GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

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MFN 193-82
JNF 53-82

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

MEB DRAFT SER QUESTIONS/RESPONSES

Attached please find General Electric's responses to the Mechanical Engineering Branch Draft SER Questions. These responses address both the "formal" questions and additional questions brought up at the meeting requiring follow up action.

Essentially all questions are addressed in this transmittal. The remaining responses will be completed by January 14, 1983. An amendment is scheduled for January 1983 to formalize those responses requiring changes to GESSAR II.

Sincerely,

Willa for

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

GGS:td

Attachments

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

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GENERAL ELECTRIC'S RESPONSES

TO

NRC'S MEB QUESTION ON

CHAPTER 3 OF GESSAR II

Question 1

Table 3.2-1

Response

No response required, Question withdrawn.

Question 2

Clarification and justification of several of the classifications in this table are requested.

Response

No response required. Question withdrawn

Question

3. Why are Nuclear Boiler System air accumulator vessels Safety Class 3 and Quality Group C instead of Safety Class 2 and Quality Group B?

RESPONSE

The reasons for classifying the Nuclear Boiler System air accumulator (for ADS valve actuation) safety class 3 and quality group c are as follows;

1) The air accumulator is not a part of the reactor coolant pressure boundary and it will not create a safety hazard as defined for safety class 2 equipment if it doesfail.
2) The air accumulator function is similar to essential service water system which provides the service function for the essential equipment. The service water system is classified safety class 3, and by the same reasoning the air accumulator is classified safety class 3 and quality group c

Question 4

Several components are listed with two possible classifications for Safety Class and/or Quality Group. For example, for the RHR System the piping within outermost isolation valves has a Safety Class of "1, 2" and Quality Group of "A/B". In these instances where multiple classifications are listed, the table should be made more specific to indicate which component subset is associated with which Safety Class and Quality Group. In most instances where multiple classifications are listed, the second classification is lower than that previously presented by GE in other FSAR's. Justify these reductions in Safety Classes and Quality Groups.

Response

When mulitple classifications are used, refer to the applicable comment note in margin (a through x). Also note reference to note (a) is a typographical error, it should be note (g).

PREVIOUS MULTIPLE CLASSIFICATIONS ARE ALSO CORRECTED IN THE FOLLOWING REVISED table 3.2-1 PAGES

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Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

MEB
#4

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
II (Continued)						
12. Valves, other - isolation valves and within outermost isolation valves	1/2	C,D	A/B	B	I	~~(a)~~ (g)
13. Valves - instrumentation beyond outermost isolation valves	2/Other	A	B/D	B/N/A	I/N/A	~~(a)~~ (g)
14. Mechanical modules - instrumentation with safety function	2	C	N/A	B	I	
15. Electrical modules with safety function	2	C	N/A	B	I	(i)
16. Cable with safety function	2	C,D,A,X	N/A	B	I	
III Reactor Recirculation System						
1. Piping	1	D	A/B	B	I	(g)
2. Pipe suspension - recirculation line	1	D	A	B	I	
3. Pipe restraints - recirculation line	2	D	N/A	B	I	
4. Pumps	1	D	A	B	I	

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
VI (Continued)						
9. Electrical equipment and devices	~~2/3~~ 2	C	N/A	~~B/NA~~ B	~~I/NA~~ I	H&D #4
10. Cable with safety function	2	C,X,A	N/A	B	I	
VII Neutron Monitoring System						
1. Electrical modules - IPRM and APRM	2	A,D,R,X	N/A	B	I	
2. Cable - IPRM and APRM	2	C,X,A	N/A	B	I	
3. Detector and tube assembly	2	D	B	B	I	
VIII Remote Shutdown System						
This system is included under groups/MPL's, II/B21, XI/E12, XVI/E51, XXVI/G51, XXVII/H13 and XXXIV/P41.						
IX Reactor Protection System						
1. Electrical modules	2	C,T	N/A	B	I	
2. Cable	2	A,C,T,X	N/A	B	I	

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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
XXV Fuel Pool Cooling and Cleanup System						
1. Vessels - filter/demineralizers	Other	R	C	B	N/A	(r)
2. Vessels - other	3	R	C	B	I	
3. Heat exchangers	3	R	C	B	I	
4. Piping	3	R,C	C	B	~~I/N/A~~ I	MEB #4
5. Pumps and pump motors	3	R	C	B	I	
6. Valves and piping - containment isolation	2	C	B	B	I	
7. Makeup system	~~3/Other~~ 3	R,O,C	C	~~B,N/A~~ B	~~I/N/A~~ I	(r) MEB #6
8. RHR connections - emergency cooling	3	A,R	C	B	I	
9. Electrical modules and cables	3	R	C	B	I	
XXVI Suppression Pool Temperature Monitoring System						
1. Electrical modules with safety functions	3	C,X	N/A	B	I	
2. Cable with safety function	3	C,X	N/A	B	I	

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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
XXXI Standby Gas Treatment System						
1. Filters	2	R	N/A	B	I	
2. Valves - ductwork	2	A,R,C	N/A	B	I	
3. Cable with safety function	2	R,A,C,X	N/A	B	I	
4. Fans and motors	2	R	N/A	B	I	
XXXII NI Chilled Water Systems						
1. Control Building	3	D,C,A	C	B	I	
2. Electrical switch gear	3	A	C	B	I	
3. Other buildings	Other	A,R,W	D	N/A	N/A	
XXXIII HPCS Service Water System						
This system is included under group/MPL XXXIV/P41.						
XXXIV Essential Service Water System						
1. Piping	2,3	O,A,C	B/C	B	I	(g)
2. Pumps	3	P	C	B	I	
3. Pump motors	3	P	N/A	B	I	

MEB #4

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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
XXXVI (Continued)						
3. Condensate header - piping and valves	2	A	B	B	I	
4. Piping and valves	Other	O	D	N/A	N/A	
5. Other components	Other	O	D	N/A	N/A	
XXXVII Suppression Pool Makeup System						
1. Valves	2	C	B	B	I	
2. Piping	2	C	B	B	I	
XXXVIII Instrument and Service Air Systems						
1. Vessels, accumulators, supporting safety-related systems	3	A,T,C	C	B	I	
2. Piping in lines between accumulators (item 1) and safety-related systems	3	A,T,C	C	B	I	
3. Pneumatic control equipment	3	A,T,C	~~Special~~ C	B	I	~~(a, c)~~ MEB #4
4. Piping and valves forming part of containment boundary	2	A,C	B	B	I	

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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
XLIV Plant Electrical Systems						
(Applicant to Supply)						
XLV Auxiliary AC Power System						
1. All components with safety function	2/3	A,C,X	N/A	B	I	(9)
XLVI Diesel Generator Systems						
1. Day tanks	3	S	C	B	I	
2. Piping and valves - fuel oil system	3	S,O	C	B	I	
3. Piping and valves - diesel service - water system	3	S	C	B	I	
4. Pumps - fuel oil system	3	S	C	B	I	
5. Pumps - diesel service - water system	3	S	C	B	I	
6. Pump motors - fuel oil system and diesel service-water system	3	A,O	N/A	B	I	
7. Diesel generators	3	S	N/A	B	I	

MEB #4

#4

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

NOTES (Continued)

e. B = the equipment shall meet the quality assurance requirements of 10CFR50 Appendix B in accordance with the quality assurance program described in Chapter 17.

N/A = Quality Assurance Requirements not applicable to this equipment.

f. I = shall be constructed in accordance with the requirements of Seismic Category I structures and equipment as described in Section 3.7, Seismic Design.

N/A = The seismic requirements for the safe shutdown earthquake (SSE) are not applicable to the equipment.

Equipment that provides no safety function but which could damage Seismic Category I equipment if it failed is analytically checked and designed to confirm its integrity against collapse when subjected to seismic loading resulting from the SSE.

g. IN ALL INSTANCES, LISTED SYSTEMS/EQUIPMENT WILL BE OF THE HIGHER CLASSIFICATI[ON] WITH THE FOLLOWING EXCEPTIONS:

~~f.~~ 1. Lines one inch and smaller which are part of the reactor coolant pressure boundary shall be Code Class 2 and Seismic Category I.

MEB #4

2. All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate and monitor safety systems shall be Safety Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

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Question 5

For the Reactor Water Cleanup System, justify the designation of Safety Class 2 and Quality Group B for the containment piping penetrations and valves.

Response

The classification of this equipment is corrected as shown on the revised table 3.2-1 (pg 3.2-27) and agreed upon in the San Jose meeting.

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
XXIII (Continued)						
4. Piping and valves within outermost containment isolation valves on pump suction	1	D,C	A	B	I	(g)
5. Pump suction and discharge piping and valves from containment isolation valves back to containment isolation valves	Other	A	C	N/A	N/A	(g)
6. Containment piping ~~penetrations and valves~~ between isolation valves	~~2~~ 1	C,A	~~B~~ A	B	I	(g)
7. Filter/demineralizer and heat exchanger piping and valves inside containment	Other	C	C	N/A	N/A	(g)
8. Piping and valves returning from containment to RHR system	2	A	B	B	I	(g)
9. Piping and valves from containment to main condenser/radwaste	Other	A,T,W	C	N/A	N/A	(g)
10. Filter/demineralizer precoat subsystem	Other	C	D	N/A	N/A	
11. Nonregenerative heat exchanger shell and piping carrying closed cooling water	Other	C	D	N/A	N/A	

MEB #5

QUESTION 6

It is the staff's position that certain systems important not identified in Regulatory Guide 1.26 should be classified Quality Group C, or its equivalent. Among these systems are: diesel fuel oil storage and transfer system, diesel engine cooling water system, diesel engine lubrication system, diesel engine starting system, and diesel engine combustion air intake and exhaust system. Justify the absence of a quality group classification of portions of those systems listed below:

Diesel Generator Cooling Water System

Diesel Generator Starting System

Diesel Generator Lubrication System

Diesel Generator Combustion Air Intake and Exhaust System

RESPONSE

The fuel oil storage and transfer, cooling water, lubrication, starting air, combustion air intake and exhaust system are quality group C classification as shown in the attached marked-up Table 3.2-1 pages 3.2-38 and 3.2-39. Compliance is by the Applicant. The diesel generator including all integral components are per the engine manufacturers' and DEMA standards and therefore, quality group classification is not applicable. Revised item 7, pump motors is not under the jurisdiction of the ASME Section III Code and quality group classification is not applicable.

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
XLIV Plant Electrical Systems						
(Applicant to Supply)						
XLV Auxiliary AC Power System						
1. All components with safety function	2/3	A,C,X	N/A	B	I	
XLVI Diesel Generator Systems						
1. Fuel oil storage and transfer ~~Day tanks~~ system	3	S,O	C	B	I	(z)
2. Cooling water system ~~Piping and valves - fuel oil system~~	3	S/A	C	B	I	(z)
5~~3~~. Lubrication system ~~Piping and valves - diesel service - water system~~	3	S	C	B	I	(z)
6~~4~~. Combustion air intake and ~~Pumps - fuel oil system~~ exhaust system	3	S,O	C	B	I	
~~5. Pumps - diesel service - water system~~	~~3~~	~~S~~	~~C~~	~~B~~	~~I~~	
7~~6~~. Pump motors - fuel oil system, ~~and~~ diesel coolant ~~service~~-water system, and lube oil system	3	A,O	N/A	B	I	
8~~7~~. Diesel generators	3	S	N/A	B	I	(z)

MEL #6

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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
XLVI (Continued)						
9 ~~8~~. Electrical modules with safety functions	3	S	N/A	B	I	
10 ~~9~~. Cable with safety functions	3	S,X	N/A	B	I	
~~10. Diesel fuel storage tanks~~	~~3~~	~~O~~	~~C~~	~~B~~	~~I~~	
3 ~~11~~. Starting air tanks, receivers, piping from & including check valve and downstream piping and valves	3	S	C	B	I	(z)
4 ~~12~~. Starting air compressor and motors	Other	S	N/A	N/A	N/A	
XLVII Combustible Gas Control System						
1. Hydrogen Recombiner System	2	C	N/A	B	I	
2. Containment/Drywell Purge System	2	A,C,E	B	B	I	
3. Containment/Drywell Mixing System	2	D,C	B	B	I	
4. Fan	2	D	B	B	I	
5. Fan motor	2	D	B	B	I	
6. Electrical modules	2	D	B	B	I	

MEB

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QUESTION 7

Information should be included on the piping and instrument diagrams to delineate in detail the system quality group classification boundaries and the boundary limits of those system portions designed seismic Category I.

RESPONSE

All pipelines on the piping and instruments diagrams are identified with a system number and material class. Included in the material class code is the quality group classification (see GESSAR Figure 1.7-4, NI Piping and Instrumentation Flow Diagram Symbols). All quality groups A, B, and C are designed seismic Category 1.

QUESTION 8

Are the reactor internals designed to NG?

RESPONSE

The reactor core support structures are designed to NG as evidence in Table 3.2.2 and Section 3.9.5.3.6. The further response of the request related to the reactor internals is provided in revised section 3.9.5.3.5. The above responses meet the requirements of SRP Section 3.9.5.

3.9.5.3.5 Stress, Deformation, and Fatigue Limits for Engineered Safety Feature Reactor Internals (Except Core Support Structure) (Continued)

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

INSERT A

3.9.5.3.6 Stress and Fatigue Limits for Core Support Structures

The design and construction of ~~stress, fatigue, and other limits for~~ the core support structures are in accordance with ASME Code Section III, Subsection NG, and are summarized in Table 3.9-10(6).

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 55a(g), is readily feasible within the physical design of the Nuclear Island. Accessibility for this inservice inspection has been provided by attentiveness of design personnel to ISI requirements (e.g., space is available for convenient placement of ISI equipment, insulation is readily removable where required, and adequate lighting is provided). Appropriate pressure taps are provided around pumps. Position indicators are provided on valves.

Details of the inservice testing program, including test schedules and frequencies, are the responsibility of the Applicant. Also, any applications for written relief from Section XI Addendum requirements, pursuant to 10CFR50, Section 55a(g)(6)(i), shall be made by the Applicant.

¶ A

The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures ~~should~~ meet the guidelines of NG-3000 and ~~be~~ are constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

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Question 9

Why are the pipe restraints on the recirculator system safety Class 2?

Response

The safety class "other" is more correct for the classification of the pipe restraints as suggested by NRC's staff. The response to this request is provided in the revised Table 3.2.1 (page 3.2-15)

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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
II (Continued)						
12. Valves, other - isolation valves and within outermost isolation valves	1/2	C,D	A/B	B	I	(a)
13. Valves - instrumentation beyond outermost isolation valves	2/Other	A	B/D	B/N/A	I/N/A	(a)
14. Mechanical modules - instrumentation with safety function	2	C	N/A	B	I	
15. Electrical modules with safety function	2	C	N/A	B	I	(i)
16. Cable with safety function	2	C,D,A,X	N/A	B	I	
III Reactor Recirculation System						
1. Piping	1	D	A/B	B	I	(g)
2. Pipe suspension - recirculation line	1	D	A	B	I	
3. Pipe restraints - recirculation line	~~2~~ other	D	N/A	~~B~~ N/A	~~I~~ N/A	MEB #9
4. Pumps	1	D	A	B	I	

QUESTION 10

Explain the Seismic Category of the fuel service equipment? Control rod grapple?

RESPONSE

The control rod grapple appears in XIX, In-Vessel Service Equipment. This grapple and the general purpose grapple included in Item XVII are not safety related and are identified with "other" on the attached mark-up. Also see ASB Question #410.31 on same topic for futher information (attached). The control rod grapple is classified as seismic N/A because it is suspended from a cable, which will not transmit the seismic response to the grapple

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
XVIII Reactor Vessel Servicing Equipment						
1. Steamline plugs	Other	C	N/A	N/A	N/A	
2. Dryer and separator sling and head strongback	2	C	B	B	I	
XIX In-Vessel Servicing Equipment						
1. Control rod grapple	~~2~~ "other"	C,R	B	N/A	N/A	(l)
XX Refueling Equipment						
1. Refueling equipment platform assembly	2	C	N/A	B	I	
2. Refueling bellows	Other	D	N/A	N/A	N/A	
3. Fuel transfer system	2/Other	C,R	B,D,N/A	B	N/A	(n)
4. Penetration sleeve	2	C,R	B/D	B	I	(o)
XXI Storage Equipment						
1. Fuel storage racks	2	C,R	N/A	B	I	
2. Defective fuel storage container	3	R	N/A	B	I	

HGB #10

* LD

#10

C F BRAUN & CO

R W Christiansen

General Electric	GESSAR ROUND 1 QUESTION	Project 6382
San Jose		

QUESTION/RESPONSE 410.31 (9.1.5) #10

QUESTION 410.31

For the fuel servicing equipment and cranes listed in Table 3.2-1 (Table 9.1-2) of your FSAR which are characterized as non-seismic Category 1, verify that they are designed not to be a missile source as a result of a safe shutdown earthquake.

RESPONSE 410.31

The Fuel Prep Machine will be identified in Table 9.1-2 as Category 1 All other hoists, tools and equipment used for servicing shall 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. Subsection 9.1.4.2.3 and Table 3.2-1 are revised accordingly.

#10

9.1.4.2.3 Fuel Servicing Equipment

The fuel servicing equipment described below has been designed in accordance with the criteria listed in Table 9.1-2. ITEMS NOT LISTED AS SEISMIC CATEGORY I SUCH AS HOISTS, TOOLS AND OTHER EQUIPMENT USED FOR SERVICING SHALL 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. 410.31

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Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XVI (Continued)						
4. Pumps	2	A	B	B	I	
5. Pump motors	2	A	B	B	I	
6. Valves - outer isolation and within	1,2	D	A/B	B	I	(g)
7. Valves - return test line to condensate storage beyond second isolation valve and vacuum pump discharge line to containment isolation valves	Other	O,A	D	N/A	N/A	(g)
8. Valves - other	2	C,A	B	B	I	(g)
9. Turbine	2	A	N/A	B	I	(m)
10. Electrical modules with safety function	2	A,[illegible]	N/A	B	I	
11. Cable with safety function	2	C,A,X	N/A	B	I	
XVII Fuel Servicing Equipment						
1. Fuel preparations machine	Other	C,R	N/A	B	I	410.31
2. General purpose grapple	Other	C,R	N/A	B	N/A	

QUESTION 11

Explain the Quality Group Classification of the Hydrogen Recombiner System.

RESPONSE

The hydrogen recombiner system quality group classification is B as shown in the attached revised Table 3.2-1 in compliance with SRP 6.2.9 II(7).

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
XLVI (Continued)						
8. Electrical modules with safety functions	3	S	N/A	B	I	
9. Cable with safety functions	3	S,X	N/A	B	I	
10. Diesel fuel storage tanks	3	O	C	B	I	
11. Starting air tanks and downstream piping and valves	3	S	C	B	I	
12. Starting air compressor and motors	Other	S	N/A	N/A	N/A	
XLVII Combustible Gas Control System						
1. Hydrogen Recombiner System	2	C	~~N/A~~ B	B	I	MEB #11
2. Containment/Drywell Purge System	2	A,C,E	B	B	I	
3. Containment/Drywell Mixing System	2	D,C	B	B	I	
4. Fan	2	D	B	B	I	
5. Fan motor	2	D	B	B	I	
6. Electrical modules	2	D	B	B	I	

QUESTION 12

Justify the classification of the polars and cask cranes.

RESPONSE

The polar crane and the cask crane have been classified as Safety Class 3, Quality Assurance requirement B and seismically qualified to retain its functional capability and maintain normal performance under conditions of 1/2 SSE (OBE) and to maintain structural integrity under SSE conditions.

Since their use is not necessary to accomplish any of the safety functions which could occur during plant operation, no higher safety class is deemed necessary.

Safety Class 3 has been assigned to these cranes because of their location in the plant and safety systems they will lift, as well as possible failure causing collision with some safety related components.

Quality Assurance requirement B construction in accordance with the QA requirements of 10CFR50, Appendix B, is necessary because of the correlation of safety classes with the other design requirements as shown in GESSAR Table 3.2-3.

Note (X) on page 3.2-53 is being changed to read -

- The cranes are designed to retain functional capability under conditions of 1/2 SSE and to retain structural integrity under conditions of SSE.

- Table 3.2-1 page 3.2-40 and 3.2-41 will be reivsed to add Seismic Category I for these cranes. Mark ups are included in the attachment.

#12

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
XLVII (Continued)						
7. Cables	2	D	B	B	I	
8. Containment/Drywell Atmosphere Monitoring System	2	D,C	N/A	B	I	
XLVIII Fire Protection System						
1. Isolation valves and piping within outmost isolation valves	2	C	R	R	I	(t)
2. Other piping and valves	Other	A,R,C,X,S	D	B	I	(t)
3. Pumps	Other	O	D	B	I	(t)
4. Pump motors	Other	O	D	B	I	(t)
5. Electrical modules	Other	A,R,C,X,S	D	B	I	(t)
6. CO_2 actuation modules	3	S	N/A	B	I	(t)
7. Cables	Other	A,R,C,X,S	D	N/A	I	
8. Sprinkles	Other	A,R,C,X,S	D	N/A	N/A	
XLIX Miscellaneous Components						
1. Containment polar crane	3	C	N/A	B	I	(x)

MEB
#12

#12

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
XLIX (Continued)						
2. Cask crane	3	R	N/A	B	I	(x) MEB #12
L Emergency Lighting	Other	A,C,X	N/A	N/A	N/A	
LI Heating, Ventilating, and Air Conditioning Systems						
(Applicant to Supply)						
LII ECCS Equipment Area Cooling						
1. Fans, flow meters, ducting valves, heat exchanges, chilling units with safety function	3	A	N/A	B	I	
2. Motors	3	A	N/A	B	I	
3. Instrumentation and controls with safety function	3	A	N/A	B	I	
4. Electrical modules	3	A	N/A	B	I	
5. Cable	3	A	N/A	B	I	

Table 3.2-1

#12

EQUIPMENT CLASSIFICATION (Continued)

NOTES (Continued)

2. A certification shall be obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

w. The condensate storage tank will be designed, fabricated, and tested to meet the intent of API Standard API 650. In addition, the specification for this tank will require: (1) 100% surface examination of the side wall to bottom joint and (2) 100% volumetric examination of the side wall weld joints.

x. The cranes are designed to hold up their loads under conditions of 1/2 SSE and to retain structural integrity ~~maintain their positions over the units~~ under conditions of an SSE.

MEB #12

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QUESTION 13

What is the natural frequency of the charcoal absorber tanks, including the support elements? How are they affected by suppression pool loads?

RESPONSE

This tank is located in the turbine building and is not affected by new loads. (See COMMENT P for further detail).

QUESTION 14

What is TEMA.

RESPONSE

TEMA stands for Tube Exchanger Manufacture Association. The C next to TEMA stands for class C.

QUESTION 14a

Provide a list of all high energy lines.

RESPONSE

A list of all high energy lines are given in Table 3.6-6 and 3.6-7 for high energy piping inside containment and outside containment respectively.

QUESTION 15

Section 3.6.2.1.4.2

The design stress and fatigue limits for Class 1 piping in the containment penetration areas are not in compliance with Standard Review Plan 3.6.2 and BTP MEB 3-1. If the maximum stress range of Equation (10) exceeds 2.4 Sm, both Equation (12) and (13) must be Less than 2.4 Sm. The cumulative usage factor must be Less than 0.1 even for Equation (10) Less than 2.4 Sm.

RESPONSE

The response to this request is provided with the revised Section 3.6.2.14 to comply with SRP 3.6.2 and BTP MEB 3-1.

#15

3.6.2 1.4.2 Piping in Containment Penetration Areas (Continued)

The cumulative usage factor ~~should~~ must be less than 0.1

MEB #15

(b) ~~If the calculated maximum stress range of Equation 10 exceeds 2.4Sm but is not greater than 3.0Sm, then the cumulative usage factor is less than 0.1.~~

(c) If the calculated maximum stress range of Equation 10 exceeds ~~3.0~~ 2.4Sm, then the stress ranges calculated by both Equation 12 and 13 of NB-3653 do not exceed 2.4Sm. ~~In addition, the cumulative usage factor is less than 0.1.~~

(d) The maximum stress as calculated by Equation 9 of NB-3652 under the loadings resulting from a postulated piping failure beyond the required restraints does not exceed 2.25Sm. Higher stresses between the isolation valves and restraints were permitted provided a plastic hinge was not formed and operability of the valves with such stresses was assured.

(2) For ASME Code Section III Class 2 piping, the following stress and fatigue limits are not exceeded.

(a) The maximum stress ranges calculated by the sum of Equations 9 and 10 of NC-3652 for normal and upset plant conditions (including an operating basis earthquake) does not exceed 0.8 (1.2Sh + Sa).

(b) The maximum stress as calculated by Equation 9 of NC-3652 under the loadings resulting from a postulated piping failure beyond the required restraints does not exceed 1.8Sh. Higher stresses between the isolation valves and restraints were permitted provided a plastic hinge was not formed and operability of the valves with such stresses

to comply with SRP 3.6.2 and BTP MEB 3-1.

QUESTION 16

It is stated in the FSAR that welded attachments to the pipe are avoided except where detailed stress analysis or tests were performed to demonstrate compliance with the stress limits given. Provide a list of all instances where this exception was taken and welded attachments were used and the results of the analyses.

RESPONSE

In the containment penetration area between isolation valves, only pipe head fitting or containment nozzle is attached to the pipe to form an anchor (i.e. flued head). Both types of attachments are required to be analyzed by detailed analysis method and documented in an ASME Code Stress Report. [redacted]

Section 3.6.1.1.4.2 is Revised to reflect this.

also attached is the closed out LRG II issue on the same subject. This will also be GE's continued position for the GESSAR Flued Head Design

#16

3.6.2.1.4.2 Piping in Containment Penetration Areas (Continued)

was assured. When the piping beyond the isolation valve is constructed in accordance with ANSI B31.1, this exception may be applied provided the pipe is either of seamless construction with full radiography of all circumferential welds or all longitudinal and circumferential welds are fully radiographed.

(3) The piping runs are straight.

(4) Welded attachments for pipe supports or other purposes were ~~are~~ avoided unless the ~~except where~~ detailed stress analyses or tests were performed to demonstrate compliance with the stress limits given in items (1) and (2). MEB #16

(5) The number of circumferential and longitudinal piping welds and branch connections are minimized. Where guard pipes are used, the enclosed portions of piping are of seamless construction and have no circumferential welds unless specific provisions for access is made to permit 100% inservice volumetric examination of all welds.

(6) The length of these portions of piping are reduced to the minimum length practical.

(7) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) do not require welding directly to the outer surface of the piping (e.g., flued integrally-forged pipe fittings may be used) except where such welds are 100% volumetrically examinable in service and a

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#16

INSPECTABILITY OF WELDED FLUED HEAD DESIGN ON MAIN STEAM LINE CONTAINMENT PENETRATION

ISSUE:

The inspectability of welded flued head design on main steam line containment penetration should be demonstrated via the following activities:

1. Verify that the plant configuration allows adequate accessability to the penetration to perform necessary inspections.

2. Determine if the penetration weld was ultrasonically examined during manufacturing. If so, report on examination results.

3. Determine if additional details exist on the welded flued head inspectability demonstrations performed at the Associated Pipe and Engineering facility in 1976 and 1977 and documented in General Electric Company Topical Report NEDO-23652, "Analysis on General Electric Designed Welded Flued Head Fitting at Containment Penetration Assembly and Provisions for Nondestructive Examination of Flued Head Fitting to Process Pipe Weld for BWR/6 Mark III - 218, 238, 251 Plants".

LRG-II RESPONSE:

The inspectability of the main steam containment penetration for LRG-II plants with welded flued head design (i.e., Clinton and Perry) has been verified as follows:

1. Main steam line containment penetrations are readily accessible to allow the performance of inspection activities. Clearance around the penetrations is sufficient to permit the use of inspection equipment. Insulation on the main steam line was designed to be removable in the area of the welds.

#16

2. Main steam line flued head penetrations were ultrasonically and radiographically examined as part of their manufacturing process. These examinations revealed no indications requiring repair.

3. In July 1976, General Electric Company conducted the feasibility study of pulse-echo ultrasonic testing (UT) to assure full volume coverage of the flued head attachment weld. This UT examination technique was repeated as a demonstration for utility, architect-engineer and NRC representatives during July 1976 and May 1977 at the Associated Pipe and Engineering facility in Compton, California. The results of the demonstration are documented in the draft report NEDO-23652. No additional documentation on the demonstrations performed is available.

The main steam line penetration of River Bend Station are not of the welded flued head design; they are forged.

MJA:lm/13K-2
1/25/82

QUESTION 17

The maximum stress range as calculated by the sum of Equation (9) and (10) should consider sustained loads, occasional loads and thermal expansion. Have occasional loads been included as per BTP MEB 3-1?

RESPONSE

Yes, the occasional loads such as thrust from relief valve and safety valve, flow transient, and earthquake have been included as required by NC-3652 ASME Section III.

QUESTION 18

Whenever two or more intermediate break locations are not selected based upon the stress or usage factor limits, a total of two intermediate locations should be selected based upon highest stress. Provide assurances that highest stress was the criteria used to select these intermediate locations.

RESPONSE

For Class 1 piping pipe break analysis procedure requires all intermediate break locations to be postulated by highest stress method. The reasonable basis referenced in Section 3.6.2.1.4.3 will be reivsed to highest stress basis.

3.6.2.1.4.3 For ASME Code Section III Class 1 Piping

Breaks are postulated to occur at the terminal ends* of the piping run or branch run. In addition, breaks are assumed to occur at any intermediate location between terminal ends where:

(1) The maximum stress range between any two loads (including the zero load set) as calculated by Equation 10 of NB-3653 for normal and upset plant conditions (including an operating basis earthquake) exceeds 2.4Sm but is not greater than 3.0Sm and the cumulative usage factor is greater than 0.1.

(2) The calculated maximum stress range of Equation 10 exceeds 3.0Sm and the stress ranges calculated by either Equation 12 or 13 of NB-3653 exceed 2.4Sm or the cumulative usage factor exceeds 0.1.

In the event that two or more intermediate locations cannot be determined by stress or usage factor limits, a total of two intermediate locations are identified on a reasonable basis** for each piping run or branch run.

*Terminal ends are extremities of piping runs that connect to structures, components, or pipe anchors that are assumed in the piping stress analysis to act as rigid constraints to piping thermal expansion. A branch connection to a main piping run is a terminal end for a branch run except when the branch and main run is modeled as a common piping system during the piping stress analysis.

**Reasonable basis is ~~one or both of the following:~~ the highest stress location.
~~(1) fitting locations, and/or~~
~~(2) highest stress or usage factor locations.~~

MEB #18

Where more than two such intermediate locations are possible, using the application of the reasonable basis, those two locations possessing the greatest damage potential is used. A break at each end of a fitting may be classified as two discrete break locations where the stress analysis is sufficiently detailed to differentiate stress at each postulated break.

QUESTION 19

The paragraph labeled (1) indicates that breaks are postulated at ends that are lightly stressed. Please modify this paragraph to reflect a more reasonable criteria.

RESPONSE

(1) Breaks are postulated to occur at terminal ends. In addition, breaks are postulated at each location where the maximum stress ranges calculated by the sum of Equations 9 and 10 of NC-3652 for normal and upset plant conditions (including and operating basis earthquake) exceeds 0.8 (1.25SH+Sa). At least two intermediate break locations are postulated based on highest stress criteria with the following exceptions. Where the maximum stress ranges of a straight run without fittings, welded attachment, or valves does not exceed 0.8 (1.25Sh+Sa), at least one intermediate break location is postulated based on highest stress criteria.

MEB #19

3.6.2.1.4.6 Piping in the Steam Tunnel and Seismic Interface Restraint (Continued)

The criteria applied to the main steam, feedwater and branch piping outboard of the containment isolation valves are as follows:

In addition breaks are postulated at each location

WHERE the maximum stress ranges of a straight run without fittings, welded attachment, or valves does not exceed 0.8 (1.25 Sa + Sh) at least one intermediate break location is postulated based on highest stress criteria

(1) Breaks are postulated to occur at terminal ends where the maximum stress ranges calculated by the sum of Equations 9 and 10 of NC-3652 for normal and upset plant conditions (including an operating basis earthquake) does not exceed 0.8 (1.25h + Sa). At least ~~one~~ two intermediate break location is postulated based on highest stress criteria with the following exceptions.

(2) Breaks are postulated in high energy branch lines connected to the main steam and feedwater lines. All branch lines are restrained to the extent necessary to protect the containment isolation valves from any loadings, jet impingement, pipe whip and other forces that could impair valve structural integrity and operability.

The criteria used to evaluate the seismic interface restraint are as follows:

(1) Mechanistic pipe breaks are considered everywhere along the main steam and feedwater lines between the outboard containment isolation valve and the seismic interface restraint.

(2) Mechanistic pipe breaks are postulated at locations in the main steam, feedwater and branch line piping where stress limits and usage factor limits are exceeded. For such postulated mechanistic breaks, the associated dynamic effects are evaluated including jet impingement, pipe whip, flooding and pressurization.

QUESTION 20

No breaks are postulated at the RHR branch line connecting to the feedwater line since this branch line is included in the piping dynamic analysis for the feedwater line. What are the maximum stress ranges calculated by the sum of Equations (9) and (10) of NC-3652 for normal and upet plant conditions and what is the cumulative usage factor at the intersection?

RESPONSE

The paragraph labeled (4) of Section 3.6.2.1.4.6 will be labeled as (3) and moved to the section under the criteria applied to the main steam, feedwater and branch piping outboard of the containment isolation valves. It will also be revised to read as follows: breaks for RHR branch line are postulated according to the requirements of paragraph (1). Terminal end is not considered at the RHR branch line connecting to the feed water line because it is included in the piping dynamic analysis of the feed water line.

3.6.2.1.4.6 Piping in the Steam Tunnel and Seismic Interface Restraint (Continued)

(3) If application of steps (1) and (2) does not result in postulating a mechanistic pipe break, an intermediate (non-mechanistic) break is postulated. The steam tunnel is designed to withstand only the environmental effects of the intermediate (non-mechanistic) break (i.e., pressure and temperature).

moves on Pg 3.6-21a (middle) → MEB #20

(~~4~~ 3) No breaks are postulated at the RHR branch line connecting to the feedwater line since this branch line is included in the piping dynamic analysis for the feedwater line.

↑ REVISE AS PER RESPONSE 20.

3.6.2.1.5 Locations of Postulated Pipe Cracks

(1) Through-wall leakage cracks are postulated at those locations that demonstrate the adequacy of separation or other means of protections from required structures, systems, and components.

(2) Through-wall leakage cracks are postulated in moderate-energy fluid system piping located within structures and compartments containing essential systems and components (except where exempted by item (3) following). The cracks are postulated to occur individually at locations appropriate to form the basis for providing required protection from the hazards of environmental conditions from fluid spraying and flooding. See Section 3.4 for the internal flooding effects analysis.

40

QUESTION 21

Please provide a list of all instances where crack propagation times or full area break opening times in excess of one millisecond were used. Also list and justify any locations where less than full break areas have been assumed.

RESPONSE

The response to this request is provided in revised Section 3.6.2.2.1.

D4

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions (Continued)

The criteria that are used for calculation of fluid blowdown forcing functions include:

(1) Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections. ~~unless physically limited by piping restraints, structural members, or piping stiffness as demonstrated by the inelastic pipe whip analysis (Subsection 3.6.2.2.2).~~

MEB #21

(2) The dynamic force of the jet discharge at the break location is based on the ~~effective~~ cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically- or experimentally-determined thrust coefficient. ~~Limited pipe displacement at the break location.~~ Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.

(3) All breaks are assumed to attain full area instantaneously ~~A~~ rise time not exceeding one millisecond is used for the initial pulse. ~~unless longer crack propagation times or rupture opening times can be substantiated by experimental data or analytical theory.~~

Blowdown forcing functions are determined by either of the two following methods:

(1) The predicted blowdown forces on pipes fed by a pressure vessel are described by transient and steady-state

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QUESTION 22

On Page 3.6-27, Paragraph (b), clarify when reflections at bends and elbows will be assumed.

RESPONSE

No reflections at bends and elbows are assumed. Wave reflections will occur at the break end, and the pressure vessel until a standby flow condition is established. Vessel and free-space conditions are used as secondary conditions. The pipe is modeled as a straight, uniform piping fixed at one end and subjected to a time-dependent thrust force at the other end. The pipe-bending-moment-deflection (or rotation) relationship used for these location is obtained from a static nonlinear cantilever - beam analysis. Further clarification is also provided in revised Section 3.6.2.2.1.

#22

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions (Continued)

forcing functions. Simply, they are as follows:

(a) The transient forcing functions at points along the pipe result from the propagation of waves (wave thrust) along the pipe and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

(b) The waves cause various sections of the pipe to be loaded with time-dependent forces. It is assumed that the pipe is one-dimensional in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections will occur at the break end. ~~changes in direction of piping, and the pressure vessel until a steady flow condition is established.~~ Vessel and free-space conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break. MEB #22

(c) The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_o) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., 0.7 P_oA).

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QUESTION 23

A thrust coefficient of 0.7 was used which is less than 1.26 as specified in Standard Review Plan 3.6.2. Please provide justification for this lower value.

RESPONSE

The response to this request is provided in revised Section 3.6.2.2.1.

23

#22, 23

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions (Continued)

forcing functions. Simply, they are as follows:

(a) The transient forcing functions at points along the pipe result from the propagation of waves (wave thrust) along the pipe and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

(b) The waves cause various sections of the pipe to be loaded with time-dependent forces. It is assumed that the pipe is one-dimensional in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections will occur at the break end, ~~changes in direction of piping, and the pressure vessel until a steady flow condition is established.~~ Vessel and free-space conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the pipe break. MEB #22

(c) The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_o) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., 0.7 P_oA). This thrust coefficient of 0.7 is determined by the sample energy balance analysis with the force history had been previously calculated. Further, the sample energy balance represents a conceptual determination of reaction force (embedment load) at the restraint which is later verified by dynamic analysis. #23

5x

QUESTION 24

Page 3.6-36, paragraph (7). Standard Review Plan 3.6.2 does not allow the use of a jet expansion zone for saturated or subcooled water blowdown. Justify this discrepancy as evident in Figure 3.6-4.

RESPONSE

The response to this request is provided in revised Section 3.6.2.3.1.

#24

#24

3.6.2.3.1 Jet Impingement Analyses and Effects on Safety-Related Components (Continued)

(6) The jet impingement force is equal to the steady-state value of the fluid blowdown force calculated by the methods described in Subsection 3.6.2.2.1.

(7) The distance of jet travel is divided into two or three regions. Region 1 (Figure 3.6-4) extends from the break to the asymptotic area. Within this region the discharging fluid flashes and undergoes expansion from the break area pressure to the atmospheric pressure. In Region 2 the jet remains at a constant diameter. For partial-separation circumferential breaks, the area increases as the jet expands; therefore, it is assumed that Region 3 never occurs. In Region 3 (except partial-separation circumferential breaks), interaction with the surrounding environment is assumed to start and the jet expands at a half angle of 10°. (Dotted lines, Figures 3.6-4 a and c).

MEB
#24

(8) ~~Moody[6] has developed a simple~~ The analytical model for estimating the asymptotic jet area for ~~steam~~ Subcooled water and saturated water~~, and steam/water blowdown conditions~~ assumes a constant jet area. For fluids discharging from a break which are below the saturation temperature at the corresponding room pressure or have a pressure at the break area equal to the room pressure, the ~~intial~~ free expansion does not occur. ~~In these cases, the jet can be assumed to expand at a 10° half angle starting at the break area. (Dotted lines, Figures 3.6-4, a and c).~~

(9) The distance downstream from the break where the asymptotic area is reached (Region 1) has been found by Moody (for circumferential and longitudinal breaks) to be approximately equal to five pipe diameters. Assuming a linear expansion from the break area to the asymptotic

QUESTION 25

Pages 3.6-47 and 48. Provide assurances that in no instances does the value of strain exceed that associated with 50% of the ultimate uniform strain energy absorption capacity as determined by dynamic testing at loading rates within 50% of the specified design loading rate as required by Standard Review Plan 3.6.2.

RESPONSE

The response of this request is provided in the revised Section 3.6.2.3.3.
This response will eliminate the confusion of the specified design loading rate as required by SRP 3.6.2.

#25

3.6.2.3.3 Loading Combinations and Design Criteria for Pipe Whip Restraints (Continued)

uniform strain for all materials which will be used for Type I components.

Deleted
MEB
#25

~~3. Dynamic material mechanical properties - The material selected exhibits tensile impact properties which are not less than 70% of the static percent elongation or 80% of the statically determined minimum total energy absorption.~~

(2) Type II restraints (e.g., clevises, brackets, pins)

(a) Materials
Material selection conforms to:

1. ASTM Specifications including consideration for brittle fracture control, or

2. ASME Code Section III, Subsection NF, B&PV Code if applicable.

(b) Inspection
Inspection conforms to:

1. ASME/ASTM requirements or process qualification and finished part surface inspection per ASTM methods, or

2. ASME Code Section III, Subsection NF, B&PV Code if applicable.

QUESTION 26

Provide details of the seismic guides referred to in paragraph (5) on page 3.6-52.

RESPONSE

Detailed drawing of the seismic guides are provided in C. F. Braun Specification 400-09 and were reviewed by the NRC at the MEB draft SER meeting.

QUESTION 27

Standard Review Plan requires the review of sketches showing the locations of the postulated pipe ruptures, including identification of longitudinal and circumferential breaks, structural barriers, if any, restraint locations, and the constrained directions in each restraint. Also to be reviewed are the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative age factor, and the calculated primary plus secondary stress range. Provide these sketches and data in the FSAR.

RESPONSE (27+[illegible])

As noted in the 3/32/81 GESSAR FDA submittal letter (Sherwood to Denton), one of the NRC concerns regarding FDAs includes anti-trust considerations. The issue is the potential problem as whether the level of detail required by the staff during the FDA review will dictate the use of particular equipment vendors in CO/OL applications that reference the FDA. This is of primary concern particularly in the "buy" area which is mostly outside the NSSS. General Electric has elected to be responsive to anti-trust concerns by identifying specific quantities as "Applicant to Supply" for those quantities which are equipment vendor dependent.

Further, General Electric has also elected to have the hydrodynamic loads handled on a site specific basis. Thus, even if there anti-trust was not a issue, many of the equipment and piping quantities would not be available generally and these quantities would have to be supplied by the applicant.

In summary, the sketches and data requested by Question 27 and the information requested by Question 42 cannot be supplied at this time, and must be supplied by the Applicant in his FSAR. This will be included in Section 1.9 as an interface requirement.

QUESTION 28

Breaks in non-nuclear high energy piping not seismically analyzed (nor qualified) should be postulated at those locations which produce the greatest effect on an essential component or structure irrespective of the fact that the high stress or fitting criteria might not require that the break be postulated. Provide assurance that the above criteria has been met.

RESPONSE

THE PARAGRAPH 3.6.2.1.4.5 WILL BE REVISED AS FOLLOWS:

3.6.2.1.4.5 FOR ANSI B31.1 PIPING

BREAKS ARE POSTULATED AT THE FOLLOWING LOCATIONS IN EACH PIPE RUN OR BRANCH RUN:

(1) AT TERMINAL ENDS OF THE RUN IF LOCATED ADJACENT TO THE PROTECTIVE STRUCTURE

(2) AT EACH INTERMEDIATE PIPE FITTING, WELDED ATTACHMENT, AND VALVE IF THE PIPING IS NOT SEISMICALLY ANALYZED.

(3) FOR SEISMICALLY ANALYZED PIPING, INTERMEDIATE BREAKS ARE POSTULATED BASED ON THE SAME CRITERIA SPECIFIED IN PARAGRAPH 3.6.2.1.4.4.

CHANGE THE EXISTING PARAGRAPH 3.6.2.1.4.6 TO A NEW NUMBER 3.6.2.1.4.7.

ADD A NEW 3.6.2.1.4.6 PARAGRAPH

3.6.2.1.4.6 APPLICABLE TO 3.6.2.1.4.3, 3.6.2.1.4.4, AND 3.6.2.1.4.5.

3.6.2.1.4.4 For ASME Code Section III Class 2 and 3 Piping

Breaks are postulated to occur at terminal ends of the piping run or branch run. In addition, breaks are assumed to occur at any intermediate location between terminal ends where the maximum stress ranges calculated by the sum of Equations 9 and 10 of NC/ND-3652 for normal and upset plant conditons (including an operating basis earthquake) exceeds 0.8 (12Sh + SA).

If two or more intermediate locations cannot be determined by stress limits, a total of two intermediate locations are identified on a reasonable basis for each piping run or branch run.

3.6.2.1.4.5 For ANSI B31.1 Piping

Replace with insert (A)

~~Breaks are postulated to occur at terminal ends of the piping run or branch run. In addition, breaks are assumed to occur at each intermediate pipe fitting, weld attachment, and valve.~~

MEB #28

INSERT (A)

3.6.2.1.4.~~6~~ 7 Piping in the Steam Tunnel and Seismic Interface Restraint

As described in Subsection 3.8.4.1.2, a seismic interface restraint is provided for both the feedwater and main steamlines. The restraint is located in the steam tunnel between the shutoff valve (in the main steam and feedwater lines) and the Auxiliary Building/ Turbine Building interface (Figure 1.2-9). The seismic interface restraint serves as a guide for the main steam and feedwater lines and provides two-dimensional lateral restraint and moment-carrying capability. The seismic interface restraint provides a seismic category transitional point between the Seismic Category I steam and feedwater piping in the Auxiliary Building and the portions of these pipes in the Turbine Building which are not designed to Seismic Category I (see Figure 3.2-1).

INSERT (A) TO page 3.6-21. MEB #28.

Breaks are postulated at the following locations in each pipe run or branch run:

(1) At terminal ends of the run if located adjacent to the protective structure.

(2) At each intermediate pipe fitting, welded attachment, and valve if the piping is not seismically analyzed.

(3) For seismically analysed piping, intermediate breaks are postulated based on the same criteria specified in paragraph 3.6.2.1.4.4.

3.6.2.1.4.6 Separating structure with High Energy Lines

~~Paragraph 3.6.2.1.4.6 is applicable to 3.6.2.1.4.3, 3.6.2.1.4.4 and 3.6.2.1.4.5~~

If a structure separates a high energy line from an essential component, ~~that~~ the separating structure ~~should~~ shall be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure irrespective of the fact that the ~~above~~ aforementioned criteria (3.6.2.1.4.3-5) might not require such a break location to be postulated.

3.6.1.3.2.2 Separation (Continued)

requirements. ~~No~~ damage was assumed to occur due to jet impingement ~~since~~ the impingement force becomes negligible beyond 30 feet. ~~[illegible]~~ LIKEWISE A 30 FOOT EVALUATION ZONE WAS ESTABLISHED FOR PIPE BREAK TO ASSURE AGAINST DAMAGE FROM A WHIPPING PIPE. ASSURANCE THAT 30 FEET REPRESENTS THE MAXIMUM FREE LENGTH WILL BE DONE AT TIME OF SPECIFIC PLANT SUBMITTAL

MEB #29

(3) Essential systems, components, and equipment at a distance less than 30 feet from any high-energy piping were evaluated to see if damage could occur to more than one essential division, preventing safe shutdown of the plant. If damage occurred to only one division of a redundant system, the requirement for redundant separation was met. Other redundant divisions are available for safe shutdown of the plant and no further evaluation was performed.

(4) If damage could occur to more than one division of a redundant essential system within 30 ft of any high energy piping, other protection in the form of barriers, shields, or embedments was used. These method of protection are discussed in Subsection 3.6.1.3.2.3.

Due to the complexities of several divisions being adjacent to high-energy lines in the drywell and steam tunnel, the requirements for separation could not be evaluated using these simplifying assumptions. For these areas, specific break locations were determined in accordance with Paragraph 3.6.2.1.4.3. If spatial separation requirements (distance and/or arrangement to prevent damage) were not met based on the evaluation of specific breaks, barriers, enclosures, shields, or restraints was necessary. These methods of protection are discussed in Subsections 3.6.1.3.2.3 and 3.6.1.3.2.4.

3.6.2.3.2.2 Pipe Displacement Effects on Safety-Related Structures, Other Systems, and Components (Continued)

pipe. Otherwise, the impacted pipe is assumed to be ruptured.

(3) If the whipping pipe impacts other components (valve actuators, cable trays, conduits, etc.), it is assumed that the impacted component is unavailable to mitigate the consequences of the pipe break event.

3.1

Add (4)

INSERT (A)

MEB #29, #30

3.6.2.3.3 Loading Combinations and Design Criteria for Pipe Whip Restraints

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low-probability gross failure in a piping system carrying high-energy fluid. The piping integrity does not usually depend on the pipe whip restraints for any loading combination. When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) will be subjected to once-in-a-lifetime loading. For the purpose of design, the pipe break is considered to be a faulted plant condition and the pipe, its restraints, and structure to which the restraint is attached are analyzed and designed accordingly.

The pipe whip restraints utilize energy absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is shown in Figure 3.6-7. The principle feature of these restraints is that they are installed with several inches of annular clearance between them and the process pipe. This allows for installation of normal piping insulation and unrestricted pipe thermal movements. Select critical locations inside primary containment are also monitored during hot functional testing to provide verification of adequate clearances prior to plant operation.

MEB #29, 30

INSERT (A) To page 3.6-xx

(4) Damages of unrestrained whipping pipe on safety-related structures, components, and systems other than the ruptured one shall be prevented by either separating of high energy systems from essential systems or providing pipe whip restraints.

QUESTION 29

Discuss how pipe whip and jet impingement effects were determined for those postulated breaks in high energy piping that are not restrained (unrestrained whipping).

RESPONSE

For breaks in unrestrained high energy piping, barriers will be designed to withstand pipe whip and or jet impingement.

We will revise Section 3.6.2.3.2.2 and Section 3G.1 in Appendix 3G as shown attached.

APPENDIX 3G

PIPE FAILURE ANALYSES

3G.1 GENERAL

This appendix describes the specific high-energy pipe-failure protection criteria from pipe whip or jet impingement provided to satisfy the requirements of Subsection 3.6.1. It demonstrates that the essential systems, components, and equipment are not adversely affected by pipe breaks to an extent that would impair the ability to shut down the plant or mitigate the consequences of the postulated pipe failure. MEB #29

The effects of spraying and flooding as a result of breaks or cracks are discussed in Subsection 3.4.1.1.

For a detailed discussion of break and crack locations and types, break exclusion areas (no break zones), guard pipes, and whip restraints, refer to Subsection 3.6.2.

QUESTION 30

Provide assurance that the tip deflection of a restrained whipping pipe will not adversely affect nearby safety-related components from performing their safety-related function.

RESPONSE

~~Response later.~~

The response to this request is provided in the revised section 3.6.2.3.22 given in the Question #29.

QUESTION 31

Provide the loads, load combinations, and stress limits that were used in the design of pipe rupture restraints. Include a discussion of the desing methods applicable to the auxiliary steel used to support the pipe rupture restraint. Provide assurance that the pipe rupture restraint and supporting structure cannot fail during a seismic event.

RESPONSE

The loads, load combinations and stress limits for the design of pipe rupture restraint are covered in paragraph 3.6.2.3.3. The design method for structural steel used to support pipe rupture restraing is covered in paragraph 3.8.4.3.2.1 and 3.8.4.3.2.3. The pipe rupture restraint and supporting structures are designed to withstand the combination of pipe rupture and seismic loads.

QUESTION 32

Provide the design criteria used for pipe rupture restraints that also support piping.

RESPONSE

No pipe rupture restraint is used to support piping except for SIRS which is designed to AISC Code.

The governing design loads for the SIRS is pipe rupture loads, or pipe support loads are negligible.

>2

QUESTION 33

Provide the basis for assuring that the feedwater isolation check valves can perform its function following a postulated pipe break of the feedwater line outside containment.

RESPONSE

The design criteria for the feedwater pipelines in the steam tunnel assures that the structural integrity is maintained. The maximum stress occurring between the two isolation check valves is limited to 2.25 Sm. The restraint for a postulated pipe break outside containment is located so that a plastic hinge is not produced, assuring the desired action of the check valve can occur. The design specificiation identifies these checks as isolation valves and requires the supplier to provide operability assurance to be verified by testing. These valves are specified to be selected for non-slam characteristics. The leak integrity of the check valves shall be demonstrated by test on analysis under most sever operating transient conditions. The valve specification requires the feedwater check valves to be ANSI Class 900, tilting disc.

The tilting disc check valve is designed to close as quickly as possible to minimize the slamming that is caused when the high velocity reverse flow is allowed to build up before the completion of closing. The disc begins to close before the flow reverses and has completed the closing before the velocity has built up to a dangerous level.

QUESTION 34

3.7.3.5.1 - What is the significance of this section?

RESPONSE

The static analysis method to be used is provided in Subsection 3.7.3.8.1.5 as indicated on revised page 3.7-40. The remainder of Subsection 3.7.3.5.1 is deleted. Figure 3.7-31 is also deleted.

QUESTION 35

3.7.3.8.1.7 - Provide a list of all instances where the first footnote applies.

RESPONSE

GESSAR II presents criteria to be used, actual analysis is to be presented by applicant. Therefore the list of piping subsystems for which the footnote applies is to be provided by applicant. The footnote has been revised to reflect that it is to be criteria, rather than imply that it has already been used.

MEB #35

3.7.3.8.1.7 Damping Ratio (Continued)

Component	Damping Ratio Operating Basis Earthquake	Safe Shutdown Earthquake (Percentage)
Large* diameter piping systems (pipe diameter greater than 12 in.)	2	2**
Small diameter piping systems (diameter equal to or less than 12 in.)	1	2

3.7.3.8.1.8 Effect of Differential Building Movements

In most cases, piping subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event. The movements may range from insignificant differential displacements between rigid walls of a common building at low elevations to relatively large displacements between separate buildings at a high seismicity site.

Differential endpoint or restraint deflections cause forces and moments to be induced into the piping system. The stress thus

*If the piping subsystem consists of only one or two spans with little structural damping, use values for small diameter piping. ~~are used.~~

MEB #35

**Conservatively changed from 3% to 2%, since 3% damping response curves are not available.

81

QUESTION 36

The number of bolt up and unbolt events is listed in Table 3.9-1 as 40 each. This is a reduction from about 120 listed in past FSARs by GE. Explain the reason for this reduction in the number of cycles considered for these events.

RESPONSE

This quotation was withdrawn and GE agreed to clarify it with revised notes on Table 3.9-1.

22

Table 3.9-1 #36

PLANT EVENTS

A. Plant Operating Events

		Operating Condition[9]	No. of Events
1.	Bolt Up[1]	Normal	40 10 MEB #36
2.	Design Hydrostatic Test	Testing	40
	a. Leak Checks at 400 psig prior to power operation, 3 cycles/startup		
3.	Startup (100°F/hr Heatup Rate)[2]	Normal	260
4.	Daily Reduction to 75% Power[1]	Normal	10,000
5.	Weekly Reduction 50% Power[1]	Normal	2,000
6.	Control Rod Pattern Change[1]	Normal	400
7.	Loss of Feedwater Heaters	Upset	80
8.	Scram:		
	a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open, and Other Scrams	Upset	220
	b. Loss of Feedwater Pumps, Isolation Valves Closed	Upset	28
	c. Turbine Bypass, Single Safety or Relief Valve Blowdown	Upset	8
9.	Reduction to 0% Power, Hot Standby, Shutdown 100°F/hr Cooldown Rate)[2]	Upset	252
10.	Unbolt[1]	Normal	40 10 MEB #36
11.	Scram:		
	a. Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open	Emergency	1[3]
	b. Automatic Blowdown	Emergency	1[3]

83

#35

Table 3.9-1

PLANT EVENTS (Continued)

NOTES:

1. Applies to reactor pressure vessel only.
2. Bulk average vessel coolant temperature change in any 1-hour period.
3. The annual encounter probability of a single event is $<10^{-2}$ for an emergency event, and $<10^{-4}$ for a faulted event.
4. One OBE event includes 10 maximum load or stress cycles.
5. One stress or load reversal cycle of maximum amplitude.
6. Applicable to main steam piping system only.
7. The number of structural feedback vibratory load cycles on the reactor vessel and internal components is 13,200 cycles of varying amplitude during the 220 events of safety/relief valve actuation. The main steam and recirculation piping system use 660 full range cycles and 880 half range cycles, which are comparable in effect to 13,200 cycles of varying magnitude. The main steam piping system uses 5460 cycles to include additional effects of acoustic wave propagation in steam during the actuations.
8. Table 3.9-2 shows the evaluation basis combinations of these dynamic loadings.
9. The ASME Code Section II service limits of Levels A, B, C, D, or testing apply to these normal, upset, emergency, faulted, and testing operating conditions, respectively.
10. The field experience and the improvement on the maintenance have resulted in substantial reduction in no. of event from ~~about~~ 120.
Approximately

MEB #36

84

QUESTION 37

Verify whether the alternate shutdown cooling mode has been considered in the design of the SRV discharging piping. Specifically, address the capability of the spring hangers to accommodate the additional weight of water.

RESPONSE

The response to this request is provided in revised Section 5.2.2.4.1 (see attached page 5.2-19 for proper insert).

5.2.2.4.1 Description (Continued)

Category I. SRV discharge line piping from the SRV to the suppression pool consists of two parts. The first is attached at one end to the safety/relief valve and at its other end to a pipe anchor; and the main steam piping, including this portion of the safety/relief valve discharge piping, is analyzed as a complete system. The second part of the SRV discharge piping extends from the anchor to the suppression pool. Because of the upstream anchor on this part of the line, it is physically decoupled from the main steam header and is therefore analyzed as a separate piping system.

As a part of the preoperational and startup testing of the main steamlines, movement of the safety/relief valve discharge lines will be monitored.

INSERT
A
MEB
#37

The SRV discharge piping is designed to limit valve outlet pressure to approximately 40 percent of maximum valve inlet pressure with the valve wide open. Water in the line more than a few feet above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, two vacuum relief valves are provided on each SRV discharge line to prevent drawing an excessive amount of water into the line as a result of steam condensation following termination of relief operation. The safety/relief valves are located on the main steamline piping rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steamlines are more accessible during a shutdown for valve maintenance.

The nuclear pressure-relief system automatically depressurizes the nuclear system sufficiently to permit the LPCI and LPCS

INSERT (A) TO page 5.2-19 MEB #37

The effect of the alternate shutdown cooling mode on SRV discharge piping shall be considered. Specifically, the resultant load distribution must be within the design capacity of the spring hangers and other support structures.

QUESTION 38 +38A

Standard Review Plan 3.9.1 requires that for each transient loading condition or combination, an acceptable ASME Code service limit has been specified; i.e., Design, Level A, Level B, Level C, or Level D. Update Table 3.9-1 to show conformance with these service limits.

RESPONSE

The response of this request is provided in the revised Table 3.9-1. The response to GE's position on the 10 cycles/events (38a) used in OBE (item 15 of Table 3.9-1) is given in the attached reference documents

38(a) Issue:

Provide the justification of 10 cycles/event on OBE of Table 3.9-1 item 15.

Response:

The justification of the 10 cycles/event is given in the attached documents. Basically, the GE fatigue evaluation of RPV and internals with 10 peak OBE cycles is conservative and appropriate. (As per Dosnall letter of 2/18/82)

84

MEB
38

Table 3.9-1

PLANT EVENTS

A. Plant Operating Events

		ASME Code Service Limit Operating Condition[9]	No. of Events
1.	Bolt Up[1]	~~Normal~~ A	40
2.	Design Hydrostatic Test	Testing	40
	a. Leak Checks at 400 psig prior to power operation, 3 cycles/startup		
3.	Startup (100°F/hr Heatup Rate)[2]	~~Normal~~ A	260
4.	Daily Reduction to 75% Power[1]	~~Normal~~ A	10,000
5.	Weekly Reduction 50% Power[1]	~~Normal~~ A	2,000
6.	Control Rod Pattern Change[1]	~~Normal~~ A	400
7.	Loss of Feedwater Heaters	~~Upset~~ B	80
8.	Scram:		
	a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open, and Other Scrams	~~Upset~~ B	220
	b. Loss of Feedwater Pumps, Isolation Valves Closed	~~Upset~~ B	28
	c. Turbine Bypass, Single Safety or Relief Valve Blowdown	~~Upset~~ B	8
9.	Reduction to 0% Power, Hot Standby, Shutdown 100°F/hr Cooldown Rate)[2]	~~Upset~~ B	252
10.	Unbolt[1]	~~Normal~~ A	40
11.	Scram:		
	a. Reactor Overpressure with Delayed Scram; Feedwater Stays On, Isolation Valves Stay Open	~~Emergency~~ C	1[3]
	b. Automatic Blowdown	~~Emergency~~ C	1[3]

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MEB #38

Table 3.9-1

PLANT EVENTS (Continued)

		ASME CODE SERVICE LIMIT Operating Condition[9]	No. of Events
12.	Improper Start of Cold Recirculation Loop	~~Emergency~~ C	1[3]
13.	Sudden Start of Pump in Cold Recirculation Loop	~~Emergency~~ C	1[3]
14.	Improper Pump Startup after Hot Standby with Reactor Drain Shut off	~~Emergency~~ C	1[3]

B. Dynamic Loading Events[8]

		Operating Condition[9]	Cycles/ Events
15.	Operating Basis Earthquake (OBE)[4] at Event 8a Pressure Peak	~~Upset~~ B	10 Cycles
16.	Safe Shutdown Earthquake (SSE)[5] at Rated Power Operating Conditions	~~Faulted~~ D	1[3] Cycle
17.	Turbine Stop Valve Closure (TSV)[6] During Event 8a	~~Upset~~ B	660 Cycles
18.	Safety Relief Valve Actuation (One, All or Automatic Depressurization System) During Event 8a	~~Upset~~ B	220 Events[7]
19.	Pipe Rupture Accident		
	Small Break Accident	~~Faulted~~ D	1[3]
	Intermediate Break Accident	~~Faulted~~ D	1[3]
	Large Break Accident	~~Faulted~~ D	1[3]

Table 3.9-1

PLANT EVENTS (Continued) MEB #38

NOTES:

1. Applies to reactor pressure vessel only.

2. Bulk average vessel coolant temperature change in any 1-hour period.

3. The annual encounter probability of a single event is $<10^{-2}$ for an ~~emergency~~ Level C event, and $<10^{-4}$ for a ~~faulted~~ Level D event.

4. One OBE event includes 10 maximum load or stress cycles.

5. One stress or load reversal cycle of maximum amplitude.

6. Applicable to main steam piping system only.

7. The number of structural feedback vibratory load cycles on the reactor vessel and internal components is 13,200 cycles of varying amplitude during the 220 events of safety/relief valve actuation. The main steam and recirculation piping system use 660 full range cycles and 880 half range cycles, which are comparable in effect to 13,200 cycles of varying magnitude. The main steam piping system uses 5460 cycles to include additional effects of acoustic wave propagation in steam during the actuations.

8. Table 3.9-2 shows the evaluation basis combinations of these dynamic loadings.

9. The ASME Code Section II service limits of Levels A, B, C, D, or testing apply to these normal, upset, emergency, faulted, and testing operating conditions, respectively.

91

38 a)

Memo from: #38

PAUL C. YIN
EXT. 51439

NUCLEAR ENERGY BUSINESS GROUP
GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA

Ruby Villa

Nov 24, 1982

MEB #38

Subject: OBE Cycle, GESSAR MEB-SER

I called Jim Brammer of NRC-MEB on Monday to explore what additional information he would need in order to close this issue. I also explained to him that Bob Bosnak's approval of GE's 10-cycle position, dated 2/13/82, is generically applicable to all BWR 6's including GESSAR. At 11 am today, Brammer called me back and assured me this issue is closed. He requests that we properly reference the above approval letter in the GESSAR transmittal.

CC: JL Embley
BA Kalmikov

PC Yin

GENERAL ELECTRIC

3-MEB

38 a)

MEB #3B(a)

OBE STRESS CYCLES USED FOR NSSS MECHANICAL EQUIPMENT DESIGN

ISSUE:

The RPV and internals fatigue evaluation is based on 10 peak Operating Basis Earthquakes (OBE) cycles. The Standard Review Plan (SRP) requires the evaluation to include contributions from 5 OBE's with 10 cycles each. Justify this deviation from the SRP.

LRG II RESPONSE:

References:
1) Letter, R. Bosnak (NRC) to R. Artigas (GE), subject, "Number of OBE Fatigue Cycles in BWR NSSS Design, dated December 2, 1981
2) Letter, R. Artigas (GE) to R. Bosnak (NRC), same subject, dated December 3, 1981

The fatigue contribution of the OBE stress cycles, to the total fatigue usage, is small. Consideration of more than 10 peak accumulated cycles would not significantly reduce the margin. This is documented in Reference 2 in response to an NRC request in Reference 1.

In addition, GE has performed studies, as documented in Reference 2, that show a plant would expect to experience less than 10 equivalent peak OBE cycles during its lifetime. Therefore, the GE fatigue evaluation of the RPV and internals with 10 peak OBE cycles is conservative and appropriate.

MJA:rf:csc/118A24

023
38 a)
MEB #3 (a)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

RECEIVED
FEB 2 3 1982
R. ARTIGAS

February 18, 1982

Mr. R. Artigas, Manager
BWR Projects Licensing
Nuclear Safety & Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

References: 1) Letter from R. Artigas to R. Bosnak dated September 17, 1981.
2) Letter from R. Bosnak to R. Artigas dated December 2, 1981.
3) Letter from R. Artigas to R. Bosnak dated December 3, 1981.

Dear Mr. Artigas:

In Reference 1, General Electric documented its presentation given to the NRC at the GE Bethesda offices on September 15, 1981. The GE presentation provided the basis for using 10 peak OBE cycles for the seismic fatigue design of BWR NSSS components. In Reference 2, we responded to the GE presentation and September 17, 1981 letter and provided a justifiable framework to resolve the use of 10 OBE cycles for all BWR plants undergoing OL review. Subsequently, in Reference 3, GE provided 1) a response spectra comparison and 2) plant-specific results which show the OBE fatigue contribution to the total cumulative usage factor. The results indicate that even for the "worst case" component, the fatigue contribution due to the OBE is almost negligible.

Results were not provided for BWR/4 plants (i.e. Limerick and Hope Creek) as indicated in your summary table. We will require that the plant-specific results for the BWR/4 plants undergoing OL review be provided to the staff when the fatigue calculations are completed.

For BWR/5 plants, GE provided fatigue results for the WNP-2 plant which is stated to have the highest usage factor amongst the BWR/5 plants. Based on our review of the WNP-2 fatigue results and of the OBE as currently defined in the WNP-2 FSAR, we conclude that the 10 peak OBE cycles used in the fatigue design of the WNP-2 plant NSSS components is acceptable.

For BWR/6 plants, it is our understanding that a generic BWR/6 component fatigue design was performed using bounding values for the 10 peak OBE cycles. These bounding values were based on a conservative OBE and typical plant-specific OBE values are much less than the OBE assumed in the bounding generic design. In addition, we understand that for each BWR/6 plant, analyses of the RPV components are performed using plant-specific data to confirm the adequacy of the generic design. Thus, based on the above understanding and on our review of the fatigue results for the generic BWR/6 design, we conclude

9x

33 a)

that the use of 10 bounding design OBE peak cycles is acceptable for the BWR/6 generic fatigue design of NSSS components.

We will include our evaluation and findings, regarding the use of 10 peak OBE cycles for fatigue design, in the Safety Evaluation Report of each BWR plant undergoing OL review.

R Bosnak

Robert J. Bosnak, Chief
Mechanical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

QUESTION 39

NUREG-0800 requires that computer programs used in analyses of seismic Category I code and non-code items have the following information provided to demonstrate their applicability and validity:

a. The author, source, dated version and facility

b. A description and the extent and limitations of its application.

c. Solutions to a series of test problems which shall be demonstrated to be substantially similar to solutions obtained from any one of sources 1 through 4, and source 5:

   1. Hand calculations
   2. Analytical results published in the literature
   3. Acceptable experimental tests
   4. By A MEB acceptable similar program
   5. The benchmark problems prescribed in Report NUREG/CR-1677, "Piping Benchmark Problems"

Please demonstrate compliance with these requirements and provide summary comparisons for the computer programs used in seismic Category I analyses.

RESPONSE

The response to this request are provided in the revised Section 3.9.1.2.

#39

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The plant events affecting the mechanical systems, components and equipment are summarized in Table 3.9-1 in two groups: (1) plant operating events during which thermal-hydraulic transients occur, and (2) dynamic loading events due to accidents and earthquakes during certain operating conditions. The number of cycles for each event are listed. The plant operating conditions are identified as normal, upset, emergency, faulted, or testing as defined in Subsection 3.9.3.1.1. Appropriate Service Levels (A, B, C, D or testing) as defined in the ASME Code, Section III are designated for design limits. The design and analysis of safety-related piping and equipment using specific applicable thermal-hydraulic transients which are derived from the system behavior during the events listed in Table 3.9-1 are documented in the design specification and/or stress report of the respective equipment. Table 3.9-2 shows the loading combinations and acceptance criteria.

3.9.1.2 Computer Programs Used in Analyses

The following sections discuss computer programs used in the analysis of all the major safety-related components. Computer programs were not used in the analysis of all components, thus, not all components are listed (e.g., main steam isolation and safety/relief valves and recirculation gate valves).

The GE computer programs are maintained either by General Electric or by outside computer program developers. In either case, the quality of the programs and the computed results are controlled.

PP. The applicant will update the below listed programs to indicate the method of verification and the version used. These programs can be divided into four categories:

INSERT (A)

MEB #39

MEB #39

3.9.1.2 Computer Programs Used in Analyses (Continued)

INSERT (A)

Deleted.

~~For each program, one or more engineers are assigned. Duties are to:~~

~~(1) keep abreast of the capability, the software contents, and the theory of the program;~~

~~(2) run test cases and maintain the reliability of the program; and~~

~~(3) advise users on the proper usage of the program and the correct interpretation of computed results.~~

~~All necessary modifications are coordinated and verified by the responsible engineers; thus, user confusion over the changes is avoided and the reliability of these programs is maintained.~~

3.9.1.2.1 Reactor Pressure Vessel and Internals

The computer programs used in the preparation of the stress report for the pressure vessel (RPV) and internals stress report are identified.

3.9.1.2.1.1 Reactor Pressure Vessel

Following are the computer programs used in the analysis to assure the structural and functional integrity of the RPV.

3.9.1.2.1.1.1 CB&I Program 7-11 - GENOZZ

The GENOZZ computer program is used to proportion barrel and double taper type nozzles to comply with the specifications of ASME Code Section III and contract documents. The program will either design such a configuration or analyze the configuration

(INSERT A TO P.3.9-2) MEB #39

PIPING PROGRAMS

The verification of the following programs will be performed in accordance with the requirements of 10CFR50, Appendix B, and SRP Section 3.9. Details of the verification of input, output, and methodology are documented in General Electric or C.F Braun Design Record Files.

GENERAL ELECTRIC

a. PISY
b. LUGST
c. DISPL
d. FTFLG01
e. SAP4
f. ANS17
g. NOZAR
h. RVFOR
i. TSFOR
j. PDA
k. EZPYP
l. LION
m. WTNOZ
n. WBHFN
o. CRDSS01
p. BILDR01

C F BRAUN & CO

a. PE30
b. PE12
c. PD61
d. PDA
e. PIPERUP

CB&I REACTOR VESSEL

The verification of the following programs is assured by contractual requirements between GE and CB&I. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-Stamped equipment is in full compliance with 10CFR50, Appendix B.

a. GENOZZ
b. NAPALM
c. 1027/BIJLAARD
d. 846
e. KALNINS
f. ASFAST
g. TEMAPR
h. PRINCESS
i. TGRV
j. E0962A
k. 984
l. GASP
m. DUNHAM'S
n. 1335
o. HAP
p. 1635
q. 953
r. 1666
s. 1684
t. E1702A
u. MESHPLOT
v. 1028
w. 1038

A DESCRIPTION OF THE COMPUTER CODES AND THEIR METHOD OF VERIFICATION FOLLOWS.

INSERT (A) Continue MEB #59

3.3.1.2 Computer Programs Used in Analyses Continued

PUMP AND MOTOR VENDOR PROGRAMS

Applicant to provide.

QUESTION 40

NUREG-0800 requires if experimental stress analysis methods are used in lieu of analytical methods, that the methods used meet the provisions of Appendix II of the ASME. Provide assurance that this is indeed true.

RESPONSE

The response to this request is provided in revised Section 3.9.1.3.

MEB #40
(40)

3.9.1.2.6.3.3.1 Program Description (Continued)

high energy piping. PIPERUP is an adaptation of the finite element method to the specific requirements of pipe rupture analysis. Straight and curved beam (elbow) elements are used to mathematically represent the piping. Axial and rotational springs are used to represent restraints. The stiffness characteristics of piping and restraints can reflect elastic/linear strain hardening material properties, and gaps between piping and restraints can be modeled.

3.9.1.2.6.3.3.2 Program Verification

The PIPERUP program has been developed and verified by Nuclear Services Corporation. General Electric's subcontractor is using Version 1.2 being run on the NOS 175 computer system.

3.9.1.3 Experimental Stress Analysis

The following subsections list those NSSS components for which experimental stress analysis is performed in conjunction with analytical evaluation. The experimental stress analysis methods are used in compliance with the provisions of Appendix II of the ASME Section III. MEB #40

3.9.1.3.1 Experimental Stress Analysis of Piping Components

The following components have been tested to verify their design adequacy:

(1) piping seismic snubbers, and
(2) pipe whip restraints.

Descriptions of the snubber and whip restraint tests are contained in Subsection 3.9.3.4 and Section 3.6, respectively.

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QUESTION 41

Please provide assurance that when Service Level D Limits are specified that the methods of anlysis used to calculate the stress and deformations conform to the methods outlined in Appendix F of the ASME Code.

QUESTION 41a)

NF supports and new load combination to include thermal and SAM ~~[illegible]~~ piping loads as primarily - etc. (SIESMIC ANCHOR MOVEMENT)

RESPONSE 41

It is required by piping analysis and support design specification that ASME Code requirements including Appendix F for faulted condition shall be met for Class 1 piping and support design. Table 3.9-2(7) shows the requirements of Appendix F. Further response to this request is provided in the revised notes 9 on Table 3.9-1.

RESPONSE 41a

GESSAR uses updated ASME Code of record (proposed February 12, 1982 as effective Date shown in the response on Question #66). This new load combination to include thermal and SAM piping load as primarily is under review.

Table 3.9-1

PLANT EVENTS (Continued)

#4.1

NOTES:

1. Applies to reactor pressure vessel only.

2. Bulk average vessel coolant temperature change in any 1-hour period.

3. The annual encounter probability of a single event is $<10^{-2}$ for an ~~emergency~~ event, and $<10^{-4}$ for a ~~faulted~~ event.

4. One OBE event includes 10 maximum load or stress cycles.

5. One stress or load reversal cycle of maximum amplitude.

6. Applicable to main steam piping system only.

7. The number of structural feedback vibratory load cycles on the reactor vessel and internal components is 13,200 cycles of varying amplitude during the 220 events of safety/relief valve actuation. The main steam and recirculation piping system use 660 full range cycles and 880 half range cycles, which are comparable in effect to 13,200 cycles of varying magnitude. The main steam piping system uses 5460 cycles to include additional effects of acoustic wave propagation in steam during the actuations.

8. Table 3.9-2 shows the evaluation basis combinations of these dynamic loadings.

9. ~~The ASME Code Section II service limits of Levels A, B, C, D, or testing apply to these normal, upset, emergency, faulted, and testing operating conditions, respectively.~~

~~The methods of analysis outlined in~~ Appendix F of the ASME Code ~~were~~ is used to determine the service Level D Limits. MEB #41

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QUESTION 42

The majority of the information in Tables 3.9-11 and 3.9-12 is listed as "Applicant to Supply". Most of this missing data appears to be items that should be included in the generic plant design FSAR such as GESSAR II. Provide an explanation of why this information cannot be provided for the generic plant design.

RESPONSE

Same response as given in Question #27.

Question 43.(3.9.3)

Table 3.9-13 indicates compliance with Regulatory Guide 1.48 which has been superseded by NUREG-0800, Section 3.9.3. Please update this table to show compliance with NUREG-0800.

Response

Table 3.9-13 will be deleted. Table 3.9-2(1) will be updated to be in conformance with NUREG-0800, Section 3.9.3.

QUESTION 44

Verify that the limits in Tables 3.9-18, 3.9-19, and 3.9-20 are in compliance with NUREG-0800, Section 3.9.3. Provide a list of all instances where the footnoted equations were used and provide the needed justification for their use.

RESPONSE

These equations are left in the text until design is completed, as they are ~~[illegible]~~ OCCAISIONALLY used, although rarely. On design completion the non-applicable equations are deleted.

In Discussion of the usage of these equations at the San Jose meeting, it was agreed that no extra explanatory footnote would be required.

QUESTION 45.(3.9.3)

The functional capability for essential systems must be assured when they are subject to loads in excess of those for which Service Level B limits are specified. By essential systems are meant those ASME Class 1, 2 and 3 and any other piping systems which are necessary to shut down the plant following or to mitigate the consequences of an accident. Please provide such criteria. In particular, have the criteria in NEDO-21985 been met?

RESPONSE

The response to this request is provided in revised Section 3.9.3.1.1.6 and References in 3.9.8.

QUESTION 45a

For safety related equipment, faulted conditions to use upset stress allowables

RESPONSE

THE ESSENTIAL SYSTEMS ARE DESIGNED TO THE FOLLOWING CRITERIA TO ASSURE THEIR FUNCTIONAL CAPABILITY WHEN THEY ARE SUBJECTED TO LOADS IN EXCESS OF THOSE FOR WHICH SERVICE LEVEL B LIMITS ARE SPECIFIED.

1. ALL ACTIVE VALVES IN THE SYSTEMS SHALL MEET LEVEL B LIMITS UNDER PLANT FAULTED CONDITION LOADS.
2. ALL PIPING SYSTEMS SHALL BE ANALYZED TO MEET LEVEL D LIMITS UNDER PLANT FAULTED CONDITION LOADS.

THE FOLLOWING RATIONALES ARE THE BASIS IN JUSTIFYING THE ADEQUENCY OF THE PIPING STRESS ANALYSIS

MEB #45(4) Response Continue.

1 THE LEVEL D LIMIT FOR PIPING IS DEVELOPED BASED ON THE STRESS LEVEL OF A PLASTIC HINGE OCCURRING AT THE FIXED END OF A CANTILEVEL BEAM. THE PERFECT ELASTIC - PLASTIC ASSUMPTION OF MATERIAL PROPERTY DOES NOT CONSIDER STRAIN HARDENING DURING THE DEVELOPMENT OF A PLASTIC HINGE. IN REALITY, A PLASTIC HINGE WILL NOT BE DEVELOPED IN THE PIPE RUN EVEN THOUGH THE STRESS REACHES THE LEVEL D LIMIT

2 PIPE RUN ARE CONTINUOUS BEAM SYSTEMS INSTEAD OF SIMPLE CANTILEVER BEAM SYSTEMS, THE PIPING SYSTEM WILL NOT DISPLACE SIGNIFICANTLY UNTIL THREE PLASTIC HINGES ARE DEVELOPED IN A SINGLE SPAN. THE STRESS REACHING THE LEVEL D LIMIT AT A SINGLE POINT WILL NOT CREATE SUFFICIENT DEFORMATION TO JEOPARDIZE THE OPERABILITY OF THE PIPING SYSTEM.

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3.9.3.1.1.4 Faulted Condition (Continued)

dynamic motion associated with a safe shutdown earthquake and hydrodynamic loads plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.1.5 Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions follows. This correlation is used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Condition	Event Encounter Probability Per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	1.0 > P > 10-2
Emergency (low probability)	10-2 > P > 10-4
Faulted (extremely low probability)	10-4 > P > 10-6

3.9.3.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event. Essential Systems which are necessary to shut down the reactor or to mitigate the consequences of an accident comply with the functional capability requirements delineated in NEDO-21985 (9) in addition to the ASME Code. MEB #85 FP

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Table 3.9-2(1)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR SAFETY-RELATED NSSS PIPING AND EQUIPMENT

Load Combination*	Operating Condition Categories: ****Design Basis	Operating Condition Categories: Eval. Only
N + SRV (all)	Upset	Upset
N + OBE	Upset	Upset
N + SSE	Faulted	Faulted
N + (OBE + SRV (ALL))	Emergency	Upset
N + (SSE + SRV (ALL))	Faulted	Faulted
N + (SBA + SRV (2))	Emergency	Emergency
N + (IBA + SRV (2))	Faulted	Faulted
N + (SBA + SRV (ADS))	Emergency	Emergency
N + (SBA/IBA + SSE + SRV (ADS))	Faulted	Faulted
***N + ($LOCA_{(1-6)}$ + SSE)	Faulted	Faulted
**N + ($LOCA_{(1-7)}$)	Faulted	
***N + ($LOCA_7$ + SSE)		Faulted

*See Legend on the following pages for definition of terms and criteria for combining loads.

**From all initial conditions.

***From rated power initial conditions.

**** ALL ASME CODE CLASS 1, 2, AND 3 PIPING SYSTEMS THAT ARE REQUIRED TO FUNCTION FOR SAFE SHUTDOWN UNDER THE LISTED POSTULATED EVENTS ARE DESIGNED TO MEET THE REQUIREMENTS OF NRC MEMORANDUM, "EVALUATION OF TOPICAL REPORT - PIPING FUNCTIONAL CAPABILITY CRITERIA", DATED JULY 17, 1980.

HEB #45

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3.9.8 References

(1) Wilson, E.L., "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties," Aerojet General Corporation, Sacramento, California, Technical Memorandum No. 23, July 1965.

(2) "PVRC Recommendations on Toughness Requirements for Ferritic Materials," WRC Bulletin No. 175.

(3) "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September 1976.

(4) "BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, November 1976.

(5) NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals. Vibration in BWR/4 and BWR/5 Plants, November 1977. Also NEDO-24057-P, Amendment 1, December 1978, and NEDE-2-P 24057 Amendment 2, June 1979.

(6) "Design and Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

(7) "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

(8) "General Electric Company, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K," Proprietary Document, NEDE-20566P, November 1975.

(9) "Functional Capability Criteria for Essential Mark II Piping", NEDO-21985, September 1978. Prepared by Battelle Columbus Laboratories for General Electric Company.

MEB
#45

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QUESTION 46

What basis is used to determine allowable piping deflections during preoperational testing of piping? When a stress analysis based on a time history is performed, which code allowables were used to establish acceptability?

RESPONSE

Question withdrawn.

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QUESTION 47

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

The essential instrumentation lines to be inspected should include (but are not limited to) the following:

a. Reactor pressure vessel level indicator instrumentation lines (used for monitoring both steam and water levels.)

b. Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow).

c. Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation).

d. Control rod drive lines inside containment (not normally pressurized by required for scram).

RESPONSE

We will revise Section 3.9.2.1.2.1.2 to include all safety related instrumentation lines that are to be included in the preoperational vibration monitoring program. We will also include in the section the requirement of visual or instrumented inspection.

3.9.2.1.2.1.1 Program Description (Continued) #47

be approved to meet the requirements of Section 3.9.2 of Regulatory Guide 1.70. Allowable deflections and measurement should be developed after completion of stress analysis. Piping exceeding Phase I acceptance limits will be treated as Phase II.

Phase II requires remedial action (add or relocate supports, etc) or proceed with time-history analysis. Apply time-history analysis to determine whether additional corrections are required.

3.9.2.1.2.1.2 Measurements

All safety-related piping and instrumentation lines ~~shall be subjected~~ to preliminary vibration measurements. These measurements shall be taken during pre-operational tests with machinery and fluid systems operating under test conditions. Any indication of persistent vibration shall be followed by recorded measurements for subsequent analysis. Visual or instrumented inspection shall be conducted to identify any excessive vibration that will result in fatigue failure. Refer to Section 14.2.12.1.77.2. Special attention shall be given to piping attached to pumps, compressors, and other rotating or reciprocating equipment. Measurements shall be taken near isolation valves, pressure control valves, and other locations where shock or high turbulence may be present. Every measurement record shall be accompanied by a sketch showing the location of the measurement point, plus a description of the system operating conditions at the time of measurement. Measured data shall include actual deflections and frequencies. The time duration of measurement shall be sufficient to indicate whether the vibration is continuous or transient. MEB #47

3.9.2.1.2.1.3 Corrective Action

If the allowables are exceeded, two options are available, whichever is deemed appropriate:

(1) take remedial action (add or relocate supports, etc); or
(2) perform time-history test of the piping system.

QUESTION 48

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

RESPONSE

Operational transients are listed in Table 3.9-2(7). Preoperational test for operational transient conditions of the non-NSSS piping systems and measurement locations will be determined [redacted]. BY JANUARY 15, 1983.

allowable criteria and testing or inspection locations will also be described at this time.

QUESTION 49

NUREG-0800, Section 3.9.2 requires that a list be provided of selected locations in the piping systems at which visual inspections and measurements (as needed) will be performed during the preoperational tests. For each of these selected locations the appropriate criteria (such as permissible peak-to-peak displacement) to be used show that the stress and fatigure limits are within the design levels should be provided. Please supply this information.

RESPONSE

Acceptance Limits

For steady-state vibration, the piping break stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for level 1 criteria and 5,000 psi for Level 2 criteria.

These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles.

For operating transient vibration,* the piping bending stress (zero to peak) due to operating transient only will not exceed $1.2S_m$ or pipe support loads are not expected to exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit insures that the total primary stress including pressure and dead weight will not exceed $1.8S_m$, the new Code Service Level B limit. Level 2 criteria are based on pipe stresses and support loads not to exceed design basis pardictions. Design basis criteria require that operating transients stresses and loads not to exceed any of the Service Level B limits including primary stress limits, fatigue analyses factor limits, and allowable loads on snubber and supports.

*Operating transient will never lead to a fatigue failure, as they are only exposed to 2 or 3 cycles only.

In the dynamic analysis, the location of highest peak stress is identified, and the modal stains and displacements as sensor locations are determined relative to the peak stress on normalized basis, such as highest peak sress in each mode is 20,000 psi. This is the allowable stress range, twice the allowable amplitude. The allowable level of vibration are obtained from the stress report, and therefore are not determined at this time.

Locations of Inspections and Devices

The main steam and recirculation piping are instrumented with transducers to measure temperature, thermal movement, and vibration deflections. During preoperational vibration testings of recirculation piping, visual observation manual measurements by hand-held vibrograph are made to supplement the remote measurements.

QUESTION 50

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenace records for snubbers be documented as follows:

Pre-Service Examination

A pre-service examination should be made on all snubbers listed in Tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9. This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a minimum verify the following:

1. There are no visible signs of damage or impaired operability as result of storage, handling, or installation.

2. The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

3. Snubbers are not seized, frozen or jammed.

4. Adequate swing clearance is provided to allow snubber movements.

5. If applicable, fluid is to be recommended level and is not leaking from the snubber system.

6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational tests exceeds six months due to unexpected situations, reexamination of items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250°F should be verified as follows:

a. During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

b. For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

c. Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepancies or inconsistences shall be evaluated for cause and corrected prior to proceeding to the next specified ____________________________________.

The above described operability program for snubbers should be included and documented in the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

50 <u>Response</u>

The applicant will submit the snubber pre-service examination and pre operational testing program. This program will include maintence records for snubbers as suggested above (January 15)

QUESTION 51

Provide justification that the design of anchors which separate seismically designed piping and non-seismic Category I piping is acceptable.

RESPONSE

ALL NON-SAFETY PIPING WHICH IS CONNECTED TO THE SAFETY PIPING BY AN INTERFACED ANCHOR IS SEISMICALLY ANALYZED. SEISMIC LOADS INDUCED BY BOTH THE SAFETY AND THE NON-SAFETY PIPING ARE INCLUDED IN THE DESIGN OF THE ANCHOR

QUESTION 52

Explain how in the design process the reinforcement thicknesses of branch connections are determined for both internal pressure and mechanical loads and incorporated into the fabricated piping. Provide assurance that all branch connections that are decoupled from the main run piping in the piping analytical model are designed and fabricated to the required reinforcement area.

RESPONSE

Branch fittings are specified in the piping material classes based on pressure/temperature requirements. Additional reinforcement is added and shown on the procurement drawings if the stress analysis of the system arrangement requires it. The analytical model includes branch loads.

At the branch connection, a stress intensification factor is used to determine the maximum stresses due to mechanical loads. If the maximum stress exceeds ASME Code limits a reinforcement such as weldolet, pad, etc will be added. When the branch line is decoupled from the run pipe, branch connection must be evaluated considering all moments from three legs. Reinforcement requirements are determined by the type of stress intensifications used to satisfy the code limits.

QUESTION 53(3.9.2)

Which plant is the prototype reactor for the GESSAR II plants?

RESPONSE

Perry is the prototype reactor for the GESSAR II.

QUESTION 54

Verify that the design and installation of the safety and pressure relief valves is according to ASME Code Section III, Subsection NB-3500 and Appendix O.

RESPONSE

1. BWR Safety/Relief Valves (Safety valves with Aux. actuating devices and pilot operated valves) are designed and manufactured in accordance with ASME Code, Section III, Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000; Code Case N-100 (spring-loaded pressure relief devices; and NB-3500 (pilot operated and power actuated pressure relief valves).

2. The design of BWR Safety/Relief Valves incorporates SRV opening nd pipe reaction louds considerations required by Appendix O, and those identified under Sub-Section NB-3658 for pressure and structural integrity, safety relief valve operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both.

Further response ~~pending~~ is provided in the revision of test on Section 3.9.3.3.1 and 3.9.3.3.2 with wording similar to SRP of 3.9.3 as attached.

MEB #5x

3.9.3.3.1 Main Steam Safety/Relief Valves (Continued)

The method of analysis applied to determine piping system response to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction thus causing momentary reactions. The resulting loads on the SRV, the main steamline, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. The Code stress limits corresponding to load combinations classification as normal, upset, emergency, and faulted are applied to the steam and discharge pipe.

INSERT 4

3.9.3.3.2 Other Safety/Relief Valves

Other Seismic Category I active SRV's are listed in Table 3.9-16. Pressure relief valves are identified in the table by the valve type "RV INLET". Vacuum relief valves are identified in the table by the valve type "VAC BREAKER".

The operability assurance program discussed in Subsection 3.9.3.2.5 applies to safety-relief valves. The qualification of relief valves is specifically outlined in Subsection 3.9.3.2.5.1.2.2

3.9.3.3.3 Nuclear Island Rupture Disks

There are no rupture disks in the Nuclear Island design which must function during and after an SSE including hydrodynamic loads.

3.9.3.4 NSSS Component Supports

3.9.3.4.1 Piping

Piping supports and their attachments are designed in accordance with Subsection NF of ASME Code Section III up to the interface of the building structure as defined in the project design specifications. The building structure component supports are designed

INSERT (A) for page 3.9-97

MEB #5x

BWR Safety/Relief Valves (Safety valves with Aux. actuating devices and pilot operated valves) are designed and manufactured in accordance with ASME Code, Section III, Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000; Code case N-100 (spring-loaded pressure relief devices; and NB-3500 (pilot operated and power actuated pressure relief valves).

The design of BWR Safety/Relief Valves incorporates SRV opening and pipe reaction louds considerations required by Appendix O, and those identified under Sub-Section NB-3658 for pressure and structural integrity, safety relief valve operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both in compliance with the requirements of SRP Section 3.9.3 ~~...~~ item (2).

15x

QUESTION 55.(3.9.3)

The last line on Page 3.9-91 makes reference to Tables 3.9.10(25) and 3.9-10(26). These references are in error. Please provide the correct references.

RESPONSE

The correct references are 3.9-2(5) and 3.9-2(6) as shown in the revised Section 3.9.3.2.5.1.2.

15-5

3.9.3.2.5.1 Procedures (Continued)

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to accomplish its intended function are described in Subsection 3.9.3.2.5.1.3.

3.9.3.2.5.1.1 Tests

Prior to installation of the safety-related valves, the following tests are to be performed: (1) shell hydrostatic test to ASME Code Section III requirements; (2) back seat and main seat leakage tests; (3) disc hydrostatic test; (4) functional tests to verify that the valve will open and close within the specified time limits when subject to the design differential pressure; and (5) operability qualification of valve actuators for the environmental conditions over the installed life. Environmental qualification procedures for operation follow those specified in Section 3.11. The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

3.9.3.2.5.1.2 Dynamic Load Qualification

The functionality of a valve during and after a seismic plus hydrodynamic event may be demonstrated by an analysis or by a combination of analysis and test. Valves shall be designed using either stress analyses or the pressure temperature rating requirements based upon design conditions. An analysis of the extended structure shall be performed for static equivalent dynamic loads applied at the center of gravity of the extended structure. See Subsection 3.9.2.2 for further details.

The maximum stress limits allowed in these analyses have confirmed structural integrity and are the limits developed and accepted by the ASME for the particular ASME Class of valve analyzed. Additional details on stress limits are listed in Tables ~~3.9-10(25)~~ 3.9-2(5) and ~~3.9-10(26)~~ 3.9-2(6).

MEB #55

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QUESTION 56

Please provide your interpretation of jurisdictional boundaries as they pertain to NF supports. Justify your position. Provide tables detailing allowables for supports.

RESPONSE

1. Jurisdictional Boundaries for NF - Supports.

The jurisdictional boundaries for NF structural supports are defined in paragraph 3.0 of NF STRUCTURAL SUPPORTS specification 300-48. The applicable paragraphs of NF-1131 and NF-1132 of the ASME Code are referenced, and typical examples of boundary delineation are presented. Additional discussion is found in specification 300-50, FIELD FABRICATION AND ERECTION OF SUPPORTS.

2. Tables Detaining Allowables for Supports.

Allowable stresses of various materials used for support design are given in the Code, Subsection NF, Table NF-2121(a)-1, and Appendix I, Tables as referenced in NF-2121(a)-1. Additionally, a survey of the use of allowable stresses is given in pages 1 and 3 of the NF SUPPORT STUDY by Braun. Manufactured components such as snubbers and U-bolts are load rated. The supplier prepares Stress Report Certificates for these components as required by specification 400-20. The allowable loads are listed in the supplier catalog.

Allowable loads of the standard support designs are given in the load tables of the standards. They are based on the Code allowable stresses. ~~Put load combinations and stress limits in the GESSAR table.~~

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QUESTION 57

Explain how relief valve loads and piping reactions are calculated for those systems other than main steam safety/relief valves (SRV) discharge piping.

RESPONSE

An in-house procedure based on Moody papers in use for calculating relief valve loads. Piping reactions are obtained from flexibility analysis with input relief valve loads.

1) F. J. Moody Flood Reaction and Impingement Loads
2) F. M. Moody Time-Dependent Pipe Forces Caused by Blowdown and Flood Stoppage.

Question 58. (3.9.3)

Please provide a statement as to the compliance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking."

Response

The response to this request is provided in revised Sections 3.9.3.1.2, 3.9.4.3 and 3.9.8.

3.9.3.1.1.7 Compliance with Regulatory Guide 1.48 #5-8

Regulatory Guide 1.48 delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of the Seismic Category I fluid system components.

As shown in Table 3.9-13, the loading combinations and design limits for the GE-supplied NSSS equipment utilized in this facility are in complete compliance with or meet the intent of Regulatory Guide 1.48 through the incorporation of the alternate approach.

3.9.3.1.2 Reactor Pressure Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel, vessel support skirt, and shroud support.

The reactor pressure vessel, vessel support skirt, and shroud support are constructed in accordance with the ASME Boiler and Pressure Vessel Code Section III. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel assembly components are classified as an ASME Safety Class 1. Complete stress reports on these components have been prepared in accordance with ASME Code requirements. NUREG-0619 (10) has also been employed for feedwater nozzle design. MEB #5.

The stress analysis is performed on the reactor pressure vessel, vessel support skirt, and shroud support for various plant operating conditions (including faulted conditions) by using the elastic methods except as noted in Subsection 3.9.1.4.3. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

#58.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations (Continued)

Table 3.9-11(1). For the non-Code components, experimental testing was used to determine the CRD performance under all possible conditions as described in Subsection 3.9.4.4. For the CRD return line nozzle, NUREG-0619[10] was also employed as an improvement. HEI #58

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of the following tests:

(1) development tests;
(2) factory quality control tests;
(3) Five-year maintenance life tests;
(4) 1.5X design life tests;
(5) operational tests;
(6) acceptance tests; and
(7) surveillance tests.

All of the tests except (3) and (4) are discussed in Section 4.6. A discussion of tests (3) and (4) follows:

(3) Five-Year Maintenance Life Tests - Four control rod drives are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and 1/6 of the cycles specified in Subsection 3.9.1.1.

Upon completion of the test program, control rod drives must meet or surpass the minimum specified performance requirements.

(4) 1.5X Design Life Tests - When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in Subsection 3.9.1.1. Two CRDs have undergone such testing in 1976. These

3.9.8 References

#58

(1) Wilson, E.L., "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties," Aerojet General Corporation, Sacramento, California, Technical Memorandum No. 23, July 1965.

(2) "PVRC Recommendations on Toughness Requirements for Ferritic Materials," WRC Bulletin No. 175.

(3) "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September 1976.

(4) "BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, November 1976.

(5) NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals. Vibration in BWR/4 and BWR/5 Plants, November 1977. Also NEDO-24057-P, Amendment 1, December 1978, and NEDE-2-P 24057 Amendment 2, June 1979.

(6) "Design and Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

(7) "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

(8) "General Electric Company, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K," Proprietary Document, NEDE-20566P, November 1975.

(9) "Functional Capability Criteria for Essential Mark II piping", NEDO-21985, September 1978, prepared by Battele Columbus Laboratories for General Electric Company.

(10) "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", NUREG-0619

MEB #58

QUESTION 59(3.9.3)

Using the guidance of NUREG-0609, provide the methodlogy used and the results of the annulus pressurization (AP) analysis (asymmetric LOCA loads) for the reactor system and affected components including the following:

1. reactor pressure vessel and supports,

2. core supports and other reactor internals,

3. control rod drives,

4. ECCS piping attached to the reactor coolant system,

5. primary coolant piping, and

6. piping supports for affected piping

The results of the above analysis should specifically address the effects of the combined loadings due to annulus pressurization and an SSE.

RESPONSE

The results of the annulus pressurization (AP) analysis will be provided by ~~March 1~~ January 15, 1983. The following is a brief description of the methodology.

1. Pressure-Time Histories

   The pressure time histories in the annulus region between the RPV and shield wall are generated from a feedwater line break and a recirculation line break. The RELAP code using nodalized mass and energy balance is used in this analysis.

2. Concentrated Force-Time Histories

The forcing function of jet impingement on the shield wall is obtained from the break flow transient cause by a feedwater line break and a recirculation line break. Forcing functions of jet reaction on RPV, jet impingement on RPV, and pipe whip restraint load on restraint anchors are obtained from the feedwater line break, the recirculation line break, and main steam line break.

3. Integrated Dynamic Analysis

Beam and shell models are used to integrate pressure-time histories and concentrated force-time histories in determining the effects on the sheild wall pedestal, vessel support, core support and internals, and control rod drives. These dynamic analyses yield displacements, forces, stresses and moments.

4. Attached Piping Analysis

Acceleration time history from the integrated dynamic analysis is used to generate response spectra for the stress analysis of the attached piping. This analysis covers ECCS lines, primary coolant piping, and associated pipe supports.

5. Load Combination for Vessel and Piping

Asymmetric LOCA loads in combination with SSE by the SRSS methodology are treated as a faulted condition for evaluation against the ASME Code and functional capability requirements. This is described in Table 3.9-2.

QUESTION 60

Describe the allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections.

RESPONSE

The allowable stress limits used for bolts are as follows.

A. Flange Connection

1. NB3230 and Table I.1.3 for ASME Class 1 Piping
2. NC3658 and Table I-7.3 for ASME Class 2 & 3 Piping

B. Equipment Anchorage and Pipe Support

1. NF3280 for NF Bolts
2. AISC Stress Limits for AISC Bolts

16x

QUESTION 61

Provide a discussion of the use of U-bolts for component supports.

RESPONSE

Due to the line contact between U-bolt and pipe they are not used to carry deadweight. Allowable loads are provided on the load capacity data sheets together with dimensions. The data sheets are published in the controlled issues of the supplier pipe support hardware catalog.

U-bolts are used to support both large and small piping. Application details are shown in Braun's design standard and in specification 400-80.

Both vertical and lateral loads can be carried by U-bolts. See the supplier catalog for allowables. Normally they are used as guides. If the lateral pipe movement is negligible, they are only used as guides.

U-bolts are designed for clearance all around the pipe, generally 16-inch around large bore and 1/32-inch around small bore. Additional clearance is provided for high temperature and large pipe diameter designs. The clearance is provided by installing a nut on both sides of the supporting member. A double nut is provided on the far side to prevent loosening. Installation and tolerances for the installed condition are given in specification 300-50.

U-bolts are not preloaded, because they are not used for axial supports of piping. No specific torque values are given. General industrial practice is used.

QUESTION 62

Identify where in the plant high strength bolts have been used.

RESPONSE

No high strength (≥170 ksi ultimate) bolts are required. The following are specified for bolting materials.

SA193, GR B7	(125 ksi) Flanges
SA193, GR B8	(75 ksi) Flanges
SA193, GR B16	(125 ksi) Flanges
SA564, Tp 630	(140 ksi) Submerged Services
SA325, GR1	(105 ksi) Supports
SA540, GR B21 CL 1	(165 ksi) Supports

QUESTION 63

Provide the inservice testing program for pumps and valves including any requests for relief from ASME Section XI requirements.

RESPONSE

Inservice testing program is by applicant. Engineer provides a design specification (400-65) to insure a testing program can be maintained with no relief required from ASME Section XI.

QUESTION 64. (3.9.5.1.2.1)

Describe those short-term and long-term actions being taken to preclude the occurrence of cracking in jet pump hold down beams as described in IE Bulletin 80-07.

RESPONSE

The response to this request is provided in revised Section 3.9.5.1.2.1.

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3.9.5.1.2 Reactor Internals

3.9.5.1.2.1 Jet Pump Assemblies

The jet pump assemblies are not core support structures but are discussed here to describe coolant flow paths in the vessel. The jet pump assemblies are located in two semi-circular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 6 and 7. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (Figure 3.9-11). The driving nozzle, suction inlet, and throat are joined together as a removable unit and the diffuser is permanently installed. High pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a hold-down clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

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3.9.5.1.2.2 Steam Dryers

The steam dryer assembly is not a core support structure or safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe below the water level. This skirt forms a seal between the

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INSERT A OF QUESTION 64 on p. 3.9-113 MEB #64

The preload on the beams will be reduced from 30 to 35 kips in accordance with General Electric recommendation. The new heat treated beams will be needed. This increases the expected life of these beams without cracking to 19-40 years. Inservice inspection of the jet pump holddown beam will be performed to detect cracking inspection frequencies will be based on a load plant experience and GE testing.

QUESTION 65

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the heated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the tehcnical specifications which will require correction action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and given an indication of valve

(7)

degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

Response

The applicant will provide the details of the leak testing program and the list of pressure isolation valves will be included in the testing program. This information will be made available within 6 months of the anticipated date of commerical operation. (This will be added to section 1.9 as an interface requirement)

3.9.3.2.3.1.1 Hydrostatic Test

All seismic-active pumps shall meet the hydrostatic test requirements of ASME Code Section III according to the class rating of the given pump. The ASME Code existant at the time of bid award applies or the designated code of record.

3.9.3.2.3.1.2 Leakage Test

The fluid pressure boundary is examined for leaks at all joints, connections, and regions of high stress such as around openings or thickness transition sections while the pump is undergoing a hydrostatic test or during performance testing. Leakage rates permitted in the design specification that are exceeded are eliminated and the component retested to establish an observed leakage rate. The actual observed leakage rate, if less than permitted, is documented and made a part of the acceptable documentation package for the component.

R The applicant will provide the details of the leak testing program and the list of pressure isolation valves will be included in the testing program.

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3.9.3.2.3.1.3 Performance Test

The vendor demonstrates that the pump is capable of meeting all hydraulic requirements while operating with flow at the total developed head, minimum and maximum head, NPSH, and other parameters as specified in the equipment specification.

Bearing temperature (except water cooled bearings) and vibration levels are also monitored during these operating tests. Both are shown to be below specified levels.

3.9.3.2.3.1.4 Dynamic Qualification

The safety-related active pumps are analyzed for operability during an SSE and hydrodynamic loading event by assuring that the pump is not damaged during the seismic event and the pump continues operating despite the SEE and hydrodynamic loads.

QUESTION 66

Table 3.2-4 indicates that the information required in identifying pressure vessel components, piping, pumps and valves, and the corresponding component code, code edition, applicable addenda, and the component order date of each ASME Section III, Class 1 & 2 component within the reactor coolant pressure boundary is scheduled to be supplied by the individual applicants. One of the advantages of the Nuclear Island concept would seem to be to minimize the amount of analysis and checking of ASME code limts to be done for each individual plant. Various places in the SAR refer to 1974 and 1977 Code versions as used for design. How will plans requiring later code versions treat the differences in code loading combinations and allowables? Will the analyses have to be redone? If code allowables cannot be met without design modifications, how will these modifications be reconciled with the design presented in GESSAR II?

RESPONSE

[redacted]

The codes of record for an FDA are currently under review. However, GE proposes to use the codes in effect at GESSAR II docketing (February 12, 1982) as codes of record.