GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

MC 682 (408) 925-5040

MFN 035-83
JNF 009-83

February 17, 1983

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

Attention: Mr. D.G. Eisenhut
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

ASSESSMENT OF UNRESOLVED SAFETY ISSUE (USI) TASK A-6 AND REVISED DRAFT RESPONSES

Attached please find an assessment of USI Task A-6 and final draft responses to selected questions of the Commission's August 25, 1982, November 15, 1982 and December 31, 1982 information requests. Only modifications (new or revised) to the response of the referenced letters are provided.

Sincerely,

Willa for

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Attachments

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

E003

LISTING OF ATTACHMENTS PROVIDED

Attachment Number	Subject
1	Assessment of USI Task A-46
2	Draft Responses to Procedures and Test Review Branch Questions
3	Draft Responses to Effluent Treatment Systems Branch Questions
4	Draft Responses to Power System Branch Questions
5	Draft Responses to Reactor Systems Branch Questions
6	Draft Responses to Equipment Qualification Branch Questions

ATTACHMENT NO. 1

ASSESSMENT OF UNRESOLVED SAFETY ISSUE TASK A-46

APPENDIX 1B

CONTENTS
(Continued)

TABLES

1B.1.2 Objective

The unresolved safety issues were initially identified in NUREG-0510 (Identification of Unresolved Safety Issues Relating to Nuclear Power Plants", January 1979). These issues are updated quarterly in NUREG 0606 ("Unresolved Safety Issues Summary"). The quarterly update provides current programmatic and schedule information and includes information relative to the implementation status of each issue for which technical resolution is complete. 9

The overall objective of this appendix is to comply with the Atomic Safety and Licensing Appeal Board decision (ALAB-444) that the Safety Evaluation Report (SER) for each plant should contain an assessment of each significant unresolved generic safety issue. The assessment should include a summary description of relevant investigative programs and the measures devised for dealing with the issues on the subject plant.

1B.1.3 238 Nuclear Island Applicability

The unresolved safety issues outlined in NUREG 0606 include all issues for which technical resolution is not considered complete by the NRC. Several apply only to pressurized water reactors~~, one applies only to operating nuclear power plants,~~ and one applies only to boiling water reactors with a Mark I Containment. The remaining unresolved safety issues which are applicable to the 238 Nuclear Island are given in Table 1B-1. The number of the generic task in the NRC program addressing each issue is given along with the section in which each issue is discussed.

TABLE 1B-1

UNRESOLVED SAFETY ISSUES

Unresolved Safety Issue	NRC Task Number	Applicable Section
Waterhammer	A-1	1B.2.2
Reactor Vessel Materials Toughness	A-11	1B.2.3
Systems Interaction in Nuclear Power Plants	A-17	1B.2.4
Safety Relief Valve Pool Dynamic Loads	A-39	1B.2.5
Seismic Design Criteria	A-40	1B.2.6
Containment Emergency Sump Reliability	A-43	1B.2.7
Station Blackout	A-44	1B.2.8
Shutdown Decay Heat Removal Requirements	A-45	1B.2.9
Safety Implications of Control Systems	A-47	1B.2.10
Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	A-48	1B.2.11
Seismic Qualification of Equipment in Operating Plants	A-46	1B.2.12

1B.2.11.3 Industry Activities and Resolution Status (Continued)

With respect to Task A-48, it is concluded that the 238 Nuclear Island can be operated, with no additional hydrogen control systems, without undue risk to the health and safety of the public.

→ INSERT FOLLOWING 2 PAGES ←

→ INSERT ←
PAGE 1B-32

1B.2.12 SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (TASK A-46)

1B.2.12.1 ISSUE DESCRIPTION

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event.

1B.2.12.2 NRC Activities

IN THE AREA OF TASK A-46, THE NRC IS ATTEMPTING to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose functions is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

ALSO, THE NRC WILL ESTABLISH GUIDELINES FOR USE IN REQUALIFYING EQUIPMENT WHOSE SEISMIC QUALIFICATION HAS BEEN FOUND TO BE INADEQUATE

1B.2.12.3 INDUSTRY ACTIVITIES AND RESOLUTION STATUS

THE PRINCIPAL OBJECTIVE OF TASK A-46 IS TO ENSURE THE ADEQUACY OF THE SEISMIC QUALIFICATION

OF EQUIPMENT IN OPERATING PLANTS WHICH HAVE NOT BEEN DESIGNED TO CURRENT SEISMIC DESIGN STANDARDS. THE 238 NUCLEAR ISLAND, HOWEVER IS designed using current seismic design criteria, and methods for seismic equipment qualification are to be latest codes and standards. Requirements for seismic equipment qualification include IEEE 344-1975 and Regulatory Guides 1.92 and 1.100. Standard Review Plans 3.2.2, 3.9.2, 3.9.3, and 3.10 have also been considered in the qualification efforts.

SINCE THE 238 NUCLEAR ISLAND HAS BEEN DESIGNED USING THE MOST UP TO DATE CRITERIA, THERE IS NO NEED TO DETERMINE THE ADEQUACY OF THE SEISMIC QUALIFICATION IN LIEU OF BACKFITTING CURRENT DESIGN CRITERI[A] IN ADDITION, SINCE CURRENT CRITERIA IS ALREADY USED FOR THE SEISMIC QUALIFICATION, THERE IS NO EQUIPMENT WHOSE QUALIFICATION SHOULD BE INADEQUAT[E]

WITH RESPECT TO TASK A-46 IT IS CONCLUDED THAT THE 238 NUCLEAR ISLAND CAN BE OPERATED WITHOUT UNDUE RISK TO THE HEALTH AND SAFETY OF THE PUBLI[C]

ATTACHMENT NO. 2

DRAFT RESPONSES TO
PROCEDURES AND TEST REVIEW BRANCH
QUESTIONS

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

Response

Subsection 14.2.7 has been revised to eliminate the detailed discussions pertaining to regulatory guide conformance. Rather, Subsection 14.2.7 now references Section 1.8 which provides the detailed discussion of the subject regulatory guides. The Section 1.8 treatment of these regulatory guides are attached.

1.8.68.1

~~1.8.16~~ Regulatory Guide 1.68.1, Revision 0, Dated December 1975

Title: Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants

This guide describes in detail the type and nature of BWR feedwater and condensate system tests that are acceptable to the staff.

Evaluation

The preoperational and startup testing of the Feedwater and Condensate Systems are not within the GE scope of services; therefore, this guide is not applicable ~~to~~ GESSAR II.

However, ~~GESSAR~~ II Chapter 14 states that a comprehensive testing program will be developed to ensure that all nuclear safety-related equipment and systems will perform in accordance with their design criteria. As individual systems are completed, they will be tested, reviewed, and approved according to predetermined and written procedures. In general, all procedures will be developed in accordance with NRC publications such as Regulatory Guide 1.68.1.

1.8.68.2

~~NN.2.2~~ Regulatory Guide 1.68.2, Revision 1, Dated July 1978

Title: Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

This guide describes an initial startup test program acceptable to the NRC staff for demonstrating the capability to shut down the hot operating and cold reactor from outside the control room by verifying that:

(1) the nuclear power plant can be safely shut down from outside the control room;

(2) the nuclear power plant can be maintained in the hot shutdown condition from outisde the control room; and

(3) the nuclear power plant has the potential for being safely cooled from hot and cold shutdown conditions from outside the control room.

Evaluation

The Remote Shutdown System is designed with the capability to accomplish the objectives of the test program outlined in the regulatory guide.

The system described in GESSAR II Section 7.4 provides remote control for reactor systems needed to carry out the shutdown function from outside the main control room and bring the reactor to cold condition in an orderly fashion.

The system provides a backup variation to the normal system used in the main control room permitting the shutdown of the reactor from outside the control room when feedwater is unavailable and normal heat sinks are lost (turbine and condenser).

1.8.68.2 Regulatory Guide 1.68.2, Revision 1, Dated July 1978 (Continued)

Activation of the relief valves and the Reactor Core Isolation Cooling (RCIC) System will bring the reactor to a hot shutdown condition after scram and isolation. During this phase of shutdown, the suppression pool will be cooled as reuqired by operating the Residual Heat Removal (RHR) System in the suppression pool cooling mode. Reactor pressure will be controlled and core decay and sensible heat rejected to the suppression pool by releasing steam through the relief valves. Reactor water inventory will be maintained by the RCIC system. This procedure will cool the reactor and reduce its pressure at a controlled rate until reactor pressure becomes low enough to discontinue RCIC operation.

The design basis assumptions for the Remote Shutdown System do not include control room damage so massive that all ECCS capabilities are lost, in fact, the design assumes all necessary cooldown and residual heat removal equipment remains operational. Paragraph C4.C is interpreted to apply only with the reactor scrammed - not with failure to scram.

1.8.95

~~1.8.38~~ Regulatory Guide 1.95, Revision 1, Dated January 1977

Title: Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

This guide describes assumptions acceptable to the Regulatory staff to be used in assessing the habitability of the control room during and after a postulated external release of chlorine. It also describes requirements for control room isolation and emergency procedures.

Evaluation

In the GESSER II design, chlorine is identified by human detection in accordance with Paragraph C.7 of Regulatory Guide 1.78. For specific cases in which a plant would be sufficiently close to a railroad or highway, analysis will be carried out to show whether or not the chlorine limits stated in Regulatory Guide 1.78 will be exceeded due to a postulated accident. If the limit could be exceeded, chlorine detection devices will be placed in the control room and intake ducts. The plan would also include an automatic isolation system.

1.8.108

~~11.1.8~~ Regulatory Guide 1.108, Revision 1 Dated August 1977

Title: Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

This regulatory guide describes diesel generator design features and testing provisions which are acceptable for verification of availability and reliability of standby power sources.

Evaluation

The design of the Standby Diesel Generator Systems are in conformance with the subject regulatory guide with the following exception to paragraph C.1.b(5): The diesel generator surveillance system is not required to have a first-out annunciation feature because annunciation of individual protective trips give the operator adequate information for correct action.

With regard to the proposed Division 1 and 2 standby power sources, the staff required GE to conform to the position outlined in Standard Review Plan Appendix 7 BTP EICSB2, Diesel Generator Reliability Qualification Testing. GE has stated its commitment to a qualification program in conformance with this commitment.

GESSAR II states that readiness of the diesels is of prime importance and will be demonstrated by periodic testing. The testing program will be designed to test the ability to start the ESF system loads as well as to run under the load long enough to bring all components of the system into equilibrium conditions. Full functional tests of the automatic control circuitry will be conducted on a periodic basis to demonstrate correct operation.

1.8.128

~~1.2.35~~ Regulatory Guide 1.128, Revision 1, Dated October 1978

Title: Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

This regulatory guide describes an installation design for Class IE batteries that is in general agreement with IEEE-484-1975.

Evaluation

The 125-volt dc systems are divided into four Class IE divisions. Each system has dc battery, battery charges, and load center distribution panels. These are designed as Class 1E equipment in accordance with applicable classes of IEEE Standard 308-1974. The plant design and layout from these dc systems will provide physical separation of the equipment, cabling, and instrumentation essential to plant safety. Each system is located in its own ventilated room and all the components are housed in a safety class structure. The battery rooms are independently ventilated to keep the gases produced due to charging of the batteries below an exposure concentration.

Therefore the Class 1E battery installations are in accordance with the Regulatory Guide with the following interpretation of Position C.1:

> The area in the immediate vicinity of the battery vent is excluded from the area defined by the phrase....."at any location within the battery area."

Fire detection instrumentation is part of the fire protection system rather than the battery system.

1.8.140

~~1.8.46~~ Regulatory Guide 1.140, Revision ~~0~~ 1, Dated ~~March 1978~~ OCTOBER 1979

Title: Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

This guide presents methods acceptable to the NRC staff for implementing the Commission's regulations in 10CFR50 and in Appendices A and 1 to 10CFR50 with regard to the design, testing, and maintenance criteria for air filtration and adsorption units installed in the normal ventilation exhaust systems of light-water-cooled nuclear power plants. This guide applies only to atmosphere cleanup systems designed to collect airborne radioactive materials during normal plant operation including anticipated operational occurrences and addresses the atmosphere cleanup systems including the various components and ductwork in the normal operating environment. This guide does not apply to post-accident engineered-safety-feature atmosphere cleanup systems that are designed to mitigate the consequences of postulated accidents. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, provides guidance for these systems.

Evaluation

The design described in GESSAR II Section 9.4.5 makes provision for air filtration and adsorption units in the Containment Purge, Exhaust, and Pressure Control System only. However the need for filtration and adsorption units is expected to be determined on a site unique basis. If filtration and adsorption units are installed in the Containment Purge, Exhaust, and Pressure Control System, they will comply with this guide. The determination of whether filtration and adsorption systems should be included on normal effluent systems is not interpreted to be within the scope of this guide. Therefore, this guide is considered not applicable unless the determination is made that filtration and adsorption are required.

ATTACHMENT NO. 3

DRAFT RESPONSES TO
EFFLUENT TREATMENT SYSTEMS BRANCH
QUESTIONS

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

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

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

Response a

Currently, the overall filter train pressure drop is monitored by an indicator and recorder in the control room. Individual components inside the filter train (demister, pre-filter, HEPA filters, and charcoal absorber) pressure drops, are monitored by indicators located in local panel. These pressure drops are monitored for each filter train.

However, to be in accordance with Regulatory Guide 1.52, a flow indicator and recorder must be added for each filter train as well as alarms. Alarms for the pressure drops must also be included

The Applicant will commit to providing the above modifications to GESSAR II via his FSAR so that the instrumentation for measuring flow rates through the ESF filter systems are in accordance with Regulatory Guide 1.52, Revision 2.

Response b

The specific type of recording device for recording pertinent pressure drop and flow rates is dependent on the Applicant and his choice of equipment vendors. Hence, the Applicant will indicate the type of recording device in his FSAR.

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

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

Response a

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

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

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

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

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

Text Modifications for 460.14

9.4.3.4 Inspection and Testing Requirements (Continued)

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

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

9.4.3.5 Instrumentation Application

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

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

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

(460.14 Cont)

11.5.1.1.2 Systems Required for Plant Operation (Continued)

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

(1) for gaseous effluent streams -

(a) plant vent discharge,

(b) offgas exhaust vent,

(c) radwaste building ventilation RMS, and

(d) turbine building ventilation RMS;*

(2) for liquid effluent streams -

(a) radwaste effluent RMS and

(b) service water effluent to cooling pond RMS;

(3) for gaseous process streams -

(a) offgas pretreatment RMS,

(b) offgas post-treatment RMS, and

(c) carbon bed vault RMS; and

(4) for liquid process streams -

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

(b) closed cooling water RMS.

* Applicant responsibility

11.5.2.1.4 Auxiliary Building Exhaust Radiation Monitoring

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

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

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

INSERT (FROM NEXT PAGE)

11.5.2.1.5 Standby Gas Treatment Radiation Monitoring System

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

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

INSERT ON PAGE 11.5-11

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

460.14

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).**

Response e

The GESSAR II source terms are consistent with BTP ETSB 11-5 for any possible failure of the offgas system. Since the offgas system has redundant active components, off-site doses are not affected by a postulated single active failure. The postulated failure of the first charcoal tank of the offgas system described in subsection 15.7.1.1, results in the higest potential iodine release. The calculated total body doses are within 500 mRem for the design basis analysis*.

---

* Note that the total body dose in Table 15.7-5 is given as 1680 mRem. However, this is a calculational error and it is actually less than 500 mRem. This will be corrected in the next amendment to GESSAR II. Table 15.7-4 also contains several typographical errors which will be corrected as shown on the attachment to this response.

Table 15.7-4

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE (DESIGN BASIS ANALYSIS) FISSION PRODUCT RELEASE TO ENVIRONMENT

Isotope	Ci	Isotope	Ci	Isotope	Ci
Cr23	1.39E-2	H3	1.01E-3	Ru103	1.07E-7
24	3.02E-2	C14	9.81E-5	105	1.08E-6
25	2.96E-2	Na24	4.18E-6	1-6	4.65E-9
I131	9.98E-3	P32	4.01E-8	Ag110m	2.25E-7
132	1.28E-1	Cr51	9.70E-7	Te129m	2.76E-8
133	7.43E-2	Mn54	4.69E-8	129	3.98E-6
134	2.87E-1	56	1.06E-4	131m	1.86E-7
135	1.19E-1	Fe59	1.47E-7	131	2.76E-7
		Co58	8.58E-6	132	1.08E-6
Kr83m	9.00E+1	60	3.74E-7	Co187	4.88E-8
85m	1.54E+2	Ni65	6.84E-7	188	1.72E-8
85	8.24E-2	Zn65	2.71E-9	Co189	6.29
87	4.19E-12	Rb88	1.42E+2	140	8.44E-8
88	4.88E-12	89	6.80E+1	141	2.84E-8
89	1.52E-13	Br89	2.96E-2	142	4.31E-8
90	3.35E-13	90	4.34E-5	La140	1.08E-4
Xe131m	8.56E-1	91	8.12E-2	142	1.09E-8
133m	8.81	92	1.30E-3	Ce141	2.00E-
133	2.14E+2	Y90	3.51E-7	143	9.87E-7
135m	3.87E+2	91m	1.61E-2	144	7.66E-8
135	6.85E+2	91	1.75E-5	Nd147	1.18E-8
137	1.87E+3	92	5.31E-6	W187	1.90E-8
138	1.26E+3	93	3.48E-6	Np289	1.81E-3
139	3.54E+3	Zr95	1.83E-7		
140	3.17E+3	97	1.73E-6		
		Nb95	1.60E-4		
		Mo99	1.69E-6		
		Te99m	8.97E-4		
		101	1.14E-3		

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

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

Response C

The leak reduction measures incorporated into the GESSAR II design are described in revised section 1A.77 (attached).

1A.77 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED-WATER REACTORS AND BOILING-WATER REACTORS (NUREG-0737 Item III.D.1.1) (Cont'd)

Response

Leak reduction measures of the

~~The~~ 238 Nuclear Island include ~~provides~~ a number of barriers to containment leakage in the closed systems outside the containment. These closed systems include:

a. Residual Heat Removal,

b. High Pressure Core Spray,

c. Low Pressure Core Spray,

d. Reactor Core Isolation Cooling,

e. Suppression Pool Cleanup (suction and return), and

f. Shutdown Service Water (supply and return).

Plant specific procedures provided by the Applicant will prescribe the method of leak testing these systems. The testing will be performed on a schedule appropriate to 10CFR50 Appendix J type B and C penetrations, that is at each refueling outage. When leakage paths are discovered including during these tests, the will be investigated [redacted] when necessary, & maintenance will be performed to reduce leakage to its lowest practical level.

INSERT →

Additionally, pressure boundary components of radioactive waste systems are purchased as augmented Class D systems to assure their capability to provide integrity.

INSERT ON PAGE 1A.77-2

In addition, Lines which penetrate the primary containment contain primary containment isolation valves which are designed in accordance with General Design Criteria 55, 56 or 57 to provide reliable isolation in the event of line breaks. These isolation provisions are discussed in Subsection 6.2.4 and contain both automatic and remote manual-closing valves.

Should a small line break develop within a space inside the secondary containment concurrent with a significant radioactive source term in the reactor water, it would be detected by the Leakage Control Systems described in Subsection 5.2.5 and the line may be isolated. Any release of radioactive material from such leaks would also be detected by process radiation monitors which would permit operation of the Standby Gas Treatment System (Subsection 6.5.1) prior to release to the environment. All lines which pass outside of the secondary containment contain Leakage Control Systems or loop seals. These systems allow the SGTS to maintain a negative pressure relative to the environment and thus limit the amount of leakage through the secondary containment. These systems are discussed in Subsections 6.5.3. Finally, expected liquid leakoff from equipment outside the containment is directed to equipment drain sumps and processed by the Radwaste System. These multiple design features of the 238 Nuclear Island provide substantial capability to limit any potential release to the environment from systems likely to contain radioactive material.

ATTACHMENT NO. 4

DRAFT RESPONSES TO
POWER SYSTEMS BRANCH
QUESTIONS

430.18
(8.3.1)

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

<u>Response</u>

See attached sheets.

FSAR Section 8.3.1.4.2.3.2 will be revised to provide additional information as shown in the attachment.

RESPONSE

The GESSAR II design does not have cable spreading areas as defined by IEEE 384, Section 5.1.3. According to that definition, a cable spreading area is a space or spaces adjacent to the main control room where instrumentation and control cables converge prior to entering the control, termination, or instrument panels. In application a cable spreading room allows the design of the trays and cabling external to the control room to proceed independently of the layout of the control room cable trays and panels. Mismatches develop in the grouping of the plant cables with respect to their required grouping for the control room. This mismatch is corrected by making the required routing transitions through interconnected trays in a cable spreading area. This is not the case for the GESSAR II Design.

For the GESSAR II design the field cables, starting at the end devices, enter the conduit and tray system and are routed back to main cable trays which are routed through the auxiliary building to the control building. This conduit and tray system may be visualized by thinking of each end device as being at the end of a tree branch. The cable then follows a branch back to a limb or a series of limbs and then finally to the trunk or main cable tray. These main tray runs were located in the best areas available for cable trays. Avoiding such things as piping systems, high energy equipment, flammable material sources and maintenance hazards was considered as tray routes were selected.

By the time a main cable tray penetrates the three hour fire rated wall of the control building, there are no cables to gather up. They are all in the appropriate tray. Each cable remains in the same tray until it peels off in a branch (tray or conduit) to the proper PGCC termination cabinet. All trays are solid metal with solid metal covers.

A listing of the power cables passing through the cable tray areas of the control building is attached. Each cable has been marked with a service number based on the following descriptions:

Service Number	Description
1	120VAC Load on the AC instrument bus
2	125VDC Load on the DC instrument bus
3	Current transformer output leads
4	120VAC Miscellaneous power (panel space heaters, description panel fear, etc.,)
5	480VAC Instrument bus transformer feed
6	480VAC Regulating transformer feed
7	480VAC Feeder to control building MCC
8	480VAC Load in control building
9	480VAC Load on control building MCC

All of the cables either serve loads or access power supplies located in the control building. Cable service numbers 1 through 4 are for 120VAC or 125VDC maximum. Cables with service numbers 5 through 9 are 480VAC.

JEM:pc/L02025-1
2/2/83

The cable tray areas are specifically designed to safely contain all types of cables which are routed through them. The areas are ventilated and have smoke venting systems. The rooms are separated from the control room by two hour equivalent fire rated walls and separated from the remainder of the plant by three hour rated walls. The rooms have fire detection systems and sprinkler systems. They are secured and not in any traffic pattern. The power trays, of which there are a total of three, are the top trays in the stacks. All trays are solid metal with solid metal covers. Any internally generated electrical fire would be confined to the originating tray. The divisional separation is such that if a fire should develop in one of the rooms, the plant operator would know immediately which safety systems he could plan on using to maintain cooling to the plant and which systems could be shutdown to commence fire fighting. He would also know that the remote shutdown panel would still be available for use.

In short, the control building cable tray areas are the safest and best place to route cables within the control building. Since the loss of either a power or control cable degrades a system, attempting to route a cable elsewhere just because it is a power cable only exposes the system to additional possibilities of damage and subsequent curtailment of operation.

See attached GESSAR II text revision for additional discussion.

JEM:pc/L02025-2
2/2/83

Power Cables Passing Through The Control Building Cable Tray Area

Division 1 Area

Cable No.	Service	Diameter	
A1 -B21C-4000-D1V3	2	.158	
A1 -D23A-4003-D1V3	9	.229	
A1 -P45 -4000-D1V3	8	1.003	1 feed, 2 conductors/ø
A1 -P45 -4018-D1V3	8	1.003	
A1 -P45 -4019-D1V3	8	1.003	
A1 -P45 -4020-D1V3	8	1.003	
A1 -P45 -4021-D1V3	8	1.003	
A1 -P45 -4022-D1V3	8	1.003	
A1 -P61 -4003-D1V3	1	.158	
A1 -R11 -4001-D1V3	6	.338	
A1 -R11 -4007-D1V3	5	.423	
A1 -R24 -4021-D1V3	7	1.003	1 feed
A1 -R24 -4022-D1V3	7	1.003	
A1 -R24 -4098-D1V3	7	1.003	
A1 -R24 -4099-D1V3	7	1.003	
A1 -R24 -4100-D1V3	7	1.003	
A1 -R24 -4101-D1V3	7	1.003	
A1 -R24 -4137-D1V3	4	.338	
A1 -R28 -4002-D1V3	2	.467	
A1 -R28 -4009-D1V3	2	.467	

Division 2 Area

Cable No.	Service	Diameter	
A1 -D23A-4007-D2V3	9		
A1 -R11 -4005-D2V3	5	.423	
A1 -R11 -4806-D2V3	5	.423	
A1 -R11 -4077-D2V3	5	.338	
A1 -R24 -4028-D2V3	7	1.003	1 feed
A1 -R24 -4029-D2V3	7	1.003	
A1 -R24 -4112-D2V3	7	1.003	
A1 -R24 -4113-D2V3	7	1.003	
A1 -R24 -4114-D2V3	7	1.003	
A1 -R24 -4115-D2V3	7	1.003	
A1 -R24 -4139 D2V3	4	.338	
A1 -R28 -4003-D2V3	2	.467	

A1 -X93 -4049-NDV3	8	.194
A1 -X93 -4050-NDV3	8	.194
A1 -X93 -4056-NDV3	8	.194
A1 -X93 -4057-NDV3	8	.194
A1 -X99C-4002-NDV3	1	.158
A1 -X99C-4007-NDV3	1	.158
A1 -X99C-4009-NDV3	1	.158
A1 -X99C-4010-NDV3	1	.158
A1 -X99C-4011-NDV3	1	.158
A1 -X99C-4012-NDV3	1	.158
A1 -X99C-4014-NDV3	1	.158
A1 -X99C-4015-NDV3	1	.158
A1 -X99C-4017-NDV3	1	.158

A1 -B13 -4001-NDV3	1	.158
A1 -B33A-4076-NDV3	2	.158
A1 -B33A-4077-NDV3	2	.158
A1 -C51C-4025-NDV3	1	.158
A1 -D17Z-4001-NDV3	1	.158
A1 -D21A-4005-NDV3	1	.158
A1 -D21A-4006-NDV3	1	.158
A1 -D21A-4009-NDV3	1	.158
A1 -D21A-4011-NDV3	1	.158
A1 -D21A-4012-NDV3	1	.158
A1 -D21A-4013-NDV3	1	.158
A1 -D21A-4014-NDV3	1	.158
A1 -D21A-4015-NDV3	1	.158
A1 -D21A-4016-NDV3	1	.158
A1 -D21A-4017-NDV3	1	.158
A1 -D21A-4018-NDV3	1	.158
A1 -D21A-4019-NDV3	1	.158
A1 -D21A-4020-NDV3	1	.158
A1 -D21A-4022-NDV3	1	.158
A1 -D21A-4024-NDV3	1	.158
A1 -D21A-4026-NDV3	1	.158
A1 -D21A-4028-NDV3	1	.158
A1 -D21A-4030-NDV3	1	.158
A1 -D21A-4032-NDV3	1	.158
A1 -D21A-4034-NDV3	1	.158
A1 -D21A-4036-NDV3	1	.158
A1 -D21A-4038-NDV3	1	.158
A1 -D21A-4040-NDV3	1	.158
A1 -D21A-4042-NDV3	1	.158
A1 -D21A-4044-NDV3	1	.158
A1 -D21A-4045-NDV3	1	.158

Division 2 Area (Cont'd)

AI -D21A-4046-NDV3	/	.158
AI -D21A-4047-NDV3	/	.158
AI -D21A-4048-NDV3	/	.158
AI -F42A-4019-NDV3	/	.196
AI -F42A-4027-NDV3	/	.158
AI -F42A-4031-NDV3	/	.158
AI -G33A-4039-NDV3	/	.158
AI -G33A-4061-NDV3	/	.158
AI -G33A-4062-NDV3	/	.158
AI -G36A-[illegible]26-NDV3	/	.423
AI -G41A-4031-NDV3	/	.158
AI -G41A-4039-NDV3	/	.158
AI -G41A-4040-NDV3	/	.158
AI -G46A-4008-NDV3	/	.158
AI -P39 -4028-NDV3	/	.194
AI -P39 -4029-NDV3	/	.194
AI -P39 -4039-NDV3	/	.423
AI -P39 -4041-NDV3	/	.423
AI -P44 -4026-NDV3	/	.158
AI -P44 -4028-NDV3	/	.158
AI -P44 -4029-NDV3	/	.542
AI -P44 -4030-NDV3	/	.403
AI -P44 -4032-NDV3	/	.158
AI -P44 -4033-NDV3	/	.542
AI -P44 -4036-NDV3	/	.158
AI -P44 -4037-NDV3	/	.158
AI -P44 -4038-NDV3	4	.158
AI -P44 -4040-NDV3	/	.158
AI -P48 -4013-NDV3	1	.158
AI -P53 -4008-NDV3	1	.158
AI -P53 -4009-NDV3	1	.158
AI -P53 -4010-NDV3	1	.158
AI -P53 -4011-NDV3	1	.158
AI -P53 -4012-NDV3	1	.158
AI -P53 -4013-NDV3	1	.158
AI -R11 -4000-NDV3	5	1.158
AI -R11 -4003-NDV3	5	.542
AI -R11 -4004-NDV3	5	.615
AI -R11 -4008-NDV3	6	.423
AI -R11 -4011-NDV3	5	.423
AI -R11 -4012-NDV3	6	.467
AI -R11 -4013-NDV3	5	.423
AI -R11 -4015-NDV3	6	.467
AI -R11 -4051-NDV3	5	.615
AI -R11 -4082-NDV3	5	.467
AI -R11 -4083-NDV3	5	.467
AI -R11 -4084-NDV3	6	.338
AI -R11 -4096-NDV3	6	.338
AI -R24 -4000-NDV3	7	1.003
AI -R24 -4044-NDV3	7	1.003
AI -R24 -4045-NDV3	7	1.003
AI -R24 -4134-NDV3	4	.338
AI -R28 -4004-NDV3	2	.615
AI -R28 -4007-NDV3	2	.615
AI -R42 -4060-NDV3	2	.615
AI -R42 -4061-NDV3	2	.615
AI -T41B-4052-NDV3	1	.158
AI -T41B-4054-NDV3	1	.158
AI -T41B-4056-NDV3	1	1.003
AI -T41B-4058-NDV3	1	.158
AI -T41B-4060-NDV3	1	.158
AI -T41D-4020-NDV3	1	.158
AI -T41D-4022-NDV3	1	.158
AI -X73A-4014-NDV3	1	.158

Text Change For 430.18

8.3.1.4.2.3.2 Other Safety-Related Systems (Continued)

(8) Detailed design basis, description, and safety evaluation aspects for a power generation control complex (PGCC) System shall be as comprehensively documented and presented in GE Topical Report, Power Generation Control Complex, NEDO-10466A and its amendments.

PGCC consists of control room panels, racks, floor sections, and termination cabinets. The floor sections are divided into ducts and the termination cabinets have metallic barriers to separate redundant Class 1E wiring.

The floor section ducts are designed so that each duct acts as a raceway and has adequate fire barriers and will contain wiring of only one division. The ducts have solid metal walls and floors, and ~~a~~ removable solid metal covers.

Cable access to the two PGCC areas is provided through two cable tray rooms located on either side of the control room. Each cable tray room is marked on drawings as containing two divisions. In actuality, Divisions 4 and 3 are imbedded in the walls of the Divisions 1 and 2 cable tray rooms, respectively. The imbedded depth is sufficient to provide a 3 hour fire rated separation.

The cable tray rooms do not contain any high energy equipment, rotating equipment, or piping which could be a potential source of missile or pipe whip. No flammable materials are stored in these rooms. Low voltage power cables (V3) are routed through the cable tray rooms to provide power for lighting transformers, regulating transformers and instrument buses within the control building. The areas are utilized for cable tray and conduit routing only, no other major equipment is housed within the cable tray rooms.

See Figures 8.3.30, 8.3.31 and 8.3.32 for physical layouts of the area.

ATTACHMENT NO. 5

DRAFT RESPONSES TO
REACTOR SYSTEMS BRANCH
QUESTIONS

440.05 (5.4.7) **In Section 5.4.7.1.5 of your FSAR, you provide a discussion of the reactor heat removal (RHR) system alternate shutdown cooling mode in which water is discharged through the automatic depressurization system (ADS) valves. Provide, or make reference to, test data confirming that the ADS valves used in your design can pass sufficient water in this mode for the most limiting conditions. Include a discussion of the applicability of the particular tests which you reference.**

Response

The description of the alternate shutdown cooling flow path presented in Section 5.4.7.1.5 has been superceeded by the EPG.

The EPG require the RHR/LPC loops with heat exchangers to be placed in suppression pool cooling.

The other low pressure injection pumps LPCS & LPCI "c" are used to force water through the S/RV.

Utilization of water directly from the suppression pool which has not been passed through the heat exchangers is also desireable from the consideration of avoiding RPV N T or head tension limit.

Test on the S/RV to verify their ability to pass suitable quantities of water for alternate shutdown cooling have been performed. The results of these tests are presented in General Electric Report Number NEDE-24988-P titled "Analysis of Generic BWR Safety/Relief Valve Operability Test Results". Tests were perfomed in response to NUREG-0737.

440.16 (6.3.2) The ECCS contains manual as well as motor-operated valves. There is a possibility that manually operated valves might be left in the wrong position and remain undetected prior to the occurrence of an accident. Examples of such valves include those pairs of normally closed valves which are in the test/drain lines between the HPCS, LPCS and LPCI isolation valves. Provide a list of all manually-operated valves in the safety-related reactor systems, including their location and type. Discuss the methods which will be used to minimize such an occurrence. It is our position that you provide indication in the control room for all critical ECCS valves (manually or motor-operated).

RESPONSE

~~A li~~. ~~of~~ All manually-operated valves in the safety-related reactor systems~~, including their location and type,~~ ~~which~~ have indication in the control room. ~~is provided as follows:~~ These valves, including their type and location, are provided below:

SYSTEM	VALVE MPL NO	LINE LOCATION	VALVE TYPE
RHR	E12-F010	20" RHR 19-EAA	20" Gate - Hand operated
	E12-F029A	18" RHR 7-BAB	18" Gate - Hand operated
	E12-F029B	18" RHR 13-BAB	18" Gate - Hand operated
	E12-F029C	18" RHR 21-BAB	18" Gate - Hand operated
	E12-F039A	12" RHR 10-EAA	12" Gate - Hand operated
	E12-F039B	12" RHR 16-EAA	12" Gate - Hand operated
	E12-F039C	12" RHR 22-EAA	12" Gate - Hand operated
LPCS	E21-F007	12" LPCS 3-EAA	12" Gate - Hand operated
	E21-FF121	4" LPCS 6-BAB	4" Globe - Hand operated
HPCS	E22-F036	12" HPCS 4-EAA	12" Gate - Hand operated
	E22-FF124	4" HPCS 20-EAB	3" Globe - Hand operated
RCIC	E51-FF210	6" RCIC 2-EAB	6" Gate - Hand operated
	E51-FF211	8" RCIC 1-AAB	8" Gate - Hand operated
	E51-FF222	3" RCIC 10-EAB	1" Globe - Hand operated
HPCS SW (ESW Div 3)	P40-FF001	8" CSSW 1-AKC	8" Butterfly - Hand operated
	P40-FF002	8" CSSW 2-AKC	8" Butterfly - Hand operated
ESW (Div 1 & 2)	P41-FF001A	10" ESW 3-ADC	10" Butterfly - Hand operated
	P41-FF001B	10" ESW 43-ADC	10" Butterfly - Hand operated
	P41-FF002A	10" ESW 4-ADC	10" Butterfly - Hand operated
	P41-FF002B	10" ESW 44-ADC	10" Butterfly - Hand operated
	P41-FF006A	10" ESW 4-ADC	10" Butterfly - Hand operated
	P41-FF06B	10" ESW 44-ADC	10" Butterfly - Hand operated

Each of the above valves is monitored by the Performance Monitoring Systems (PMS) for individual alarming of "not fully open". These valves are then grouped by system and division with a status light in the control room for system level indication of "manual valve misaligned". In addition to the status light, a connection is made to the "system out of service" alarm such that an alarm results whenever the status light is on.

The positions of normally closed safety-related reactor system manually-operated valves which are not part of an ECCS loop (such as on the test/drain lines between isolation valves, etc.) are monitored implicity by the Reactor Coolant Pressure Boundary and ECCS System Leakage Detection System described in subsection 5.2.5. If a pair of these normally closed valves were inadvertently left open, the leakage detection system would activate an alarm in the control room and allow the operators to take appropriate corrective actions. The capability of the leak detection system is summarized in the following table:

Location	Detection	Leakage Rate Within Activates Alarm
Drywell	Drywell Floor Drain Sump	5 gpm*
	Drywell Equipment Drain Sump	25 gpm
External to Drywell (Within Containment)	Containment Floor Drain Sump	5 gpm
	Containment Equipment Drain Sump	25 gpm
Outside of Containment (Aux. Building, Main Steam Tunnel, and Turbine Building)	Area Temperature	25 gpm
	Floor Drain (HPCS, RHR, RCIC, and LPCS Pump Rooms)	5 gpm

*The sensitivity within the drywell is 1 gpm within 1 hour. The 5 gpm is for activation of control room alarm.

ATTACHMENT NO. 6

DRAFT RESPONSES TO
EQUIPMENT QUALIFICATION BRANCH
QUESTIONS

271.02
(3.9)

a. The GESSAR II FSAR does not contain substantial discussion about the development of hydrodynamic loads for purposes of equipment qualification or how the loads are handled in the qualification. The limited discussion of this subject in Section 3.9 indicates that the hydrodynamic loads will be represented by response spectra. But no discussion is presented as to how the response spectra are developed or how the hydrodynamic loads are combined with seismic loads. If these loads are combined by performing an SRSS summation, the results may be less than conservative. Discuss more thoroughly the treatment of hydrodynamic loads, seismic loads, and their combination.

b. Persistently throughout Section 3.9 the statement is made that if equipment can be shown to have natural frequencies greater than 33 Hz, it can be considered rigid. This, of course, may not be true if the equipment is subjected to hydrodynamic loads which have a frequency content greater than 33 Hz. Where hydrodynamic loading is mentioned, a frequency of 60 Hz as the cutoff frequency should be provided since hydrodynamic loads often contain higher frequencies.

Response a

The development of hydrodynamic loads, including those for the purposes of equipment qualification, will be provided by the Applicant. The hydrodynamic loads which will be represented by response spectra will also be provided by the Applicant. Hydrodynamic and seismic loads are combined using the methodolgy of NUREG-0484 as described in NEDE-24326-1-P.

Response b

The 33 Hz criteria was not intended for hydrodynamic load combination. For these dynamic loads, a frequency of at least 60 Hz is used. GESSAR II will be revised as indicated (attached).

3.9.2.2.2.17 Other Nuclear Island ASME III Equipment

of interest at a rate no greater than one octave per minute. If no resonances are located, then the equipment is considered as rigid and single frequency tests at every 1/3 octave frequency interval are acceptable. Also, if all natural frequencies of the equipment are greater than 33 Hz for seismic loads and 60 Hz for hydrodynamic loads, the equipment may be considered rigid and analyzed statically as such. In this static analysis, the dynamic forces on each component were obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration at 33 Hz. The dynamic stresses were then added to the operating stresses and a determination made of the adequacy of the strength of the equipment. The search for the natural frequency is done analytically if the equipment shape can be defined mathematically and/or by prototype testing.

If the equipment is a rigid body while its support is flexible, the overall system can be modeled as a single-degree-of-freedom system consisting of a mass and a spring. The natural frequency of the system was computed; then the acceleration was determined from the floor response spectrum curve using the appropriate damping value. A static analysis was then performed using this acceleration value. In lieu of calculating the natural frequency, the peak acceleration from the spectrum curve was used. The critical damping values for welded steel structures from Table 3.7-1 were employed.

In case the equipment cannot be considered as a rigid body, it can be modeled as a multi-degree-of-freedom system. It is divided into a sufficient number of mass points to ensure adequate representation. The mathematical model can be analyzed using modal analysis technique or direct integration of the equations of motion. Specified structural damping is used in the analysis unless justification for other values can be provided. A stress analysis was performed using the appropriate inertial forces or equivalent static loads obtained from the dynamic analysis of each mode.

271.04
(3.10)

a. The GE position on fatigue effects due to hydrodynamic loading should be discussed. The argument for using only one OBE-intensity earthquake instead of five as stipulated in the IEEE 344-1975 Standard for seismic fatigue evaluation may be acceptable for certain plants. The use of only one OBE, however, should not be used on a generic basis. It should rather be justified for each plant where only one OBE is used.

b. The qualification program should address the degree of aging or environmental degradation that pieces of equipment could potentially incur prior to the occurrence of dynamic loading. The program should assure that the equipment has undergone its maximum expected amount of aging before the dynamic loads are applied in the qualification of the equipment. Surveillance and maintenance programs needed to assure that the equipment does not age to a degrees worse than qualified to should be described.

c. Sequential testing needs to be discussed more thoroughly. The discussion should make clear that seismic and hydrodynamic tests follow other environmental testing on the equipment. The sequence of exploratory, seismic and hydrodynamic loads and how this sequence properly qualifies the equipment for all loads incurred during the life of the equipment should be included in the discussion.

Response a

The GE method for handling fatigue effects due to hydrodynamic loading is presented in Section 4.4.2.5 of NEDE-24326-1-P.

The justification for using only one OBE instead of five for GESSAR II is given in the response to MEB (DSER) Item No. 38a. However, as described in Subsection 4.4.2.5.1(d) of NEDE-24326-1-P, 5 OBEs combined with the appropriate hydrodynamic loads are utilized for equipment qualification.

271.07
(3.9)

It is stated that if the equipment is a rigid body while its support is flexible, the overall system can be modeled as a single-degree-of-freedom system. A substantiation is required in order to classify the system as single-degree-of-freedom.

Response

If the equipment is a rigid body, it means that there is no amplification of the dynamic vibration. With the support being flexible, the amplification will come only from the spring action of the support. In this case it is sufficient to have a single-degree-of-freedom model to represent the overall system. Use of such a model will be substantiated on a case-to-case basis.

271.08
(1.8)

a. It is stated that closely spaced modes are combined by either the Double Sum Method or an algebraic sum of such modes. The Double Sum Method is acceptable according to Regulatory Guide 1.92, but an algebraic sum could be inappropriate. If the modes are added algebraically, cancellation would occur among modes having opposing signs. This cancellation could result in a non-conservative calculated total response. Justify the use of the algebraic sum method.

b. Two deviations from SRP 3.7.3 criteria are given. Justification for allowing these deviations should be provided.

Response a

The statement that closely spaced modes are combined by either the Double Sum Method or an algebraic sum of such modes is in error. As described in Subsection 3.7.3.7, only the Double Sum Method is employed. (As mentioned in Response b to this question, section 1.8 has been revised to address only regulatory guides. Thus, this "problem" statement, addressing SRP Section 3.7.2 has been eliminated.)