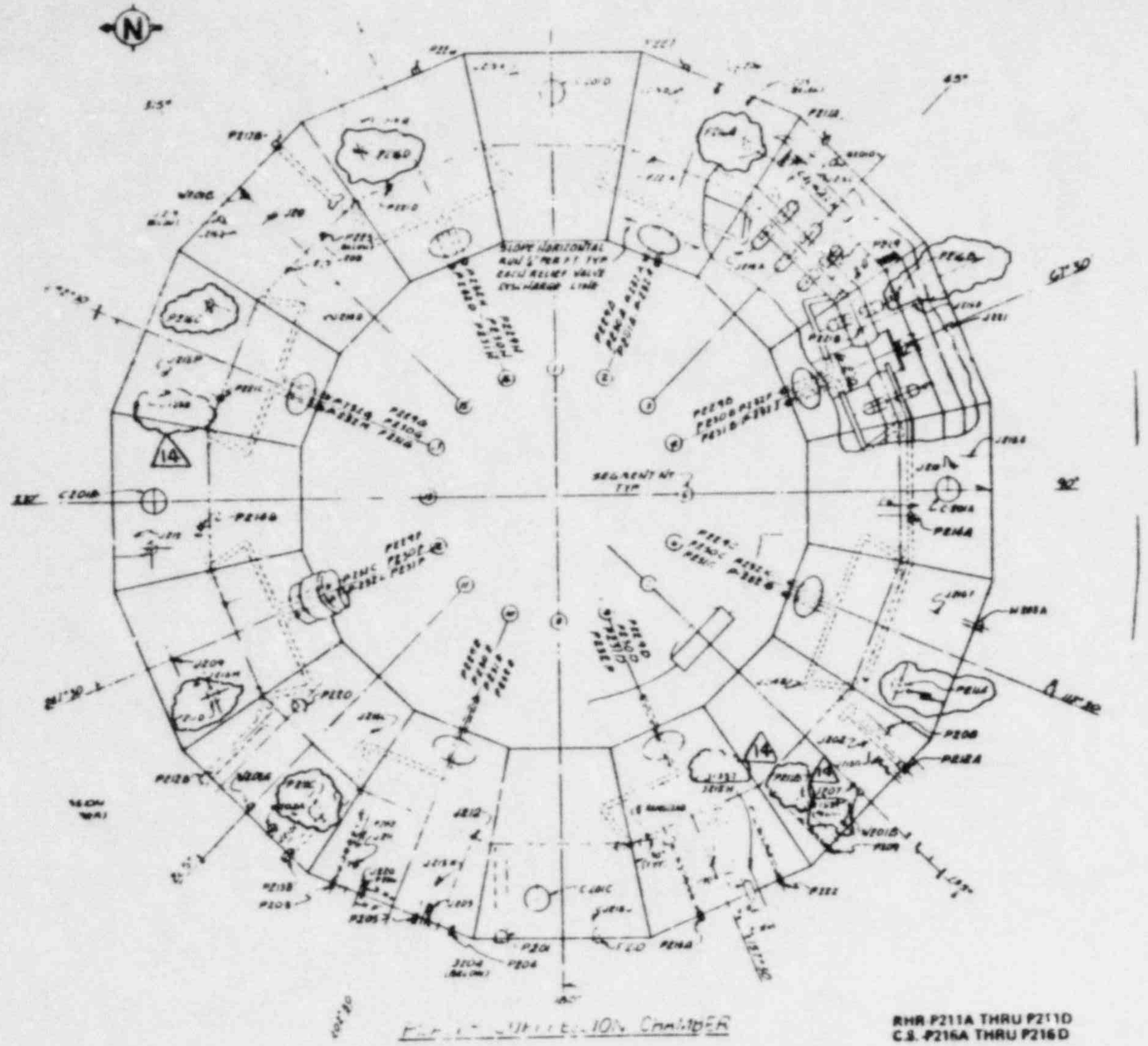


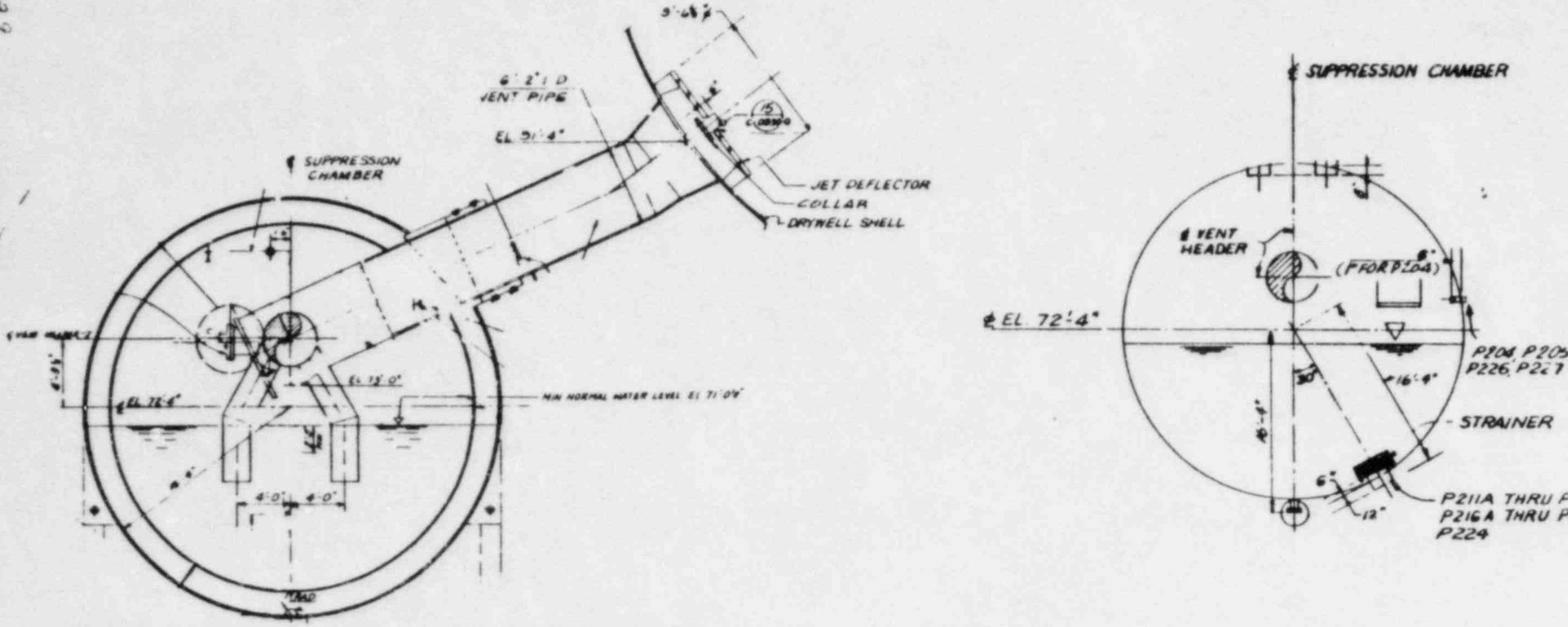
RHR P211A THRU P211D
C.S. P216A THRU P216D

HOPE CREEK
GENERATING STATION

SUPRESSION CHAMBER

FIGURE 2-4

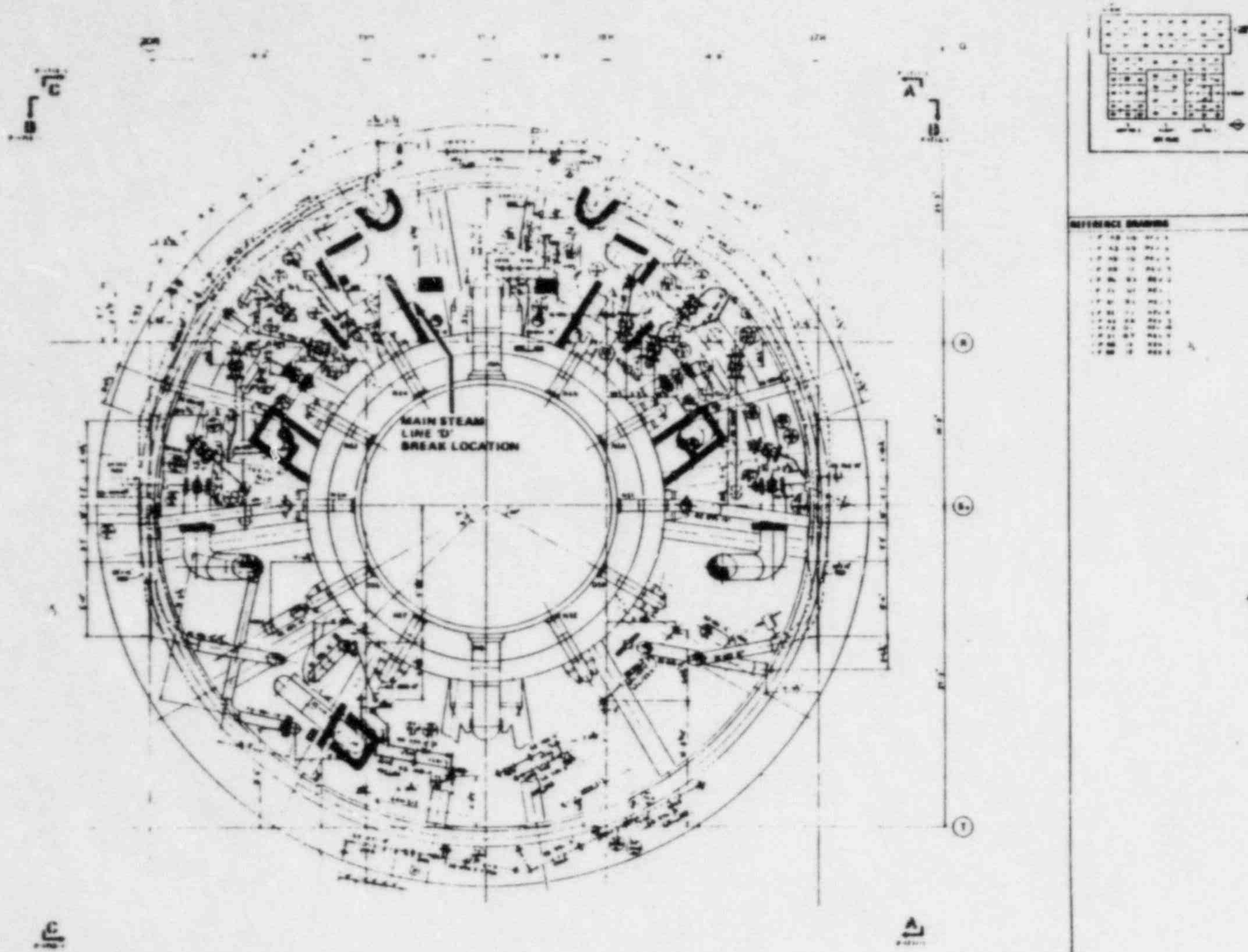
DSER OPEN ITEM 129



HOPE CREEK
GENERATING STATION

SUPRESSION CHAMBER
STRAINERS

FIGURE 2.5



HOPE CREEK
GENERATING STATION

BREAK LOCATIONS

FIGURE 3-1

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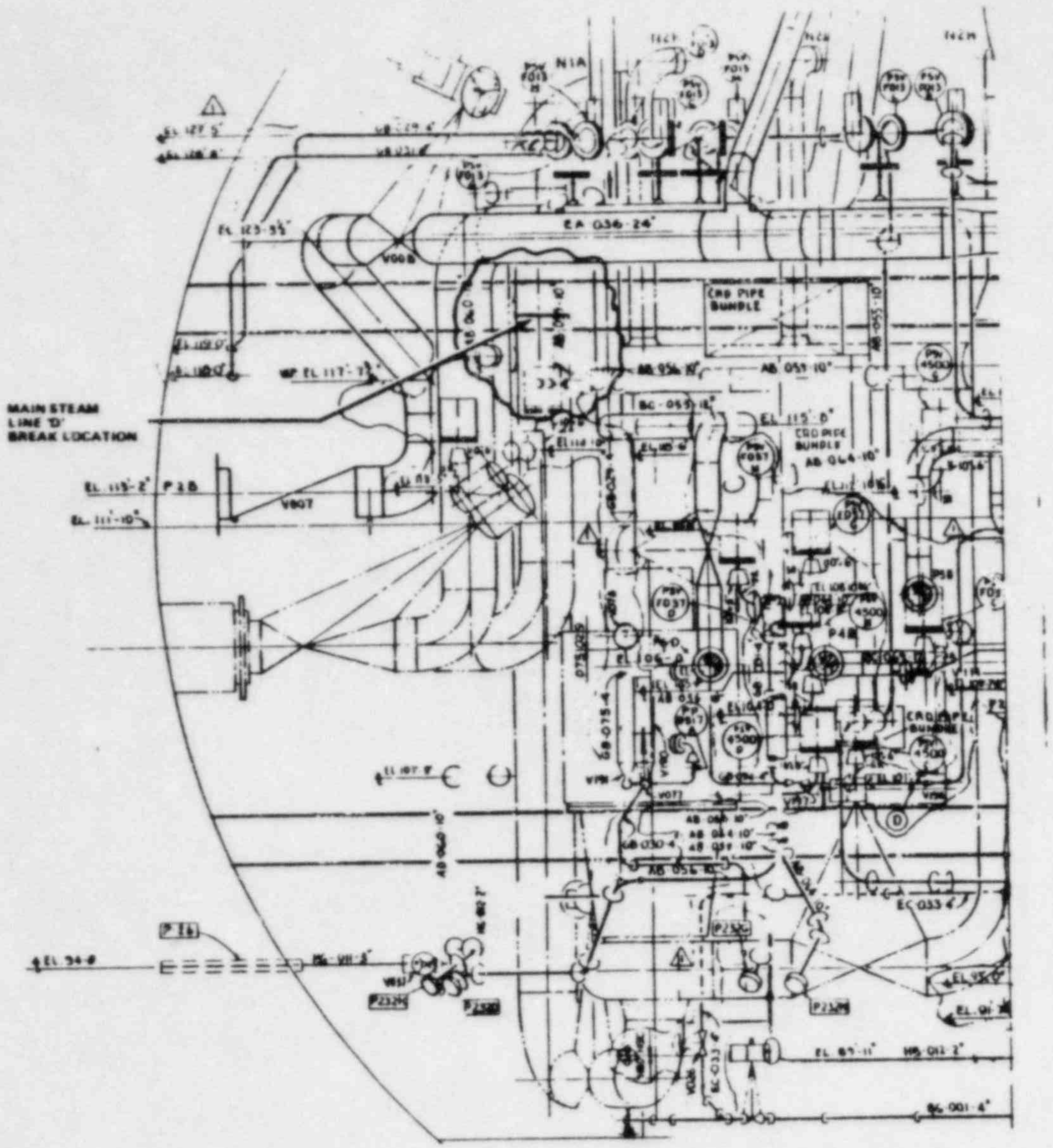
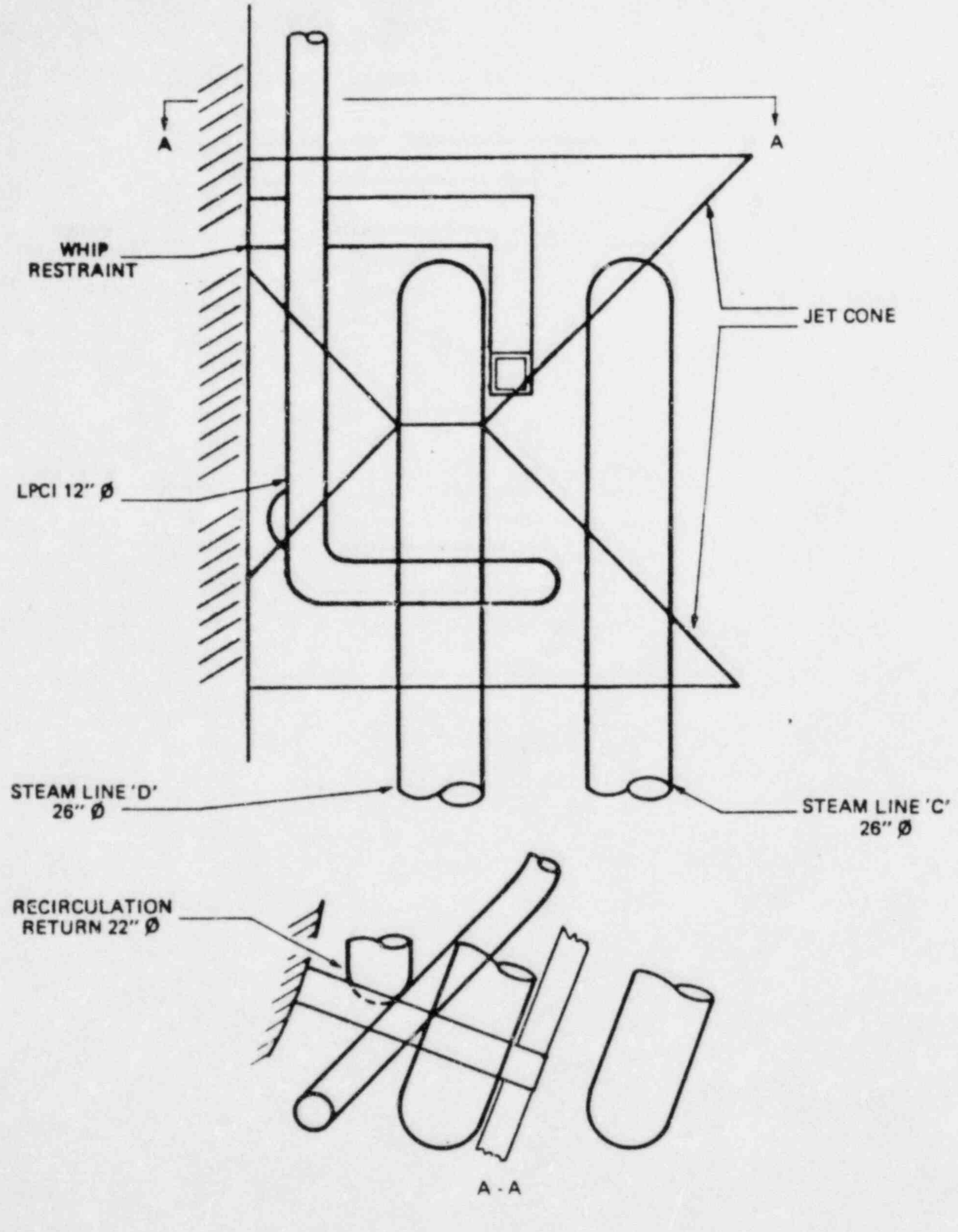


HOPE CREEK
GENERATING STATION

MAIN STEAM LINE

FIGURE 3-2



HOPE CREEK
GENERATING STATION

MAIN STEAM "D"

DSER OPEN ITEM 12?

FIGURE 3-3

6.2.2.2.2 Effects of Insulation on System Performance

Fiberglass blanket sections covered with 22-gauge, 304 stainless steel jacketing insulate structures, equipment, and piping within the primary containment. This form of insulation is not expected to create a debris-clogging problem for containment cooling operation after a LOCA. The vendor has studied the performance of the materials during a simulated design basis accident (DBA). The results of the study have been accepted by the NRC as Topical Report OCF-1, Nuclear Containment Insulation System. As indicated in the study:

a. The fiberglass insulation will not deteriorate or lose its mechanical integrity during a LOCA.

b. The operation of the emergency spray systems will not wash off or dislodge the insulation from piping, equipment, or structures.

c. Only those segments of insulation that are subjected to the violent forces of a component rupture, jet impingement, or pipe whip could be expected to become potential clogging debris. ~~No more than 50 blanket sections of the containment insulation inventory are expected to be so affected in a postulated LOCA.~~

INSERT d →

d. ~~Even if some of the blanket sections are transported to the suppression chamber ring header and from there to the suppression pool, the fiberglass will not clog the suction strainers. If whole blanket sections should lodge on a strainer, the material is porous and will not impede the suction flow. Since the insulation will sink, and since the strainer nozzles are offset up from the suppression chamber bottom, the insulation should not come to rest on the strainer inlet area. If some of the blanket sections should be shredded during the LOCA, the individual fibers will not bind together or to strainer surfaces due to the inert nature of the material. Therefore, fiberglass insulation does not constitute a potential strainer clogging problem.~~

e. If some fibers do pass through the 0.125-inch strainer mesh, the study has shown that pump function and spray nozzle performance are not affected. Particles of this size or smaller will not impair the safe function of

DSER OPEN ITEM 129

INSERT d

The only path for insulation to enter the suppression pool is through the vent pipes and the downcomer ring header. The jet deflectors prevent debris from entering the vent pipes directly. Floor grating, structural steel, and components in the drywell will retain insulation debris and prevent it from reaching the floor or the vent pipes. Only a small portion of the insulaton debris generated will actually be available for transport to the suppression pool. The openings at the jet deflectors will prevent all but small fragments from entering the vent pipes.

Much of the insulation debris that is transported to the suppression pool is fast settling, the remainder settles more slowly. After the initial blowdown the insulation debris that reaches the suppression pool begins to settle to the bottom. The flow velocity created by the ECCS pump operation for the bulk of the suppression pool is very low, therefore, only a portion of the insulation debris in the suppression pool will collect on the ECCS strainers. The strainers are located above the bottom of the suppression pool and the velocities generated at the bottom of the suppression pool are not sufficient to reentrain insulation that has settled except very near the strainers.

An evaluation of the transport and accumulation of postulated insulation debris was performed in accordance with the guidance of NUREG 0897 (issued for comment). This evaluation was transmitted under separate cover (R. L. Mittl, PSE&G, to A. Schwencer, NRC, dated May 15, 1984).

The flow restriction caused by the insulation accumulation on the strainers in this analysis does not adversely effect operation of the ECCS pumps.

DSER OPEN ITEM 129

DSER Open Item No. 152 (DSER Section 9.4.4)

RADIOACTIVITY MONITORING ELEMENTS

Turbine enclosure ventilation system

RESPONSE

This item is not an open item per telephone conversation between J. M. Ashley (PSE&G) and John Ridgely (NRC-ASB) on March 22, 1984.

TELEPHONE NOTES

PSE&G Hope Creek Licensing (Bethesda)

Date: March 22, 1984

From: J.M. Ashley

To: D. Wagner, J. Ridgely (ASB)

Subject: HCGS DSER Open Items

Discussion

Ashley called to find out what NRC concerns existed with respect to FSAR Sections 3.5.1.2 (Item 30), 9.2.2 (Item 145) and 9.4.4 (Item 152).

Ridgely explained that these items were inadvertently listed as open items in the listing of open items at the front of the DSER. The NRC has no outstanding concerns with the sections.

J. M. Ashley

DSER OPEN ITEM 152

DSER Open Item No. 154 (DSER Section 9.5.1.4.a)

METAL ROOF DECK CONSTRUCTION CLASSIFICATION

Metal roof deck construction is noncombustible, but is not listed as "acceptable for fire" in the UL Building Materials Directory and, therefore, is not consistent with Section C.5.a(10) of BTP CMEB 9.5-1. The staff will require the applicant to provide metal roof deck construction that is classed "acceptable for fire" in the UL Building Materials Directory or that meets the criteria for Class 1 roof deck systems in the FM system approval guide.

RESPONSE

Metal roof deck construction for HCGS meets the criteria for Class 1 roof deck systems outlined by Factory Mutual's systems approval guide. Therefore, HCGS complies with Section C.5.a(10) of Branch Technical Position CMEB 9.5-1.

FSAR Section 9.5.1.1.7 has been revised to address compliance with the above requirement.

9.5.1.1.6 Cable Spreading Room

The cable spreading room (CSR) is separated from other areas of the plant by 3-hour-rated fire barriers. Fire detection and an automatic preaction sprinkler system are provided. Cabling to the remote shutdown panel room is independent of the CSR and provides the necessary means to attain a safe shutdown if the CSR is lost.

9.5.1.1.7 Building Materials Selection

Interior walls, partitions, structural components, thermal insulation materials, and radiation shielding materials are noncombustible. Areas containing systems or equipment required for safe shutdown are either unfinished or finished with noncombustible materials.

Suspended acoustical ceiling panels are Underwriters Laboratories, Inc (UL)-listed and have a flame spread, fuel contribution and smoke development rating of 25 or less. Suspended ceiling supports are noncombustible.

listed as Class I
Metal deck roof construction is noncombustible and ~~has reinforced concrete slab over metal deck form work.~~
by Factory Mutual (FM) Systems approval guide.

9.5.1.1.8 Protection from Transformer Fires

Medium and low voltage-amperage transformers located indoors are dry and air-cooled. Oil-filled medium voltage-amperage transformers (main and station service transformers) are located outdoors near the turbine building and circulating water pump structure. The turbine building and circulating water pump structure contain nonsafety-related systems.

All main and station service transformers are provided with individual water spray systems and are separated from each other by a 1-hour fire barrier. Each transformer has a collection dike and drainage outlet for collecting transformer oil spills and fire suppression system water and draining it to the oily waste drainage system.

DSER OPEN ITEM 154

DSER Open Item No. 159 (DSER Section 9.5.1.5.a)

PRIMARY AND SECONDARY POWER SUPPLIES FOR FIRE DETECTION SYSTEM

Primary and secondary power supplies for the detection system in accordance with NFPA 72D, which the staff references in Section C.6.a (6) of its guidelines, have not been provided. The staff will require that primary and secondary power supplies for the fire detection system satisfy the provisions of Section 2220 of NFPA 72D.

RESPONSE

As indicated in Section 9.5.1.2.15, the fire detection system is supplied from an inverter system which has batteries and SDG-backed MCCs as power supplies. Section 9.5.1.6.16 has been revised to provide discussion on compliance with NFPA 72D requirements.

removal of a detection device from a detector circuits, and power failure. If any of the above problems occur, a fire detection system trouble is annunciated locally and in the main control room. Plant operation will periodically test the system for proper functioning, similar to inservice testing of other plant systems.

At HCGS, the fire and smoke detection system is in compliance with NFPA 72D except that the operation and supervision of the system is not the sole function of the plant operator. The plant operator's duties cover operation of the generating station and monitoring and supervising the fire protection systems.

9.5.1.6.15 Paragraph C.6.a.(3)

Paragraph C.6.a.(3) requires that the fire detectors be installed in accordance with NFPA 72E.

At HCGS, the location of early warning fire and smoke detectors was determined and performed under the direction of a registered fire protection engineer. The location of the fire and smoke detectors complies with the guidelines of NFPA 72E except for the location of ionization and photoelectric detectors in high-bay areas. The detectors are not located in each bay formed by deep beams. NFPA 72E allows detector locations to be determined based on engineering judgement considering ceiling shape, ceiling surfaces, ceiling height, configuration of contents, combustible characteristic and ventilation.

At locations in areas where composite construction is used, the diffusion of combustion particulates throughout the compartment volume produced during the incipient and smoldering stages of the fire will negate the effect of beam depth and result in acceptable levels of detection coverage.

9.5.1.6.16 Paragraph C.6.a.(6)

Paragraph C.6.a.(6) requires primary and secondary power supplies be provided for the fire detection system and electrically operated control valves conforming to NFPA 72D.

INSERT A →

DSER OPEN ITEM 159

INSERT A

At HCGS, the fire detection system is supplied with uninterruptible 120 volt ac power fed from an inverter type system which has three power supplies. The normal or primary power supply is from an offsite source and the alternate or secondary power supply is from a 4-hour station battery supply. A third power supply serves as backup to the primary and secondary power supplies. In addition, both the primary and backup power supplies are connected to buses which are backed by standby diesel generators (SDGM). The buses are disconnected from the SDGs during a LOCA event; however, the buses can be reconnected to the SDGs under administrative control. Figure 8.3-11, Sheet 3, is the single line diagram of the power supplies to the fire detection system equipment. Therefore, the fire detection system is furnished with power supplies which meet the NFPA 72D requirements.

DSER OPEN ITEM 159

DSER Open Item No. 161 (DSER Section 9.5.1.5.b)

FIRE WATER VALVE SUPERVISION

Supervision has not been provided for all valves in the fire protection water supply system in accordance with NFPA 26. To meet staff guidelines in the Section C.6.c of BTP CMED 9.5-1, the type of valve supervised and the frequency at which its position is verified should be as listed.

RESPCNSE

All valves in the fire protection water supply system are supervised in accordance with NFPA 26 except locked valves which are inspected to verify valve condition. FSAR Section 9.5.1.2.3.4 has been revised to indicate this.

The outdoor, underground yard loop was designed in accordance with NFPA 24. The yard loop consists of 12-inch diameter cement mortar-lined ductile iron pipe that extends around the power block. Post-indicator valves are provided for sectional control. Two-way hydrants, controlled by individual curb box valves, are installed on the yard loop at maximum intervals of 250 feet. A hose house is provided for each hydrant and equipped with 200 feet of hose, fittings, and accessories in accordance with NFPA 24.

9.5.1.2.3.4 Water Supply for Automatic and Manual Sprinkler/Spray Systems

Automatic and manual sprinkler/spray systems headers are connected to the in-plant loops that are fed from the main underground fire protection water piping or yard loop by two separate supplies. The in-plant loops are 8-inch and 10-inch lines. Since the in-plant loops are fed by two separate supplies, they are considered an extension of the main underground yard loop. Automatic and manual sprinkler/spray systems and hose standpipe systems serving a single safety-related area have takeoffs from an in-plant loop, separated by sectional control valves. The header arrangement is such that, by manual positioning of the sectional valves, no single piping failure can impair both the primary and backup fire protection provided for a single area.

AC power supply for sprinkler/spray system control panels including the fire status panel in the control room is provided from a non-Class 1E inverter. The inverter is fed by non-Class 1E batteries and non-Class 1E motor control centers (MCCs) backed by standby diesel generators (SDGs). Loss of normal ac power will not prevent the panels or systems from operating.

An outside screw and yoke (OS&Y) gate valve for each sprinkler and deluge system is located adjacent to the system control or alarm valve. The branch connection into the building is provided with a post-indicator valve at the connection to the yard loop. Each sprinkler and deluge system is provided with local water flow alarms and remote annunciation in the main control room.

INSERT →

9.5.1.2.4 Wet Pipe Sprinkler Systems

Wet pipe sprinkler systems are provided for the plant areas listed in Table 9.5-2. The density coverage and installation for

DSER OPEN ITEM 161

SER Open Item No. 161

INSERT

Control and sectionalizing valves in the fire protection water system are either electrically supervised or administratively controlled in accordance with NFPA 26. The valves that are electrically supervised are those valves that control the water suppression system and the valves in the fire pump suction and discharge lines located in the fire pump house. These valves are shown on Figures 9.5-13 through 9.5-16 and 9.5-18. The electrically supervised valves are provided with normally open contacts that close in the event of valve movement. The electrical supervision signal is indicated on the local control panels and registers as a system trouble on the fire protection status panel in the main control room.

The valves that will be administratively controlled are the post indicator valves in the yard area that provide sectional control of the fire main loop and fire water supply lines branching into various buildings, the sectional valves in the in-plant loop and supply piping, and the valves that control the water supply to standpipe and hose systems. These valves are padlocked in the open or closed position so that they cannot be inadvertently operated. The control valves for the standpipe and hose system in the reactor building and intake structure are normally closed to maintain these systems in a dry condition.

Valves are either electrically supervised, or locked and inspected monthly. documentation recording this inspection will be made.

DSER Open Item No. 162 (DSER Section 9.5.1.5.c)

DELUGE VALVES

The applicant is not providing approved deluge valves for the deluge systems. This is not in accordance with NFPA 13 which the staff references in its guidelines. The staff will require the applicant to provide deluge valves approved by a nationally recognized testing laboratory as components for fire protection systems, as specified by NFPA 13, which the staff references in Section C.6.c of BTP CMEB 9.5-1.

RESPONSE

The deluge valves for the HCGS water spray, protection and deluge system are Viking Corporation Model D-5 water control valves which are UL listd per the 1983 UL Fire Protection Equipment Directory. Sections 9.5.1.2.5, 9.5.1.2.6, 9.5.1.2.7 and 9.5.1.2.8 have been revised to reflect this.

Section 9.5.1.2.7 has been revised to clarify that the electric-motor-operated valves in the preaction water spray systems are not deluge valves.

the sprinkler systems are in accordance with NFPA 13. Each sprinkler system is provided with an alarm check valve or flow switch that annunciates in the main control room. OS&Y gate valves serving as shutoff valves to automatic sprinkler systems are supervised with any problems annunciated in the main control room.

Wet pipe sprinkler system operation is initiated upon a rise in ambient temperature to the melting point of fusible links on sealed sprinkler heads, thus causing the spray heads to open. The flow of water through an alarm check valve or flow switch energizes a local alarm and registers an alarm condition on the fire monitor panel in the main control room. Once initiated, the wet sprinkler system operation is terminated manually by shutting either a gate valve external to the hazard or a post-indicator valve outdoors.

9.5.1.2.5 Water Spray Systems

Water spray systems are provided for the plant areas or equipment listed in Table 9.5-2.

The water spray systems have directional solid cone UL-listed spray nozzles or perforated pipe. The water flow is controlled by deluge valves. A system alarm and a valve position alarm for supervised OS&Y gate valves for each spray system are provided in the main control room. Spray densities and installation complies with NFPA 13 and 15.

Operation of the automatic spray systems is initiated by a temperature sensor. This sensor detects a rapid rise in ambient temperature and/or attainment of a fixed high temperature and releases a tripping device to open the deluge valve, thus supplying water under pressure to the open spray nozzles. Actuation of a sensor also initiates a local alarm, and registers the alarm condition on the fire protection status panel in the main control room, independently of water flow in the system. Water flow in the system initiates a local alarm and registers the system-actuated condition on the fire protection status panel in the main control room independent of the detection alarm.

Manual release of the deluge valve tripping device also initiates local and remote water flow alarms. System operation is terminated by manually closing a gate valve external to the hazard area.

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Operation of the manual water spray systems is initiated by a pushbutton on the local panel and opening a normally closed OS&Y gate valve. A temperature sensor detects a fixed high temperature and registers the alarm condition on the local control panel and in the main control room. The system is activated by operating a pushbutton on the local control panel and opening the OS&Y gate valve. Water flow in the system initiates a local alarm and registers the system-actuated condition in the main control room independent of the detection alarm. System operation is terminated by manually closing the OS&Y gate valve external to the hazard area.

9.5.1.2.6 Deluge Systems

Deluge systems are provided for the diesel fuel tank rooms as listed in Table 9.5-2.

The deluge systems have open sprinkler heads. Water flow is controlled by a UL-listed deluge valve actuated by a local manual switch, and a normally closed OS&Y gate valve. A system alarm, and a valve position alarm on supervised OS&Y gate valves for each deluge system is provided in the main control room. The density coverage and installation for the systems are in accordance with NFPA 13.

Water flow in the system initiates a local alarm and registers the system-actuated condition on the fire protection status panel in the main control room.

System operation is terminated by manually closing a gate valve external to the hazard area.

9.5.1.2.7 Manual Preaction Water Spray Systems

Manual preaction water spray systems serve the reactor building. Specific equipment covered is listed in Table 9.5-2.

Individual hazards are protected by fixed water spray nozzles on dry piping at atmospheric pressure. Each individual system is controlled by an electric-motor-operated valve. ~~serving as a deluge valve.~~ Each electric-motor-operated ~~deluge~~ valve is connected to a common header system within the reactor building. The header is pressurized to 20 psig with air. Water supply to the header is controlled by a

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UL-listed (deluge)

preaction valve assembly with a building penetration having one normally open isolation valve. Fixed temperature line-type heat detectors are used and any fire condition is annunciated in the main control room. Alarms are provided for low air pressure and for the closed condition of supervised OS&Y gate valves. Water flow is detected by a pressure switch downstream of the motor-operated valve and is annunciated in the main control room. Design and installation comply with NFPA 13 and 15.

The manual preaction water spray system operation is initiated by either a fixed high temperature thermostat, or a manual switch in the main control room, which actuates the preaction valve and charges the system with water to the inlet of each individual hazard valve. No water is discharged through the closed hazard valve at this time. Any fire condition is annunciated in the main control room.

High temperature due to fire condition at any individual hazard activates the local fixed high-high temperature thermostat and annuniates a fire condition locally and in the main control room. The hazard valve (motor-operated gate valve) is opened manually by a pushbutton on the local control panel and water is discharged onto the hazard.

When the fire is controlled and the environment is cooled to a temperature below the thermostat, the fire alarms at the local control panel are silenced and the fire indicating lights in the main control room are turned off. The discharge of water may then be stopped by a manual pushbutton on the local control panel which closes the hazard valve (motor-operated gate valve). The ball-type drip valve at the low point of the open piping system automatically drains the system downstream of the hazard valve into the radwaste drainage system. The main header remains pressurized.

The system is capable of being reset and returned to normal status without entering the hazard area as follows:

a. The water spray portion of the system is reset and drained automatically.

b. The fire main gate valve is manually closed and the normally closed drain valve is manually opened, draining the piping downstream of the preaction valve assembly into the radwaste drainage system.

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c. The preaction valve is manually reset, the header is pressurized with supervisory air, and when the fire main valve is manually opened, the entire system is returned to full-service status.

9.5.1.2.8 Preaction Sprinkler Systems

Preaction sprinkler systems are provided for plant areas or equipment as listed in Table 9.5-2.

UL-listed (deluge)

Preaction sprinkler system operation is initiated by heat actuating devices located in the hazard area, which actuates the preaction valve and charges the system with water up to the closed, fusible link sprinkler heads. No water is discharged to the hazard area at this time. Any fire condition is annunciated in the main control room. High temperature due to fire condition melts one or more of the fusible link sprinkler heads and water discharges onto the hazard.

When the fire is controlled, water discharge is terminated by manually closing the fire main gate valve, and the normally closed test and drain valves are opened, draining the system. Used sprinkler heads are replaced, the preaction valve is manually reset, and the header is pressurized with supervisory air.

9.5.1.2.9 Wet Standpipes and Hose Stations

Wet standpipes for fire hoses are designed in accordance with NFPA 14. Standpipes are installed adjacent to stairwells, exits, and other points in all normally accessible areas in plant buildings. Four-inch standpipes are provided for three or more hose connections, and 3-inch standpipes are provided for one or two hose connections. The standpipe hose connections are equipped with 1-1/2-inch hose valves and 75 or 100 feet of 1-1/2-inch woven jacket lined hose with spray nozzles.

Wet standpipes are maintained in a dry condition in the reactor building and the intake structure.

Adjustable spray nozzles with shutoff capabilities, UL-listed for Class C fires, are provided.

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DSER Open Item No. 163 (DSER Section 9.5.1.5.c)

MANUAL HOSE STATION PIPE SIZING

Manual hose stations are located throughout the plant in accordance with NFPA 14. Three-inch-diameter piping is used to serve up to two hose stations in some areas. This does not meet staff guidelines. The staff will require the applicant to provide 4-in. diameter piping consistent with the guidelines in Section C.6.c(4) of BTP CMEB 9.5-1.

RESPONSE

At HCGS the pipe size for the wet standby system meets NFPA 14 requirements. Also, as stated in FSAR Section 9.5.1.6.21, all standpipe connections to the in-plant loop are 4-inch diameter and feed multiple hose connections. Except for one instance, branches off the standpipes that feed one or two hose connections are 3-inch diameter. The as-built plant configuration has one 3-inch standpipe connection, with 3 hose connections to it. These 3 hose stations are not all on the same floor and all 3 hose stations could not be used to fight the same fire. This 3 inch branch has been evaluated and found acceptable to meet NFPA 14 flow and pressure requirements. As stated in FSAR Section 9.5.1.6.19, the fire water supply can provide water at the required flow and pressure to supply any hydraulically designed sprinkler or deluge system and all hoses which can be used to fight the same fire.

FSAR Section 9.5.1.2.9 has been revised to clarify the design of HCGS wet standpipe system.

c. The preaction valve is manually reset, the header is pressurized with supervisory air, and when the fire main valve is manually opened, the entire system is returned to full-service status.

9.5.1.2.8 Preaction Sprinkler Systems

Preaction sprinkler systems are provided for plant areas or equipment as listed in Table 9.5-2.

Preaction sprinkler system operation is initiated by heat actuating devices located in the hazard area, which actuates the preaction valve and charges the system with water up to the closed, fusible link sprinkler heads. No water is discharged to the hazard area at this time. Any fire condition is annunciated in the main control room. High temperature due to fire condition melts one or more of the fusible link sprinkler heads and water discharges onto the hazard.

When the fire is controlled, water discharge is terminated by manually closing the fire main gate valve, and the normally closed test and drain valves are opened, draining the system. Used sprinkler heads are replaced, the preaction valve is manually reset, and the header is pressurized with supervisory air.

9.5.1.2.9 Wet Standpipes and Hose Stations

INSERT A

Wet standpipes for fire hoses are designed in accordance with NFPA 14. Standpipes are installed adjacent to stairwells, exits, and other points in all normally accessible areas in plant buildings. Four-inch standpipes are provided for three or more hose connections, and 3-inch standpipes are provided for one or two hose connections. The standpipe hose connections are equipped with 1-1/2-inch hose valves and 75 or 100 feet of 1-1/2-inch woven jacket lined hose with spray nozzles.

INSERT B

INSERT C →

Wet standpipes are maintained in a dry condition in the reactor building and the intake structure.

Adjustable spray nozzles with shutoff capabilities, UL-listed for Class C fires, are provided.

DSER OPEN ITEM 163

At HCGS, each sprinkler and deluge system is provided with an OS&Y gate valve adjacent to the system automatic control or alarm valve. The branch connection into the building is provided with a post indicator valve at the connection to the fire main loop. Each sprinkler and deluge system is provided with local water flow alarms (using pressure or flow switch) and remote annunciation in the main control room.

An OS&Y gate valve is provided at each branch (off the in-plant fire main loop) supplying the sprinkler, deluge or standpipe system.

The standpipe systems are not provided with water flow alarms. Waterflow in the standpipe systems is indicated by pump running annunciation in the main control room without automatic system actuation annunciation.

9.5.1.6.21 Paragraph C.6.c.(4)

Paragraph C.6.c.(4) requires individual standpipes be at least 4 inches in diameter for multiple hose connections and 2 1/2 inches in diameter for single hose connections.

At HCGS, all standpipe connections to the in-plant loop are 4 inch diameter for standpipes feeding multiple hose connections. See Figures 9.5-13 through 9.5-18. But branches off the standpipes, that feed two or less hose connections, are 3 inches in diameter. As stated in Section 9.5.1.6.19, the fire water supply can provide water at the required flow and pressure to supply any sprinkler or deluge system and all the hoses which can be used to fight the same fire.

INSERT D

9.5.1.6.22 Paragraph C.6.c.(4)

Paragraph C.6.c.(4) requires that provisions be made to supply water at least to standpipes and hose connections for manual firefighting in areas containing equipment required for safe plant shutdown in the event of a safe shutdown earthquake (SSE). The firewater piping serving such hose stations should be analyzed for SSE loading, and should be provided with supports to ensure system pressure integrity.

DSER OPEN ITEM 163

DSER Open Item No. 163

Insert A

to provide 65 psig at the topmost outlet of the hose standpipe system with 100 gpm flowing from the outlet.

Insert B

except in one instance where 3 hoses are connected to a 3 inch branch. (These 3 hose stations are not on the same floor and cannot be used to fight the same fire.)

Insert C

The fire water supply can provide water at the required flow and pressure to supply any hydraulically designed sprinkler or deluge system and all the hose streams which can be brought to bear on the same fire. See Sections 9.5.1.6.19 and 9.5.1.6.21.

Insert D

except in one instance where 3 hoses are connected to a 3 inch branch. These 3 hose stations are not all on the same floor and could not be used to fight the same fire. This 3 inch branch has been evaluated and found acceptable to meet NFPA 14 pressure and flow requirements.

DSER Open Item 164 (DSER Section 9.5.1.6.e)

REMOTE SHUTDOWN PANEL VENTILATION

The remote shutdown panel room is supplied by an HVAC system that also supplies the control room. A single fire could disable both areas.

To meet our guidelines in Section C.7.f of BTP CMEB 9.5-1, we will require the applicant to provide a ventilation system for the remote shutdown panel that is isolated from the main control room.

RESPONSE

The remote shutdown panel (RSP) room and the control room are served by independent HVAC systems as described in FSAR Sections 9.4.3.1.3 and 9.4.3.2.1 respectively.

The RSP room and its associated HVAC unit are separated by a 3 hour fire-rated barrier. A 3 hour fire-rated floor between the main control room HVAC and RSP room HVAC supply units meets the criteria for separation outlined in Branch Technical Position CMEB 9.5.1. A single fire will not affect both the MCR and RSP rooms.

DSER Open Item No. 165 (DSER Section 9.5.1.6.g)

EMERGENCY DIESEL GENERATOR DAY TANK PROTECTION

A 550-gal. fuel oil day tank is provided in each diesel generator room. No enclosure or dike is provided for the day tanks. This is not consistent with staff guidelines. The staff will require that the applicant protect the day tanks in accordance with its guidelines in Section C.7.i of BTP CMEB 9.5-1.

RESPONSE

A dike has been provided for each day tank. Section 9.5.1.2.26 has been revised to indicate this.

provided in the vicinity of battery rooms. The "low air flow" remote alarm in the control room registers fan failure conditions. Automatic fire detectors alarm locally, and alarm and annunciate in the main control room.

9.5.1.2.25 Turbine Lubrication and Control Oil Storage and Use Areas Fire Protection

The turbine lubrication and control oil storage and use areas are remote from areas containing safety-related systems, and are separated by 3-hour fire barriers with Class A fire doors.

The condenser area, secondary condensate pump area, RFPT lube oil reservoir and purifier rooms, and RFPT rooms are protected with wet pipe sprinkler systems. The lube oil storage room and the main turbine lube oil reservoir and purifier room are protected by a water spray system.

9.5.1.2.26 Diesel Generator Area Fire Protection

The diesel generators are separated from each other and other areas of the plant by 3-hour fire barriers with Class A fire doors.

One 550-gallon capacity diesel generator fuel oil day tank is located in each diesel generator room. An automatic fixed carbon dioxide total flooding system is provided in each diesel generator room. Manual water hose stations are provided as a backup fire suppression system. Photoelectric and infrared detectors that alarm locally and annunciate in the main control room are provided in each diesel generator room.

INSERT A →

Each diesel generator room drains via normally closed isolation valves to a common drainage sump pump basin that has a sump pump capable of discharging 100 gpm.

The normal ventilation system can be used for manual smoke venting. Each supply and return duct is provided with ETL-operated fire dampers.

DSER OPEN ITEM 165

SER Open Item No. 165

Insert A

Each fuel oil day tank is provided with a dike which surrounds the floor area under the tank. The dike area is also below the equipment access grating that surrounds the diesel generator. Three sides of the dike are made of a 6 inch channel that is bolted to the floor with a neoprene oil-resistant gasket. The fourth side is the east wall of the diesel generator room. The floor area within the dike is sloped to a sump area located in the middle of the dike area. No drain is provided to drain the dike area or the sump. However, the dike has sufficient capacity to hold 110 percent of the contents of the fuel oil day tank.

DSER Open Item No. 182 (DSER Section 15.9.10)

TMI-2 ITEM II.K.3.18

The applicant should specify which option they are planning to implement. Either option 2 or option 4 is acceptable to the staff.

RESPONSE

Option 4 will be implemented at HCGS, as indicated by our response to Question 421.12.

DSER Open Item No. 185 (DSER Section 7.2.2.2)

TRIP SYSTEM SENSORS AND CABLING IN TURBINE BUILDING

The applicant is required to describe the separation utilized between redundant channels listed below and demonstrate that the design can withstand the effects of missiles, HELB and seismic events in a way that is consistent with satisfying the safety analysis described in FSAR Chapter 15.

IDENTIFICATION	DESCRIPTION	LOCATION
SB-PT-N052A	Main (Turbine) Stop Valve Closure and Turbine Control Valve Fast Closure Trips Bypass	Turbine Building
SB-PT-N052B	Main (Turbine) Stop Valve Closure and Turbine Control Valve Fast Closure Trips Bypass	Turbine Building
SB-PT-N052C	Main (Turbine) Stop Valve Closure and Turbine Control Valve Fast Closure Trips Bypass	Turbine Building
SB-PT-N052D	Main (Turbine) Stop Valve Closure and Turbine Control Valve Fast Closure Trips Bypass	Turbine Building
SM-PT-N076A	MSIV*-Low Steam Line Pressure Trip (PCRVICS)**	Turbine Building
SM-PT-N076B	MSIV*-Low Steam Line Pressure Trip (PCRVICS)**	Turbine Building
SM-PT-N076C	MSIV*-Low Steam Line Pressure Grip (PCRVICS)**	Turbine Building
SM-PT-N076D	MSIV*-Low Steam Line Pressure Trip (PCRVICS)**	Turbine Building
SM-PT-N075A	MSIV*-Low Condenser Vacuum Trip (PCRVICS)**	Turbine Building
SM-PT-N075B	MSIV*-Low Condenser Vacuum Trip (PCRVICS)**	Turbine Building

DSER Open Item No. 185 (Cont'd)

SM-PT-N075C	MSIV*-Low Condenser Vacuum Trip (PCRVICS)**	Turbine Building
SM-PT-N075D	MSIV*-Low Condenser Vacuum Trip (PCRVICS)**	Turbine Building

* MSIV = Main Steam Isolation Valve

** PCRVICS = Primary Containment and Reactor Vessel Isolation Control System

RESPONSE

The response to Question 421.17 addresses the concerns of RPS sensors located in non-seismic structures (turbine building).

DSER Open Item No. 190 (DSER Section 7.2.2.7)

REGULATORY GUIDE 1.75

We asked the applicant to provide an overview of the plant electrical distribution system with emphasis on the reactor protection system (i.e., reactor trip, engineered safety features actuation and supporting features) instrumentation including the sensors, logic, and actuation relay power supplies and divisional separation as a background for addressing FSAR Chapter 7 concerns.

In a meeting, the applicant provided a response to this question. The staff reviewed this response and found it to be acceptable if reference is made to figures and a table of the FSAR are revised.

We require that the applicant augment this response to make reference to Figures 8.3-8, 8.3-9, 8.3-11, and 8.3-13 and revise Figure 7.2-1 of the FSAR.

RESPONSE

The response to Question 421.9 provides the requested information concerning an overview of the plant electrical distribution system. Figure 7.2-1 was revised in response to Question 421.14.

DSER Open Item No. 192 (DSER Section 7.2.2.9)

REACTOR MODE SWITCH

We require the applicant to augment the response concerning the reactor mode switch to indicate that HCGS has responded to IE Notice 83-42 and that the modified mode switch will be installed prior to fuel load. In addition, we require the applicant to clarify the response regarding the use of operational procedures as the primary method of controlling rod movement during refueling.

RESPONSE

The response to Question 421.26 addresses the concerns about the reactor mode switch installed at HCGS.

DSER Open Item No. 194 (DSER Section 7.3.2.2)

STANDARD REVIEW PLAN DEVIATIONS

The staff has reviewed the applicant's response concerning SRP deviations and has concluded that they are acceptable with the exception of the ESF equipment area cooling system and the SSEAVS. For these systems, the applicant is required to provide additional justification or show system applicability to the Standard Review Plan Table 7.1.

RESPONSE

The response to Question 421.2 provides the justification for any deviations between HCGS control systems design and SRP Table 7-1 requirements.

DSER Open Item No. 197 (DSER Section 7.3.2.5)

MICROPROCESSOR, MULTIPLEXER AND COMPUTER SYSTEMS

We require the applicant to expand the response regarding the Bailey 862 modules and to provide a typical set of drawings and the instruction manuals for the Bailey Model 862.

RESPONSE

The response to Question 421.6 provides the requested information concerning the reliability of the Bailey 862 equipment. Typical drawings and Bailey 862 instruction manuals were provided to the NRC as additional documents during the January 13 ICSB meetings

DSER Open Item No. 200 (DSER Section 7.4.2.2)

REMOTE SHUTDOWN SYSTEM

The applicant is required to confirm that the HCGS design meets the staff's guidance for remote shutdown capability.

RESPONSE

The response to Question 421.38 identifies how the HCGS remote shutdown systems design meets the NRC staff's guidances for remote shutdown capability.

DSER Open Item No. 205 (DSER Section 7.5.2.4)

PLANT PROCESS COMPUTER SYSTEM

Pending final revisions to FSAR Sections 7.5 and 7.7. These revisions should provide clarification of the safety categorization of the information systems addressed in these sections.

RESPONSE

The response to Question 421.55 addresses the concerns regarding the plant process computer system.

DSER Open Item No. 209 (DSER Section 7.7.2.3)

CREDIT FOR NON-SAFETY RELATED SYSTEMS IN CHAPTER 15 OF THE FSAR

The peak vessel pressures resulting from the analyses of the transients without taking credit for nonsafety-related structures, systems, and components are bounded by the peak pressure limit of the overpressure protection system as described in the Hope Creek FSAR.

The staff is reviewing the applicant's response relating to this concern and will report its finding in a future SER.

RESPONSE

The response to Question 421.54 identifies which of the nonsafety-grade systems/components that may be actuated during the course of anticipated operational occurrences (transients) are included in the Technical Specifications.

DSER Open Item No. 210 (DSER Section 7.7.2.4)

TRANSIENT ANALYSIS RECORDING SYSTEM

The applicant is required to document the test result details regarding the GETARS I remote multiplexer unit and its associated electrical isolation. The staff is presently reviewing the information provided by the applicant in response to our request for additional information and will document the results of this review in a future safety evaluation report (SER).

RESPONSE

The response to Question 421.49 fully describes the transient analysis recording system being used at HCGS.

DSER Open Item No. 218 (DSER Section 9.5.1.1)

FIRE HAZARDS ANALYSIS

GDC #3 requires: "Fire fighting systems shall be designed to assure that rupture or inadvertent operation does not significantly impair the safety capability of those structures, systems, and components." To satisfy this requirement, the applicant has designed components required for hot shutdown so that rupture or inadvertent operation of fire suppression systems will not adversely affect the operability of these components. Where necessary, appropriate protection is provided to prevent impingement of water spray on components required for hot shutdown. Redundant trains of components that are susceptible to damage from water spray are physically separated so that manual fire suppression activities will not adversely affect the operability of components not involved in the postulated fire. However, the staff is concerned that the mechanism by which fire and fire fighting systems may cause the simultaneous failure of redundant or diverse trains has not been adequately considered in the design. The staff will require that the applicant identify such mechanisms that were considered in his fire hazards analysis and the measures taken to preclude the fire or fire-suppressant-induced failure of redundant or diverse safety trains.

RESPONSE

In response to Appendix R and IE Notice 83-41, each suppression system covering safety related areas was reviewed for spurious actuation either by seismic induced error or operator error. Those systems with closed heads were found acceptable as is, since a second failure of one or more heads would be required to discharge water. This applied to the cable spread room at elevation 77 ft. and the intake structure service water pump rooms. The automatic CO_2 systems cover the control equipment mezzanine at elevation 117 ft.-6 in., the diesel generator fuel oil tanks and the diesel generators. Spurious actuation of any diesel generator CO_2 system will cause a trip of that DG set but since each DG set is separated, this will not prevent safe shutdown utilizing the remaining diesel generators or offsite power. Spurious actuation of the control equipment mezzanine CO_2 system will not affect the cable therein or safe shutdown from the main control room or the remote shutdwon panel room, since the mezzanine only contains cable.

The open head type systems were changed to manual operation. These are the FRVS charcoal filter system, main control room emergency charcoal filter system, and the diesel fuel oil tank room systems. The deluge valves are manually actuated by pushbutton on the local panel. The OS&Y gate valve is also kept closed to prevent spurious actuation. Please refer to Section 9A.4.1.2.2.

Drainage and flooding caused by automatic or manual fire fighting has been considered and will not prevent safe shutdown. Please refer to Section 9.5.1.1.9 and the reply to Question 640.9.

QUESTION 640.9 (SECTION 14.2.12)

Modify FSAR Subsection 14.2.12.1.29 (KC-Fire Protection - Deluge) to provide assurance that:

1. Upon automatic sprinkler actuation, adequate drainage in the affected spaces is provided to preclude flooding (including expected hand-held hose volume).

2. A walk-down of plant equipment is conducted to identify potential incidences where the actuation of fire suppression systems could cause damage to or inoperability of systems important to safety.

See IE Information Notice 83-41: Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment, June 22, 1983.

RESPONSE

~~The results of our findings on adequacy of the drainage to preclude flooding, upon automatic sprinkler actuation, will be available May 1984~~.

Section 14.2.12.1.29.b has been revised to include a prerequisite walkdown of the fire protection system to identify potential areas where the fire protection system could cause damage.

Section 14.2.12.1.29.b has been revised to address the provision to drain areas where automatic sprinkler actuation might affect safe shutdown equipment.

2. The system responds to simulated fire signals.

3. The refrigeration system operates to maintain pressure and temperature as specified by the manufacturer's technical instruction manual.

14.2.12.1.29 KC-Fire Protection - Deluge

a. Objective

The test objective is to verify the capability of the fire protection system to deliver water to the sprinkler system, pre-action and deluge systems, hose stations, and hydrants at rated pressure and flow.

b. Prerequisites

1. Component tests have been completed and approved.

2. System instrumentation has been calibrated and approved.

3. AC and dc power are available.

4. The diesel fire pump local fuel oil storage tank is in service.

5. Adequate fire protection water supply is available.

6. A walkdown has been performed to identify components or areas that may be susceptible to damage due to actuation of the deluge system.

INSERT A →

c. Test Method

1. All valves, controls, alarms, interlocks, and logic are checked for proper operation.

2. Normal system flow paths are verified.

DSER OPEN ITEM 218

Insert A

7. Floor drains have been provided to remove the expected fire fighting water flow from automatic sprinkler systems, hand hose lines, etc. Temporary build up of water in the affected spaces will not flood safe shutdown equipment.

DSER OPEN ITEM 218

HOPE CREEK
DSER OPEN ITEM RESPONSE

TECHNICAL SPECIFICATION ITEM 4.4.5 (TS-3):

Core Flow Monitoring for Crud Effects

Crud deposition causes gradual flow reductions in some light water reactor cores. However, measurement of core flow by jet pump pressure drop and core plate pressure drop will provide adequate indication of such flow reductions, if they should occur. Technical Specifications will require that the core flow be checked at least once every 24 hours to detect flow reduction.

RESPONSE

Crud deposition is assumed in the General Electric methods used for the design of fuel and for calculating pressure drop. This assumption has no significant impact on the CPR results. Thus, crud deposition is conservatively accounted for when predicting fuel performance. The build up of crud occurs very slowly, especially in the early years of fuel life, therefore, such a short inspection frequency is not justifiable.

HOPE CREEK
DSER OPEN ITEM RESPONSE

LICENSE CONDITION ITEM 4.2 (LC-1): Fuel Rod Internal Pressure Criterion

The applicant must demonstrate that a fuel rod internal pressure criterion which allows the internal fuel rod pressure to exceed system pressure will not (1) lead to fuel system damage during normal operation and AOOs, (2) prevent control rod insertion when required, (3) result in an underestimate of the number of fuel failures in, or radiological consequences of, postulated accidents or (4) lead to loss of coolable geometry.

RESPONSE

General Electric has proposed an alternative internal pressure criterion that would satisfy the SRP requirement and would resolve this issue generically. The NRC staff is presently reviewing General Electric's proposal as part of its review of an amendment to GESTAR II (NEDE-24011). Completion of that review is expected by April 1984, and General Electric will pursue this matter to a complete resolution.