

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

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MFN 194-82
JNF 54-82

December 17, 1982

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

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

Gentlemen:

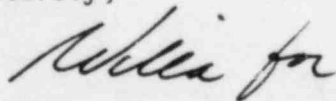
SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR II)
DOCKET NO. STN 50-447

Reference: Letter from G.G. Sherwood (GE) to D.G. Eisenhut (NRC)
pertaining to draft responses to the Commission's
August 25, 1982 request for additional information on
GESSAR II, dated November 10, 1982.

Attached please find final draft responses to the Commission's August 25, 1982 information request. Only modifications (new or revised) to the responses of the referenced letter are provided. If a response is not included in one of the attachments, then the response provided on November 10, 1982 should be considered as the final draft response. Attachment No. 1 summarizes the status of these responses.

Except for the Structural Engineering Branch response, this transmittal completes the Commission's August 25, 1982 request for additional information on GESSAR II. As indicated on Attachment No. 1, the Structural Engineering Branch responses will be provided early January 1983 in conjunction with the audit. An amendment is scheduled for January 1983 to formalize the responses.

Sincerely,



Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

GGs:td

Attachments

cc: F.J. Miraglia (w/o attachments)
D.C. Scaletti

C.O. Thomas (w/o attachments)
L.S. Gifford (w/o attachments)

*E003
Limited Dist*

ATTACHMENT NO. 1

Status of Final Draft Responses to
Commission's 8/25/82 Request

<u>Branch</u>	<u>Final Draft Response</u>
Hydrologic and Geotechnical Engineering	11/10/82 Letter
Chemical Engineering	11/10/82 Letter
Radiation Assessment	Attachment No. 2*
Effluent Treatment Systems	Attachment No. 3
Auxiliary Systems	Attachment No. 4
Power Systems	Attachment No. 5
Procedures and Test Review	Attachment No. 6
Structural Engineering	Submittal in early Jan. 1982 in connection with SEB 11/30/82 - 12/2/82 audit action items

*Attachments to this letter

ATTACHMENT NO. 2

DRAFT RESPONSES TO
RADIATION ASSESSMENT BRANCH
QUESTIONS

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

Response:

Section 12.1.1.3.1 will be revised as indicated on the attached mark-up.

TEXT CHANGE FOR 471.04**12.1.1.1 Design and Construction Policies (Continued)**

(Description on onsite inspections to determine that the design and operation keeps radiation exposures ALARA is, where required, the responsibility of the Applicant.)

12.1.1.2 Operation Policies

(Description of operational policies to maintain occupational doses ALARA is the responsibility of the Applicant.)

12.1.1.3 Compliance with 10CFR20 and Regulatory Guides 8.8, 8.10 and 1.8

Compliance of the Nuclear Island design with Title 10 of the Code of Federal Regulations, Part 20 (10CFR20), is ensured by the compliance of the design and operation of the facility within the guidelines of Regulatory Guides 8.8, 8.10 and 1.8.

12.1.1.3.1 Compliance with Regulatory Guide 8.8

The design of the Nuclear Island fully meets the intent of Revision ³ of Regulatory Guide 8.8, and reflects the commitment of General Electric and its subcontractors. Examples of compliance with all items in Section C.3 of the regulatory guide are delineated in Subsection 12.3.1 of this SAR. Design features of the Nuclear Island allow the Applicant to comply easily with the recommendations of Subsection C.4 of the guide. For instance, provisions are made in systems such as the Reactor Water Cleanup System (RWCS) to allow flushing of the piping in shielded cubicles before entry, and to use remote reach rods. Breathing air headers are provided in areas where past experience indicates airborne radioactivity has been a problem. Design provisions allow for remote operation of fuel handling and radwaste cask filling.

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

Response

Insert the following paragraph at the end of subsection 12.1.2.1:

The features of the plant design which ensure that the plant can be operated and maintained with ALARA exposures will also serve to assist in achieving ALARA exposures during the decommissioning process. Examples of features which will assist in maintaining low occupational exposures during decommissioning include the following:

- (a) Provisions for draining, flushing, and decontaminating equipment and piping.
- (b) Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of crud traps.
- (c) Shielding which provides protection during maintenance or repairs and during decommissioning operations.

471.05 continued

- (d) Provision of means and adequate space for utilization of moveable shielding.
- (e) Separation of more highly radioactive equipment from less radioactive equipment and provision of separate shielded compartments for adjacent items of radioactive equipment.
- (f) Provision of access hatches for installation or removal of plant components.
- (g) Provision of design features such as the Reactor Water Cleanup system and the Condensate Demineralizer to minimize crud buildup.

471.06
(12.2.2)

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

Response

Add the following to Subsection 12.2.2.2:

A conservative assessment of airborne activity in the containment during normal plant operation was performed by assuming a continuous steam leak rate of 2000 pounds per hour from the safety relief valves to the suppression pool and neglecting the benefit of the suppression pool cleanup system. The estimated airborne concentrations are acceptably low as illustrated by the summary results in Table 12.2-24.

471.06 Continued

Events that result in venting of steam through the Safety Relief Valves cause fission products to be released to the suppression pool. ^{Sub-}Section 15.2.4.5 provides a discussion of the analysis of a worst-case transient chosen to envelope the radiological consequences of these events and includes resulting estimates of doses received by personnel while leaving the containment during this event. The transient analyzed was based on closure of the Main Steam Isolation Valves with subsequent depressurization of the reactor vessel. All personnel are expected to be out of the containment within a period of less than four minutes after Safety Relief Valve actuation. Following this period, activity continues to build up in the containment as depressurization continues. Subsequent re-entry to the containment for plant restart can be carefully controlled and would be performed with full breathing protection if necessary. Table ^{12.2-25} presents the results of airborne activity calculations for the enveloping transient which are oriented toward re-entry considerations. The values at 8 hours are representative of short-term, low-occupancy re-entry while the 48 hour values are representative of longer term re-entry when continuous occupancy is desirable. Containment purge initiation at 0 hours was assumed for the latter case.

Table 12.2-24

Normal Operation Maximum and Typical Radiological Parameters

<u>Radiological Parameter</u>	<u>Maximum (Sumps Area)</u>	<u>Typical (CRD Area)</u>
Maximum C/MPC For Any Isotope (Excluding Noble Gases)	0.051	0.034
Total C/MPC For All Isotopes (Excluding Noble Gases)	0.079	0.054
Airborne Dose Rates:		
Lens of Eye, mrem/hr	0.01	0.003
Skin beta, mrem/hr	0.02	0.01

Table 12.2-25
Containment Re-entry Parameters with
Suppression Pool Cleanup

<u>Radiological Parameter</u>	<u>Value at 0 hrs</u>	<u>Value at 48 hrs</u>
Maximum C/MPC For Any Isotopes (excluding noble gases)	1190 (Rb-88)	0.12 (Rb-88)
Total C/MPC For All Isotopes (excluding noble gases)	1190 (Rb-88)	0.13 (Rb-88)
Airborne dose rates:		
Lens of eye, mrem/hr	1100	1.4
Skin beta	4600	74

471.07
(12.2.2)

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

Response

Any residual gases which might be entrapped in the vessel head can be vented prior to head removal. It is recommended by General Electric that this practice normally be followed during refueling or other operations which involve removal of the vessel head.

471.08
(12.3.1)

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

Response

Typical nickel and cobalt contents of the principal materials which are in contact with the coolant are given ~~in Table 1.~~

Table 1.

471.08 continued.

Carbon steel is used in a large portion of the system piping and equipment outside of the Nuclear Steam Supply System. Carbon steel is typically low in nickel content and contains a very small amount of cobalt impurity.

Stainless steel is used in portions of the system such as the reactor internal components, recirculation system piping, and heat exchanger tubes where high corrosion resistance is required. The nickel content of stainless steels is in the 8 to 10.5 percent range and is controlled in accordance with applicable ASME material specifications. No cobalt content limits have been specified for stainless steels used within the reactor vessel with the exception that low-cobalt (less than 0.05 percent) XM-19 alloy has been used in the control rod drives. Cobalt

471.08 continued

content in stainless steels used outside the vessel has not been limited to 0.2 percent; however, a previous review of many materials certifications indicated an average cobalt content of only 0.15 percent in austenitic stainless steels.

Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750 which have high nickel content are used in some reactor vessel internal components. These materials are used in applications for which there are special requirements to be satisfied (such as possessing specific thermal expansion characteristics along with adequate corrosion resistance) and for which no suitable alternate low-nickel material is available. Cobalt content in Inconel X750 used in the fuel assemblies is limited to 0.05 percent.

Stellite is used for hard facings of components

471.08 continued

which must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. An alternative material (Colmonoy) has been used for some hard facings in the core area.

Table 1
Typical Nickel and Cobalt Content of Materials

Material	Nickel (%)	Cobalt (%)
Carbon steel	0.25	1% of Ni
Stainless steel	10	1% of Ni
Ni-Cr-Fe (Inconel 600, Inconel X750)	70	1% of Ni
Stellite 6	3	58

471.09

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

Response

The most comprehensive study of cobalt inputs to the BWR system which is available was carried out under sponsorship of the Electric Power Research Institute. Results of the study were published in ERI NP-2263 (Reference 11). This report contains tabulations by system component of the surface areas of materials exposed to the coolant, cobalt and nickel contents of these materials, and estimates of the material corrosion rates and corresponding cobalt input rates to the system.

471.10

Provide the results of cost/benefit analyses evaluating the effects of reducing the cobalt content of cobalt contributing materials and components (e.g., the reactor vessel internals at core vicinity, the reactor pressure vessel cladding, the primary recirculation loop and the feedwater piping). This cost/ benefit evaluation should be done for the cobalt content reduced to 0.25, 0.10 and 0.05 weight percent. In addition, correct the radiation survey data for the ⁶⁰Cobalt housing in Table 12.2-19 of your FSAR, which indicates 3000 mr/hr before cleaning and 4000 mr/hr after cleaning.

Response

The present state of knowledge concerning the mechanisms of cobalt transport and deposition is inadequate to support quantitative prediction of the benefits of reducing the cobalt contents of cobalt bearing materials. Existing analytical models assume that the buildup of radiation levels on the piping is directly correlated with the cobalt-60 concentration of the reactor water. Recent operating plant data, however, reveal numerous contradictions to this assumption including instances where the trend in radiation levels has been upward while the cobalt-60 concentration has been decreasing or stable and a range

471.10 continued

of radiation levels which spans nearly a factor of 20 observed at plants which have approximately equal cobalt-60 concentrations. It now appears that some other as-yet undetermined aspect (or aspects) of water chemistry is rate-controlling with respect to incorporation of cobalt-60ⁱⁿ deposits on out-of-core surfaces. Consequently, it is not clear that any dose rate reduction would be obtained based solely on reduction of material cobalt content. While there is presently no way to quantify the potential benefits of such reductions, it is anticipated that ongoing evaluations of operating plant data will lead to further revisions and refinements of the analytical models. Reliable cost-benefit analyses may therefore become feasible in the future and, if favorable, lead to implementation of cobalt content limits.

The after-cleaning dose rate in Table 12.2-19 will be corrected from 4000 mR/hr to 2000 mR/hr as shown on the attached markup.

Table 12.2-19
RADIOACTIVE SOURCES IN CONTROL ROD DRIVE SYSTEM

Control Rod Drive Radiation Survey Data

<u>Component</u>	<u>Gamma Dose Measured at Contact MR/hr</u>			
	<u>Before Cleaning</u>		<u>After Cleaning</u>	
	<u>Maximum</u>	<u>Average</u>	<u>Maximum</u>	<u>Average</u>
Spud	10,000	600	500	110
Filter	23,000	3,500	20,000	300
Collet Housing	3,000	1,800	2,000 4,000	700
Outer Cylinder	1,200	60	80	40
Strainer	8,000	1,800	1,000	500
Flange	1,000	200	400	150

Control Blade Principal Isotopes

Curies (135 Gwd/Te 7-Days Cooled)

<u>Isotope</u>	<u>Ci/Blade</u>
Cr51	1.4E5
Mn54	9.1E3
Fe55	1.6E5
Co58m	7.7E3
Co58	8.8E3
Co60	1.1E5
Ni63	5.0E3
Total	4.4E5

QUESTION 471.11

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

RESPONSE 471.11

Response to this question is included in ^{modified} Subsection 12.3.2.3(4), Plant Shielding Description - Fuel Building.

12.3.2.3 Plant Shielding Description (Continued).

operation, dose rates in the pump area are less than 1 mR/hr. During an isolation transient, however, dose rates in the area temporarily increase to 700 mR/hr. Due to the nature of the event, egress from the area can be accomplished well before dose rates reach this level.

Access to equipment in this area is not required during this occurrence. *An individual in this area will know that the dose rate is increasing since a local-mounted area radiation monitoring sensor, converter, indicating auxiliary unit, and audio alarm are provided.*

The two redundant SGTS filter units are located in separately shielded cubicles in the Fuel Building.

Shielding of the fan from the filter unit provides a lower radiation area for fan maintenance. Operation of the SGTS does not require entry into the SGTS filter area.

- (5) Control Room - The dose rate in the control room is much less than 1 mR/hr during normal reactor operating conditions. The outer walls of the Control Building are designed to attenuate radiation from radioactive materials contained within the Shield Building and from possible airborne radiation surrounding the Control Building following a LOCA. The walls provide sufficient shielding to limit the direct-shine exposure of control room personnel following a LOCA to a fraction of the 5-Rem limit as is required by 10CFR50, Appendix A, Criterion 19. Shielding for the outdoor air cleanup filters is also provided to allow temporary access to the mechanical equipment area of the Control Building following a LOCA, should it be required.
- (6) Radwaste Building - Shielding for the Radwaste Building is designed to limit radiation levels in the open corridors, control room and HVAC and Electrical Equipment

471.12
(12.3.2)

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

Response

The radiation zones depicted in Figures 12.3-16, 12.3-17, 12.3-18, and 12.3-19 should be less than 1 millirem per hour zones. The "POST-LOCA" labeling of these drawings should also be removed.

471.13
(12.2.2)

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

Response

The assumption of 24 drywell purges per year was chosen to provide a conservative basis for bounding the estimated annual releases via this path. In view of the fact that the I-131 releases thus obtained are small compared to those from other release paths no further refinement of the assumptions was attempted.

471.14
(12.3.2)

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

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

Response

See revised subsection 12.3.2.3
(Insert Item B)

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

Response

See revised subsection 12.3.2.3
(Insert Items A and C)

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

Response

The concrete shielding dimensions of the spent fuel transfer tube shielding are shown on revised Figure 19.3.12.3-6.

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

Response

See revised subsection 12.3.2.3 and Figure 19.3.12.3-3.

12.3.2.3 Plant Shielding Description (Continued)

demineralizer cubicle, which is infrequently required, is via a stepped shield plug at the top of the cubicle. The bulk of the piping and valves for the filter demineralizers is located in an adjacent shielded valve gallery. Backflushing and resin application of the filter demineralizers are controlled from an open corridor, where dose rates are less than 1 mR/hr. The RWCS backwash receiving tank is also separately shielded from the other components of the RWCS, including the tank discharge pump, which allows maintenance of the pump without direct exposure to the spent resins contained in the backwash tank. A shielded labyrinth entry to the backwash tank cubicle reduces the dose rates at the entry to less than 1 mR/hr.

The fuel storage pool is shielded to maintain accessible areas around the pool at less than 1 mR/hr. Transfer of fuel from the Reactor Building to the Fuel Building is through an inclined fuel-transfer tube. A reinforced concrete structure surrounding the tube reduces radiation levels to less than 5 mR/hr during fuel transfer. Access to the fuel transfer tube is through a hatch shielded by a stepped, composite concrete and lead shield plug.

INSERT ITEM A.

TP
The main steamline, RWCS piping and RHR piping are routed through the shielded steam tunnel. The steam tunnel walls are designed to reduce the dose rates to less than 1 mR/hr in areas along the sides of the steam tunnel and less than 5 mR/hr both above and below the steam tunnel.

INSERT ITEM

B

Shielding of the Traversing Incore Probe (TIP) modules is provided by wing walls and a stepped, shielded floor at the (+)11 ft 0 in. elevation above. During normal

12.3.2.3 Plant Shielding Description (Continued)

operation, dose rates in the pump area are less than 1 mR/hr. During an isolation transient, however, dose rates in the area temporarily increase to 700 mR/hr. Due to the nature of the event, egress from the area can be accomplished well before dose rates reach this level. Access to equipment in this area is not required during this occurrence.

The two redundant SGTS filter units are located in separately shielded cubicles in the Fuel Building. Shielding of the fan from the filter unit provides a lower radiation area for fan maintenance. Operation of the SGTS does not require entry into the SGTS filter area.

INSERT
ITEM C

TP

- (5) Control Room - The dose rate in the control room is much less than 1 mR/hr during normal reactor operating conditions. The outer walls of the Control Building are designed to attenuate radiation from radioactive materials contained within the Shield Building and from possible airborne radiation surrounding the Control Building following a LOCA. The walls provide sufficient shielding to limit the direct-shine exposure of control room personnel following a LOCA to a fraction of the 5-Rem limit as is required by 10CFR50, Appendix A, Criterion 19. Shielding for the outdoor air cleanup filters is also provided to allow temporary access to the mechanical equipment area of the Control Building following a LOCA, should it be required.
- (6) Radwaste Building - Shielding for the Radwaste Building is designed to limit radiation levels in the open corridors, control room and HVAC and Electrical Equipment

INSELT
ITEM A

The applicant will provide unique high radiation signs at the entry to the fuel transfer cubicle to warn of the high radiation levels within the cubicle during fuel transfer.

- Lead shielding is provided above the refueling bulkhead, under the path of spent fuel being transferred from the reactor to the upper fuel storage pool. The shielding provides radiation protection for personnel ~~not~~ working in the upper levels of the drywell during the spent fuel transfer process. The shielding consists of stainless steel clad ^{lead} sheets running parallel to the 180° azimuth varying in thickness from 4 inches of lead directly below the transfer path to 1.5 inches of lead at locations beyond the boundaries of the transfer path. During the transfer process, temporary rails are erected to ensure that a ~~fallen~~ dropped fuel assembly remains on the 4 inch thick lead section.

~~INTERLOCK~~

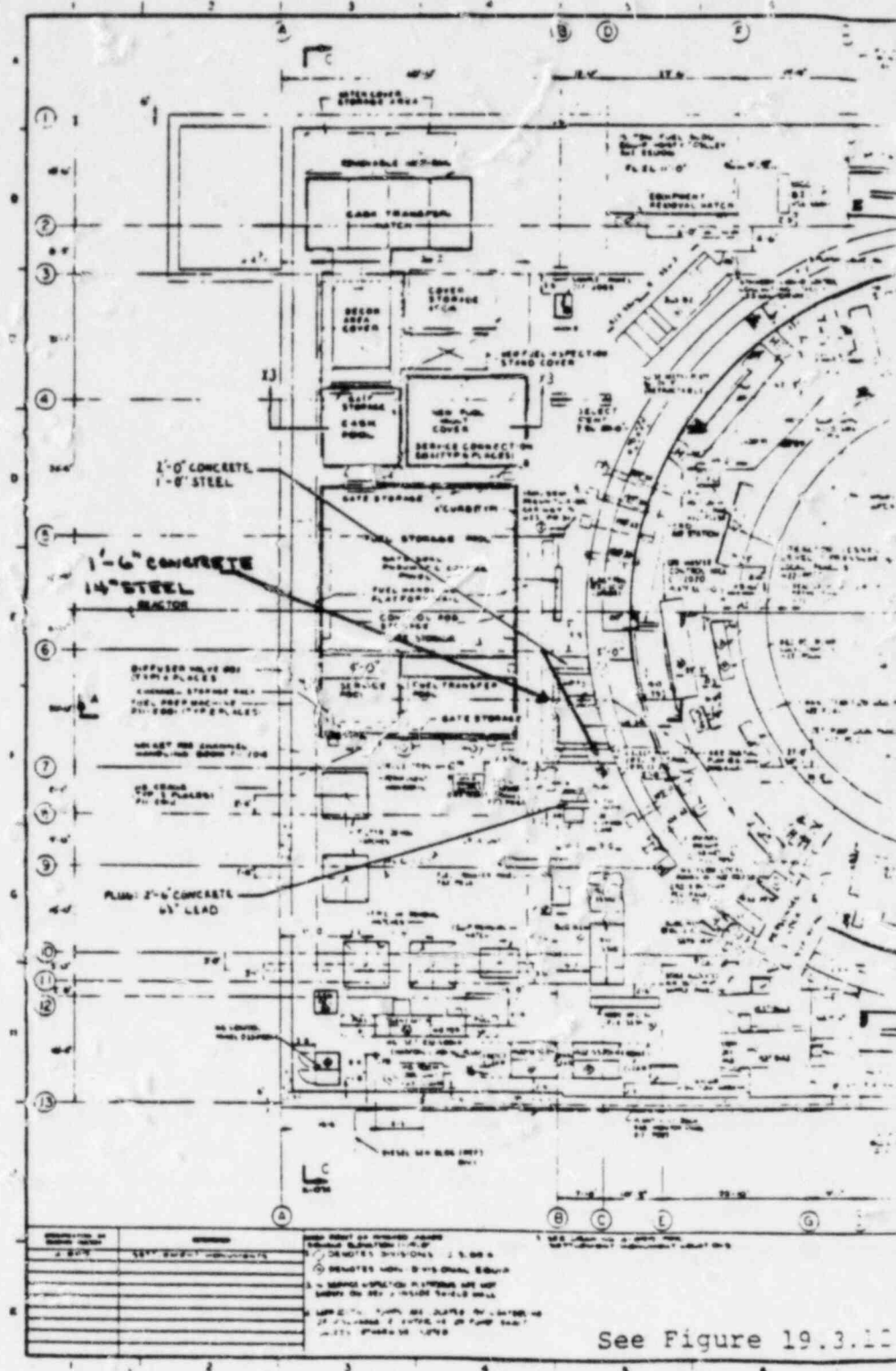
AN INTERLOCK IS PROVIDED TO PREVENT SPENT FUEL PASSAGE WHEN EITHER OF THE SHIELD PLUGS, AT 11 FOOT OR 26 FOOT ELEVATION, ARE NOT CLOSED AND LOCKED.

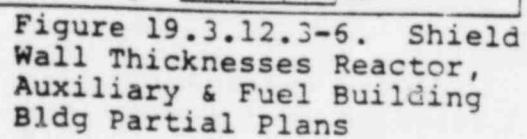
CONVERSELY, INTERLOCKS ARE PROVIDED WITH THE 11 FOOT AND 26 FOOT ELEVATION SHIELD PLUGS TO PREVENT SHIELD PLUG REMOVAL WHEN SPENT FUEL PASSAGE CAN OCCUR.

OR UNLOCKING

Insert Item C

Access to the lower fuel transfer tube cubicle is through a hatch, shielded by a stepped, composite concrete and lead shield plug located at coordinate FS of Figure 19.3.12.3-3. Entry into the shield annulus is made at the Shield Building opening also located at coordinate FS. Upon entering the shield annulus, the ladder shown on Figure 19.3.12.3-6 at coordinate C40 is used to climb to the level of the lower fuel transfer tube valve located in the shield annulus. The applicant will provide unique high radiation signs at the entry of the fuel transfer cubicle to warn of the high radiation levels within the cubicle during fuel transfer.





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

Response

See revised subsection 12.3.2.3

471.16

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

Response

and personnel exposure

The following release terms were used as a basis for shielding calculations.

	% Core Inventory
Reactor Coolant	
Noble Gases	100%
Halogens	50%
All Others	1%
Containment Atmosphere	
Noble Gases	100%
Halogens	25%

471.17

In paragraph (4) of Item II.8.2 of NUREG-0737, we state that you should submit post-accident dose rate maps for potentially occupied areas and indicate the projected doses to individuals who must be in vital areas for certain necessary occupancy times. Accordingly, provide post-accident radiation zone maps and the estimated doses received by individuals assigned to perform the following functions:

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

Response

Add new sections 12.3.5 and 12.3.6
(Insert A)

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

Response

The estimated exposures to personnel as a result of sampler operation ~~assuming~~
~~Regulatory Guide 1.19 Source Terms~~ one hour after the accident are as follows:

Source	3 ft from sampler
Liquid Sample	17.5 ^{mR} MR *
Gas Sample	60 ^{mR} MR *

Exposure due to background radiation is unique and will be supplied by the applicant.

* Assuming a sample collecting time of 10 minutes.

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

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

Response:

The locations of the post-accident sampling area and the sample analysis area will be specified by the applicant. The applicant will provide the estimated doses received by personnel performing analyses of the samples.

12.3.5 POST-ACCIDENT ACCESS REQUIREMENTS

The locations requiring access to mitigate the consequences of an accident during the 100 day Post-accident period are the control room, the technical support center, the three manual valves in the Auxiliary and Fuel Building, and the ADS bottles in the Fuel Building. The three manual valves are the ESW supply to the hydrogen mixing blowers of the Combustible Gas Control system and a Drywell Bleed-off Vent system valve. Access to the ADS bottles is required to replace them, in the event that the air supply in the bottles is depleted. Replacement of the bottles ~~are~~ is not required prior to 7 days following the accident.

The control room and the technical support center, which are continuously occupied, are designed to reflect the criteria in Section 3.1.2.2.10.

The doses received by personnel entering the Fuel and Auxiliary Building to operate the ~~3~~ three manual valves and to replace the ADS air bottles are listed in the table below.

For entry into the Fuel Building, it is assumed that self-contained full facepiece respirators operating in the pressure mode is used. The exposure time anticipated for operating each valve is 5 minutes, and for exchanging the ADS bottles is 30 minutes.

<u>OPERATION</u>	<u>GAMMA DOSE</u>	<u>BETA DOSE</u>	<u>THYROID DOSE</u>
OPERATING ESW VALVE (P41-FF17) IN FUEL BUILDING	0.3	0.2	0.1
OPERATING DRYWELL BLEED-OFF VENT VALVE (T41-FF051) IN FUEL BUILDING	0.3	0.2	0.1
OPERATING ESW VALVE (P41-FF116) IN AUXILIARY BUILDING	1.9	-	-
ADS BOTTLE REPLACEMENT - FUEL BUILDING	0.9	0.5	0.4

12.3.6 POST-ACCIDENT RADIATION ZONE MAPS

The Post-accident radiation zone maps for the areas in the Shield Building, Auxiliary Building, ^{Diesel Generator Building} and Fuel Building are presented in Figures 12.3-20 through 12.3-28. The zone maps represent the maximum gamma dose rates that exist in these areas during the Post-accident period. These dose rates do not include the airborne contribution in the Auxiliary, Fuel, and Diesel Generator Building.

The radiation zone number assigned to each zone is explained in the table below.

<u>ZONE DESIGNATION</u>	<u>MAXIMUM POST-ACCIDENT DOSE RATE (R/hr)</u>
I	< 0.5
II	< 5
III	< 50
IV	< 500
V	< 5000
VI	> 5000

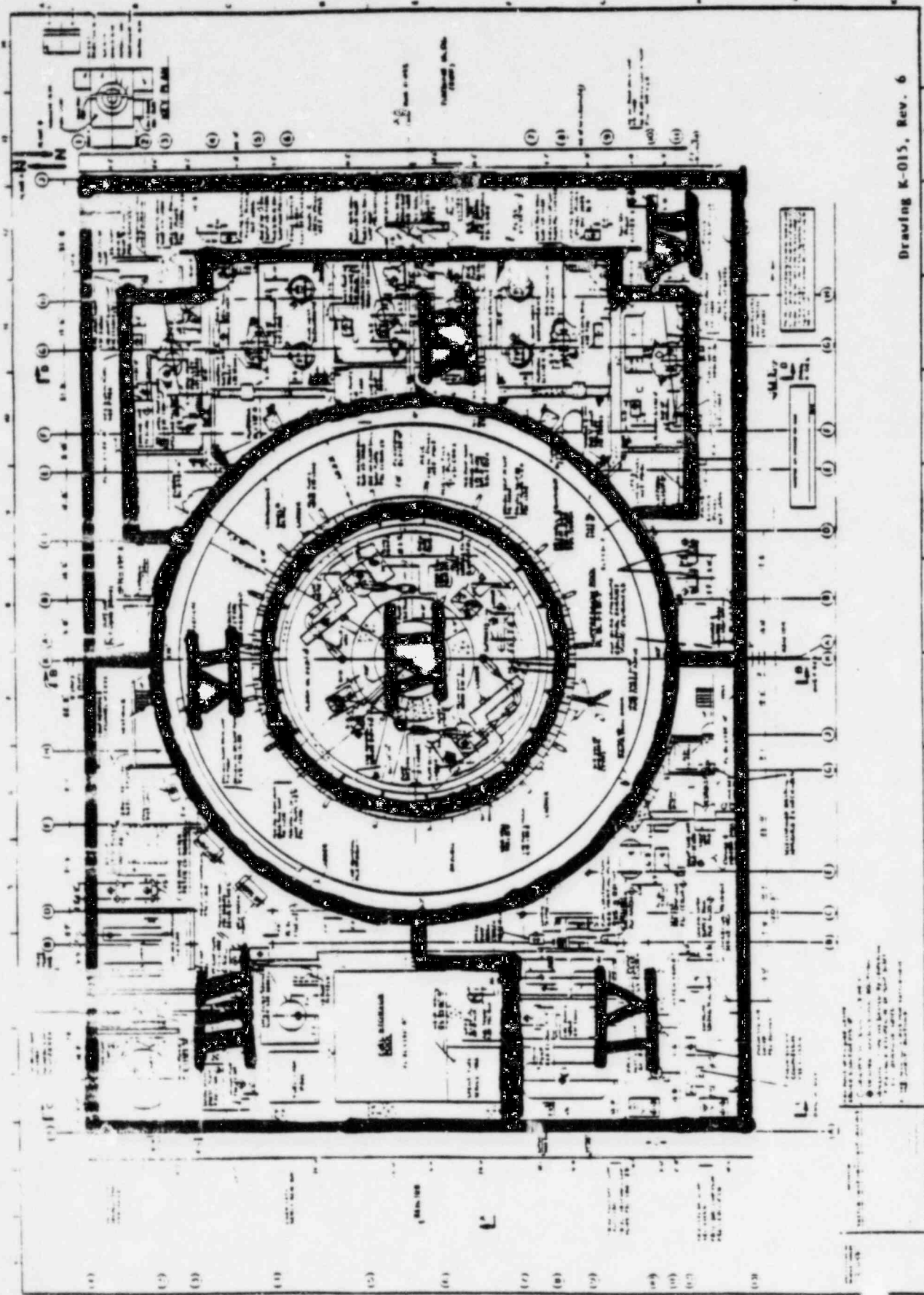
2

Post-accident zone maps of the Control Building are not presented since the Building is continuously occupied and is designed to reflect the criteria established in Section 3.1.2.2.10.

12.3.6.1 TIME DEPENDENT POST-ACCIDENT ZONE DOSE RATES

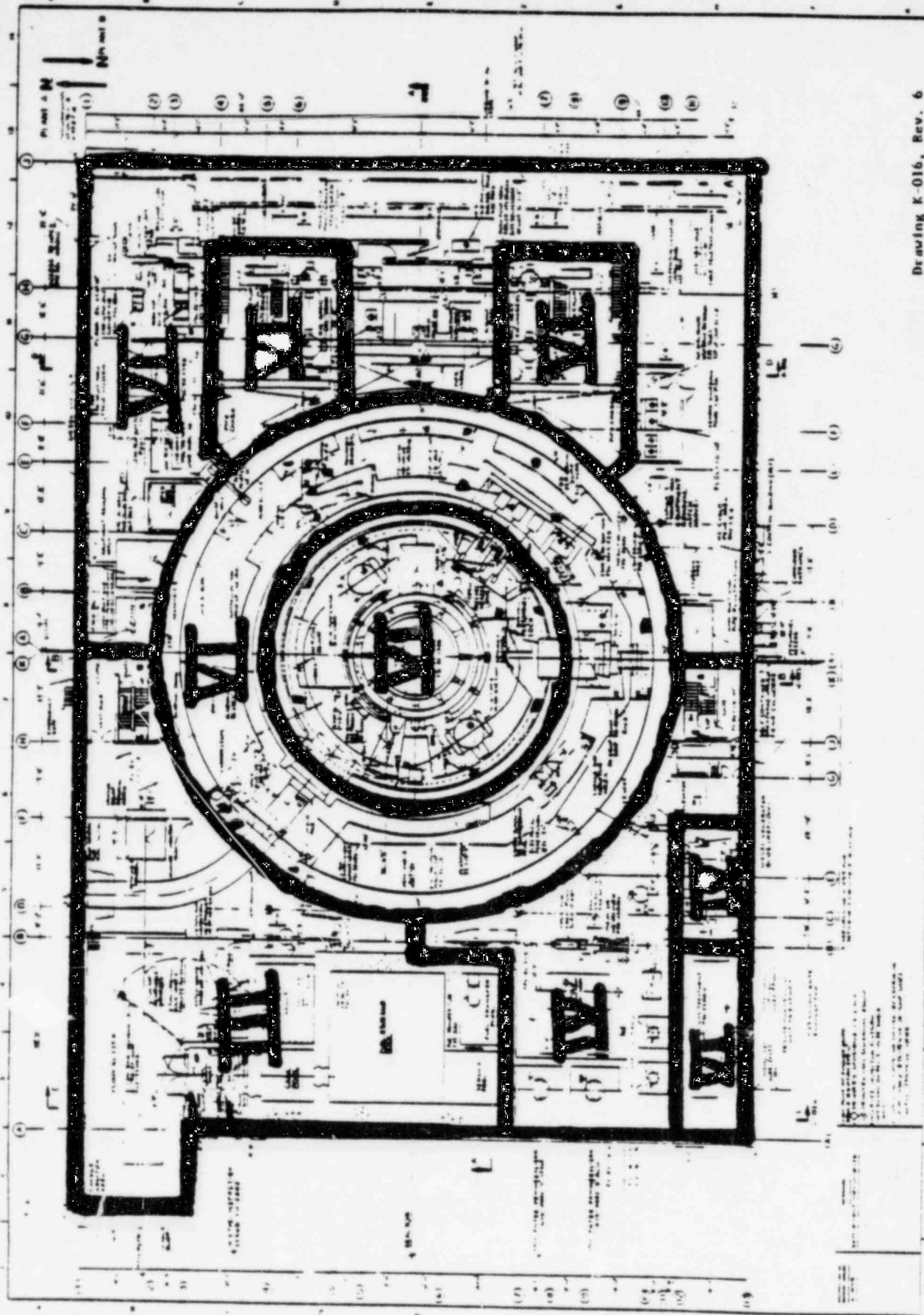
The time-dependent dose rates for the zones identified in Figures 12.3-20 through 12.3-28 are described in Figures 12.3-29 through 12.3-34. The curves can be used to predict the dose rates in selected areas of the plant during the Post-accident period.

Q2 II
238 NUCLEAR ISLAND



REACTOR, AUXILIARY &
FUEL BUILDING ARRGT
FIGURE 12.3-2.0 Plan at E1(-) 32'-0"

CESSAR II
238 NUCLEAR ISLAND

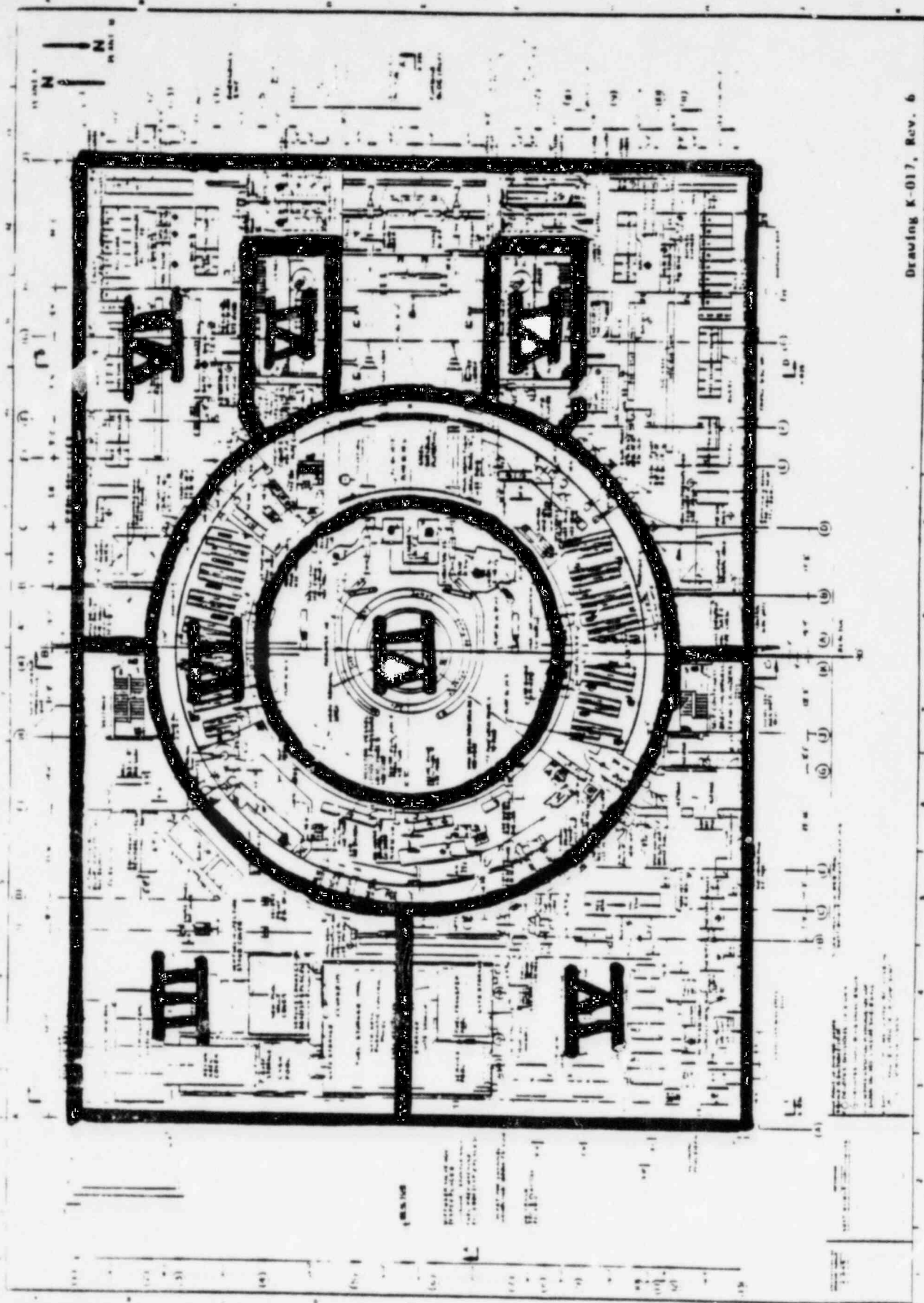


Drawing K-016, Rev. 6

Reactor, Auxiliary
& Fuel Building Arrgt
Plan at El(-16'-10")

FIGURE 12.3-21

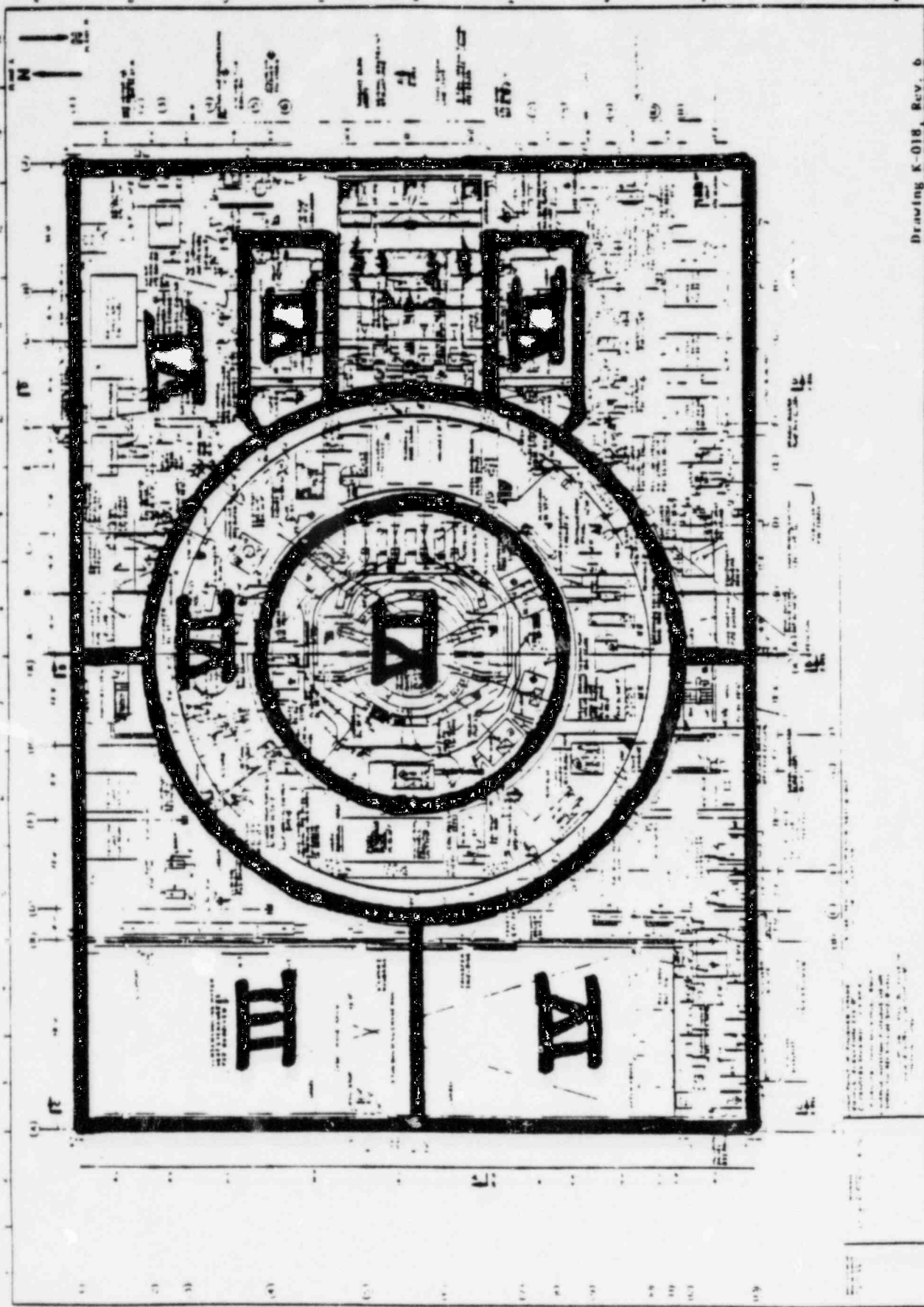
CESSAB II
218 NUCLEAR ISLAND



Reactor, Auxiliary
& Fuel Building Arrgt
Plan at El 11'-0"

FIGURE 12.3-22

12. 11
218 NUCLEAR ISLAND



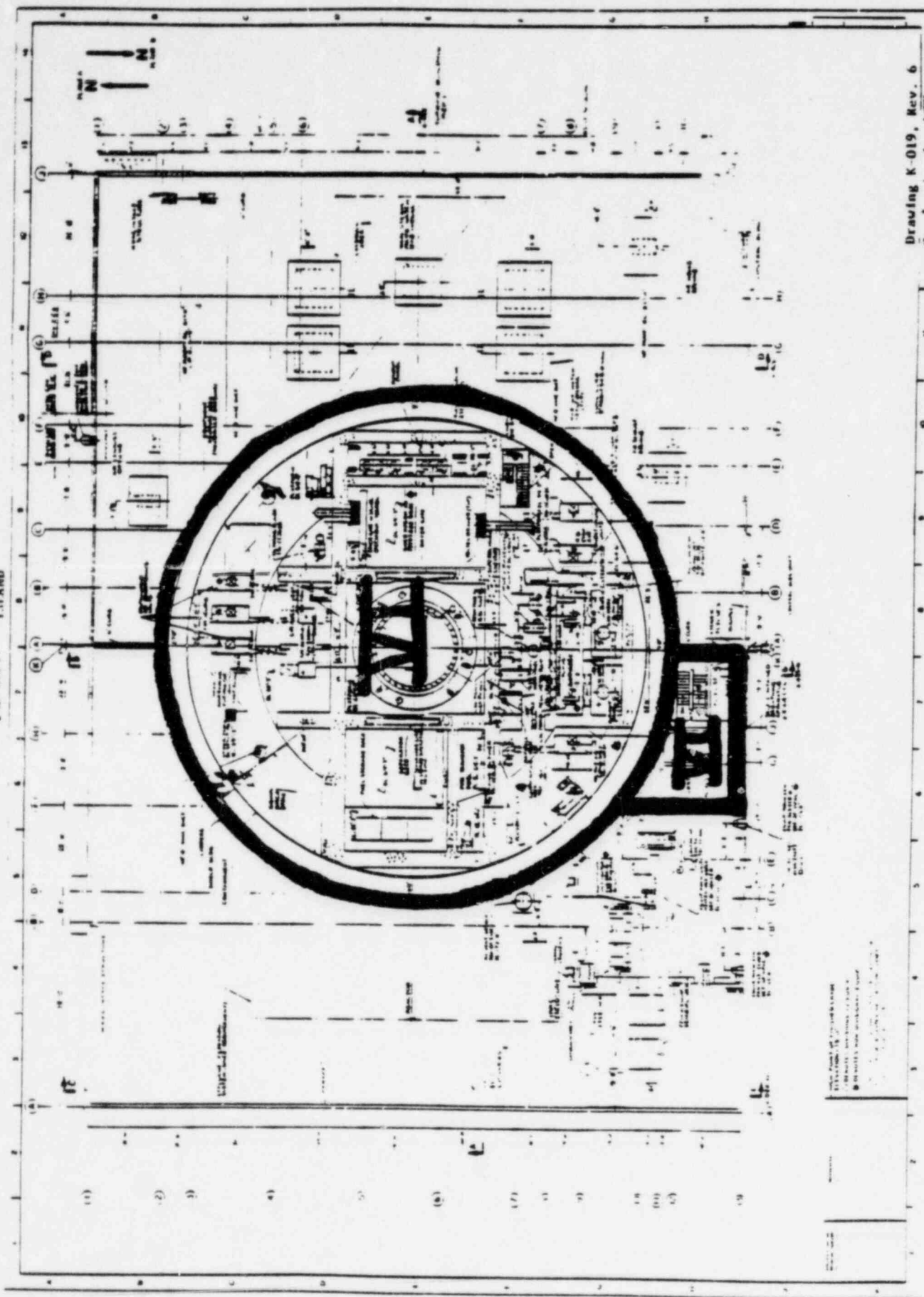
Drawing K-018, Rev. 6

Reactor, Auxiliary
& Fuel Building

Plan at El 28'-6"

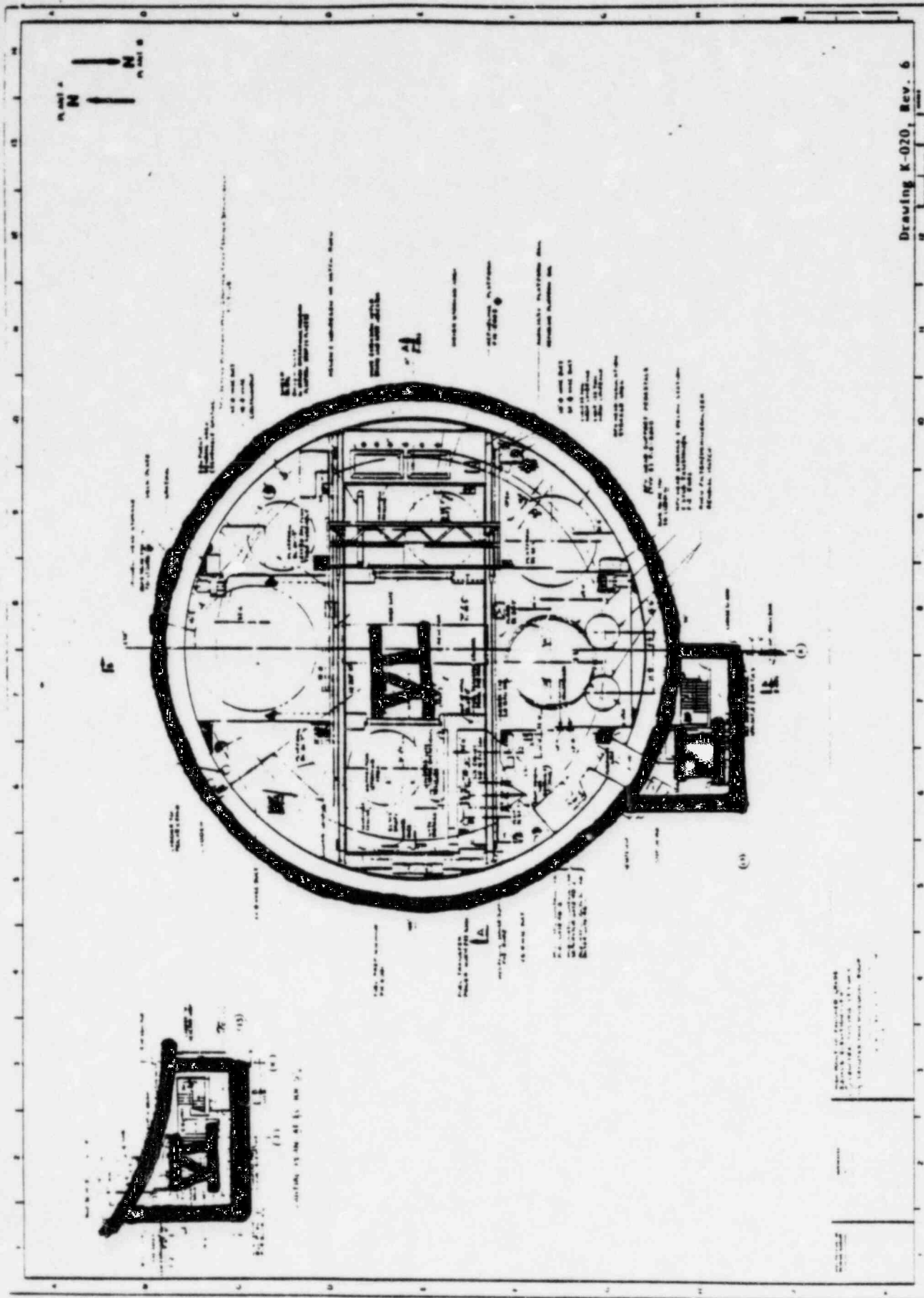
1.2-12

FIGURE 12.3-2.3



Drawing K-019, Rev. 6
Reactor, Auxiliary
& Fuel Building
Plan at Y1 50'-0"
1.2-51

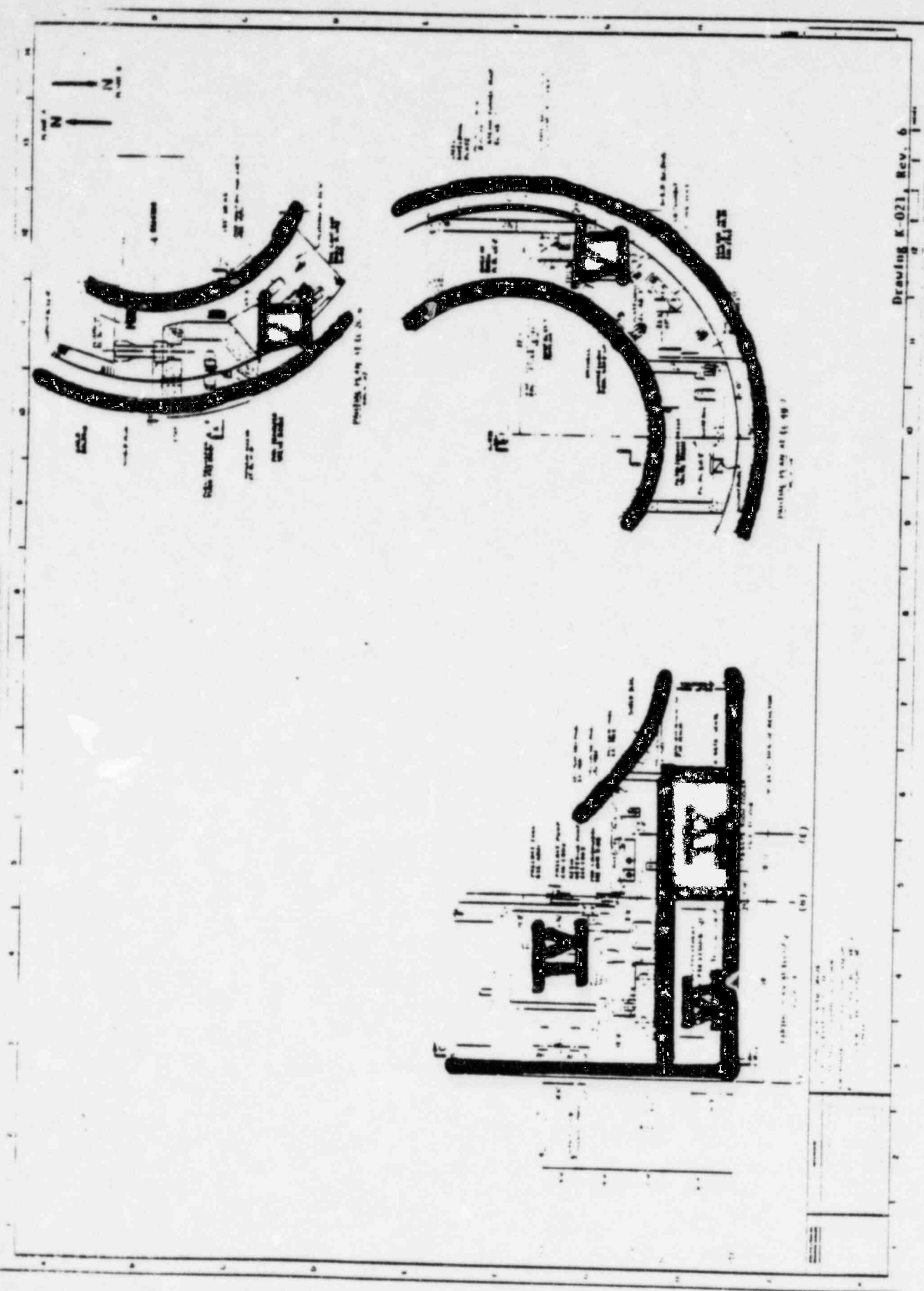
FIGURE 12.3-24



Reactor, Auxiliary
4 Fuel Building
Plan at E1 864'-7"

1.2-54

FIGURE 12. 3-25



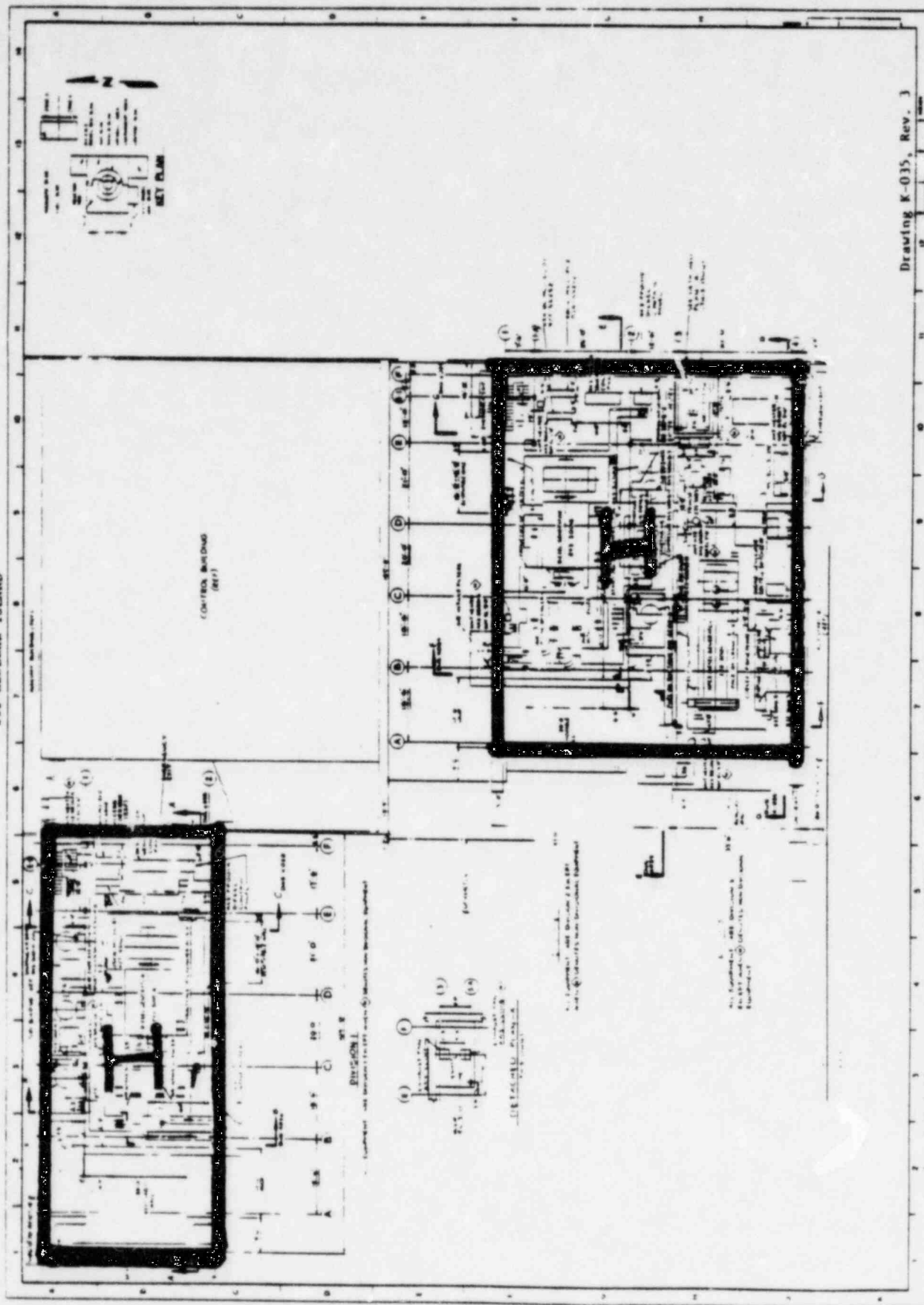
Drawing K-021, Rev. 6

Reactor, Auxiliary
& Fuel Building
Bldg Arrgt Partial Plans

FIGURE 12.3-26

22A/007
REV. 0

CESSAR II
232 NUCLEAR ISLAND



1. Diesel Generator
Bldgs Div 1,2&3
Plan at El (-)6'-10"
1.2-65

FIGURE 12.3-27

CESSAR II
238 NICHOLAS ISLAND

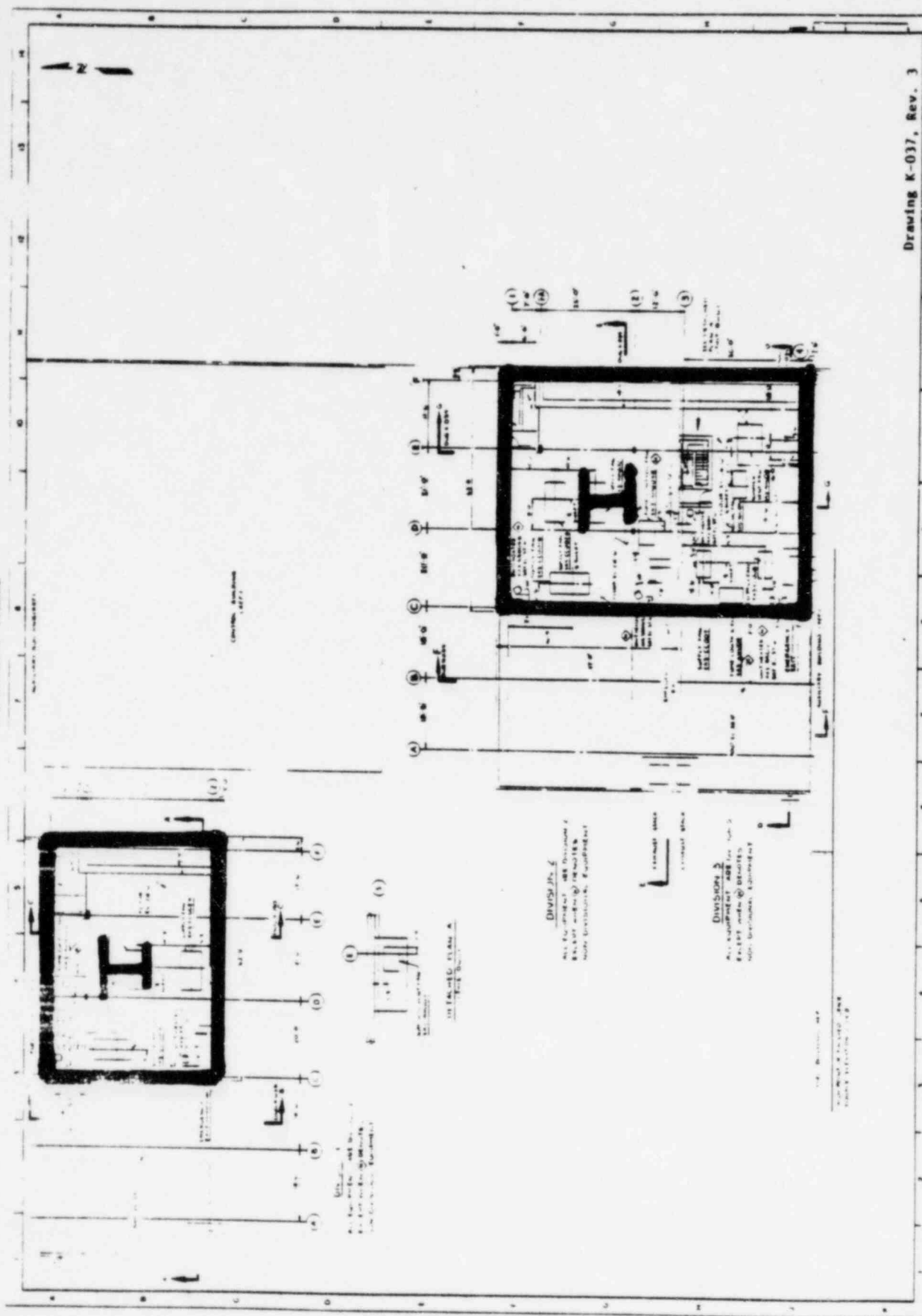


FIGURE 12.3-28 1.2-67

FIGURE 12.3-29

AUXILIARY BUILDING - ECCS ROOMS AND 432'-0" CORRIDOR

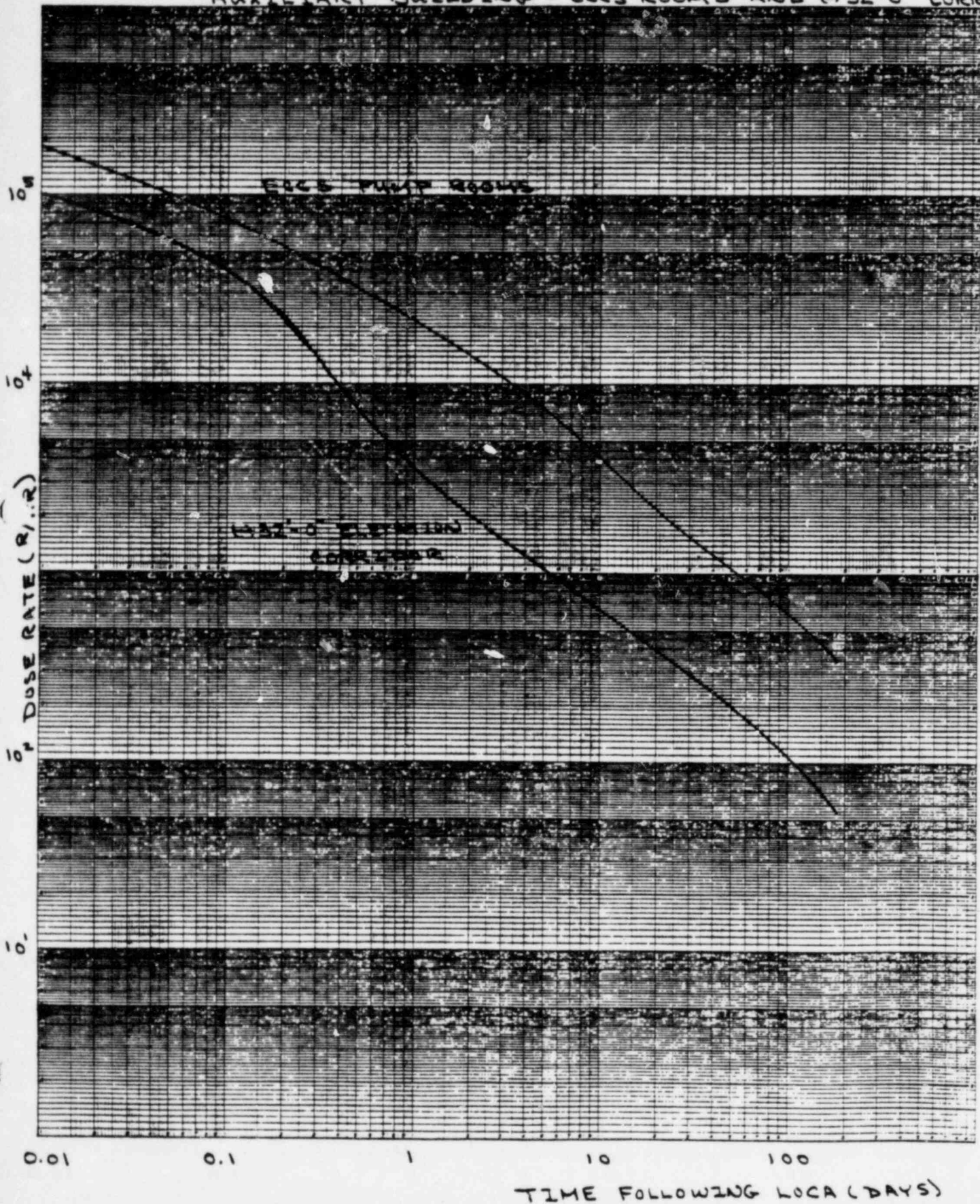


FIGURE 12.3-30

AUXILIARY BUILDING - ABOVE +6'-10" EL. OUTSIDE SECONDARY CONT.

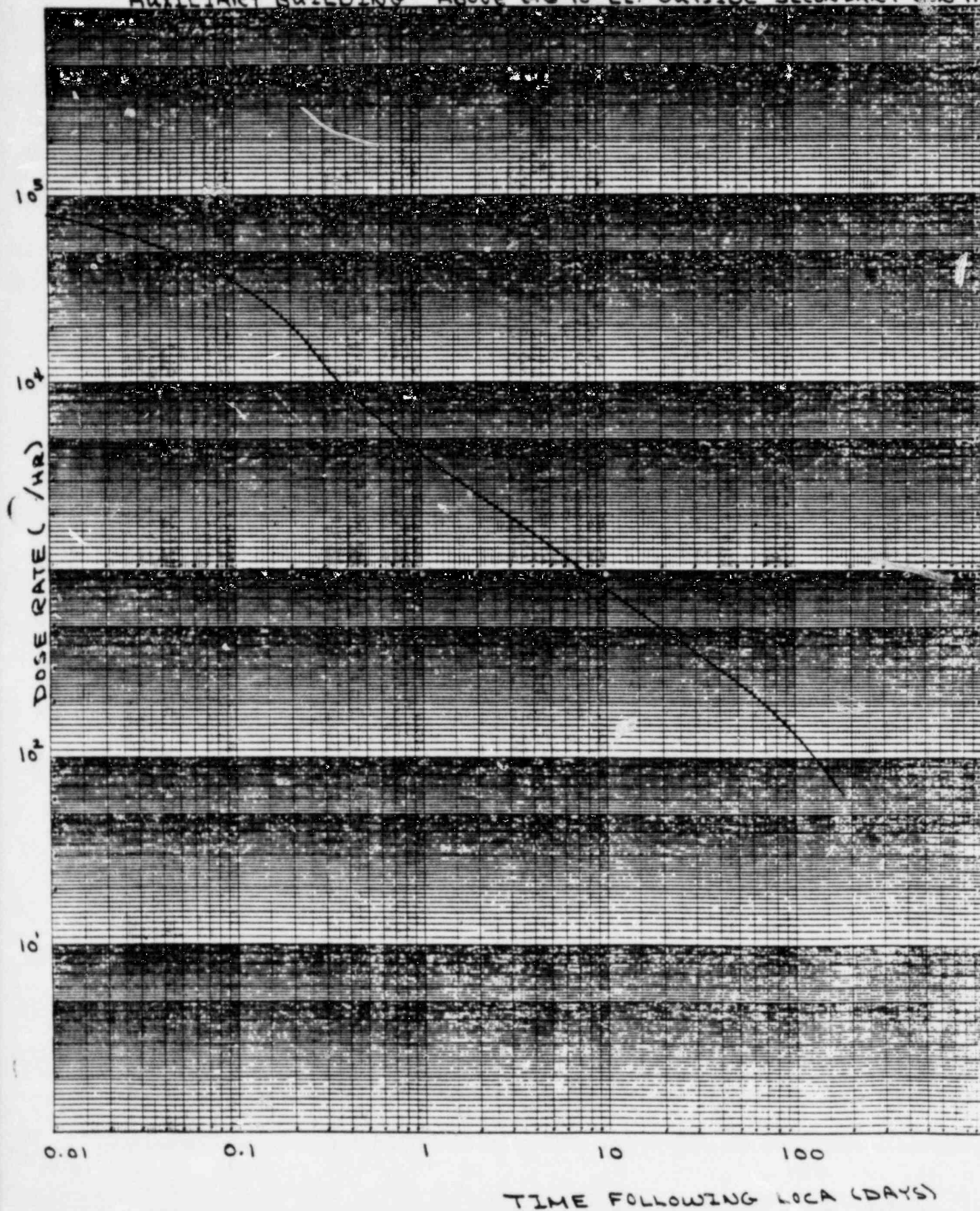


FIGURE 12.3-31

FUEL BUILDING - SGTS FILTER CUBICLE

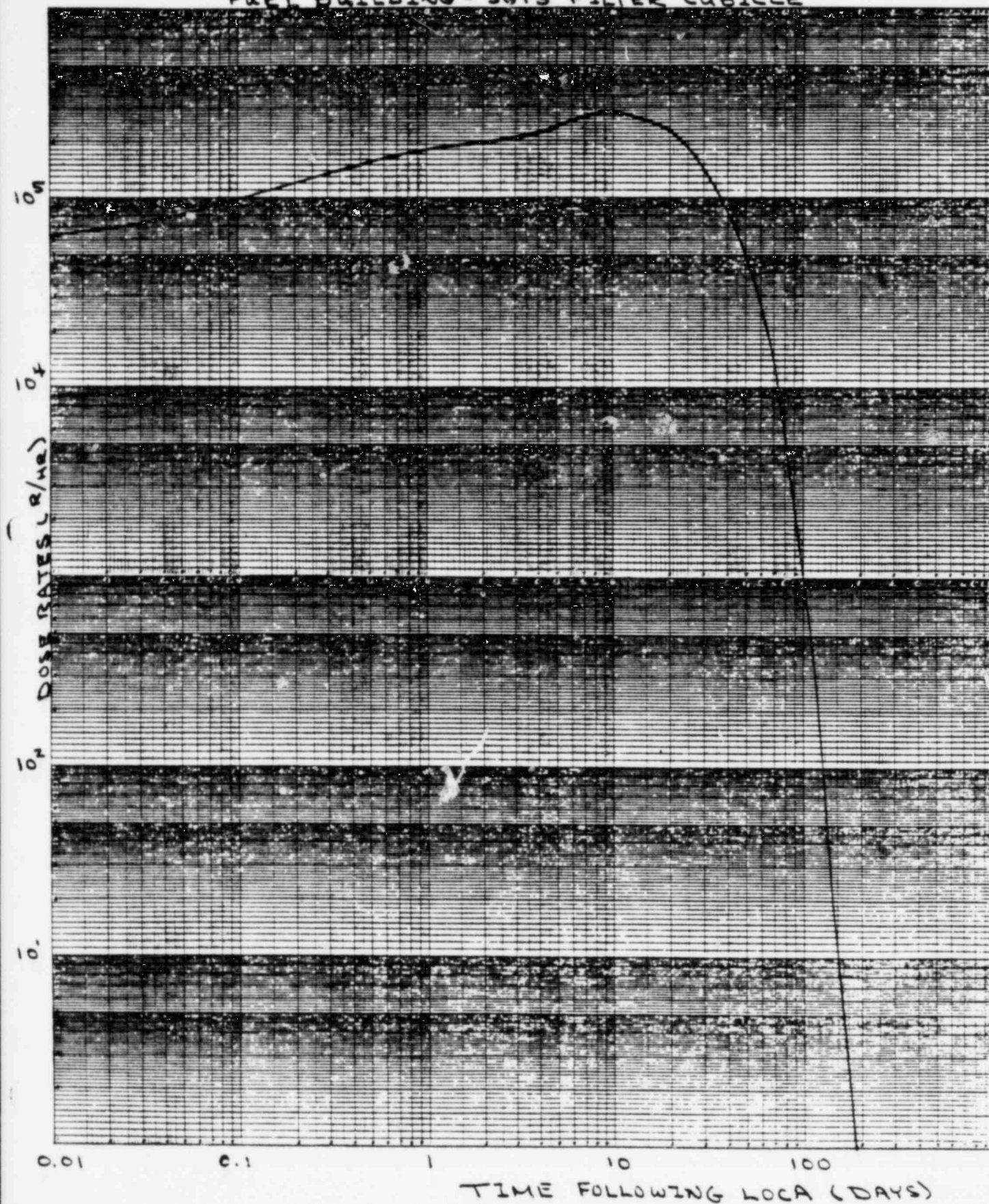


FIGURE 12.3-32

FUEL BUILDING-OUTSIDE SGTS FILTER CABICLE

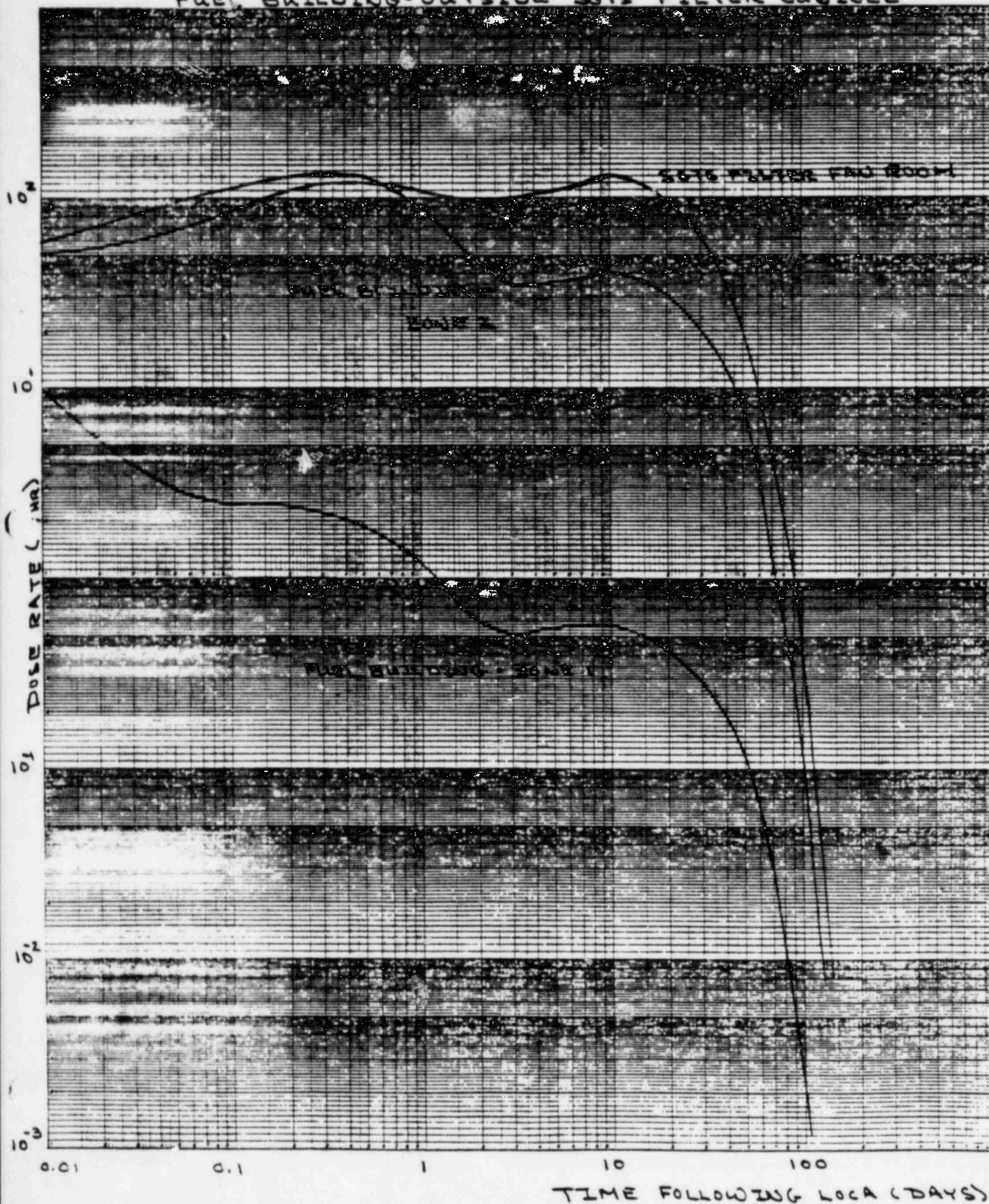


FIGURE 12.3-33

GAMMA DOSE RATES

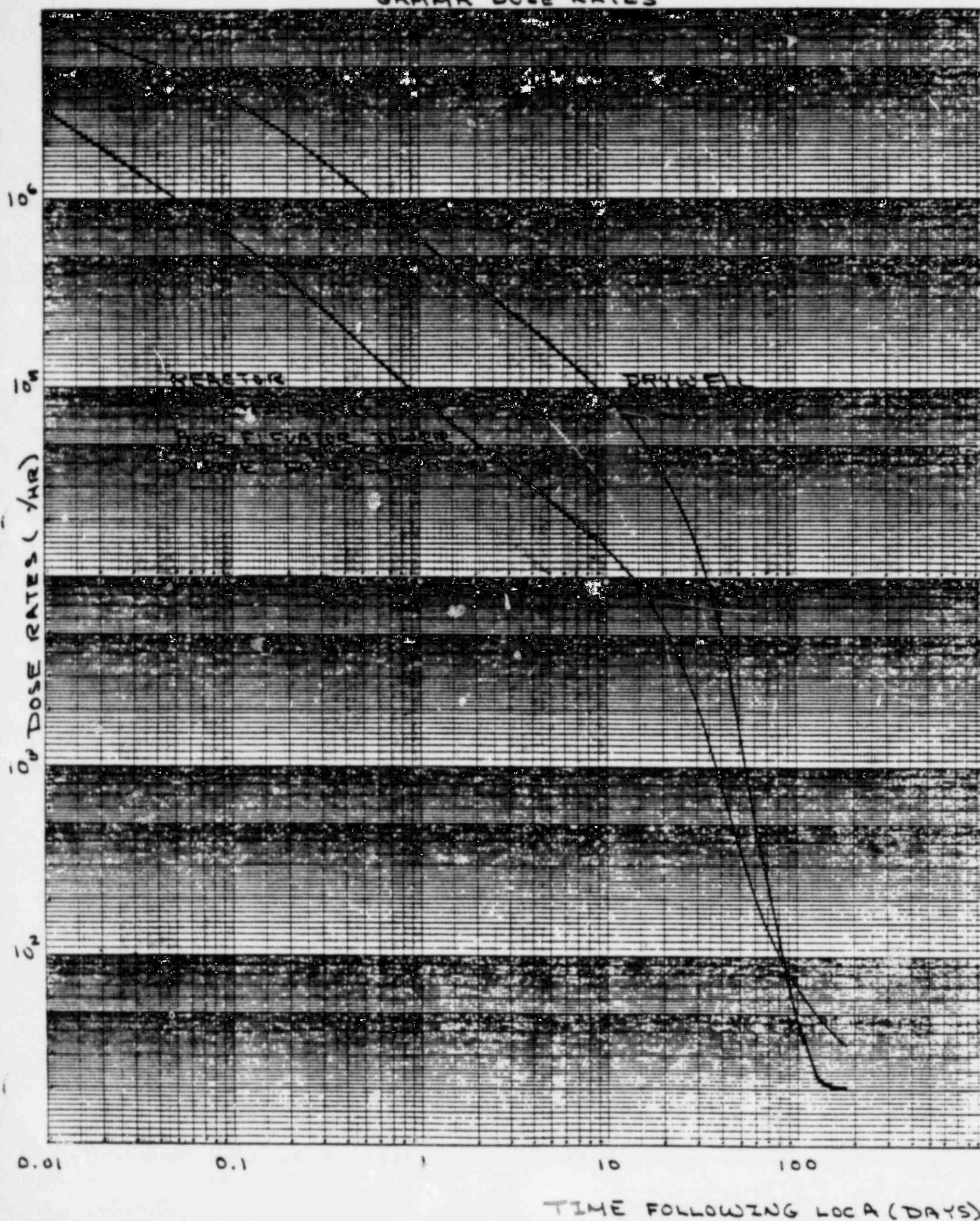
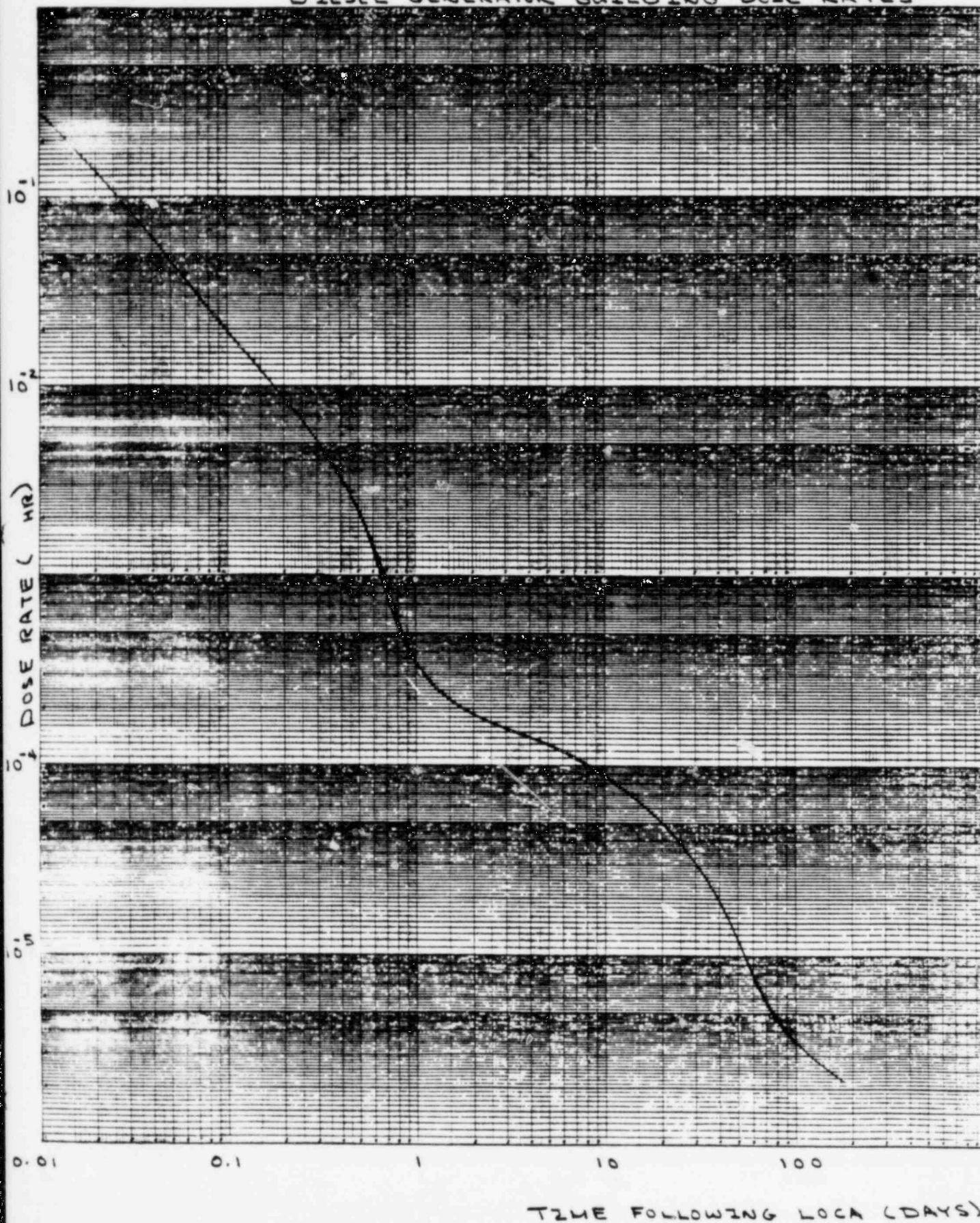


FIGURE 12.3-34

DIESEL GENERATOR BUILDING DOSE RATES



12.3.⁷₈ References

1. N. M. Schaeffer, "Reactor Shielding for Nuclear Engineers", TID-25951, U. S. Atomic Energy Commission (1973).
2. J. H. Hubbell, "Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 KeV to 100 GeV", NSRDS-NBS29, U. S. Department of Commerce, August 1969.
3. "Radiological Health Handbook", U.S. Department of Health, Education, and Welfare, Revised edition, January 1970.
4. "Reactor Handbook", Volume III, Part B, E. P. Blizzard, U. S. Atomic Energy Commission (1962).
5. Lederer, Hollander and Perlman, "Table of Isotopes", Sixth Edition, (1968).
6. M. A. Capo, "Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source", APEX-510, November 1958.
7. "Reactor Physics Constants", Second Edition, ANL-5800, U.S. Atomic Energy Commission, July 1963.
8. ENDF/B-III and ENDF/B-IV Cross Section Libraries, Brookhaven National Laboratory.
9. PDS-31 Cross Section Library, Oak Ridge National Laboratory.
10. DLC-7, ENDF/B Photo Interaction Library.

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

Response

This information will be provided by the applicant.

471.19 /

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

Response

Sections 1AA.2 and 1AA.3.3 have been revised to include the post-accident sampling station and the sample analysis area as indicated in the attached mark-ups.

1AA.2 SUMMARY OF SHIELDING DESIGN REVIEW (Continued)

rooms and pumps and valves per Table 1AA-1. All vital equipment will be environmentally qualified. It is also shown that this exposure envelope is not time dependent after about 100 days.

- c) It is not necessary for operating personnel to have access to any place other than the Control Room and, the post-accident sampling station, the sample analysis area, and three manual valves in the Auxiliary and Fuel Buildings to operate the equipment of interest during the 100 day period. The manual valves are for essential service water supply (one in each division) to the hydrogen mixing blowers of the Combustible Gas Control system and a Drywell Bleed-off Vent System valve. These valves are considered accessible on a controlled exposure basis. Direct shine from the containment is less than one R/hr within four hours post-accident.
- d) The control room is designed to be accessible post-accident.
- e) Access to radwaste is not required, but the Radwaste Building is accessible since primary containment sump discharges are isolated and secondary containment sump pump power is shed at the onset of the accident. Thus, fission products are not transported to radwaste. The hydrogen control system is operated from the Control Room; the 238 Nuclear Island does not have a containment isolation reset control area or a manual ECCS alignment area. These functions are provided in the Control Room.

* Information related to the post-accident sampling station and the sample analysis area will be provided by the applicant and is not included in the detailed discussion which follows.

1AA.3.3 Post Accident Operation (Continued)

For purposes of this review the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shutdown and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the Control Room, except for the ^{post accident sampling station, the sample analysis area, and} several manual valves noted earlier.

471.20

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

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

Response

High temperature Filters have not been implemented in the BWR design. While some test data indicates that magnetic Filters may be of benefit in maintaining Feedwater quality in plants with pumped-Forward heater drains, there is presently no demonstrated benefit with respect to control of radiation buildup.

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

Response

Stainless steel is specified for heat exchange surfaces where it is necessary to minimize corrosion. Carbon steel (which has a very low cobalt content) is used for piping and equipment in the balance of the system.

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

Response

In general, the primary system configuration is dictated by requirements other than reduction of corrosion product inputs such as functional performance, safety, reliability, etc. However, satisfaction of engineering design objectives such as minimum pressure drops, low cost, and ~~the~~ efficient utilization of space tend to result in satisfaction of the objective of minimizing surface areas in contact with the coolant as well.

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

Response

Seamless pipe has been used where practical in order to eliminate some of the longitudinal welds which require inservice inspection and thereby reduce exposures of personnel performing inservice inspections.

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

Response

The present reactor water cleanup system is adequate to meet existing water quality requirements. Within the limitations of present uncertainties regarding the mechanisms of crud buildup, this system is considered to be near optimum with respect to balancing cleanup effectiveness against loss of plant thermal efficiency and system cost. It has not been demonstrated that increased cleanup system capacity will be effective in reducing crud-related radiation levels.

f. Provisions for injecting oxygen into the feedwater line.

Response

Provisions for injection of oxygen into the Feedwater have not been implemented in the design. It has not been demonstrated that implementation of this feature will be effective in reducing crud buildup. Two recent tests of oxygen injection at operating plants provided no evidence of a beneficial effect on radiation levels. The incentive, if any, for oxygen injection is lower in a plant with Forward-pumped heater drains as a result of the fact that the feedwater oxygen content in such plants is expected to be moderately high.

ATTACHMENT NO. 3

DRAFT RESPONSES TO
EFFLUENT TREATMENT SYSTEMS BRANCH
QUESTIONS

460.09

460.09
(1.3)
(1.2)
(1.3)
(1.4)

Provide a table in Section 1.3 of your FSAR comparing the design features of the liquid, gaseous and solid radwaste systems with each position of Regulatory Guide 1.143, Revision 1 (October 1979). Justify each position for which an exception is taken. If information is provided in other sections of the FSAR for the individual items, cross-references to these sections is acceptable. We consider compliance with Section C.5 of Regulatory Guide 1.143 to be essential. Verify whether you satisfy our acceptance criteria for concentrations of radioactive constituents in accordance with Item II of section 15.7.3 of the Standard Review Plan (SRP). Our position is that limiting doses to 0.5 rems, as stated in Section 11.3.2.20 of your FSAR, is not an acceptable alternative.

RESPONSE: 460.09

Regulatory Guide 1.143 Revision 1 (Oct 1979) furnishes seismic design guidance acceptable to the NRC staff. GE has implemented⁽¹⁾ all the positions of this guide with the exception of C.2.1.3.

2.1.3 Those portions of the gaseous radwaste treatment system that are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems, should be designed to the seismic design criteria given in regulatory position 5 of this guide. For the systems that normally operate at pressures above 1.5 atmospheres (absolute), these criteria should apply to isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system (e.g., waste gas storage tanks in the PWR) and to the building housing this equipment. For systems that operate near ambient pressure and retain gases on charcoal adsorbers, these criteria should apply to the tank support elements (e.g., charcoal delay tanks in a BWR) and the building housing the tanks.

Support of the off gas delay tanks are analyzed to show no failures with the conditions of:

- a) no natural frequencies between 2 and ³³ Hz.
- b) a stress in the supports, based on a horizontal static equivalent force equal to the OBE less than 1.33 times the allowable stress level of AISC "Manual of Steel Construction" 7th edition 1970 and
- c) the tanks are located on the basement of the building housing them.

These exceptions are justified because pressure boundary failure is very unlikely as supported by the following facts:

1. The system design pressure is at least 350 psi while the system operating pressure is 6.7 psig.
2. The material of construction is required to demonstrate high notched stress ductility.
3. All pressure retaining butt welds are 100% radiographed.
4. The system must pass a 10^{-5} atm cc/sec helium leak test.
5. No single failure of an active mechanical component can allow bypass of the primary charcoal absorber.

In addition these criteria have been accepted in NUREG 0124 and the acceptance criteria

of SRP 11.3 as defined in ETSB 11.5 is met because no single action mechanical component will allow "by pass of the main decay portion" (ie "the charcoal delay beds") of the offgas system.

In addition

Even assuming two active component failures (ie. both by pass valves) and the failure of the H₂H₂H₂ radiation isolation system, and either (7+1) brings the expected offgas release rate of 25,000 $\mu\text{Ci/sec}$ [30 min] or 100 $\mu\text{Ci/sec}$ x Megawatt Thermal, as specified in ETSB 11.5 and the physically unrealizable infinite cloud model of R.G. 1.3 and the $\frac{\bar{X}}{Q}$ specified for accidents (not that for normal events as defined in the SRP) the site dose is less than 375 mRem.

Detail compliance with the acceptance criteria of Section 15.7.3 of the SRP is applicants responsibility because geology and hydrology considerations are site dependent.

CUSTOMER	PAGES	PAGE
APPARATUS	JOB	
DATE	BY	ITEM
<p><u>QUESTION 460.10</u></p> <p>Add sections for effluent radiation monitors and engineered safety feature (ESF) filters in Table 3.2-1 of your FSAR. Also add to this table, under appropriate sections, the recombiners in the off-gas system and the process radiation monitors themselves.</p> <p><u>RESPONSE 460.10</u></p> <p>Effluent radiation monitoring is included in Group X, Process Radiation Monitor System, of Table 3.2-1. ESF filters are in Group XXXI, Standby Gas Treatment System, of Table 3.2-1. The recombiners in the off-gas system are included with the pressure vessels in Group XXX, Offgas System, of Table 3.2-1. The process radiation monitors themselves are included with the electrical modules in Group X, Process Radiation Monitor System, of Table 3.2-1.</p> <p>ok with 4.6.6 to SCFS</p>		

Table 3.2-1
EQUIPMENT CLASSIFICATION (Continued)

Principal Component ^a	Safety Class ^b	Location ^c	Quality Group Classification ^d	Quality Assurance Requirement ^e	Seismic Category ^f	Comments
X Process Radiation Monitor System (includes gaseous and liquid effluent monitoring)						
1. Electrical modules main steamline and reactor building ventilation monitors <i>with safety functions (includes monitors)</i>	2	A, C, T, X (R ₁)	N/A	B	I	
2. Cable main steamline and containment ventilation safety monitors <i>with safety functions</i>	2	A, C, T, X (R ₁)	N/A	B	I	
XI RHR System						
1. Heat exchangers - primary side	2	A	B	B	I	
2. Heat exchangers - secondary side	3	A	C	B	I	
3. Piping within outermost isolation valves	1,2	C	A/B	B	I	(g)
4. Piping beyond outermost isolation valves	2	A	B	B	I	(g)
5. Pumps	2	A	B	B	I	
6. Pump motors	2	A	N/A	B	I	
7. Valves - isolation, LPCI line	1	D, A	A/B	B	I	(g)
8. Valves - isolation, other	2	D, A	B	B	I	(g)

3.2-19

CESSAR II
238 NUCLEAR ISLAND

22A7007
Rev. 4 JP

460.11
(6.5.1)

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

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

Response

The instrumentation for measuring flow rates through the ESF filter systems for the SGTS is furnished by the applicant as part of the Gaseous Effluent Monitoring System (GEMS).

The instrumentation for measuring flow rates through the ESF filter systems for the Control Building Outdoor Air Cleanup (CB OAC) System is provided in accordance with Regulatory Guide 1.52, Revision 2 (March 1978) except for the record suggestion as discussed in the following part b.2).

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

Response

- 1) The type of recording device which ~~is~~ provided for recording the flow rates through the ESF filter systems for the SGTS is furnished by the applicant as part of the GEMS. No recording device ~~is~~ provided for recording the pertinent pressure drops in the control room for the SGTS. A recording device is not needed since the standby unit will automatically start if either of the operating unit's pertinent pressure drops reaches its limiting condition.
- 2) No recording device ~~is~~ provided for recording the pertinent pressure drops and flow rates in the control room for the CB OAC System. A recording device is not needed since the standby unit will automatically start if the operating unit's flow rate reaches its lower limiting condition.

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

Response

The charcoal, filters, demister, electric heating coil and centrals for the SGTS and the CBACS are replaced by removing them from the units. The housings for the SGTS and the CBACS are all welded stainless steel housings and are not removed as an intact unit or in section. This position was explained in GESSAR I, as follows:

c-2-j Both the standby gas treatment units and the emergency air makeup cleaning units are not removable as intact unit. High activity accumulating elements can decay safely in place prior to removal as safe radwaste. Removal of the charcoal will be done pneumatically into standard solid radwaste containers with minimum exposure to operating personnel.

A change in the position stated in GESSAR I will have a major effect upon the design.

... unit

RESPONSE TO QUESTION 460.11 CONT.

The CBOARS unit is a front loading type of unit. The unit is manufactured by CTI and meets the requirements of RG 1.52.

The unit is designed such that the prefilters and the HEPA filters are installed into the unit from the front of the unit.

The filters are bolted into holding frames from the front of the unit. The filter openings, in the front of the unit, are covered by access plates. The access plates are removed from the unit to remove the filters. This operation is done from the front of the unit without entering into the unit.

The ESE atmosphere cleanup system will be totally enclosed.

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

Response

~~The~~ The ESE atmosphere cleanup systems housing leakage test is specified to be performed in accordance with provisions of Section 6 of ANSI N 510-1975 and in accordance with C.2.1 of RG 1.52. ~~Because~~ The purchase specification for the ESE atmosphere cleanup systems require the manufacturer to perform the required factory

test specification for these systems will require the field test to be performed to verify the system meets the requirements of ANSI ON 500-1975.

Compliance with the test requirements is the responsibility of the applicant

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

Response

With regard to the position C.3.b of the NRC Regulatory Guide 1.52, Revision 2 (March 1978), and ANSI N 509-1976, automatic overtemperature cutoff switches are furnished for the air heaters in lieu of the manual overtemperature cutoff switches. These automatic overtemperature cutoff switches will not be accessible following a postulated LOCA. This is acceptable since they reset automatically and do not require manual action of any kind. The temperature set point should not exceed 225 °F per ANSI N 509-1976. The applicant will be required to comply.

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

4.1

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

4.2

- 1) Total Steam Flow = 15.4×10^6 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.

- 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 = $-3^\circ F$, Dewpt = $-65^\circ F$

K_{bXe} at above conditions = 2032 cc/gm,

K_{bKr} = 93 cc/gm

- A) No cryogenic offgas system

5 & 6) H_2O used for turbine gland seal has no appreciable activity

- 7) See GESSAR II Section 11.3

- b. Release data for tritium from operating BWR's does not support your conclusions regarding release via: (1) the gaseous pathway as compared to the liquid pathway; or (2) the total release. In fact, for a number of operating BWR's, tritium releases are significantly higher than your estimate. Accordingly, verify your estimates for tritium release via the gaseous and liquid pathways using actual release data.
- c. Verify and correct the N-16 concentration given in Table 11.1-4 of your FSAR. Additionally, verify and correct, as appropriate, the reactor water concentrations for Na-24, P-32, Cr-51, Mn-54 and Zr-65 since these are significantly lower than the corresponding concentrations given in NUREG-0016, Revision 1.
- d. Add Fe-55 to Table 11.1-5 of your FSAR.

Response

b. THE TRITIUM RELEASES THAT HAVE BEEN REPORTED VARY SO GREATLY THAT ANY SPECIFIC VALUE IS QUESTIONABLE. WITHIN THE ACKNOWLEDGED ERROR, TRITIUM IN WATER RELEASED = 5×10^4 Ci/YR, TRITIUM IN GAS = 5×10^4 Ci/YR.

c. WE WILL CHANGE TABLE 11.1.4 TO SHOW N-16 REACTOR WATER CONCENTRATION OF 6.0×10^{-4} μ Ci/g. FOR LIQUID & 5.0×10^{-4} μ Ci/g FOR GAS. WILL ALSO CHANGE 11.1.5 TO SHOW THE FOLLOWING:

ELEMENT	CONCENTRATION
Na-24	1.0×10^{-2} (μ Ci/g)
P-32	2.0×10^{-4} (μ Ci/g)
Cs-51	6.0×10^{-3} "
Mn-54	7.0×10^{-5} "
* (New) - Fe-65	4.0×10^{-3} "
Zn-65	2.0×10^{-4} "

* D SEE (C) ABOVE

460.13
(11.2)

Provide additional information on the following items applicable to the liquid waste management system:

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

Response: 460.13 a

*Resin rinse and cleanup phase separator decant
inputs are reported in column 8 of
Table 11.2-4.*

~~Approved by [signature]~~

- b. Your inputs for chemical laboratory waste, laboratory wash water and laundry drains are low in comparison with the corresponding values given in NUREG-0015, Revision 1, on a per reactor basis. Verify and correct, as appropriate, these inputs.

Response:

Laundry drain input is in agreement with NUREG-0015. Values reported in FSAR for laboratory wash water and chemical waste input will be changed to correspond to NUREG 0015.

See attached corrections for GESSAR

TABLE 11.2-4

& FIGURE 11.2-16

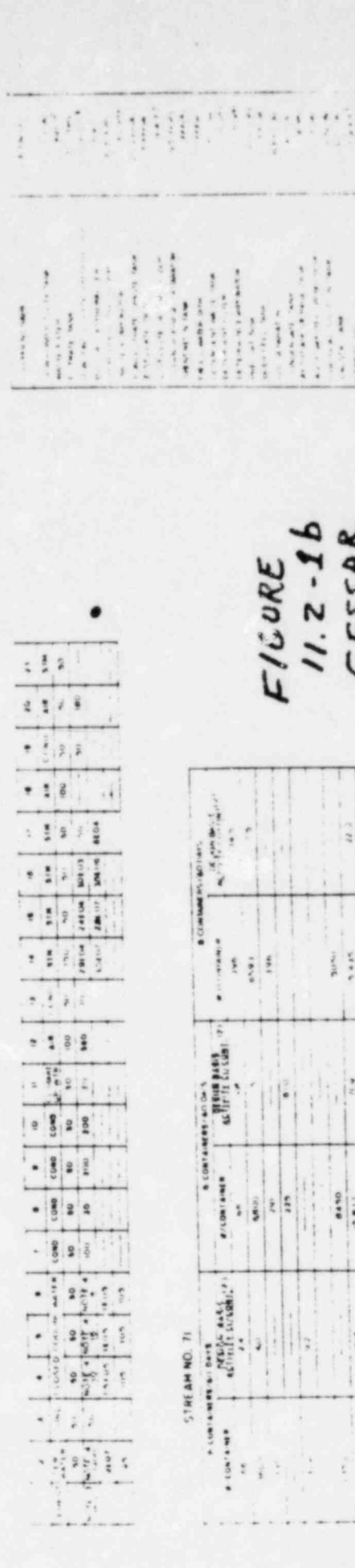
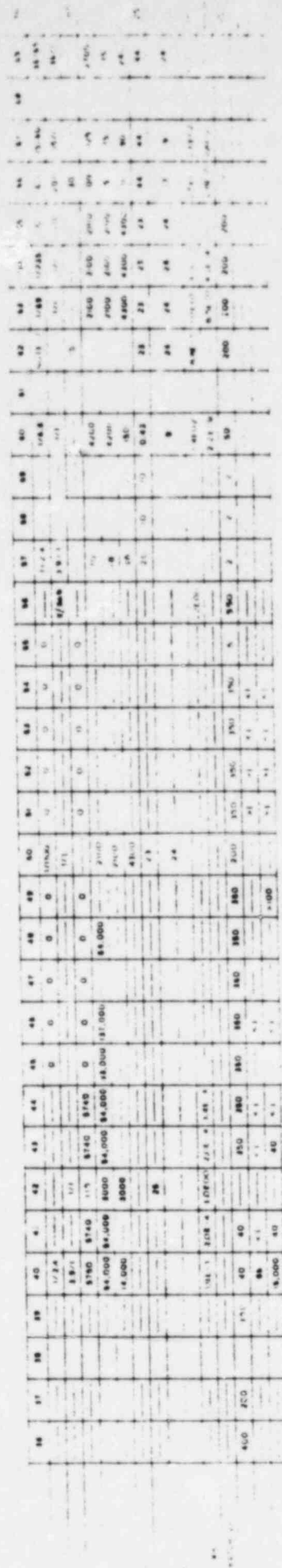
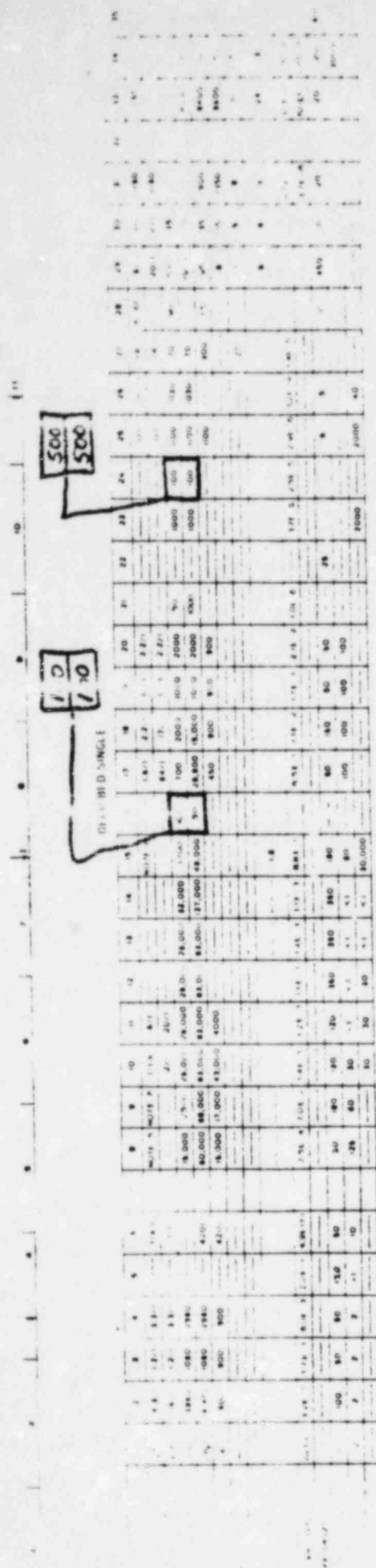


FIGURE
11.2-1b
GESSAR

C. 79E AM NO. 71

Table 11.2-4

... used for a unit of a unit plant.

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

RESPONSE: 460.13 C

Compliance to NUREG 0102 is discussed in Section I.2.43 (Page 1.8-82). The regenerant system and ultrasonic resin cleaner is the responsibility of the applicant (TABLE 1.9-1) but radwaste inputs are a safety interface (TABLE 1.9-2).

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: 460.13 d

As discussed in Section 11.2.3, the liquid wastewater 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.

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

Response:

"Waste collector" was changed to
Low Conductivity in revised paragraph
11.2.2.2.

GESSAR REVISION

11.2.2.2 High Conductivity Subsystem

This subsystem collects and processes dirty radwaste, i.e., water of relatively high conductivity and solids content. Floor drains, and condensate demineralizer regeneration solutions, are typical of wastes found in this subsystem. These wastes are collected, chemically adjusted to a basic pH as required, and concentrated in a forced-circulation concentrator with a submerged, steam-heated element to reduce the volume of water containing contaminants and to decontaminate the distillate. The distillate is demineralized to remove soluble contaminants carried over from the evaporator and then combined with the ^{Low conductivity} ~~waste collector~~ subsystem just prior to the second waste demineralizer, unless high conductivity dictates recycling. Conductivity instrumentation on the effluent of the distillate demineralizer determines the routing of this stream to either the waste demineralizer or the distillate tank for recycle.

460.13

460.13

If desired, the distillate need not be combined with the ^{Low} ~~waste~~ ^{conductivity} ~~collector~~ subsystem. This alternate route is through the excess water tanks (Subsection 11.2.2.3) to condensate storage or discharge from the plant.

6

Need correction

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

Response :

The excess water tanks are shown in Figure 11.2-2K with dedicated input sources. Discharge piping is arranged so that excess water from the low conductivity subsystem cannot be discharged to the environment (Paragraph 11.2.2.3).

460.13

9. Since your PSI 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.

460.13

h

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

Response: 460.13 h

Section 12.3.1.1 and Section 9.3.3 discuss provisions for containing radioactive spills.

460.13

i

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: 460.13 i

The concentration of radionuclides in the excess water storage tanks is reported in attached table. This table will be added to the revised version of FSAR.

Typographic corrections for I-131

concentration in the concentrated waste tank will also be made in the revised FSAR.

Table 12.2-13 (Continued)
EXCESS WATER TANKS AB/5

Source Volume = 100000 gal.
Total Curies = 0.56

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies
BR-83	3.8E-04	SR-89	1.5E-02	ZR-95	1.9E-04	NA-24	3.6E-04
BR-84	1.8E-05	SR-90	1.3E-03	ZR-97	6.0E-06	P-32	8.5E-05
BR-85	0.	SR-91	8.0E-03	NB-95	6.0E-05	CR-51	2.6E-03
I-131	9.0E-02	SR-92	2.5E-03	RU-103	9.0E-05	MN-54	2.5E-04
I-132	2.0E-02	Y-90	1.3E-03	RU-106	1.4E-05	MN-56	1.0E-03
I-133	4.1E-02	Y-91M	5.5E-03	RH-103M	9.0E-05	CO-58	2.9E-02
I-134	7.0E-04	MO-99	2.5E-04	RH-106	1.4E-05	CO-60	3.2E-03
I-135	1.6E-02	TC-99M	5.5E-03	LA-140	3.6E-02	FE-59	4.6E-04
TOTAL	1.7E-01	TC-101	3.8E-06	CE-141	1.5E-05	NI-65	6.0E-10
		TE-129M	1.5E-03	CE-143	1.5E-05	ZN-65	1.2E-05
		TE-132	1.9E-02	CE-144	1.9E-04	ZN-69M	4.9E-06
		CS-134	8.5E-04	PR-143	1.4E-04	AG-110M	3.8E-04
		CS-136	3.8E-04	ND-147	4.5E-05	W-187	9.5E-04
		CS-137	1.4E-03	TOTAL	3.7E-02	TOTAL	3.8E-02
		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				

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	.8E-03	NB-95	3.7E-03	CR-51	1.4E-02
I-131	2.6E-01	SR-92	.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
TOTAL	2.8E-01	TC-101	2.7E-07	CE-141	1.0E-03	NI-65	5.6E-07
		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	TOTAL	1.9E-01	TOTAL	2.1E-01
		CS-138	3.9E-06				
		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				

○ Delete -

12.2-34

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42A/00/
Rev. 0

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

Response: (460.13j)

Tables 11.1-2 and 11.1-3 report the source terms used in calculating the radioactive concentrations of the various supply water streams processed in the radiowaste system.

Assumptions used in calculating the estimated release for the liquid effluent is discussed in Section 11.2.3

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

The filter unit, marked future, on Figure 9.4.7 is to filter the containment exhaust if operational measurement of radioactive emission indicate that filtration is needed to meet Appendix I limits.

This exception to the GESSAR PDA requirement for the filter unit was negotiated between TVA and the NRC for the Hartsville and Phipps Bend STRIDE units. (GE to provide exact reference).

The Nuclear Island design provides space and provisions for the addition of the filter units.

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

RESPONSE b

NOBLE GAS, TRITIUM + ARGON 41 RELEASE RATES AFTER 30 MINUTES ARE BASED ON ANS 18.1, REV. 1 INPUT DECAYED FOR 30 MINUTES. THE NUMBER IS 2.6×10^4 $\mu\text{Ci/sec}$. THE 5.3×10^4 $\mu\text{Ci/sec}$ IS FROM ANS 18.1, REV. 0.

A DETAILED DESCRIPTION OF TABLE 12.2-22 IS FOUND IN PARAGRAPH 11.3.4.6. ~~HOWEVER, IF ANY RELEASE RATES ARE SHOWN ON THE DRAWING, THE TITLE IS CHANGED TO "DESIGN BASES ANNUAL RELEASE RATES OF NOBLE GASES + IODINES FOR ENVIRONMENTAL IMPACT EVALUATIONS"~~

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

Response

The following effluent pathways are currently designed with flowrate measuring devices in the sampling system or duct:

- (1) offgas pretreatment (see Fig. 7.6-10a)
- (2) offgas posttreatment (see Fig. 7.6-10b)
- (3) offgas vent pipe (see Fig. 7.6-10b)
- (4) containment ventilation discharge monitoring (Fig. 7.6-10d)
- (5) shield annulus HVAC (Figs. 9.4-9, 6.5-1)

The following pathways do not have flow measuring devices dedicated distinctly for Process Radiation monitoring but have the maximum rated flows associated with the duct shown on their respective ventilation drawing. Additional flow measuring devices for these ducts are within the applicants scope.

- (1) Fuel Building HVAC
- (2) Containment space - Refuel mode
- (3) Auxiliary Building HVAC
- (4) Control Building HVAC

The Standby Gas Treatment system and the Radwaste Building HVAC system are noted (per DWG on 7.6-11a) as having flow measuring devices to be installed by the customer.

- d. Since the off-gas system is located in the turbine building which is not within the scope of your design, state whether the design of the off-gas system lies within your scope. If not, state whether the off-gas system you have described is an interface requirement for the balance of plant.
- e. State whether the source terms you have used to evaluate off-site doses due to a postulated failure of the off-gas system are consistent with Branch Technical Position ETSP 11-5 (July 1981).
- f. State whether the seismic criteria for the proposed off-gas system will conform to Section C.5 of Regulatory Guide 1.143. In responding to this question, a cross-reference to another section of your FSAR is acceptable.

Response

d - YES

e. A FAILURE OF THE OFFGAS SYSTEM ACCORDING TO BRANCH TECHNICAL
ETSP 11-5 IS CAUSED BY A SINGLE FAILURE OF AN ACTIVE COMPONENT,
SINCE THE ~~OFF~~ OFFGAS SYSTEM IS REDUNDANT IN ALL ACTIVE COMPONENTS.
THERE IS NO REQUIREMENT TO MEET THE BRANCH TECHNICAL POSITION'S

f - SEE ANSWER TO 460.09 B

460.15

a

460.15
(11.4)

Provide additional information on the following items applicable to the solid radwaste system:

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

RESPONSE: 460.15 a

Wet solid waste is described in paragraph 11.4.2.3.1. Isotopic breakdown is shown in TABLE 11.4-3 and estimated number of containers is shown in Figure 11.2-16

460.15

b

- b. Experience with operating BWR's indicates that a deep bed condensate polishing system can generate a significantly higher volume of solidified "wet" solid wastes (i.e. about 41,000 cubic feet for a 1400 Mwe plant) than that presented in Table 11.4-2 of your FSAR. Accordingly, verify that your inputs to Table 11.4-2 of your FSAR are correct.

RESPONSE: 460.15 b

Our information for operating BWR's in the 1700 to 2300 MW_T plant size indicates good agreement with the data presented in TABLE 11.4-2.

Domestic operating plants in the power range of 3300 MW_T are Brown Ferry and Peach Bottom. Neither ~~one~~^{have} deep bed condensate polishing systems.

[Signature]

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

Added to text as shown below:

11.4.1 Design Bases

~~REDACTED~~

11.4.1.1 Power Generation Design Bases

The solid waste management system provides the capability for solidifying and packaging wastes from the reactor water cleanup system, the fuel pool cooling and cleanup system, the liquid rad-waste system, resins, and particulate wastes from the condensate cleanup system. Wastes from these systems will consist of spent resin, evaporator bottoms, diatomaceous earth, and other filtering media.

The solid waste management system also provides a means of compacting and packaging miscellaneous dry radioactive materials, such as paper, rags, contaminated clothing, gloves, and shoe coverings and for packaging contaminated metallic materials and incompressible solid objectives such as small tools and equipment parts.

The solid waste management system is designed so that failure or maintenance of any frequently used component shall not impair system or plant operation. Storage is provided ahead of process units to allow hold-up in case of delay for maintenance.

Drum capping and sample retrieval are performed locally. The operating philosophy of the solid radwaste control system is manual start and automatic stop with all functions interlocked to provide a fail-safe mode of operation.

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

Response: 460.15 d.

- (1) section 12.3.1.1 and Section 9.3.3 discuss provisions for containing radioactive spills.
- (2) Applicant will provide process parameters, operating procedures, and sampling and test procedures necessary to comply with Solidification Process Control Program. The applicant will maintain appropriate records showing conformance with the established parameters.
- (3) Heat tracing of concentrate piping and tanks is shown on Figures 11.2-2j and 11.2-2h.
- (4) Provisions for pipe and equipment flushing are described in Chapter 11 and shown on the specific P&ID for that system.
- (5) All tanks or equipment that uses compressed air for transport are vented to the vessel vent system which is routed to HEPA filters. See Figure 11.2-2M.
- (6) storage capacities for spent resin and cleanup phase separators are reported in Figure 11.2-1b.
- (7) Storage capacity for solid waste is discussed in paragraph 11.4.1.2

460.15

e

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

RESPONSE: 460.15 e

~~The equipment used to process dry solid waste is the responsibility of the applicant.~~

The response to this requirement will be supplied by the applicant. Reference: Subsection 11.4.2.3.2.

[Signature]
11/15/82

46.16
(11.5)

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

- a. Provide in tabular columns, the sampling frequency, the minimum analysis frequency and the sensitivity in Ci/cc for the following airborne effluents and process streams:
 1. Grab sampling for the principal gamma emitters and tritium for the plant vent, turbine building vent and radwaste building ventilation system effluents.
 2. Grab sampling for the principal noble gas gamma emitters for the off-gas system, the drywell purge system and the fuel building ventilation system effluents.
 3. Grab sampling for iodine in process streams for the off-gas treatment system; the drywell purge system; the auxiliary, fuel, radwaste and turbine buildings vent systems; the evaporator vent systems; and the pre-treatment liquid radwaste tank vent gas systems.
 4. Continuous sampling of the effluents for iodines, particulates and gross alpha emitters for the plant vent, turbine building vent and radwaste building vents.

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

Response

The technical specifications for radiological effluents for grab sampling frequency, minimum analysis frequency and the sensitivity should be provided by the applicant during his submittal of the waste sampling and analysis program in conformance with R.G. 1.21. The consistency of these items with NUREG -0473 Rev.2 will need to be ascertained by the applicant at that time. The tables presented in GESSAR II, i.e. 11.5.4, 5, 6 and 7, are intended as a basic guide but not as a complete substitute for an approved sampling program.

The turbine building monitoring and sampling provisions are within the applicants scope.

b. For liquid effluents and process streams:

1. Add your proposed grab sampling provisions for the service water and the detergent drain tank effluents to Table 11.5-6 of your FSAR.
2. Add your grab sampling provisions in the process liquid streams for the component cooling water system and the laboratory and sample system waste systems in Table 11.5-4 of your FSAR. Clearly indicate whether the fuel pool filter-demineralizer includes both spent fuel and refueling pools.
3. It is our position that your grab sampling and the associated analysis should identify the isotopic composition and determine the concentrations of the principal radionuclides and determine the concentration of the alpha emitters in addition to determining the gross radioactivity for all liquid effluents and process streams.
4. Explain what you mean by the waste sample tanks and the floor drain sample tank to which you refer in Table 11.5-6 of your FSAR. We find these references to be unclear since the discharge to the environment from the liquid radwaste system can only be from either the excess water tank or the detergent drain tank according to your system description.
5. Add the radionuclide Fe-55 to the isotopic analyses of effluent and process streams.

Response

1. The grab sampling provisions for the detergent drain tank are listed in table 11.5.6. The grab sampling provisions for service water will be within the applicants scope.

2. The grab sampling provisions for the component cooling water system and the laboratory and sample system waste systems will be in the applicants scope. The fuel pool filter - demineralizer has the capability to include both the spent fuel and refueling pools.

3. Grab sampling and the associated analyses for radiological concentrations are within the applicants scope.

4. The entries for waste sample tanks and the floor drain sample tank ~~entry~~ in Table 11.5-6, will be deleted.

5. Specific isotopic analyses are within the applicants scope. However, Regulatory Guide 1.21 Rev. 1 section C.10. notes that for certain radionuclides, for example Fe 55, it may be more appropriate to calculate its release concentrations based on previously calculated ratios and ones that are updated periodically to insure an accurate ratio.

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

Response

It is unknown at this time whether all effluent monitors will conform with ANSI N13.10 (1974) since the scope of supply for process radiation monitors will be determined by the applicant. The majority of GE designed Process Radiation Monitors was completed prior to the issuance of ANSI N13.10 (1974) and although it is anticipated that they could meet most, if not all, of ANSI N13.10 (1974), the parameters and units used in the design specifications may be expressed in different terms than the ANSI document.

460.17

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

WE AGREE THAT CS-137 SHOULD BE INCLUDED
IN THE ANALYSIS FOR POSTULATED TANK FAILURES. ALSO,
DUE TO THE WIDE DIVERSITY OF SITE GEOLOGICAL & HY-
DROLOGICAL PARAMETERS, RADIOLOGICAL CONSEQUENCES WILL
BE EVALUATED BY THE INDIVIDUAL APPLICANT.

:

ATTACHMENT NO. 4

DRAFT RESPONSES TO
AUXILIARY SYSTEMS BRANCH
QUESTIONS

410.07
(3.5.1)

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

Response

Response to the accumulator portion of this question is provided in revised Subsections 3.5.1.1.2.2 and 3.5.1.2.2. As described under item(s) of Subsection 3.5.1.1.2.2, bolts have only a small amount of stored energy and are of no concern as potential missiles.

3.5.1.2.4 Evaluation of Potential Gravitational Missiles Inside Containment

Gravitational missiles inside the containment have been considered as follows:

Seismic Category I systems, components, and structures are not potential gravitational missile sources.

Non-seismic items and systems inside containment are classified as follows:

a. Cable Tray

- ✓ All cable trays for both Class IE and non-class IE circuits are seismically supported whether or not a hazard potential is evident.

b. Conduit and Non-Safety Pipe

410.06

NON-CLASS IE CONDUIT IS SEISMICALLY SUPPORTED IF IT IS IDENTIFIED AS A POTENTIAL HAZARD TO SAFETY-RELATED EQUIPMENT. ALL REACTOR ISLAND NON-SAFETY CLASS PIPING IS SEISMICALLY ANALYZED WITH THE EXCEPTION OF ^{THE} RADWASTE BUILDING.

c. Equipment for Maintenance

1

All other equipment, such as hoists, that is required during maintenance will either be removed during operation, moved to a location where it is not a potential hazard to safety related equipment, or seismically restrained to prevent it from becoming a missile.

3.5.1.1.2.2 Missile Analyses (Continued)

divisional equipment makes the design acceptable. All safe shutdown functions in the Reactor Island design have redundant backups and these redundant items are separated either by considerable distance or a missile-proof barrier. Based on this, the probability of a valve bonnet missile striking both Division 1 and 2 vital targets for safe shutdown is extremely low making the resultant probability much less than 10^{-7} times per year.

- (2) Valve Stems - All the isolation valves installed in the reactor coolant systems have stems with a back seat which eliminates the possibility of ejecting valve stems even if the stem threads fail. Since a double failure of highly reliable components would be required to produce a valve stem missile, the overall probability of occurrence is less than 10^{-7} times per year. Hence valve stems can be dismissed as a source of missiles.

Moderate energy vessels less than 275 psig are not credible missile source.

410.07

- (3) Pressure Vessels - The pneumatic system air bottles are designed for 2500 psig to ASME Code Section III requirements. The bottles are not considered a credible source of missiles for the following reasons:

- (a) The bottles are fabricated from heavy-wall rolled steel.
- (b) The operating orientation is vertical with the ends facing concrete slabs. The bottles are topped with steel covers thick enough to preclude penetration by a missile.
- (c) The fill connection is protected by a permanent steel collar.

3.5.1.2.1 Rotating Equipment (Continued)

By an analysis similar to that in Subsection 3.5.1.1.1, it is concluded that no other items of rotating equipment inside the containment have the capability of potential missiles. All other pumps are incapable of achieving an overspeed condition.

3.5.1.2.2 Pressurized Components

FOR EXAMPLE, THE ADS ACCUMULATORS ARE DESIGNED FOR 200 PSIG UPSET PRESSURE (< 275 PSI) TO ASME SECTION III REQUIREMENTS AND ARE THEREFORE NOT CONSIDERED A CREDIBLE MISSILE SOURCE.

410.07

Identification of potential missiles and their consequences outside containment are specified in Subsection 3.5.1.1.2. The same conclusions may be drawn for pressurized components inside of containment. ^ One additional item is control rod drives (CRD) under the reactor vessel. The CRD mechanisms are not credible missiles. The CRD housing supports (Section 4.6) are designed to prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel. Since these housing supports are in close proximity to the drive housing and the supports have been designed specifically for the separation event, there is no reason to consider the CRD mechanisms as credible missiles.

3.5.1.2.3 Missile Barriers and Loadings

Credit is taken in some cases of rotating and pressurized components generating missiles for missile-consequence mitigation by structural walls and slabs. Penetration of the following walls and slabs by potential missiles is not considered credible:

- (1) drywell wall,
- (2) weir wall,
- (3) upper pool walls and floor,
- (4) reactor pedestal, and
- (5) other interior walls and slabs.

410.16

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

Response:

The appropriate GESSAR II sections will be revised to include the additional information provided on the Clinton docket in the course of the Clinton review pertaining to the Scram discharge system. The following summarizes this information:

- 1) Modifications will be implemented (and GESSAR II revised accordingly) to the Scram discharge system that will comply with the criteria enumerated in the Generic Safety Evaluation Report - BWR Scram Discharge System.
- 2) A discussion will be included in GESSAR II describing the effects of a drive/cooling water pressure control valve failure (closed or open)
- 3) Control Rod Drive (CRD) system specifications presently comply to the requirements of NUREG-0619 for the deletion of the CRD return line. Demonstration by test, that CRD flow to the reactor vessel is equal to or greater than the boil-off rate as discussed in NUREG-0619, is no longer a requirement.
- 4) GESSAR II Figure 4.6-5 will be updated to provide a complete P&ID of the control rod hydraulic system as modified.

Modifications to the GESSAR II text
are shown on the following pages.

TEXT CHANGES FOR 410.16

4.6.1.1.2.4.2.2 Accumulator Charging Pressure (Continued)

During normal operation, the flow control valve maintains a constant system flow rate. This flow is used for drive flow and drive cooling.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the drive pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 16 gpm (the sum of the flow rate required to insert 4 control rods) normally passes from the drive water pressure stage through eight solenoid-operated stabilizing valves (arranged in parallel) into the cooling water header. The flow through two stabilizing valves equals the drive insert flow for one drive; that of one stabilizing valve equals the drive withdrawal flow for one drive. When operating a drive(s), the required flow is diverted to the drives by closing the appropriate stabilizing valves, at the same time opening the drive directional control and exhaust solenoid valves. Thus, flow through the drive pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

INSERT
(A) →

4.6.1.1.2.4.2.4 Cooling Water Header

The cooling water header is located downstream from the drive/cooling pressure valve. The drive/cooling pressure control valve is manually adjusted from the control room to produce the required drive/cooling water pressure balance.

INSERT

(A)

pressure control valve (PCV),

If the ~~PCV~~ were to fail to a full-open position, the cooling water pressure would increase and the drive water pressure would decrease. The resulting cooling water pressure increase could cause control rods to drive inward. The existence of rod drifts would be alarmed to the control room operator for appropriate action. The resulting drop in drive water pressure would make normal control and notch movements impossible but would not affect the ability of the scram function.

Conversely, if the ~~PCV~~ PCV were to fail to a full-closed position, the cooling water pressure would decrease while the drive water pressure would increase. The reduction in cooling water pressure (and flow) would eventually lead to high CRD temperatures being alarmed in the control room. The CRD system's scram function would not be affected by the increase in drive water pressure.

In both of the failure cases described above, the manually operated bypass PCV in conjunction with isolation gate valves located upstream and downstream of the PCV would enable the operators to take corrective action.

4.6.1.1.2.4.2.4 Cooling Water Header (Continued)

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive/cooling pressure control valve will maintain the correct drive pressure and cooling water pressure, independent of reactor vessel pressure. Changes in setting of the pressure control valves are required only to adjust for changes in the cooling requirements of the drives, as the drive seal characteristics change with time. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long-term exposure to reactor temperatures. The temperature of each drive is indicated and recorded, and excessive temperatures are annunciated in the control room.

4.6.1.1.2.4.2.5 Scram Discharge Volume

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

During normal plant operation, the scram discharge volume is empty and vented to atmosphere through its open vent and drain valve. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Lights in the control room indicate the position of these valves.

During a scram, the scram discharge volume partly fills with water discharged from above the drive pistons. After scram is completed, the CRD seal leakage from the reactor continues to flow into the scram discharge volume until the discharge volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the

4.6.1.1.2.4.2.5 Scram Discharge Volume (Continued)

drive. When the initial scram signal is cleared from the reactor protection system (RPS), the scram discharge volume signal is overridden with a keylock override switch, and the scram discharge volume is drained and returned to atmospheric pressure.

Remote manual switches in the pilot valve solenoid circuits allow the discharge volume vent and drain valves to be tested without disturbing the RPS. Closing the scram discharge volume valves allows the outlet scram valve seats to be leak-tested by timing the accumulation of leakage inside the scram discharge volume.

INSERT
(B) →

Seven liquid-level switches activated by six transmitters connected to the instrument volume, monitor the volume for abnormal water level. They are set at three different levels. At the lowest level, a switch actuates to indicate that the volume is not completely empty during post-scram draining or to indicate that the volume starts to fill through leakage accumulation at other times during reactor operation. At the second level, two switches produce a rod withdrawal block to prevent further withdrawal of any control rod when leakage accumulates to half the capacity of the instrument volume. The remaining four switches are interconnected with the trip channels of the Reactor Trip System and will initiate a reactor scram should water accumulation fill the instrument volume.

4.6.1.1.2.4.3 Hydraulic Control Units

Each hydraulic control unit (HCU) furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. Operation of the electrical system that supplies scram and normal control rod positioning signals to the HCU is described in Subsection 7.7.1.2 (Rod Control and Information System).

INSERT

(B)

Redundant Scram Discharge Volume (SDV) vent and drain valves are provided, ~~as part of the SDV modifications done for the LRG II plants.~~ The redundant SDV valve configuration assures that no single active failure can result in an uncontrolled loss of reactor coolant. An additional solenoid operated pilot valve controls the redundant vent and drain valve. The vent and drain system is therefore sufficiently redundant to avoid a failure to isolate the SDV due to solenoid failure. The vent and drain valve's opening and closing sequences are controlled to minimize excessive hydro-dynamic forces.

All SDV piping is required to be continuously sloped from its high point to its low point.

A vent line is provided as part of the scram discharge system to assure proper drainage in preparation for scram reset. ~~The LRG II position is to provide~~ a dedicated vent line with a nonsubmerged discharge to the atmosphere. Furthermore, additional vent capability is provided by the vent line vacuum breakers. The vacuum breakers are required to have a differential pressure no greater than 5 inches of water.

The SDV vent and drain lines are required to be dedicated lines that discharge into the Radwaste System. Vacuum breakers on the SDV vent line and shut-off valves on the SDV vent and drain lines preclude water from siphoning back into the SDV from the Radwaste System.

The SDV and associated vent and drain piping is classified as important to safety and required to meet the ASME Section III Class II and Seismic Category I requirements.

4.6.1.1.2.5.3 Scram (Continued)

The CRD accumulators are necessary to scram the control rods within the required time. Each drive, however, has an internal ball-check valve which allows reactor pressure to be admitted under the drive piston. If the reactor is above 600 psi, this valve ensures rod insertion in the event the accumulator is not charged or the inlet scram valve fails to open. The insertion time, however, will be slower than the scram time with a properly functioning scram system.

The CRDS, with accumulators, provides the following scram performances at full power operation, in terms of average elapsed time after the opening of the RPS trip actuator (scram signal) for the drives to attain the scram strokes listed:

From Full-Over (Notch Position 48) To:

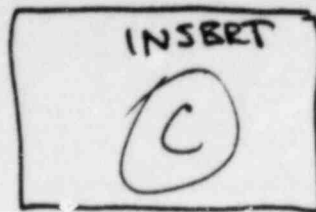
Notch Position	44	28	12
Stroke (in.)	12	60	108
Time (sec)	0.28	0.91	1.620

4.6.1.1.2.6 Instrumentation

The instrumentation for both the control rods and control rod drives is defined by that given for the rod control and information system. The objective of the rod control and information system is to provide the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled. The system allows the operator to manipulate control rods.

INSERT
C

The design bases and further discussion are covered in Chapter 7, "Instrumentation and Control System."



Diverse, and redundant level sensing instrumentation on the Scram Discharge Instrument Volume (SDIV) is provided for the automatic scram function. SDIV water level is measured by utilization of both float sensing and pressure sensing devices. Instrument taps have been relocated from the vent and drain piping to the SDIV to protect the level sensing instrumentation from the flow dynamics in the scram discharge system. Each SDIV has a redundant instrument loop. A one-out-of-two twice logic is employed for the automatic scram function. This instrumentation arrangement assures the automatic scram function on high SDIV water level in the event of a single active or passive failure. ~~These SDIV modifications will be implemented in the LRG II plants.~~

The SDIV scram level instrumentation arrangement and trip logic allows instrument adjustment or surveillance without bypassing the scram function or directly causing a scram. Each level instrument can be individually isolated without bypassing the scram function. ~~A one-out-of-two twice trip logic is employed. LRG II plants Technical Specifications will ensure that the scram function is not bypassed during repair, replacement, adjustment or surveillance of any system component.~~

Supervisory instrumentation and alarms such as accumulator trouble, scram valve air supply low pressure, and scram discharge volume not drained alarms, are adequate and permit surveillance of the scram system's readiness.

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

410.19 RESPONSE

THE GESSAR ^{II} NEW AND SPENT FUEL STORAGE RACKS
WILL BE HIGH DENSITY. SOME MODIFICATION IN THE
WRITE-UP WILL OCCUR AS A RESULT OF QUESTION
410.23 RESPONSE.

THE LETTER OF FEB 12, 1982 WAS ^{incorrect} ~~NOT CORRECT~~ WITH
RESPECT TO FUEL STORAGE.

THE GESSAR II REWRITES IS SHOWN ON THE
FOLLOWING PAGES.

①

9. AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

THE NEW FUEL AND SPENT FUEL STORAGE RACKS ARE THE SAME HIGH DENSITY DESIGN. THE NEW FUEL RACKS CAN BE USED FOR EITHER DRY OR SUBMARGE STORAGE OF FUEL. THE FOLLOWING WILL DESCRIBE THE DESIGN OF THESE RACKS, ^{BUT} SINCE THEY ARE IDENTICAL, INFORMATION FOR NEW FUEL RACKS WILL ONLY BE PRESENTED WHERE APPROPRIATE. THE DETAILED ANALYSIS OF THE RACK DESIGN IS CONTAINED IN SECTION 9.1.2 FOR SPENT FUEL STORAGE.

9.1.1 NEW FUEL STORAGE

9.1.1.1 DESIGN BASES

9.1.1.1.1

- Nuclear Design

A full array of loaded new fuel racks is designed to be subcritical, by at least 5% Δk . Neutron-absorbing material, as an integral part of the design, is employed to assure that the calculated k_{eff} , including biases and uncertainties, will not exceed 0.95 under all normal and abnormal conditions, or 0.98 under optimum moderation.

(a) Monte Carlo techniques are employed in the calculations performed to assure that k_{eff} does not exceed 0.95 under all normal and abnormal conditions.

(b) The assumption is made that the storage array is infinite in all directions. Since no credit is taken for neutron leakage, the values reported as effective neutron multiplication factors are, in reality, infinite neutron multiplication factors.

- ②
- (c) The biases between the calculated results and experimental results, as well as the uncertainty involved in the calculations, are taken into account as part of the calculational procedure to assure that the specific k_{eff} limit is met.

9.1.1.1.2 STORAGE DESIGN

- 4a) The new fuel storage racks provided in the new fuel storage vault provide storage for 30% of one full core fuel load.
- 4b) The new fuel modules are designed and arranged so that fuel assemblies and bundles can be handled efficiently during refueling operations.

9.1.1.1.3 MECHANICAL AND STRUCTURAL DESIGN (Figure 9.1-1)

- 4a) The new fuel storage racks contain storage space for 30% of one full core of fuel assemblies (with channels) or bundles (without channels). They are designed to withstand all credible static and seismic loadings.
- 4b) The racks are designed to protect the fuel assemblies and bundles from excessive physical damage which may cause the release of radioactive materials in excess of 10CFR20 and 10CFR100 requirements, under normal and abnormal conditions caused by impacting from either fuel assemblies, bundles or other equipment.
- 4c) The racks are constructed in accordance with the Quality Assurance Requirements of 10CFR50, Appendix B.
- 4d) The racks are categorized as Safety Class 2 and Seismic Category I.

SEC SECTION 9.1.1.3 FOR

ADDITION DISCUSSION OF MECHANICAL
DESIGN BASES AND ANALYSIS

9.1.1.1.4 Thermal-HYDRAULIC DESIGN

See SECTION 9.1.2.1.4 FOR ~~THE~~ DISCUSSION.

9.1.1.1.5 MATERIAL CONSIDERATIONS

See SECTION 9.1.2.1.5 FOR DISCUSSION

9.1.1.1.6 DYNAMIC ANALYSIS

See SECTION 9.1.2.1.6 FOR DISCUSSION

9.1.1.1.7 IMPACT ANALYSIS

See SECTION 9.1.2.1.7 FOR DISCUSSION

9.1.1.2 Facilities Description (NEW FUEL STORAGE)

- (1) The location of the new fuel storage facility within the complex is shown in Section 1.2.
- (2) The new fuel storage racks are top entry racks designed to maintain the new fuel while precluding the possibility of criticality under normal and abnormal conditions. The upper tieplate of the fuel element rests against the module to provide lateral support. The lower tieplate sits in the bottom of the rack, which supports the weight of the fuel.
- (3) The rack arrangement is designed to prevent accidental insertion of fuel assemblies or bundles between adjacent racks. The storage rack is designed to provide accessibility to the fuel bail for grappling purposes. Nominal fuel spacing from center to center is 6.56 inches by 6.56 inches.
- (4) The floor of the new fuel storage vault is sloped to a drain located at the low point. This drain removes any water that may be accidentally and unknowingly introduced into the vault. The drain is part of the floor drain subsystem of the liquid radwaste system.
- (5) The radiation monitoring equipment for the new fuel storage area is described in Subsection 7.1.1.6.2.

9.1.1.3 Safety Evaluation

5

9.1.1.3.1 Criticality Control

The design of the new fuel storage racks, which includes neutron-absorbing materials, provides for an effective multiplication factor (k_{eff}) for both normal and abnormal storage conditions equal to or less than 0.95. To ensure design criteria are met, the following normal and abnormal new fuel storage conditions were analyzed:

(6)

9.1.1.3.1 Criticality Control (Continued)

- (1) normal positioning in the new fuel array, and
- (2) eccentric positioning in the new fuel array (Figure 9.1-2).

The new fuel storage area will accommodate fuel ($k_{inf} \leq 1.35$ at 20°C in standard core geometry) from a multi-unit BWR facility with no safety implications.

9.1.1.3.2 Structural Design

- (1) The new fuel vault contains one 13x17 fuel storage rack, which provides storage for a maximum of 221 fuel assemblies or bundles.
- (2) The new fuel storage racks are designed to be supported above the vault floor by a support structure.

Since the racks are freestanding (i.e., no supports above the base), the support structure also provides the required stability.

- (3) The racks include individual solid tube storage compartments which provide lateral restraints over the entire length of the fuel assembly.
- (4) The weight of the fuel assembly or bundle is supported axially by the rack lower support.
- (5) The racks are fabricated from stainless steel. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications at the time of equipment order.

(7)

9.1.1.3.2 Structural Design (Continued)

- (6) The nominal center-to-center spacing for the fuel assemblies or bundles between rows is 6.56 inches. The maximum spacing between racks is 2.0 inches.
- (7) Lead-in guides at the top of the storage spaces provide guidance of the fuel during insertion.
- (8) The racks are designed to withstand, while maintaining the nuclear safety design basis, the impact force generated by the vertical free-fall drop of a fuel assembly from a height of 6 ft.
- (9) The rack is designed to withstand a pullup force of 4000 lb and a horizontal force of 1000 lb. There are no readily definable horizontal forces in excess of 1000 lb and, in the event a fuel assembly should jam, the maximum lifting force of the fuel-handling platform grapple (assumes limit switches fail) is 3000 lb.
- (10) The new fuel storage racks require no periodic special testing or inspection for nuclear safety purposes.

9.1.1.3.3 Protection Features of the New Fuel Storage Facilities

The new fuel storage vault is housed in the Fuel Building (Subsection 3.8.4). The vault and Fuel Building are Seismic Category I structures. The Fuel Building provides protection from severe natural phenomena such as tornadoes, tornado missiles, floods and high winds. Fire protection features are described in Subsection 9.5.1 and Appendix 9A.

The storage rack structure is designed to withstand the impact resulting from a falling weight. Tests using a simulated fuel bundle of the correct weight and size have been conducted to



8

9.1.1.3.3 Protection Features of the New Fuel Storage Facilities
(Continued)

verify that the rack casting can withstand the impact from a bundle dropped from a maximum allowable height above the array. Procedural fuel-handling requirements and equipment design dictate that no more than one bundle at a time can be handled over the storage racks and at a maximum height of 6 ft above the upper rack. Therefore, the racks cannot be displaced in a manner causing critical spacing as a result of impact from a falling object.

The five-ton general-purpose building crane can traverse the full length of the fuel building. A corridor is provided along the shield building side (not over) of the pools and vault; roof hatches are provided in the vicinity of the FPPCU equipment. This permits removal of major equipment by way of the hatch, thus eliminating the need to move these components along or over the pools and vault. The shipping cask cannot be lifted or moved above the new fuel vault because of inadequate clearance.

Should it become necessary to move major loads along or over the pools, administrative controls will require that the load be moved over the empty portion of the spent fuel pool and to avoid the area of the new fuel storage vault.

New fuel is carried to the new fuel vault and placed in the storage rack using the fuel-handling platform. During positioning of new fuel into the new fuel racks, the grapple is always above the upper fuel rack casting, and the grapple interfaces only with the fuel bundle bail and could not engage the fuel rack. Thus, the transfer devices used for new fuel handling to the new fuel vault cannot impose uplift loads on the rack castings.



9.1.2 SPENT FUEL STORAGE (HIGH DENSITY)

9.1.2.1 DESIGN BASES

9.1.2.1.1 NUCLEAR DESIGN

- (1) A full array in the loaded spent fuel rack is designed to be subcritical, by at least 5% Δk . Neutron-absorbing material, as an integral part of the design, is employed to assure that the calculated k_{eff} , including biases and uncertainties, will not exceed 0.95 under all normal and abnormal conditions.
 - (a) Monte Carlo techniques are employed in the calculations performed to assure that k_{eff} does not exceed 0.95 under all normal and abnormal conditions.
 - (b) The assumption is made that the storage array is infinite in all directions. Since no credit is

(10)

taken for neutron leakage, the values reported as effective neutron multiplication factors are, in reality, infinite neutron multiplication factors.

- (c) The biases between the calculated results and experimental results, as well as the uncertainty involved in the calculations, are taken into account as part of the calculational procedure to assure that the specific k_{eff} limit is met.

9.1.2.1.2 Storage ^{Design}

- ~~10/1~~ The fuel storage racks provided in the spent fuel storage pool provide storage for 326% of one full core fuel load.
- ~~10/1~~ The fuel storage racks provided in the containment pool provide storage for 68% of one full core fuel load.
- ~~10/1~~ The spent fuel racks are designed and arranged so that fuel assemblies and bundles can be handled efficiently during refueling operations.

9.1.2.1.3 MECHANICAL AND ^{Design} Structural (Figure 9.1-1)

- ~~10/1~~ The spent fuel storage racks in the Fuel Building and Containment contain storage space for 394% of one full core of fuel assemblies (with channels) or bundles (without channels). They are designed to withstand all credible static and seismic loadings.
- ~~10/1~~ The racks are designed to protect the fuel assemblies and bundles from excessive physical damage which may cause the release of radioactive materials in excess of 10CFR20 and 10CFR100 requirements, under normal and abnormal conditions caused by impacting from either fuel assemblies, bundles or other equipment.

11/1 The racks are constructed in accordance with the Quality Assurance Requirements of 10CFR50, Appendix B.

YB/4 The racks are categorized as Safety Class 2 and Seismic Category I.

11/1 The pool level is maintained by structural concrete walls with a stainless steel liner. The bottoms of the pool gates are sufficiently high to maintain the water level over the spent fuel storage racks for adequate shielding and cooling. All pool fill and drain lines enter the pool above the safe shielding water level. Redundant anti-siphon vacuum breakers are located at the high point of the pool circulation lines to preclude a pipe break from siphoning the water from the pool and jeopardizing the safe water level.

The racks include individual solid tube storage compartments, which provide lateral restraints over the entire length of the fuel assembly or bundle. (12)

The weight of the fuel assembly or bundle is supported axially by the rack fuel support.

The racks are fabricated from stainless steel. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications at the time of equipment order.

The nominal center-to-center spacing for the fuel assemblies or bundles between rows is 6.56 inches. The ~~maximum~~ spacing between racks is 2.0 inches.

NOMINAL

Lead-in guides at the top of the storage spaces provide guidance of the fuel during insertion.

The racks are designed to withstand, while maintaining the nuclear safety design basis, the impact force generated by the vertical free-fall drop of a fuel assembly from a height of 6 ft.

The rack is designed to withstand a pullup force of 4000 lb and a horizontal force of 1000 lb. There are no readily definable horizontal forces in excess of 1000 lb and in the event a fuel assembly should jam, the maximum lifting force of the fuel-handling platform grapple (assumes limit switches fail) is 3000 lb.

~~was with cage section~~
The fuel storage racks are designed to handle irradiated fuel assemblies. The expected radiation levels are well below the design levels.

Handwritten signature/initials

~~Structural and Mechanical~~ (cont'd)

In accordance with Regulatory Guide 1.29, the high density fuel storage racks are designated Seismic Category 1. Structural integrity of the rack has been demonstrated for the load combinations below using linear elastic design methods.

The applied loads to the rack are:

- (1) dead loads, which are weight of rack and fuel assemblies, and hydrostatic loads;
- (2) live loads - effect of lifting an empty rack during installation;
- (3) thermal loads - the uniform thermal expansion due to pool temperature changes;
- (4) seismic forces of OBE and SSE;
- (5) accidental drop of fuel assembly from maximum possible height (6 ft above rack); and
- (6) postulated stuck fuel assembly causing an upward force of 3000 lb.

The load combinations considered in the rack design are:

- (1) live loads;
- (2) dead loads plus OBE;
- (3) dead loads plus SSE; and
- (4) dead loads plus fuel drop

Thermal loads were not included in the above combinations because they were negligible due to the design of the rack (i.e., the rack is attached only at its base and is free to expand/contract under pool temperature changes).

The loads experienced under a stuck fuel assembly condition are less than those calculated for the seismic conditions and, therefore, have not been included as a load combination.

The storage racks are attached to the support structure by bolting, sufficient to counteract the tendency to overturn from horizontal loads and to lift from vertical loads. The analysis of the rack assumed an adequate supporting structure, and loads were generated accordingly.

Stress analyses were performed by classical methods based upon shears and moments developed by the dynamic method discussed in Subsection. Using the given loads, load conditions and analytical methods, stresses were calculated at critical sections of the rack and compared to acceptance criteria referenced in ASME Section III subsection NF. Compressive stability was calculated per AISI code¹ for light gage structures.

The loads in the three orthogonal directions were considered to be acting simultaneously and were combined using the SRSS method suggested in Regulatory Guide 1.92. The loads due to the OBE event are approximately 90% of those

W. J. N. P.

~~W. J. N. P.~~ (Cont'd)

due to an SSE event, and allowable stress levels for OBE are 50% of SSE, therefore making the OBE event the limiting load condition except for stability, where SSE acceptance criteria of 67% of critical buckling strength is required.

Under fuel drop loading condition, the acceptance criterion is that, although deformation may occur, K_{eff} must remain <0.95 . The rack is designed such that, should the drop of a fuel assembly damage the tubes and dislodge a plate of poison material, the K_{eff} is still <0.95 as required.

The effect of the gap between the fuel and the storage tube has been taken into account on a local effect basis. Dynamic response analysis shows that the fuel contacts the tube over a large portion of its length, thus preventing an overloaded condition of both fuel and tube.

Vertical impact load of the fuel onto its seat has been considered conservatively as being slowly applied without any benefit for strain rate effects.

High Density Fuel Storage Rack

The high density fuel storage rack is designed to provide sufficient natural convection coolant flow to remove 88,000 Btu/hr/bundle of decay heat.

The fuel bundle rests in the storage space with the lower tieplate extending through the hole in the support plate. The water needed to cool each fuel bundle flows by natural circulation up through the lower tieplate and the bundle. Corner slots are provided in the support plate to permit cooling of the annular gap between a channeled fuel bundle and storage tube wall.

The outer and central tube rows are supported on the base by the fittings, which have large coolant holes in all four sides. These holes, as well as the remaining rows of tubes, are open to the water plenum formed between the

base plate and the fuel supports. This plenum has four large openings in the base plate, permitting water flow from the support structure below.

The support structure must be designed to provide an adequate flow rate to prevent water reaching excessive temperatures (212°F). The flow rate is dependent on the decay heat load, the ΔP losses through the structure and the losses through the rack and bundle.

9.1-3 9-1
Figure 9-3 and Table 9-1 are supplied to the utility to allow proper sizing of the flow area required in the support structure. ~~the details on the analysis supporting the temperature increase curve.~~

In the storage pool, the bundle decay heat is removed by recirculation flow to an outside heat exchanger so that a favorable pool temperature can be.

~~7-15~~

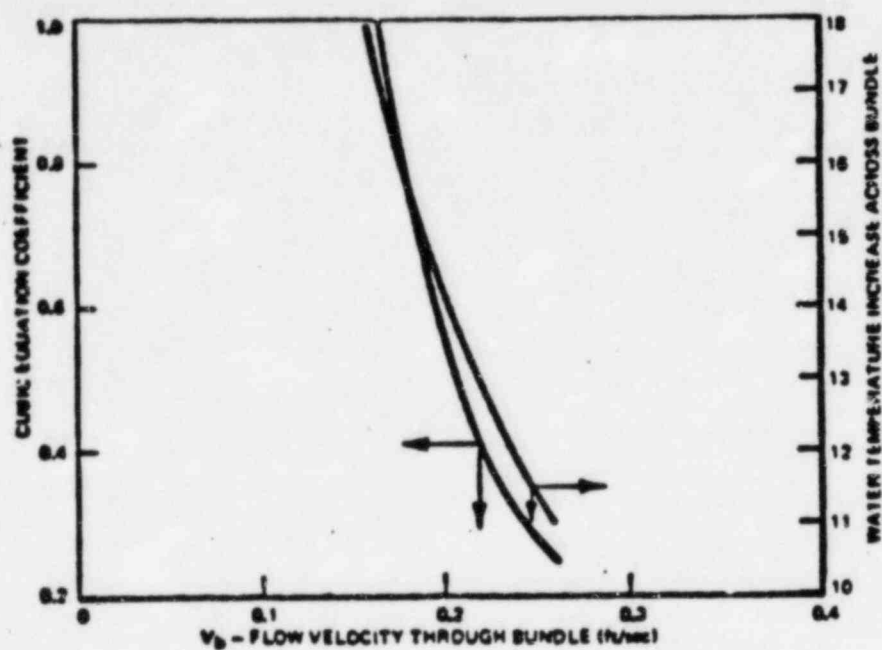


Figure 9.1-3 Cubic Equation Coefficient versus Flow Velocity Through Bundle and Water Temperature Increase Across Bundle

maintained. Although the design pool exit-temperature (to heat exchanger) is far below boiling, the coolant temperature within the rack could reach the bulk boiling temperature if the naturally induced bundle flow is not enough (due to a high flow resistance) to carry away the decay heat generated by the spent fuel. It is necessary to evaluate the rate of the naturally circulated flow to determine the maximum rack exit temperature.

The parameters which will affect the water flow through the high-density fuel storage rack and consequently the water temperature exiting the top of the storage space are:

- (1) hole size through the fitting;
- (2) flow area through the base plate;
- (3) flow resistance through the bundle;
- (4) height of the module above the pool liner;
- (5) support structure restriction to horizontal flow under module; and
- (6) loading pattern of fuel in pool (e.g., fresh fuel loaded in center of array would result in higher cooling water exit temperatures.

The analysis was performed with the bundle flow channel in place, since this is the most restrictive (all bundle cooling flow must enter through the lower tieplate orifice). Also, heat is generated in the water space between the channel and the tube by gamma capture in the water and metal, thus creating a need for additional flow openings into this space. Heat generation rates for the BWR bundle irradiated 44 GWd/Mt and cooled seven days were calculated using the ORIGEN⁴ computer code. These rates are:

Bundle	66,000 Btu/hr
Zr Channel	752 Btu/hr
H ₂ O Space	2,510 Btu/hr
Stainless Boral Tube	256 Btu/hr

~~END~~

In no case does the cooling water exit temperature at the top of the rack approach boiling. With exit water at 115°F and the pool return water temperature at 100°F, the cladding temperature will be 122°F and the Boral tube centerline temperature will be 105°F.

~~9.1.2.1~~ Factors Influencing Temperature Increase Across the Bundle

These factors are listed in previous sections. The magnitude of the effects of each of these factors is discussed below. The following relationships, each relating to one of the factors, are used to solve the cooling water temperature increase as it flows upward through the bundle. The driving force to generate flow through the bundle is given by:

$$h_c = 1 - \left(\frac{H_c + 1 + \rho_c}{2\rho_c} \right) \quad v_b = \frac{h_c M Q}{2C_p \left(\frac{A_b}{144/\rho_c} \right)^2}$$

This force is equal to the various pressure drops the water encounters in getting up through the bundle, or $v_b H_1$, where H_1 is the sum of these pressure drops given below:

Bundle Head Loss: $H_b = 7.95 \left(\frac{v_b^2}{2g} \right) \left(\frac{A_b}{A_k} \right)^2$

Holes in Castings: $H_c = \frac{v_b^2}{2g} \left[\frac{1}{c^2} \left(\frac{A_b}{A_2} \right)^2 + 2k_b \left(\frac{A_b}{A_1} \right)^2 \right]$

END

Base Plate:

$$h_{bp} = \frac{42^2 v_b}{2g} \left[K_c \left(\frac{A_b}{A_2} \right)^2 + \left(\frac{A_b}{A_1} \right)^2 K_b \right]$$

Area Under Module:

$$h_m = \sum_1^n \frac{(a+b)FLa^2}{abg} \left(\frac{A_b}{A} \right)^2 v_b^2$$

Reduction of area under module due to support structure:

$$h_{mr} = \sum_1^n \frac{a_n^2 \left(\frac{A_b}{A_1} \right)^2 v_b^2}{2g} \left[K_c + \left(1 - \frac{A_2}{A} \right)^2 \right]$$

9.1.2, 1A.2

Definition of the above terms is given in Subsection 4.4.2. The factor a_n is the ratio of flow rates to the previous quadrants and the flow rate to the quadrant in question. A quadrant in this case is one-fourth of the bundles in a module which is $169/4 = 42$ bundles for a 13×13 module. A quadrant of a module is used, since the support structures essentially divide the module into four equal areas. Assume the module quadrant in question is four quadrants from the edge of the pool array. The cooling water to this quadrant must flow horizontally under the four other module quadrants and supply cooling water to these modules. If the heat load in each quadrant is equal, then the flow to the outer quadrant is five times the flow to the quadrant in question and $a_5 = 5$. As we move closer to the quadrant in question, a_n becomes 5, 4, 3, 2 and finally, 1.

Thus, \sum_1^5 is the sum of the five pressure loss factors given above. These relationships may be summed up as a cubic equation having the following form:

$$av_b^3 + \gamma v_b^2 + \alpha v_b + \delta = 0$$

where

$$\theta = \sum_{i=1}^4 K_i$$

$\gamma = 0$, since there are no V^2 terms;

$$a = h_c \left(1 - \frac{M_t + l + p_c}{2p_c} \right); \text{ and}$$

$$b = \frac{h_c M_Q}{2C_p \left(\frac{A_b}{144} \right) p_c^2}$$

For a given geometry of fuel, inlet water temperature and heat from the bundle a and b will be constants and θ is the only coefficient that changes. Thus, under the conditions above, defining θ will set the value of V_b . The cubic was solved for a series of arbitrary values for θ and the results plotted in Figure 2-3. Referring to this plot and knowing the value for θ permits rapid determination of V_b and the temperature increase across the bundle.

A solution is presented for the case where the module is supported 8 in. above the floor and the support structures occupy 25% of the area under the module. Using the relationships above and the other factors as defined, the amount each factor contributes to θ is as follows:

<u>Factor</u>	<u>θ</u>
Bundle Head Loss	0.257
Base Plate	0.0137
Holes in Castings	0.0127
Area Under Module	0.0188
Reduction of Area under Module	0.0513
Total	0.354

$$= -1.12 \times 10^{-3}$$

$$= -4.66 \times 10^{-3}$$

and the equation to be solved is:

$$0.354V_b^3 + 1.129 \times 10^{-3}V_b - 4.66 \times 10^{-3} = 0$$

Referring to Figure 3-1 for a coefficient of 0.354, $V_b = 0.231$ ft/sec and the temperature increase of the cooling water is 12.3°F .

It can be noted that the minimum temperature increase will be determined when $\theta = 0.257$, which is the pressure drop through the bundle alone, and this increase will be 11.1°F .

The effects of changing the design parameters can be quickly determined using the above relationships. The results obtained will be conservative due to the high bundle heat loads assumed and assumptions made as to module location in the pool.

Design of the storage tube in the module, the support castings, the support plate, and the base plate is fixed. However, details of the module support structure will probably vary between facilities. For convenience, the coefficients are tabulated in Table 4-1 for various module heights above the floor and for reductions of this area due to the presence of support structures.

9-1

Table 4-3 9-1

CUBIC EQUATION COEFFICIENTS FOR VARIOUS HEIGHTS OF MODULE
 ABOVE THE FLOOR AND VARIOUS REDUCTIONS OF AREA BETWEEN
 MODULE AND FLOOR

Height Above Floor (in.)	Cubic Equation Coefficient				Area Under Module
	10%	20%	25%	30%	
14	0.0135	0.0154	-	0.0185	3.73×10^{-3}
10	0.0265	0.0301	-	0.0362	9.86×10^{-3}
8	-	-	0.0513	-	1.88×10^{-2}
6	0.0736	0.0836	-	0.100	4.37×10^{-2}

where area reduction = $\left(1 - \frac{A_2}{A}\right) \times 100$.

If it is assumed that the module is 10 in. above the floor, with a 20% area reduction in the area between module and floor and with all other parameters as given above, then the temperature increase is determined as follows:

	<u>0</u>
Area Under Module	0.07986
Area Reduction Under Module	0.0301
Base Plate	0.0137
Holes in Castings	0.0127
Bundle	0.257
Total 0	0.323

9.1-3
 From Figure 3-3, $V_b = 0.238$ ft/sec and temperature increase is 11.8°F. When the design results in values that fall between the data given in Table 4-3, an interpolation can be performed to get the correct value.

9-1

~~Table 4-3~~

9.1.2.1.4.2

~~SECRET~~
~~GE COMPANY PROPRIETARY~~
~~CLASS III~~

24

Definition of Terms

- A_3 Projected free area normal to horizontal flow beneath module, exclusive of structure, ratioed for one quadrant of module (in.^2).
- A Area normal to horizontal flow beneath the module, including structure, ratioed for one quadrant of module (in.^2).
- A_1 Flow area through base plate/42 (in.^2).
- A_2 Hole area through fitting = $\frac{\pi}{4} (D_o)^2 0.4 \text{ in.}^2$.
- A_b Flow area through bundles = 15.353 in.^2 .
- A_k Arbitrary area used in bundle friction correlation = 10 in.^2 .
- a Width of module perpendicular to direction of flow (ft).
- a_n Ratio flow at support point n to flow at module quadrant referenced; for this calculation $n = 4$ and $a_n = 1, 2, 3, 4$.
- b Height of module above pool floor (ft)
- C Orifice coefficient = 0.61
- C_p Specific heat of water = $1.0 \text{ Btu/lb-}^\circ\text{F}$.
- f Friction factor = 0.0085.
- g Gravitational constant 32.2 ft/sec^2 .
- H_b Head loss through bundle (ft H_2O).
- H_c Head loss through holes in fittings (ft H_2O)

~~9.2~~ 2 Definition of Terms (Cont'd)

- H_{bp} Head loss through area in base plate (ft H_2O).
- H_m Head loss through area under module (ft H_2O).
- H_{mr} Head loss through area under module restrictions (ft H_2O).
- h_c Effective depth of cold water over entrance point into bundle = 13.5 ft in this example.
- K_b Head loss coefficient due to bends = 0.45.
- K_c Head loss due to area contraction = 0.21.
- l Intercept in ρ versus t correlation = 63.45 lb/ft³.
- M Slope of ρ versus t correlation = -0.0145 lb/ft³-°F.
- ρ_c Density of water = 62.00 lb/ft³ (at 100°F).
- Q Heat evolution rate from bundle = $\frac{68,000}{3,600}$ Btu/sec.
- t Inlet water temperature (100°F).
- V_b Velocity of water through bundle (ft/sec).

9.12.1.5
~~Material Considerations~~ Material Considerations

in solution heat treated condition,
in accordance with the latest
issue of the applicable ASTM
specification at the time of
equipment order.

All structural material used in the fabrication of the MDFSS is ~~Type 304~~
stainless steel. This material was chosen due to its corrosion resistance
and its ability to be formed and welded with consistent quality. Boral plates,
used as a neutron absorber, are an integral nonstructural part of the basic
fuel storage tube. These plates are sandwiched between the inner and outer
wall of the storage tube and are not subject to dislocation or removal,
deliberate or inadvertent. The inner and outer walls of the storage tube are
welded together at each end, thereby isolating the Boral plates from direct
contact with spent fuel pool water. At the normal pool water operating
temperatures of 60 to 150°F, there is no significant deterioration or corro-
sion of stainless steel or Boral.

(27)

9.1.1.1.5
~~Properties~~ Material (cont'd)

Presence of the neutron absorber material in the fabricated fuel storage module will be verified by visual examination and dimensional inspection. The tube design allows visual verification that material exists in the space provided for placement of the neutron absorber material. The thickness of the Boral plates is nonstandard, providing a statistical significance between the thickness of the Boral and commercially produced aluminum or steel sheets, therefore enabling confirmation of Boral presence by dimensional inspection. In addition, use of non-Boral plates in fabrication of fuel storage tubes will cause tube deformation. Deformed tubes will not be accepted by inspection or by fabrication fixtures used for assembly of a fuel storage module. Dimensional reinspection at neutron absorber plate locations can be performed at the pool site. These data would be compared with the dimensional results obtained during fabrication of the individual tubes and of the module assembly. A visual reinspection of each Boral plate location can also be performed. Acceptance of the above inspections will ensure that Boral plates are contained in the fuel storage module sufficient to maintain the neutron multiplication factor at or less than 0.95 with a 95% confidence level.

~~next section~~ The storage tube and the integral neutron absorber material are permanently marked with identification traceable to the material certifications. The fuel storage tube assembly containing the neutron absorber material is compatible with the environment of treated water and provides a design life of 40 years, including allowances for corrosion.

Corrosion data and industrial experience confirm that aluminum and Boral have acceptable corrosion-resistant properties for the proposed application. (2.5)

See Reference

9.1.2.1.5
~~ANALYSIS~~ DYNAMIC ANALYSIS

~~9.1.1.5.1~~ Analysis Method

~~9.1.1.5.1~~ Input Excitation

The high density fuel storage rack was analyzed using the GE standard BWR/6 seismic response spectra. The spectra were derived by determining the dynamic

response of the GE standard plant over a wide range of soil conditions with a D.3g SSE excitation. This implies that the rack is designed for the worst site condition, and considerably higher seismic margin is achievable for more optimal sites.

9.1.1.6.1.2
Modeling

(1) Horizontal

Various pool arrangements were analyzed to determine the worst-case hydrodynamic mass effect. The case chosen is that of eight racks in a rectangular pool with a 14-in. spacing to the wall and 2-in. spacing between the modules. This arrangement was modeled as two lumped mass cantilever columns as shown in Figure 4-5^{9.1-4} where Nodes 1 through 12 represent the group of eight racks and Nodes 13 through 25 represent the pool wall to account for interaction effects. Node 12 represents the joint of the tube and fitting in the rack.

(2) Vertical

Because of the much higher natural frequency (<33 Hz) in the vertical direction, the vertical response force was determined statically and no special modeling is necessary.

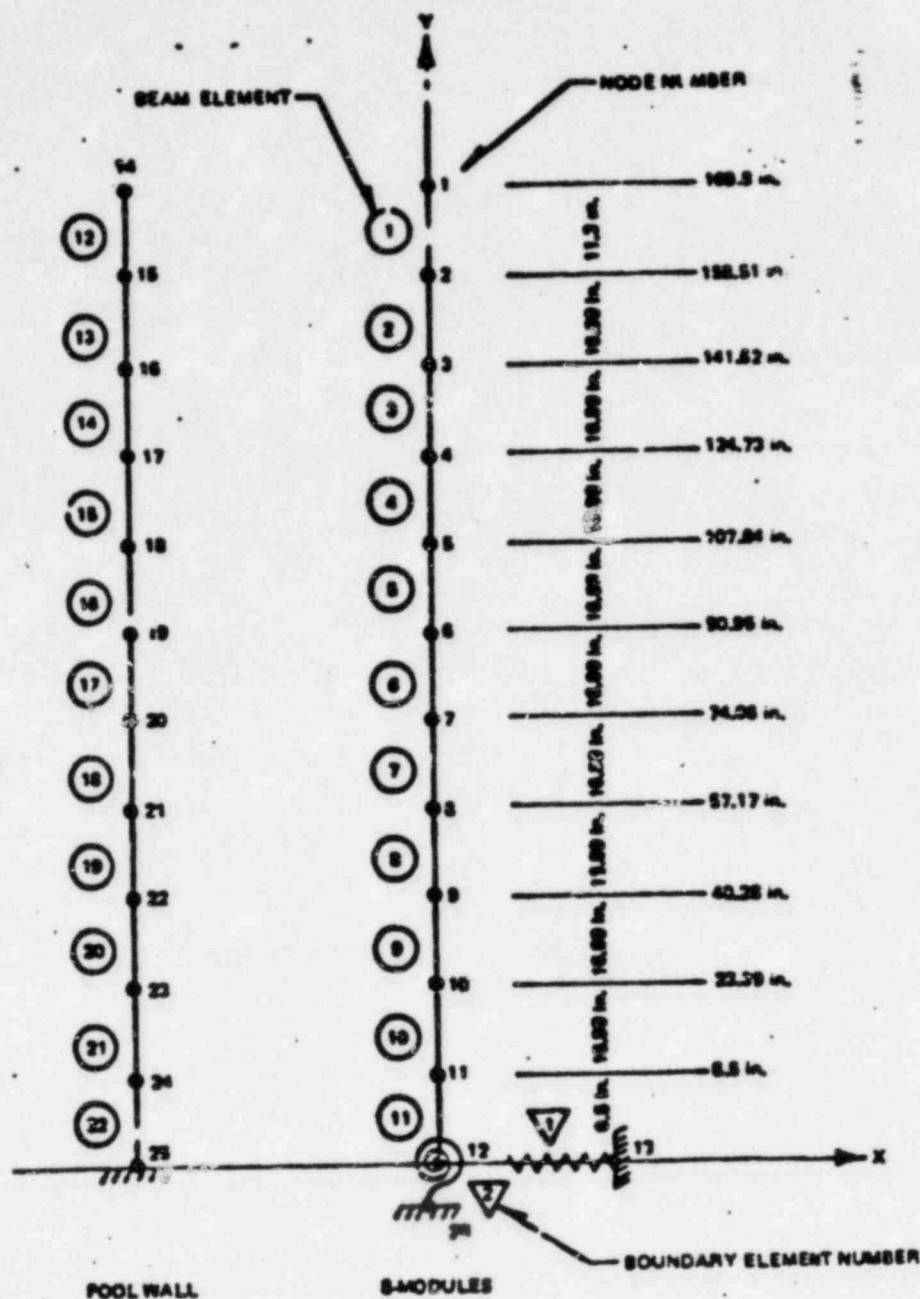
9.1.1.5.1.3
Analysis

The total mass matrix of each rack for the analysis is equal to its structural mass matrix plus the hydrodynamic mass matrix. Conservative structural damping values of Regulatory Guide 1.61 are used without any added damping due to fluid effects. The WATER-01 computer program (GE Company Proprietary) was used to determine the hydrodynamic mass of one rectangular body inside another rectangular body.

The governing dynamic equation for a water-filled rectangular container subject to ground excitation is:

$$(M_s + M_h) \ddot{x} + (c) \dot{x} + (k) x = -(M_s + M_h) \ddot{y}$$

30



9.1-4
 Figure 4-5. DYSEA Computer Model

where

M_s = a diagonal matrix representing lumped structural mass;

M_h = a nondiagonal hydrodynamic mass matrix which causes coupled motion among the racks and between the racks and the pool wall;

c and k = damping and stiffness matrices, respectively, of the system;

\ddot{x} , \dot{x} , x = the acceleration, velocity and displacement vectors of the system relative to the support motion;

\ddot{y} = the rack base support acceleration excitation.

The response spectrum analysis method in the DYSEA computer program is used to calculate the response forces due to the OBE or SSE horizontal acceleration. The horizontal response spectra used is shown in Figure 4-6 for the frequency range of 1 to 30 Hz.

9.1-5

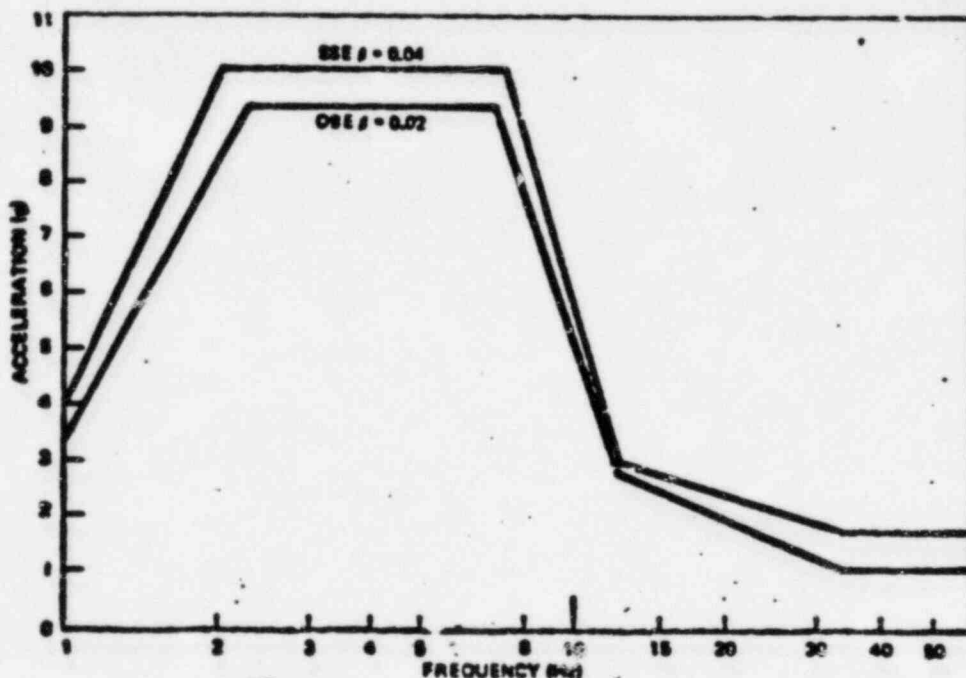


Figure 4-6 Horizontal Spectrum (GE Company Proprietary)

9.1-5

9.1.3.1.6.1.4
 Results

-Using the input and methods described above, the maximum response forces at the tube-to-fitting connection for each rack are as shown below:

Event	Damping (%)	M Moment (in.-lb)	Shear (lb)	Vertical Force (lb)
OBE	2	1.83×10^7	1.62×10^5	1.90×10^5
SSE	4	1.96×10^7	1.79×10^5	2.80×10^5

The first natural frequency of the rack with base fixed is 12.8 Hz.

9.1.1.1.7 IMPACT ANALYSIS

9.1.1.1.7.1 Vertical Impact Analysis

Vertical impact analysis is required because the fuel assembly is held in the storage rack by its own weight without any mechanical holddown devices. Therefore, when the downward acceleration of the storage rack exceeds 1.0g (g = gravitational acceleration), contact between the fuel assembly and the storage rack is lost. Impact occurs when fuel-assembly/storage-rack contact is re-established. Although it is very unlikely for most utility plants to have storage rack vertical seismic accelerations exceeding 1.0g, such large acceleration values can occur for certain unfavorable combinations of sites and building designs; thus, the need for vertical impact analysis.

4.3.1.1.1 Input Excitation

The input excitation consists of several representative acceleration time histories at the refueling pool floor. These time histories are generated from using actual site and plant structure models. The time histories have been normalized to 0.15g OBE and 0.30g SSE for those plants whose seismic requirements are less than 0.15g OBE and 0.30g SSE. For those plants whose seismic requirements are more than 0.15g OBE and 0.30g SSE, the time histories were not reduced. The time histories are then time scaled and amplitude scaled

4-10

such that a broad spectrum is obtained. The above procedure has been adopted in order to encompass all known plants where the potential for utilizing the high density storage rack exists. The enveloped spectra of the time histories are shown in Figure 4-7.

~~9.1-6~~

9.1-6

4-3.1.2 Analysis Method

Since impact analysis requires a nonlinear system model, the time history direct integration method is used to determine the system response. The integration scheme uses a central difference procedure. The integration time step size is selected small enough to ensure conveyance to the correct solution.

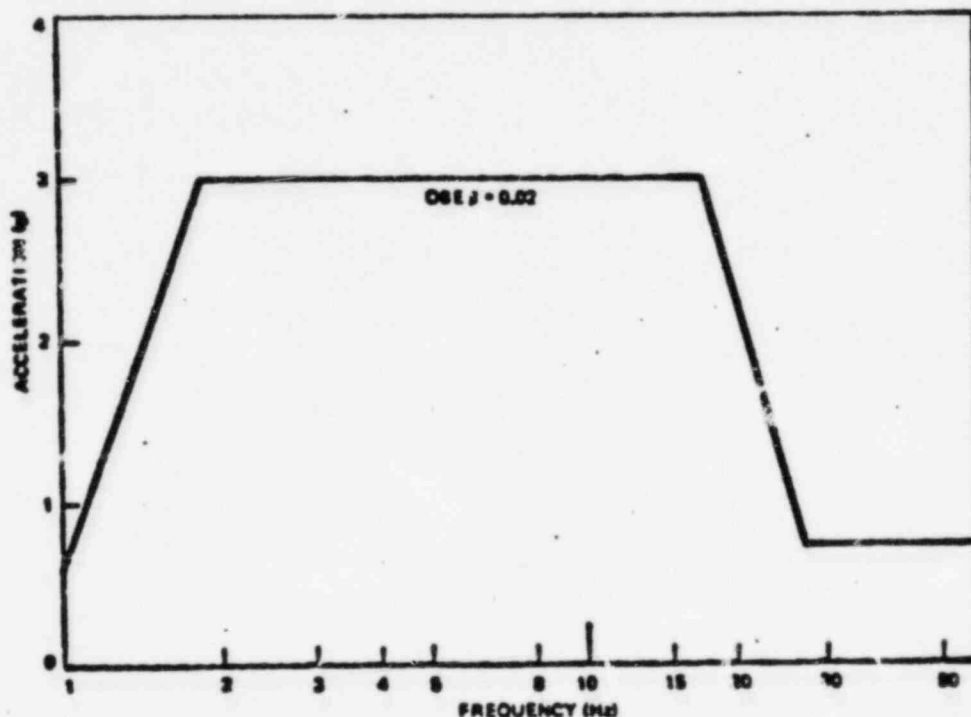


Figure 4-7 Vertical Spectrum (GE Company Proprietary)

9.1-6

~~3-1~~

3.1.3 Modeling

For the vertical impact analysis of the fuel bundle, three types of elements are used:

- (1) lumped mass gap element to represent the fuel assembly;
- (2) the variable water mass element to represent the fuel rack support plate and fuel interaction effects; and
- (3) the linear spring-damper element to represent the top flange of the I-beam in subfloor support system, which consists of a series of fabricated I-beams arranged in a rectangular array.

The webs and bottom flanges of the I-beams are calculated to be extremely stiff in the vertical direction and are represented by the same rigid base as the fuel storage pool floor.

The hydrodynamic effects result in a lowered natural frequency of the fuel assembly and are accounted for in the analysis. The percentage of critical damping is taken to 7% for SSE condition and 4% for OBE conditions.

3.1.4 Design Adequacy Evaluation

Assuming that all of the fuel bundles stored in a fully filled high density rack will vibrate in-phase with behavior similar to the response of a single fuel bundle, the dynamic design loads of the high density fuel rack with 169 fuel bundles were calculated and used for design structural adequacy evaluation.

4.1.2 Horizontal Impact Analysis

Horizontal impact analysis is required because a clearance exists between the fuel assembly and the high density fuel storage rack tube walls. It is

~~3.1.4~~

Expected that the fuel assemblies have equal probability of being in contact with any one of the four sides. Thus, the impact forces are expected to be equally likely on all four surfaces of the fuel rack and fuel assembly. The random nature of the horizontal impacts thus precludes significant gross loading of the fuel storage rack. Thus, only localized deformations of the fuel storage rack cells need to be considered.

~~4.3.2.1~~ Input Excitation

For horizontal impact analysis, the fuel assembly is modeled as a series of parallel beams representing the channel and the fuel rods supported by spacers. The local flexibility of the fuel storage rack cells is determined through a standard structural analysis using the SAP-4 computer code.

The hydrodynamic effects between the fuel assembly and the storage rack are accounted for by a fluid mass element. A lumped-mass gap element is used between the fuel assembly and the fuel storage rack. The lumped-mass gap element is also used between the fuel rods and the channel to simulate the clearance between the fuel rods and the channel. The upper and lower tieplates are assumed to be attached to the channel due to the tight fit between the tieplates and the channel. The overall bending stiffness of the fuel channel is represented by linear spring damper elements. Because of the relatively high gross bending stiffness of the fuel rack in the horizontal direction, the rack is considered fixed as far as gross bending is concerned.

~~4.3.2.2~~ Design Adequacy Evaluation

The horizontal impact response determined using the model and analysis method described above is used to determine the dynamic loads on the fuel rack cells. The local loading on the individual storage rack cells is used for design structural adequacy evaluation.

~~4.3.2.3~~
~~Design Adequacy Evaluation~~

9.1.2² Facilities Description (SPENT FUEL STORAGE)

- (1) The spent fuel storage racks provide storage in the containment and spent fuel pools for spent fuel received from the reactor vessel during the refueling operation. The spent fuel storage racks are top entry racks designed to maintain the spent fuel while precluding the possibility of criticality under normal and abnormal conditions. The upper tieplate of the fuel element rests against the rack to provide lateral support. The lower tieplate sits

in the bottom of the rack, which supports the weight of the fuel.

- (2) The rack arrangement is designed to prevent accidental insertion of fuel assemblies or bundles between adjacent modules. The storage rack is designed to provide accessibility to the fuel bail for grappling purposes. Nominal fuel spacing from center to center is 6.56 inches by 6.56 inches.
- (3) The location of the spent fuel pool and the containment pool within the complex is shown in Section 1.2.

410.25
(9.1.3)
GE

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

410.25 RESPONSE

THE GESSAR TEXT HAS BEEN CORRECTED TO DELETE
THE REFERENCE TO ALUMINUM RACKS (SEE ATTACHED).
WE HAVE ELECTED NOT TO CHANGE OUR WATER
CHEMISTRY REQUIREMENTS.

9.1.3.2 System Description (Continued)

drained from the inclined transfer tube during downward fuel transfer, as well as the volume of water above the skimmer weirs, which drains from the pools following a temporary loss of circulation.

Clarity and purity of the pool water are maintained by a combination of filtering and ion exchange. The filter-demineralizers maintain total dissolved heavy element content (Cu, Ni, Fe, Hg, etc.) at 0.1 ppm or less with a pH range of 6.0 to 7.5 for compatibility with fuel storage racks and other equipment. Each filter unit in the filter-demineralizer subsystem has adequate capacity to maintain the desired purity level of the pools under normal operating conditions. The flow rate is designed to be approximately that required for two complete water changes per day for the fuel transfer and storage pools. The maximum system flow rate is twice that needed to maintain the specified water quality. Water may be returned to condensate storage after being filtered and demineralized. 410.25

The FPCCU System is designed to remove suspended or dissolved impurities from the following sources:

- (1) dust or other airborne particles;
- (2) surface dirt dislodged from equipment immersed in the pool;
- (3) crud and fission products emanating from the reactor during refueling;
- (4) debris from inspection or disposal operations; and
- (5) residual cleaning chemicals or flush water.

410.29
(b)(1.4)

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

Response (New)

There are no differences between the BBSAR II and Perry fuel handling system designs except for the Applicant dependent components such as the spent fuel cask and cask crane).

ATTACHMENT NO. 5

DRAFT RESPONSES TO
POWER SYSTEMS BRANCH
QUESTIONS

430.27
(8.3.2)

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

Response

See GESSAR II section 8.3.2.1.1 & Fig. 8.3-18

The number of cells in each battery bank (either Class 1E or non-Class 1E) is 60 cells for the divisions 1, 2, and 4 and the non-divisional batteries.

The operating voltage range for Division 3 (HPCS) Class 1E dc loads is 112.5V to 137.5V with 125V dc nominal voltage. The maximum equalizing charge voltage for Division 3 (HPCS) 125Vdc battery is 137.4 volts. Voltage at the end of two-hour design discharge will be provided by the applicant. Division 3 battery charger is rated for 240/480V AC input with 132 volts (nominal), 100 amps dc output. Division 3 dc battery has 60 cells.

The charger is also capable of automatically regulating output voltage within $\pm 1/2\%$ of its rated value at any load between 0 and 100%, with the ac power feeding the charger deviating from the rated voltage by $\pm 10\%$. Thus the Division 3 battery charger will not be affected by the voltage sag which occurs when the HPCS pump is started on the DG. The 125V DC battery will be able to maintain the bus voltage.

All dc loads connected on the division 3 dc bus are rated for operation in the voltage range of 112.5V to 137.5V.

8.3.2.1.1
Five independent 125 VDC systems are provided to supply Reactor Island normal and emergency DC power for each unit as appropriate. Four of the five 125 VDC systems are Class 1E power. The fifth system supplies non-Class 1E power.

The DC power systems provide adequate power for station emergency auxiliaries and for control and switching during all modes of operation. The operating voltage range of Class 1E dc loads is 110V to 140V.

The maximum equalizing charge voltage for Class 1E batteries is 140Vdc.

The dc system minimum discharge voltage at the end of the two hour discharge period is 1.83Vdc per cell.

The 125 VDC systems provide a reliable control and switching power source for the Class 1E systems.

All batteries are sized so that required loads will not exceed 80% of nameplate rating, or warranted capacity at end-of-installed-life with 100% design demand. Each 125 VDC battery is provided with two chargers, each of which is capable of recharging its battery from a discharged state to a fully charged state while handling the normal, steady-state DC load.

Battery sizes are specified as:

- (1) Battery E, Division 1 - 1950 A-hr at 8-hr rate; 2080A for 1 min
- (2) Battery F, Division 2 - 1500 A-hr at 8-hr rate; 1620A for 1 min
- (3) Battery G, Division 3 (HPCS) - 400 A-hr at 8-hr rate; 500A for 1 min
- (4) Battery H, Division 4 - 425 A-hr at 8-hr rate; 550A for 1 min
- (5) Battery I, nondivisional - 2550 A-hr at 8-hr rate



8.3-181/8.3-182

430.41
(8.3)

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

Response

The response to this question will be provided by the Applicant. At that time General Electric and the diesel-generator vendors will supply recommendations to the Applicant.

430.54
(9.5.3)

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

Response

Illumination levels of the control room and the remote shutdown panel are addressed in Chapter 13.

430.61
(9.5.4)

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

Response

Division 1 and 2 day tank vents terminate 6-inches beyond building wall into the dock area. These terminations are protected by the roof over the dock area and by the wall around the dock area. Division 3 vent terminates on a 45 degree down slope at the outside surface of the Diesel Generator Building wall. Bird screens cover all three terminations. All three terminations will be located on drawing K-036 (Figure 1.2-19) and the attached Section H-H added to drawing K-037 (Figure 1.2-20).

R W Christiansen

General Electric

San Jose

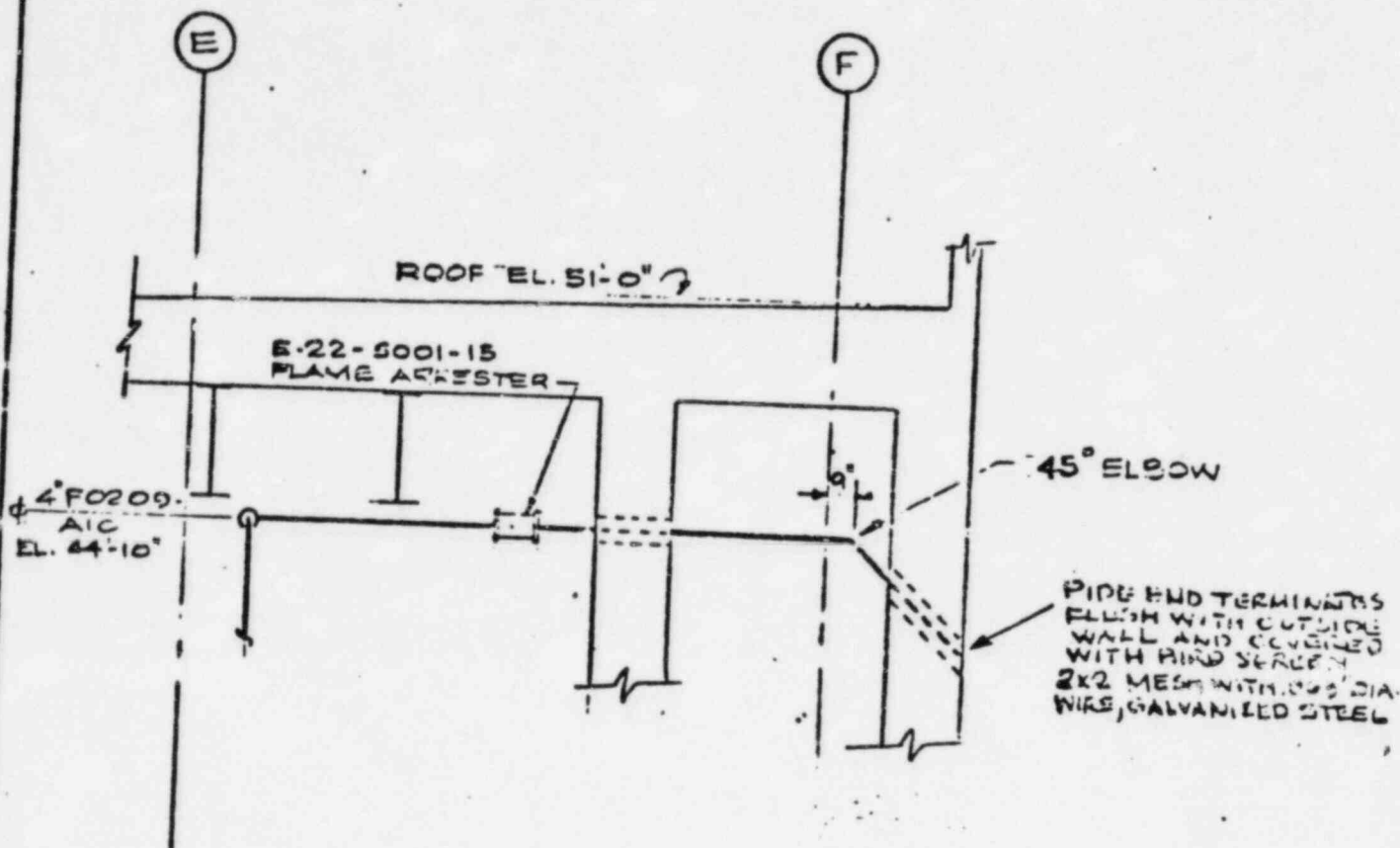
CESSAR

ROUND 1 QUESTIONS

Project 6382-3

September 29, 1982

QUESTION 430.61



SECTION H-H

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

Response

There are no high-energy lines in the diesel-generator rooms. See section 3.6.

The moderate-energy lines are the diesel starting air, fuel oil, intake air, lube oil vent, essential service water and the equipment drain systems. All the above moderate-energy lines are only associated with their respective divisions.

3 The engine exhaust lines are exempted from high energy lines because ^{their expected} ~~the~~ usage does not exceed 2% of the time over the 40 years design life. Also, failure of the exhaust lines and consequential damages are confined to each respective division.

430.68 (cont'd)

- ~~e. The BOP fuel oil transfer pump minimum capacity is gallons per minute & is by the Applicant. (The discharge head requirement for those portions of the system associated with the nuclear island is 16 PSIG for Div 1 + 2 and 25 PSIG for Div 3.~~
- 2 d. The minimum quantity of fuel to be stored for each diesel-generator shall be based on 7 days supply at maximum specific fuel oil consumption See 9.5.4.1.1(c).
- e. The diesel fuel oil quality standards which must be met is the ASTM specification D975 for Diesel fuel oils, #2 diesel fuel, with not less than 35 cetane number. The above meet the requirement of Reg Guide 1.137.
- 2 c. The BOP fuel oil transfer pump minimum capacity shall be not less than the specific fuel oil consumption of the diesel engine which does not exceed 0.38 pounds of fuel per net horsepower hour. ^{See 430.57} The minimum discharge head shall be not less than 16 PSIG for Div 1 + 2 and 25 PSIG for Div 3. Compliance is by the Applicant.

QUESTION 430.78
(9.5.5)

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

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

RESPONSE

Division 3 diesel-generator cooling water system is ^{specified} ~~designed~~ to maintain the engine in a warm standby condition in accordance with the quick start reliability requirements. The specific details of the system functions to achieve this will be provided by the applicant depending on the type of the keepwarm system furnished for a particular engine.

During the standby mode of the diesel-generator, the jacket water temperature is maintain above ambient temperature by means of an electric jacket water heater for Div 1 and 2 and an electric immersion heater for Div 3. The heated water is circulated by a jacket water keep warm pump for Div 1 and 2, and by gravity and natural circulation for Div 3.

The design jacket water temperatures at the anticipated ambient temperatures of 60°^{minimum} F. in the diesel-generator rooms is 120° F. Compliance is ^{by the Applicant.} The excess capacity, if any, in the jacket water heating system is by the Applicant. ^{10% margin is recommended.} See section 9.5.5.2, page 9.5-28, as attached in 430.69.

3

R W Christiansen

General Electric

San Jose

GESSAR
ROUND 1 QUESTIONSProject 6382-P
September 29, 1982

QUESTION/RESPONSE 430.93 (9.5.5)

QUESTION 430.93

Provide enlarged and more detailed plan and elevation views of the Division 3 Diesel Generator Air Start System Air Compressors. Show the intake, the exhaust, the cooling system and the fuel supply for the diesel engine-driven compressor. Incorporate these enlarged views into the appropriate drawings in Section 1.2 of your FSAR.

RESPONSE 430.93

The diesel engine driven air compressor is an air-cooled type and requires no cooling water. The fuel supply is provided by a tank locally mounted on the air compressor base. The air intake is through a filter mounted on the compressor head. The diesel engine exhaust is piped to the Diesel Generator Building stack. The attached detail will be added to drawing K-035 (Figure 1.2-18).

R W Christiansen

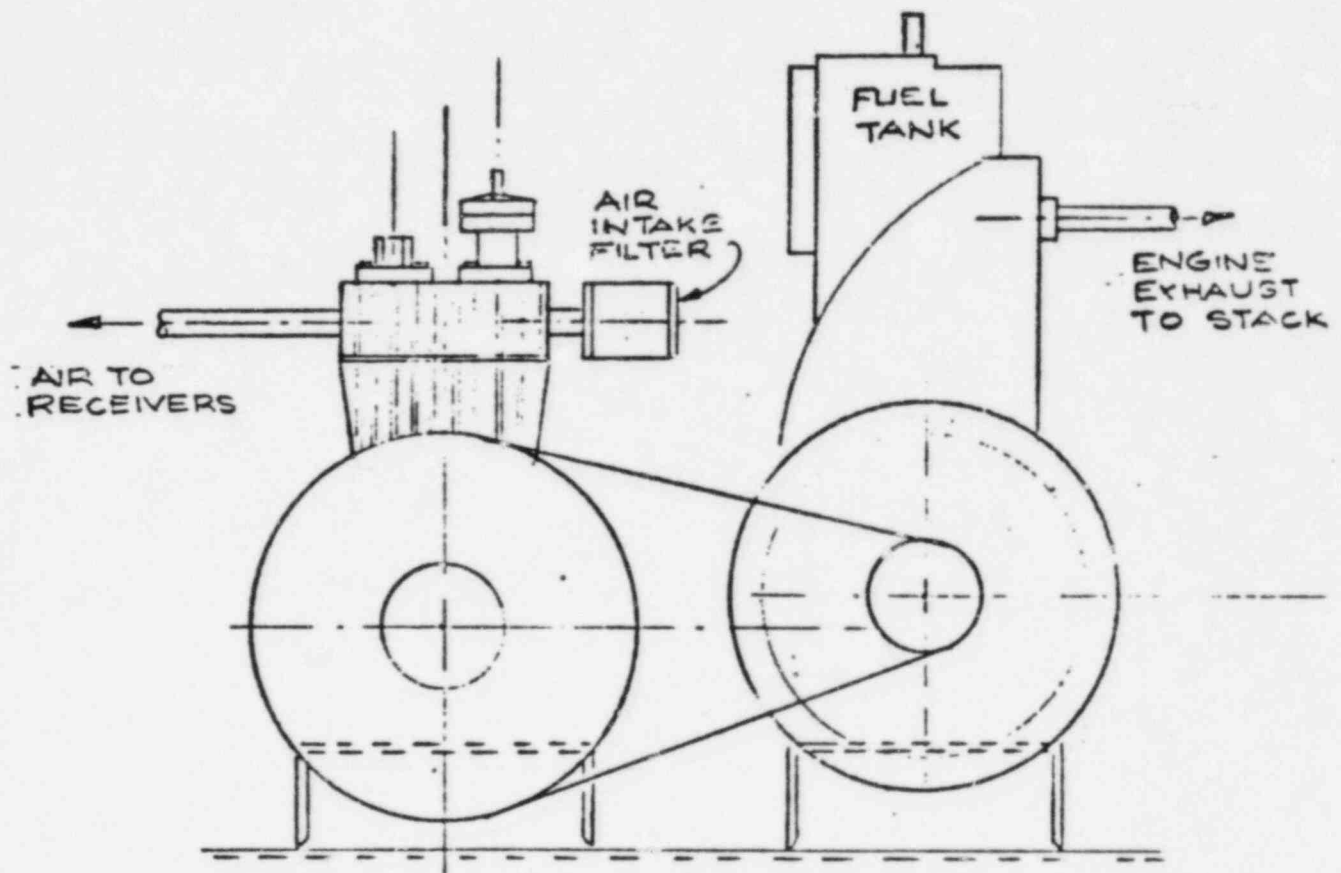
General Electric

CESSAR
ROUND 1 QUESTIONS

Project 6382-
September 29, 1961

San Jose

QUESTION 430.93



DETAIL 1

STARTING AIR COMPRESSOR E22-5001-9
AIR-COOLED DIESEL ENGINE-DRIVEN

430.101
(9.5.7)

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

Response

It was specified for the Div 1 and 2 diesel-generator lube oil system that the annunciator shall include annunciation of the lube oil pressure low and lube oil high differential pressure. All the rest of the alarms necessary shall be by the Vendor. Figure 9.5-16 is an exact replica of Delaval's drawing 29-513-17021 Brook SPR 52000 and 300. Therefore, the Vendor should provide their drawing showing complete alarm system, in such that Figure 9.5-16 could be revised to show complete alarm system. Afterward, Section 9.5.7.4 could also be revised to comply with Figure 9.5-16.

Our drawings are not intended to show all instrumentation within the Applicant's scope of supply. See response to 430.09 and revised FSAR section 8.3.1.1.8.1.5 for alarms. The text takes precedent over the figures or drawings.

QUESTION 430.105
(9.5.7)

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

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

RESPONSE

The implementation of MI-9644 recommendation to be answered by the applicant.

Division 3 diesel generator has a continuously operating soakback pump which provides lubrication to the turbo-charger parts in the standby condition.

For the Division 1 & 2 diesel engines
a continuously pressurized prelubrication system
is specified unless the vendor can demonstrate
that it is not required and is undesirable.
In some engines possible leakage ^{prelubrication} during shutdown
~~along~~ the valve stems into the cylinders could
result in damage to the engine on a subsequent
start. Confirmation of compliance to the
requirements is by the Applicant.

430.111
(9.5.7)

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

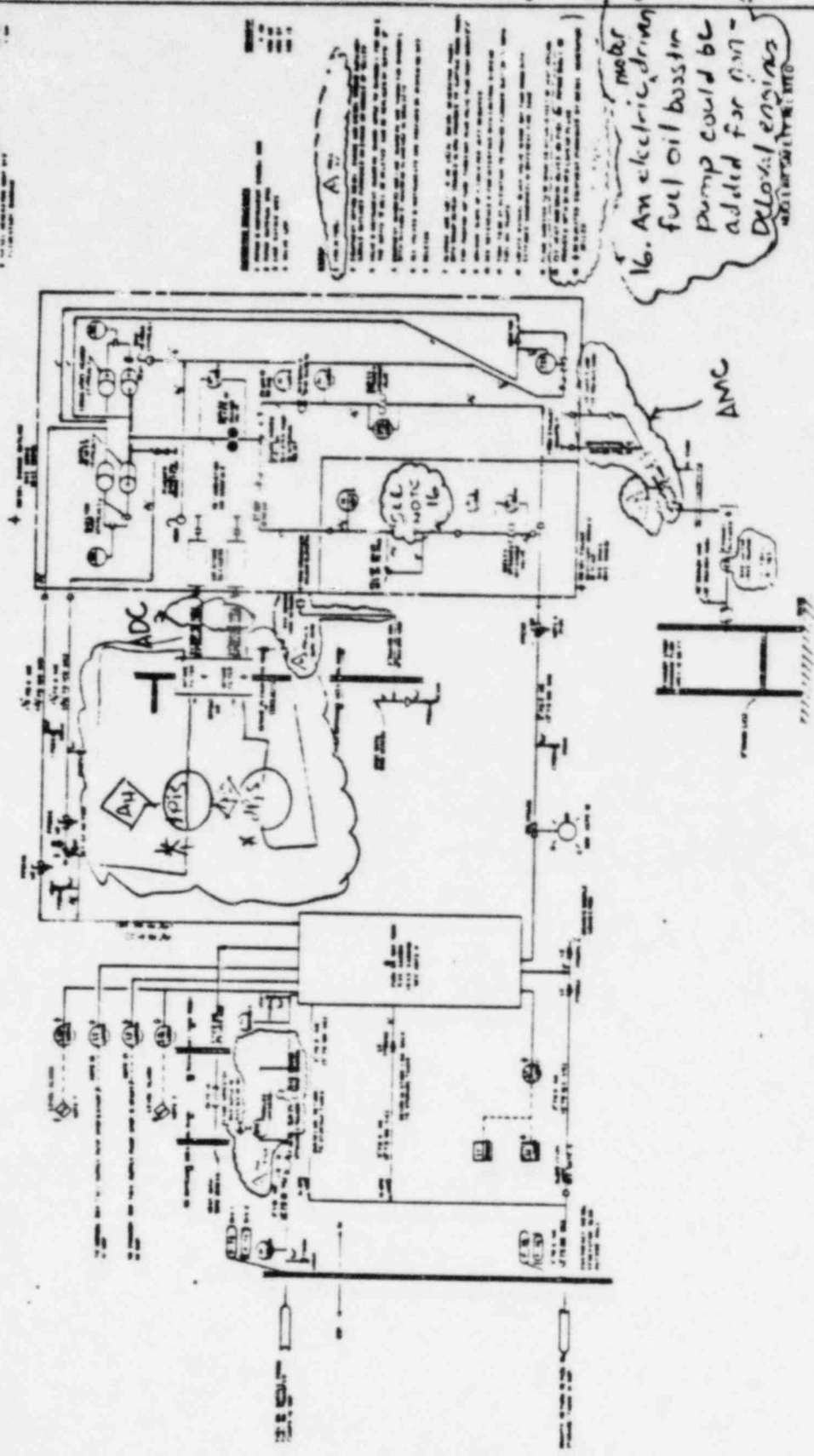
The combustion air intake and exhaust systems shown in Figure 9.5-10 is complete. All instrumentation and controls associated with the systems but are not shown are by applicant. A differential pressure gauge across each filter including high differential alarm will be added to Figure 9.5-10 as shown on the attached marked up in 430.66. See response to 430.112.

130.66 ATTACHMENT
130.111 ATTACHMENT

Drawing K-132, Rev 6

Figure 9.5-10. Division 1 and 2 Diesel Fuel Transfer, Intake and Exhaust System P&I Diagram

9.5-73/9.5-74



430.112
(9.5.8)

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

Response

The Div 1 and 2 air intake filters are provided with a differential pressure gauge across each filter, ^{including a high differential pressure alarm}. Additional instrumentation, controls, sensors, and alarms provided in the design of the diesel engine combustion air intake and exhaust system are by Applicant as required. ~~Vendors requirements applicable to a particular situation and environment. See response to 430.09 For alarms and revised F&E section 8.3.1.1, 8.1.5.~~

Div. 3 (HPCS) DG air intake filter is not provided with a pressure gauge or a differential pressure switch. ~~However, additional instrumentation, sensors and alarms can be provided, if required by the applicant.~~

430.115
(9.5.8)

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

Response

Applicant will

The ~~Vendor~~ should provide a more specific discussion on the effect of decreasing barometric pressure on diesel-engine performance.

It was specified that the diesel-engine should be able to stand a maximum outdoor tornado-induced pressure change of 2 psi per second for Div 1 and 2 diesel-generator.

& (-) 3 PSI. ^{The total duration of this transient is from 4 to 41 secs.} See FSAR Section 3.3.2.

9

Div 3 (HPCS) DG is specified to operate at the environmental conditions equivalent to the altitude of 1000 feet above sealevel. It is also designed to operate at an ambient atmospheric pressure to approximately (-) 0.25 inch of water.

430.117
(9.5.8)

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

Response

Potential fire in the Division 2 and 3 diesel-generator building will not degrade the quality of the diesel combustion air, thereby permitting the remaining diesel-generator to provide its full rated power for the following reasons: Each Division 2 and 3 DG buildings are independent from each other and will be totally isolated from each other in case of occurrence of fire in either building. The location of each air intake are far away as possible so that in case of fire in one building the remaining air intake will not be affected.

See Figure 1.2-18 for location of Div 1 & 2 air intake and Fig 1.2-19 for Div 3

ATTACHMENT NO. 6

DRAFT RESPONSES TO
PROCEDURES AND TEST REVIEW BRANCH
QUESTIONS

ATTACHMENT NO. 6

DRAFT RESPONSES TO
PROCEDURES AND TEST REVIEW BRANCH
QUESTIONS

640.07

1. Emergency response information system (ERIS) tests.

ONLY CHANGE

Response

The Applicant's FSAR will describe the ERIS tests.

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

Response

These tests are covered by Subsection 14.2.12.3.1.

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

Response

A preoperational test procedure for expansion, vibration and dynamic effects has been added as Subsection 14.2.12.1.75.