GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

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MFN 048-83
JNF 013-83

March 9, 1983

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Attention: Mr. D.G. Eisenhut, Director
Division of Licensing

Gentlemen:

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR II)
DOCKET NO. STN 50-447

REVISED DRAFT RESPONSES AND MATERIALS UPDATE

Attached please find revised final draft responses to selected questions of the Commission's August 25, 1982 and November 15, 1982 information requests and the Mechanical Engineering Branch draft SER meetings. Only modifications (new or revised) to the responses of the referenced letters are provided. Also attached is a draft of a GESSAR II update relative to the latest revisions to Regulatory Guides 1.31 and 1.44 and NUREG-0313.

Sincerely,

Wells for

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Attachments

cc: F.J. Miraglia (w/o attachments)
D.C. Scaletti
C.O. Thomas (w/o attachments)
L.S. Gifford (w/o attachments)

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LISTING OF ATTACHMENTS PROVIDED

Attachment Number	Subject
1	Draft Responses to Structural and Geotechnical Branch Questions
2	Draft Responses to Power Systems Branch Questions
3	Draft Responses to Meterology and Effluent Treatment Branch Questions
4	Draft Responses to Reactor Branch Questions
5	Draft Responses to Mechanical Engineering Branch Questions
6	Draft Update of GESSAR II Relative to R.G.'s 1.31 and 1.44 and NUREG-0313

Attachment No. 1

Draft Responses to
Structural and Geotechnical
Branch Questions

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

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

220.04 and 220.05 Responses

The structural response to this load is evaluated using equivalent static forces obtained by the procedure in Reference 6 for rigid missiles, and the procedure in Reference 7 for deformable missiles

Ductility in flexure of slabs and shapes is used Ductility ratios do not exceed 10. The above text will be added to Section 3.5.3.2 and will meet the SRP requirements. For Steel, requirements of Appendix A of SRP Section 3.5.3 are met. Ductility concepts were not used in Concrete shear analysis.

Reference 7 (J. D. Riera, On the Stress Analysis of Structures Subjected to Aircraft Impact Forces, Nuclear Engineering and Design, North Holland Publishing Co, Vol. 8. 1968) will be added.

1/24

220.08
(3.7.1)

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

220.08 - Response

[redacted]

The damping factors indicated in Table 3.7-1 were used in the response analysis of various structures and systems and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing and presented in Section 3.10.[redacted] These values are consistent with those given in NRC Regulatory Guide 1.61.

When developing seismic design data for the SSE, the higher damping values of Regulatory Guide 1.61 were not used. The SSE data was obtained by doubling the OBE values which were based on the lower damping values. In the design process, the stress levels have been assessed and found sufficiently high to justify the use of the damping factors in Table 3.7-1.

Section 3.7.1.3 will be revised as indicated above.
Table 3.7-1 will be revised as indicated.

Table 3.7-1

CRITICAL DAMPING RATIOS FOR DIFFERENT MATERIALS

Item	Percent Critical Damping	
	OBE Condition	SSE Condition
Reinforced concrete structures	4.0	7.0
Welded structural assemblies	2.0	4.0
Equipment	2.0	3.0
Bolted or riveted structural assemblies	4.0	7.0
Vital piping systems		
- diameter greater than 12 in.	2.0	3.0
- diameter less than or equal to 12 in.	1.0	2.0
Reactor pressure vessel, support skirt, shroud head, separator	2.0	4.0
Guide tubes and CRD housings	1.0	2.0
Fuel	6.0	6.0
Steel frame structures	2.0	4.0

delete

1/24

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

220.10

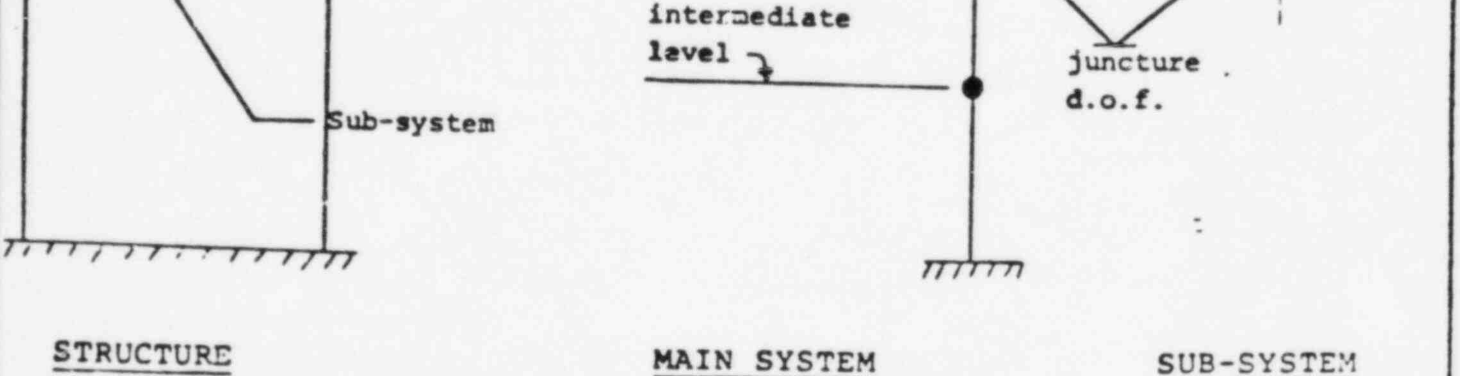
The report on the study of polar crane interaction with the stee containment is [redacted]. provided below.

DYNAMIC INTERACTION BETWEEN CONTAINMENT AND POLAR BRIDGE CRANE GIRDER

CONCEPT

Dynamic interaction between any two structural systems depends on their relative masses and stiffnesses.

The structural system in question, namely, the steel containment with the crane girder was divided into two systems. A main system consisting of the containment alone, and a sub-system consisting of the polar crane and the crane bridge. The main system was idealized as a 3-mass system with masses concentrated at the c.g. of the containment ellipsodial head, crane girder level and an intermediate level. The sub-system was idealized as a 2-mass system with masses concentrated at the center and at an extreme trolley position, the former representing the mass of the crane bridge and the later representing the trolley with L.L. To study dynamic interaction of the two systems in all possible modes of excitation, three different types of excitation were considered. They were vertical excitation, horizontal lateral excitation, and torsional excitation. For each of these excitations, the two systems were reduced to corresponding equivalent single d.o.f. systems by condensing out the non-juncture degrees of freedom. These effective masses and stiffnesses yielded the frequencies for the main system and for the sub-system for each of the three modes of excitation. Using the existing literature and the developed mass and frequency ratios, the percent error involved in decoupling the two systems and modifying the main system with the mass of the sub-system lumped into it was studied.



IDEALIZATION

CRANE GIRDER-CONTAINMENT INTERACTION

TVA STRIDE

November 5, 1974

SUMMARY OF RESULTS

	Main System			Sub-System			Mass Ratio	Frequency Ratio	% error in the eigenvalue for the modified main system compared with that of the complete system	Approximation Acceptable or not
	Effective Mass, M_M^*	Effective Stiffness, K_M^*	Frequency f_M	Effective Mass, M_S^*	Effective Stiffness, K_S^*	Frequency f_S	M_S^*/M_M^*	$\frac{f_S}{f_M}$		
Vertical Excitation	151.8 $\frac{K\text{-}SEC^2}{FT}$	1.20×10^6 K/FT	14.15 CPS	11.06 $\frac{K\text{-}SEC^2}{FT}$	2374 K/FT	2.33 CPS	0.07	0.16	< 10	Acceptable
Horizontal Lateral Excitation	151.8 $\frac{K\text{-}SEC^2}{FT}$	0.21×10^6 K/FT	5.88 CPS	11.06 $\frac{K\text{-}SEC^2}{FT}$	139 K/FT	0.56 CPS	0.07	0.10	< 10	Acceptable
Torsional Excitation	551,819 $\frac{K\text{-}SEC^2\text{-}FT}{RAD}$	15.84×10^8 K-FT/RAD	8.53 CPS	29,750 $\frac{K\text{-}SEC^2\text{-}FT}{RAD}$	0.5×10^6 $\frac{K\text{-}FT}{RAD}$	0.65 CPS	0.05	0.08	< 10	Acceptable

In conclusion, interaction between the steel containment and the crane can be ignored and the mass of the crane etc can be lumped into the containment model at that level for all types of excitation.

OVALING MODES OF CRANE GIRDER

Due to the non-axisymmetric point loads resulting from the polar bridge crane, the crane ring-girder and the steel containment shell can exhibit ovaling modes of vibration.

The frequencies of these modes have been computed using standard formulae. The exact shape of a given ovaling mode of vibration consists of a curve which is a sinusoid on the developed circumference of the ring. For these computations the ring-girder is assumed to act as a structural composite with a tributary shell section. The results are summarized below.

OVALING MODES

MODE OF VIBRATION	WITHOUT CRANE		WITH CRANE	
	RAD/SEC	CPS	RAD/SEC	CPS
n* = 2	25.4	4.04	18.8	3.00
n = 3	71.9	11.44	53.3	8.48
n = 4	138.0	21.96	102.0	16.23

*n = number of full sine waves along the circumference.

In conclusion, judging from the high frequencies and nature of the respective mode shapes, the ovaling modes have very little modal responses as well as very small participation factors and hence are not significant. In addition, the ovaling modes have been found to have hardly any coupling with the beam modes of vibration.

REFERENCES:

1 Pickel, T W, Jr, "Evaluation of Nuclear System Requirements for Accommodating Seismic Effects", Nuclear Engineering and Design, Vol 20, 1972.

2 Den Hartog, J P, "Mechanical Vibrations", 4th edition, McGraw-Hill, 1956.

3 Bechtel Power Corporation, "Topical Report - Seismic Analyses of Structures and Equipment for Nuclear Power Plants", BC-TOP-4, Rev 2, June 1974.

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

RESPONSE

The response to this question is being addressed in conjunction with question 220.44. An additional eight cases of SSI analyses are being performed using the elastic half-space method with R.G.1.60 motion applied at the foundation level to demonstrate that the current GESSAR II design envelopes and design parameters bound those produced by these new analyses. The fundamental frequencies of structures, equipment and components will be limited above the low frequency range (≤4 Hz) within which the response spectra obtained from the elastic half-space method exceed the response spectra obtained from the GESSAR II finite-element method. The results of these new analyses will be provided to the staff in April 1983 to demonstrate that the GESSAR II finite element method analyses satisfies the intent of the staff's acceptance criteria regarding the input level of the motion.

220.20
(3.8.2)
(3.8.3)
(3BA8.4)

Provide the following information applicable to pool dynamic loads, their load combinations and the analysis of these loads:

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

Response a

A finite element model ~~has been~~ is used to represent the Reactor Building. Time history analyses ~~have been~~ are preformed for pool dynamic loads. Response acceleration time histories ~~were~~ are obtained at selected node points including the reactor pressure vessel supports. Soil elements are used in the math model to account for the soil structure interaction.

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

Response b

The structures adjacent to the Reactor Building are expected to experience insignificant amount of impact from the pool dynamic loads. This is to be confirmed by the Applicant.

c. Your procedures for combining static and alternating dynamic loads (Section 3BA.8.4) do not agree with our positions on this matter. (Refer to Sections 3.8.2 and 3.8.3 of the SRP.) Discuss the effect of this deviation. In addition, indicate whether your method of analysis includes the effects of fluid-structure interaction in the manner specified in the last paragraph of Item II.3.a of Section 3.8.3 of the SRP; i.e., whether you comply with the Appendix to Section 3.8.1 of the SRP. (Refer to Question 220.25)

Response c

Subsection 3BA.8.4 will be deleted from Appendix 3B. Section 3.8 provides the loads and load combinations for the structures which are in compliance with Sections 3.8.2 and 3.8.3 of the SRP.

In the pool dynamic load analysis, hydrodynamic mass has been used to cover the structure-fluid interaction. GESSAR II will be revised (as attached) to comply with the appendix to Section 3.8.1 of the SRP.

TEXT CHANGE FOR 220.20(c)

3.8.3.3.1.1 Loads (Continued)

Floor at El 11 ft., 0 in.	350 psf
Floor at El (-) 5 ft., 3 in.	250 psf
Stairs and platforms	100 psf
Floor at RPV head laydown area	1000 psf
Floor at equipment hatches	1000 psf
Floor in upper pool (in addition to water weight)	1000 psf except for 1500 psf in fuel storage area

P_o = normal operating or shutdown differential pressures between the inside and outside of the drywell (negligible)

add → SRV discharge,

T_o = thermal effects during normal operating conditions including plate or liner expansion, equipment and pipe reactions, and temperature gradients using the combination of internal and external temperatures which would produce the most critical transient or steady-state thermal gradient. The maximum normal operating drywell temperature is 150°F.

R_o = pipe reactions during normal operation or shutdown conditions based on the most critical transient or steady state condition.

P_v = pressure difference between the drywell interior and exterior of the structure (i.e., containment) considering both interior pressure changes due to heating or cooling and containment atmospheric pressure variations (0.8 psig) on positive exterior pressure.

Construction loads = loads applied to the structure from start to completion of construction for construction load combinations (the definitions of D, L, and T_o are applicable but the construction value is used).

3.8.3.3.1.1 Loads (Continued)

drywell walls and slabs are calculated on the basis of the transient bulk gas and liquid temperatures on each side. (See Chapter 6 for drywell temperature diagrams.) Add → Includes SRV discharge thermal effects.

R_a = pipe reaction from thermal conditions generated by the postulated pipe rupture, including R_o

Y_r = reactions from high-energy pipe break. None are on the drywell as these pipes are not anchored at the drywell.

Y_j = load on the structure due to jet impingement from a ruptured high-energy pipe

Y_m = the energy resulting from the impact of a ruptured high-energy pipe on the structure

R_v = SRV discharge loads, vent clearing, chugging, pool stratification, pool swell loads, etc. (For details see Appendix 3B.)

H_a = water pressure resulting from internal containment flooding

W = loads generated by the design wind

3.8.3.3.1.2 Shrinkage and Cyclic Effects

The drywell wall is designed for all loads listed above, the strains induced by concrete shrinkage and for 500 cycles of temperature variation from 60° to 150°F due to startup and shutdown during the 40-year life of the plant. Creep effects are addressed in Subsection 3.8.3.4.1.1.

3.8.3.3.1.3 Load Combinations

Structural concrete design load combinations are defined in this subsection. Nonstructural concrete and pool liner design load combinations are the same but a load factor of 1.0 is used throughout for the applicable load cases and normal operating conditions including hot and cold shutdowns.

3.8.3.3.1.3.1 Load Combinations for Service Load Conditions

The following load combinations are based on the working stress design method:

(1) test $1.0D + 1.0L + 1.0P_t + 1.0T_t$;

(2) construction $1.0D + 1.0L + 1.0T_o + W$;

(3) normal $1.0D + 1.0L + 1.0T_o + 1.0R_o + 1.0P_v$; and

(4) severe environmental $1.0D + 1.0L + 1.0T_o + 1.0F_{eqo} + 1.0R_o + 1.0P_v + R_v$.

3.8.3.3.1.3.2 Load Combinations for Factored Load Conditions

(1) Extreme environmental $1.0D + 1.0L + 1.0T_o + 1.0F_{eqs} + 1.0P_v + R_o$;

1.25 change!

(2) abnormal $1.0D + 1.0L + 1.5P_a + 1.0T_a + 1.0R_a + 1.0R_v$;

(3) abnormal $1.0D + 1.0L + 1.0P_a + 1.0T_a + 1.25R_a + 1.0R_v$;

(4) abnormal/severe environmental $1.0D + 1.0L + 1.25P_a + 1.0T_a + 1.25\ F_{eqo} + 1.0R_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ R_v$;

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

Response d

Finite element non-linear dynamic analysis (ANSYS) was made to determine the effects of negative pressures on the bottom liner plate in the suppression pool. The effects were combined with the effects of high temperature and seismic loads. [redacted]

The 16 ft deep foundation mat is heavily reinforced and is very rigid compared to the liner span between anchorage embedments (range from 27 to 36 inches)

The deformation of the foundation mat in the 27 to 36 inch liner span is negligible compared to the strains of the liner undergoing non-linear deformation due to negative LOCA pressure. Therefore the cracking of the concrete in the foundation mat is negligible and is not considered in the analysis.

220.25
(3.8.2)

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

RESPONSE

This response will be provided in June 1983 as part of the severe accident design review schedule.

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

Response

The GESSAR II design requirements for all of the piping in the annular region between the shield building and the containment vessel (the shield annulus) meet or exceed the requirements of BTP MEB 3-1 for fluid system piping in containment penetration areas for which breaks or cracks need not be postulated. The comparative requirements are:

1. For high-energy ASME Code Section III Class 1 piping, the GESSAR II requirements are the same as BTP MEB 3-1.

2. For high-energy ASME Code Section III Class 2 piping, GESSAR II design requirements are more conservative than BTP MEB 3-1 in that GESSAR II adds the additional requirement that the piping run is straight.

3. For moderate-energy ASME Code Section III Class 2 piping, GESSAR II conservatively applies the same requirements as for ASME Code Section III Class 2 piping.

In addition to meeting these stringent requirements, guard pipes are provided for high-energy piping in the shield annulus. Thus, even in the unlikely event that one of these pipes did fail, its guard pipe would direct the fluid into the drywell and there would be no flooding in the shield annulus.

All of the piping runs in the shield annulus are straight except for one 2-inch moderate energy line. Although this line meets the "no crack" requirements of BTP MEB 3-1, applying the more conservative GESSAR II design requirements, it would be necessary to postulate a crack. However, a crack in this line would be accommodated by the shield annulus drain. Hence, the flooding elevation would be negligible and there would be no threat of containment buckling.

220.43
(3.8.5)

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

220.43 Response

GESSAR will be revised as indicated below for the factors of safety against sliding for the reactor, the auxiliary and the control buildings.

See attached revised first paragraph of 3.8.5.4.1 and revised Figure 3.8-75. The design meets the requirements of SRP 3.8.5 Item II.5.

For the reactor building, ~~we have additionally used~~ the passive pressure resistance acting on that part of the foundation mat which is deeper than the adjacent auxiliary and fuel building foundations was additionally used. This item ~~raised~~ resulted in the safety factor ~~to~~ of 1.19 in the pseudo-static calculation.

For the auxiliary and control buildings the minimum safety factor of 1.1 is obtained using time-history approaches. The method is outlined in San Onofre 2 and 3, NRC question 131.20, and Appendix 3.7C-G of the FSAR.

3.8.5.3.2 Auxiliary, Fuel, Control, Radwaste, and Diesel Generator Buildings Foundations

The foundation loads and load combinations for these structures are discussed in Subsection 3.8.4.3.

3.8.5.4 Design and Analysis Procedures

The soil and structural settlements have been calculated at various points of each building. These values are included in the piping design specification for each individual piping system. In each specification, there is a table listing load combination requirements and stress limits. The differential settlement has been listed as a loading condition. The piping systems are designed for this settlement. The acceptance criteria are specified in the form of stress limits. They are in accordance with the appropriate sections of the ASME Code which is identified in the piping specifications.

3.17

3.8.5.4.1 Reactor Building Foundation

The design of the Reactor Building foundation is concerned primarily with determining shear and moments in the reinforced concrete and determining the interaction of the substructure with the underlying foundation. For a reactor building foundation supported on soil or rock, the pertinent aspects in the design are to maintain the bearing pressures within allowable limits, particularly due to overturning forces, and to ensure that there is adequate frictional force to prevent sliding of the structure when subjected to lateral loads.

and passive resistance

The design loads considered for analysis of the base slab foundation are the worst resulting forces from superstructures due to static and dynamic load combinations and such loads directly applied on the base slab as dead, live, seismic, hydrostatic, internal pressure, and temperature loads. The post-LOCA flooding condition has

BUILDING	OVERTURNING S. F. (1)	SLIDING S. F. (2)	BUOYANCY S. F. (3)	REMARKS
REACTOR BUILDING	10.0	1.19	4.6	
AUXILIARY BUILDING	5.9	1.10 (4)	1.9	
FUEL BUILDING	4.4	1.13	2.7	
CONTROL BUILDING	10.8	1.10 (4)	8.2	SHALLOW FOUNDATION
RADWASTE BUILDING	6.7	1.74	1.5	
DIESEL GENERATOR BLDG DIV I	22.9	1.40	6.0	SHALLOW FOUNDATION
DIESEL GENERATOR BLDG DIV II & III	10.3	1.30	6.6	SHALLOW FOUNDATION

(1) OVERTURNING S. F. $= \dfrac{\text{RESISTING POTENTIAL ENERGY}}{\text{OVERTURNING KINETIC ENERGY DUE TO SSE}}$

(2) SLIDING S. F. $= \dfrac{\text{RESISTING FORCE}}{\text{BUILDING SEISMIC SHEAR FORCE (SSE)}}$

(3) BUOYANCY S. F. $= \dfrac{\text{BUILDING WEIGHT}}{\text{BUOYANCY FORCE}}$

(4) The minimum S.F against sliding is obtained by the use of time-history approaches.

FACTORS OF SAFETY FOR OVERTURNING, SLIDING, AND BUOYANCY

Figure 3.8-75. Factors of Safety - Overturning, Sliding ~~Floating~~ and Buoyancy

220.44 (3A.5.2) (Fig.3A-18)

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

RESPONSE

See response to question 220.09.

Audit
Action Item 4

With respect to the drywell design, list those provisions of CC-3000, Section III, Div. 2 Code which are not applicable and the technical basis for the non-applicability. (Ref: Section 3 B.3.2.1.(1) of GESSAR II)

Response

Provisions of CC-3000 (ACI359, ASME Sec III Div 2)

Not applicable to Drywell Design (Ref - 3.8.3.2.1)

a) METALLIC LINER, CC-3121 GENERAL This paragraph is not applicable because the drywell face shells are not the nominal leakage-prevention liners that are addressed by the code. The shell plates are intended to carry design loads and are designed to act like steel reinforcement.

b) FOUNDATION REQUIREMENTS, CC-3561 (a) GENERAL This is a standard design covering a range of postulated sites. Therefore the design need not be based on information provided by laboratory or field tests of some specific subsurface strata. The Applicant will show that his site falls within the postulated range of sites.

c) CC-3720 LINER This paragraph does not apply because, as noted for CC-3121 above, the shell plates are meant to act as reinforcement. Thus, shell plate design is controlled by allowable stresses (CC-3400) and load factors (Table CC-3230-1)

Audit
Action Item 5

With respect to the analysis of the equipment hatch opening of the Shield Building investigate the effect of not considering the curvature on the adequacy of design. Also discuss how the effects of the three components of earthquake motion are accounted for in the hatch opening analysis and the basis thereof.

(Ref: Section 3.8.3 Concrete and Structural Steel Internal Structures)

Response.

The analysis of the equipment hatch opening of the Shield Building was originally performed considering the curvature of the Shield Building (During the SEB Audit it was established curvature was not considered; this was however in error.)

A description of the mathematical model and the method of accounting for the three components of earthquake motion follows:

The finite element method is used to evaluate forces and moments for design of the equipment hatch opening.

A portion of shield building cylindrical shell was analyzed using a curved math model. and allowing for effects of

the hatch opening in the properties of the shell

← Displacements and rotations from shield building model output for various loads and loading combinations have been imposed on boundary nodes for the model of the hatch opening itself. These displacements and rotations provide continuity with the remaining shield building.

Three independent X, Y and Z components of seismic force were incorporated in the computer input, and combined in the computer analysis with other loads for desired load combinations. Either vertical upward or downward seismic load was combined with other loads so that the maximum design force ~~is~~ was calculated. Horizontal seismic loads were also added with the above loads utilizing the absolute sum method.

Audit
Information Request 1

Provide information about the impact analysis for automobile impact on structures (Ref. Section 3.5.3 Barrier Design Procedures)

Response

The impact analysis for automobile impact on structures is attached. All structures subject to this impact load were analyzed as described; specific calculations for the Auxiliary Building walls are attached.

The method utilizes the energy balance technique which involves using the strain energy of the target at maximum structural response to balance the residual kinetic energy of the target resulting from missile impact. A plastic collison is considered, meaning the missile remains in contact with the target. In the case of automobile impact experimental data are available to enable definition of a force-time function. The equation of motion is then solved for the maximum displacement The Bechtel topical report, BC-TOP-9A, on design of structures for missile impact was extensively used in this work.

It is concluded that the exterior building walls sustain the crash of automobile missile during a tornado.

AUTOMOBILE MISSILE IMPACT ON AUXILIARY BUILDING WALLS

1. CRITERIA - (FROM GESSAR II)

STRIKING VELOCITY - V_s = 100 ft/sec
WEIGHT - W_m = 4000 #
FRONTAL AREA - A = 20 ft^2

AUTOMOBILE MISSILES ARE TO BE CONSIDERED AT ALTITUDES LESS THAN 30' ABOVE GRADE LEVEL.

2. ASSUMPTIONS -

A PLASTIC COLLISION IS ASSUMED, I.E., THE AUTOMOBILE MISSILE AND THE WALL ACQUIRE SAME VELOCITY AFTER IMPACT.

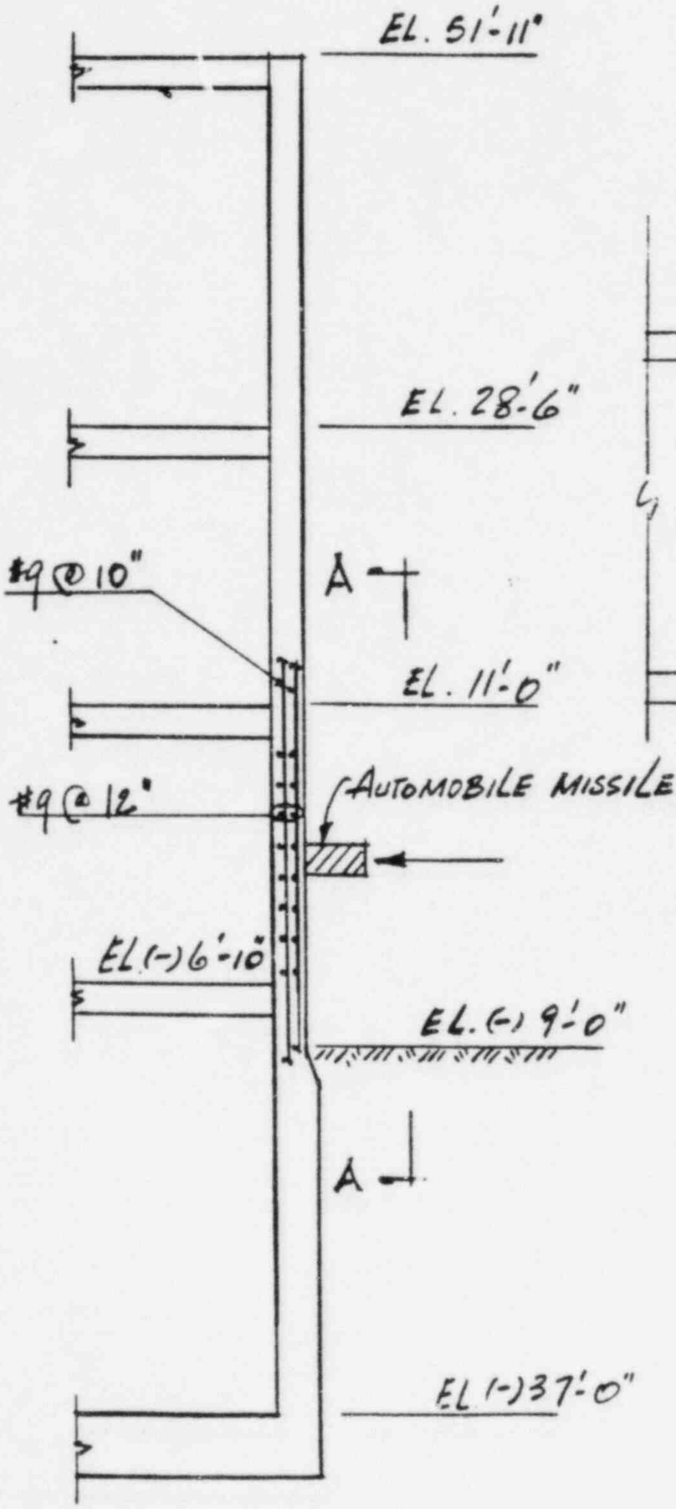
3 FORCE - TIME HISTORY FOR AUTOMOBILE CRASH

ACCORDING TO APPENDIX D OF BC-TOP-9A REV. 2, SEPT., 1974, THE FORCE-TIME HISTORY FOR AN AUTOMOBILE FRONTAL CRASH IS APPROXIMATELY AS FOLLOWS

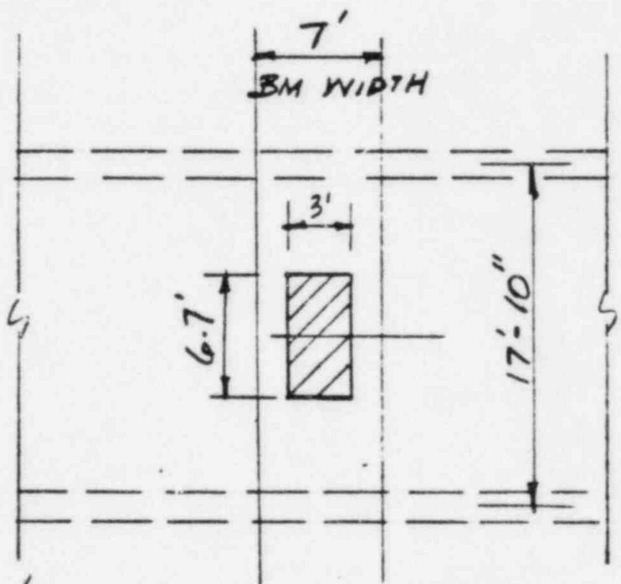
$$F(t) = 0.625\, V_s W_m \sin 20t \quad (0 \le t \le 0.0785 \text{ sec})$$

$$F(t) = 0 \quad (t > 0.0785 \text{ sec})$$

4. STRENGTH OF AUXILIARY BUILDING EXTERIOR WALLS

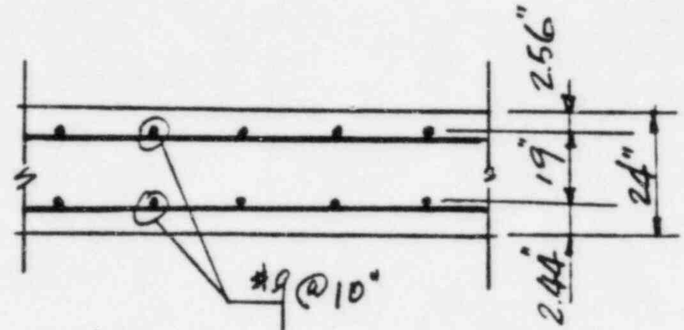


EXTERIOR WALL ELEVATION



SECT. A-A

CONSIDER THE WALL AS A ONEWAY SLAB SPANNED BETWEEN THE FLOOR SLABS



AUTOMOBILE FRONTAL AREA = $20^{\#}$

ASSUME $3' \times 6.7' = 20.1^{\#} \doteq 20^{\#}$

BEAM WIDTH $= 3' + 2 \times 2' = 7'$

$d = 24" - 2.44" = 21.56"$

$d' = 2.56"$

$$p = \frac{A_s}{bd} = \frac{8.4}{84 \times 21.56} = 0.00464$$

$$\frac{1.4\sqrt{f_c'}}{f_y}\left(\frac{t}{d}\right)^2 = \frac{1.4 \times (3000)^{\frac{1}{2}}}{60000} \times \left(\frac{24}{21.56}\right)^2 = 0.00158$$

$$\frac{0.25\, f_c'}{f_y} = \frac{0.25 \times 3000}{60000} = 0.0125$$

ACCORDING TO P4-2, BC-TOP-9A, REV 2. FOR MEMBERS WITH TENSION AND COMPRESSION STEEL, REINFORCING STEEL MUST SATISFY THE FOLLOWING CRITERIA

$$\frac{1.4\sqrt{f_c'}}{f_y}\left(\frac{t}{d}\right)^2 \leq \frac{A_s}{bd}$$

$$\frac{A_s - A_s'}{bd}\left(\frac{t}{d}\right)^2 \leq \frac{0.25 f_c'}{f_y}$$

FROM THE ABOVE CALCULATIONS, $\frac{A_s}{bd} = 0.00464 > 0.00158$

AND $A_s = A_s'$. THEREFORE. THE WALL REINFORCEMENTS MEET THE REQUIRED CRITERIA.

$$pn = .00464 \times 9.06 = .042$$

$$\frac{p'}{p} \doteq 1$$

$F = 0.03$ (FROM CHART IN P4-8 BC-TOP-9A)

AVERAGE MOMENT INERTIA

$$I_A = \frac{1}{2}\left(\frac{bt^3}{12} + Fbd^3\right)$$

$$= \frac{1}{2}\left(\frac{1}{12}\times 84 \times 24^3 + 0.03 \times 84 \times 21.56^3\right)$$

$$= 61012 \text{ in}^4$$

FOR $f_c' = 3000$ psi, $f_y = 60000$ psi, $p = 0.00464$, DEPTH OF EQUIVALENT RECTANGULAR STRESS BLOCK WITHOUT COMPRESSION REINFORCEMENTS CAN BE CALCULATED AS FOLLOWS

$a/d = 0.106 + \frac{14}{50} \times 0.012 = 0.109$ (REF. TO FLEXURE 1.1 ACI DESIGN HANDBOOK)

$a = 0.109 \times 21.56 = 2.36''$

a $(0.236'')$ is LESS THAN d' $(2.56'')$

SO. THE ULTIMATE MOMENT CAPACITY OF THE WALL CAN BE CONSERVATIVELY EVALUATED AS FOLLOWS

$$M_u = 0.9 A_s f_{dy} (d - d')$$

WHERE f_{dy} = ALLOWABLE DYNAMIC YIELD STRESS FOR REINF. STEEL.
$= 60{,}000$ psi

$$M_u = 0.9 \times 8.4 \times 60 \times (21.56 - 2.56)/12$$
$$= 718.2 \ ^{1K}$$

5. REQUIRED STRAIN ENERGY OF WALLS TO STOP THE AUTOMOBILE MISSILES

THE REQUIRED STRAIN ENERGY TO STOP THE WALL-MISSILE COMBINATION IS THE SUM OF THE KINETIC ENERGY OF THE MISSILE AND THE WALL MASSES AT THE END OF THE DURATION OF IMPACT.

FOR FRONTAL IMPACT OF AN AUTOMOBILE, THE FORCE-TIME HISTORY IS DEFINED (SEE SECT. 3). SO WE CAN USE THE FOLLOWING FORMULA TO EVALUATE THE REQUIRED STRAIN ENERGY, E_s

$$E_s = \frac{(M_m + M_e)}{2 M_e^2} I^2$$ — REF. TO P. 3-5 BC-TOP-9A, REV. 2

WHERE

M_m = MASS OF MISSILE $4000/g$

M_e = EFFECTIVE MASS OF TARGET DURING IMPACT

$$= (D_x + 2T)(D_y + 2T)\frac{\gamma_c T}{g}$$

$$= (6.7 + 2\times2)(3 + 2\times2)\frac{145\times2}{g}$$

$$= 21721/g$$

I = APPLIED IMPULSE

$$= \int_0^{t_i} F(t)\,dt$$

$$= \int_0^{0.0785} (0.625\, V_s W_m \sin 20t)\,dt$$

$$= -\frac{1}{20}\times 0.625\, V_s W_m \left[\cos 20t\right]_0^{0.0785}$$

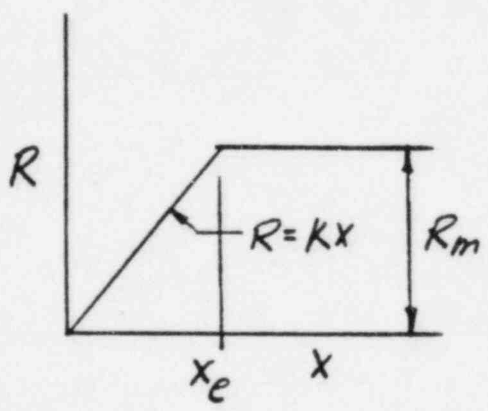
$$= -\frac{1}{20}\times 0.625\times 100\times 4000\,[0-1]$$

$$= 12500 \text{ \#-SEC}$$

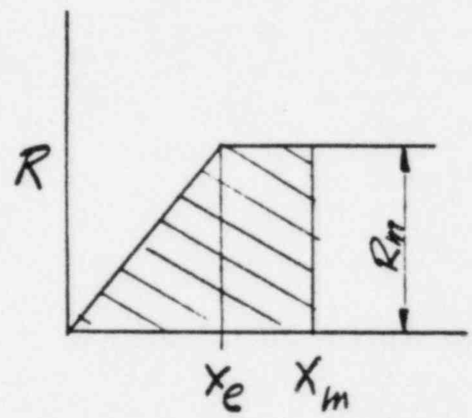
$$E_s = \frac{(4000 + 21721)}{2\times(21721)^2}\times 12500^2 = 4259 \text{ \#-ft}$$

6. MAXIMUM WALL DISPLACEMENTS

THE RESISTANCE - DISPLACEMENT FUNCTION, $R(x)$, FOR A CONCENTRATED LOAD IS CONSIDERED TO BE ELASTO - PLASTIC AS FOLLOWS

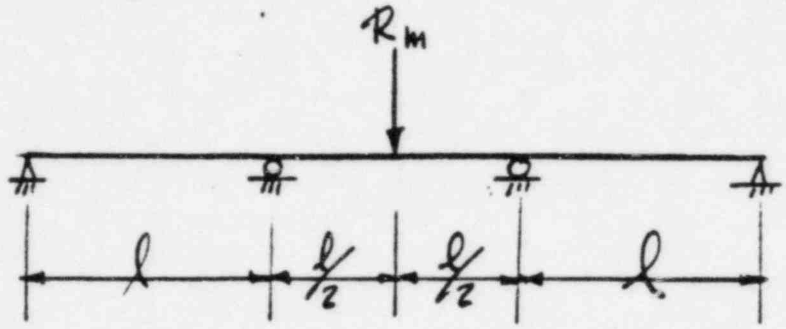


RESISTANCE - DISPLACEMENT FUNCTION



AVAILABLE STRAIN ENERGY

THE ESTIMATED MAXIMUM WALL RESPONSE IS DETERMINED BY EQUATING THE AVAILABLE WALL STRAIN ENERGY TO THE REQUIRED STRAIN ENERGY AND SOLVING FOR THE MAXIMUM DISPLACEMENT X_m.



THE ABOVE MULTI-SPAN BEAM IS USED TO ANALYZE THE PLASTIC RESISTING FORCE (R_m) AND YIELD DISPLACEMENT OF THE WALL.

$$R_m = \frac{4(M_u^+ + M_u^-)}{L}$$

$$X_e = \frac{0.011\, R_m L^3}{EI}$$

REFER TO P4-5, BC-TOP-9A REV. 2

$$R_m = \frac{4 \times (718.2 + 718.2)}{17.83} \doteq 322^{\text{KIPS}}$$

$$X_e = \frac{0.011 \times 322 \times 10^3 \times (17.83 \times 12)^3}{3.2 \times 10^6 \times 61012}$$

$$= 0.178''$$

FROM THE CURVE SHOWN IN PAGE 7, THE AVAILABLE WALL STRAIN ENERGY

$$E_A = R_m \left(X_m - \frac{X_e}{2}\right)$$

BY EQUATING THE AVAILABLE WALL STRAIN ENERGY TO THE REQUIRED STRAIN ENERGY. WE OBTAIN THE WALL MAXIMUM

DISPLACEMENT AS FOLLOWS.

$$X_m = \frac{E_3}{R_m} + \frac{X_e}{2}$$

$$= \frac{4259 \times 12}{322000} + \frac{0.178}{2}$$

$$= 0.248''$$

6. THE REQUIRED DUCTILITY RATIO

$$\mu_r = \frac{X_m}{X_e}$$

$$= 0.248/0.178 = 1.393$$

ACCORDING TO P.4-7, BC-TOP-9A, REV. 2, THE MAXIMUM ALLOWABLE DUCTILITY RATIO FOR CONCRETE BEAM IS 10. FROM THE ABOVE CALCULATIONS, THE REQUIRED DUCTILITY RATIO IS 1.393, FAR LESS THAN 10. THEREFORE, THE WALL CAPACITY IS ADEQUATE TO SUSTAIN THE CRASH OF AUTOMOBILE MISSILE.

7. SHEAR STRESSES

ACCORDING TO SECTION 3. FORCE-TIME HISTORY, THE MAXIMUM FORCE $F_{max} = 0.625\, V_s W_m$

$$= 0.625 \times 100 \times 4$$

kips

PUNCHING SHEAR FORCE = 250 kips

$$v_u = \frac{V_u}{\phi\, b_o\, d}$$

$$= \frac{250000}{0.85 \times 2 \times [(36+21.56)+(80.4+21.56)]\,21.5}$$

$= 42.8$ psi < 220 psi OK

8. CONCLUSIONS - THE AUXILIARY BUILDING EXTERIOR WALLS ARE ABLE TO SUSTAIN [illegible] CRASH OF AUTOMOBILE MISSILE GENERATED [illegible]RNADO WITHOUT EXCESSIVE DISPLACEMENT.

9. REFERENCES

Bechtel TOPICAL REPORT - DESIGN OF STRUCTURES FOR MISSILE IMPACT (BC-TOP-9A, REVISION 2)

Attachment No. 2

Draft Responses to
Power Systems
Branch Questions

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

Response

The [redacted] GESSAR II design is based on a stable grid voltage conditions with maximum fluctuation of ±5% at the interface points (Refer to Note 4 of fig. 8.3-2 "6900 V S/L Buses E & F".

The Applicant is requested to perform a voltage analysis of his system and guarantee that he can meet the above requirement.

With reference to the Branch Technical Position, PSB-1, [redacted] the following comments apply:

Paragraph B.1

The [redacted] GESSAR II design provides Bus Undervoltage relaying set at 70% for divisions 1 & 2, and 73% for division 3 (HPCS), to detect a loss of the offsite power at the class IE buses and isolate these buses from the BOP. The feeder undervoltage relaying provide a second level of undervoltage protection which satisfy the following requirements as stated in PSB-1

B.1.a) The selection of undervoltage and time delay set points will be determined from an analysis of the voltage requirements of the class IE loads at all onsite system distribution levels;

B.1.b.1) The [redacted] GESSAR II design provides a time delay setting for the feeder undervoltage relaying to insure the existance of a sustained degraded voltage condition. Following this delay, if the degraded voltage condition still [redacted] persists, loss of offsite power operational sequence is [redacted]. started. This sequence includes automatic tripping of the [redacted], offsite power feeder breaker, starting of the HPCS D/G [redacted] and transfer of the div.3 bus from offsite to the [redacted] D/G. Offsite breaker open indication, D/G running [redacted] indication and the D/G breaker closed indications [redacted] Provide the control room operator the necessary information

B.1.b.2) The [redacted] GESSAR II design does not provide a second time delay as suggested in the PSB-1. [redacted] HPCS (div.3) bus is automatically separated [redacted] from the offsite power system upon sustained [redacted]. degraded voltage condition as described in B.1.b.1 above.

GESSAR II complies.

B.1.C.1&2) [redacted]. Refer to Fig. 8.3-2 "6900 Volt Single Line Buses E & F", and Fig. 8.3-14a "HPCS Power System – Simplified-One Line Diag."

B.1.C.3) HPCS is an independent bus and has no [redacted]. ~~Refer to GESSAR II section 8.3.1.1.7(2)~~ common trippings with Div. 1 or 2. No load shedding ~~"Load Shedding and Sequencing on Class 1E Buses"~~. is required for the HPCS (div.3) bus loads.

B.1.C.4) The voltage sensors (feeder undervoltage relaying) [redacted] automatically initiate the disconnection of the offsite power sources whenever the voltage set point and time delay limits have been exceeded. [redacted]

B.1.C.5) The [redacted] GESSAR II design provides means for device testing and calibrating during plant operation.

B.1.C.6) GESSAR II complies.

B.1.d) The Technical Specifications will include limiting conditions for operations, surveillance requirements and ~~[illegible]~~ will be provided by the Applicant.

The trip set points and the allowable values for the second-level voltage protection sensors and associated time delay devices are plant unique items which ~~[illegible]~~ will be provided by the Applicant.

Paragraph B.2

~~[illegible] provide automatic load shedding in case of sustained degraded voltage about [illegible] rated.~~ HPCS (div. 3) bus loads require no load shedding.

Paragraph B.3

~~[illegible]~~ A voltage drop calculation was performed which shows that all safety related buses down to the 480 volt level, provide full load operation voltages for safe continuous operation of all energized loads, based on a BOP supply voltage of 6.9 KV ± 5%.

Paragraph B.4

This recommended testing procedure will be addressed in the pre-operational test procedures.

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

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

Response

During periodic testing of the HPCS (Div. 3) diesel generator, the D/G is paralleled with the offsite power system. During such a test should a loss of offsite power occur, a LOOP signal would probably not be generated because the D/G would attempt to provide power to the offsite system through the closed offsite power feed breaker. In this case the offsite feed breaker will trip either by offsite feeder overcurrent or by generator overcurrent with voltage restraint (following a time delay). This will automatically separate the HPCS bus from the offsite system. The D/G will continue operating with the governor control changed from droop to isochronous mode and the voltage regulator changed to automatic mode. With these actions complete the D/G will be ready to accept its required load for LOOP conditon.

QUESTION 430.12
(8.3.1)

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

Explain the use of the D1 through D4 inputs shown in Figure 8.3-25 of your FSAR, coming via isolators into the load drivers of the "B" scram solenoid circuits.

Response

GESSAR II design for the RPS scram and MSIV solenoid circuits is same as one used for Clinton Power Station. The specific parameters, such as wire size, circuit protective devices, conduit grounding and resistivity for GESSAR II design will be the same as ones used in the Clinton analysis. Furthermore the GESSAR II design provides a redundant circuit protection for each scram and MSIV solenoid power circuit.

The scram solenoid and MSIV circuits are run in separate conduits. Cables from other circuits are not run in these conduits. Within the PGCC the conduit is flexible and since the circuits are non-divisional the flexible conduit is routed within non-divisional PGCC ducts. There is no mixing of divisions.

Optical isolaters have been provided for electrical isolation within the panel between 1E and non-1E interfaces of the logic circuit. The power supply feeding the "B" solenoids is of the same type as the one feeding the "A" solenoids. Solenoid "A" is fed from non-1E bus A via an inverter and an EPA. Power is maintained within 1E parameters and the equipment is used for the power supply system is of high quality, ± 1/2% voltage regulation.

Solenoid "B" is fed from non-1E bus "B" power supply similar to bus "A". It is acceptable to run "A" and "B" solenoid power circuits together since the isolation is provided in the logic cabinets. Separate ~~non-class~~ non-1E power supplies are provided to enhance plant availability.

Figure 8.3-25 will be corrected on the basis of the above discussion. See attached marked copy.

For more discussion of the NSPS Power Distribution refer to Question 430.14 response.

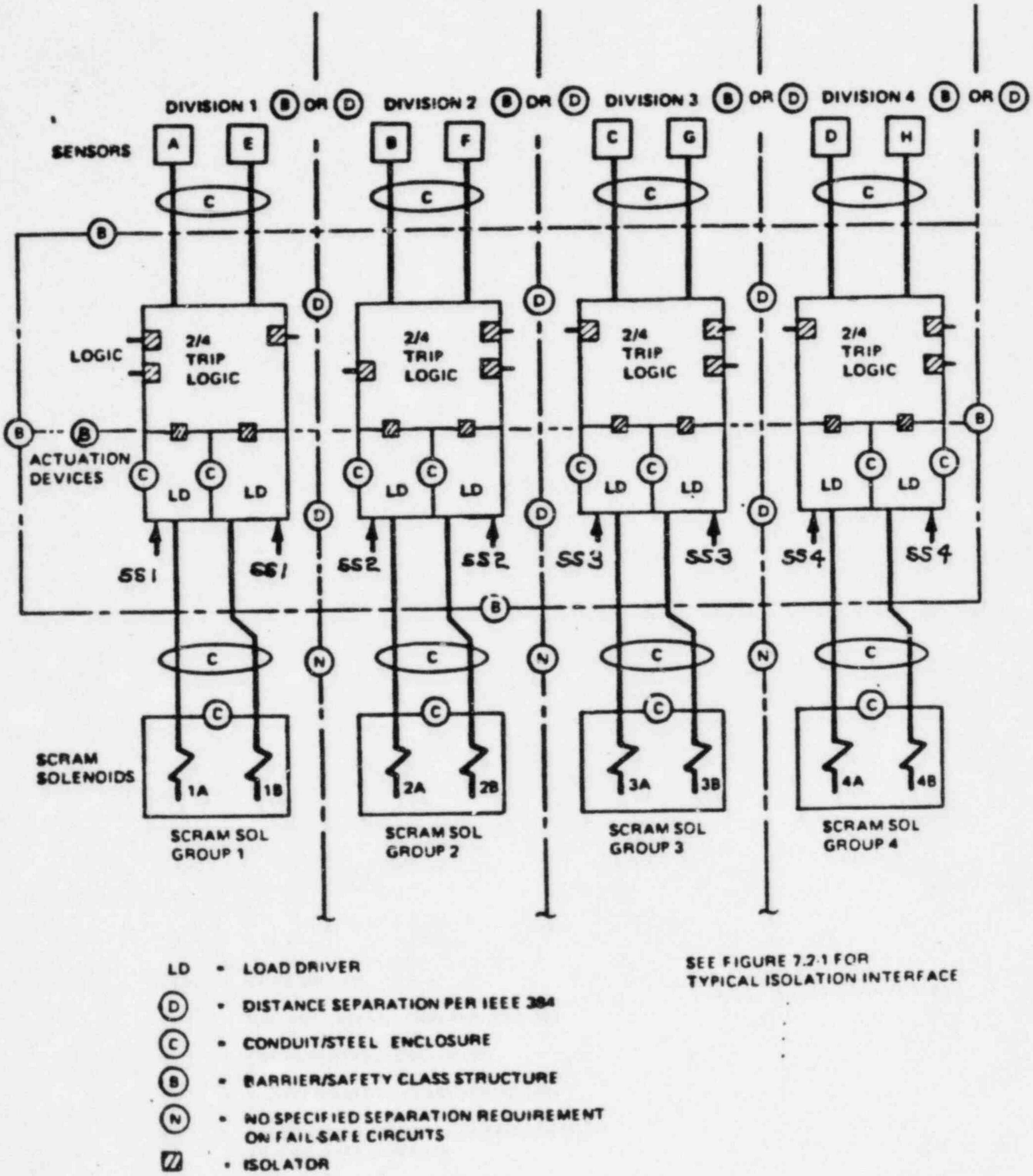


Figure 8.3-25. RPS Separation Scheme

Attachment No. 3

Draft Responses to
Meterology and Effluent Treatment
Branch Questions

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

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

Response

Gas

4.1

1) Maximum Power = 3730 Mwt
2) H^3 production = 50 Ci/yr gas
50 Ci/yr liquid

2

1) Total Steam Flow = 16.4×10^6 lbs./hour
2) Total RPV Inventory = 5.13×10^5 lbs

4.6

1) Holdup Time from Main Condenser to offgas system = 0.0012 hrs.
(Holdup pipe delay time = 10 min.)
2) See GESSAR II Section 11.3 for description of Offgas System performance
- Reduces H_2 from 45#/hr to 0.01#/hr
- Reduces Xe & Kr from 1.2 Ci/sec to 5.1×10^{-5} Ci/sec
I^- at Main Turbine Condenser = 6.3 Ci/yr
3) Mass of charcoal = 24.5 tons
Temperature of charcoal = -3°F, Dewpt = -65°F
K_{bxe} at above conditions = 2032 cc/gm,
K_{bkr} = 93 cc/gm

4) No cryogenic offgas system
5 & 6) H_2O used for turbine gland seal has no appreciable activity
7) See GESSAR II Section 11.3

Response to 460.12 a (Continued)

Gas (Continued)

GESSAR II design does not include filter units. If required, the Applicant will provide these units and the charcoal absorber thickness. Refer to response to 460.19a.

Liquid

1. Requested information for condensate/feedwater system is reported in response to question 460.13 c.

2. "Average" daily flow rates for low conductivity, high conductivity and chemical waste is reported in response to question 460.13 2. Note that the specific waste inputs can occur at various times and for short durations. The daily flows reported represent the expected yearly inputs averaged on a daily basis.

3. PCA inventory for halogens, Soluble/Insoluble fission products and activation products of liquid waste are reported in TABLE 12.2-13

4. Design base df's for radwaste processing equipment is reported in TABLE 11.2-2 and discussed in section 11.2.2.4 .

5. Estimated releases of liquid to the environment is discussed in Section 11.2.3

6. Demineralizer condensate gland seal steam flow is 15,400 lb/hr (per section 2.2.6.1 of NUREG-0016, this sealing steam is considered clean). The holdup time is no less than 30 seconds.

460.13

460.13
d

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

Response: 460.13d

As discussed in Section 11.2.3, the liquid radwaste system is designed with adequate margins so that no liquid waste need to be discharged even under a wide variety of anticipated operational occurrences. Thus no liquid detergent discharge is expected. It is recognized however that unusual operating conditions such as abnormally high inputs of detergent waste or in the event that the detergent evaporator is out of service could necessitate discharge to the canal. The extent of these occurrences would be operational dependent and is the responsibility of the applicant.

A commitment to limit the discharge to the canal is provided on the attached revision to GESSAR II page 11.2-2.

11.2.1.2 Design Criteria (Continued)

specified in 10CFR50, Appendix I. Liquid discharge to the canal may be initiated only from one excess water storage tank and require passage through an administratively closed and locked valve as well as manual initiation. No single error or failure will result in discharge.

The design will maintain occupational exposure as low as practicable in accordance with Nuclear Regulatory Commission Regulatory Guide 8.8 while operating with the design basis fuel leakage.]

The pressure retaining process equipment in the liquid radwaste system is designed with a rated pressure and temperature of 150 psig and 212°F. The evaporator and heating element, however are designed for 350°F. The collection and storage tanks are atmospheric. The mixed-bed demineralizers and evaporators are pressure vessels.

In response to the requirements of General Design Criterion 60, the liquid radwaste system provides one discharge line to the canal for the release of liquid waste with the flow rate of this effluent stream controlled by a flow control valve with feedback from a flow measuring system. Additionally, radiation monitoring equipment is placed on this line to assure that excess activity is not allowed to be discharge in the event of an operational variance. A high radiation signal from this monitor will close the discharge line. The single discharge line is sourced only by the Detergent Drain Tank, a very low level radioactivity source, or one of the two Excess Water Tanks which usually contain condensate quality water. These two sources may not discharge simultaneously because of interlocks on the two valves allowing access to the discharge line. The Excess Water Tank serves as a large (100,000-gal surge capacity) to allow temporary holdup, again precluding the need for discharge of any waste volume for which there is no immediate room available in operating tankage such as Condensate Storage. 11.1

INSERT

INSERT ON PAGE 11.2-2

By administrative control, the discharge from this single discharge line to the canal is limited to a maximum of 5 gpm. This discharge can be increased up to 50 gpm provided the Applicant can demonstrate the discharge will meet the requirements of 10CFR20.

460.13

8

460.13

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

Response:

Information reported on page 11.2-30 is for a single unit radwaste system. The equipment list is essentially the same for a single or dual unit radwaste system. Capacity of some equipment would increase slightly for a dual unit facility.

The detergent waste tank size will be increased to 10,000 gal each as indicated on the attached Figure 11.2-1b.

DEEP BED SINGLE

10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35
1/1.8	6/1	—	—	—	NOTE 6	—	1.8/1	2.2/1	1.1/1	2.2/1	—	—	—	—	1/1	—	1/4.3	6/50	6/1	1/1	1/60		1/85	1/10	—
2/1	20/1	—	—	—	—	—	84/1	17/1	1.1/1	2.2/1	—	—	—	—	1/1	—	1/4.3	—	20/1	2/1	1/60		1/1	1/10	—
26,000	26,000	26,000	26,000	32,000	1700	<1	700	2000	1000	2000	50	—	1000	100	1100	1030	70	90	570	35	—		—	—	—
63,000	63,000	63,000	63,000	127,000	46,000	90	26,000	15,000	1000	2000	1000	—	1000	100	1100	1030	70	—	1900	70	—		8400	5500	—
63,000	4000	—	—	—	—	—	450	900	900	900	—	—	—	—	1100	—	300	15	95	35	1900		8400	5500	—
—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	8	15	1350		8600	260	—
—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	25	—	1	5	8		11	0.57	—
—	—	—	—	—	1.8	—	—	—	—	—	—	—	—	—	—	—	—	—	8	8	9		24	13	—
1.4E-1	1.2E-1	1.1E-1	1.4E-3	1.1E-3	6.8E-1	—	6.5E-1	2.1E-2	1.7E-1	2.1E-2	1.0E-6	—	3.7E-8	2.5E-5	2.8E-8	3.2E-9	1.4E-5	—	—	—	4.5E01		3.0E00	1.7E-4	—
—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	3.7E08		6.0E07	2.8E03	—
120	120	350	350	350	180	—	50	150	50	50	—	25	—	—	5	5	—	2	450	10	120		120	120	400
30	<1	<1	<1	<1	80	—	100	100	100	100	—	—	—	—	—	—	—	—	—	—	—		—	1000	—
30	30	30	<1	<1	80,000	—	—	—	—	—	—	—	2000	—	2000	40	—	—	—	—	—		—	—	—

45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70
0	0	0	0	0	1/1100	0	0	0	0	0	—	1/2.4	—	—	1/6.8		NOTE 7	1/69	1/235	0	6/1	15/60		36/65	—
—	—	—	—	—	1/1	—	—	—	—	—	8/369	3.9/1	—	—	1/1		—	1/1	1/1	1/1	20/1	15/1		38/1	—
0	0	0	0	0	—	0	0	0	0	0	—	—	—	—	—		15	—	—	—	30	—		—	85
63,000	127,000	—	64,000	—	2100	—	—	—	—	—	—	70	—	—	4200		—	2100	2100	2100	100	1125		2700	—
—	—	—	—	—	2100	—	—	—	—	—	—	18	—	—	4200		—	2100	2100	2100	5	75		75	—
—	—	—	—	—	8300	—	—	—	—	—	—	36	—	—	150		—	4300	4300	4300	11	90		240	—
—	—	—	—	—	23	—	—	—	—	—	—	20	10	10	0.43		23	23	22	23	44	44		44	25
—	—	—	—	—	24	—	—	—	—	—	—	—	—	—	9		24	24	24	24	17	9		24	—
—	—	—	—	—	—	—	—	—	—	—	2.2E00	—	—	—	1.4E02		8.8E-1	1.0E01	4.7E-3	—	2.5E01	2.6E02		6.7E00	1.9E00
—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	2.2E09		—	6.5E07	4.0E04	—	2.8E05	2.4E07		1.7E08	—
350	350	350	350	350	300	350	350	350	350	5	550	2	2	2	50		200	200	200	200	—	—		—	100
—	<1	—	—	—	—	>1	>1	<1	<1	—	—	—	—	—	—		—	—	—	—	—	—		—	—
—	<1	—	—	>100	—	>1	>1	<1	<1	—	—	—	—	—	—		—	—	—	—	—	—		—	—

	12	13	14	15	16	17	18	19	20	21
[illegible]	AIR	COND	STM	STM	STM	STM	AIR	COND	AIR	STM
	100	50	150	50	50	50	100	50	50	50
	560	10	2.9E04	2.4E04	3.0E03	50	—	50	180	—
—	—	—	3.5E07	7.8E07	3.5E06	8E04	—	—	—	—
—	—	—	—	—	—	—	—	—	—	—

	8 CONTAINERS / 60 DAYS	
BASIS FT³/CONT. (2)	# / CONTAINER	DESIGN BASIS ACTIVITY CI/CONT (2)
	396	14.5
	6597	7.5
—	396	—
	—	—
—	—	—
—	—	—
—	6050	—
	15,435	22.0

EQUIPMENT NAME	SIZE EACH	QUANTITY
LOW CONDUCTIVITY TANK	[illegible] GAL	2
WASTE FILTER	40 FT²	2
FILTRATE TANK	3000 GAL	1
LOW CONDUCTIVITY DEMINERALIZER	180 FT³	1
BACKUP DEMINERALIZER	180 FT³	1
HIGH CONDUCTIVITY TANK	18,000 GAL	3
WASTE EVAPORATOR	40 GPM	2
CONCENTRATED WASTE TANK	25,000 GAL	1
DISTILLATE TANK	3000 GAL	1
DISTILLATE DEMINERALIZER	180 FT³	1
CLEANUP PHASE SEPARATOR	12,500 GAL	2
SPENT RESIN TANK	13,000 GAL	1
EXCESS WATER TANK	50,000 GAL	2
DETERGENT WASTE TANK	1500 GAL	2
DETERGENT FILTER	25 GPM	1
DETERGENT EVAPORATOR	5 GPM	1
PRECOAT TANK	2000 GAL	1
BODY FEED TANK	300 GAL	1
OIL SEPARATOR	6300 GAL	2
CONDENSATE TANK	1000 GAL	1
ANTIFOAM STORAGE TANK	2 GAL	1
ANTIFOAM MEASURING TANK	0.1 GAL	1
CHEMICAL ADDITION TANKS	300 GAL	3
CAUSTIC TANK	2000 GAL	1
CEMENT SILO	600 FT³	1

10,000 GAL

SUPPORTING DOCUMENTS
1. RADWASTE SYSTEM PD

MPL ITEM NO.
G17-1020

Figure 11.2-1b. Liquid Waste Management System Process Flow Diagram

Response to 460.13i

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

Response

The concentrations of radionuclides in the excess water storage tanks is reported in the attached table. Source terms are for a combination of inputs from the high and low conductivity ~~system~~ subsystem.

Typograpic corrections for I-131 concentration in the concentrated waste tank are shown in the attached Table 12.2-13.

EXCESS WATER TANKS A815

Source Volume = 100000 gal.
Total Curies = 0.56

Halogens

Isotope	Curies
BR-83	3.8E-04
BR-84	1.8E-05
BR-85	0.
I-131	9.0E-02
I-132	2.0E-02
I-133	4.1E-02
I-134	7.0E-04
I-135	1.6E-02
TOTAL	1.7E-01

Soluble Fission Products

Isotope	Curies
SR-89	1.5E-02
SR-90	1.3E-03
SR-91	8.0E-03
SR-92	2.5E-03
Y-90	1.3E-03
Y-91M	5.5E-03
MO-99	2.5E-04
TC-99M	5.5E-03
TC-101	3.8E-06
TE-129M	1.5E-03
TE-132	1.9E-02
CS-134	8.5E-04
CS-136	3.8E-04
CS-137	1.4E-03
CS-138	1.2E-04
BA-137M	1.3E-03
BA-139	1.2E-03
BA-140	3.1E-02
BA-141	1.5E-05
BA-142	1.0E-06
NP-239	2.2E-01
TOTAL	3.2E-01

Insoluble Fission Products

Isotope	Curies
ZR-95	1.9E-04
ZR-97	6.0E-06
NB-95	6.0E-05
RU-103	9.0E-05
RU-106	[illegible]E-05
RH-103M	9.0E-05
RH-106	1.4E-05
LA-140	3.6E-02
CE-141	1.5E-05
CE-143	1.5E-05
CE-144	1.9E-04
PR-143	1.4E-04
ND-147	4.5E-05
TOTAL	3.7E-02

Activation Products

Isotope	Curies
NA-24	3.6E-04
P-32	8.5E-05
CR-51	2.6E-03
MN-54	2.5E-04
MN-56	1.4E-03
CO-58	2.9E-02
CO-60	3.2E-03
FE-59	4.6E-04
NI-65	6.0E-10
ZN-65	1.2E-05
ZN-69M	4.9E-06
AG-110M	3.8E-04
W-187	9.5E-04
TOTAL	3.8E-02

Table 12.2-13 (Continued)

CONCENTRATED WASTE TANK A700

Source Volume = 3000 gal. normal, 25,000 gals. full
Total Curies = 29

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies
BR-83	2.5E-04	SR89	9.4E-02	ZR-95	1.2E-03	NA-24	4.0E-04
BR-84	1.9E-06	SR-90	8.9E-03	ZR-97	7.5E-06	P-32	4.4E-04
BR-85	4.7E-10	SR-91	4.8E-03	NB-95	3.7E-03	CR-51	1.4E-02
I-131	2.6E-01	SR-92	2.6E-04	RU-103	5.2E-04	MN-54	1.5E-03
I-132	6.2E-02	Y-90	8.9E-03	RU-106	8.7E-05	MN-56	9.5E-05
I-133	1.5E-00	Y-91M	3.3E-03	RH-103M	5.2E-04	CO-58	1.7E-01
I-134	1.0E-04	MO-99	7.3E-02	RH-106	8.7E-05	CO-60	2.0E-02
I-135	7.3E-02	TC-99M	1.7E-03	LA-140	1.8E-01	FE-59	2.6E-03
		TC-101	2.7E-07	CE-141	1.0E-03	NI-65	5.6E-07
TOTAL	2.8E-01	TE-129M	7.9E-03	CE-143	3.1E-05	ZN-65	8.4E-05
		TE-132	5.9E-02	CE-144	1.2E-03	ZN-69M	5.0E-06
		CS-134	6.1E-03	PR-143	6.9E-04	AG-110M	2.3E-03
		CS-136	1.9E-03	ND-147	2.2E-04	W-187	1.6E-03
		CS-137	9.4E-03				
		CS-138	3.9E-06	TOTAL	1.9E-01	TOTAL	2.1E-01
		BA-137M	9.4E-03				
		BA-139	5.4E-05				
		BA-140	1.6E-01				
		BA-141	6.8E-07				
		BA-142	1.3E-07				
		TOTAL	1.0E-00				

0 delete —

(NOTE: LAST PAGE ONLY CHANGED)

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

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

Response a

The GESSAR II design does not include the filter units (HEPA filters and charcoal absorber) in the primary containment purge and exhaust. Space for a filter unit is provided in the event an Applicant chooses to include one.

GE considers that the filter unit in primary containment purge and exhaust is not needed. During normal plant operation, the STGS which has filter units, can be initiated to exhaust flow from the primary containment if it is necessary.

The Nuclear Island HVAC design ~~will~~ does not provide space, etc. for installing exhaust filter units, other than the primary containment purge and exhaust system. All ventilation exhaust have process radiation monitors in the exhaust stream that will detect the release of radioactivity. In event that high level of radioactivity is detected, the ventilation exhaust will automatically be shut off and the Standby Gas Treatment System will be automatically actuated to ventilate that area.

The above applies to the secondary containment buildings; that is, Shield Building Annulus, ECCS/RWCU Pump Rooms of the Auxiliary Building and the Fuel Building and the primary containment.

GESSAR II will be revised (as attached) to clarify Nuclear Island exhausts.

Text Modifications for 460.14

9.4.3.4 Inspection and Testing Requirements (Continued)

filters, fans and redundant components to assure system availability. The tests include determination of differential pressures and filter efficiencies, control setpoints and signals, alarm functioning, modulation valve performance, airflow rates, damper functioning, airflow switch operation, isolation butterfly valve functioning and thermal performance of heaters and coolers. Test connections are provided for sampling and monitoring the above-noted categories of performance.

The balance of the system is proven operable by its use during operation. Standby equipment can be tested to ensure proper operation on demand. Equipment layout provides easy access for inspection and testing.

9.4.3.5 Instrumentation Application

Instrumentation and controls for the Auxiliary Building pressure control systems [Figure 9.4-3 (K-163)] are designed for automatic operation. The system fans are started from manual pushbutton stations in the main control room. Airflow failure, sensed by an airflow switch, actuates an alarm, which starts the standby fan and repositions the associated dampers.

Exhaust air from the Auxiliary Building ECCS Area Pressure Control System is continuously monitored for radioactivity. A high level of activity or an ECCS operating signal automatically starts the SGTS, stops the supply and exhaust fans, closes their associated dampers, closes the air supply isolation valves and directs the exhaust air to the SGTS.

The ECCS recirculating fan coil cooling units for RHR pump rooms A, B and C, RCIC, HPCS and LPCS pump rooms are interlocked to start when the pump they protect is started. Also, manual override from pushbutton stations in the main control room is provided.

(460.14 cont)

17

11.5.1.1.2 Systems Required for Plant Operation (Continued)

The radiation monitoring systems (RMS) provided to meet these objectives are:

(1) for gaseous effluent streams -

(a) plant vent discharge,

(b) offgas exhaust vent,

(c) radwaste building ventilation RMS, and

(d) turbine building ventilation RMS;*

(2) for liquid effluent streams -

(a) radwaste effluent RMS and

(b) service water effluent to cooling pond RMS;

(3) for gaseous process streams -

(a) offgas pretreatment RMS,

(b) offgas post-treatment RMS, and

(c) carbon bed vault RMS; and

(4) for liquid process streams -

(a) RHR service water system RMS (loops A and B) and

(b) closed cooling water RMS.

* Applicant responsibility

11.5.2.1.4 Auxiliary Building Exhaust Radiation Monitoring

This system monitors the radiation level exterior to the Auxiliary Building ventilation exhaust duct of the Auxiliary Building ECCS Area Pressure Control System. The system consists of two redundant instrument subsystems, channel A and channel B, which are physically and electrically independent of each other. Each channel consists of a local detector, a converter and a main control room radiation monitor. Power for channel A is supplied from 120-vac RPS Bus E. Power for channel B is supplied from 120-vac RPS Bus F.

Each radiation monitor provides two trip circuits: one for upscale (high) radiation or an inoperative circuit and one for downscale. The upscale/inoperative trip of channel A initiates opening of the exhaust to the SGTS valve, closing of the exhaust to the plant vent valve, closing of the ECCS corridor exhaust valve, and the closing of the RWCU corridor supply valve for Division 1. The same trip also initiates startup of the SGTS, Division 1. The trip of channel B monitor initiates the actuation of the corresponding valves for Division 2 and startup of the SGTS for Division 2.

High radiation and downscale control room annunciators are actuated by the signals from the monitors. Each control room radiation monitor visually displays the radiation level.

INSERT (FROM NEXT PAGE)

11.5.2.1.5 Standby Gas Treatment Radiation Monitoring System

This system monitors the radiation level at the SGTS exhaust duct.

The detectors are physically located downstream of the exhaust and heat removal fans and dampers on the exhaust ducts for Division 1 and Division 2.

INSERT ON PAGE 11.5-11

The exhaust from the Auxiliary Building electrical areas, corridors, & ~~steam tunnel~~ and Elevator Tower HVAC System (Figure 9.4-4a) is through two louvered roof vents and is not monitored. Only the steam tunnel has a potential for gaseous radioactive releases requiring monitoring. The Auxiliary Building portion of the steam tunnel is isolated from the rest of the Auxiliary & ~~Turbine~~ Building and is open to the Applicant's Turbine Building portion of the steam tunnel via the Seismic Interface Restraint Structure. Monitoring a gaseous releases from this area will be accomplished by Turbine Building vent monitoring. The Turbine Building vent monitoring is the responsibility of the Applicant. The control rod drive maintenance area source has been determined to be not significant. The remaining areas exhausted by this system contain no radioactive sources and are isolated from the potentially radioactively contaminated areas of the Auxiliary Building.

Attachment No. 4

Draft Responses to
Reactor Systems
Branch Questions

440.01 Indicate whether the design of your proposed 238 Nuclear Island conforms to the LRG-II positions. If there are any known exceptions at this time, so indicate.

Response

As described in Appendix 1E (Sections 1E.1 through 1E.13), the GESSAR II positions conform to all of the Reactor Systems Branch LRG-II positions with one minor exception; 5-RSB. The following will replace the GESSAR II Position on page 1E.5-2:

"Leak detection capabilities are discussed in the response to NRC question 480.27. Each ECCS room is separate and water-tight. Any suppression pool water loss is therefore limited to the flooding of the largest volume room and redundant equipment in other rooms is protected from flooding. Any leakage from the first isolation valve will not result in a long term equilibrium suppression pool level below the NPSH requirements for the RHR system. The strainers on the intake of the suction line will remain submerged to a depth greater than 7 feet which has been determined to be sufficient for continued operation of the RHR pumps."

440.20 (15.0) We state in the SRP (e.g., in Section 15.1) that for anticipated transients, the most limiting plant systems single failure shall be identified and assumed in the analysis. Accordingly, describe the worst single failure for each events analyzed in Chapter 15 of your FSAR. Provide analyses including these postulated failures for the five most limiting events identified in your FSAR.

Response

The five most limiting analyzed Chapter 15 transients are:

1. Loss of Feedwater Heater-Manual Flow Control (Subsection 15.1.1)

2. Feedwater Control Failure-Maximum Demand (Subsection 15.1.2)

3. Pressure Regulation Downscale Failure (Subsection 15.2.1)

4. Generator Load Rejection with Failure of Bypass (Subsection 15.2.2)

5. Turbine Trip with Failure of Bypass (Subsection 15.2.3)

In reviewing the expected sequence of events utilized in simulating the plant performance for each of these transients, it was determined that postulating a single active safety-related component failure does not alter the transients. For the feedwater control failure - maximum demand transient in which credit is taken for full turbine bypass capacity, a single active component failure would result in the loss of one of the turbine bypass paths. However, the consequence of loosing one bypass path is not expected to result in fuel failure.

Attachment No. 5

Draft Responses to
Mechanical Engineering
Branch Questions

QUESTION 48

Which operational transients will be used for preoperational testing of the non-NSSS piping systems? Which system will be monitored and what locations will be instrumented?

RESPONSE

The operational transients to be used for peroperational testing of the non-NSSS piping systems will be provided by the Applicant. The systems to be monitored and what locations to be instrumented are addressed in the revised subsection 3.9.2.1.2. (Attached.)

(Text change to Question 48)

3.9.2.1.2 PREOPERATIONAL TESTING OF NON-NSSS PIPING

3.9.2.1.2.1 PREOPERATIONAL VIBRATION TESTING

This subsection defines the general requirements for vibration testing of piping systems as specified in Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors". Specific vibration testing requirements are defined in ANSI/ASME OM3-1982*, "Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems". An outline of that standard is given here. Preparation of detailed test specifications by the Applicant will require consulting the complete standard. Instrumentation locations will be provided by the Applicant in accordance with the vibration monitoring group selection (Subsection 3.9.2.1.2.1).

Piping systems to be tested are classified into three vibration monitoring groups, according to required degree of test sophisications. Vibration Monitoring Group 1 (VMG1) requires precise test instrumentation plus some degree of mathematical analysis, where as VMG3 may require only visual observations by competent personnel. VMG2 permits simpler instrumentation than VMG1, such as hand-held on temporarily mounted displacement meters or accelerometers.

3.9.2.1.2.1.1 VIBRATION MONITORING GROUP SELECTION

In selecting monitoring groups for various systems or piping configuration, a general rule is that the most rigorous testing (VMG1) should be applied to systems where vibration has the greatest safety implication. As an example, VMG1 is applied to all Safety Class 1 Piping. VMG1 may also be applicable to Safety Class 2 and 3, if vibration-producing elements (pumps, compressors, relief valves, etc) are present. Table 3.9-24 lists systems being preoperationally tested for vibration. The system classification for vibration testing shall be made by applicant. Preliminary testing may require changes in group selection if unforeseen problems develop.

---

* As revised by the minutes of the Subgroup on Piping Systems (SCVM) meeting held at St. Petersburg, Flordia on December 8 and 9, 1982.

3.9.2.1.2.1.2 ACEPTANCE CRITERIA -- VMGI

For steady - state vibrations, the maximum calculated alternating stress intensity S_{alt} should be limited to:

For ASME Class 1 piping:

$$S_{alt} = \frac{M}{Z} C_2 K_2 F_S \leq \alpha S_{el}$$

Where C_2 = Secondary stress index as defined in ASME Code.

K_2 = Local stress index as defined in ASME Code.

M = Maximum zero-to-peak dynamic moment due to vibration only, or in combination with other loads as required by system design specification.

F_S = Factor of safety applicable to class 1 piping (=1.3). Where the the user demonstrates analytically or by experience that the VG-2 methods are inherently conservative by at least a Factor of 1.3, the factor of safety, Fs, need not be included in the calculation of Salt.

α = 0.8 for materials covered by Figure I-9.1 of Section III of the ASME code.

α = 0.6 for materials covered by Figure I-9.2 of Section III of the ASME code.

S_{el} = Endurance limit (Sa) from Figure I-9.1 or I-9.2 of Section III of the ASME code.

Z = Section modulus of the pipe.

For ASME Class 2 and 3, and ANSI B31: $S_{alt} = C_2 K_2 \frac{M}{Z} F_s \leq \alpha S_{el}$

Where C_2K_2=2i, i = Stress intersification factor, as defined in Subsections NC and ND of the ASME Code or B31.

For transient vibrations, the and maximum alternating stress should be limited as follows.

For ASME Class 1 piping, S_a= allowable alternating peak stress value from Figure I-9.1 or I-9.2 using N_v where $N_v = \frac{(EVLC)}{U_v}$

EVLC = Equivalent number of maximum anticipated vibratory load cycles.

U_v = Unused usage factor = 1-U

U = Cumulative usage factor from ASME Class 1 analysis, which excluded vibratory code.

The maximum alternating stress intensity S_{alt} shall be limited to 0.8 SEL for carbon steel, or 0.6 S_{EL} for stainless steel. For ASME Class 2, 3, and B31 piping, the stresses shall be evaluated the same as for steady state vibration. Alternately, the appropriate ANSI Code shall be used to evaluate the stresses for transient vibrations.

3.9.2.1.2.1.3 ACCEPTANCE CRITERIA -- VMG-2

For testing utilizing deflection measurements, acceptance is based on the following equation.

$$S_{allow} = \frac{\alpha S_{EL}}{10,000\, C_2 K_2 F_s} \delta_n$$

Where S_{allow} = allowable zero to peak deflection limit

S_n = Value of deflection obtained

all other symbols are the same as given for VMG1. For testing utilizing velocity measurement, acceptance is based on the following equations.

$$V_{allow} = \frac{C_1 C_4}{C_3 F_S} \left[\frac{3.64 \times 10^3 \alpha S_{EL}}{C_2 K_2} \right]$$

Where V_{allow} = Allowable velocity, in/second

C_1 = Correction factor to compensate for the effect of concentrated weights along the characterestic span (sec. Fig. 10 of OM3-1982)

C_3 = a correction factor accounting for pipe contents and insulation

$$= 1.0 + \frac{W_F}{W} + \frac{W_{INS}}{W}$$

Where

W = weight of the pipe per unit length (lb/ft)

W_F = weight of the pipe contents per unit length (lb/ft)

W_{INS} = the weight of the insulation per unit length (lb/ft)

= 1.0 for pipe without insulation and either empty or containing steam

C_4 = correction factor for end conditions different from fixed ends and for configurations different from straight spans

= 1.0 for a straight span fixed at both ends, but conservative for any practical end conditions for straight spans of pipe

= 1.33 for cantilever and simply supported pipe span

= 0.74 for equal leg Z-bend

= 0.83 for equal leg U-bend

Appendix D of OM-3 presents examples of correction factors C_1 and C_4 for typical piping spans along with a combination of these factors to provide an initial screening method.

3.9.2.1.2.1.4 ~~A~~CEPTANCE CRITERIA -- VMG3

The acceptability of piping is determined by visual observation, or employing simple devices such as rules, optical wedge, or spring scale. If the level of vibration is too small to be perceived and the possibility of damage is judged to be minimal, the system is acceptable. The judgement as to acceptability can be made only by evaluation of all the following facts as to their effects on piping stress.

a) Vibration magnitude and location

~~a) Vibration magnitude and location~~

b) Proximity to sensitive equipment

c) Branch connection behavior

d) Capability of nearby component supports

Any unique operational characteristics of the systems shall be considered in the evaluations.

If unacceptable vibration levels are indicated by the method listed above, the system must be reclassified as either VMG2 or VMG1.

3.9.2.1.2.1.5 CORRECTIVE ACTION

Should the piping vibration exceed the acceptance criteria, correction action must be taken to make the system acceptable. This action may consist of adding supports, reducing forcing functions, determining and modifying resonant sections, or changing operating conditions. After corrective action is taken, additional testing shall be performed to determine if the vibrations have been sufficiently reduced to satisfy the acceptance criteria.

3.9.2.1.2.2 PREOPERATIONAL THERMAL EXPANSION AND DYNAMIC TESTING

Preoperational thermal expansion and dynamic testing is provided in Subsection 14.2.12.1.75.

Table 3.9-24

PIPING SYSTEMS TO BE VIBRATION TESTED/INSPECTED (PREOPERATIONAL)

Reactor Feedwater System
Reactor Water Cleanup System
Standby Liquid Control System
Residual Heat Removal System
Reactor Core Isolation Cooling System
Reactor Recirculation System
Controlled Drive Hydraulic Systems
Low Pressure Core Spray System
High Pressure Core Spray System
Fuel Pool Cooling and Cleanup System
Leak Detection System
Liquid and Solid Radwaste Systems
Neutron Monitoring System
Offgas System
Upper Pool Storage System
N I Chilled Water System
Demineralized Water and Condensate Distribution
Essential Service Water System
Heated Water Distribution
HPCS Service Water System
Suppression Pool Make-up System
Suppression Pool Cleanup
Essential Bldg. Chilled Water
RHR Service Water Systems
Condensate System
Reactor Protection System

Attachment No. 6

Draft Update of GESSAR II
Relative to R.G.'s 1.31 and 1.44
and NUREG-0313

1.8.31 Regulatory Guide 1.31, Revision 3, Dated April 1978

Title: Control of Ferrite Content in Stainless Steel Weld Metal

This guide describes an acceptable method of implementing the requirements of General Design Criterion 1 of Appendix A to 10CFR50 and Appendix B to 10CFR50 when fabricating and joining austenitic stainless steel components and systems.

Evaluation

As discussed in Subsection 5.2.3, the GESSAR II design complies with this guide but, as discussed below, uses the alternate method of weld pad deposition for coated electrodes as defined in ASME Section III NB2400.

GESSAR II implements the provisions of this regulatory guide. As a result of discussions with the NRC staff, GE conducted a program to demonstrate that the controls applied to filler metal provide adequate delta ferrite in production welds. This concept was accepted by the NRC and is stated in Revision 3, with corollary scope and application requirements.

For GESSAR II plants, all austenitic stainless steel weld filler metal for Class 1, Class 2, and core support components will be supplied with ferrite compositions as defined by Regulatory Guide 1.31, Rev. 3. The weld filler metal additionally must comply with the chemical analysis requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The weld pad deposition technique for coated electrodes stated in ASME Section III, NB2400 is considered an acceptable alternate to the AWS A5.4 technique recommended in Regulatory Guide 1.31, Rev. 3.

[NEW]

1.8.44 Regulatory Guide 1.44, Revision 0, Dated May 1973

Title: Control of the Use of Sensitized Stainless Steel

This guide describes acceptable methods of implementing the requirements of GDC 1 and 4 of Appendices A and B to 10CFR50, with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors.

Evaluation

The GESSAR II design complies with this regualtory guide and with the guidelines of NUREG 0313, revision 1 as well.

All applications of nuclear grade stainless steel are specified as either 304L or 316L (or LN) grade. See revised GESSAR subsections 5.2.3.4 and 4.5.2 for additional discussion. As stated, the General Electric Company is complying with the intent of R. G. 1.44 by controlling the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking through the use of IGSCC resistant materials. In addition, stress rule evaluation is being performed on GE scope of supply to predict other areas where IGSCC might be possible due to high stress. This effort will allow appropriate modifications to be made where appropriate.

Induction Heating Stress Improvement (IHSI) treatment on stainless steel weld joints to preclude intergranular stress corrosion cracking will also be considered and implemented by GE and plant owners when approved the NRC.

[NEW]

4.5 REACTOR MATERIALS

4.5.1 Control Rod System Structural Materials

4.5.1.1 Material Specifications

INSERT ①

a. Material List

The following material listing applies to the control rod drive mechanism supplied for this application. The position indicator and minor nonstructural items are omitted.

(1) Cylinder, Tube and Flange Assembly

		Exception
Flange	ASME SA182 Grade F304 -	< 200°F
Plugs	ASME SA182 Grade F304 -	< 200°F
Cylinder	ASTM A269 Grade TP 304 -	< 200°F
Outer Tube	ASTM A269 Grade TP 304	< 200°F
Tube	ASME SA351 Grade CF-3 -	
Spacer	ASME SA351 Grade CF-3 -	

(2) Piston Tube Assembly

Piston Tube	ASME SA479 or SA249 Grade XM-19	
Nose	ASME SA479 Grade XM-19	
Base	ASME SA479 Grade XM-19	
Ind. Tube	ASME SA312 Type 316 -	(dry environment)
Cap	ASME SA182 Grade F316 -	"

(3) Drive Line Assembly

Coupling Spud	Alloy X-750
Compression Cylinder	ASME SA479 or SA249 Grade XM-19
Index Tube	ASME SA479 or SA249 Grade XM-19
Piston Head	ARMCO 17-4 PH or its equivalent

INSERT 1

Grades F304 & F316 are retained as Control Rod System Structural Mater by the exceptions allowed under Regulatory Guide 1.44.

The specific exceptions are the following:

1) Parts do not see temperature above 200°F during normal operation.

2) Parts are in a non-corrosive enviroment (dry enviroment)

3) Parts are subjected to low stress.

4) Parts exposed to special processing have been under surveillance program that demonstrate that there is no problem in regard to intergranular stress corrosion

4.5.1.1 Material Specifications (Continued) Exceptio

Piston Coupling	ASTM A312 Grade TP 304 or ASTM A269 Grade TP 304	Low Stre
Magnet Housing	ASTM A312 Grade TP 304 or ASTM A269 Grade TP 304 or ASTM A312, A249, or A213 TP 316L –	Low Str

(4) Collet Assembly

Collet Piston	ASTM A269 TP 304 or ASTM A312 TP 304 –	Surveillance program
Finger	Alloy X-750	
Retainer	ASTM A269 TP 304 –	
Guide Cap	ASTM A269 TP 304 –	

(5) Miscellaneous Parts

Stop Piston	ARMCO 17-4 PH or its equivalent
O-Ring Spacer	ASTM A240 Type 304 –
Nut	ASME SA479 Grade XM-19
Collet Spring	Alloy X-750
Ring Flange	ASME SA182 Grade F304
Buffer Shaft	ARMCO 17-4 PH or its equivalent
Buffer Piston	ARMCO 17-4 PH or its equivalent
Buffer Spring	Alloy X-750
Nut (hex)	Alloy X-750

The austenitic 300 series stainless steels listed under ASTM/ASME specification number are all in the annealed condition (with the exception of the outer tube in the cylinder, tube and flange assembly), and their properties are readily available. The outer tube is approximately 1/8 hard, and has a tensile of 90,000/125,000 psi, yield of 50,000/85,000 psi and minimum elongation of 25%.

The balance of Chapter 4 (pages 4.5-3 through 4.5-13/4.5-14) will be reviewed and updated as required to be in compliance with RG 1.44 Rev. 0, RG. 1.31 Rev. 3, and NUREG-0313 Rev. 1. Any exceptions will be noted in a manner similar to that done for revised subsection 4.5.1.

4.5.1.1 Material Specifications (Continued)

The coupling spud, collet fingers, buffer spring, nut (hex), and collet spring are fabricated from Alloy X-750 in the annealed or equalized condition, and aged 20 hours at 1300°F to produce a tensile of 165,000 psi minimum, yield of 105,000 psi minimum, and elongation of 20% minimum. The piston head, stop piston, buffer shaft, and buffer piston are ARMCO 17-4 PH (or its equivalent) in condition H-1100 (aged 4 hours at 1100°F), with a tensile of 140,000 psi minimum, yield of 115,000 psi minimum, and elongation of 15% minimum.

These are widely used materials, whose properties are well known. The parts are readily accessible for inspection and replaceable if necessary.

All materials, except SA479 or SA249 Grade XM-19, have been successfully used for the past 10 to 15 years in similar drive mechanisms. Extensive laboratory tests have demonstrated that ASME SA479 or SA249 Grade XM-19 are suitable materials and that they are resistant to stress corrosion in a BWR environment.

b. Special Materials

No cold-worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the Control Rod Drive (CRD) system. ARMCO 17-4 PH (or its equivalent) (martensitic precipitation hardened stainless steel) is used for the piston head, stop piston, buffer shaft, and buffer piston. This material is aged to the H-1100 condition to produce resistance to stress corrosion cracking in the BWR environments. ARMCO 17-4 PH (or its equivalent) (H-1100) has been successfully used for the past 10 to 15 years in BWR drive mechanisms.

4.5.1.2 Austenitic Stainless Steel Components

a. Processes, Inspections and Tests

Two special processes are employed which subject selected 300 Series stainless steel components to temperatures in the sensitization range:

(1) The cylinder and spacer (cylinder, tube and flange assembly) and the retainer (collet assembly) are hard surfaced with Colmonoy 6 (or its equivalent).

(2) The collet piston and guide cap (collet assembly) are nitrided to provide a wear-resistant surface.

Nitriding is accomplished using a proprietary process. Components are exposed to a temperature of about 1080°F for about 20 hours during the nitriding cycle.

Colmonoy (or its equivalent) hard-surfaced components have performed successfully for the past 10 to 15 years in drive mechanisms. Nitrided components have been used in CRDs since 1967. It is normal practice to remove some CRDs at each refueling outage. At this time, both the Colmonoy (or its equivalent) hard-surfaced parts and nitrided surfaces are accessible for visual examination. In addition, dye penetrant examinations have been performed on nitrided surfaces of the longest service drives. This inspection program is adequate to detect any incipient defects before they could become serious enough to cause operating problems.

Regulatory Guide 1.44

Discussion of the degree of conformance to Regulatory Guide 1.44 is provided in Subsection 4.5.2.4.

4.5.1.2 Austenitic Stainless Steel Components (Continued)

b. Control of Delta Ferrite Content

Discussion of this subject and the degree of conformance to Regulatory Guide 1.31 is presented in Subsection 4.5.2.4.

4.5.1.3 Other Materials

These are discussed in Subsection 4.5.1.1.b.

4.5.1.4 Cleaning and Cleanliness Control

4.5.1.4.1 Protection of Materials During Fabrication, Shipping and Storage

All the CRD parts listed above (Subsection 4.5.1.1) are fabricated under a process specification which limits contaminants in cutting, grinding and tapping coolants and lubricants. It also restricts all other processing materials (marking inks, tape etc.) to those which are completely removable by the applied cleaning process. All contaminants are then required to be removed by the appropriate cleaning process prior to any of the following:

(1) Any processing which increases part temperature above 200°F.

(2) Assembly which results in decrease of accessibility for cleaning.

(3) Release of parts for shipment.

The specification for packaging and shipping the Control Rod Drive provides the following:

4.5.1.4.1 Protection of Materials During Fabrication, Shipping and Storage (Continued)

The drive is rinsed in hot deionized water and dried in preparation for shipment. The ends of the drive are then covered with a vapor tight barrier with dessicant. Packaging is designed to protect the drive and prevent damage to the vapor barrier. Audits have indicated satisfactory protection.

Semiannual examination of the humidity indicators of ten percent of the units is required to verify that the units are dry and in satisfactory condition. This inspection shall be performed with a GE-Engineering designated representative present. Position indicator probes are not subject to this inspection.

Site or warehouse storage specifications require inside heated storage comparable to level B of ANSI N45.2.2.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37.

Regulatory Guide 1.37

General Compliance or Alternate Approach Assessment: For Commitment and Revision Number, see Section 1.8.

4.5.2 Reactor Internal Materials

4.5.2.1 Material Specifications

Materials used for the Core Support Structure:

Shroud Support - Nickel-Chrome-Iron-Alloy, ASME SB166 or SB168.

4.5.2.1 Material Specifications (Continued)

Shroud, core plate, and grid - ASME SA240, SA182, SA479, SA312, SA249, or SA213 (all Type 304L).

Peripheral fuel supports - ASTM A312 Grade TP-304, A479 Type-316L, ASME SA312 Grade Type-304L

Core plate and top guide studs and nuts, and core plate wedges - ASME SA479, SA193 Grade B8A, SA194 Grade 8A (all Type-304)

Control rod drive housing - ASME SA312 TP-304, SA182 Type-304, and ASME SB167 Type Alloy 600.

Control rod guide tube - ASME SA358 Grade 304, SA312 Grade TP-304; ASTM A358 Grade 304, A312 Grade TP-304, A351 Grade CF8, A249 TP-304.

Orificed fuel support - ASTM A249 TP-304, A240 TP-316L, A479 TP-316L.

Materials Employed in Other Reactor Internal Structures.

(1) Shroud Head and Separators Assembly and Steam Dryer Assembly

All materials are TP-304, 304L or 316L stainless steel.

Plate, Sheet and Strip	ASTM A240, TP-304, 304L or 316L
Forgings	ASTM A182 Grade F304 or 304L
Bars	ASTM A276 TP-304 or 316L
Pipe	ASTM A312 Grade TP-304

4.5.2.1 Material Specifications (Continued)

Tube	ASTM A269 Grade TP-304
Castings	ASTM A351 Grade CF8

(2) <u>Jet Pump Assemblies</u>

The components in the Jet Pump Assemblies are a Riser, Inlet Mixer, Diffuser, and Riser Brace. Materials used for these components are to the following specifications:

Castings	ASTM A351 Grade CF8 and ASTM SA351 Grade CF3
Bars	ASTM A276 TP-304, ASTM A479 TP-316L ASTM A637 Grade 688
Bolts	ASTM A193 Grade B8 or B8M and ASME SA479 TP-316L
Sheet and Plate	ASTM A240 TP-304, and ASME SA240 TP-304L, 316L
Pipe	ASTM A358 TP-304, 316L and ASME SA312 Grade TP-304, 316L
Forged or Rolled Parts	ASME SA182, Grade F304, F316L, ASTM B166, and ASTM A637 Grade 688.

4.5.2.1 Material Specifications (Continued)

Materials in the Jet Pump Assemblies which are not austenitic stainless steel are listed below:

a. The Inlet Mixer Adaptor casting, the wedge casting, bracket casting adjusting screw casting, and the Diffuser collar casting are hard surfaced with Stellite 6 (or its equivalent) for slip fit joints.

b. The Diffuser is a bimetallic component made by welding an austenitic stainless steel ring to a forged Alloy 600 ring, made to Specification ASTM B166.

c. The Inlet-Mixer contains a pin, insert, and beam made of Alloy X-750 to Specification ASTM A637 Grade 688.

All core support structures are fabricated from ASME specified materials, and designed in accordance with requirements of ASME Code, Section III, Subsection NG. The other reactor internals are noncoded, and they are fabricated from ASTM or ASME specification materials. Material requirements in the ASTM specifications are identical to requirements in corresponding ASME material specifications.

4.5.2.2 Controls on Welding

Core support structures are fabricated in accordance with requirements of ASME Code Section III, Subsection NG. Other internals are not required to meet ASME Code requirements. Requirements of ASME Section IX BPV Code, are followed in fabrication of core support structures.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products

Wrought seamless tubular products for CRD housings, and peripheral fuel supports, were supplied in accordance with ASME Section III, Class CS, which requires examination of the tubular products by para- radiographic and/or ultrasonic methods according to paragraph NG-2550.

Wrought seamless tubular products for other internals were supplied in accordance with the applicable ASTM or ASME material specifications. These specifications require a hydrostatic test on each length of tubing.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel - Regulatory Guide Conformance

Regulatory Guide 1.31: Control of Stainless Steel Welding

Cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 FN minimum and 8.0 FN average (Ferrite Number). This ferrite content is considered adequate to prevent any micro-fissuring (Hot Cracking) in austenitic stainless steel welds. This procedure complies with the requirements of Regulatory Guide 1.31.

Regulatory Guide 1.44: Control of the Use of Sensitized Stainless Steel

Proper solution annealing of the 300 series austenitic stainless steel is verified by testing per ASTM-A262, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel - Regulatory Guide Conformance (Continued)

Steels." Welding of austenitic stainless steel parts is performed in accordance with Section IX (Welding and Brazing Qualification) and Section II Part C (Welding Rod Electrode and Filler Metals) of the ASME Boiler and Pressure Vessel Code. Welded austenitic stainless steel assemblies require solution annealing to minimize the possibility of the sensitizing. However, welded assemblies are dispensed from this requirement when there is documentation that welds are not subject to significant sustained loads and assemblies have been free of service failure. Other reasons, in line with the regulatory guide, for dispensing with the solution annealing are that assemblies are exposed to reactor coolant during normal operation service which is below 200°F temperature or assemblies are of material of low carbon content (less than 0.025%). These controls are employed in order to comply with the intent of the Regulatory Guide 1.44.

Regulatory Guide 1.37: Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Exposure to contaminant is avoided by carefully controlling all cleaning and processing materials which contact stainless steel during manufacture and construction. Any inadvertent surface contamination is removed to avoid potential detrimental effects.

Special care is exercised to insure removal of surface contaminants prior to any heating operation. Water quality for rinsing, flushing, and testing is controlled and monitored.

The degree of cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel - Regulatory Guide Conformance (Continued)

Regulatory Guides 1.31, 1.44, and 1.37

General Compliance or Alternate Approach Assessment: For Commitment and Revision Number, see Section 1.8.

4.5.2.5 Other Materials

Hardenable martensitic stainless steel and precipitation hardening stainless steels are not used in the reactor internals.

Materials, other than Type-300 stainless steel, employed in reactor internals are:

(1) SA479 Type XM-19 stainless steel;

(2) SB166, 167, and 168, Nickel-Chrome-Iron (Alloy 600); and

(3) SA637 Grade 688 Alloy X-750.

Alloy 600 tubing plate, and sheet are used in the annealed condition. Bar may be in the annealed or cold-drawn condition.

Alloy X-750 components are fabricated in the annealed or equalized condition and aged when required.

Stellite 6 (or its equivalent) hard surfacing is applied to some austenitic stainless steel castings using the gas tungsten arc welding or plasma arc surfacing processes.

All materials, except SA 479 Grade XM-19, have been successfully used for the past 10 to 15 years in BWR applications. Extensive laboratory tests have demonstrated that XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

4.5.3 Control Rod Drive Housing Supports

All CRD housing support subassemblies are fabricated of ASTM-A-36 structural steel, except for the following items:

	Material
Grid	ASTM A441
Disc springs	Schnorr Type BS-125-71-8 (or its equivalent)
Hex bolts and nuts	ASTM A307
6 in. x 1 in. x 3/8 in. tubes	ASTM A500 Grade B

For further CRD housing support information, see Subsection 4.6.1.2.

5.2.3.3.4 Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes (Continued)

Electrodes are distributed from sealed containers or ovens as required. At the end of each work shift, unused electrodes are returned to the storage ovens. Electrodes which are damaged, wet, or contaminated are discarded. If any electrodes are inadvertently left out of the ovens for more than one shift, they are discarded or reconditioned in accordance with manufacturer instructions.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress/Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization

Regulatory Guide 1.44 addresses 10CFR50, Appendix A, GDCs 1 and 4, and Appendix B, requirements to control the application and processing of stainless steel to avoid severe sensitization that could lead to stress/corrosion cracking.

All austenitic stainless steel is purchased in the solution-heat-treated condition in accordance with applicable ASME and ASTM specifications.

Cooling rates from solution heat treating temperatures are required to be rapid enough to prevent sensitization. Non-sensitization is verified using ASTM A262, Practice A methods.

Material changes have been made to minimize the possibility of intergranular stress/corrosion cracking (IGSCC). All welded wrought austenitic stainless steel in the reactor coolant pressure boundary which could be susceptible to stress/corrosion cracking is low carbon nuclear grade Type 304 or Type 316 L or LN ~~or 316L~~ with 0.02% maximum

5.2.3.4.1.1 Avoidance of Significant Sensitization (Continued)

carbon content ~~.~~ and nitrogen control. There is no piping which is service sensitive or nonconforming as defined in NUREG-0313, revision 1.

~~For manual welds with the gas tungsten arc (GTAW) and shielded metal arc (SMAW) welding processes, the heat input was limited by weaving and welding technique restrictions. Non-weaving (stringer bead) techniques were used where possible. When required, weaving was controlled to meet the following bead width limits: for GTAW, the lesser of five times the filler wire diameter or 7/16 inch; for SMAW, the lesser of four times the electrode core wire diameter or 5/8 inch.~~ For machine, automatic, and manual welding, ~~with processes except GTAW and SMAW, heat input was restricted to 50,000 joules per inch.~~ interpass temperature ~~was~~ is restricted to 350°F for all stainless steel welds. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings ~~were~~ are required by specification to have a minimum of 5% ferrite.

~~Whenever any wrought austenitic stainless steel was heated to temperatures over 800°F by means other than welding or thermal cutting, the material was re-solution heat treated.~~

These controls were used to avoid severe sensitization and to comply with Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel.

For commitment and revision, see Section 1.8.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of causing stress/corrosion cracking of austenitic stainless steel components was avoided by