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March 14, 1983

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Docket Nos.: 50-352
50-353

Subject: Limerick Generating Station, Units 1 & 2
Draft Safety Evaluation Report (DSER)
from NRC Mechanical Engineering Branch

Reference: Letter, A. Schwencer to E. G. Bauer, Jr.
dated February 7, 1983

Dear Mr. Schwencer:

Transmitted herewith are draft responses and related FSAR page changes to open items contained in the subject DSER. This material is provided in draft form at the request of Ms. Li, NRC staff reviewer, in advance of a meeting the week of March 21, 1983. We plan to formally incorporate these page changes into the FSAR subsequent to the March 21, meeting.

Very truly yours,

Eug. J. Bradley

Eugene J. Bradley

CC: See Attached Service List

B001
Limited
Dist

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LIMERICK
MEB SER QUESTIONS

3.2 Classification of Structures, Components, and Systems

QUESTION NO.	FSAR SECTION	TECHNICAL AREA	RESPONSIBLE ORGANIZATION
1	3.2.1	Mech.	B
2	3.2.1	RPV Int.	GE
3	3.2.1	Mech.	B
4	3.2.1	React. Control Sys.	GE
5	3.2.2	CS Structures	GE
6	T3.2-1	Plant Des.	B
7	T3.2-1	RPV Skirt	GE
8	T3.2-1	CRD Hsg.	GE
9	T3.2-1	RPV Int.	GE
10	T3.2-1	Mech.	B
11	T3.2-1	Mech.	B
12	T3.2-1	Mech.	B
13	T3.2-1	Mech.	B
14	T3.2-1	C&I Equip.	GE/B

RDP:cal:hmm/K022218*-3
3/10/83

DRAFT

QUESTION NO. 1
(3.2.1, Page 3.2-2)

It is the staff's position that those portions of the steam system of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop valve, and connected piping of 2½ inches nominal pipe size or larger up to and including the first valve normally closed or capable of automatic closure during all modes of normal reactor operation should be classified seismic Category I. Your use of remotely operated manual valves in lieu of normally closed or automatic valves is not in conformance with Regulatory Guide 1.29. Provide additional assurance that your exception has an equivalent level of safety.

RESPONSE

The use of remotely operated manual valves in lieu of normally closed or automatic valves is justifiable for the following reasons:

Those portions of the steam system of Limerick extending from the outermost containment isolation valve up to but not including the turbine stop valves, and connected piping of 2½" nominal pipe size or larger up to and including the remotely operated manual valves are classified seismic Category I. In addition, these valves are Class 1E powered, and the controls are installed on seismic Category I panels located in the control room for ready operator access to remotely close the valves when required.

During normal plant operation, in case of a pipe break downstream of any one of the remotely operated manual valve, radiation monitors in the turbine enclosure exhaust will detect radiation and alert the operator in the control room. Temperature elements will also show an increase in temperature.

Each of the three remotely operated manual valves in question is downstream of the MSIV's which automatically close in the event of a large pipe break in the main steam line.

A main steam isolation event is annunciated in the control room, where the operator can manually initiate the MSIV-LCS within 20 minutes. At the same time the operator can close the three manual valves. This time period is consistent with loading requirements of the Class 1E electrical buses and provides reasonable time for operator action.

Even assuming the unlikely event of a pipe break downstream of any one of these remotely operated manual valves coincident with a LOCA, during the time period before the MSIV-LCS is actuated, the radiation doses are well below the guideline values of 10CFR 100. The activity levels in the residual steam would be comparable to normal operation activity levels. Core activity would not be transported past the already closed MSIVs due to the transport delay time of the residual steam and water.

DRAFT

QUESTION NO. 1 (CONT'D)

The MSIV-LCS, as discussed in Section 6.7, is designed with sufficient capability to control leakage from MSIVs. Flow into the MSIV-LCS is induced by blowers which will maintain the pressure in the steam lines slightly negative with respect to the atmosphere, thus ensuring that the MSIV leakage passes through the blower and into the Main Steam Tunnel where the reactor enclosure recirculation system (RERS) collects and processes it before release to the atmosphere through the Standby Gas Treatment System (SGTS).

Remote manual operation of the three valves in question is consistent with the intent of Regulatory Guide 1.96 which allows the MSIV-LCS to be a remotely operated manual system due to the transport delay time of the containment atmosphere in passing through the main steam isolation valves.

The main steam to condenser hotwell steam spargers isolation valve, HV-109, was incorrectly identified as an open valve. Section 3.2.1 is revised to correct this.

The remote manual valves are the following (shown in Figure 10.3-1):

1. Main steam to air ejectors isolation valve, HV-150

2. Main steam to steam seal evaporator isolation valve, HV-111

3. Main steam to reactor feed pump turbine high pressure steam supply valve, HV-108

4. Main steam to condenser hotwell steam spargers isolation valve, HV-109

MEB-1

Consistent with Regulatory Guide 1.26, the turbine bypass valve test is not designed to seismic Category I requirements.

d. Regarding paragraph C.1.m of the guide, reactivity control systems, such as the reactor manual control systems, that are not required to function following an SSE are not seismic Category I.

e. Structures, systems, and components that do not have safety-related functions, but whose failures reduce the function of plant safety features, are designed to seismic Category I criteria, but are not subject to the quality assurance program discussed in paragraph C.4 of the guide. Such structures, systems, and components are classified seismic Category IIA.

f. Paragraph C.3 of the Regulatory Guide recommends seismic Category I design requirements be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system requires division of the system into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, Paragraph C.4 of the Regulatory Guide states that "the pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both these guidelines are considered to be met adequately by applying the following practices to such items:

(1) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

DRAFT

QUESTION NO. 2
(3.2.1, Page 3.2-2)

What parts of the reactor internals and the reactor core are not designed to seismic Category I standards?

RESPONSE

In the reactor systems, all non-safety related reactor internal structures are not classified as seismic Category I. These internals are listed below:

1. Feedwater Spargers
2. Initial Startup Neutron Sources
3. Surveillance Sample Holders
4. In-Core Instrument Housings
5. Steam Dryer
6. Shroud Head and Separator Assembly
7. Guide Rods

DRAFT

QUESTION NO. 3
(3.2.1, Page 3.2-3)

Explain the statement "Consistent with Regulatory Guide 1.26, the turbine bypass valve test is not designed to Seismic Category I requirements".

RESPONSE

The statement as written with "....Valve Test....." is a typographical error. Section 3.2.1 is revised accordingly.

The remote manual valves are the following (shown in Figure 10.3-1):

1. Main steam to air ejectors isolation valve, HV-150

2. Main steam to steam seal evaporator isolation valve, HV-111

3. Main steam to reactor feed pump turbine high pressure steam supply valve, HV-108

4. Main steam to condenser hotwell steam spargers isolation valve, HV-109

Consistent with Regulatory Guide 1.26, the turbine bypass valve ~~test~~ chest is designed to Quality Group D. In accordance with Regulatory Guide 1.29 for the turbine stop valve, the turbine bypass valve chest is not designed to seismic Category I requirements. MEB-3

d. Regarding paragraph C.1.m of the guide, reactivity control systems, such as the reactor manual control systems, that are not required to function following an SSE are not seismic Category I.

e. Structures, systems, and components that do not have safety-related functions, but whose failures reduce the function of plant safety features, are designed to seismic Category I criteria, but are not subject to the quality assurance program discussed in paragraph C.4 of the guide. Such structures, systems, and components are classified seismic Category IIA.

f. Paragraph C.3 of the Regulatory Guide recommends seismic Category I design requirements be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system requires division of the system into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, Paragraph C.4 of the Regulatory Guide states that "the pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both these guidelines are considered to be met adequately by applying the following practices to such items:

(1) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

 DRAFT

QUESTION NO. 4
(3.2.1, Page 3.2-3)

Regulatory Guide 1.29 requires that systems or portions of systems required for reactor shutdown should be classified seismic Category I. Justify not classifying manual reactivity control systems in this manner.

RESPONSE

The LGS is installing the Redundant Reactivity Control System (RRCS) which automatically responds to an Anticipated Transient Without Scram (ATWS) event or can be manually initiated. This system meets Regulatory Guide 1.29 and Seismic Category I requirements. In addition, control rods and control rod drives are designed per seismic Category I and R.G. 1.29. Subsequently, Subsection 3.2.1d is not applicable, and is deleted from the text as shown in the attachment.

The remote manual valves are the following (shown in Figure 10.3-1):

1. Main steam to air ejectors isolation valve, HV-150

2. Main steam to steam seal evaporator isolation valve, HV-111

3. Main steam to reactor feed pump turbine high pressure steam supply valve, HV-108

4. Main steam to condenser hotwell steam spargers isolation valve, HV-109

Consistent with Regulatory Guide 1.26, the turbine bypass valve test is not designed to seismic Category I requirements.

~~d. Regarding paragraph C.1.m of the guide, reactivity control systems, such as the reactor manual control systems, that are not required to function following an SSE are not seismic Category I.~~ MEB-

d ~~e~~. Structures, systems, and components that do not have safety-related functions, but whose failures reduce the function of plant safety features, are designed to seismic Category I criteria, but are not subject to the quality assurance program discussed in paragraph C.4 of the guide. Such structures, systems, and components are classified seismic Category IIA.

e ~~f~~. Paragraph C.3 of the Regulatory Guide recommends seismic Category I design requirements be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system requires division of the system into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, Paragraph C.4 of the Regulatory Guide states that "the pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both these guidelines are considered to be met adequately by applying the following practices to such items:

(1) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

DRAFT

QUESTION NO. 5
(3.2.2, Table 3.2-1, Page 1)

Justify not classifying the core support structure Quality Group B.

RESPONSE

Limerick core support structure was designed and procured prior to the issuance of Subsection NG of the ASME Code, Section III. However, an earlier draft of ASME Code was used as a guide in developing the design criteria in lieu of Subsection NG. These criteria are presented in FSAR Section 3.9.5.3. Subsequent to the issuance of Subsection NG, comparisons were made to assure that pre-NG design meets the equivalent level of safety as presented by Subsection NG (Quality Group 'B').

DRAFT

QUESTION NO. 6
(3.2.2, Table 3.2-1)

It is the staff's position that pipe supports should have the same quality group classification as the fluid system for which they must function. Justify all cases in which ASME Class 1, 2 or 3 piping or component supports have not been given a quality group classification commensurate with the piping or component classification.

RESPONSE

For the design of Limerick pipe supports, the Code B31.7 is used for Nuclear Class 1, 2 and 3 piping and the Code B31.1 is used for non-nuclear piping. These codes were applicable at the time of the procurement of these supports (June 1971).

The stress limits used in the design are given in Table 3.9-21, which demonstrates a conservative basis for the Limerick pipe support design.

DRAFT

QUESTION NO. 7
(Table 3.2-1)

Justify not providing a quality group classification for the reactor vessel support skirt.

RESPONSE

The reactor vessel support skirt is designed to the ASME Code Section III, Subsection NF criteria. The RPV support skirt does not contain water, steam or radioactive material; therefore, quality group classification (per R.G. 1.26) is not applicable.

DRAFT

QUESTION NO. 8
(Table 3.2-1)

Justify not providing a quality group classification for the control rod drive housing supports.

RESPONSE

Control rod drive housing supports are designed in accordance with the requirements of AISC code. These supports do not fall under the jurisdictional boundary of the Subsection NF of ASME Code.

In addition, the CRD housing supports do not contain water, steam or radioactive material; therefore, quality group classification (per R.G. 1.26) is not applicable.

DRAFT

QUESTION NO. 9
(Table 3.2-1)

Justify not providing a quality group classification for the reactor internals.

RESPONSE

Reactor internal structures are designed per manufacturer's standard as shown in the table. For core support structure design basis, see response to Question No. 5. These internal structures do not contain water, steam or radioactive material, therefore, quality group classification (per R.G. 1.26) is not applicable.

Accordingly, Table 3.2-1 is corrected.

TABLE 3.2-1

LGS DESIGN CRITERIA SUMMARY

SYSTEM/COMPONENT [40]	FSAR SECTION	SOURCE OF SUPPLY [1]*	LOCATION [2]*	QUALITY GROUP CLASSIFICTION [3]*	PRINCIPAL CODES AND STANDARDS [4]*	SEISMIC CATEGORY [5]*	Q-LIST [6]*	COMMENTS
I **NSSS**								
A. **Reactor System**	4,5							
1. Reactor vessel		GE	C	A	III-1	I	Y	[7]
2. Reactor vessel support skirt		GE	C	-	III-1	I	Y	
3. Reactor vessel appurtenances, pressure retaining portions		GE	C	A	III-1	I	Y	
4. CRD (control rod drive) housing supports		GE	C	-	MP STD	I	Y	
5. Reactor internal structures, engineered safety features		GE	C	-	MP STD	I	Y	
6. Reactor internal structures, other		GE	C	-	MP STD	II	N	[8] MEB-9
7. Control rods		GE	C	-	MP STD	I	Y	
8. Core support structure		GE	C	-	MP STD	I	Y	
9. Power range detector hardware		GE	C	B	III-2	I	Y	
10. Fuel assemblies		GE	C	-	MP STD	I	Y	
B. **Nuclear Boiler System**	4,5							
1. Vessels, level instrumentation condensing chambers		GE	C	A	III-1	I	Y	[9]
2. Vessels, air accumulators		P	C	C	III-3	I	Y	
3. Piping, relief valve discharge		P	C	C	III-3	I	Y	
4. Piping and valves, reactor coolant pressure boundary (RCPB)		GE/P	C,R	A	III-1	I	Y	[7], [9]
5. Pipe supports, main steam		P	C	-	III-1	I	Y	[7]
6. Mechanical components, instrumentation with safety function		GE	C	B	MP STD	I	Y	[11]
7. Electrical modules, with safety function		GE	C,R,CS	-	IEEE-323, 344	I	Y	[11], [12]
8. Quenchers and quencher supports		P	C	C	III-3	I	Y	
C. **CRD Hydraulic System**	4.6.1							
1. Control rod drives		GE	C	-	III-2	I	Y	
2. Hydraulic control unit including scram accumulators		GE	R	-	MP STD	I	Y	[14]

* Refer to Notes at the end of this table.

DRAFT

QUESTION NO. 10
(Table 3.2-1)

Explain the use of ASME III Class 3 piping that is not Seismic Category I in the spent fuel pool cooling system.

RESPONSE

During the original design and purchase of the spent fuel pool cooling system piping, the design guidance was based on Safety Guide 26, dated March 23, 1972. Safety Guide 26 stated that any system carrying radioactive fluid should be designed to ASME III. Subsequent revisions to Safety Guide 26 (now Regulatory Guide 1.26) did not impose such a stringent requirement; however, the purchase order had been issued for ASME III pipe. As discussed in Section 9.1.3, those portions of the spent fuel pool cooling system that are required to be safety related are designed as Seismic Category I.

QUESTION NO. 11
(Table 3.2-1)

What part of the Emergency Service Water System is Seismic Category IIA?

RESPONSE

Sheets 1, 2 and 3 of Figure 9.2.2 indicate the parts of the Emergency Service Water System that are Seismic Category IIA. Non-Seismic Category I drain and vent lines and capped ends extending from Seismic Category I piping are Seismic Category IIA downstream of the last isolation valves.

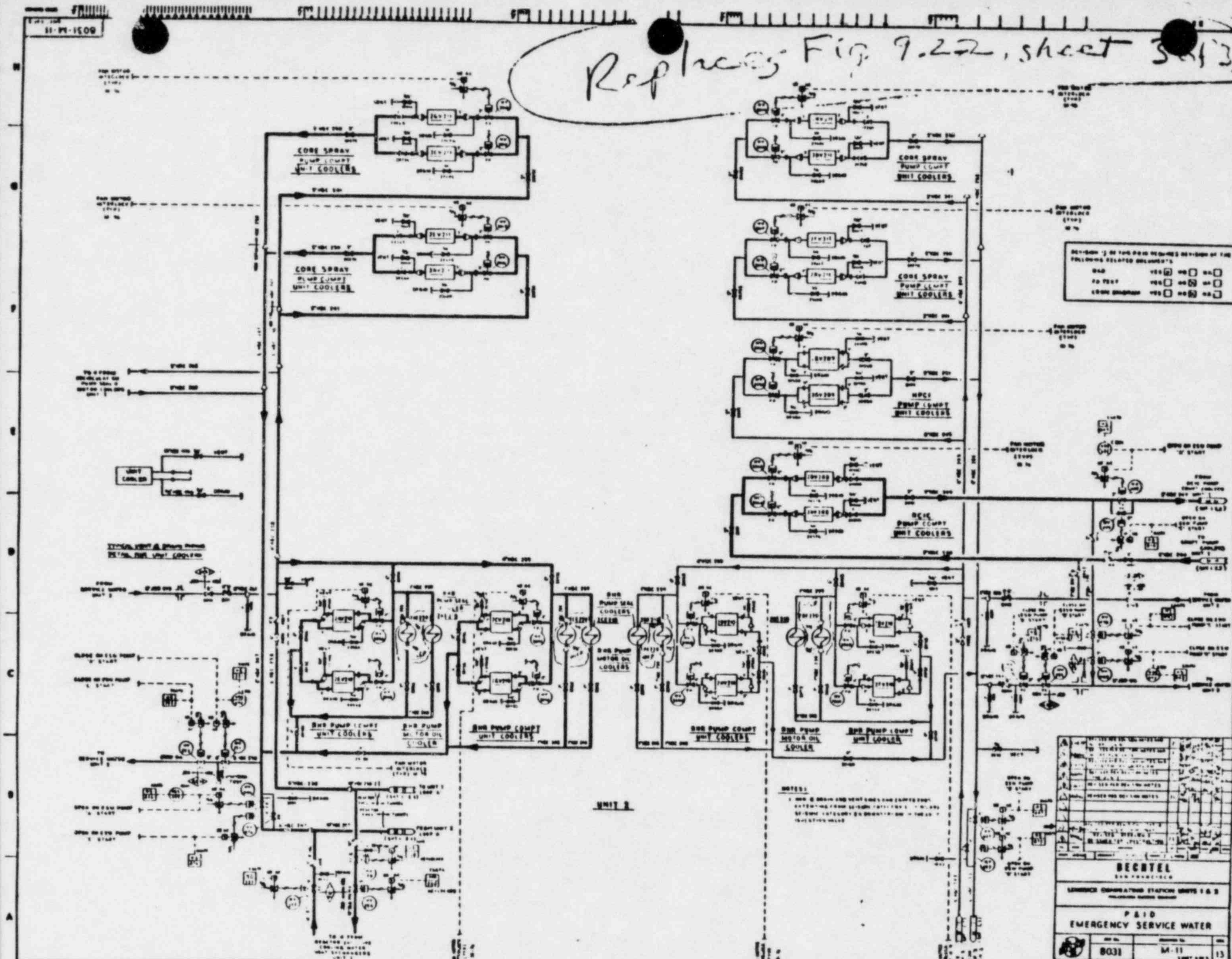
The operator may elect to provide ESW to the following non-seismic Category I equipment:

(a) RECW heat exchanger,

(b) TECW heat exchanger, and

(c) Reactor recirculation pump seal and motor oil coolers.

ESW flow to and from these components is controlled by redundant key-locked remote manual valves for (a) and (b) and lock-closed manual valves for (c).



Replaces Fig 9.2-2, sheet 3 of 3
CORE SPRAY PUMP COMPT UNIT COOLERS
CORE SPRAY PUMP COMPT UNIT COOLERS
CORE SPRAY PUMP COMPT UNIT COOLERS
CORE SPRAY PUMP COMPT UNIT COOLERS
HPCI PUMP COMPT UNIT COOLERS
RCIC PUMP COMPT UNIT COOLERS
UNIT COOLER
TYPICAL VENT & DRAIN PIPING DETAIL FOR UNIT COOLERS
RHR PUMP COMPT UNIT COOLERS
RHR PUMP MOTOR OIL COOLER
RHR PUMP SEAL COOLERS
RHR PUMP MOTOR OIL COOLERS
UNIT 2
NOTES:
BECHTEL
SAN FRANCISCO
P&ID
EMERGENCY SERVICE WATER
8031
M-11
13

MEB-11

QUESTION NO. 12
(3.2.2, Table 3.2-1)

There are systems of light-water cooled reactors important to safety that are not identified in Regulatory Guide 1.26 that the staff considers should be classified Quality Group C. Examples of these systems are: diesel fuel oil storage and transfer system; diesel engine cooling water system, diesel engine lubrication system, diesel engine starting system, diesel engine combustion air intake and exhaust system, and instrument and service air systems required to perform a safety function; and certain ventilation plant systems. Gas treatment systems which are considered as engineered safeguards systems should be classified Quality Group B.

Justify the lack of quality group classification for many of the above diesel-generator system components, the standby gas treatment system, the control structure ventilation system, the auxiliary switchgear room, and HVAC equipment room.

RESPONSE

The quality group classification for the diesel generator auxiliary system piping and components is addressed in the attached response to NRC Question No. 430.75.

The safety related HVAC fans, coils, filters, plenums, dampers, and duct work are not classified as piping or pressure vessels and as such do not fall under the jurisdiction of ASME III, Class 3 as an applicable design standard with the exception of the condenser on the control room chillers, which is classified as ASME III, Class 3.

In addition to the applicable design standards as listed for safety related systems in Table 3.2-1, the safety related plant ventilation systems, including duct work and hangers, are Q-listed, designed to Seismic Category I standards, require mill test certification to ASTM Standards and are environmentally qualified where applicable.

MEB-12

<u>QUESTION 430.75</u> (Sections 3.2, 9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8)

Diesel generator auxiliary systems piping and components are classified in the FSAR text and Table 3.2.1 as conforming to ASME Section III Class 3, ANSI B31.1, or manufacturer's standard. It is not entirely clear where the respective classifications begin or end. In any event, this is not acceptable. We require the entire diesel generator auxiliary systems to be designed to ASME Section III Class 3, or Quality Group C, in accordance with Regulatory Guide 1.26. Revise your FSAR accordingly. Also, provide the industry standards that were used in the design, manufacture, and inspection of the diesel engine mounted piping and components. Revise the appropriate P&IDs to show where quality group changes occur.

<u>RESPONSE</u>

The diesel generator auxiliary systems are the following:

a. Fuel oil system (Figure 9.5-8)

b. Cooling water system which includes the jacket water cooling loop and the air cooler coolant loop (Figure ~~9.~~ 9.5-9)

c. Starting system (Figure 9.5-10)

d. Lubrication system (Figure 9.5-11)

e. Combustion air intake and exhaust system (Figure 9.5-12)

Piping and equipment ~~in~~ these systems is provided in accordance with ASME Section III Class 3, ANSI B31.1, and manufacturer's standards as indicated on the above referenced figures and Table 3.2-1.

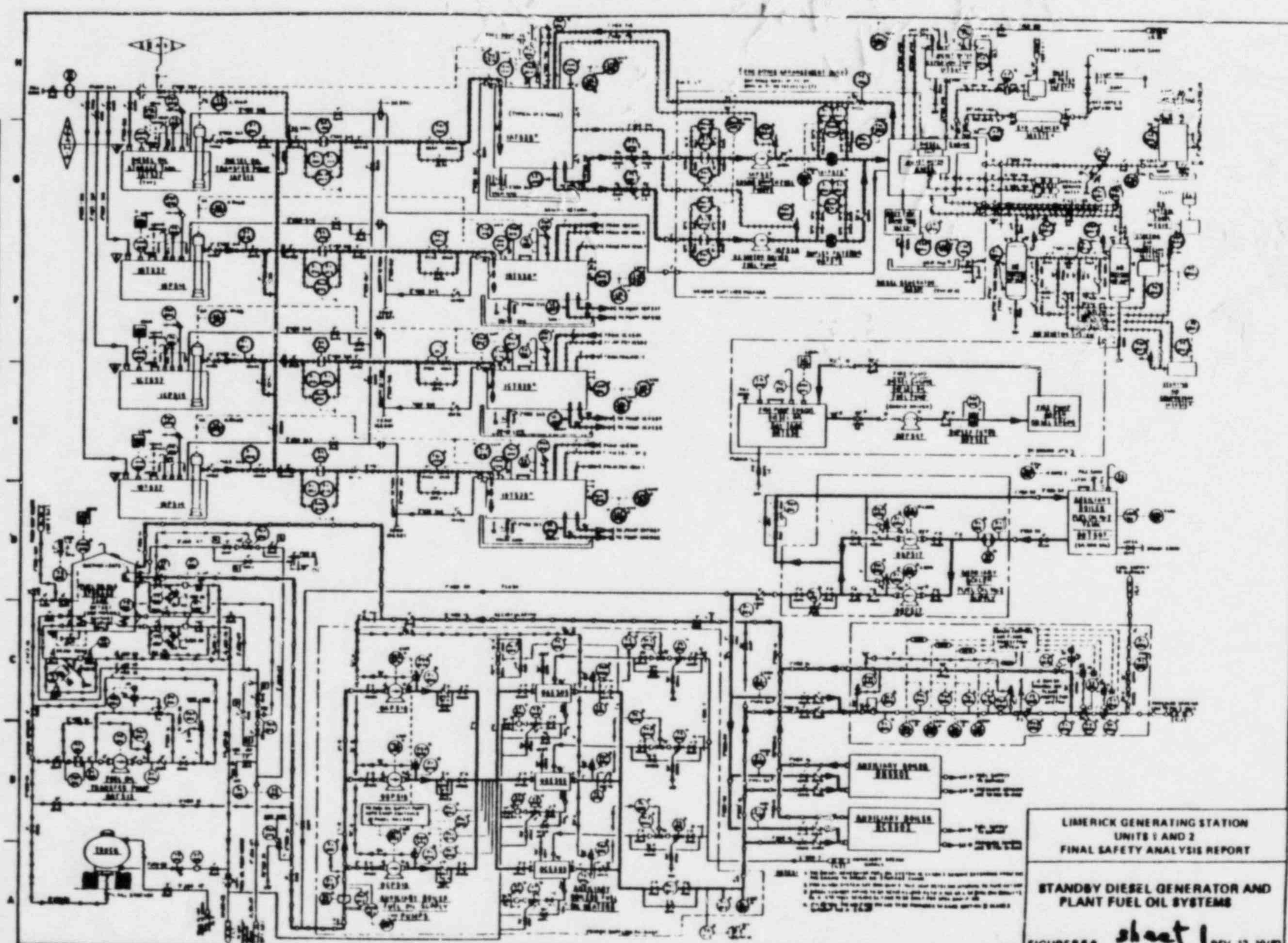
All piping and equipment has been designed to withstand seismic accelerations and operating loads, regardless of design code. The manufacturer has developed a highly reliable engine piping system over the 44 years that the design of this basic engine has been in use.

The design code used for each piping seismic segment or component meets or exceeds the commitment made in the Limerick PSAR, Appendix A and Figure A.2.3.

430.75

MZB-12

REFERENCE DRAWINGS	
P & I D NO.	FSAR FIG. NO.
[illegible]	[illegible]



LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

STANDBY DIESEL GENERATOR AND PLANT FUEL OIL SYSTEMS

FIGURE 9.5-6 sheet 1 REV. 12, 10/82

ATTACHMENT TO Q. 430.75

CHAPTER 9

MEB-12

FIGURES (Cont'd)

DRAFT

QUESTION NO. 13
(Table 3.2-1, Page 14)

Appendix A of SRP 3.2.2 requires that main steam leads from, and including, the turbine stop valves to the turbine casing shall be Quality Group D + QA or certification. Why have you omitted these lines from your 'Q' list?

RESPONSE

The main steam leads from, and including, the turbine stop valves to the turbine casing are Quality Group D and have QA certification.

Qualifications with respect to certification requirements have been met and are listed in Table 3.2-1, Item VIII.B.2 and Notes 29, 30 and 31 to the table.

QUESTION NO. 14
(Table 3.2-1, Pages 16-18)

Why are there no Quality Group Classification on instrumentation and control systems for engineered safety feature systems?

RESPONSE

Table 3.2-1, Note 45 (on Page 38 of Rev. 15, 12/82) discusses the qualification of instrumentation and control components, which states:

> "Equipment is qualified in accordance with the conformance statements made in Section 7.2, 7.3, 7.4, 7.5 and 7.6 in reference to IEEE-279 Paragraph 4.4 and IEEE-323".

Note 9 of Table 3.2-1, which discusses qualification of the instrument lines, is referenced for applicable NSSS and ESF systems instrument lines.

LIMERICK
MEB SER QUESTIONS

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

QUESTION NO.	FSAR SECTION	TECHNICAL AREA	RESPONSIBLE ORGANIZATION
15	3.6.1.1	Mech.	B/GE
16	3.6.2	Plant Des.	B
17	3.6.2.1.1.1	Pipe Break Crit.	GE/B
18	3.6.2.1.1.1	Plant Des.	B
19	3.6.2.1.1.1	Pipe Break Crit.	B
20	3.6.2.1.1.1	Mech.	B
21	3.6.2.1.1.2,3	Pipe Break Crit.	GE/B
22	3.6.2.1.1.4	Plant Des.	B
23	3.6.2.1.1.5	Mech.	B
24	3.6.2.1.3	Mech.	B
25	3.6.2.1.3	Pipe Break Crit.	GE/B
26	3.6.2.1.3	Mech.	B
27	3.6.2.1.3	Break Area	GE/B
28	3.6.2.1.3	Mech.	B
29	3.6.2.1.3	Break Geometry	GE/B
30	3.6.2.1.3	Pipe Break Crit.	GE/B
31	3.6.2.2.2	Recirc. Sys. Prop.	GE
32	3.6.2.2.2f	Recirc. Sys.	GE
33	3.6.2.2.2	Plant Des.	B
34	3.6.2.3	Plant Des.	B
35	3.6.2.4	Restraint	GE
36	3.6.2.5.2	Plant Des.	B
37	3.6.2.5.2	Plant Des.	B
38	3.6 (T&F)	Mech.	B

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DRAFT

QUESTION NO. 15
(3.6.1.1, Page 3.6-4)

Provide the analytical or experimental data used to demonstrate the capability of a pipe impacted by a larger diameter pipe or an equal diameter pipe with greater wall thickness to survive the impact without loss of pressure boundary integrity.

RESPONSE

To date, experimental or analytical data is not used in the Limerick design to justify the use of criteria other than those stated in Paragraph f of Section 3.6.1.1. If such experimental or analytical data is used in the future, the NRC will be informed.

DRAFT

QUESTION NO. 16
(3.6.2, Page 3.6-30)

What is meant by the statement "Terminal ends of the piping runs extending beyond these portions of high-energy piping are considered to originate at a point adjacent to the required moment-limiting restraints"?

RESPONSE

The above statement is clarified below and incorporated in Section 3.6.2.1.1.1.

> "Terminal ends of high energy piping which penetrate the containment shell are considered to originate beyond the containment isolation valve and its first moment limiting restraint, both inboard and outboard".

Terminal ends of high energy piping which penetrate the containment are considered to originate beyond the containment isolation valve and its first moment-limiting restraint, both inboard and outboard.

3.6.2 DETERMINATION OF PIPE FAILURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED PIPING FAILURES

Information concerning break and crack location criteria and methods of analysis is presented in this section. The location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated ruptures of high-energy and moderate-energy piping inside and outside the primary containment.

3.6.2.1 Criteria Used to Determine Pipe Break and Crack Locations and Their Configurations

3.6.2.1.1 Break Locations in High-Energy Fluid System Piping

3.6.2.1.1.1 Piping in Containment Penetration Areas

High-energy pipes penetrating the primary containment are provided with moment-limiting restraints that are located reasonably close to the containment isolation valves and are designed to withstand the loadings resulting from a pipe break either inboard of the inboard isolation valve or outboard of the outboard isolation valve so that neither isolation valve operability nor leaktight integrity of the containment penetration would be impaired as a result of such pipe breaks. ~~Terminal ends of the piping runs extending beyond these portions of high-energy piping are considered to originate at a point adjacent to the required moment-limiting restraints.~~ MEB-16

Breaks are not postulated in these portions of high-energy piping in containment penetration areas provided that the following design stress and fatigue limits are satisfied:

For ASME B&PV Code, Section III, Class 1 Piping

a. The stress intensity range S_n, calculated for normal and upset conditions by equation (10) of paragraph NB-3653, does not exceed 2.4 S_m, or

b. The stress intensity range S_n, calculated for normal and upset conditions by equation (10) of paragraph NB-3653 exceeds 2.4 S_m but does not exceed 3.0 S_m and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1, or

c. The stress intensity range S_n, calculated for normal and upset conditions by equation (10), exceeds 3.0 S_m, but the stress intensity ranges computed by equations (12) and (13) of paragraph NB-3653 are less than 2.4 S_m and the cumulative usage factor associated with normal, upset, and testing conditions is less than 0.1

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QUESTION NO. 17
(3.6.2.1.1.1, Page 3.6-30)

Branch Technical Position MEB 3-1 requires that when breaks and cracks are not postulated in high-energy ASME Class I piping in containment penetration areas, the following limits must be met:

a) The maximum stress range between any two load sets (including the zero load set) should not exceed 2.4 S_m, and should be calculated by Eq. (10) in Paragraph NB-3653, ASME Code, Section III, for those loads and conditions thereof for which Level A and Level B stress limits have been specified in the system's design specification, including an operating basis earthquake (OBE) event transient. The S_m is design stress intensity as defined in Article NB-3600 of the ASME Code Section III.

   If the calculated maximum stress range of Eq. (10) exceeds 2.4 S_m, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 should meet the limit of 2.4 S_m.

b) The cumulative usage factor should in all cases be less than 0.1.

   Revise your break exclusion criteria to include these load set and design stress intensity requirements.

RESPONSE

The Limerick design postulates breaks using the above criteria with one exception. When Equation (10) exceeds 2.4 Sm but not greater than 3.0 Sm, no break is postulated unless the cumulative usage factor exceeds 0.1. The breaks are always postulated whenever the usage factor exceeds 0.1 regardless of stress. This position is consistent with BTP-MEB 3-1, Rev. 0 (11/24/75).

Furthermore, the Limerick piping design is similar to Susquehanna and LaSalle which have received operating licenses (OL) in 1982. Therefore, the Limerick design should be adequate for safety.

DRAFT

QUESTION NO. 18
(3.6.2.1.1.1, Page 3.6-31)

When stresses in ASME III, Class 2 piping beyond the break exclusion area exceed $1.8S_h$ as calculated by Equation (9), and when the piping is constructed in accordance with ANSI B31.1, the piping shall be seamless with full radiography of all circumferential welds, or all longitudinal and circumferential welds shall be fully radiographed. Provide a commitment to this requirement.

RESPONSE

The Limerick design is in compliance with this requirement as clarified in the revised Section 3.6.2.1.1.1.

f. For piping constructed in accordance with ANSI B31.1 all welds shall be fully radiographed.

Inservice inspection of the reactor coolant pressure boundary is discussed in Section 5.2.4. MEB-18

3.6.2.1.1.2 Recirculation System Piping

Pipe breaks in the recirculation system are postulated to occur at the following locations:

a. Terminal ends of a piping run or branch run

b. At intermediate locations between terminal ends where the maximum stress range between any two load sets (including the zero load set), as calculated according to subarticle NB-3600 (ASME B&PV Code, Section III) for upset plant conditions and an independent OBE event, meets the following requirements:

  1. The stress range, as calculated using equation (12) or (13), exceeds 2.4 S_m.

  2. The stress range calculated using equation (10) exceeds 2.4 S_m but is less than 3.0 S_m, and the cumulative usage factor exceeds 0.1.

  3. The stress range calculated using equation (10) exceeds 3.0 S_m, and the cumulative usage factor exceeds 0.1.

c. If two or more intermediate break locations cannot be determined by stress or usage factor limits, two intermediate locations are selected on a reasonable basis. This basis includes consideration of fitting locations and/or highest stress or usage factor locations. Where more than two such intermediate locations are possible using the application of the above reasonable basis, those two locations possessing the greatest damage potential are used. A break at each end of a fitting can be classified as two discrete break locations when the stress analysis is sufficiently detailed to differentiate stresses at each postulated break.

3.6.2.1.1.3 Class 1 Piping (Other Than Recirculation System Piping and Piping in Containment Penetration Areas)

Breaks in Class 1 piping (ASME B&PV Code, Section III) are postulated to occur at the following locations:

a. At terminal ends of piping runs or branch runs

b. At intermediate locations between terminal ends, as determined by one of the two following criteria:

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QUESTION NO. 19
(3.6.2.1.1.1, Page 3.6-31)

Where you have employed welded support attachments in break exclusion areas, commit to performing detailed stress analyses or tests to demonstrate compliance with the applicable stress limits.

RESPONSE

In Limerick design, there are no welded support attachments within the break exclusion area.

DRAFT

QUESTION NO. 20
(3.6.2.1.1.1, Page 3.6-31)

Provide details of, and justification for instances in which 100% volumetric weld examination in break exclusion piping will not be performed.

RESPONSE

In all break exclusion piping of Limerick, 100% volumetric weld examination will be performed.

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QUESTION NO. 21
(3.6.2.1.1.2, 3; Page 3.6-3, 33)

Branch Technical Position MEB 3-1 requires that breaks be postulated in ASME III Class I piping, other than containment penetration areas according to the following criteria:

a) At terminal ends.

b) At intermediate locations where the maximum stress range as calculated by Eq. (10) and either (12) or (13) exceeds 2.4 Sm.

c) At intermediate locations where the cumulative usage factor exceeds 0.1.

d) If two intermediate locations cannot be determined by (b) and (c) above, two highest stress locations based on Eq. (10) should be selected. If the piping run has only one change or no change of direction, only one intermediate location should be postulated.

Revise your pipe break location criteria to conform to these requirements.

RESPONSE

The Limerick design has postulated breaks using the above criteria with one exception. When Equation (10) exceeds 2.4 Sm but not greater than 3.0 Sm, no break is postulated unless the cumulative usage factor exceeds 0.1. The breaks are always postulated wherever the usage factor exceeds 0.1 regardless of stress. This position is consistent with BTP-MEB 3-1, Rev. 0 (11/24/75).

Furthermore, the Limerick piping design is similar to Susquehanna and LaSalle, which have received operating licenses (OL) in 1982. Therefore, the Limerick design should be adequate for safety.

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QUESTION NO. 22
(3.6.2.1.1.4, Page 3.6-33)

The staff contends that in ASME Class 2 and 3 piping systems where intermediate break locations are postulated at each pipe fitting, and the piping system contains no fittings, valves, or welded attachments, a break should be postulated at each extreme of the piping run adjacent to the protective structure. Add this criteria to your intermediate break location postulation methodology.

RESPONSE

Section 3.6.2.1.1.4 is revised to reflect the above criteria.

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1. At each location of potential high stress such as pipe fittings (elbows, tees, reducers, etc) valves, and welded attachments

2. At each location where, for normal and upset load conditions, the following stress and fatigue limits are not met:

   (a) The stress intensity range S_n, calculated by equation (10) of paragraph NB-3653, does not exceed 2.4 S_m.

   (b) The stress intensity range S_n, as calculated by equation (10) of paragraph NB-3653, exceeds 2.4 S_m but is less than 3.0 S_m, and the cumulative usage factor is less than 0.1.

   (c) The stress intensity range S_n exceeds 3.0 S_m, but the stresses computed by equation (12) and (13) of paragraph NB-3653 are less than 2.4 S_m, and the cumulative usage factor is less than 0.1.

   When the above stress and fatigue criteria result in less than two intermediate break locations, a minimum of two separated locations are chosen based on highest stress, as calculated by equation (10) of paragraph NB-3653. The two locations are chosen with a difference in stress of at least 10% or, if stresses differ by less than 10%, the two locations are separated by a change in direction of the pipe run. Where the piping consists of a straight run without fittings, valves, or welded attachments, a minimum of one location is chosen on the basis of highest stress.

3.6.2.1.1.4 Class 2 and 3 Piping (Other Than Recirculation System Piping and Piping in Containment Penetration Areas)

Breaks in Class 2 and 3 piping (ASME B&PV Code, Section III) are postulated to occur at the following locations:

a. At terminal ends of piping runs or branch runs

b. At intermediate locations between terminal ends, as determined by one of the ~~two~~ three following criteria: MEB-22

   1. At each location of potential high stress, such as pipe fittings (elbows, tees, reducers, etc), valves, and welded attachments

3. at each extreme of the piping run adjacent to basic protective structures, when the piping system contains no fitting, valve, or welded attachment.

2. At each location where the maximum stress range, as calculated by the sum of equations (9) and (10) of paragraph NC-3652, considering normal and upset plant conditions, exceeds $0.8(1.2S_h + S_A)$.

When the above stress and fatigue criteria result in less than two intermediate break locations, a minimum of two separated locations are chosen based on highest stress, as calculated by the sum of equations (9) and (10) of paragraph NC-3652. The two locations are chosen with a difference in stress of at least 10% or, if stresses differ by less than 10%, the two locations are separated by a change in direction of the pipe run. Where the piping consists of a straight run without fittings, valves, or welded attachments, a minimum of one location is chosen on the basis of highest stress.

MEB-22

3.6.2.1.1.5 Nonnuclear Class Piping

Breaks in nonnuclear class piping are postulated to occur at the following locations:

a. At terminal ends of piping runs or branch runs

b. At each intermediate location of potential high stress, such as pipe fittings (elbows, tees, reducers, etc), valves, and welded attachments

Alternatively, the break locations for nonnuclear class piping can be selected according to the same criteria used for Class 2 and 3 piping, provided that all necessary analyses are made.

3.6.2.1.2 Crack Locations in Moderate-Energy Fluid System Piping

Through-wall leakage cracks are postulated to occur in moderate-energy piping located in areas containing essential systems and components. Cracks are postulated to occur at terminal ends of piping runs or branch runs, and at intermediate locations selected in accordance with either of the two following criteria:

a. At locations of potential high stress, such as pipe fittings (elbows, tees, reducers, etc), valves, and welded attachments

b. For Class 1 piping (ASME B&PV Code, Section III), at locations where the maximum stress range as calculated by equation (9) of paragraph NB-3652 exceeds $0.6S_m$, and for Class 2 or 3 piping (ASME B&PV Code, Section III) or nonnuclear piping, at locations where the maximum stress range as calculated by the sum of equations (9) and (10) of paragraph NC-3652 exceeds $0.4(1.2S_h + S_A)$.

 DRAFT

QUESTION NO. 23
(3.6.2.1.1.5, Page 3.6-34)

Provide assurances that breaks in non-seismic Category I piping have been postulated at those locations that would result in the maximum amount of damage and that all safety related systems and components have adequate protection from these piping breaks.

RESPONSE

The only high energy seismic Category II piping in the control structure is the portion of the steam supply line to the offgas recombiner preheater that is located within the recombiner compartments (elev. 180 feet). The only high energy seismic Category II piping in the reactor enclosure is the RWCU piping inside the RWCU filter/demineralizer compartments and the RWCU holding pump compartments (elev. 313 feet). In both cases, the walls of the compartments are capable of withstanding the pipe whip and jet impingement forces and the compartment pressurization that could result from breaks at the most adverse locations. Since there are no safety related components located in these compartments, safety related components will not be affected by high energy pipe breaks within these compartments.

Other safety related structures, such as the spray pond pump structure and the diesel-generator enclosures, do not contain high energy seismic Category II piping.

Safety related components are protected from the effects of high-energy pipe breaks in nonsafety related structures by the walls that separate the safety related structures from the nonsafety related structures. To the extent necessary to prevent unacceptable damage to safety related components, these walls are designed to withstand the pipe whip and jet impingement forces and compartment pressurization that could result from breaks at the most adverse locations in high energy seismic Category II piping within the nonsafety related structures.

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QUESTION NO. 24
(3.6.2.1.3, Page 3.6-35)

Is there any unrestrained whipping pipe located inside containment?

RESPONSE

The following high energy piping systems in the drywell are not provided with pipe whip restraints:

a. Reactor Vessel Drain Line (4" DCA-101)

b. Main Steam Drain Lines (2" & 3" DBA-105)

c. RPV Head Vent Line (2" DBA-108)

d. Standby Liquid Control Injection Line (2" DCA-112)

Each of these lines has been analyzed to verify that in the event of a pipe break, damage to structures, systems, or components needed for safe shutdown will not occur. Therefore, the ability to shut the reactor down safely will be maintained if a break occurs in any of these lines.

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QUESTION NO. 25
(3.6.2.1.3, Page 3.6-35)

Circumferential breaks should be postulated whenever the maximum stress range is exceeded and the circumferential stress is less than 1.5 times the axial stress, regardless of whether the cumulative usage factor is less than 0.1. Alter your break posulations to include this requirement.

RESPONSE

The rules for exemption of certain break orientations, which are based solely on stress and independent of calculated cumulative usage factor, are described in the Section 3.6.2.1.3. This section is corrected to reflect the above criteria. The revised criteria are consistent with the Branch Technical Position MEB 3-1 and further clarified below:

At each of these postulated break locations, consideration is given to the occurrence of either a longitudinal split or circumferential break. Both types of breaks are considered if the maximum stress ranges in the circumferential and axial directions are not significantly different. Only one type of break is considered as follows:

1. If the results of a detailed stress analysis indicate that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break is postulated.

2. If the analysis indicates that the maximum stress range in the circumferential direction is at least 1.5 times that in the axial direction, only a longitudinal split is postulated.

The above criteria notwithstanding, cracks are not postulated in those portions of moderate-energy piping located in the following areas:

a. Areas in which high-energy pipe breaks are postulated, provided that moderate-energy piping cracks would not result in more severe environmental conditions than the high-energy pipe breaks.

b. Between containment isolation valves, provided that:

   1. The piping meets the requirements of subarticle NE-1120 of the ASME B&PV Code

   2. The maximum stress range for Class 1 piping (ASME B&PV Code, Section III) as calculated by equation (9) of paragraph NB-3652 does not exceed $0.6S_m$, and the maximum stress range for Class 2 and 3 (ASME B&PV Code, Section III) or nonnuclear piping as calculated by the sum of equations (9) and (10) of paragraph NC-3652 does not exceed $0.4(1.2S_h + S_A)$.

3.6.2.1.3 Types of Breaks and Cracks in Fluid System Piping

Circumferential Breaks

A circumferential break is assumed to result in (a) severance of a high-energy pipe on a plane perpendicular to the pipe axis, and (b) separation amounting to at least a one-diameter lateral displacement of the ruptured piping ends unless physically limited by piping restraints, structural members, or piping stiffness. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

Circumferential breaks are postulated in high-energy fluid system piping of nominal pipe size greater than 1 inch, at the locations determined by the criteria listed in Section 3.6.2.1.1, except where ~~the cumulative usage factor is less than 0.1 and~~ it can be shown that the maximum stress is in the circumferential direction and is at least 1.5 times the longitudinal stress, in which case only a longitudinal break is postulated. MEB-25, 30

Longitudinal Breaks

A longitudinal break is assumed to result in an axial split parallel to the pipe axis, without causing pipe severance. The break opening area is assumed to be equal to the effective cross-sectional flow area of the pipe at the break location. The split is assumed to be oriented so that the jet reaction force causes out-of-plane bending of the piping configuration. Piping movement is assumed to occur in the direction of the jet reaction unless limited by piping restraints, structural members, or piping stiffness.

Longitudinal breaks are postulated in high-energy fluid system piping of nominal pipe sizes of 4 inches and larger, at the locations determined by the criteria listed in Section 3.6.2.1.1, with the following exceptions. Longitudinal breaks are not postulated:

a. At terminal ends, provided the piping at the terminal ends contains no longitudinal pipe welds

b. At intermediate break locations chosen to satisfy the criterion for a minimum number of break locations

c. At locations where ~~the cumulative usage factor is less than 0.1 and~~ it can be shown that the maximum stress is in the longitudinal direction and is at least 1.5 times the circumferential stress, in which case only circumferential breaks need to be postulated. MEB-25, 30

Through-Wall Leakage Cracks

Through-wall leakage cracks are postulated to occur in moderate-energy fluid system piping exceeding a nominal pipe size of 1 inch, at the locations determined by the criteria listed in Section 3.6.2.1.2. A crack is assumed to occur at any orientation about the circumference of a pipe. Fluid flow from a crack is based on a circular opening with an area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width.

3.6.2.2 Analytical Models to Define Forcing Functions and Response Models (Recirculation System Only)

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon the fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for recirculation system piping are presented below.

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

a. 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

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QUESTION NO. 26
(3.6.2.1.3, Page 3.6-35)

State where you have taken credit for a less than one pipe diameter displacement in the event of a circumferential break and discuss the analysis performed in such a case.

RESPONSE

All analyses to determine the blowdown (reaction) force on the segment of piping that contains a circumferential break are based on unobstructed discharge from 100% of the cross-sectional area of the pipe. This is consistent with the assumption of a one-diameter lateral displacement of the ruptured piping ends. The only pipe break analyses that have involved lateral displacements of less than one pipe diameter are analyses concerning jet impingement forces. In certain cases where pipe whip restrains are located on both sides of a postulated circumferential break, and the design of the restraints prevents the two ends of the break from achieving a one-diameter displacement, credit is taken for one end of the broken pipe causing partial blockage of the fluid being discharged from the opposite side of the break. Similarly, in certain cases where one side of the break is an RPV nozzle safe-end and the other side of the break is restrained from achieving a one-diameter displacement relative to the nozzle, credit is taken for partial blockage of the fluid discharging from the restraint pipe end. This methodology can result in a reduction of the jet impingement force on potential impingement targets.

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QUESTION NO. 27
(3.6.2.1.3, Page 3.6-35)

List and justify any longitudinal breaks that are assumed to be less than full area breaks.

RESPONSE

In the Limerick design, all breaks are assumed to attain full pipe break area instantaneously.

This is reflected in the revised text.

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 may be taken into account, as applicable, in the reduction of jet discharge.

b. ~~A rise time not exceeding one millisecond is used for the initial pulse.~~ all breaks are assumed to attain full pipe break area instantaneously. MEB-27

Blowdown forcing functions are determined by either of two methods as described below.

Moody Model

The predicated blowdown forces on pipes fed by a pressure vessel can be described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Ref 3.6-4. These are simply described 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 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.

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_0) 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., $0.7P_0A$).

d. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

DRAFT

QUESTION NO. 28
(3.6.2.1.3, Page 3.6-35)

Provide assurance that longitudinal breaks are postulated at two diametrically opposite points on the piping circumference.

RESPONSE

Longitudinal breaks are postulated to occur at two diametrically opposed points on the piping circumference. Section 3.6.2.1.3 is revised to clarify this assumption.

(but not concurrently) at two diametrically opposed points on the piping circumference

The above criteria notwithstanding, cracks are not postulated in those portions of moderate-energy piping located in the following areas:

a. Areas in which high-energy pipe breaks are postulated, provided that moderate-energy piping cracks would not result in more severe environmental conditions than the high-energy pipe breaks.

b. Between containment isolation valves, provided that:

   1. The piping meets the requirements of subarticle NE-1120 of the ASME B&PV Code

   2. The maximum stress range for Class 1 piping (ASME B&PV Code, Section III) as calculated by equation (9) of paragraph NB-3652 does not exceed $0.6S_m$, and the maximum stress range for Class 2 and 3 (ASME B&PV Code, Section III) or nonnuclear piping as calculated by the sum of equations (9) and (10) of paragraph NC-3652 does not exceed $0.4(1.2S_h + S_A)$.

3.6.2.1.3 Types of Breaks and Cracks in Fluid System Piping

Circumferential Breaks

A circumferential break is assumed to result in (a) severance of a high-energy pipe on a plane perpendicular to the pipe axis, and (b) separation amounting to at least a one-diameter lateral displacement of the ruptured piping ends unless physically limited by piping restraints, structural members, or piping stiffness. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

Circumferential breaks are postulated in high-energy fluid system piping of nominal pipe size greater than 1 inch, at the locations determined by the criteria listed in Section 3.6.2.1.1, except where the cumulative usage factor is less than 0.1 and it can be shown that the maximum stress is in the circumferential direction and is at least 1.5 times the longitudinal stress, in which case only a longitudinal break is postulated.

Longitudinal Breaks

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A longitudinal break is assumed to result in an axial split parallel to the pipe axis, without causing pipe severance. The break opening area is assumed to be equal to the effective cross-sectional flow area of the pipe at the break location. The split is assumed to be oriented so that the jet reaction force causes out-of-plane bending of the piping configuration. Piping movement is assumed to occur in the direction of the jet reaction

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QUESTION NO. 29
(3.6.2.1.3, Page 3.6-36)

What geometry is assumed for the opening of a longitudinal break?

RESPONSE

For high energy lines, the jet discharge is calculated assuming an opening with break area of 100% of the pipe cross-sectional area.

DRAFT

QUESTION NO. 30
(3.6.2.1.3, Page 3.6-36)

Longitudinal pipe breaks should be postulated whenever the maximum stress range is exceeded and the circumferential stress is greater than 1.5 times the axial stress regardless of whether the cumulative usage factor is less than 0.1. Change your break postulation methodology to reflect this requirement.

RESPONSE

See response to Question No. 25.

DRAFT

QUESTION NO. 31
(3.6.2.2.2, Page 3.6-39)

How is the mass/inertia and stiffness properties of the recirculation system represented?

RESPONSE

The mass/inertia and stiffness properties of the recirculation system in the pipe dynamic analysis (PDA model) are represented as described below:

A generic representation of the pipe in any given analysis is shown in Figure 1. If the stiffness of the piping segment located between A and B is such that:

- the slope of BD at B = 0, then in the analysis, the pipe is treated as built-in at B.

- the slope of BD at B ≠ 0 (considerably different), then in the analysis, the pipe is considered to have a fixed, simple support (pinned end) at B.

To analyze the pipe with both ends supported (Figure 2a) with the above computer model, two simplifications are made in the piping dynamic analysis (PDA) program. First, an equivalent point mass is assumed at D instead of pipe length DE. The inertia characteristics of this mass rotating around point B are calculated to be identical to those of pipe length DE rotating around point E. Secondly, an equivalent resisting force is calculated for any deflection for the case of a built-in end from the bending moment-angular deflection relationship for pipe length DE. This equivalent force is subtracted from the applied thrust force when calculating the net energy. The new model resulting from these simplifications is shown in Figure 2b.

The PDA computer program is described in the FSAR Section 3.9.1.2.2.6. In addition, the details of the pipe break analysis computer model and criteria are documented in the GE Licensing Topical Report NEDO-23649 (August 1977) which has been reviewed by the NRC.

Question NO. 31 (Cont'd)

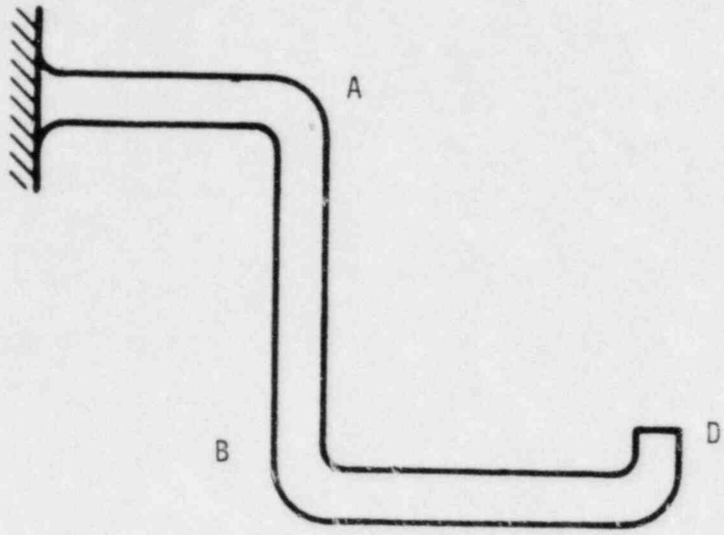
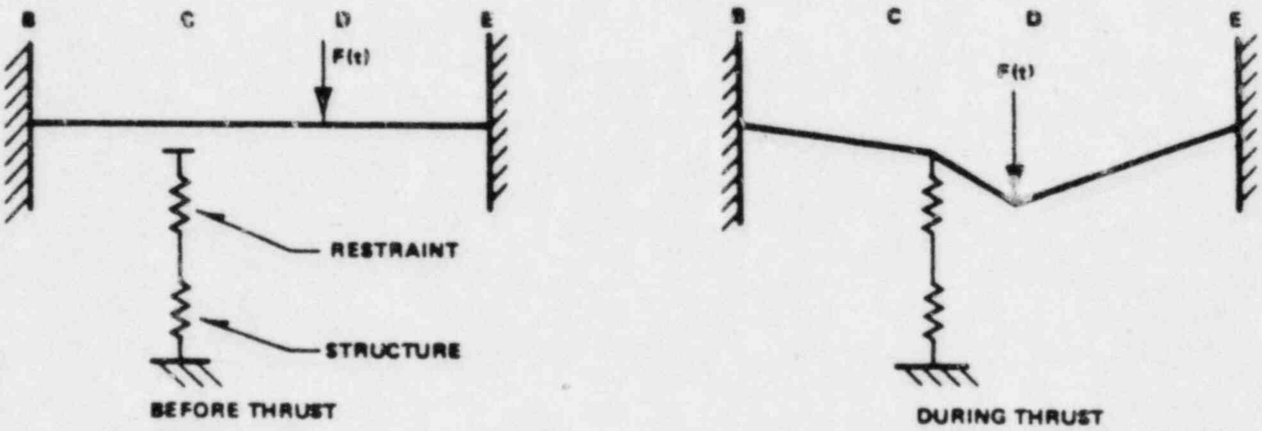
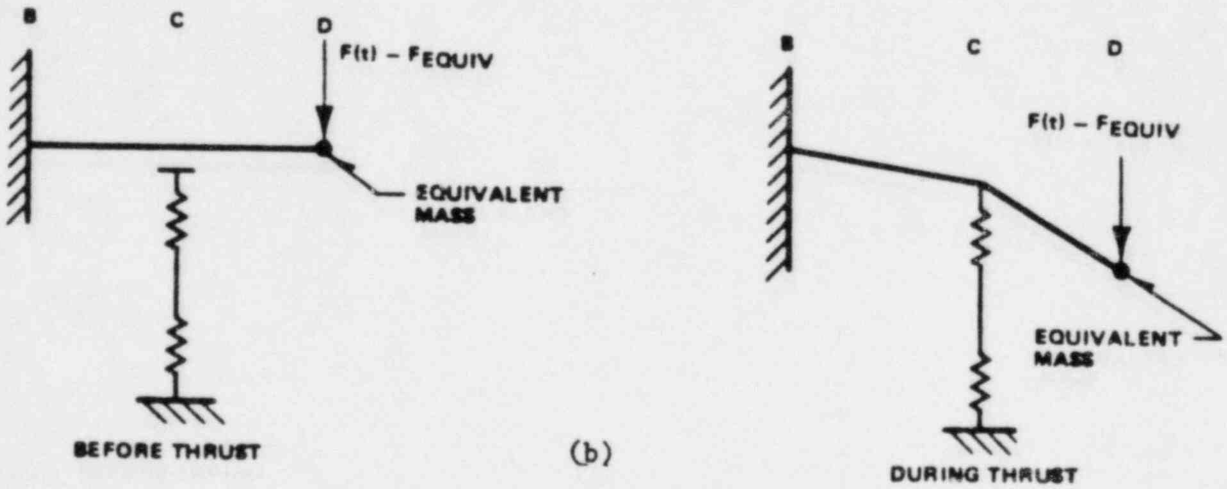


Figure 1 - Generic Representation of Pipe



(a)



(b)

Figure 2 - Representation of Pipe with Both Ends Built-In

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QUESTION NO. 32
(3.6.2.2.2.f, Page 3.6-40)

What limits are used to ensure operability?

RESPONSE

None of the components (such as vessel safe ends and valves), attached to the broken recirculation piping system, are required for safe shutdown or serve a safety function to protect the structural integrity of an essential component following a DBA.

Accordingly, the text is revised.

safety function, or whose failure would not further escalate the consequences of the accident, are not designed to meet limits imposed by the ASME B&PV Code for essential components under faulted loading. However, if these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure operability, if required, will be met.

MEB-32

The pipe whip analysis was performed using the PDA computer program (Ref 3.6-6). PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used for the pipe and the restraint. Similar to the plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of pipe motion are formulated using an energy consideration, and the equations are numerically integrated in small time steps to yield time-history information of the deformed pipe.

A comprehensive verification has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. This is described in Ref 3.6-7. Part of this verification program included an independent analysis of the recirculation system piping for the 1969 Standard Plant Design by Nuclear Services Corporation (NSC), under contract to General Electric Company. The recirculation system piping was chosen for study due to its complex piping arrangement and assorted pipe sizes. The NSC analysis included elastic-plastic pipe properties, elastic-plastic restraint properties, and gaps between the restraint and pipe as documented in Ref 3.6-7. The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load-deflection curve supplied by GE. This approximation is demonstrated in Figure 3.6-36. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint-to-structure loads than the GE analysis. The deflection limit used by NSC is the design

DRAFT

QUESTION NO. 33
(3.6.2.2.2, Page 3.6-40)

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

RESPONSE

The basis for assuming that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside containment is described below:

a) The normal operating pressure of the valves is 1155 psig. Each valve is designed, however, to withstand a differential pressure of 2132 psi across the seat. Design pressure, temperature and ASME Code class are shown below:

Valve	Design Pressure (psig)	Design Temperature (°F)	ASME Code Class
1F010A,B	2132	459	1
1F074A,B	2132	459	1
1F032A,B	2132	459	2

The valves are also seismically and dynamically qualified.

b) If a break were to occur between valves 1F074 and 1F032 (Figure 5.1-3), redundant check valves 1F010 and 1F074 (Figure 5.1-3) would have to fail to cause a LOCA outside containment. If a break were to occur upstream of 1F032, redundant check valves 1F010, 1F074 and 1F032 would have to fail. The probability of catastrophic failure of two or three of these check valves accompanying the subject break is considered to be extremely small.

c) A leakage detection system is provided in the reactor enclosure area containing the two outboard check valves 1F074 and 1F032 to alert the operator of a leak so that corrective action can be initiated. Section 5.2.5 contains a description of leak detection provisions. A postulated pipe break would be expected to provide warning indications and not an instantaneous double-ended failure that could theoretically generate unusually large dynamic loads.

d) Motor operated gate valve 1F011, located inside containment on the feedwater line, can be closed by an operator to isolate a broken line if two or three of the check valves described above fail.

In summary, a double-ended shear of a feedwater line outside containment with an instantaneous opening of the break area is not considered to be a credible design basis for the feedwater check valve.

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QUESTION NO. 34
(3.6.2.3, Page 3.6-42)

It is the staff's position that the loading condition of a piping system prior to rupture should be 102% of full power. Change your assumed loading condition or justify the lower value.

RESPONSE

The basis of selecting 100% power as the loading condition of a piping system prior to rupture is justified as follows:

1. Pipe rupture analysis state-of-the-art involves several conservative steps and assumptions in all phases of break design (e.g., probability of break, the postulated speed of break propogation, the structural material properties and the structural stability characteristics of pipe break restraint structures).

2. For those portions of piping systems which are normally pressurized during normal plant operation at power mode, the thermodynamic states in the piping systems are those of full (100%) thermal power.

3. There is a much higher probability for scheduled plant operation at 100% power (or less) than at higher ratings.

The combined effects of these conservatisms result in designs sufficiently capable of sustaining breaks at higher power levels.

DRAFT

QUESTION NO. 35
(3.6.2.4, Page 3.6-45)

Provide a detailed discussion of how you have evaluated?

a) impact and rebound due to pipe whip

b) elastic and inelastic deformation of piping and restraints

c) support boundary conditions.

RESPONSE

a) Considerable testing and analyses have demonstrated that potential rebound does not cause unacceptable increases in restraint deformation following the first quarter cycle loading for the GE restraint design and piping system experiencing blowdown thrust forces. Tests were performed on a 12-inch pipe size restraint with two primary loading configurations which represent the typical conditions during the postulated pipe rupture. Any other loading condition results in a combination of these two extremes. These loading configurations are:

   1. Load applied perpendicular to the restraint frame base against the cable; and

   2. Load applied parallel to the base against one side of the frame.

b) Non-linear and time-independent stress-strain relations are used for the pipe and the restraint. A static non-linear canti-lever beam analysis is used for these locations to obtain the relationship between the pipe bending moment and deflection (or rotation).

c) Support boundary conditions are described in Section 3.6.2.4.6.

The pipe dynamic analysis (PDA) computer program is described in Section 3.9.1.2.2.6 and a GE report NEDE-10813 on PDA was reviewed by the USAEC.

DRAFT

QUESTION NO. 36
(3.6.2.5.2, Page 3.6-46)

Provide a list of all instances where a pipe restrain touches ᴊ pipe during normal operation. Justify this practice.

RESPONSE

In all instances where piping contacts a restraint during normal operation, the restraint is included in the piping thermal and dynamic analysis model.

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 DRAFT

QUESTION NO. 37
(3.6.2.5.2, Page 3.6-46)

Provide a more detailed discussion of the design limits used to verify operability of a component that is protected by an operability restraint.

RESPONSE

Section 3.6.2.5.2 is revised to include the following information.

"The operability of the isolation valves protected by operability restraints is assured by limiting the pipe break dynamic stress in the adjacent pipe. Stresses at the junction of this component with the pipe are limited to the dynamic yield strength of the pipe material (1.1 Sy). Between the containment penetration inboard/outboard isolation valves, pipe dynamic stress is limited to be less than 2.25 Sm."

3.6.2.5.1 Design Loading Combinations

The design loading combinations applied in the design of pipe whip restraints are categorized with respect to the plant operating conditions which are identified as normal, upset, emergency, and faulted as described in Section 3.9.3.1.1. Pipe break is considered as a faulted plant condition.

3.6.2.5.2 Design Stress Limits

Operability Restraints - When restraints for piping are designed so that contact between pipe and restraint will occur during normal plant conditions, the design loading combinations for normal, upset, emergency, and faulted conditions are applicable In evaluating the supports and restraints for Class 1, 2, and 3 (ASME B&PV Code, Section III), the design stress limits applied in evaluating loading combinations for normal, upset, emergency, and faulted (except for pipe rupture) conditions are those given in Tables 3.9-12 and 3.9-16. After rupture of the supported pipe occurs, the piping system is no longer within the jurisdiction of ASME Section III because the pressure boundary has been breached. The restraints are evaluated for pipe rupture loads as described in Section 3.6.2.3. INSERT MEB-37

Independent Restraints - When restraints are designed solely to control movement following a postulated pipe rupture and to function independently of the normal support system, only the design pipe rupture loads are applicable.

To ensure that restraints function independently of the normal support system, the motions of the intact pipe due to all normal and upset plant conditions and the vibratory motion of the SSE are calculated and used to specify a minimum clearance between the pipe and the restraint. Wherever possible, gaps between pipes and restraints are maximized to avoid possible contact during plant operation. Where a particular location requires minimizing a gap, special features are provided to permit adjustment of the gap size during hot functional testing.

Independent restraints are evaluated for the pipe rupture loads as described in Section 3.6.2.3.

3.6.2.6 Guard Pipe Assembly Design Criteria

Guard pipe assemblies are not used in this plant.

3.6.3 DEFINITIONS

Certain terms used in Sections 3.6.1 and 3.6.2 have specified meanings as described below.

MEB-37

The operability of the isolation valves protected by operability restraints is assured by limiting the pipe break dynamic stress in the adjacent pipe. Stresses at the junction of this component with the pipe are limited to the dynamic yield strength of the pipe material (1.1 Sy). Between the containment penetration inboard/outboard isolation valves, pipe dynamic stress is limited to be less than 2.25 Sm.

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QUESTION NO. 38
(3.6, Tables & Figures)

Provide a schedule for completing all tables and figures.

Break locations for all high energy pipe breaks should be shown on the restraint drawings. In addition, the break exclusion area should also be shown on the applicable drawings.

RESPONSE

All tables and figures in Section 3.6 are scheduled for completion in 1983.

The piping isometric drawings listed in the index of figures for Chapter 3 will identify the locations at which breaks in high energy pipe are postulated to occur. These same drawings also show the break exclusion zones applicable to high energy piping in the containment penetration areas.

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3.7 Seismic Design

QUESTION NO.	FSAR SECTION	TECHNICAL AREA	RESPONSIBLE ORGANIZATION
39	3.7.3.2.1	OBE Fatigue Cycles	GE
40	3.7.3.2.2	Civil	B
41	3.7.3.6	3-Seismic Components (R.G. 1.92)	GE
42	3.7.3.7.1	Closely Spaced Modes (R.G. 1.92)	GE
43	3.7.3.12	Civil	B
44	3.7.3.13	Plant Des.	B

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QUESTION NO. 39
(3.7.3.2.1, page 3.7-18)

Section 3.7.3.2.1 of the LGS FSAR arrives at only one OBE intensity earthquake for design of the NSSS systems and components. Justification is required for this conclusion. Specifically, the applicant is required to provide a response to the letter from R. Boscnak (NRC) to R. Artigas (GE) dated February 18, 1982.

RESPONSE

For the NSSS piping, 50 peak OBE cycles are used.

For other NSSS equipment and components, a generic study serves as the basis for 10 peak OBE cycles. In response to the referenced letter the results of the fatigue calculations for the most limiting BWR 4 component are shown below:

BWR/4 RPV FEEDWATER NOZZLE(2)

Loading	Fatigue Usage
10 OBE Cycles	0.006
All Others (1)	0.967
Total	0.973

Accordingly, FSAR is revised as attached.

(1) All other fatigue contributions due to SRV, thermal, operating transients, etc.

(2) The most limiting calculation for the BWR/4 product line.

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3.7.3.2.1.1 NSSS Piping
Fifty peak OBE cycles are postulated for fatigue evaluation.
3.7.3.2.1.2 Other NSSS equipment and components

b. Seismic load computation based upon the tray frequency and the design spectra

c. Calculation of the tray allowable capacity

d. Evaluation of the tray capacity by interaction formula

3.7.3.1.4 Supports for seismic Category I HVAC Ducts and Cable Trays

The supports for HVAC ducts and cable trays are analyzed by the response spectrum method (see Ref 3.7-2).

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Determination of Number of Earthquake Cycles (NSSS) MEB-39

To evaluate the number of cycles which exist within a given earthquake, a typical BWR enclosure-reactor dynamic model was excited by three different recorded time histories: May 18, 1940, El Centro NS component 29.4 sec; 1952, Taft N 69° W component, 30 sec; and March 1957, Golden Gate S 80° E component, 13.2 seconds. The modal response is truncated so that the response of three different frequency bandwidths could be studied: 0-10 Hz; 10-20 Hz; and 20-50 Hz. This is done to give a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior as given in Table 3.7-18 was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. ~~This relationship is graphically shown in Figure 3.7-43.~~ MEB-39

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake was found in the following manner:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis.

b. The number of cycles which the component experiences are found from Table 3.7-18 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, 0.5% (0.005) of these cycles are conservatively assumed to be at the peak load and 4.5% (0.045) at or above three quarter peak. The remainder of the cycles have negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-year life of a plant. Fatigue evaluation due to the SSE is not necessary, since it is a faulted condition, and thus the evaluation is not required by ASME Code, Section III. MEB-39

The OBE is an upset condition, and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants show that during a 40-year life, it is probable that five earthquakes with intensities of one-tenth of the SSE intensity, and one earthquake of approximately 20% of the proposed SSE intensity, will occur. ~~Therefore, the probability of even an OBE is extremely low.~~ To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, ~~one OBE intensity earthquake is~~ ten-peak OBE cycles are postulated for fatigue evaluation.

Table 3.7-19 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.2.2 Determination of the Number of Earthquake Cycles (Non-NSSS)

In general, the design of the equipment is not fatigue controlled, because the equipment is elastic, and the number of cycles in an earthquake is low.

Equipment that is qualified by analysis is designed to remain elastic during the earthquake. Any fatigue effects in tested equipment are accounted for by the duration of the test. Consequently, the number of cycles of the earthquake is accounted for.

In order to conduct a fatigue evaluation for nuclear Class I piping, the number of cycles for a given load set is obtained. This is done by considering ten maximum stress cycles per earthquake and five OBEs and one SSE to occur within the life of the plant.

3.7.3.3 <u>Procedure Used for Modeling</u>

3.7.3.3.1 Procedure Used for Modeling (NSSS)

3.7.3.3.1.1 Modeling of Piping Systems

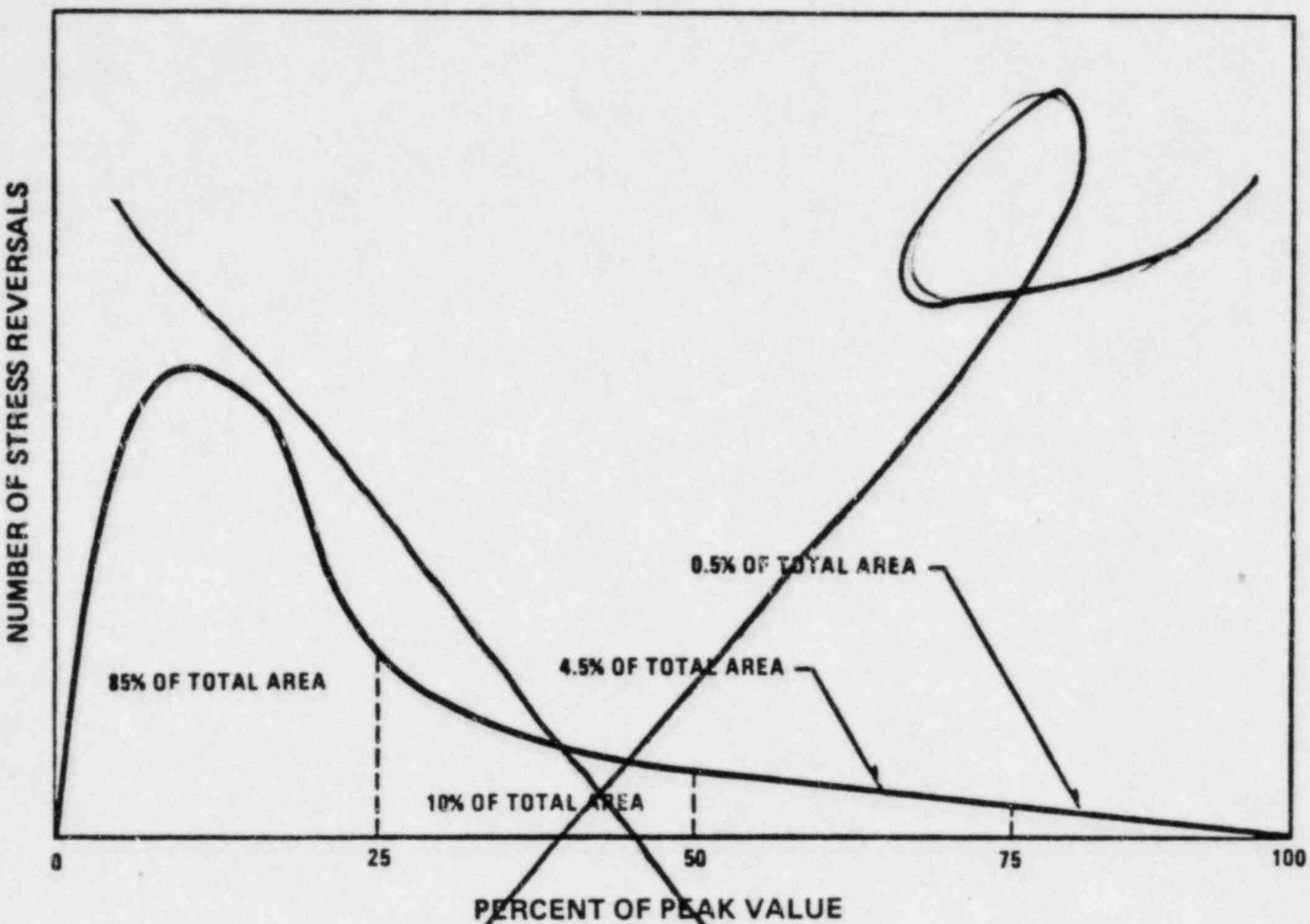
The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at the nodes connected by weightless elastic members representing the physical properties of each segment. The pipe lengths between mass points are no greater than the length which would have a natural frequency of 33 Hz, when calculated as a simply supported beam. All

TABLE 3.7-18

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT FOR NSSS SYSTEMS AND COMPONENTS

MEB-39

	FREQUENCY ~~BANK~~ BANDWIDTH (Hz)		
	0^{+} - 10	10 - 20	20 - 50
Total number of seismic cycles	168	359	643
Number of seismic cycles (0.5% of total) ~~0.5% cycles~~ between 75% and 100% of peak loads	0.8	1.8	3.2
Number of seismic cycles (4.5% of total) ~~4.5% cycles~~ between 50% and 75% of peak loads	7.5	16.2	28.9



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~~DENSITY OF STRESS REVERSALS~~

FIGURE 3.7-43

MEB-39

TABLE 3.9-2

PLANT EVENTS

EVENT NO. NORMAL, UPSET, AND TESTING CONDITIONS	NO. OF CYCLES
1. Bolt-up[1]	123
2. Design hydrostatic test	130
3. Startup (100°F/hr heatup rate)[2]	120
4. Daily reduction to 75% power [1]	10,000
5. Weekly reduction to 50% power [1]	2,000
6. Control rod pattern change[1]	400
7. Loss of feedwater heaters	80
8. 50% ~~SSE~~ ℓ OBE event at rated operating conditions	10/50 ~~10~~ ℓ [3]
9. Scram:	
a. Turbine-generator trip, feedwater on, isolation valves stay open	40
b. Other scrams	140
10. Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)[2]	111
11. Unbolt	123
12. Preop blowdown	10
13. Natural circulation startup	3
14. Loss of ac power, natural circulation restart	5

MEB-39

TABLE 3.9-2 (Cont'd)

EMERGENCY CONDITIONS	NO. OF CYCLES
15. Scram:	
Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1[4]
16. a. Automatic Blowdown	1[4]
b. Loss of feedwater pumps, isolation valves closed	5
c. Single safety or relief valve blowdown	8
17. Improper start of cold recirculation loop	1[4]
18. Sudden start of pump in cold recirculation loop	1[4]
19. Improper startup with reactor drain shut off	1[4]
FAULTED CONDITION	
20. Pipe rupture and blowdown	1[4]
21. Safe shutdown earthquake at rated operating conditions	1[4]

---

(1) Applies to RPV only.
(2) Bulk average vessel coolant temperature change in any one-hour period.
(3) ~~Includes 10 maximum load cycles per event.~~ MEB-39
(4) The annual encounter probability of the one cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

Fifty peak OBE cycles for NSSS piping, 10 peak OBE cycles for other NSSS equipment and components.

DRAFT

QUESTION NO. 40
(3.7.3.2.2, Page 3.7-19)

The reasoning that fatigue is not important for equipment because the equipment remains elastic is not valid. Change this section to indicate a more correct approach.

RESPONSE

Section 3.7.3.2.2 is revised to clarify the approach.

The SSE has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-year life of a plant. Fatigue evaluation due to the SSE is not necessary, since it is a faulted condition, and thus the evaluation is not required by ASME Section III.

The OBE is an upset condition, and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants show that during a 40-year life, it is probable that five earthquakes with intensities of one-tenth of the SSE intensity, and one earthquake of approximately 20% of the proposed SSE intensity, will occur. Therefore, the probability of even an OBE is extremely low. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, one OBE intensity earthquake is postulated for fatigue evaluation.

Table 3.7-19 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.2.2 Determination of the Number of Earthquake Cycles (Non-NSSS)

MEB-40

In general, the design of the equipment is not fatigue controlled. ~~because the equipment is elastic, and the number of cycles in an earthquake is low.~~ fatigue is not a controlling factor because.

For ~~E~~equipment ~~that is~~ qualified by analysis, ~~is designed to remain elastic during the earthquake.~~ ¶ Any fatigue effects in tested equipment are accounted for by the duration of the test. Consequently, the number of cycles of the earthquake is ~~accounted for.~~ considered.

In order to conduct a fatigue evaluation for nuclear Class I piping, the number of cycles for a given load set is obtained. This is done by considering ten maximum stress cycles per earthquake and five OBEs and one SSE to occur within the life of the plant.

3.7.3.3 <u>Procedure Used for Modeling</u> INSERT NEXT PAGE

3.7.3.3.1 Procedure Used for Modeling (NSSS)

3.7.3.3.1.1 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at the nodes connected by weightless elastic members representing the physical properties of each segment. The pipe lengths between mass points are no greater than the length which would have a natural frequency of 33 Hz, when calculated as a simply supported beam. All

a) The equipment is designed to remain below 90% of the yield strength of the material for the extreme loading condition.

b) The number of stress cycles considered is 60 (5 OBE and 1 SSE events at 10 cycles each). Based on ASME Section III, Appendix I Criteria (Figure I-9-1), this number of cycles will not result in a reduction of allowable stresses. MEB-40

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QUESTION NO. 41
(3.7.3.6, Page 3.7-22)

Section 3.7.3.6 of the LGS FSAR states that for NSSS systems, the absolute sum of the largest horizontal response and the vertical response was used for response spectrum methods while the algebric sum of contribution due to two earthquake components was used for time history methods. Regulatory Guide 1.92 requires that the square-root-of-the-squares of three components of the earthquake motion be used. The applicant is requested to justify the approach used in the LGS analysis. In addition, describe how the vertical response spectrum is determined.

RESPONSE

The text is revised to include the following:

1. Three Components of Earthquake Motion

   The simultaneous use of three components of earthquake motion was not a design basis requirement of the construction permit for this plant. However, the NSSS systems and components are evaluated to the requirement of Regulatory Guide 1.92.

   a. Response Spectrum Method

      The individual response spectra in each orthogonal direction are obtained by the SRSS combination of the colinear contribution due to the three directions of earthquake motion.

   b. Time-History Method

      When the time-history method of analysis is used, the time-history responses from each of the three components of the earthquake motion are combined algebrically at each time step.

2. Effects of Parameter Variations on Floor Response Spectra

   To account for potential variations in the primary structure frequencies, the computed floor response spectra are peak-broadened by ±15%.

interaction is not used in the dynamic analysis. A simplified lumped mass method using a fixed base model is used. However, for a more refined analysis of containment and reactor enclosure, the underlying foundation medium is considered to interact with the structure. The equivalent soil spring constant and damping coefficient are computed in accordance with the formulae of Table 3-2 of Ref 3.7-2, and the analysis carried out by the methods discussed in Appendix D of Ref 3.7-2. The resulting structure-foundation interaction coefficients are listed in Table 3.7-17.

3.7.2.5.1 Floor Response Spectra (NSSS) See Section 3.7.3.6.1.a.

MEB-41

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.2 Floor Response Spectra (Non-NSSS)

The time-history method of analysis was used to develop the floor response spectra. A discussion of the technique of finding the nodal time history and then producing the spectrum may be found in Sections 4.2 and 5.2 of Ref 3.7-2.

3.7.2.6.1 NSSS See Section 3.7.3.6.1

MEB-41

3.7.2.6 Three Components of Earthquake Motion

3.7.2.6.2 Non-NSSS

The response spectrum method was used in seismic analysis of structures. Independent analyses are performed for the vertical and two horizontal (east-west and north-south) directions. For design purposes, the response value used is the maximum value obtained by adding the response due to the vertical earthquake with the larger value of the response due to one of the horizontal earthquakes by the absolute sum method.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Combination of Modal Responses (NSSS)

MEB-42

See Section 3.7.3.7.1

~~When the response spectra method of modal analysis is used, all modes are combined by the SRSS method. (See Section 3.7.3.7.1.1).~~

3.7.2.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses (i.e., shears, moments, deflections, accelerations, and inertia forces) are combined by either the sum of the absolute values method, or by the square root of the sum of the squares method with consideration of consideration of closely spaced modes. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by ten percent or less of the lower frequency. When the SRSS method is used, USNRC Regulatory Guide 1.92 shall be adopted for the combination of modal responses.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

The turbine enclosure is the only non-Category I structure close to seismic Category I structures. It is designed to withstand an

SSE without the structural elements exceeding the yield strength. Dynamic analysis of this structure was done by the response spectrum method.

The remaining non-Category I structures are designed for seismic loads according to the Uniform Building Code (UBC) (Ref 3.7-3). The non-Category I structures are analytically checked to ensure that they will not collapse on, or otherwise impair the integrity of, adjacent seismic Category I structures when subjected to the design seismic loads.

Structural separations have been provided to ensure that interaction between Category I and non-Category I structures does not occur. The minimum separation gap between the buildings is twice the relative displacement except at two locations (constituting less than 1% of the total contact area), where it is only 1.7 times the relative displacement.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

MEB-41

INSERT

To account for variations in the structural frequencies owing to uncertainties in the material properties of the structure and to approximations in the modeling techniques used in the seismic analysis, the computed floor response spectra are smoothed, and peaks associated with each of the structural frequencies are broadened. In lieu of making a parametric study considering changes in the material properties and other variables, the spectrum is broadened on either side of the peak value by 15% of the frequency at which the peaks occur.

3.7.2.10 Use of Constant Vertical Static Factors

Vertical seismic system multi-mass dynamic models are used to obtain vertical response loads for the seismic design of seismic Category I structures. Therefore, constant vertical static factors are not used to account for vertical response to earthquakes for the seismic design of Category I structures.

3.7.2.11 Methods Used to Account for Torsional Effects

Torsional effects for the reactor enclosure, diesel-generator enclosure, spray pond pumphouse, and radwaste enclosure are accounted for as follows:

A static analysis is performed to account for torsion on these structures. The eccentricity is determined using the distance between the center of mass and the center of rigidity of the individual structure. The inertial force from the response spectrum analysis is applied at the center of mass. The resulting torsional moment is equal to the inertial force times the eccentricity. The shear forces due to the torsional moment

3.7.2.9.1 Effects of Parameter Variations on Floor Response Spectra (NSSS)

To account for potential variations in the primary structure frequencies due to uncertainties in material properties of the soil and structure, soil structure interaction techniques, approximation in damping, and approximation in dynamic modeling, the computed floor response spectra are peak-broadened by ±15%. This is consistent with the requirements of Regulatory Guide 1.122, although this regulatory guide is not the design basis requirement for LGS construction permit.

3.7.2.9.2 Effects of Parameter Variations on Floor Response Specta (Non-NSSS)

MEB-41

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frequency equal to or greater than 33 Hz, it is considered as rigid. If the natural frequency of the component falls within the broadened peak of the response spectrum curve, then it is designed to take the applied load.

3.7.3.5 Use of Equivalent Static Load Method of Analysis (Non-NSSS)

The equivalent static load method is used when the natural frequency of the equipment is not determined. If the equipment can be adequately represented by a single degree of freedom system, then the applied inertia load is equal to the weight of the equipment times the peak value of the response spectrum curve. Seismic acceleration coefficients for multi-degree of freedom systems, which may be in the resonance region of the amplified response spectra curves, are increased by 50% to account conservatively for the increased modal participation.

Appendix D of BP-TOP-1 (Ref 3.7-4) discusses the use of equivalent static load method of analysis as applicable to piping.

3.7.3.6 Three Components of Earthquake Motion

3.7.3.6.1 Three Components of Earthquake Motion (NSSS)

→ INSERT NEXT PAGE

~~a. Response spectrum method~~

~~The simultaneous use of three components of earthquake motion is not a design basis requirement of the construction permit for this plant. The total seismic responses is predicted by combining the response calculated from analyses due to one horizontal and one vertical seismic input. For this case, where the response spectrum method of seismic analysis is used, the basis for combining the loads from the two analyses is given below:~~ MEB-4)

~~1. The peak responses of the different modes for the same earthquake excitations do not occur at the same time.~~

~~2. The peak responses of a particular mode due to earthquake excitations from different directions do not occur at the same time.~~

~~3. The peak stresses due to different modes and due to different excitations may not occur at the same location, nor in the same direction.~~

~~To implement the above, the two translation components of earthquake excitations are combined by finding the~~

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The simultaneous use of three components of earthquake motion was not a design basis requirement of the construction permit for this plant. However, the NSSS systems and components are evaluated to the requirement of Regulatory Guide 1.92.

a. Response Spectrum Method

Response spectra generated by GE are developed considering three component of earthquake motion. The individual response spectra in each orthogonal direction are obtained by the SRSS combination of the colinear contribution due to the three directions of earthquake motion. These are used to predict the total response at each frequency.

b. Time-History Method

When the time-history method of analysis is used, one of the following options is used to obtain the peak value of any particular response of interest.

1. When maximum colinear contributions due to the three directions of earthquake motion are calculated separately, the total response is obtained as the SRSS combination of the colinear values.

2. When colinear time history responses from each of the three components of the earthquake motion are calculated individually by the step-by-step method and then combined algebraically at each time step, the maximum response is obtained as the peak value from the combined time solution.

3. Finally, when a response at each time step is calculated directly based on the simultaneous application of the three earthquake components, the maximum response is determined by scanning the combined time-history solution.

The components of earthquake motion must be statistically independent for Options 2 and 3. Also, the time-history method precludes the need to consider closely spaced modes.

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absolute sum of all responses of interest (e.g., strain, displacement stress, moment, shear, etc) from seismic motion, in one horizontal (x or z) and one vertical direction (y), i.e., $|x| + |y|$ or $|y| + |z|$. The design is made for the larger of the two sums $|x| + |y|$ or $|y| + |z|$.

b. Time history method

The algebraic sum of contributions (to displacements, loads, stresses, etc) due to the two earthquake components is calculated for each natural mode for each time interval of analysis. The time interval is less than or equal to 0.2 of the smallest period of interest. The maximum values of all time intervals are the design displacements, accelerations, loads, or stresses.

3.7.3.6.2 Three Components of Earthquake Motion (Non-NSSS)

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For equipment, cable trays, and supports for cable trays and HVAC ducts, the three spatial components of the earthquake are considered in the same manner as for structures (described in Section 3.7.2.6).

The criteria used for combining the results of horizontal and vertical seismic responses for piping systems are described in Section 5.1 of Ref 3.7-4.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Combination of Modal Responses (NSSS)

When the response spectra method of modal analysis is used, all modes are combined by the SRSS method. The SRSS combination of modal responses is defined mathematically as:

$$R = \left[\sum_{i=1}^{n} (R_i)^2 \right]^{1/2} \qquad (3.7\text{-}6)$$

where

R = Combined response

R_i = Response in the i mode

n = Number of modes considered in the analysis

3.7.3.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses of equipment are combined by the SRSS method. The absolute values of two closely spaced modes are added first

DRAFT

QUESTION NO. 42
(3.7.3.7.1, Page 3.7-23)

How are closely spaced modes combined for NSSS systems and components?

RESPONSE

Regulatory Guide 1.92 is not a design basis for the construction permit of this plant; however, all NSSS systems and components are evaluated by using the double sum method with absolute sign for combination of closely spaced modes, consistent with R.G. 1.92.

The relevant text is revised.

interaction is not used in the dynamic analysis. A simplified lumped mass method using a fixed base model is used. However, for a more refined analysis of containment and reactor enclosure, the underlying foundation medium is considered to interact with the structure. The equivalent soil spring constant and damping coefficient are computed in accordance with the formulae of Table 3-2 of Ref 3.7-2, and the analysis carried out by the methods discussed in Appendix D of Ref 3.7-2. The resulting structure-foundation interaction coefficients are listed in Table 3.7-17.

3.7.2.5 Development of Floor Response Spectra

The time-history method of analysis was used to develop the floor response spectra. A discussion of the technique of finding the nodal time history and then producing the spectrum may be found in Sections 4.2 and 5.2 of Ref 3.7-2.

3.7.2.6 Three Components of Earthquake Motion

The response spectrum method was used in seismic analysis of structures. Independent analyses are performed for the vertical and two horizontal (east-west and north-south) directions. For design purposes, the response value used is the maximum value obtained by adding the response due to the vertical earthquake with the larger value of the response due to one of the horizontal earthquakes by the absolute sum method.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Combination of Modal Responses (NSSS)

See Section 3.7.3.7.1

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~~When the response spectra method of modal analysis is used, all modes are combined by the SRSS method. (See Section 3.7.3.7.1.1).~~

3.7.2.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses (i.e., shears, moments, deflections, accelerations, and inertia forces) are combined by either the sum of the absolute values method, or by the square root of the sum of the squares method with consideration of consideration of closely spaced modes. Two consecutive modes are defined as closely spaced when their frequencies differ from each other by ten percent or less of the lower frequency. When the SRSS method is used, USNRC Regulatory Guide 1.92 shall be adopted for the combination of modal responses.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

The turbine enclosure is the only non-Category I structure close to seismic Category I structures. It is designed to withstand an absolute sum of all responses of interest (e.g., strain, displacement stress, moment, shear, etc) from seismic motion, in one horizontal (x or z) and one vertical direction (y), i.e., $|x| + |y|$ or $|y| + |z|$. The design is made for the larger of the two sums $|x| + |y|$ or $|y| + |z|$.

b. Time history method

The algebraic sum of contributions (to displacements, loads, stresses, etc) due to the two earthquake components is calculated for each natural mode for each time interval of analysis. The time interval is less than or equal to 0.2 of the smallest period of interest. The maximum values of all time intervals are the design displacements, accelerations, loads, or stresses.

3.7.3.6.2 Three Components of Earthquake Motion (Non-NSSS)

For equipment, cable trays, and supports for cable trays and HVAC ducts, the three spatial components of the earthquake are considered in the same manner as for structures (described in Section 3.7.2.6).

The criteria used for combining the results of horizontal and vertical seismic responses for piping systems are described in Section 5.1 of Ref 3.7-4.

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3.7.3.7 Combination of Modal Responses

INSERT

~~3.7.3.7.1 Combination of Modal Responses (NSSS)~~

~~When the response spectra method of modal analysis is used, all modes are combined by the SRSS method. The SRSS combination of modal responses is defined mathematically as:~~

$$R = \left[\sum_{i=1}^{n} (R_i)^2 \right]^{1/2} \qquad (3.7\text{-}6)$$

~~where~~

~~R = Combined response~~

~~R_i = Response in the i mode~~

~~n = Number of modes considered in the analysis~~

3.7.3.7.2 Combination of Modal Responses (Non-NSSS)

The modal responses of equipment are combined by the SRSS method. The absolute values of two closely spaced modes are added first

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3.7.3.7.1 - Combination of Modal Responses (NSSS)

All piping and equipment analyzed or supplied by GE are evaluated to the requirements of Regulatory Guide 1.92.

When the response spectra method of modal analysis is used, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10 percent) are combined by the square root of the sum of the squares (SRSS) as described in Section 3.7.3.7.1a. Closely spaced modes are combined by the double sum method with absolute sign as described in Section 3.7.3.7.1b.

In the time-history method of dynamic analysis, the vector sum at every time step is used to calculate the combined response. The use of the time-history method precludes the need to consider modal spacing.

a. Square Root of the Sum of the Squares

The square root of the sum of the squares (SRSS) method is defined mathematically as:

$$R = \left[\sum_{i=1}^{n} (Ri)^2 \right]^{\frac{1}{2}}$$

Where:

R = Combined response
Ri = Response due to the ith mode
n = Number of modes considered in the analysis

b. Procedure of Combining Closely Spaced Modal Response

This method is defined mathematically as:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| E_{ks} \right]^{\frac{1}{2}}$$

Where R is the representative maximum value of a particular response of a given element to a given component of excitation, R_k is the peak value of the response of the element due to the kth mode, and N is the number of significant modes considered in the modal response combination. In addition, R_s is the peak value of the response of the element attributed to sth mode. Also,

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$$E_{ks} = \left[1 + \left[\frac{w'_k - w'_s}{\beta'_k w_k + \beta'_s w_s} \right]^2 \right]^{-1}$$

in which

$$w'_k = w_k \left[1 - \beta_k^2\right]^{0.5} \quad \text{and} \quad \beta'_k = \beta_k + \frac{2}{t_d w_k}$$

Where w_k and β_k are the modal frequency and the damping ratio in the kth mode, respectively, and t_d is the duration of the earthquake.

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QUESTION NO. 43
(3.7.3.12, Page 3.7-25)

Please provide a more detailed discussion of your analysis procedures for buried seismic Category I piping. Provide an example of an analysis.

RESPONSE

A detailed discussion of the procedures and its application for analyzing buried seismic Category I piping is provided in the FSAR Reference 3.7-2 "Seismic Analyses of Structures and Equipment for Nuclear Power Plants", BC-TOP-4A, Rev. 3, Bechtel Power Corporation, San Francisco, California (November 1974), Section 6. This report has been reviewed and accepted by NRC.

DRAFT

QUESTION NO. 44
(3.7.3.13, Page 3.7-25)

Please provide a discussion of the techniques used to design anchors that separate seismic Category I and non-seismic Category I systems.

RESPONSE

Seismic boundary anchors are designed for the combined loads generated from both sides of a boundary anchor. The loads from the seismic Category I side are actual calculated loads and the loads from the non-seismic Category I side are determined by one of the following:

1. The actual calculated seismic loads if the non-seismic side piping is dynamically analyzed for seismic events,

2. The actual calculated loads if the non-seismic side piping is designed to a conservative simplified seismic design criteria (e.g., by simplified span methods such as those used for designed of small piping), or

3. The loads determined by the plastic capability of the piping.

In accordance with this response, Section 3.7.3.13.2 is revised.

3.7.3.13.2 Interaction of other Piping with Seismic Category I Piping (Non-NSSS)

The techniques used to consider the interaction of seismic Category I piping with non-Category I piping are discussed in Section 3.4 of Ref 3.7-4.

3.7.3.14 Seismic Analysis for Reactor Internals (NSSS)

The modeling of RPV internals is discussed in Section 3.7.2.3.1.2. The damping values are given in Table 3.7-1. A comparison of seismic responses is shown in Table 3.7-4.

3.7.3.15 Analysis Procedures for Damping

3.7.3.15.1 Analysis Procedures for Damping (NSSS)

Analysis procedures for damping are discussed in Section 3.7.2.15.1.

3.7.3.15.2 Analysis Procedure for Damping (Non-NSSS)

If the equipment damping is unknown, the response spectrum curve for 0.5% damping is used to arrive at a conservative seismic loading. The damping values used for the OBE are increased for the SSE, where sufficient justification is established.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison With NRC Regulatory Guide 1.12 Rev 1

The seismic instrumentation program complies with Regulatory Guide 1.12 Rev.1, except for the item listed below:

Response spectrum recorders are not supplied as discrete instruments. A response spectrum analyzer, permanently installed in the control room, presents more complete information than that presented by response spectrum recorders. Recorded data from the triaxial time-history accelerographs are fed into the response spectrum analyzer to produce earthquake spectra immediately following an earthquake. All locations where response spectrum recorders are required by the regulatory guide are monitored by time-history accelerographs. This system achieves the intent of Regulatory Guide 1.12 Rev 1.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation is provided for Unit 1 only, as essentially the same response is expected at Unit 2.

a. Seven triaxial time-history accelerographs

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INSERT FOR 3.7.3.13.2

Seismic boundary anchors are designed for the combined loads generated from both sides of a boundary anchor. The loads from the seismic Category I side are actual calculated loads and the loads from the non-seismic Category I side are determined by one of the following:

1. The actual calculated seismic loads if the non-seismic side piping is dynamically analyzed for seismic events,

2. The actual calculated loads if the non-seismic side piping is designed to a conservative simplified seismic design criteria (e.g., by simplified span methods such as those used for designed of small piping), or

3. The loads determined by the plastic capability of the piping.

NOT USED

3.7-4 "Seismic Analysis of Piping Systems", BP-TOP-1, Rev. 3, Bechtel Power Corporation, San Francisco, California (January 1976).

3.7-5 L. K. Liu, "Seismic Analysis of the Boiling Water Reactor", Symposium on Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1971.

3.7-6 N.M. Newmark, "Design Criteria for Nuclear Reactors Subject to Earthquake Hazards", Proc IAEA Panel on Aseismic Design and Testing of Nuclear Facilities, Japan Earthquake Engineering Promotion Society, Tokyo, Japan (1967).

LIMERICK
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3.9 Mechanical Systems and Components

QUESTION NO.	FSAR SECTION	TECHNICAL AREA	RESPONSIBLE ORGANIZATION
45	3.9.1.1.1	CRD Trans.	GE
46	3.9.1.1.1	CRD Hsg. Trans.	GE
47	3.9.1.1.3	HCU-OBE Cycles	GE
48	3.9.1.1.5	M.S. Trans (Startup/ Shutdown Cycles)	GE
49	3.9.1.1.2-11	Normal/Upset Trans.	GE
50	3.9.1.1.9	SRV-Pool Cycles	GE/B
51	3.9.1.1.9	SRV-Scram Cycles	GE
52	3.9.1.2	Comp. Programs	GE/B
53	3.9.1.3	Exp. Stress Anal.	GE
54	3.9.1.4	Elast.-Plastic Anal.	GE/B
55	3.9.1.4.1	CRD Tests	GE
56	3.9.2	Piping Vib. Program/ FW Cracking (NUREG 0619)	GE/B/PECO
57	3.9.2.1	Level 1 & 2 Criteria	GE/B
58	3.9.2.1.b	Plant Des.	B
59	3.9.2.4	Prototype Reactor	GE
60	3.9.2.5	LOCA + SSE	GE
61	3.9.2.1a.3	Snubbers-Vib. Control	GE/B
62	3.9.2.1	Vib. & Pre-op Test Crit.	B/GE
63	3.9.3	AP Descrip. (NUREG 0609)	GE/B
64	3.9.3.1	Piping Func. Capability (NEDO 21985)	GE/B
65	3.9.3.1.6	Recirc. Pump	GE
66	3.9.3.1	NL-LC (NUREG 0800)	GE
67	3.9.3.1	SRV Lines-Fatig. Anal. (Quenchers)	B
68	3.9.3.1, T3.9-6	NL-LC & AC Table	GE

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QUESTION NO.	FSAR SECTION	TECHNICAL AREA	RESPONSIBLE ORGANIZATION
69	3.9.3.1, T3.9-6	NL Schedule	GE/B
70	3.9.3.3.2	Plant Des.	B
71	3.9.3.4.1	Supp. Eval. (High Cyc. Fat.)	B/GE
72	3.9.3.4.1	Bolts Allowables	GE/B
73	3.9.3.4	Component Support Primary & Secondary Stresses	GE/B
74	3.9.3.3	PR. Relief Devices	B/GE
75	3.9.3.4	NF Boundaries	B/GE
76	3.9.3.4	Buckling Crit.-Comp. Supp. & RPV Skirt	GE/B
77	3.9.3.4.1	Snubbers-Strength	B/GE
78	3.9.3.4.1	Snubbers-Spec.	B
79	3.9.3.4.1	Snubbers-Spec.	B
80	3.9.3.4.1	Snubbers-Spec.	B
81	3.9.3.4.1	Snubbers-Spec.	B
82	3.9.3	Plant Des.	B
83	3.9.3.1, T3.9-6	NL-LC & AC Table	GE/B
84	3.9.4.2	CRD Comp.	GE
85	3.9.5	Jet Pump Beam	GE
86	3.9.5.1	RPV Int.	GE
87	3.9.6.1	Plant Des.	B
88	3.9.6.1	Plant Des.	B
89	3.9.6.1	Plant Des.	B

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QUESTION NO. 45
(3.9.1.1.1, Page 3.9-2)

Explain the absence of upset and emergency category transients for the control rod drive.

RESPONSE

The transient categories are added to Section 3.9.1.1.1.

3.9.1.1.1 Control Rod Drive (CRD) Transients

The normal and test service load cycles used for design purposes in the 40-year life of the CRD are as follows:

	Transient	Category	Cycles
a.	Reactor startup and shutdown	Normal/Upset	120
b.	Vessel pressure tests	Normal/Upset	130
c.	Vessel overpressure	Normal/Upset	10
d.	Scram test plus startup scrams	Normal/Upset	300
e.	Operational scrams	Normal/Upset	300
f.	Jog cycles	Normal/Upset	30,000
g.	Shim/drive cycles	Normal/Upset	1,000

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In addition to the above cycles, the following have been considered in the design of the CRD:

	Transient	Category	Cycles
j	OBE*	Upset	10
k	SSE	Faulted	1
h.	Scram with inoperative buffer	Normal/Upset	10
i.	Scram with stuck control blade	Normal/Upset	1

All ASME Class 1 components of the CRD have been analyzed according to ASME Section III B&PV Code. The capability of CRD to withstand other emergency and faulted conditions is verified by test rather than analysis.

* The frequency of occurrence of this transient indicates the emergency category. However, for conservatism, the OBE was analyzed as an upset condition. Ten peak stress cycles are postulated.

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QUESTION NO. 46
(3.9.1.1.2, Page 3.9-2)

Justify the differences between the number of transients for the control rod drive and the control rod drive housing.

RESPONSE

The worst case load on the CRD housing is thermal, while the worst case load on CRD is mechanical. Per plant design requirements, at least 200 scram cycles are considered. However, the CRD design has considered more than 200 cycles for additional conservatism.

Also, see the revised Section 3.9.1.1.2 and the response to Question No. 45.

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3.9.1.1.2 CRD Housing and Incore Housing Transients

Transients classifications and number of their cycles considered in the CRD housing and incore housing design and fatigue analysis are as follows:

	Transient	Category	Cycles
a.	Normal startup and shutdown	Normal/Upset	120
b.	Vessel pressure tests	Normal/Upset	130
c.	Vessel overpressure tests	Normal/Upset	~~70~~ 10
d.	Interruption of feedwater flow	Normal/Upset	80
e.	Scram	Normal/Upset	200
f.	Operating basis earthquake (OBE) ~~(1)~~	Normal/Upset	~~1~~ 10
g.	Safe shutdown earthquake (SSE) ~~(2)~~	~~Emergency~~ Faulted	1
	CRD Housing Only		
h.	Stuck rod scram	Normal/Upset	1
i.	Scram with no buffer	Normal/Upset	~~1~~ 10

(1) The frequency of this cycle indicates an emergency category. However, for conservatism, this OBE condition is analyzed as an upset.

(2) SSE is a faulted condition; however, in the stress analysis report, it is treated as an emergency with lower stress limits.

DRAFT

QUESTION NO. 47
(3.9.1.1.3, Page 3.9-4)

Justify using only one OBE cycle for the hydraulic control unit.

RESPONSE

This is a typographical error, see attached revision of Section 3.9.1.1.3.

3.9.1.1.3 Hydraulic Control Unit Transients

	Transient	Category	Cycles
a.	Reactor startup and shutdown	Normal/Upset	120
b.	Scram tests	Normal/Upset	300
c.	Operational scrams	Normal/Upset	300
d.	Jog cycles	Normal/Upset	30,000
e.	Scram with stuck scram discharge valve	Emergency	1
f.	OBE	Upset	~~1~~ 10
g.	SSE	Faulted	1

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3.9.1.1.4 Core Support and Reactor Internals Transients

Cycles considered in the reactor internals design and fatigue analysis are listed in Table 3.9-2.

3.9.1.1.5 Main Steam System Transients

Transients considered in the main steam piping stress analysis are as follows:

	Transient	Category	Cycles
a.	Startup	Normal	120
b.	Loss of feedwater pumps, isolation valves closed	Upset	10
c.	Scram	Upset	180
d.	Shutdown	Normal	111
e.	Hydrostatic test	Test	3
f.	Design hydrotest	Test	130
g.	Operating basis earthquake (OBE)	Upset	50
h.	Turbine stop valve closure (TSV)	Upset	120
i.	Relief valve lift (RVL) (at 3 cycles per actuation)	Upset	34,200

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QUESTION NO. 48
(3.9.1.1.5, Page 3.9-4)

Why are there 120 startups and 111 shutdowns?

RESPONSE

In the design of NSSS piping system, 120 startup transients and 111 shutdown transients are considered as defined in Sections 3.9.1.1.5 and 3.9.1.1.6. Out of the 9 transients not counted for the shutdown, 8 are due to SRV blowdown and 1 due to automatic depressurization.

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QUESTION NO. 49
(3.9.1.1.2-11, Pages 3.9-3 to 8)

Why do many of the transients listed have two classifications, i.e., normal/upset?

RESPONSE

Whenever a transient is categorized with two classifications, i.e., normal/upset, the most limiting of the two is considered in the design.

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QUESTION NO. 50
(3.9.1.1.9, Page 3.9-6)

How many cycles due to suppression pool dynamics are included in the analysis?

RESPONSE

In the Limerick RPV, RPV internals and piping New Loads Adequacy Evaluation, at least 7700 SRV cycles are considered to account for the pool dynamic loads.

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QUESTION NO. 51
(3.9.1.1.9, Page 3.9-6)

Why are only 180 scram cycles considered?

RESPONSE

The 180 scram cycles, including scram due to turbine trips, were chosen for design based on plant operating data available and projected at the time of the Limerick plant design. Most recent operating plant history from 21 operating reactors shows an average of 2.3 scrams per year due to turbine trip and 2.7 scrams per year for other reasons (200 scrams total for 40 years). The low average of scram was 0.9 per year due to turbine trip and 2.3 per year for other reasons (128 total). Since the data for the 21 plant included the early time period when plant shakedown occurs, the low average value of 128 scrams is most representative of the total scrams expected during the service life of the reactor. For conservatism, 180 scram cycles are used.

The text is revised to show the correct transient category.

c.	Shutdown (100°F/hr, pressure decrease to 0 psig, 270°F/hr between 375°F and 330°F)	Normal/Upset	120
d.	Scram	Normal / Upset ~~Emergency/Faulted~~	180 MEB-51
e.	System pressure and temperature decay is from 1000 psig and 546°F, to 35 psig and 281°F within 15 seconds.	Emergency/Faulted	1
f.	System temperature change is from 546°F to 375°F within 3.3 minutes, and from 375°F to 281°F at a rate of 300°F/hr. Pressure change is from 1000 to 35 psig.	Emergency/Faulted	1
g.	System temperature change is from 546°F to 375°F within 10 minutes, and from 375°F to 281°F, at a rate of 100°F/hr. Pressure change is from 1000 to 35 psig.	Emergency/Faulted	8
h.	System temperature change is from 546°F to 583°F within 2 seconds, from 583°F to 538°F within 30 seconds, and from 538°F to 400°F with return to 546°F at a rate of 100°F/hr. Pressure change is from 1000 to 1350 psig, thence to 240 psig, with return to 1000 psig.	Emergency/Faulted	1
i.	System temperature changes greater than 30°F, are from 561°F to 500°F within 7 minutes, and from 500°F to 400°F, with return to normal operating temperature of 546°F, at a rate of 100°F/hr. Pressure change is from 1000 to 1180 psig, to 240 psig, with return to normal operating pressure of 1000 psig.	Emergency/Faulted	10

Paragraph NB3552 of the ASME III Code excludes various transients, and provides means for combining those which are not

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QUESTION NO. 52
(3.9.1.2, Page 3.9-9)

In order for the staff to assess the applicability and validity of computer programs used in dynamic and static analyses of seismic Category I Code and non-Code items, the following information is required:

a) The author, source, dated version and facility.

b) A description, and the extent and limitation of its application.

c) The computer program 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 calculation

2. analytical results published in the literature

3. acceptable experimental tests

4. by an MEB acceptable similar program

5. the benchmark problems found in NUREG/1677

RESPONSE

The NSSS programs can be divided into two categories:

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output and methodology is documented in GE Design Record Files.

a.	PIPST01	j.	FAP 71	s.	PDA
b.	MASS	k.	CREEP PLAST	t.	EZPYP
c.	SNAP (MULTISHELL)	l.	ANSYS	u.	LION4
d.	GASP	m.	SAP4	v.	SIMOK
e.	NOHEAT	n.	ANSI-7	w.	DISPL
f.	FINITE	o.	NOZAR	x.	WTNOZ
g.	DYSEA	p.	TSFOR	y.	SPECA04
h.	SHELL 5	q.	RVFOR	z.	GEAPL01
i.	HEATER	r.	PISYS	aa.	POSUM
				ab.	FTFLG

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. 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.

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Pump Motor Vendor Programs

a. RTRMEC

CB&I Programs

a. 7-11-GENOZZ
b. 9-48-NAPALM
c. 1027
d. 846
e. 781-KALNINS
f. 979-ASFAST
g. 766-TEMAPR
h. 767-PRINCESS
i. 928-TGRV
j. 962-E0962A
k. 984
l. 992-GASP
m. 1037-DUNHAM'S
n. 1335
o. 1606 & 1657-HAP
p. 1634N

In accordance with the response, the FSAR is revised.

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3.9.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves.

	Transient	Cycles
a.	50-575-50°F at 100°F/hr	300
b.	±29°F between limits of 50°F and 575°F, instantaneous	600
c.	±50°F between limits of 50°F and 546°F, instantaneous	200
d.	546°F to 375°F, instantaneous	30
e.	546°F to 281°F, instantaneous	2
f.	130°F to 546°F, instantaneous	1
g.	110% design pressure at 575°F	1
h.	1300 psi at 100°F installed hydrostatic test	130
i.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific nuclear steam supply system (NSSS) components (computer programs were not used in the analysis of all components, thus, not all components are listed). The computer programs can be divided in to two categories: MEB-52

Computer programs are maintained either by GE or by outside computer program developers. In either case, the quality of the programs and the computed results are controlled. One or more engineers are assigned to each program. Duties are:

a. To keep abreast of the capability, the software contents and the theory of the program

b. To run test cases and maintain the reliability of the program

c. To 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, users' confusion over changes is avoided, and the high reliability of these programs is maintained.

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GE Programs

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The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output and methodology is documented in GE Design Record Files.

a. PIPST01
b. MASS
c. SNAP (MULTISHELL)
d. GASP
e. NOHEAT
f. FINITE
g. DYSEA
h. SHELL 5
i. HEATER
j. FAP 71
k. CREEP PLAST
l. ANSYS
m. SAP4
n. ANSI-7
o. NOZAR
p. TSFOR
q. RVFOR
r. PISYS
s. PDA
t. EZPYP
u. LION4
v. SIMOK
w. DISPL
x. WTNOZ
y. SPECA04
z. GEAPL01
aa. POSUM
ab. FTFLG

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. 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.

Pump Motor Vendor Programs

a. RTRMEC

CB&I Programs

a. 7-11-GENOZZ
b. 9-48-NAPALM
c. 1027
d. 846
e. 781-KALNINS
f. 979-ASFAST
g. 766-TEMAPR
h. 767-PRINCESS
i. 928-TGRV
j. 962-E0962A
k. 984
l. 992-GASP
m. 1037-DUNHAM'S
n. 1335
o. 1606 & 1657-HAP
p. 1634N

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3.9.1.2.1 Reactor Vessel and Internals MEB-52

3.9.1.2.1.1 Reactor Vessel

The computer programs used in the preparation of the reactor vessel stress report are identified, and their use summarized in the following paragraphs.

3.9.1.2.1.1.1 Chicago Bridge and Iron (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 the ASME Code, Section III, and contract documents. The program either designs such a configuration or analyzes the configuration input to comply to code. If the input configuration does not comply with the specifications, the program modifies the design and redesigns it to yield an acceptable result.

3.9.1.2.1.~~2~~1.2 CB&I Program 9-48 - NAPALM

The basis for the Nozzle Analysis Program--All Loads Mechanical (NAPALM) is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program provides analyses at each mechanical load point of application. The maximum stress intensity is calculated for both the inside and outside surfaces at each reference location. The program measures the maximum stress intensity for both the inside and outside surfaces of the nozzle, as well as their angular locations as measured from the 0° reference location. The principle stresses are also listed. Stresses resulting from each component of loading (bending, axial, shear, and torsion) are listed, as well as the loadings which cause these stresses.

3.9.1.2.1.~~3~~1.3 CB&I Program 1027

This program is a computerized version of the analysis method contained in Ref 3.9-1.

Part of this program provides for the determination of the shell stress intensities (S) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. Eight S values are calculated, one at each of four cardinal points, for both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces). With the determination of each S, the components of that S (two normal stresses, σ_x and σ_y, and shear stress τ) are also determined. This program provides the same information as the manual calculation, and the input data is essentially the geometry of the vessel and attachment.

3.9.1.2.1.~~4~~1.4 CB&I Program 846

This program computes the required thickness of a hemispherical head with a large number of circular parallel penetrations, by means of the area replacement method, in accordance with the ASME

Code, Section III.

In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

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3.9.1.2.1.~~5~~ 1.5 CB&I Program 781 - KALNINS

The KALNINS thin shell program is used to establish the shell influence coefficient, and to perform the detailed stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axisymmetric loading:

a. Preload condition

b. Internal pressure

c. Thermal load

This program is a thin elastic shell program for shells of revolution developed by Dr. A. Kalnins of Lehigh University. Extensive revisions and improvements have been made by Dr. J. Endicott, to yield the CB&I version of this program.

The basic method of analysis was published by Professor Kalnins (Ref 3.9-2).

3.9.1.2.1.~~6~~ 1.6 CB&I Program 979 - ASFAST

The ASFAST program performs the stress analysis of axisymmetric, bolted closure flanges between the head and cylindrical shell.

3.9.1.2.1.~~7~~ 1.7 CB&I Program 766 - TEMAPR

This program reduces any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient has the same average temperature, and the same temperature-moment as the given temperature gradient. The input consists of the wall thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV.

3.9.1.2.1.~~8~~ 1.8 CB&I Program 767 - PRINCESS

The PRINCESS program calculates the maximum alternating stress amplitudes from a series of stress values, by the method in Section III of the ASME B&PV Code.

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3.9.1.2.1.~~8~~ 1.9 CB&I Program 928 - TGRV

The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel.

The TGRV program is a highly modified version of the TIGER heat transfer program, written about 1958 at Knolls Atomic Power Laboratory, by A.P. Bray.

The program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments, and continually summed. Linear mathematics is used to solve the mesh network for every time interval. Three basic forms of heat transfer (conduction, radiation, and convection), as well as internal heat generation, are included in the analysis.

TGRV calculates and outputs the steady state or transient temperature distributions in a given structure, as a function of time. The program inputs are any odd-shaped structure which can be represented by a three-dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates.

3.9.1.2.1.~~10~~ 1.10 CB&I Program 962 - E0962A

Program E0962A is one of a group of programs (E0953A, E1606A, E0962A, E0992N, E1037N, and E0984N) which are used together to determine the temperature distribution and stresses in pressure vessel components, using the finite element method.

Program E0962A is primarily a plotting program. Using the nodal temperatures calculated by program E1606A or Program E0928A, and the node and element cards for the finite element model, it calculates and plots lines of constant temperature (isotherms). These isotherm plots are used as part of the stress report to present the results of the thermal analysis. They are also useful in determining at which points in time the thermal stresses should be determined.

In addition to its plotting capability, the program can also determine the temperatures of some of the nodal points by interpolation. This feature of the program is intended primarily for use with the compatible TGRV and finite element models that are generated by program E0953A.

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3.9.1.2.1.~~11~~ 1.11 CB&I Program 984

Program 984 is used to calculate the stress intensity of stress differences, on a component level, between two different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by Section III of the ASME Code.

3.9.1.2.1.~~12~~ 1.12 CB&I Program 992 GASP

The GASP program, originated by Professor E.L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane or axisymmetric structures of arbitrary geometry, and is written in FORTRAN IV. See Ref 3.9-3, for a detailed account.

GASP structures may have arbitrary geometry, and have linear or nonlinear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

A structure to be analyzed is broken up into a finite number of discrete elements or "finite-elements", which are interconnected at a finite number of "nodal-points" or "nodes." The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress-model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

3.9.1.2.1.~~13~~ 1.13 CB&I Program 1037 - DUNHAM'S

DUNHAM'S program is a finite ring element stress analysis program. It determines the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads, or nonaxisymmetric loads represented by a Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle nonaxisymmetric loads (which requires that each node have three degrees of freedom), while the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.~~14~~ 1.14 CB&I Program 1335

To obtain stresses in the shroud support, the baffle plate must be made a continuous circular plate. The program makes this modification and allows the baffle plate to be included in CB&I program 781 (KALNINS) as two isotropic parts, with an orthotropic portion at the middle (where the diffuser holes are located).

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3.9.1.2.1.~~15~~ 1.15 CB&I Programs 1606 and 1657 - HAP

The HAP program is an axisymmetric nonlinear heat analysis program. It is a finite element program, used to determine nodal temperatures in a two-dimensional or axisymmetric body subject to transient disturbances. Programs 1606 and 1657 are identical, except that 1606 has a larger storage area allocated, and can thus be used to solve larger problems. The model for program 1606 is compatible with CB&I stress programs 992 (GASP) and 1037 (DUNHAM'S). INSERT 2

3.9.1.2.1.16 CB&I Program 1635

Program 1635 offers the following three features to aid the stress analyst in preparing a stress report:

a. It generates punched card input for program 767 (PRINCESS) from the stress output of program 781 (KALNINS).

b. It writes a stress table in a format that can be incorporated into a final stress report.

c. It has the option to remove through-wall thermal bending stress, and report these results in a stress table similar to the one mentioned above.

3.9.1.2.1.17 CB&I Program 953

This program is a general purpose program, which does the following:

a. It prepares input cards for the thermal model.

b. It prepares the node and element cards for the finite element model.

c. It sets up the model in such a way that the nodal points in the TGRV model correspond to points in the finite element model. They have the same number, so that there is no possibility of confusion in transferring temperature data from one program to the other.

3.9.1.2.2 Piping

The computer programs used in the analysis of NSSS piping systems within GE's scope of supply are identified, and their use summarized in the following paragraphs.

3.9.1.2.1.1.16 CB&I Program 1634N

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This program is used to analyze thin cylindrical shells subjected to local loading beyond the range where Bijlaard's curves are directly applicable, i.e., R/t >300.

This program computes stress and displacements in thin-walled elastic cylindrical shells subjected to mechanical loading such as radial loads, longitudinal and circumferential moments.

3.9.1.2.1.2 Reactor Internals

3.9.1.2.1.2.1 Core Plate Beam Buckling - PIPST01

PIPST01 is a computer program which calculates approximate core plate beam buckling capability. It uses the Rayleigh-Ritz energy method to determine the applied moment needed to begin yielding and then finally to buckle a given tee beam. The tee beam models a segment of a BWR/2-5 core plate with a stiffener beam. The pressure differential across the plate that would have created this moment is calculated for a given length of beam or size of core plate.

Generic dimension and material properties are all input by the user.

3.9.1.2.1.2.2 Other Programs

Other computer codes used for the analysis of the internal components are:

a. MASS
b. SNAP (MULTISHELL)
c. GASP
d. NOHEAT
e. FINITE
f. DYSEA
g. SHELL 5
h. HEATER
i. FAP 71
j. CREEP PLAST
k. ANSYS

Detailed descriptions of these programs are given in Section 4.1.4.1.

3.9.1.2.2 Piping

3.9.1.2.2.1 Structural Analysis Program - SAP 4

SAP 4 is a general Structural Analysis Program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements, and to compute the stresses of each element of the structure. The structure can be composed of unlimited numbers of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress, brick, thick shell, spring, or axisymmetric elements. The program can treat thermal and various forms of mechanical loading, as well as internal element loading. The dynamic analysis includes mode superposition, time history, and response spectrum analyses. Earthquake loading, as well as time-varying pressure, can be treated. The program is very versatile and efficient in solving large and complex structural systems. The output contains displacements of each nodal point, as well as stresses at the surface of each element.

3.9.1.2.2.2 Component Analysis - ANSI-7

The ANSI-7 Computer Program determines stress and accumulative usage factors in accordance with NB-3600 of the ASME Code, Section III. The program performs stress analyses in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.2.3 Area Reinforcement - NOZAR

The computer program Nozzle Area Reinforcement Program (NOZAR) performs an analysis of the required reinforcement area for openings. The calculations performed by NOZAR are in accordance with the rules of the 1974 edition of Section III of the ASME Code.

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3.9.1.2.2.4 Dynamic Forcing Functions

~~3.9.1.2.2.4.1 Relief Valve Discharge Pipe Forces~~

~~See Section 3.9.1.2.6.5 for descriptions of the computer programs used to analyze relief valve discharge pipe forces.~~

3.9.1.2.2.4.~~2~~1 Turbine Stop Valve Closure - TSFOR

The TSFOR program computes the time history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

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3.9.1.2.2.4.2 Relief Valve Discharge Pipe Forces Computer Program/RVFOR

The relief valve discharge pipe connects the relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time dependent forces to develop in the pipe wall. This computer program computes the transient fluid mechanics and the resultant pipe forces using the method of characteristics.

3.9.1.2.2.5 Piping Analysis Program/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include: distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option of response spectrum analysis and the results are documented in a report to the Commission, "PISYS Analysis of NRC Benchmark Problems", NEDO-24210, August 1979.

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3.9.1.2.2.5 Integral Attachment - LUGST

The LUGST program evaluates stresses in the pipe wall that are produced by loads applied to the integral attachments. The program was prepared on the basis of Welding Research Council Bulletin 198.

3.9.1.2.2.6 Piping Dynamic Analysis Program - PDA

The pipe whip analysis was performed using the PDA computer program. PDA is used to determine the response of a pipe subjected to the thrust force occuring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end, subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used to model the pipe and the restraint. Similar to the popular elastic-hinge concept, bending of the pipe is assumed to occur at the fixed end, and at the location supported by the restraint, only.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using moment-rotation relations, nonlinear equations of motion are formulated using energy considerations, and the equations are numerically integrated in small time steps to yield the time-history of the pipe motion.

3.9.1.2.2.7 Piping Analysis Program - EZPYP

EZPYP links the ANSI-7 and SAP programs together. The EZPYP program can be used to run several SAP cases by making user specified changes to a basic SAP pipe model. By controlling files and SAP runs, the EZPYP program makes it possible to perform a complete piping analysis in one computer run.

3.9.1.2.2.8 Thermal Transient Program - LION4

The LION4program is used to compute radial axialthermal gradients in piping. The program calculates a time history of ΔT_1,ΔT_2, Ta, and Tb (defined in ASME Section III, Class 1 piping analysis) for uniform and tapered pipe wall thickness.

3.9.1.2.2.9 Synthetic Time History Program - SIMOK

The SIMOK program provides a time history that is equivalent to an input response spectrum. The synthetic time history is used to generate a new spectrum that is plotted with the input spectrum, to verify that the time history and spectrum are equivalent. Synthetic time histories are used in a multiple input analysis of the piping.

3.9.1.2.2.10 Differential Displacement Program - DISPL

The DISPL program provides differential movements at each piping attachment point, based on building modal displacements.

3.9.1.2.2.11 WTNOZ Computer Program

WTNOZ is a timeshare program for piping weight calculations. MEB-52

3.9.1.2.3 ~~Recirculation~~ Pumps and Motors

3.9.1.2.3.1 Recirculation Pumps

No computer programs were used in the design of the recirculation pumps.

3.9.1.2.~~4~~ 3.2 Core Spray Pumps and Motors

The RTRMEC computer program is used in the analysis of a motor design representative of (or very similar in mechanical construction to) the core spray pump motor.

RTRMEC calculates and displays the results of a mechanical analysis of a motor rotor assembly acted upon by external forces at any point along the shaft (rotating parts only). The shaft deflection analysis, including magnetic and centrifugal forces, was conducted. The calculation for the seismic condition assumes that the motor is operating, and that the seismic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor-shaft assembly. Note that the distributed motor assembly weight is lumped at the various stations. The shaft weight at a station is the sum of one-half the weight of the incremental shaft length just before the station, plus one-half the weight of the adjacent incremental shaft length just after the station. Bending and shear effects are accounted for in the calculations.

3.9.1.2.5 Residual Heat Removal (RHR) Heat Exchangers

The following are the computer programs used in dynamic and static analysis to determine the structural and functional integrity of the RHR heat exchangers. MEB-52

a. Support Load Seismic Analysis (ED-6)

This program computes the total loads at the upper and lower supports of the RHR heat exchanger. This program takes into account the heat exchanger flooded weight, seismic loads (either OBE or SSE), and the allowable nozzle loads; and sets up the worst combination of these loads. By maximizing seismic loads together with nozzle

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The FTFLG computer program was used to analyze the flange joints connecting the pump bowl castings. The description of this program is provided in Subsection 3.9.1.2.5.3.

3.9.1.2.4 Dynamic Loads Analysis

3.9.1.2.4.1 Acceleration Response Spectrum Program/SPECA04

The SPECA04 computer program generates acceleration response spectrum, consistent with R.G. 1.122 for an arbitrary input of time history of piecewise linear accelerations, i.e., to compute maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time histories from a random file. It also has the capability of generating the broadened/enveloping spectra in conformance with R.G. 1.122 when the spectral points are generated equally spaced on a logrithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analysis.

3.9.1.2.4.2 Forces and Moment Time-Histories Program/GEAPL01

The GEAPL01 computer program converts distributed asymmetric pressure time histories over a given area into equivalent time varying nodal forces and moments for use as input to perform dynamic analysis of a system. The overall resultant forces and moment-time histories at specified points of resolution can also be obtained from GEAPL01.

3.9.1.2.5 Residual Heat Removal (RHR) Heat Exchangers

3.9.1.2.5.1 Structural Analysis Program - SAP4

SAP4 is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Subsection 3.9.1.2.2.1.

3.9.1.2.5.2 Beam Element Data Processing Program/POSUM

POSUM is used to process SAP4 generated data. POSUM is a computer code designed to process SAP4 generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended for use on RHR heat exchangers with four nozzles or core spray pumps with two nozzles.

3.9.1.2.5.3 Effects of Flange Joint Connections/FTFLG

The flange joints connecting the pump bowl castings are analyzed using FTFLG program. This program uses the local forces and moments determined by SAP4 to perform flat flange calculations in accordance with the rules set forth in Appendix II and Section III of ASME Code.

loads, maximum conservative moments and forces at the upper and lower supports are calculated.

b. Stress Analysis of Supports (ED-8)

This program performs a full stress analysis of the upper and lower supports of the RHR heat exchanger. The stresses in the supports (both upper and lower) caused by loads resulting from seismic and nozzle loads are computed in the support load program (ED-6), and are used as input values for this program. This program computes the membrane stresses on the shell of the heat exchanger by using Bijlaard's analysis, as well as the net section stresses (shear, tensile, bearing) on the lower support plate and upper lugs. It also computes the stresses on the welds holding the supports to the shell of the heat exchanger.

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3.9.1.2.6 Seismic Category I Items Other than NSSS

A list of computer programs used in the non-NSSS system components is provided in Table 3.9-3. This list consists of computer programs developed and/or owned by Bechtel Power Corporation (BPC), and of computer programs that are recognized and widely used in industry.

The Bechtel developed and/or owned computer programs are documented, verified, and maintained by Bechtel, and meet the requirements of 10 CFR, Part 50, Appendix B. A brief description of each of these Bechtel programs is provided below.

3.9.1.2.6.1 ME101, Linear Elastic Analysis

Program Description: ME101 is a finite element computer program that performs linear elastic analyses of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically.

The output may be used directly for piping design, for conformation to code, and for other regulatory requirements. Two piping codes, ASME B&PV Code 1974 and B31.1 Summer 1973 addenda, are incorporated into the program to the extent of computing flexibility factors, stress intensification factors, and stresses. ME101 may be used for static and seismic analysis of piping systems. Static analysis considers one or more of the following: thermal expansion, deadweight, uniformly distributed loads, externally applied loads (forces, moments, displacements, and rotations). Seismic analysis is based on standard normal

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QUESTION NO. 53
(3.9.1.3, Page 3.9-23)

Provide assurance that all experimental stress analysis performed on seismic Category I Code or non-Code items meets provisions of Appendix II of Section III of the ASME Code.

RESPONSE

Experimental stress analysis is not used for Limerick seismic Category I code or non-code systems and components. Accordingly, Section 3.9.1.3.1 is revised to reflect this response.

3.9.1.3.1 Experimental Stress Analysis of ~~Piping Components~~ NSSS Seismic Category I Items

No experimental stress analysis methods are used.

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

~~a. Piping seismic shock suppressors~~

~~b. Pipe whip restraints~~

~~Descriptions of the support and whip restraint tests are contained in Sections 3.9.3.4 and 3.6, respectively.~~

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~~3.9.1.3.2 Orificed Fuel Support: Vertical and Horizontal Load Tests~~

~~A series of vertical and horizontal load tests were performed on the Orificed Fuel Support (OFS) to verify the design. Results from these tests indicate that the component and seismic loading of the OFS are well below the stress limit allowances, with a safety margin of 1.26 for normal and upset conditions, and of 1.5 for the faulted condition. (The allowable stress limits were arrived at by applying a 0.65 quality factor to the ASME code allowables of 1.5 Sm for upset and 1.5 X .7Su for faulted.)~~

~~3.9.1.3.3 Control Rod Drive~~

~~(Not applicable.)~~

3.9.1.3.~~4~~2 Seismic Category Items Other Than NSSS

No experimental stress analysis methods are used.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

All seismic Category I equipment is evaluated for the faulted loading conditions. However, emergency stress limits rather than faulted stress limits are used in many cases. In all cases, actual stresses are within the code specified limits. The following paragraphs in this section show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in Sections 3.9.3 and 3.9.5, and Table 3.9-6.

Sections 3.9.2.2 and 3.7 discuss the treatment of dynamic loads resulting from the postulated SSE. Section 3.9.2.5 discusses the dynamic analysis of loads on NSSS equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas, and no cases have been identified where design limits, such as clearance limits, are violated.

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QUESTION NO. 54
(3.9.1.4, Page 3.9-24)

Provide details of any elastic-plastic analysis you may have used in evaluating seismic Category I equipment for Service Level D Limits.

RESPONSE

Elastic-plastic analysis has not been used in evaluating the Limerick's seismic Category I systems and components for compliance with service Level D Limits. The stress levels of these components are below the ASME allowable stress.

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QUESTION NO. 55
(3.9.1.4.1, Page 3.9-24)

Provide details of references of testing done in lieu of analysis on control rod drives.

RESPONSE

To verify the CRD performance, three types of tests were performed: (a) oscillatory displacement of lower CRD housing, (b) displacement of core support structure, and (c) fuel channel deflection.

a) A test was conducted with the lower CRD flange oscillating with a 2 inch peak-to-peak displacement. No adverse effects were observed during the normal continuous drive-in or jog operation.

b) To simulate the seismic interaction, the core plate and top guide structures of the test vessel were displaced relative to the CRD housing center line. The results showed no effect in CRD performance.

c) The test vessel fuel channels were deflected to simulate the seismic interactions. The test was performed with fuel channel deflections up to 1.5 inches which are greater than the expected deflection values. Since the CRD and control rod were not permanently deformed, the drive operability was maintained.

The load criteria, calculated and allowable stresses for various operating conditions will be summarized in Table 3.9-6v upon completion of New Loads Adequacy Evaluation (NLAE) program.

See revised Subsection 3.9.1.4.1.1.

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3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives

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The ASME Section III Code components of the CRD have been analyzed for abnormal conditions ~~h. and i.~~ shown in Section 3.9.1.1.1. ~~The loads and stresses are within the elastic limits of the material.~~ The loading criteria, calculated and allowable stresses for various operating conditions are summarized in Table 3.9-6V.
~~No analysis has been made for the non-code components of the CRD for the abnormal condition~~. The design adequacy of non-code components of the CRD has been verified by analysis and extensive testing programs on component parts, specially instrumented prototype drives, and production drives. The testing included postulated abnormal events, as well as the service life cycle listed in Section 3.9.1.1.1.

3.9.1.4.1.2 Hydraulic Control Unit

The hydraulic control unit (HCU) was analyzed for the SSE faulted condition. The analysis of the HCU under faulted condition loads establishes the structural integrity of the system.

3.9.1.4.1.3 CRD Housing

DRAFT

QUESTION NO. 56
(3.9.2)

1. 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 startup 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 but required for scram).

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

RESPONSE

1. Vibration Monitoring Program

   A. Instrumentation and Small Lines

   The instrumentation lines identified in the question are included in the Limerick Vibration Monitoring Program. The piping was evaluated as having very low potential for steady state vibration, based on a well-supported seismic design configuration and on favorable (no-flow) environmental vibration conditions. None of the conditions needed for direct independent excitation of vibration throughout these lines is in effect during operation. However, vibratory motion at the junction with the process pipe (NSSS or BOP) or reactor pressure vessel could occur and is monitored by vibration sensors mounted on the process pipe. Wherever feasible, physical inspection for vibration will be made by the test engineer during the preoperational phase of startup testing.

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B. NSSS Piping (Main Steam and Recirculation)

Flow Transients

As currently documented in Section 3.9.2.1a.1.4, the main steam and recirculation piping systems are tested for the following operating flow transients:

a. Recirculation pump starts;

b. Recirculation pump trip at 100% rated flow;

c. Turbine stop valve closure at 100% power; and

d. Manual discharge of each SRV at 1,000 psig and at planned transient tests that result in SRV discharge.

Locations of Inspections and Devices

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

2. In compliance with NUREG-0619, the Limerick design has incorporated the resolution presented in NEDE-21821, "BWR Feedwater Nozzle/Sparger Final Report", March 1978.

The Limerick feedwater nozzle has been modified. The new configuration is the triple-sleeve with two sister-ring seals and an unclad nozzle. This assures the longest ISI intervals per NUREG-0619.

The CRD return line is not part of the Limerick design.

The above information is incorporated in the revised text.

If the test measurements indicate failure to meet Level 2 criteria, the following corrective actions are taken after completion of the test:

a. Installation Inspection: A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. Snubbers are located close to the midpoint of the total travel range at the operating temperature. Hangers are in their operating range between the hot and cold settings. If the vibration exceeds limits, the source of the vibration must be identified. Actions, such as suspension adjustment, are taken to correct any discrepancies.

b. Instrumentation Inspection: The instrumentation installation and calibration are checked, and any discrepancies are corrected.

c. Repeat Test: If a. or b. above identify a malfunction or discrepancy that could account for failure to comply with Level 2 criteria, and appropriate corrective action is taken, the test may be repeated.

d. Documentation of Discrepancies: If the test is not repeated, the discrepancies found under actions a. or b. above are documented in the test evaluation report and correlated with the test condition. The test is not considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1a.6 Measurement Locations for Main Steam and Recirculation Piping

~~Remote shock and vibration measurements are made in the three orthogonal directions near the first downstream safety/relief valve on each steam line, and in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve. During preoperational testing prior to fuel load, visual inspection of the piping is made, and any visible vibration measured with a hand-held instrument.~~

~~For each of the selected remote measurement locations, Level 1 and 2 deflection and acceleration limits are prescribed in the startup test specification. Level 2 limits are based on the results of the stress report, adjusted for operating mode and instrument accuracy; Level 1 limits are based on maximum allowable code stress limits.~~

Remote vibration measurements during initial startup will be made for each of the main steam lines and recirculation lines. The locations of the measurements will be described in the startup test specification.
During preoperational testing of recirculation piping, visual observation and manual measurements by hand-held vibrograph are made to supplement the remote measurements.

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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 holddown 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.

3.9.5.1.1.9 Steam Dryers

The steam dryer assembly is not a core support structure. 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 and 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 wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head. MEB-56

3.9.5.1.1.10 Feedwater Spargers

The feedwater nozzle and sparger design follows the resolution presented in Ref. 3.9-23.

These components are not core support structures. They are discussed here to describe flow paths in the vessel. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle, and is shaped to conform to the curvature of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers, and is discharged radially inward, mixing the cooler feedwater with the downcomer flow from the steam separators and steam dryer, before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus, and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.1.11 Core Spray Lines

This component is not a core support structure. It is discussed here because the core spray lines are the means for directing flow to the core spray nozzles, which distribute coolant during accident conditions.

3.9-11 E. Wilson, and S. R. Nickell, "Application of the Finite Element Method to Heat Conduction Analysis," Nuclear Engineering and Design, 4 (1966).

3.9-12 E. Wilson, "Structural Analysis of Axisymmetric Solids," AIAA Journal, 3 (112) (December 1965).

3.9-13 P. J. Schneider, Temperature Response Charts, John Wiley and Sons, Inc. (1963).

3.9-14 Sample Analysis of a Class I Piping System, prepared by the Working Group on Piping (SGD, ScIII) of the ASME Boiler and Pressure Vessel Code (December 1971).

3.9.15 Not used.

3.9-16 BWR Fuel Channel Mechanical Design and Deflection, NEDE-21354-P, General Electric Company (September 1976).

3.9-17 BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings, NEDE-21175-P, General Electric Company (November 1976).

3.9-18 Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, NEDE-24057-P (Class III) and NEDO-24057 (Class I), General Electric Company (November 1977).

3.9-19 Design and Performance of G.E. BWR Jet Pumps, APED-5460, General Electric Company, Atomic Power Equipment Department (July 1968).

3.9-20 H. H. Moen, Testing of Improved Jet Pumps for the BWR/6 Nuclear System, NEDO-10602, General Electric Company, Atomic Power Equipment Department (June 1972).

3.9-21 Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K, NEDE-20566, Proprietary Document, General Electric Company.

3.9-22 Stress Report for the Target Rock 6 x 10 Relief Valve, Model # 7567F; 4th Edition, McGraw-Hill, 1965.

3.9-23 Boiling Water Reactor Feedwater Nozzle/Sparger Final Report, NEDE-21821, General Electric Company (March, 1978).

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DRAFT

QUESTION NO. 57
(3.9.2.1, Pages 3.9-29 to 31)

Provide the actual stress limits to be used for both Level 1 and Level 2 of your piping vibration test program.

RESPONSE

NSSS

For steady-state vibration, the piping peak 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 will not exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit ensures 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 predictions. 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 usage factor limits, and allowable loads on snubbers.

These limits for NSSS piping are incorporated in the FSAR as Section 3.9.2.1a.4.3 following the revised definition of Level 1 and Level 2 criteria.

BOP (Non-NSSS)

Vibration stresses will be consistent with the limits of the American National Standard, ANSI/ASME OM3-1982. These limits are based on the piping design fatigue curves for up to 10^6 cycles of vibration given in ASME Section III, Appendix I. To account for fatigue with higher cycles, the design fatigue strength of carbon steels will be reduced by applying a factor of 0.8 and furthermore employing a safety factor of 1.3. Austenitic pipe steels design fatigue strength reduction factor will be 0.6, and is further reduced by employing a safety factor of 1.3. Piping stress indices (K_2C_2) and intensification factors (2i) as applicable to each particular system are also applied in accordance with the standard.

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each size (i.e. 10 kips, 20 kips, 50 kips, etc.) will also be qualified and tested for design and faulted condition loadings, prior to shipment to field. Snubbers will be tested to allow free piping movements at low velocity. During plant startup, the snubbers will be checked for improper settings and checked for any evidence of oil leak.

The criteria for vibration displacements is based on the assumed linear relationship between displacements, snubber loads and magnitudes of applied loads, for any function and response of the system. Thus the magnitudes of limits of displacements, snubber loads, and nozzle loads are all proportional. Maximum displacements (Level 1 limits) are established to prevent the maximum stress in the piping systems from exceeding the normal and upset primary stress limits, and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system. These limits will be compared with the field measured piping displacements. Method of acceptance is as explained in Section 3.9.2.1a.4.

3.9.2.1a.4 Test Evaluation and Acceptance Criteria for Main Steam and Recirculation Piping

The piping response to test conditions is considered acceptable if the organization responsible for the stress report reviews the test results, and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report, and/or that the tests verify that piping stresses are within code limits (ASME Section III, NB-3600). Acceptable deflection ~~and acceleration~~ limits are determined after the completion of the piping systems stress analysis and are provided in the startup test specifications.

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs. MEB 57

~~3.9.2.1a.4.1 Level 1 Criteria~~

~~If, in the course of the tests, measurements indicate that the piping is responding in a manner that makes test termination prudent, the test is terminated. Level 1 criteria establish bounds on movement that, if exceeded, make a test hold or termination mandatory. The limits on movement are based on maximum allowable code stress limits.~~

INSERT

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3.9.2.1a.4.1 Level 1 Criteria

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1a.4.2 Level 2 Criteria

If the Level 2 criteria are satisfied for both steady state and operating transient vibrations, there will be no fatigue damage to the piping system due to steady state vibration and all operating transient vibrations are bounded by the values in the stress report.

Exceeding the Level 2 specified pipe motion requires that the responsible piping design engineer be advised. Plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Detailed evaluation is needed to develop corrective action or to show that the measurements are acceptable. Depending upon the nature of such resolution, the applicable tests may or may not be repeated.

3.9.2.1a.4.3 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 will not exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit ensures 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 predictions. 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 usage factor limits, and allowable loads on snubbers.

3.9.2.1a.4.2 Level 2 Criteria

Conformance with Level 2 criteria demonstrates that the piping is responding in a manner consistent with the stress report predictions. Failure to meet Level 2 criteria does not mean that the piping response is unsatisfactory; it means that the system is not responding in accordance with theoretical predictions, and further analyses based on test results are necessary. Level 2 criteria are intended to screen out test results that are consistent with predictions and need no analytical review, from those that need review.

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3.9.2.1a.5 Corrective Actions for Main Steam and Recirculation Piping

During the course of the tests, the remote measurements are regularly checked to determine compliance with Level 1 criteria. If trends indicate that Level 1 criteria may be violated, the measurements are monitored at more frequent intervals. The test is held or terminated as soon as Level 1 criteria are violated. As soon as possible after the test hold or termination, the following corrective actions are taken:

a. Installation Inspection: A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. Snubbers are located close to the midpoint of the total travel range at the operating temperature. Hangers are in their operating range between the hot and cold settings. If vibration exceeds the criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection: The instrumentation installation and calibration is checked, and any discrepancies are corrected. Additional instrumentation is added, if necessary.

c. Repeat Test: If actions a. or b. identify discrepancies that could account for failure to meet Level 1 criteria, the test is repeated.

d. Resolution of Findings: If the Level 1 criteria are violated on the repeat test, or no relevant discrepancies are identified in a. or b., the organization responsible for the stress report reviews the test results and the criteria to determine if the test can be safely continued.

DRAFT

QUESTION NO. 58
(3.9.2.1b, Page 3.9-34)

Provide a list of sensor type and location and measurement locations for BOP piping vibration, thermal expansion and dynamic effects testing.

RESPONSE

Interim test specifications governing the scope of startup testing of BOP piping are in preparation. These test specifications are intended to be the repository for all primary information relating to the scope, objectives, methods, measurements, and criteria for evaluation of the test results. The BOP piping systems are being categorized in terms of the following:

a. Test environment (hot deflection, steady state vibration or dynamic transient response).

b. Test measurements (remotely monitored, visual or none required due to small expected response to test environment).

c. The appropriate testing phase (preoperational or power ascension).

Table 3.9-7 contains a list of systems included in the test program.

DRAFT

QUESTION NO. 59
(3.9.2.4, Page 3.9-47)

Provide a statement in LGS FSAR explicitly identifying the prototype reactor for LGS.

RESPONSE

The prototype reactor for LGS is the Browns Ferry-3 design docketed on July 31, 1968.

Accordingly, Section 3.9.2.4 is revised.

P The prototype reactor for Limerick is the Browns Ferry-3 design docketed on July 31, 1968. See Ref. 3.9-18.

on the prototype design, including the shroud, shroud head, core support structures, the jet pumps, and the peripheral control rod drive and incore guide tubes. Access is provided to the reactor lower plenum.

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Reactor internals for Limerick are substantially the same as the internal design configurations which have been tested in prototype BWR/4 plants. Results of the prototype tests are presented in a Licensing Topical Report, Ref. 3.9-18. This report also contains additional information on the confirmatory inspection program.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces (Figure 3.9-2), a comparison is made between the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a comprehensive vertical dynamic model of the RPV and internals, with 12 degrees of freedom. Only motion in the vertical direction is considered here; hence, each structural member (between two mass points) can only have an axial load. Besides the real masses of the RPV and core support structures, allowance is made for the water inside the RPV.

Typical curves of the variation of pressures during a steam line break are shown in Figure 3.9-2. The accident analysis method is described in Section 3.9.5.2.

The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis, and is detailed in Section 3.7.2.1. The dynamic components of forces from these loads are combined with dynamic force components from other dynamic loads (including seismic), all acting in the same direction, by the square root of the sum of the squares (SRSS) methods. This resultant force is then combined with other steady-state and static loads on an absolute sum basis to determine the design load in a given direction.

The loads and load combinations acting upon the jet pumps and LPCI coupling are listed in Section 3.9.3.1.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiating the instrumented vibration test program for the prototype plant, extensive dynamic analyses of the reactor

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QUESTION NO. 60
(3.9.2.5, Page 3.9-48)

Verify that the actual loads considered are a LOCA in combination with the SSE.

RESPONSE

The load combination and acceptance criteria tables for ASME Code Class 1, 2 and 3 piping and components have listed LOCA + SSE as one of the load combinations considered. This is documented in Table 3.9-6 (attached to the response to Question No. 68) for NSSS and Table 3.9-11 for BOP.

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QUESTION NO. 61
(3.9.2.1a.3, Page 3.9-29)

List all instances where snubbers are used to control steady-state vibration.

RESPONSE

Snubbers have not been used to control steady-state vibration in the Limerick design.

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QUESTION NO. 62
(3.9.2.1, Page 3.9-29)

Piping vibration, thermal expansion, and dynamic effects testing is done during a preoperational testing program. The purpose of these tests is to assure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the plant's preoperational and startup testing program, the applicant must test various piping systems for abnormal, steady-state or transient vibration and for restraint of thermal growth. Systems to be monitored include 1) ASME Code Class 1, 2 and 3 piping systems, 2) high energy piping systems inside seismic Category I structures, 3) high energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level, and 4) seismic Category I portions of moderate energy piping systems located outside containment. The piping vibration test program must comply with the ASME Code, Section III paragraphs NB-3622.3, and ND-3622.3 which require that the applicant be responsible, by observations during startup or initial operations, for ensuring that the vibration of piping systems is within acceptable levels. This vibration might be due to plant transients or might be associated with steady-state plant operation. This steady-state vibration, whether flow-induced or caused by nearby vibrating machinery, could cause 10^8 or 10^9 cycles of stress in the pipe during its 40-year life.

For this reason, the staff requires that the stresses associated with steady-state vibration be limited to 50% of the alternating stress intensity, S_a at 10^6 cycles as defined in the ASME Code, Appendix I, Figure I-9.1 and I-9.2. In addition, pipe whip restraint initial clearances will be checked as will snubber response. The test program should consist of a mixture of instrumented measurements and visual observation by qualified personnel. The applicant will be required to provide a summary of the results of this test program upon its completion.

Provide assurances that your preoperational testing complies with the above position.

RESPONSE

NSSS

See response to Question No. 57 and revised FSAR Sections attached thereto for piping vibration test programs acceptance limits and Level 1 and Level 2 criteria.

BOP (Non-NSSS)

The startup test program specifications describe in detail for piping which is instrumented for remote monitoring of vibrations and thermal expansion, and piping which is accessible for preoperational or startup walkdown testing by test personnel. The test criteria limit the permissible pipe vibratory stress to the allowable limits prescribed in the industry standard for startup testing of nuclear power systems, ANSI/ASME OM3-1982. Section 3.9.2.1b.2 is revised accordingly.

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QUESTION NO. 62 (CONT'D)

The LGS startup testing program requires that the following conditions be demonstrated per Regulatory Guide 1.70:

1) Thermal expansion is free from significant and unacceptable restraint not accounted for in the design.

2) Piping vibration is within acceptable limits for long term vibratory stress.

3) Dynamic transient response of the piping is within the limits set by the ASME Code design stress analysis.

the allowable limits prescribed in the industry standard for startup testing of Nuclear Power Systems; ANSI/ASME OM3-1982,

acceptable, if the vibration is not significant, or as questionable, if the vibration is significant. The lines with questionable steady-state vibration are monitored as applicable by suitable instrumentation to determine the system response.

The type of any necessary instrumentation is determined by the design engineer, so that the maximum amplitude and frequency response of the piping system can be determined. The instrumentation does not screen out the significant frequencies.

The acceptance criterion for the steady vibration tests is that the maximum measured amplitude of the piping vibration does not induce more stress in the pipe than the endurance limit of the material. By limiting the maximum stress in the pipe due to steady state vibration below ~~the endurance limit (allowable stress corresponding to 10^6 cycles or greater)~~, the steady-state vibration induced stress does not contribute to reducing piping fatigue life. MEB-62

When required, additional restraints are provided to reduce the steady-state vibration, and to keep the stresses below the acceptance criteria levels.

Table 3.9-7 provides a reference to the appropriate test descriptions in Chapter 14.

3.9.2.2a Seismic Qualification of NSSS Safety-Related Mechanical Equipment

This section describes the criteria for seismic qualification of safety-related mechanical equipment, and the qualification testing and/or analysis applicable to this plant for all the major components, on a component by component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit, for example, ECCS pumps. These modules are generally discussed in this section, rather than in Sections 3.10 and 3.11. Electrical supporting equipment, such as control consoles, cabinets, and panels, which are part of the NSSS, are discussed in Section 3.10.

3.9.2.2a.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its function during and after an earthquake is demonstrated by tests and/or analysis. Selection of testing, analysis, or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, the seismic Category I mechanical equipment operations are performed simultaneously with vibratory testing. Where simultaneous testing is not practical, the operation and/or loads are simulated by mathematical analysis, and these loads are applied in addition to the physical test loads.

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QUESTION NO. 63
(3.9.3)

Using the guidance of NUREG-0609, provide the methodology 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 reactor asymmetric loads analysis will be documented in a self-contained appendix to Section 3.9 and/or in the appropriate section of Design Assessment Report (DAR) upon completion.

The following is a brief description of the methodology:

a. 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 COPDA computer code (NE699/D2), which models the effects of inertia, was employed in this analysis. This computer code is discussed in the NRC approved Bechtel's Topical Report BN TOP-4, Rev. 1.

b. Concentrated Force-Time Histories

The forcing function of jet impingement on the shield wall is obtained from the break flow transient caused by a feedwater line break and a recirculation line break. Likewise, the 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 and the recirculation line break.

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QUESTION NO. 63 (CONT'D)

c. Integrated Dynamic Analysis

GE computer codes are employed to integrate the pressure-time histories and concentrated force-time histories in determining the effects on the shield wall pedestal, vessel support, core support and internals, and control rod drives. These dynamic analyses yield accelerations, forces, and moments.

d. Attached Piping Analysis

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

e. Load Combinations for Vessel and Piping

The asymmetric LOCA loads in combination with SSE by the SRSS methodology are treated as a faulted condition for evaluation against the ASME Code. This is described in revised Table 3.9-6 for NSSS and in Table 3.9-11 for BOP.

QUESTION NO. 64
(3.9.3.1, Page 3.9-52)

The functional capability for essential systems must be assured when they are subjected 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 shutdown the plant following or to mitigate the consequences of an accident. Please provide such criteria. In particular, for both NSSS and BOP, have the criteria in NEDO-21985 bee met?

RESPONSE

For BOP piping, the functional capability of essential systems complies with NEDO-21985. Table 3.9-12 is revised to reflect this requirement.

For NSSS piping, the functional capability of essential systems is being evaluated per NEDO-21985 as part of the New Loads Adequacy Evaluation. The appropriate FSAR revision will be provided upon completion of the New Loads evaluation.

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TABLE 3.9-12

DESIGN CRITERIA FOR ASME CODE CLASS 1 NON-NSSS PIPING

CONDITION	STRESS LIMITS[1]
Design	NB-3221 and NB-3652
Normal and upset	NB-3223 and NB-3654
Emergency	NB-3224 and NB-3655
Faulted	NB-3225 and NB-3656

[1] As specified by ASME Code Section III, 1971 through Winter 1972 Addenda

(2) Functional capability of essential piping is assured per NEDO-21985 September 1978.

DRAFT

QUESTION NO. 65
(3.9.3.1.6, Page 3.9-53)

Why are the recirculation pumps designed to Section VIII Division I of the ASME Code?

RESPONSE

At the time the Limerick recirculation pump was procured, Section III design rules for pumps were under development. Therefore, the design was based on the standard industry rules as described in Section 3.9.3.1.6, including ASME Code Section VIII, Division 1 for thickness calculations of pressure retaining parts and for sizing the pressure retaining bolts.

DRAFT

QUESTION NO. 66
(3.9.3.1, Pages 3.9-52 to 62)

Standard Review Plan 3.9.3 of NUREG-0800 requires that internal parts of components, such as valve discs and seats, and pump shafting, subjected to dynamic loading during operation of the component should be included in consideration of loading combinations, system operating transients, and stress limits. Provide assurance that this has been done.

RESPONSE

The FSAR update incorporating the New Loads Adequacy Evaluation (NLAE) results for the NSSS systems and components will be provided upon NLAE program completion. As part of the standardized New Loads package, each component or equipment documented in the Table 3.9-6 series will have five entries for each of the ASME Code service levels:

1. Code Criteria,

2. Limiting Load Combination,

3. Limiting Stress Type,

4. Allowable Stress, and

5. Calculated Stress

DRAFT

QUESTION NO. 67
(3.9.3.1)

The safety relief valve discharge piping and downcomers are ASME Class 2 and 3 components, a fatigue analysis is not required in their design by the ASME Section III Boiler and Pressure Vessel Code. However, a through wall leakage crack in these lines resulting from fatigue caused by SRV actuations and small LOCA conditions would allow steam to bypass the pressure suppression pool. This could result in an unacceptable over-pressurization of the containment. We, therefore, require that the applicant perform a fatigue evaluation on these lines in accordance with the ASME Class 1 fatigue rules.

RESPONSE

Fatigue evaluation of the unsubmerged portion of safety relief valve discharge piping and downcomers in the wetwell has been performed in accordance with the ASME Class 1 fatigue rules.

DRAFT

QUESTION NO. 68
(3.9.3.1, Table 3.9-6)

Appendix A of SRP 3.9.3 requires that Class 1, 2 and 3 components, component supports, and Class CS core support structures shall meet a service limit not greater than Level D when subjected to the appropriate combination of loadings resulting from (1) sustained loads, (2) either the DBPB, MS/FWPB, or LOCA, and (3) the SSE. This loading combination does not appear in Table 3.9-6 for all of the above components. Provide more explicit loading combinations and show what service limits are met.

RESPONSE

Load combination and acceptance criteria for NSSS ASME Code Class 1, 2 and 3 piping and components are listed in the attached revision to Table 3.9-6.

MEB-68

TABLE 3.9-6

~~(Page 1 of 2~~

LOADING COMBINATIONS, STRESS LIMITS, AND ALLOWABLE STRESSES

This table lists the design loading combinations, allowable stresses, and calculated stresses, for the major mechanical safety-related components in the plant. Various parts of the table are referenced in Section 3.9. The format in some parts of the table is not consistent, since variation of analytical method and depth of detail necessary to demonstrate the safety aspects of various components differs. The table is divided into the following parts:

3.9-6 Load Combinations and Acceptance Criteria for ASME Code Class 1, 2, and 3 NSSS Piping and Equipment

a. Reactor Pressure Vessel and Shroud Support Assembly

b. Reactor Vessel Internals and Associated Equipment

c. Reactor Water Cleanup Heat Exchangers

d. Class 1 Main Steam Piping

e. Class 1 Recirculation Loop Piping

f. Not Used

g. Main Steam Relief Valves

h. Main Steam Isolation Valve

i. Recirculation Pump

j. Reactor Recirculation System Gate Valves

k. Class 3 Safety/Relief Valve Discharge Piping

l. Standby Liquid Control Pump

m. Standby Liquid Control Tank

n. ECCS Pumps

o. RHR Heat Exchanger

p. RWCU Pump

q. RCIC Turbine

r. RCIC Pump

s. Fuel Storage Racks

t. High Pressure Coolant Injection Pump

u. Not Used

v. Control Rod Drive Housing

w. Jet Pumps

x. Not Used

y. LPCI Coupling

z. Not Used

aa. Control Rod Guide Tube

ab. Incore Housing

ac. Reactor Vessel Support Equipment; CRD Housing Support

ad. NOT USED

ae. HPCI Turbine

TABLE 3.9-6

LOAD COMBINATION AND ACCEPTANCE CRITERIA
FOR ASME CODE CLASS 1, 2 AND 3 NSSS PIPING AND EQUIPMENT

MEB-68

LOAD COMBINATION	DESIGN BASIS	EVALUATION BASIS	SERVICE LEVEL
N + $SRV_{(ALL)}$	Upset	Upset	(B)
N + OBE	Upset	Upset	(B)
N + OBE + $SRV_{(ALL)}$	Emergency	Upset	(B)
N + SSE + $SRV_{(ALL)}$	Faulted	Faulted	(D)*
N + SBA + SRV	Emergency	Emergency	(C)*
N + SBA + $SRV_{(ADS)}$	Emergency	Emergency	(C)*
N + SBA/IBA + OBE + $SRV_{(ADS)}$	Faulted	Faulted	(D)*
N + SBA/IBA + SSE + $SRV_{(ADS)}$	Faulted	Faulted	(D)*
N + LOCA** + SSE	Faulted	Faulted	(D)*

LOAD DEFINITION LEGEND

Normal(N) - Normal and/or abnormal loads depending on acceptance criteria.

OBE - Operational basis earthquake loads.

SSE - Safe shutdown earthquake loads.

SRV - Safety/relief valve discharge induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling).

SRV_{ALL} - The loads induced by actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).

SRV_{ADS} - The loads induced by the actuation of safety/relief valves associated with Automatic Depressurization System which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.

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MEB-6B

TABLE 3.9-6

LOAD COMBINATION TABLE (CONT'D)

LOCA	-	The loss of coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping).
$LOCA_1$	-	Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	-	Pool swell impact loads on piping and components located above the suppression pool water upper surface.
$LOCA_3$	-	Oscillating pressure induced loads on submerged piping and components during condensation oscillations.
$LOCA_4$	-	Building motion induced loads from chugging.
$LOCA_5$	-	Building motion induced loads from main vent air clearing.
$LOCA_6$	-	Vertical and horizontal loads on main vent piping.
$LOCA_7$	-	Annulus pressurization loads.
SBA	-	The abnormal transients associated with a Small Break Accident.
IBA	-	The abnormal transients associated with an Intermediate Break Accident.

*All ASME Code Class 1, 2 and 3 piping that are required to function for safe shutdown under the postulated events are designed to meet the requirements described in NEDO-21985.

**The most limiting case of load combinations among $LOCA_1$ through $LOCA_7$.

DRAFT

QUESTION NO. 69
(3.9.3.1)

The suppression pool hydrodynamic loads must be reconciled and the results documented when the load definition are finalized. Provide a commitment to submit the results of this reconciliation.

RESPONSE

Many responses in this MEB-SER package address the LGS hydrodynamic loads. Results of the New Loads Adequacy Evaluation (NLAE) will be documented in the FSAR for NRC submittal upon program completion.

QUESTION NO. 70
(3.9.3.3.2, Page 3.9-76)

Are there any open discharge systems mounted on ASME Class 3 systems? If so, has Regulatory Guide 1.67 been used in the design of these systems?

RESPONSE

In the Limerick design, there are no relief valve stations mounted on ASME Class 3 systems that have open discharge systems with limited discharge pipes. Therefore, Regulatory Guide 1.67 is not applicable. Section 3.9.3.3.2 is revised to reflect this.

to the suppression pool is established. This period includes clearing the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the MSRV cause the MSRV discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise time-history solution of the fluid flow equation, to generate a time-history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification, and the value of ASME flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves is assumed in the analysis, because simultaneous discharge is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Figure 3.9-3 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge.

The method of analysis applied to determine piping system response to MSRV operation is time-history integration. The forces are applied at locations on the piping system where the fluid flow changes direction, thus causing momentary reactions. The resulting loads on the MSRV, the main steam line, and the discharge piping are combined with loads due to other effects, as specified in Section 3.9.3.16. The code stress limits corresponding to load combination classifications of normal, upset, emergency, and faulted, are applied to the main steam lines and MSRV discharge piping.

3.9.3.3.2 Design and Installation Details for Mounting of Pressure Relief Devices in ASME Code Class 1, 2, and 3 Systems (Non-NSSS)

MEB-70

The design of the pressure relieving devices can be grouped into two categories: open discharge and closed discharge.

a. Open Discharge

There are no open discharge pressure relieving devices mounted on ASME Code Class 1, 2 and 3 systems with limited runs of discharge piping.

b. Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank, or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and

DRAFT

QUESTION NO. 71
(3.9.3.4.1, Page 3.9-77)

Paragraph NF-3132.3 requires that supports be evaluated for high cycle fatigue. Provide assurances that this has been done.

RESPONSE

In the Limerick design, no supports are subject to high cycle fatigue. Further justification is provided in response to Question No. 77.

DRAFT

QUESTION NO. 72
(3.9.3.4.1, Page 3.9-77)

Provide the allowables used for bolts for supports and piping.

RESPONSE

1. Component Support Bolting (NSSS)

   (a) RWCU Pump

   The support bolting of this pump which is not essential to safety is designed for the effects of pipe load and SSE load to the requirements of the ASME Code, Section III, Appendix XVII. The stress limits of 0.41Sy for tension and 0.15Sy for shear are used.

   (b) RCIC/SLC Pumps and RCIC Turbine

   The equipment-to-base plate bolting satisfies the following design criteria:

   For Normal and Upset conditions, 1.0S is used for primary membrane and 1.5S for primary membrane plus bending, where S is the allowable stress limit from the ASME Code Section III, Appendix I, Table I-7.3. For Emergency and Faulted conditions, stresses shall be less than 1.2 times the allowable limits for "Normal and Upset" given above.

   (c) Flanged Connection Bolting

   There are no flange-type connections in component supports.

2. Piping Supports and Pipe Mounted Equipment (Valves and Pump) Supports (NSSS)

   The supports are hanger and snubber type (including clamps) linear standard components as defined by the ASME Code Section III, Subsection NF. The bolts used in these supports meet criteria of NF-3280 for Service Levels A and B and NF-3230 for Service Levels C and D.

   For Service Levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum allowable stresses at temperature.

The above NSSS response is incorporated in the FSAR as Section 3.9.3.4.1.7.

3. Component Support Bolting (Non-NSSS)

   The support bolt allowables are in accordance with the design code of the pipe as follows:

DRAFT

QUESTION NO. 72 (CONT'D)

Piping Classification	Bolt Allowable Code
Nuclear Class 1, 2, and 3	ANSI-B31.7
Non-Nuclear Piping	ANSI-B31.1
Wall Through-Bolts	AISC

For flanged connections, the bolt allowables used in the piping are those ASME Code Section III, 1979 Summer Addenda, Sections NB, NC and ND for Class 1, 2 and 3 respectively.

d. Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary seismic forces, (i.e., OBE and SSE), system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

3.9.3.4.1.2 RHR and Core Spray Pump Supports

The core spray and RHR pumps have been tested in the shop and are tested as described in Section 3.9.3.2a. These tests provide the adequacy of the support structure for the pump assembly under operating conditions. Furthermore, the stress calculation summary provided in Section 3.9.3.1 defines the stress levels in the critical support areas, namely, the pressure boundary parts and the non-pressure boundary parts. The stress level margins prove the adequacy of the equipment.

3.9.3.4.1.3 RCIC Turbine Supports

The RCIC turbine assembly is analyzed as described in Section 3.9.3.1a. The calculation summary defines the stress levels in the critical support areas, namely, the stop valve yoke and the pedestal dowel pins and bolts. The substantial stress level margins prove the adequacy of the equipment.

3.9.3.4.1.4 Reactor Water Cleanup System Pump Supports

The pump pedestal bolts are analyzed as discussed in Section 3.9.3.1b. Loads from seismic dead weight, connecting pipes, and temperature are considered.

3.9.3.4.1.5 HPCI Turbine Supports

The HPCI turbine assembly is analyzed as described in Section 3.9.3.1a. The calculation summary in Table 3.9-6(ac) defines the stress levels in the critical support areas, namely, the stop valve yoke and the pedestal dowel pins and bolts. The substantial stress level margins prove the adequacy of the equipment.

3.9.3.4.1.6 Reactor Pressure Vessel Support Skirt MEB-76

See Insert (Page 3.9-79a) attached to Q #76 Response.

3.9.3.4.1.7 Bolting Stress Limits (NSSS) MEB-72

INSERT attached (Page 3.9-79b).

MEB 72

3.9.3.4.1.7.1 Component Support Bolting

(a) RWCU Pump

The support bolting of this pump which is not essential to safety is designed for the effects of pipe load and SSE load to the requirements of the ASME Code, Section III, Appendix XVII. The stress limits of 0.41Sy for tension and 0.15Sy for shear are used.

(b) RCIC/SLC Pumps and RCIC Turbine

The equipment-to-base plate bolting satisfies the following design criteria:

For Normal and Upset conditions, 1.0S is used for primary membrane and 1.5S for primary membrane plus bending, where S is the allowable stress limit from the ASME Code Section III, Appendix I, Table I-7.3. For Emergency and Faulted conditions, stresses shall be less than 1.2 times the allowable limits for "Normal and Upset" given above.

(c) Flanged Connection Bolting

There are no flange-type connection in component supports.

3.9.3.4.1.7.2 Piping Supports and Pipe Mounted Equipment (Valves and Pump) Supports

The supports are hanger and snubber type (including clamps) linear standard components as defined by the ASME Code Section III, Subsection NF. The bolts used in these supports meet criteria of NF-3280 for Service Levels A and B and NF-3230 for Service Levels C and D.

For Service Levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum allowable stresses at temperature.

DRAFT

QUESTION NO. 73
(3.9.3.4)

Does the design criteria for component supports in Limerick systems categorize the stresses produced by seismic anchor point motion of piping and the thermal expansion of piping as primary or secondary? It is the staff's position that for the design of component supports, the stresses produced by seismic anchor point motion of piping and the thermal expansion of piping should be categorized as primary stresses.

RESPONSE

For pipe supports, reactions produced by primary and secondary pipe loads are categorized as primary. The primary and secondary loads are summed and compared to the load rating to ensure the rating is not exceeded. Since no distinction is made between primary and secondary loads, and load rated components are designed to primary limits or qualified by testing, the supports meet primary stress criteria for primary and secondary loads combined.

DRAFT

QUESTION NO. 74
(3.9.3.3, Page 3.9-76,77)

Include a description of the computer program or calculational procedures utilized in the analysis of pressure relief devices by time-history or equivalent static solution, respectively. What dynamic load factor is used in the equivalent static method?

RESPONSE

NSSS

See the revised Subsection 3.9.1.2.2.4.2 attached to the response to Question No. 52 (Computer Programs) for description of the Relief Valve Discharge Pipe Forces Computer Program/RVFOR. The equivalent static method is not used.

BOP (Non-NSSS)

The analysis of pressure relief devices on seismic Category I systems is performed using time history methods. Forcing functions in terms of segment force time histories are generated using Bechtel computer programs NE805 and NE452 (Sections 3.9.1.2.6.6 and 3.9.1.2.6.5, respectively).

Structural piping response to these generated forcing functions is calculated by the time history method using Bechtel computer program ME101. FSAR Section 3.9.1.2.6.1 is revised to include a detailed description of the ME101 program.

loads, maximum conservative moments and forces at the upper and lower supports are calculated.

b. Stress Analysis of Supports (ED-8)

This program performs a full stress analysis of the upper and lower supports of the RHR heat exchanger. The stresses in the supports (both upper and lower) caused by loads resulting from seismic and nozzle loads are computed in the support load program (ED-6), and are used as input values for this program. This program computes the membrane stresses on the shell of the heat exchanger by using Bijlaard's analysis, as well as the net section stresses (shear, tensile, bearing) on the lower support plate and upper lugs. It also computes the stresses on the welds holding the supports to the shell of the heat exchanger.

3.9.1.2.6 Seismic Category I Items Other than NSSS

A list of computer programs used in the non-NSSS system components is provided in Table 3.9-3. This list consists of computer programs developed and/or owned by Bechtel Power Corporation (BPC), and of computer programs that are recognized and widely used in industry.

The Bechtel developed and/or owned computer programs are documented, verified, and maintained by Bechtel, and meet the requirements of 10 CFR, Part 50, Appendix B. A brief description of each of these Bechtel programs is provided below.

3.9.1.2.6.1 ME101, Linear Elastic Analysis

MEB-74

Program Description: ME101 is a finite element computer program that performs linear elastic analyses of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically.

The output may be used directly for piping design, for conformation to code, and for other regulatory requirements. Two piping codes, ASME B&PV Code 1974 and B31.1 Summer 1973 addenda, are incorporated into the program to the extent of computing flexibility factors, stress intensification factors, and stresses. ME101 may be used for static and seismic analysis of piping systems. Static analysis considers one or more of the following: thermal expansion, deadweight, uniformly distributed loads, externally applied loads (forces, moments, displacements, and rotations). Seismic analysis is based on standard normal

INSERT

MEB-74

ME101 is a finite-element computer program that performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering. ME101 performs a thorough check of the input prior to analysis. In addition, the program automatically modifies the geometry to improve the finite-element model.

The output may be used directly for piping design, for conformation to Code, and for other regulatory requirements. Two piping codes, ASME B&PV Code, 1974, and ANSI B31.1, Summer 1973 Addenda, are incorporated in ME101 to the extent of computing flexibility factors, stress intensification factors, and stresses.

ME101 performs static and dynamic load analysis of piping systems, effective weight calculations, and ASME B&PV Code, Section III Class 2 and 3 and ANSI B31.1 Code stress checks.

Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly-distributed loads, and externally-applied forces, moments, imposed displacements and rotations, individual force loads, static seismic (uniform directional acceleration) loads, or seismic anchor movement analysis.

Dynamic analysis is based on the standard normal superposition techniques. The input excitation may be in the form of seismic response spectra or time-dependent loading functions. In the single or multiple response spectrum analysis, the user may request modal synthesis by square root of the sum of the squares (SRSS) method or by NRC Regulatory Guide 1.92 closely spaced mode 10% (EQuation 4) method. ME101 can consider further differential damping for large and small pipe according to NRC Regulatory Guide 1.61. Various methods of eigenvalue solution are available. Determinant search or subspace iteration considers all data points as mass points. In the time-history analysis, the excitation may be in the form of arbitrary nodal forces, support displacements, rotations, or support accelerations that are not necessarily in phase.

ME101 checks stresses from design loads versus allowable stresses according to ASME/ANSI Code equations. The user may request design load checks for sustained loads, occassional loads, multimode thermal expansion and pipe break, except for time-history load cases.

The ME101 restraint load summary report prints the support load results from several load cases together in the same report, except for time-history load cases.

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MEB-74

The general loading combinations capability for ME101 can combine the results of several load cases together, according to certain algebraic rules, to form a new load case. The new load case resulting from this may be used in stress comparisons or restraint load summaries, except for time-history load cases. ME101 has the capability of saving load case results on a tape and using these results in late runs for stress checks, restraint load summary reports, and general loading combinations, except for time-history load cases.

For piping configurations with optional node numbering, ME101 generates isometric plots. The user may obtain plots on ZETA or CALCOMP plotters on a Tektronix 4014 graphics terminal, or on an RMS-600 printer/plotter.

ME101 uses out-of-core techniques for both static and response spectra analysis and has no practical limitations to the number of equations or band width. However, the use of very large systems may become prohibitive due to cost of computation. The maximum number of mode shapes allowable for response spectra analysis is currently 125.

This program considers the zero period acceleration effect in seismic response analysis. It accepts coordinate and keyword data in English or Metric units.

The current UNIVAC version, 31, of ME101 is being used for Limerick.

This piping program's development has begun in July 1975, and since then it has been used on a number of other projects.

The ASME Benchmark Problem 1 demonstrates the solution for natural frequencies of a three-dimensional structure, as described in Reference 3.9.4.

Natural frequencies, in hertz, from ME101 and Reference 3.9-4, are as follows:

Mode	Reference 3.9-4	ME101
1	110	112
2	117	116
3	134	138

A total of 26 test problems were used for the verification of the ME101 results. These verification problems have been compared against one of the following:

a. ME632, Computer Program, "Seismic Analysis of Piping Systems", VERB MODB, 1976, Bechtel International Corporation, San Francisco, CA.

MEB-74

b. "Pressure Vessel and Piping 1972 Computer Programs Verification", The American Society of Mechanical Engineers.

c. Hand Calculations

d. EDS Superpipe, EDS Nuclear, San Francisco, CA.

e. NUPIPE-IIM, Nuclear Services Corporation Piping Analysis Program, Campbell, CA.

f. TPIPE, A Computer Program for Analysis of Piping Systems, PMB Systems Engineering, San Francisco, CA.

g. ADINA, A Computer Program, Massachusetts Institute of Technology, Boston, MA.

h. MSC/NASTRAN Program, McNeal Schwendler Corporation, Los Angeles, CA.

i. EASE2 Program, Engineering/Analysis Corporation, San Francisco, CA.

j. ANSYS, Swanson Analysis System, Inc., 1975, Elizabeth, PA.

The J1 version of ME101 also includes seven NRC benchmarked problems, as referenced in NUREG/CR-1677, dated August 1980.

mode techniques, and uses response spectrum data. Two methods of eigenvalue solution are available. Determinant Search or Subspace Iteration considers all data points as mass points. Kinematic Reduction and Householder QR considers masses only at specified data points, and in designated directions. Differential seismic anchor movement analyses and effective weight calculation of restraints and anchors are also provided.

ME101 generates isometric plots of the piping configuration, with optional node numbering. The plots are obtained by either a ZETA or CALCOMP 1036 plotter.

The program uses out-of-core solution techniques for both static and dynamic analysis, and has no practical limitations to the number of equations or to the band width. However, very large systems may become prohibitive due to cost of computation. The maximum number of mode shapes allowable is currently 125.

Program Version and Computer: The current UNIVAC version (C3) of ME101 is being used by BPC.

Extent of Application: ME101 is a piping program developed by BPC. Its development began in July 1975, and is being continuously supported by BPC. It has been used by various BPC projects.

Test Problems: The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three dimensional structure as described in Ref 3.9-4.

The following table lists the natural frequencies from ME101 and Ref 3.9-4.

Natural Frequency Comparisons, Hz

Mode No.	Ref 3.9-4	ME 101
1	110	112
2	117	116
3	134	138

3.9.1.2.6.2 ME632, Piping System Analysis

MEB 74

Program Description: ME632 performs stress analyses of 3-dimensional piping systems. The effects of thermal expansion, uniform load of the pipe, pipe contents and insulation, concentrated loads, movements of the piping system supports, and other external loads, such as wind and snow, may be considered. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program.

DRAFT

QUESTION NO. 75
(3.9.3.4, Page 3.9-77)

Provide a graphic summary of your interpretation of Subsection NF boundaries.

RESPONSE

Section 3.9.3.4.2 is revised and Figures 3.9-9 and 3.9-10 are added to show the jurisdictional boundaries for the pipe supports.

See the response to Question No. 81 for the affected text revision.

CHAPTER 3

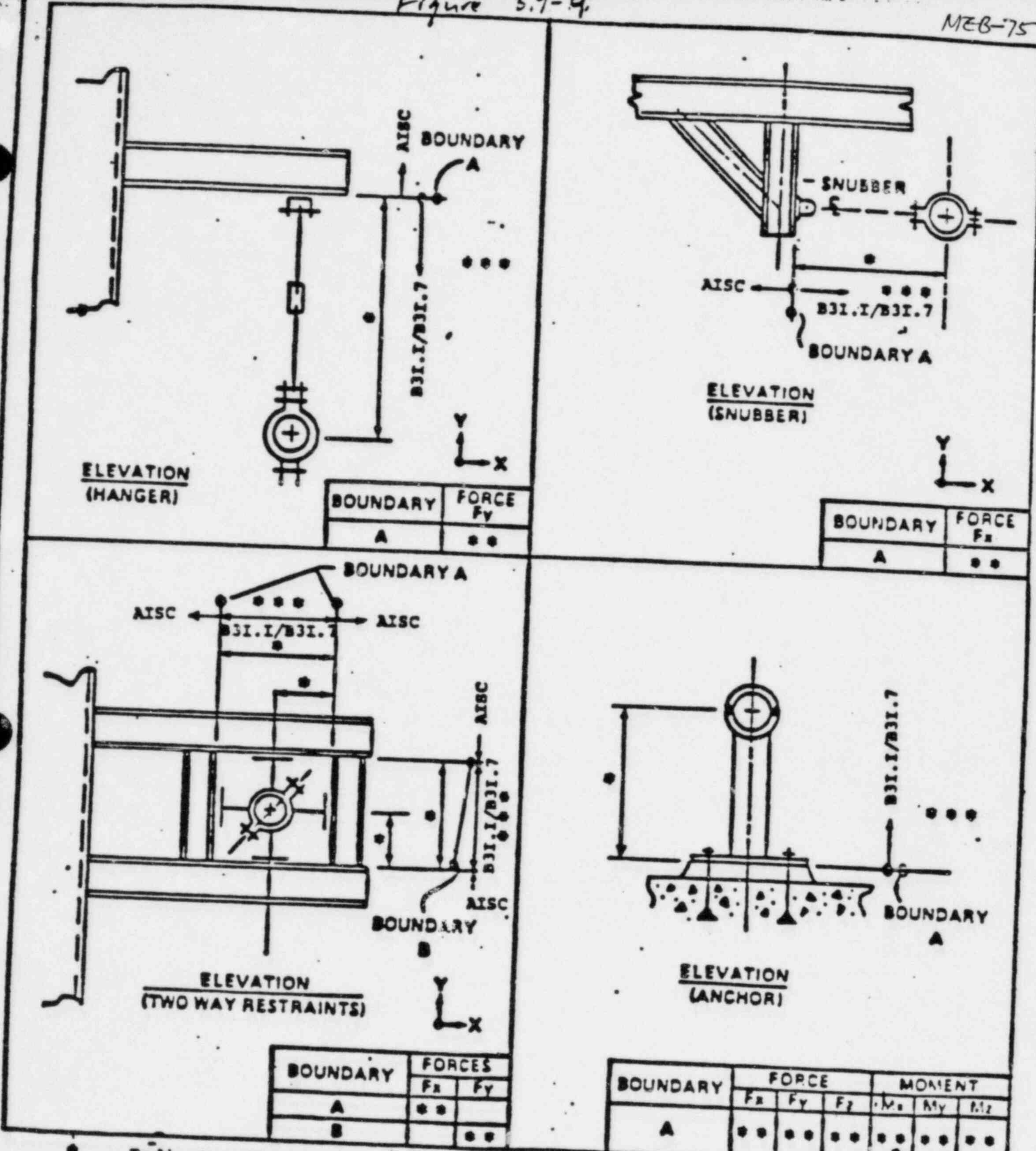
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

FIGURES (Cont'd)

MEB-75

Figure 3.9-9.

MEB-75



BOUNDARY	FORCE Fy
A	**

BOUNDARY	FORCE Fx
A	**

BOUNDARY	FORCES Fx	Fy
A	**	
B		**

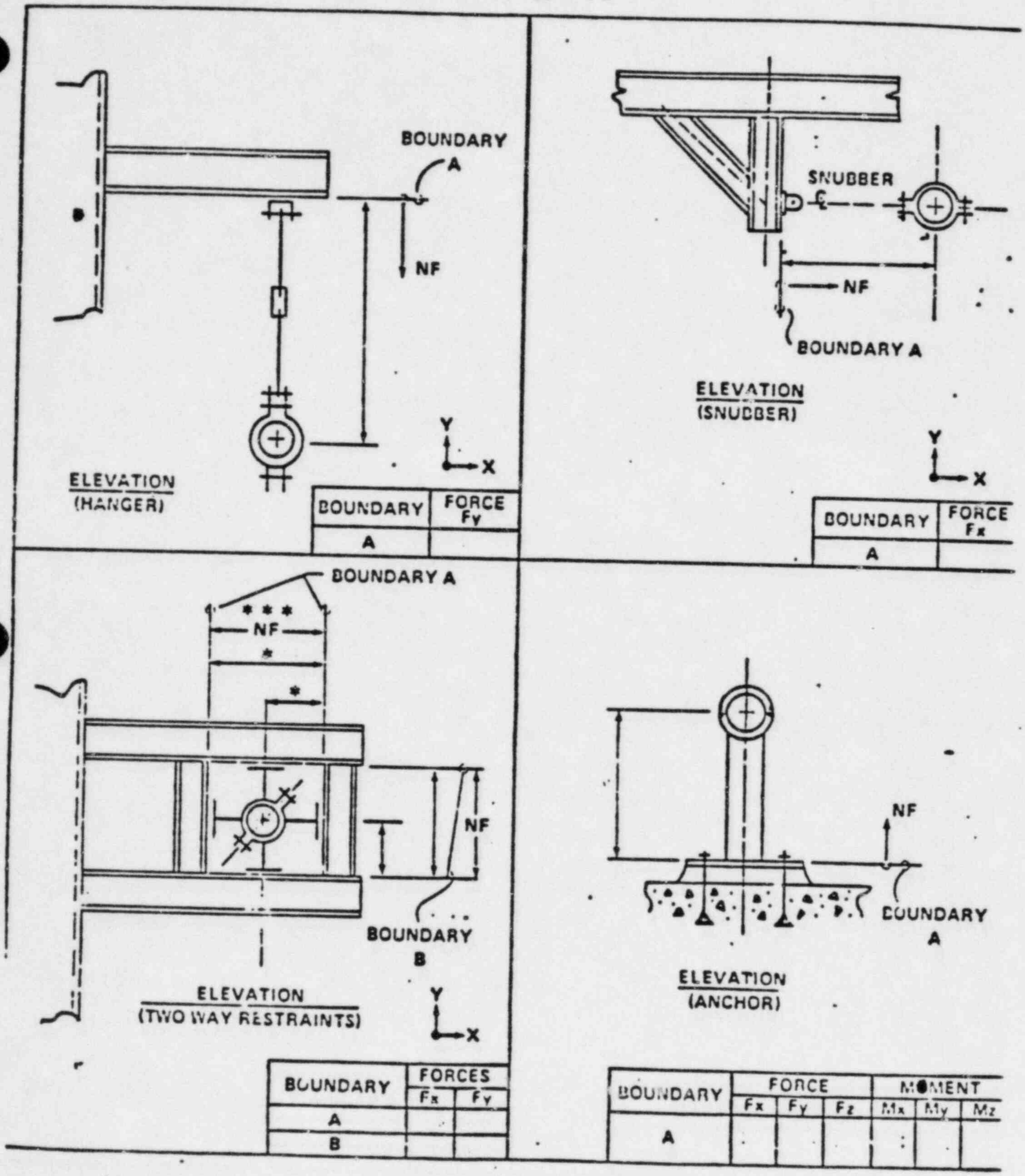
BOUNDARY	FORCE Fx	Fy	Fz	MOMENT Mx	My	Mz
A	**	**	**	**	**	**

* Indicates dimensional location of B31.1/B31.7 elements
** Indicates loads imposed on the B31.1/B31.7 boundary
*** Indicates B31.1/B31.7 boundaries (including weld, if any)
• B31.7. CODE FOR NUCLEAR CLASS 1, 2 & 3 PIPING SYSTEMS.
• B 31.1 CODE FOR NON-NUCLEAR PIPING SYSTEMS
• Snubbers, compensating starts, and struts comply with all the requirements of ASME Section III, Subsection NF; they are not commercially available to meet the requirements of ANSI B31.1.

JURISDICTIONAL BOUNDARY BETWEEN PIPE SUPPORTS AND SUPPORTING STRUCTURE FOR GENERAL R.O.P.

FIGURE ~~2~~ 3.9-10

MEB-75



~~-35-~~

C?7

JURISDICTIONAL BOUNDARY
BETWEEN PIPE SUPPORTS
AND SUPPORTING STRUCTURE

~~Sheet 1 of 2~~

~~Rev.1~~

DRAFT

QUESTION NO. 76
(3.9.3.4)

Provide the bases for the allowable buckling loads including the buckling allowable stress limit under faulted conditions for all NSSS and BOP ASME Class 1 component supports including the reactor vessel support skirt.

RESPONSE

NSSS

The Class 1 linear type component supports are required by the suspension design specification to be designed in accordance with the rules of the ASME code, Section III, Subsection NF. To avoid buckling in the component supports, Appendices F and XVII of the ASME code require that the allowable loads shall be limited to two-thirds of the critical buckling loads.

The critical buckling loads for Class 1 component supports subjected to faulted loads which are more severe than normal, upset and emergency loads, are determined by the vendor using the methods described in Appendix F of the ASME code.

Per design specification, the permissible compressive load on the reactor vessel support skirt cylinder (modeled as plate and shell type component support) is limited to 90 percent of the load which produces yield stress, divided by the safety factor for the condition being evaluated. The effects of fabrication and operational eccentricity is included. The safety factor for faulted conditions is 1.125.

An analysis of reactor pressure vessel support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Code Section III, Paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The faulted condition analyzed included the compressive loads due to the design bases maximum earthquake, the overturning moments and shears due to the jet reaction load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure expansion of the reactor vessel. The expected maximum earthquake loads for the Limerick reactor vessel support skirts are less than 50% of the maximum design bases loads used in the buckling analysis described; therefore, the expected faulted loads are well below the critical buckling limits of Paragraph F-1370(c) for this reactor vessel support skirt. The expected earthquake loads for this reactor are determined using the seismic dynamic analysis methods described in Section 3.7.

In accordance with this response, piping supports are addressed in the revised Section 3.9.3.4.1.1. The RPV support skirt is addressed in the added Section 3.9.3.4.1.6.

BOP (Non-NSSS)

For BOP Class 1 component supports, the allowable stress is limited to two-thirds of the critical buckling stress.

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The requested information for BOP, ASME Class 1, 2 and 3 component supports is provided in Table 3.9-21.

resulting stresses are determined by using either a time-history computer solution, or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. Water slug effects are also considered.

Time-history dynamic analysis is performed for the discharge piping and its supports. The effect of the loading on the header is also considered. The design load combinations for a given transient are shown in Table 3.9-11, and the design criteria and stress limits are shown in Tables 3.9-12 and 3.9-16.

3.9.3.4 Component Supports

3.9.3.4.1 Supports furnished with the NSSS.

3.9.3.4.1.1 Piping

Piping supports are designed in accordance with Subsection NF of ASME Code Section III. Supports are either designed by load rating per paragraph NF-3260, or to the stress limits for linear supports per paragraph NF-3231. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports, as no fatigue evaluation is necessary to meet the ASME code requirements. MEB-76

The design criteria and dynamic testing requirements for component supports are as follows:

a. Component Supports

All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they are ~~have been~~ installed. All component supports are designed in accordance with the rules of Subsection NF of the ASME code.

b. Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

To avoid buckling in the component supports, Appendix F and XVII of the ASME Code require that the allowable loads be limited to two-thirds of the critical buckling loads. The critical buckling loads for Class 1 component supports subjected to the faulted condition are more severe than normal, upset, and emergency loads.

c. Snubbers

~~The design load on snubbers includes those loads caused by seismic forces (OBE and SSE), system anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.~~

~~The snubbers are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.~~

~~The snubbers are also tested dynamically, to ensure that they can perform as required, in the following manner:~~

1. ~~The snubber is subjected to either a force or displacement that varies similar to the sine wave.~~

2. ~~The frequency (Hz) of the input motion or force is verified at small increments within the specified range.~~

3. ~~The resulting relative displacements and corresponding loads across the working components, including end attachments, are recorded.~~

4. ~~The test is conducted with the snubber at various temperatures.~~

5. ~~The peak load in both tension and compression is equal to or higher than the rated load.~~

6. ~~The duration of the test at each frequency is specified.~~

   ~~The snubber is tested dynamically at a frequency within a specified frequency range, and at a minimum specified temperature for the faulted load. Test duration is specified. Snubbers are tested for various abnormal environmental conditions.~~

   ~~Upon completion of the above abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.~~

MEB-76

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MEB-76

C. Snubbers

Snubbers are not supplied by GE, however, required load capacity and snubber location for NSSS piping systems are determined by GE as a part of the NSSS piping system design and analysis scope.

1. Required Load Capacity and Snubber Location

The entire piping system, including valves and the suspension system between anchor points, is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as springs with a given stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (OBE and SSE), system anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, and other hydrodynamic forces (SRV, LOCA, AP).

The snubber location and loading direction are first decided by estimation so that the stresses in the piping system have acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system as described above.

The spring constant required by the suspension design specification for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are not in agreement, they are brought into agreement, and the system analysis is redone to confirm the snubber loads.

If the stiffness of the backup structure for the snubber is not large compared to that of the snubbers, the reduced effective snubber stiffness (spring constant) is used in the analysis to account for backup structure flexibility.

2. Snubber Design Specification

See Subsection 3.9.3.4.2.1.1.

d. Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary seismic forces, (i.e., OBE and SSE), system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

3.9.3.4.1.2 RHR and Core Spray Pump Supports

The core spray and RHR pumps have been tested in the shop and are tested as described in Section 3.9.3.2a. These tests provide the adequacy of the support structure for the pump assembly under operating conditions. Furthermore, the stress calculation summary provided in Section 3.9.3.1 defines the stress levels in the critical support areas, namely, the pressure boundary parts and the non-pressure boundary parts. The stress level margins prove the adequacy of the equipment.

3.9.3.4.1.3 RCIC Turbine Supports

The RCIC turbine assembly is analyzed as described in Section 3.9.3.1a. The calculation summary defines the stress levels in the critical support areas, namely, the stop valve yoke and the pedestal dowel pins and bolts. The substantial stress level margins prove the adequacy of the equipment.

3.9.3.4.1.4 Reactor Water Cleanup System Pump Supports

The pump pedestal bolts are analyzed as discussed in Section 3.9.3.1b. Loads from seismic dead weight, connecting pipes, and temperature are considered.

3.9.3.4.1.5 HPCI Turbine Supports

The HPCI turbine assembly is analyzed as described in Section 3.9.3.1a. The calculation summary in Table 3.9-6(ac) defines the stress levels in the critical support areas, namely, the stop valve yoke and the pedestal dowel pins and bolts. The substantial stress level margins prove the adequacy of the equipment.

3.9.3.4.1.6 Reactor Pressure Vessel Support Skirt MEB-76

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MEB-76

3.9.3.4.1.6 Reactor Pressure Vessel Support Skirt

The permissible compressive load on the reactor vessel support skirt cylinder (modeled as plate and shell type component support) is limited by the design specification to 90 percent of the load which produces yield stress, divided by the safety factor for the condition being evaluated. The effects of fabrication and operational eccentricity is included. The safety factor for faulted conditions is 1.125.

An analysis of reactor pressure vessel support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Code Section III, Paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The faulted condition analyzed included the compressive loads due to the design basis maximum earthquake, the overturning moments and shears due to the jet reaction load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure expansion of the reactor vessel.

Subsequently, based on currently defined faulted condition loads, the maximum compressive stress in the support skirt including axial* and bending** loads is less than the faulted condition allowable of Appendix F (Paragraph F-1325) determined by the methods of NB 3133.6 of the ASME Code.

The loading criteria, stress criteria, calculated and allowable stresses are summarized in Table 3.9-6(a).

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*Axial loads include weight, fuel interaction, seismic SSE and the maximum of condensation oscillation, chugging and vent clearing due to a loss of coolant accident (LOCA).

**Bending loads include seismic SSE and jet reaction, jet impingement and annulus pressurization due to a loss of coolant accident (LOCA).

DRAFT

QUESTION NO. 77
(3.9.3.4.1c, Page 3.9-78,80)

(1) Have you considered fatigue strength of snubbers used as shock and vibration arrestors or as dual purpose snubbers?

(2) Describe measures taken to ensure that thermal growth does not exceed snubber lock-up velocity.

RESPONSE

(1) Snubbers for Limerick are used to arrest shock due to seismic and other dynamic transient events. Under such applications, the snubbers will be subjected to a limited number of load cycles.

Snubbers are not designed for vibration control. Therefore, no fatigue evaluation has been performed.

Steady state vibration conditions will be identified during the preoperational test program. If the snubbers are used to correct such conditions, they will be evaluated for acceptability under those conditions.

Additionally, the snubber inservice inspection program assures that any potential malfunction due to fatigue-type failure will be detected.

(2) All Limerick snubbers are acceleration sensitive, mechanical snubbers. The acceleration threshold of these snubbers is well above the anticipated thermal growth rate of the piping system.

DRAFT

QUESTION NO. 78
(3.9.3.4.1.C, Pages 3.9-78,80)

Describe how snubber support structures flexibility has been accounted for. How has end clearance and lost motion been considered?

RESPONSE

The methodology utilized for the stress analysis of seismic Category I, 2½" and larger piping systems is as follows:

1. For systems designed to seismic and hydrodynamic loads, the flexibility of the pipe supports are considered in the piping stress analysis. A stiffness tolerance criteria is used to facilitate support design and installation.

2. For systems designed to seismic loads, only the supports are considered as rigid members in the piping stress analysis model and are designed such that their fundamental frequencies in the direction of the applied load is within the rigid range of the seismic response spectra.

End clearance and lost motion are limited by specification to a value of 0.04" (see Section 3.9.3.4.2.1). End clearance and lost motion are not considered in the piping stress analysis. Instead, a linear-average snubber stiffness is used in combination with that of the snubber support structure. Section 3.9.3.4.2 is revised to include this information. See the response to Question No. 81 for text changes.

DRAFT

QUESTION NO. 79
(3.9.3.4.1.C, Pages 3.9-78,80)

Provide the information to be included in snubber Design Specifications.

RESPONSE

Section 3.9.3.4.2 is revised to include the snubber design specifications information. See response to Question No. 81 for text changes.

DRAFT

QUESTION NO. 80
(3.9.3.4.1.c, Page 3.9-78,80)

Provide assurance that snubbers will be verified for proper installation and operability (not locked up) prior to preoperational testing.

RESPONSE

Pre-installation, installation and post-installation inspections of snubbers will be performed before a pre-operational test. Additional inspections are required if more than six months have elapsed between the last inspection and initial system heat-up. Tabless 14.2-4 and Section 3.9.3.4.2 are modified to include this pre-operational requirements for snubbers. Revised Table 14.2-4 is attached. The revised Section 3.9.3.4.2 is included in Question No. 81 response.

<u>Acceptance Criteria</u>

a. Proper movement of affected components is verified.

b. Feeders or branches sustain an adequate share of the decaying air supply as required by the operational mode.

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(P-100.3) MECHANICAL SNUBBER TESTING

TEST OBJECTIVE

The test objective is to verify adequate pre-service examination to mechanical snubbers on all safety related systems.

PRE-REQUISITES

All pre-installation, installation, and post-installation inspections have been performed on mechanical snubbers by designated inspection organizations.

TEST METHOD

Verify through document review that all inspection activities have been completed, verified, and signed. Reviews will be made by systems and additional visual inspections will be made if original inspections are performed more than 6 months prior to initial heatup of the system.

ACCEPTANCE CRITERIA

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

(2) Location, orientation, position setting, and configuration are according to design drawings and specifications.

(3) Snubbers are not seized, frozen, or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

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

If inspection for items 1 and 4 are performed more than 6 months prior to initial system heatup, reverify and document.

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QUESTION NO. 81
(3.9.3.4.1.c, Pages 3.9-78,80)

Provide a list of all systems utilizing snubbers in the FSAR.

This list should include:

1) number of snubbers utilized

2) type of each snubber

3) whether the snubber is constructed to Subsection NF

4) whether the snubber is used as a shock, vibration, or dual purpose snubber

5) for snubbers identified in (4) above, whether the snubber and component were evaluated for fatigue strength.

RESPONSE

See response to Question No. 75 for Item 3 and response to Question No. 77 for Items 4 and 5.

Section 3.9.3.4.2 is revised to address the snubber design, analysis and tests. Table 3.9-17 is added to include the following information pertaining to the snubber design:

(a) System ID,

(b) Snubber Type,

(c) Fabricator, and

(d) Rated Load Range

Note that the revision to Section 3.9.3.4.2 also includes the revision as results of Question Nos. 75, 78, 79 and 80.

MEB-81

3.9.3.4.2 Supports Not Furnished with the NSSS

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Component supports consist of spring hangers, rigid hangers, restraints and shock suppressors. Spring hangers are designed for component gravity loads in cold and in hot conditions of the piping system. Rigid hangers and restraints are designed for loads due to gravity, thermal expansion, seismic, and other dynamic events associated with upset, emergency, and faulted plant conditions. Shock suppressors are designed for seismic and other dynamic events associated with upset, emergency, and faulted plant conditions.

The shock suppressors are dynamically tested for sinusoidal forcing function, and where applicable, the relative displacement across the suppressor versus time is recorded. At least two suppressors of each size and model are tested at both room temperature and elevated temperature for sinusoidal forcing function. Experimental or analytical data are obtained on suppressor cyclic life versus applied cyclic force, at up to 50% of the normal rated load, at frequencies of 3 Hz, 15 Hz, and 33 Hz. Functioning of the shock suppressor under 300°F temperature for a short duration under rated load is demonstrated.

The design loading combinations for ASME Code Class 1, 2, and 3 component supports, categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted, are given in Table 3.9-21. This table also provides the stress limits for each plant operating condition.

3.9.4 CONTROL ROD DRIVE SYSTEM

The discussion in this Section includes the CRD mechanism (CRDM), the hydraulic control unit (HCU), the condensate supply system, and the scram discharge volume, and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information on CRD System

Descriptive information on the CRD system is contained in Section 4.6.

3.9.4.2 Applicable CRD System Design Specifications

The CRD system is designed to meet the functional design criteria as outlined in section 4.6, and consists of the following:

a. Locking piston CRD

b. HCU

c. Hydraulic power supply (pumps)

3.9.3.4.2.1 Design Basis

Subsection NF of ASME Code Section III is not used for the Limerick design. The codes used instead are ANSI B31.7 for nuclear class piping and ANSI B31.7 for non-nuclear class piping. For a graphical definition of jurisdictional boundaries between pipe supports and supporting structures, refer to Figures 3.9-9 and 3.9-10. MEB-75

The design loading combinations for supports for ASME Code Class 1, 2 and 3 components, categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted are given in Table 3.9-21. This table also provides the stress limits for each plant operating condition. The loads imposed on the ASME Class 1, 2 and 3 active valves and pumps are limited to valves below the code allowable loads to ensure operability of the active components by the design of the supports. The supports are designed to remain elastic under the maximum loads. The minor local deformations associated with the elastic deformation of the support will not impair operability of the active components. MEB-81

3.9.3.4.2.2 Snubbers

Snubbers are used in seismic Category I systems. Both inside and outside containment snubbers are the mechanical type. The mechanical snubbers are purchased from Pacific Scientific Corporation with load ratings appropriate for the design conditions and load combinations. A summary of the snubber design is provided in Table 3.9-17.

3.9.3.4.2.2.1 Analytical Methods

The methodology utilizied for the stress analysis of seismic Category I, 2½" and larger piping systems is as follows:

a. For systems designed to seismic and hydrodynamic loads, the flexibility of the pipe supports are considered in the piping stress analysis. A stiffness tolerance criteria is used to facilitate support design and installation.

b. For systems designed to seismic loads, only the supports are considered as rigid members in the piping stress analysis model and are designed such that their fundamental frequencies in the direction of the applied load is within the rigid range of the seismic response spectra.

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3.9.3.4.2.2.2 Snubber Design Specification

The purchase specification of shock suppressors (snubbers) covers the following criteria for supplier's performance qualification tests and load tests. Only mechanical snubbers are specified for LGS.

(a) The friction resistence of the suppressor to normal pipe movement shall be a maximum of 1 percent of the rated load of the unit or 5 lb., whichever is greater.

(b) The suppressor shall limit the acceleration of the pipe to a maximum of 0.02g when subjected to any load up to the normal rated load.

(c) The total lost movement at the suppressor shall not exceed ±.040 inches due to any applied dynamic cycle load from 3 to 33 cps up to the rated load at the unit.

(d) The suppressor shall be designed for an exposure to a temperature of 40°F prior to initial startup and 200°F during continuous operations and to a radiation dose of 6.4×10^7 rads during the life of the plant. Functioning of the schock suppressor under 340°F temperature for a short duration under rated load shall be demonstrated.

3.9.3.4.2.2.3 Snubber Performance Test

Production Test: This type of test is required to be performed on each unit.

(a) Check unit to confirm that it operates freely over the total stroke.

(b) Measure and record the force required to initiate motion over the stroke in tension and compression.

(c) On units which allow movement after the initial suppression of load, determine that the maximum acceleration level is not exceeded. This requirement must be met in both tension and compression at room temperature.

(d) Measure and record lost motion of the snubber mechanism.

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<u>Qualification Tests</u>: These types of tests are to be performed on randomly selected production models. These tests are used to demonstrate the required load performance (load rating) and specified displacement when subjected to dynamic load cycling. Also included in these tests are low temperature, high temperature, humidity, radiation and faulted load conditions.

3.9.3.4.2.3 Struts

See Section 3.9.3.4.1.1d.

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CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

TABLES (Cont'd)

MEB-81

TABLE 3.9-17

SEISMIC CATEGORY I SYSTEM SNUBBER DESIGN INFORMATION

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SYSTEM ID	TYPE OF SNUBBER	FABRICATOR	RATED LOAD RANGE
Main Steam and Relief Valve Discharge	Mechanical	Pacific Scientific	(Later)
Main Steam-Turbine Loads and Bypass Condenser	Mechanical to	Pacific Scientific	
Feedwater	Mechanical	Pacific Scientific	
Standby Liquid Control	Mechanical	Pacific Scientific	
RHR	Mechanical	Pacific Scientific	
RHR Relief Valve to Suppression Pool	Mechanical	Pacific Scientific	
Core Spray	Mechanical	Pacific Scientific	
HPCI Steam Inlet	Mechanical	Pacific Scientific	
HPCI	Mechanical	Pacific Scientific	
RCIC Steam Inlet	Mechanical	Pacific Scientific	
RCIC	Mechanical	Pacific Scientific	
Fuel Pool Cooling	Mechanical	Pacific Scientific	
Reactor Water Cleanup	Mechanical	Pacific Scientific	

TABLE 3.9-17 (CONT'D)

SEISMIC CATEGORY I SYSTEM SNUBBER DESIGN INFORMATION

SYSTEM ID	TYPE OF SNUBBER	FABRICATOR	RATED LOAD RANGE
RHR Service Water	Mechanical	Pacific Scientific	(Later)
Diesel Generator System	Mechanical	Pacific Scientific	
ECCS Pump Loop	Mechanical	Pacific Scientific	
MS Isolation Valve Leakage Control Sys.	Mechanical	Pacific Scientific	
Containment Instrumentation	Mechanical	Pacific Scientific	

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QUESTION NO. 82
(3.9.3.4.6, Page 3.9-80)

Have BOP supports been designed to Subsection NF?

RESPONSE

The BOP supports are not designed to Subsection NF. For the pipe support design, the ANSI B31.7 Code is used for nuclear Class 1, 2 and 3 piping; and the ANSI B31.1 Code is used for non-nuclear piping.

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QUESTION NO. 83
(3.9.3.1, Table 3.9-6)

Provide a table summarizing service limits for ASME Code Class 1, 2, 3 and CS components and their supports including piping.

RESPONSE

See response to Question No. 68 and the attached revision to Table 3.9-6, load combination and acceptance criteria for ASME Code Class 1, 2, and 3, NSSS piping and equipment including core support components. For BOP ASME Code Class 1, 2 and 3 piping and supports, see Table 3.9-21 and the revised Tables 3.9-12 and 3.9-16.

TABLE 3.9-12

DESIGN CRITERIA FOR ASME CODE CLASS 1 NON-NSSS PIPING

MEB-83

CONDITION	STRESS LIMITS[1]
Design	NB-3221 and NB-3652
Normal and upset	NB-3223 and NB-3654
Emergency	NB-3224 and NB-3655
Faulted	NB-3225 and NB-3656

[1] As specified by ASME Code Section III, 1977 EDITION THROUGH SUMMER 1979 ADDENDA.

TABLE 3.9-16

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 PIPING

CONDITION	STRESS LIMITS[1]
Design, normal, upset, and emergency	The piping shall conform to the requirements of Section III, paragraphs NC-3600 and ND-3600.
Faulted	The piping shall conform to requirements of ASME Code Case 1606.

[1] As specified by ASME Code Section III, 1971 through Winter 1972 Addenda EXCEPT THE FOLLOWING:

NUCLEAR CLASS 2 AND 3 FLANGES ARE ANALYZED IN ACCORDANCE WITH ASME CODE SECTION III 1977 EDITION THROUGH 1979 SUMMER ADDENDA.

MEB-83

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QUESTION NO. 84
(3.9.4.2, Page 3.9-81)

Verify that CRD components forming part of the reactor coolant pressure boundary are treated as Class 1.

RESPONSE

The CRD components forming part of the reactor coolant pressure boundary are classified as an appurtenance, Class 2 and seismic Category I as shown in Table 3.2-1.

Accordingly, Section 3.9.4.2 is revised as attached.

d. Interconnecting piping

e. Flow and pressure and isolation valves

f. Instrumentation and electrical controls

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Those components of the CRD forming part of the primary pressure boundary are designed according to ASME B&PV Code, Section III, class 2.

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1; and the components are designed according to the codes and standards governing the individual quality groups.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Section 3.9.1.1, faulted conditions in Section 3.9.1.4, seismic testing in 3.9.2.2a. and loading combinations in Table 3.9-6(v).

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRDs and CRD housings are evaluated analytically, and the design load combinations and stress limits are listed in Table 3.9-6. For the noncode components, experimental testing is used to determine the CRD performance under all possible conditions, as described in Section 3.9.4.4.

3.9.4.3.1 Control Rod Drive Housing Supports

The CRD housing support system functions are described in Section 4.6.1.2.

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of the following tests:

a. Development tests

b. Factory quality control tests

c. 5-year maintenance life tests

d. 1.5% design life tests

e. Operational tests

f. Acceptance tests

g. Surveillance tests

DRAFT

QUESTION NO. 85
(3.9.5)

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

Philadelphia Electric Company will reduce the preload on the beams from 30 to 25 kips in accordance with General Electric recommendations. This increases the expected life of the beams to 19-40 years. In-service inspection of the jet pump hold down beam will be performed to detect cracking. Inspection frequencies will be based on a lead plant experience and GE testing, and will be such that any crack initiation will be detected prior to beam failure.

Accordingly, Section 3.9.5.1.1.8 is revised.

3.9.5.1.1.8

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 holddown 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. MEB-85

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3.9.5.1.1.9 Steam Dryers

The steam dryer assembly is not a core support structure. 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 and 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 wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

3.9.5.1.1.10 Feedwater Spargers

These components are not core support structures. They are discussed here to describe flow paths in the vessel. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle, and is shaped to conform to the curvature of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers, and is discharged radially inward, mixing the cooler feedwater with the downcomer flow from the steam separators and steam dryer, before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus, and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.1.11 Core Spray Lines

This component is not a core support structure. It is discussed here because the core spray lines are the means for directing flow to the core spray nozzles, which distribute coolant during accident conditions.

INSERT FOR SECTION 3.9.5.1.1.8

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Philadelphia Electric Company will reduce the preload on the beams from 30 to 25 kips in accordance with General Electric recommendations. This increases the expected life of the beams to 19-40 years. In-service inspection of the jet pump hold down beam will be performed to detect cracking. Inspection frequencies will be based on a lead plant experience and GE testing, and will be such that any crack initiation will be detected prior to beam failure.

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QUESTION NO. 86
(3.9.5.1)

Verify that the design and analysis of your reactor internals is equivalent to Subsection NG.

RESPONSE

Limerick reactor internals were designed and procured prior to the issuance of Subsection NG of the ASME Code, Section III. However, an earlier draft of ASME Code was used as a guide in the design of the reactor internals. These criteria are presented in Section 3.9.5.3 and were used in-lieu of Subsection NG. Subsequent to the issuance of Subsection NG, comparisons were made to assure that the pre-NG design meets the equivalent level of safety as presented by Subsection NG.

DRAFT

QUESTION NO. 87
(3.9.6.1, Page 3.9-95)

Pumps and valves that are not to be inservice tested and are safety-related Code Class 1, 2, or 3 must be specifically identified in a request for relief containing the following information:

a. Identify component for which relief is requested:

   (1) Name and number as given in FSAR

   (2) Function

   (3) ASME Section III Code Class

   (4) For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000.

b. Specifically identify the ASME Code requirement that has been determined to be impractical for each component.

c. Provide information to support the determination that the requirement in item (b) is impractical; i.e., state and explain the basis for requesting relief.

d. Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.

e. Provide an explanation as to why the proposed inservice testing will provide an acceptable level of quality and safety and not endanger the public health and safety.

f. Provide the schedule for implementation of the procedure(s) in item (d).

RESPONSE

With the exception of subparagraph (e), the concerns expressed in the above Question No. 87 have been addressed in the Pump and Valve Inservice Testing Program Plan which was forwarded by letter from Mr. John Kemper (PECO) to Mr. A. Schwencer, Chief Licensing Branch No. 2, Division of Licensing, NRC, dated December 28, 1982. The response to paragraph (e) is given below.

All required leak rate tests are conducted per ASME Section XI, IWV-3420. Exercise and stroke time tests for some valves are delayed to cold shutdown or refueling as permitted by IWV-3411 and IWV-3521 because they are either inaccessible during plant operations or their exercising would put the plant in an unsafe condition during normal operations. These valves are all identified in specific relief requests. Delaying these tests to cold shutdown or refueling will provide an acceptable level of quality and safety and not endanger the public health and safety.

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QUESTION NO. 88
(3.9.6.1, Page 3.9-95)

The applicant must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the revised rules of 10CFR, Part 50, Section 50.55a, paragraph (g).

RESPONSE

Section 3.9.6 is revised to reflect the above compliance.

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addenda (this being the code in effect six months prior to the LGS construction permit date of June 1974). That publication does not require preservice and inservice testing of pumps and valves to ensure operational readiness. The requirements for inservice testing of pumps and valves were added as Subsections IWV and IWP to ASME B&PV Code, Section XI, Summer 1973 Addenda, effective December 30, 1973. The preservice testing program for assessing operational readiness of pumps and valves is conducted, however, to the extent practical within design limitations, so that it complies with the intent of the 1980 ~~1974~~ Edition of ASME B&PV Code, Section XI, with addenda through the ~~summer of 1975~~. Winter of 1980. MEB-88

Operational readiness of pumps and valves is assessed in the first 120-month inservice tests. These tests will comply, to the extent practical within design limitations, with editions of ASME B&PV Code, Section XI, and with addenda in effect no more than twelve (12) ~~six~~ months prior to the commercial operation of the LGS.

During successive 120-month periods, inservice tests of pumps and valves for assessing operational readiness comply, to the extent practical within design limitations, with editions of ASME B&PV Code, Section XI, and with addenda in effect no more than ~~six~~ twelve (12) months prior to each 120-month period.

3.9.6.1 Inservice Testing of Pumps

Safety-related ASME Section III Class 1, 2, and 3 pumps are inservice tested where practical, in accordance with Subsection IWP of ASME B&PV Code, Section XI, to establish and detect changes in the hydraulic and mechanical reference parameters. Pumps to be tested and their respective Section III Code Class are listed in Table 3.9-31.

The pump test program meets, to the extent practical, the requirements for establishing pump reference values in accordance with IWP-3000 of ASME B&PV Code, Section XI. The allowable ranges of inservice test quantities and corrective actions are in accordance with IWP-3200 of ASME B&PV Code, Section XI.

The frequency and duration of periodic tests for each pump are discussed in the technical specifications, and are in accordance with IWP-3300, IWP-3400, and IWP-3500 of ASME B&PV Code, Section XI.

The methods of measurement are in accordance with IWP-4000 of ASME B&PV Code, Section XI and are included in the inservice testing program.

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QUESTION NO. 89
(3.9.6.1, Page 3.9-95)

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated 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 for RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure system. 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 IVW-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 technical specifications which will require corrective 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 position on leak rate limiting conditions for operation is that leak rates must be equal to or less than 1 gallon per minute (GPM) for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

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.

The applicant has not yet submitted its program for the pre-service and inservice testing of pumps and valves; therefore, we have not yet completed our review.

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RESPONSE

The concerns expressed in this question have been addressed in the Pump and Valve Inservice Testing Program Plan which was forwarded by letter from Mr. John Kemper (PECO) to Mr. A. Schweneer, Chief Licensing Branch No. 2, Division of Licensing, NRC, dated December 28, 1982.

The Limiting Conditions for Operation (LCO) are included in the Technical Specifications leak rate testing of RHR and Class II and III valves are included in the Pump and Valve Inservice Testing Program Plan transmitted to the NRC.