GE Nuclear Energy

LICENSING AND CONSULTING SERVICES

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

DATE: May 11,1990
MEMO: EEN-9019P

TO: Distribution

FROM: E. Earl Nichols
ABWR-SSAR Licensing Program Manager
(408) 925-4288

SUBJECT: ABWR SSAR AMENDMENT 11 TRANSMITTAL CORRECTIONS

Reference: Submittal of Amendment 11, Non-Proprietary Information, to GE's ABWR SSAR, MFN No. 037-90, dated May 23, 1990.

Submittal of Amendment 11, Proprietary Information, to GE's ABWR SSAR, MFN No. 038-90, dated May 2, 1990.

Attached is a corrected (correcting the date of issue from May 23 to May 2) copy of the subject non-propreiteary transmittal letter. Please replace your copy of this letter with the attached copy.

In addition, there were a couple of printing errors that occurred in some of the packages sent out. Please make the following changes to the pages in the Amendment 11 package so that you will be assured of having the correct pages in your copy of the ABWR SSAR.

Replacement

- Pull all Chapter 3 pages from Amendment 11 package and replace with attached Chapter 3 pages.
- Pull all Chapter 19B & 19G pages from Amendment 11 package and replace with attached Chapter 19B & 19G pages.
- Pull Chapter 20 pages, from 20.3-1.2 through 20.3-131, and replace with attached Chapter 20 pages, from 20.3-1.2 through 20.3-131.
- Pull Chapter 20 page 20.3-288.1 and replace it with 20.3-290.1 (Proprietary Page).

Additions

- Insert Chapter 11 pages: 11.5-ii,iii 11.5-1,2 11.5-7,8 11.5-9,10 11.5-11,12
- Insert Chapter 20 page: 20.3-141

CC: R. Mitchell, J.N. Fox

GE Nuclear Energy

May 2, 1990 (Corrected Date)

MFN No. 037-90

Docket No. STN 50-605

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Charles L. Miller, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **Submittal of Amendment 11, Non-Proprietary Information, to GE's ABWR SSAR**

Reference: Submittal of Amendment 11, Proprietary Information, to GE's ABWR SSAR, MFN No. 038-90, dated May 2, 1990

Dear Mr. Miller:

Enclosed are thirty-four copies of selected sections of Chapter 1, *Introduction and General Description of Plant*, Chapter 3, *Design of Structures, Components, Equipment, and Systems*, Chapter 5, *Reactor Coolant System and Connected Systems*, Chapter 6, *Engineered Safety Features*, Chapter 7, *Instrumentation and Control Systems*, Chapter 9, *Auxiliary Systems*, Chapter 10, *Steam and Power Conversion*, Chapter 11, *Radioactive Waste Management*, Chapter 13, *Conduct of Operation*, Chapter 14, *Initial Test Program*, Chapter 19, *Response to Severe Accident Policy Statement*, Chapter 20, *Question and Response Guide*, of the Standard Safety Analysis Report (SSAR) for the Advanced Boiling Water Reactor (ABWR).

This submittal includes additions covering the following: Response to standby gas treatment system questions, includes modifications to the design; Addition of a flammability control system to the atmospheric control system; Addition of the initial test program for the turbine island and radwaste facilities; Addition of the reactor service and turbine water systems; Resolution of draft safety evaluation report open items for Chapters 4, 5 and 6 (except for preservice/inservice inspection plan); Resolution of preliminary draft safety evaluation report items for Chapters 3 (except for inservice testing plan); Expansion of Chapter 7 that includes turbine island and radwaste facility instrumentation and controls; Balance of Chapter 7 and 10 question responses; Update of applicable USIs/GSIs; and Closeout of design related emergency preparedness issue.

In addition there are other changes to various sections of the SSAR which are identified, along with the above changes, on the page change instruction sheet.

Please note that all or parts of the following sections contain information that is designated as General Electric Company proprietary information: 7.7, 19.5, 19B and 20.3. This information is being submitted under separate cover.

Sincerely,

G.L. Sozzi, Acting Manager
Licensing and Consulting Services

cc: D. R. Wilkins (GE)
T. E. Murley (NRC)
D. M. Crutchfield (NRC)
D. J. McGoff (DOE)
F. J. Miraglia, Jr. (NRC)
D. C. Scaletti (NRC)
K. E. Stahlkopf (EPRI)
F. A. Ross (DOE)

EEN9018

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 SUMMARY DESCRIPTION

This section contains an evaluation of the principal design criteria of the ABWR Standard Plant as measured against the NRC General Design Criteria for Nuclear Power Plants, 10CFR50 Appendix A. The general design criteria, which are divided into six groups with the last criterion numbered 64, are intended to establish minimum requirements for the principal design criteria for nuclear power plants.

The NRC General Design Criteria were intended to guide the design of all water-cooled nuclear power plans; separate BWR-specific criteria are not addressed. As a result, the criteria are subject to a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of the ABWR design to the interpretation of the criteria is discussed. For each criterion, a specific assessment of the plant design is made and a complete list of references is included to identify where detailed design information pertinent to that criterion is treated in this safety analysis report (SAR).

Based on the content herein, the design of the ABWR design fully satisfies and is in compliance with the NRC General Design Criteria.

3.1.2 EVALUATION AGAINST CRITERIA

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

3.1.2.1.1.1 Criterion 1 Statement

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

3.1.2.1.1.2 Evaluation Against Criterion 1

Safety-related and non-safety-related structures, systems, and components are identified on Table 3.2-1. The total quality assurance program is described in Chapter 17 and is applied to the safety-related items. The quality requirements for non-safety-related items are controlled by the quality assurance program described in Chapter 17 in accordance with the functional importance of the item. The intent of the quality assurance program is to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures as well as the documentation of the foregoing by keeping appropriate records. The total quality assurance program is responsive to and in conformance with the intent of the quality-related requirements of 10CFR50 Appendix B.

Structures, systems, and components are identified in Section 3.2 with respect to their location, service and their relationship to the safety-related or non-safety related function to be performed. Recognized codes and standards are applied to the equipment per the safety classifications to assure meeting the required safety-related function.

Documents are maintained which demonstrate that all the requirements of the quality assurance program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided, and the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that any desired item of information is retrievable for reference. These records will be maintained during the life of the operating licenses.

The detailed quality assurance program is in conformance with the requirements of Criterion 1.

For further discussion, see the following sections:

	Chapter/ Section	Title
(1)	1.2	General Plant Description
(2)	3.2	Classification of Structures, Components, and Systems

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

3.1.2.1.2.1 Criterion 2 Statement

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structure systems and components shall reflect:

(1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated;

(2) Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) The importance of the safety functions to be performed.

3.1.2.1.2.2 Evaluation Against Criterion 2

Since the ABWR design is designated as a standard plant, the design bases for safety-related (See Subsection 3.1.2.1.1.2) structures, systems, and components, cannot accurately reflect the most severe of the natural phenomena that have been historically reported for each possible site and their surrounding areas. However, the envelope of site-related parameters which blanket the majority of potential sites in the conterminous United States is defined in Chapter 2. The design bases for these structures, systems, and components reflect this envelope of natural phenomena including appropriate combinations of the effects of normal and accident conditions with this envelope. The design bases meet the requirements of Criterion 2.

Detailed discussion of the various phenomena considered and design criteria developed are presented in the following sections:

	Chapter/ Section	Title
(1)	2.0	Summary of Site Characteristics
(2)	3.2	Classification of Structures, Components, and Systems
(3)	3.3	Wind and Tornado Loadings
(4)	3.4	Water Level (Flood) Design

TABLE 3.2-1

CLASSIFICATION SUMMARY

The classification information is presented System-wise* in the following order:**

Table 3.2-1 Item No.	MPL Number**	Title
B		**Nuclear Steam Supply Systems**
B1	B11/J10 J11/J12	Reactor Pressure Vessel System*/Fuel*
B2	B21	Nuclear Boiler System*
B3	B31	Reactor Recirculation System
C		**Control and Instrument Systems**
C1	C11/C12	CRD System*
C2	C31	Feedwater Control System
C3	C41	Standby Liquid Control System
C4	C51	Neutron Monitoring System*
C5	C61	Remote Shutdown System
C6	C71	Reactor Protection System*
D		**Radiation Monitoring Systems**
D1	D11	Process Radiation Monitoring* System
D2	D23	Containment Atmospheric Monitoring System*
E		**Core Cooling Systems**
E1	E11	RHR System*
E2	E22	High Pressure Core Flooder System*
E3	E31	Leak Detection and Isolation System*
E4	E51	RCIC System*
F		**Reactor Servicing**
F1	F11	Fuel Servicing Equipment
F2	F13	RPV Servicing Equipment
F3	F14	RPV Internal Servicing Equipment
F4	F15	Refueling Equipment
F5	F16	Fuel Storage Equipment
G		**Reactor Auxiliary Systems**
G1	G31	Reactor Water Cleanup System
G2	G41	Fuel Pool Cooling and Cleanup System
G3	G51	Suppression Pool Cleanup System
H		**Control Panels**
H1	H11	Main Control Room Panel*
H2	H21	Local Control Panels*

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

*** *Only those systems that are in the ABWR Standard Plant scope are included in this table.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Table 3.2-1 Item No.	MPL Number**	Title
J	**Nuclear Fuel**	
		See Item B1
K	**Radioactive Waste Systems**	
K1	K11	Radioactive Drain Transfer System
K2	K17	Radwaste System
N	**Power Cycle Systems**	
N1	N11/N21 N22/N25 N26/N27 N31/N32 N33/N34 N35/N36 N37/N38 N39/N41 N42/N43 N44/N51 N61/N71 N72	Power Conversion System
N2	N62	Offgas System
P	**Station Auxiliary Systems**	
P1	P13	Makeup Water System (Condensate)
P2	P21	Reactor Building Cooling Water System*
P3	P22	Turbine Building Cooling Water System
P4	P24/P25	HVAC Cooling Water Systems*
P5	P41	Reactor Cooling Water System
P6	P42	Turbine Cooling Water System
P7	P51/P52 /P54	Instrument/Service Air /High Pressure Nitrogen Systems
R	**Station Electrical Systems**	
R1	R42	DC Power Supply*
R2	R10/R11/ R22	Auxiliary AC Power System*
R3	R43	Emergency Diesel Generator System*
R4	R52	Lighting and Servicing Power Supply

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B1 Reactor Pressure Vessel System/ Fuel Assemblies						
1. Reactor vessel	1	C	A	B	I	
2. Reactor vessel support skirt and stabilizer	1	C	A	B	I	
3. Reactor vessel appurtenances -pressure retaining portions	1	C	A	B	I	(g)
4. Supports for CRD housing, in-core housing and recirculation internal pump	1	C	A	B	I	
5. Reactor internal structures - feedwater, RHR/ECCS high pressure core flooder spargers	3	C	B	B	I	
6. Reactor internal structures- safety related components including core support structures (See Subsection 3.9.5)	3	C	---	B	I	
7. Reactor internal structures - non-safety related components (See Subsection 3.9.5)	N	C	---	---	---	
8. Control rods	3	C	---	B	I	
9. Power range detector hardware including startup range detector	3	C	---	B	I	
10. Fuel assemblies	3	C	---	B	I	
11. Reactor Internal Pump Motor Casing	1	C	A	B	I	
B2 Nuclear Boiler System						
1. Vessels - level instrumentation condensing chambers	2	C	B	B	I	

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TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
2. Vessel - air accumulators (for ADS and SRVs)	3	C	C	B	I	
3. Piping including supports - safety/relief valve discharge	2/3	C	B/C	B	I	(h)

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B2 Nuclear Boiler System (Continued)						
4. Piping including supports - main steamline (MSL) and feedwater (FW) line within outermost isolation valve	1	C,SC	A	B	I	
5. Piping including supports - MSL and FW from outermost isolation valve to and including seismic interface restraint and FW from outermost isolation to and including shutoff valve	2	SC	B	B	I	
6. Piping including supports - MSL from the seismic interface restraint to the turbine stop valve	N	T	B	---	---	
7. Deleted						
8. Piping - FW beyond seismic interface restraint	N	T	D	---	---	
9. Seismic Interface Restraint - MSL/FW	2	SC	B	B	I	
10. Pipe whip restraints - MSL/FW	3	SC,C	---	B	---	
11. Piping including supports - other within outermost isolation valves						
a. RPV head vent	1	C	A	B	I	(g)
b. RPV head spray	1	C	A	B	I	(g)
c. Main steam drains	1	C,SC	A	B	I	(g)
12. Piping including supports - other beyond outermost isolation valves						
a. RPV head vent	N	C	D	---	---	
b. RPV head spray	N	SC	D	---	---	
c. Main steam drains	N	SC	D	---	---	

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TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
C5 Remote Shutdown System						
Components of this system are included under B2, E1, E4, G3, H2, and P2.						
1. Electrical modules with safety-related function	3	C,SC,RZ, X	---	B	I	
2. Cable with safety related function	3	RZ	---	B	I	
C6 Reactor Protection System						
1. Electrical modules with safety-related function	3	SC,X,T, RZ	---	B	I	
2. Cable with safety related functions		3	SC,X,T, RZ	---	B	I
3. Electrical Modules, other	N	T,X	---	---	---	(u)
4. Cable, other	N	T,X	---	---	---	(u)
D1 Process Radiation Monitoring System (includes gaseous and liquid affluent monitoring)						
1. Electrical modules - with with safety-related functions (includes monitors)	3	SC,X,RZ	---	B	I	
2. Cable with safty-related functions	3	SC,X,RZ	---	B	I	
3. Electrical Modules, other	N	T,SC,RZ, X,W	---	---	---	(u)
4. Cable, other	N	T,SC,RZ, X,W	---	---	---	(u)

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
D2 Containment Atmospheic Monitoring System						
1. Component with safety-related	3	C,SC	---	B	I	
E1 RHR System						
1. Heat exchangers-primary side	2	SC	B	B	I	
2. Heat exchangers including supports-secondary side	3	SC	C	B	I	
3. Piping including supports * within outermost isolation valves	1/2	C,SC	A/B	B	I	(g)
4. Containment spray piping including supports and spargers, within and including the outermost isolation valves	2	C	B	B	I	
4a. Piping including supports beyond outermost isolation valves	2/3	SC	B/C	B	I	(g)
5. Main Pumps including supports	2	SC	B	B	I	
6. Main Pump motors	3	SC	---	P	I	
7. Valves - isolation, (LPFL line) including shutdown suction line isolation valves	1	C,SC	A	B	I	(g)
8. Valves - isolation, other (pool suction valves and pool test return valves)	2	C,SC	B	B	I	(g)
9. Valves beyond isolation valves	2/3	SC	B/C	B	I	(g)

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* *The RHR/ECCS low pressure flooder spargers are part of the reactor pressure vessel system, see Item B1.5.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
E1 RHR System (Continued)						
10. Mechanical modules with safety-related functions	3	SC	C	B	I	
11. Electrical modules with safety-related function	3	C,SC,X	---	B	I	
12. Cable with safety-related function	3	C,SC,X	---	B	I	
13. Other mechanical and electrical modules	N	C,SC,X	---	---	---	
14. Jockey pumps including supports	2	SC	B	B	I	
15. Jockey pump motor	N	SC	---	---	---	

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TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
E2 High Pressure Core Flooder						
1. Reactor pressure vessel injection line and connected piping including supports within outermost isolation valve*	1/2	C,SC	A/B	B	I	(g)
2. All other piping including supports **	2/3	SC,O	B/C	B	I	(g)
3. Main Pump	2	SC	B	B	I	
4. Main Pump motor	3	SC	---	B	I	
5. Valves - outer isolation and within the reactor pressure vessel injection line and connected lines	1	C,SC	A	B	I	(g)
6. All other valves	2/3	SC	B/C	B	I	(g)
7. Electrical modules with safety-related function	3	C,SC,X	---	B	I	
8. Cable with safety-related function	3	C,SC,X	---	B	I	
E3 Leak Detection and Isolation System						
1. Temperature sensors	3/N	C,SC	---	B/---	I/---	(z)
2. Temperature switches	3/N	X	---	B/---	I/---	(z)
3. Pressure transmitters	3/N	C,SC	---	B/---	I/---	(z)
4. Pressure switches	3/N	X	---	B/---	I/---	(z)
5. Differential pressure transmitters (flow)	3/N	C,SC	---	B/---	I/---	(z)

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* *The ECCS high pressure core flooder spargers are part of the Reactor Pressure Vessel System, see Item B1.5.*

** *Pool suction piping, suction piping from condensate storage tank, test line to pool, pump discharge piping and return line to pool.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
F2 RPV Servicing Equipment						
1. Steamline plugs	N	SC	---	---	---	
2. Dryer and separator strongback and head strongback	N	SC	---	---	---	
F3 RPV Internal Servicing Equipment						
1. Control rod grapple	N	SC	---	---	---	
F4 Refueling Equipment						
1. Refueling equipment platform assembly	N	SC	---	---	I	(bb) 210.13
2. Refueling bellows	N	SC	---	---	---	
F5 Fuel Storage Equipment						
1. Fuel storage racks - new and spent	N	SC	---	---	I	(bb) 210.15
2. Defective fuel storage container	N	SC	---	---	---	(bb) 210.15
G1 Reactor Water Cleanup System						
1. Vessels including supports (filter/demineralizer)	N	SC	C	---	---	210.20
2. Regenerative heat exchangers including supports carrying reactor water	N	SC	C	---	---	210.20
3. Cleanup recirculation pump, motors	N	SC	C	---	---	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
G1 Reactor Water Cleanup System (Continued)						
4. Piping including supports and valves within and including outermost containment isolation valves on pump suction	1	C,SC	A	B	I	(g)
5. Pump suction and discharge piping including supports and valves from containment isolation valves back to shut-off valves at feedwater line connections	N	SC	C	---	---	(g)
6. Piping including supports and valves from feedwater lines to and including shut-off valves	2	SC	B	B	I	(g)
7. Piping including supports and valves to main condenser	N	SC,T	C	---	---	(g)
8. Non-regenerative heat exchanger tube inside and piping including supports and valves carrying process water	N	SC	C	---	---	(g)
9. Non-regenerative heat exchanger shell and piping including supports carrying closed cooling water	N	SC	D	---	---	
10. Filter/demineralizer precoat subsystem	N	SC	D	---	---	
11. Filter demin holding pumps including supports - valves and piping including supports	N	SC	C	---	---	

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TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
4. Turbine bypass piping including supports	N	T	D	--	--	
5. Turbine stop valve, turbine bypass valves, and the main steam leads from the turbine control valve to the turbine casing	N	T	D	--	--	(l)(n)(o)
6. Feedwater system components beyond outboard shutoff valve	N	T	D	N/A	N/A	
7. Turbine generator	N	T	--	--	--	
8. Condenser	N	T	--	--	--	
9. Air ejector equipment	N	T	--	--	--	
10. Turbine gland sealing system components	N	T	D	--	--	
N1 Power Conversion System (Later)						
N2 Offgas System						
1. Pressure vessels including supports	N	T	--	--	--	(p)(q)
2. Atmospheric tanks including supports	N	T	--	--	--	(p)(q)
3. 0-15 psig tanks including supports	N	T	--	--	--	(p)(q)
4. Heat exchangers including supports	N	T	--	--	--	(p)(q)
5. Piping including supports and valves	N	T	--	--	--	(p)(q)
6. Pumps including supports	N	T	--	--	--	(p)(q)

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P1 Makeup Water System (Condensate)						
1. Piping including supports and valves forming part of the containment boundary	2	C	B	B	I	
2. Condensate storage tank including supports	N	O	D	---	---	(w)
3. Condensate header - piping including supports and valves	2	SC	B	B	I	
4. Piping including supports and valves		N	O	D	---	---
5. Other components	N	O	D	---	---	
P2 Reactor Building Cooling Water System						
1. Piping and valves forming part of primary containment boundary	2	SC,C	B	B	I	(g)
2. Other safety-related piping, including supports pumps and valves	3	SC,C	C	B	I	
3. Electrical modules with with safety-related function	3	SC,C,X	---	B	I	
4. Cable with safety-related function	3	SC,C,X	---	B	I	
5. Other mechanical and electrical modules	N	SC,C,X,M	---	---	---	
P3 Turbine Building Cooling Water System	N	T	D	---	---	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P4 HVAC Cooling Water Systems						
1. Chillers, pumps, valves, and piping including supports-Safety-related HVAC support	3	SC,X	C	B	I	
2. Chillers, pumps, valves, and piping including supports - non-safety related HVAC support	N	C,SC,RZ,	---	---	---	
3. Piping including supports and valves forming part of containment boundary	2	C,SC	B	B	I	
4. Electrical modules and cable with safety-related function	3	SC,X	---	B	I	
5. Other mechanical and electrical modules	N	C,SC,RZ, T,X	---	---	---	
P5 Reactor Cooling Water System						
1. Safety-related piping including supports, piping and valves	3	U,O,X	C	B	I	
2. Electrical modules and cables with safety-related function	3	U,O,X	---	B	I	
3. Other non-safety related mechanical and electrical modules	N	U,O,X	---	---	---	
P6 Turbine Cooling Water System						
1. Non-safety related piping including supports, piping and valves	N	P,O,T	---	---	---	

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TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
2. Eletrical modules and cables with non-safety related function	N	P,O,T	---	---	---	
P7 Instrument/Service Air Systems						
210.20 1. Containment isolation including supports valves and piping	2	C	B	B	I	
2. Other non-safety related mechanical and electrical components	N	SC,RZ, X,T,H, W,C	---	---	---	
P8 High Pressure Nitrogen Systems						
1. Containment isolation including supports valves and piping	2	C	B	B	I	
2. Piping including supports with safety-related function	3	SC,C	C	B	I	
3. Electric modules with safety-related functions	3	RZ,X	---	B	I	
4. Cable with safety-related function	3	SC,RZ, X	---	B	I	
5. Other non-safety related mechanical and electrical components	N	SC,RZ, X	---	---	---	
R1 DC Power Supply - Nuclear Island						
435.54 1. 125 volt batteries, battery racks, battery chargers, and distribution equipment	3	SC,C,X, RZ	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
2. Control power cables (including under ground cable system, cable splices, connectors and terminal blocks)	3	SC,C,X, RZ	---	B	I	
3. Conduit and cable trays and their supports	3	SC,C,X, RZ	---	B	I	
4. Protective relays and control panels	3	SC,X,RZ	---	B	I	
5. Containment electrical penetrations assemblies	3	SC,C	---	B	I	
6. Motors	3	SC,C,X, RZ	---	B	I	
R2 Auxiliary AC Power System						
1. 6900 volt switch gear	3	SC,X,RZ	---	B	I	
2. 480 volt load centers	3	SC,X,RZ	---	B	I	
3. 480 volt motor control centers	3	SC,X,RZ	---	B	I	
4. 120 VAC safety related distribution equipment including inverters	3	SC,X,RZ	---	B	I	
5. Control and power cables (including underground cable systems, cable splices, connectors and terminal blocks)	3	SC,C,X RZ	---	B	I	
6. Conduit and cable trays and their supports	3	SC,C,X RZ	---	B	I	
7. Containment electrical penetration assemblies	3	SC,C,X RZ	---	B	I	
8. Transformers	3	SC,C,X RZ	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
9. Motors	3	SC,C,X,RZ	---	B	I	
10. Load sequencers	3	SC,X,RZ	---	B	I	
11. Protective relays and control panels	3	SC,X,RZ	---	B	I	
12. Valve operators	3	SC,C,X,RZ	---	B	I	
R3 Emergency Diesel Generator System						
1. Starting air receiver tanks piping including supports from and including check valve and downstream piping including supports and valves	3	RZ	C	B	I	(y)
2. Starting air compressor and motors	N	RZ	---	---	---	
3. Combustion air intake and exhaust system	3	RZ,O	C	B	I	
4. Safety-related piping including supports valves - fuel oil system, diesel cooling water system, and lube oil system	3	RZ,O	C	B	I	
5. Pump motors - fuel oil system, diesel cooling water system and lube oil system	3	RZ,O	---	B	I	
6. Diesel generators	3	RZ	---	B	I	(y)
7. Mechanical and electrical modules with safety-related functions	3	RZ,O,X	---	B	I	
8 Cable with safety-related functions	3	RZ,O,X	---	B	I	
9. Other mechanical and electrical modules	N	RZ,O	---	---	---	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
R4 Lighting and Servicing Power Supply						
1. Emergency Lighting	N	SC,C,X RZ	---	---	---	
T1 Primary Containment System						
1. Primary containment vessel (PCV) - reinforced concrete containment vessel (RCCV)	2	C	B	B	I	
2. Vent system (vertical flow channels and horizontal discharges	2	C	B	B	I	
3. Suppression chamber/drywell vacuum breakers	2	C	B	B	I	
4. PCV penetrations and drywell steel head	2	C	B	B	I	
5. Upper and lower drywell airlocks	2	C,SC	---	B	I	
6. Upper and lower drywell equipment hatches	2	C,SC	---	B	I	
7. Lower drywell access tunnels	2	C	---	B	I	
8. Suppression chamber access hatch	2	C,SC	---	B	I	
9. Safety related instrumentation	2	C,SC	---	B	I	
T2 Containment Internal Structures						
1. Reactor vessel stabilizer truss	3	C	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
T2 Containment Internal Structures (Continued)						
2. Support structures for safety-related piping including supports and equipment	3	C	---	B	I	
T3 RPV Pedestal and Shield Wall						
1. RPV pedestal and shield wall	3	C	---	B	I	
2. Diaphragm floor	3	C	---	B	I	
T4 Standby Gas Treatment System						
1. All equipment except deluge piping and valves	3	SC,C,RZ	---	B	I	
2. Deluge piping and valves	N	SC	---	---	---	

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NOTES

a. A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors and mechanical modules include turbines, strainers, and orifices.

b. 1, 2, 3, N = Nuclear safety-related function designation defined in Subsections 3.2.3 and 3.2.5.

c. C = Primary Containment
H = Service Building
M = any other location
O = Outdoors onsite
RZ = Reactor Building Clean Zone (balance portion of the reactor building outside the Secondary Containment Zone)
SC = Secondary Containment portion of the reactor building
T = Turbine Building
W = Radwaste Building
X = Control Building
F = Firewater Pump House
U = Ultimate Heat Sink Pump House
P = Power Cycle Heat Sink Pump House

d. A,B,C,D = Quality groups defined in Regulatory Guide 1.26 and Subsection 3.2.2. The structures, systems and components are designed and constructed in accordance with the requirements identified in Tables 3.2-2 and 3.2-3.

--- = Quality Group Classification not applicable to this equipment.

e. B = the quality assurance requirements of 10CFR50, Appendix B are applied in accordance with the quality assurance program described in Chapter 17.

--- = Requirements of 10CFR50, Appendix B are not applicable.

f. I = The design requirements of Seismic Category I structures and equipment are applied as described in Section 3.7, Seismic Design.

--- = The seismic design requirements for the safe shutdown earthquake (SSE) are not applicable to the equipment. However, the equipment that is not safety related but which could damage Siesmic Category I equipment if its structural integrity failed is checked analytically and designed to assure its integrity under seismic loading resulting from the SSE.

g. 1. Lines one inch and smaller which are part of the reactor coolant pressure boundary shall be ASME Code Section III, Class 2 and Seismic Category I.

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

3. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate and monitor safety systems shall be Code Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

NOTES (Continued)

4. All other instrument lines:

   i Through the root valve the lines shall be of the same classification as the system to which they are attached.

   ii Beyond the root valve, if used to actuate a safety system, the lines shall be of the same classification as the system to which they are attached.

   iii Beyond the root valve, if not used to actuate a safety system, the lines may be Code Group D.

5. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system may be Code Group D.

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6. All safety-relaed instrument sensing lines shall be in conformance with the criteria of Regulatory Guide 1.151.

h. Relief valve discharge piping shall be Quality Group B and Seismic Category I.

   Safety/relief valve discharge line (SRVDL) piping from the safety/relief valve to the quenchers in the suppression pool consists of two parts: the first part is attached at one end to the safety/relief valve and attached at its other end to the diaphragm floor penetration. This first portion of the safety/relief valve discharge piping is analyzed with the main steam piping as a complete system. The second part of the safety/relief valve discharge piping extends from the penetration to the quenchers in the suppression pool. Because of the penetration on this part of the line, it is physically decoupled from the main steam piping and the first part of the SRVDL piping and is, therefore, analyzed as a separate piping system.

i. Electrical devices include components such as switches, controllers, solenoids, fuses, junction boxes, and transducers which are discrete components of a larger subassembly/module. Nuclear safety-related devices are Seismic Category I. Fail-safe devices are non-Seismic Category I.

j. The control rod drive insert lines from the drive flange up to and including the first valve on the hydraulic control unit are Safety Class 2, and non-safety related beyond the first valve.

k. The hydraulic control unit (HCU) is a factory-assembled engineered module of valves, tubing, piping, and stored water which controls two control rod drives by the application of pressures and flows to accomplish rapid insertion for reactor scram.

   Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Groups A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connection to conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

NOTES (Continued)

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but of the remaining parts and details. For example: (1) all welds are LP inspected; (2) all socket welds are inspected for gap between pipe and socket bottom; (3) all welding is performed by qualified welders; and (4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer standards and proven design techniques which are not explicitly defined within the codes for Quality Groups A, B, or C. This is supplemented by the QC technique described.

l. The turbine stop valve is designed to withstand the SSE and maintain its integrity.

m. The RCIC turbine is not included in the scope of standard codes. The assure that the turbine is fabricated to the standards commensurate with safety and performance requirements, General Electric has established specific design requirements for this component which are as follows:

1. All welding shall be qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.

2. All pressure-containing castings and fabrications shall be hydrotested at 1.5 times the design pressure.

3. All high-pressure castings shall be radiographed according to:

ASTM E-94	
E-141	
E-142	maximum feasible volume
E-71, 186 or 280	Severity level 3

4. As-cast surfaces shall be magnetic-particle or liquid-penetrant tested according to ASME Code, Section III, Paragraphs NB-2575, NC-2576, or NB-2576, and NC-2576.

5. Wheel and shaft forgings shall be ultrasonically tested according to ASTM A-388.

6. Butt welds shall be radiographed and magnetic particle or liquid penetrant tested according to the ASME Boiler and Pressure Vessel Code. Acceptance standards shall be in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph NB-5340, NC-5340, NB-5350, or NC-5350, respectively.

7. Notification shall be made on major repairs and records maintained thereof.

8. Record system and traceability shall be according to ASME Section III, NCA-4000.

9. Control and identification shall be according to ASME Section III, NCA-4000.

10. Procedures shall conform to ASME Section III, NB-5100 and NC-5100.

11. Inspection personnel shall be qualified according to ASME Section III, NB-5500 and NC-5500.

NOTES (Continued)

n. All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards is used as an alternate to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those defined in Paragraph 136.4, Nonboiler External Pipng, ANSI B31.1.

o. The following qualifications are met with respect to the certification requirements:

   1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valve to turbine casing utilizes quality control procedures equivalent to those defined in GE Publication GEZ-4982A, General Electric Large Steam Turbine Generator Quality Control Program.

   2. A certification obtained from the manufacturer of these valves and steam loads demonstrates that the quality control program as defined has been accomplished.

p. Regulatory Guide 1.143 furnishes complete design guidance relating to seismic and quality group classification and quality assurance provisions for radioactive waste management systems, structures and components.

q. Detailed seismic design criteria for the offgas system are provided in Section 11.3.

r. The portions of the MSL from the second isolation valve to the turbine stop valve and the first valve in branch lines shall be designed so that the SSE does not cause structural interaction or failure that could degrade the functioning of a Seismic Category I structure system or component to an unacceptable safety level.

s. Not used

t. There is a limited quality assurance program for the Fire Protection System.

u. Special seismic qualification and quality assurance requirements are applied.

v. Not used.

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

x. The cranes are designed to hold up their loads under conditions of OBE and to maintain their positions over the units under conditions of SSE.

y. All off-engine components are constructed to the extent possible to the ASME Code, Section III, Class 3.

z. Components associated with a safety-related function (e.g., isolation) are safety-related.

aa. Structures which support or house safety-related mechanical or electrical components are safety-related.

bb. A quality assurance requirements shall be applied to ensure that the design, construction and testing requirements are met.

3.7 SEISMIC DESIGN

All structures, systems, and equipment of the facility are defined as either Seismic Category I or non-Seismic Category I. The requirements for Seismic Category I identification are given in Section 3.2 along with a list of systems, components, and equipment which are so identified.

All structures, systems, components, and equipment that are safety-related, as defined in Section 3.2, are designed to withstand earthquakes as defined herein and other dynamic loads including those due to reactor building vibration (RBV) caused by suppression pool dynamics. Although this section addresses seismic aspects of design and analysis in accordance with Regulatory Guide 1.70, the methods of this section are also applicable to other dynamic loading aspects, except for the range of frequencies considered. The cutoff frequency for dynamic analysis is 33 Hz for seismic loads and 80 ZHz for suppression pool dynamic loads. The definition of rigid system used in this section is applicable to seismic design only.

The safe shutdown earthquake (SSE) is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

(1) the integrity of the reactor coolant pressure boundary;

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition; and

(3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

The operating basis earthquake (OBE) is that earthquake which, considering the regional and local geology, seismology, and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produce vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. During the OBE loading condition, the safety-related systems are designed to be capable of continued safe operation. Therefore, for this loading condition, safety-related structures, and equipment are required to operate within design limits.

The seismic design for the SSE is intended to provide a margin in design that assures capabili- ty to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the required systems and components do not lose their capability to per- form their safety-related function. This is referred to as the no-loss-of-function criterion and the loading condition as the SSE loading condition.

Not all safety-related components have the same functional requirements. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, elastic behavior of this structure under the SSE loading condition is ensured. On the other hand, there are certain structures, components, and systems that can suffer permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain contents and allow fluid flow.

Table 3.2-1 identifies the equipment in various systems as Seismic Category I or non-Seismic Category I.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design earthquake loading is specified in terms of a set of idealized, smooth curves called the design response spectra in accordance with Regulatory Guide 1.60.

Figure 3.7-1 shows the standard ABWR design values of the horizontal SSE spectra applied at the ground surface in the free field for damping ratios of 2.0, 5.0, 7.0 and 10.0% of critical

values of the vertical SSE spectra applied at the ground surface in the free field for damping ratios of 2.0, 5.0, 7.0, and 10.0% of critical damping where the maximum vertical ground acceleration is 0.30 g at 33Hz, same as the maximum horizontal ground acceleration.

The design values of the OBE response spectra are one-half* of the spectra shown in Figures 3.7-1 and 3.7-2. These spectra are shown in Figures 3.7-3 through 3.7-20.

The design spectra are constructed in accordance with Regulatory Guide 1.60. The normalization factors for the maximum values in two horizontal directions are 1.0 and 1.0 as applied to Figure 3.7-1. For vertical direction, the normalization factor is 1.0 as applied to Figure 3.7-2.

3.7.1.2 Design Time History

The design time histories are synthetic acceleration time histories generated to match the design response spectra defined in Subsection 3.7.1.1.

The design time histories considered in GESSAR (Reference 1) are used. They are developed based on the method proposed by Vanmarcke and Cornell (Reference 2) because of its intrinsic capability of imposing statistical independence among the synthesized acceleration time history components. The earthquake acceleration time history components are identified as H1, H2, and V. The H1 and H2 are the two horizontal components mutually perpendicular to each other. Both H1 and H2 are based on the design horizontal ground spectra shown in Figure 3.7-1. The V is the vertical component and it is based on the design vertical ground spectra shown in Figure 3.7-2.

The magnitude of the SSE design time history is equal to twice the magnitude of the design OBE time history. The OBE time histories and response spectra are used for dynamic analysis and evaluation of the structural Seismic System; the OBE results are doubled for evaluating the structural adequacy for SSE. For development of floor response spectra for Seismic Subsystem analysis and evaluation, see Subsection 3.7.2.5.

The response spectra produced from the OBE design time histories are shown in Figures 3.7-3 through 3.7-20 along with the design OBE response spectra. The closeness of the two spectra in all cases indicates that the synthetic time histories are acceptable.

The response spectra from the synthetic time histories for the damping values of 1, 2, 3 and 4 percent conform to the requirement for an enveloping procedure provided in Item II.1.b of Section 3.7.1 of NUREG-0800 (Standard Review Plan, SRP). However, the response spectra for the higher damping values of 7 and 10 percent show that there are some deviations from the SRP requirement. This deviation is considered inconsequential, because (1) generating an artificial time history whose response spectra would envelop design spectra for five different damping values would result in very conservative time histories for use as design basis input, and (2) the response spectra from the synthetic time histories do envelop the design spectra for the lower damping values. This is very important because the loads due to SSE on structures should use 7 percent damping for concrete components, but are obtained by ratioing up the response from the OBE analysis involving the lower damping. The OBE analysis uses only the lower damping values (up to 4%), which are consistent with the SRP requirements (See Subsection 3.7.1.3).

* *The OBE given in Chapter 2 is one-third of the SSE, i.e., 0.10 g, for the ABWR Standard Nuclear Island design. However, as discussed in Chapter 2, a more conservative value of one-half of the SSE, i.e., 0.15 g, was employed to evaluate the structural and component response.*

3.7.3.8.2.2 Effect of Differential Building Movements

The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor-point displacement are used in a static analysis to determine the additional stresses due to relative anchor-point displacements. Further details are given in Subsection 3.7.3.8.1.8.

3.7.3.9 Multiple Supported Equipment Components With Distinct Inputs

The procedure and criteria for analysis are described in Subsections 3.7.2.1.3 and 3.7.3.3.1.3.

3.7.3.10 Use of Constant Vertical Static Factors

All Seismic Category I subsystems and components are subjected to a vertical dynamic analysis with the vertical floor spectra or time histories defining the input. A static analysis is performed in lieu of dynamic analysis if the peak value of the applicable floor spectra times a factor of 1.5 is used in the analysis. A factor of 1.0 instead of 1.5 can be used if the equipment is simple enough such that it behaves essentially as a single degree of freedom system. If the fundamental frequency of a compoent in the vertical direction is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed statically using the zero-pe-sponse spectrum.

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3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are included for Seismic Category I subsystems similar to that for the piping systems discussed in Subsection 3.7.3.3.1.2.

3.7.3.12 Buried Seismic Category I Piping and Tunnels

For buried Category I buried piping systems and tunnels the following items are considered in the analysis:

(1) The inertial effects due to an earthquake upon buried systems and tunnels will be adequately accounted for in the analysis. In case of buried systems sufficiently flexible relative to the surrounding or underlying soil, it is assumed that the systems will follow essentially the displacements and deformations that the soil would have if the systems were absent. When applicable, procedures, which take into account the phenomena of wave travel and wave reflection in compacting soil displacements from the ground displacements, are employed.

(2) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., are considered. When applicable, procedures utilizing the principles of the theory of structures on elastic foundations are used.

(3) When applicable, the effects due to local soil settlements, soil arching, etc., are also considered in the analysis.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

In certain instances, non-Seismic Category I piping may be connected to Seismic Category I piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at Seismic Category I valves which may or may not be physically anchored. Since a dynamic analysis must be modeled from pipe anchor point to anchor point, two options exist:

(1) specify and design a structural anchor at the Seismic Category I valve and analyze the Seismic Category I subsystem; or, if impractical to design an anchor,

(2) analyze the subsystem from the anchor point in the Seismic Category I subsystem through the valve to either the first anchor point in the non-Seismic Category I subsystem; or to sufficient distance in the non-Seismic Category I Subsystem so as not to significantly degrade the accuracy of analysis of the Seismic Category I piping.

Where small, non-Seismic category piping is directly attached to Seismic Category I piping, its effect on the Seismic Category I piping is accounted for by lumping a portion of its mass with the Seismic Category I piping at the point of attachment.

Furthermore, non-Seismic Category I piping (particularly high energy piping as defined in Section 3.6) is designed to withstand the SSE to avoid jeopardizing adjacent Seismic Category I piping if it is not feasible or practical to isolate these two piping systems.

3.7.3.14 Seismic Analysis for Reactor Internals

The modeling of RPV internals is discussed in Subsection 3.7.2.3.2. The damping values are given in Table 3.7-1. The seismic model of the RPV and internal is shown in Figure 3.7-32.

3.7.3.15 Analysis Procedures for Damping

Analysis procedures for damping are discussed in Subsection 3.7.2.15.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with NRC Regulatory Guide 1.12

The seismic instrumentation program is consistent with Regulatory Guide 1.12.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment are used to measure plant response to earthquake motion:

(1) three triaxial time-history accelerographs (THA);

(2) three peak-recording accelerographs (PRA);

(3) two triaxial seismic triggers;

(4) one seismic switch (SS);

(5) four response spectrum recorders;

(6) recording and playback equipment; and

(7) annunciators.

The location of seismic instrumentation is outlined in Table 3.7-7.

3.7.4.2.1 Time-History Accelerographs

Time-history accelerographs produce a record of the time-varying acceleration at the sensor location. This data is used directly for analysis and comparison with reference information and may be, by calculational methods, converted to response spectra form for spectra comparisons with design parameters.

Each triaxial acceleration sensor unit contains three accelerometers mounted in an orthogonal array (two horizontal and one vertical). All acceleration units have their principal axes oriented identically. The mounted units are oriented so that their axes are aligned with the building major axes used in development of the mathematical models for seismic analysis.

One THA is located on the reactor building (RB) foundation mat, El (-) 13.2 M, at the base of an RB clean zone for the purpose of measuring the input vibratory motion of the foundation mat. A second THA is located in an RB clean zone at El (+) 26.7 M on the same azimuth as the foundation mat THA. They provide date on the frequency, amplitude, and phase relationship of the seismic response of the reactor building structure. A third THA is located in the free field at the finished grade approximately 160 M from any station structures with axes oriented in the same direction as the reactor building accelerometers.

Two seismic triggers, connected to form redundant triggering, are provided to start the THA recording system. They are located in the free field at the finished grade 160 M from the reactor building. The trigger unit consists of orthogonally mounted acceleration sensors that actuate relays whenever a threshold acceleration is exceeded for any of the three axes. The trigger is engineered to discriminate against false starts from other operating inputs such as traffic, elevators, people, and rotating equipment.

SECTION 3.9

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analyzed for the faulted loading conditions. The ECCS and SLC pumps are active ASME Class 2 components. The allowable stresses for active pumps are provided in a footnote to Table 3.9-2.

The reactor coolant pressure boundary components of the reactor recirculation system (RRS) pump motor assembly, and recirculation motor cooling (RMC) subsystem heat exchanger are ASME Class 1 and Class 3, respectively, and are analyzed for the faulted loading conditions. All equipment stresses are within the elastic limits.

3.9.1.4.7 Fuel Storage and Refueling Equipment

Storage, refueling, and servicing equipment which is important to safety is classified as essential components per the requirements of 10CFR50 Appendix A. This equipment and other equipment which in case of a failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite-element analysis to generate loadings. This analysis utilizes appropriate floor response spectra and combines loads at frequencies up to 33 Hz for seismic loads and up to 60 Hz for other dynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC, allowables.

3.9.1.4.8 Fuel Assembly (Including Channel)

GE BWR fuel assembly (including channel) design bases, and analytical and evaluation methods including those applicable to the faulted conditions are the same as those contained in References 1 and 2.

3.9.1.4.9 ASME Class 2 and 3 Vessels

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 vessels. The equivalent allowable stresses using elastic techniques are obtained from NC/ND-3300 and NC-3200 of the ASME Code Section III. These allowables are above elastic limits.

3.9.1.4.10 ASME Class 2 and 3 Pumps

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 pumps. The equivalent allowable stresses for nonactive pumps using elastic techniques are obtained from NC/ND-3400 of the ASME Code Section III. These allowables are above elastic limits. The allowables for active pumps are provided in a footnote to Table 3.9-2.

3.9.1.4.11 ASME Class 2 and 3 Valves

Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions for Class 2, and 3 valves. The equivalent allowable stresses for nonactive valves using elastic techniques are obtained from NC/ND-3500 of ASME Code, Section III. These allowables are above elastic limits. The allowables for active valves are provided in a footnote to Table 3.9-2.

3.9.1.4.12 ASME Class 1, 2 and 3 Piping

Elastic analysis methods are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from Appendix F (for Class 1) and NC/ND-3600 (for Class 2 and 3 piping) of the ASME Code Section III. These allowables are above elastic limits. The allowables for functional capability of the essential piping are provided in a footnote to Table 3.9-2.

3.9.1.5 Inelastic Analysis Methods

Inelastic analysis is only applied to ABWR components to demonstrate the acceptability of three types of postulated events. Each event is an extermly low-probability occurence and the equipment affected by these events would not be reused. These three events are:

(1) Postulated gross piping failure.

(2) Postulated blowout of a reactor internal recirculation (RIP) motor casing due to a weld failure.

(3) Postulated blowout of a control rod drive (CRD) housing due to a weld failure.

The loading combinations and design criteria for pipe whip restraints utilized to mitigate the effects of postulated piping failures are provided in Subsection 3.6.2.3.3.

In the case of the RIP motor casing failure event, there are specific restraints applied to mitigate the effects of the failure. The mitigation arrangement consists of lugs on the RPV bottom head to which are attached two long rods for each RIP. The lower end of each rod engages two lugs on the RIP motor/cover. The use of inelastic analysis methods is limited to the middle slender body of the rod itself. The attachment lugs, bolts and clevises are shown to be adequate by elastic analysis. The selection of stainless steel for the rod is based on its high ductility assumed for energy absorption during inelastic deformation.

The mitigation for the CRD housing attachment weld failure is by somewhat different means than are those of the RIP in that the components with regular functions also function to mitigate the weld failure effect. The components are specifically:

(1) Core support plate

(2) Control rod guide tube

(3) Control rod drive housing

(4) Control rod drive outer tube

(5) Bayonet fingers

Only the cylindrical bodies of the control rod guide tube, control rod drive housing and control rod drive outer tube are analyzed for energy absorption by inelastic deformation.

Inelastic analysis for there latter two events together with the criteria used for evaluation are consistent with the procedures described in Subsection 3.6.2.3.3 for the different components of a pipe whip restraint. Figure 3.9-6 shows the stress-strain curve used for the blowout restraints.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The overall test program is divided into two phases; the preoperational test phase and the initial startup test phase. Piping vibration, thermal expansion and dynamic effects testing will be performed during both of these phases as described in Chapter 14. Subsections 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11 relate the specific role of this testing to the overall test program. Discussed below are the general requirements for this testing. It

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should be noted that because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow. However, the more specific requirements for the design and testing of the piping support system are described in Subsection 3.9.3.4.1.

3.9.2.1.1 Vibration and Dynamic Effects Testing

The purpose of these tests is to confirm that the piping, components, restraints and supports of specified high- and moderate-energy systems have been designed to withstand the dynamic effects of steady state flow-induced vibration and anticipated operational transient conditions. The general requirements for vibration and dynamic effects testing of piping systems are specified in Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors". More specific vibration testing requirements are defined in ANSI/ASME OM3, "Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems".Preparation of detailed test specifications will be in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below. 210.31

3.9.2.1.1.1 Measurement Techniques

There are essentially three methods available for determining the acceptability of steady state and transient vibration for the affected systems. These three measurement techniques are visual observation, local measurements, or remotely monitored/recorded measurements. The technique used in each case will depend on such factors as the safety significance of the particular system, the expected mode and/or magnitude of the vibration, the assessability of the system during designated testing conditions, or the need for a time history recording of the vibratory behavior. Typically, the systems where vibration has the greatest safety implication will be subject to more rigorous testing and precise instrumentation requirements and, therefore, will require remote monitoring techniques. Local measurement techniques, such as the use of a hand-held vibrometer, are more appropriate in cases where it is expected that the vibration will be less complex and of lessor magnitude. Many systems that are assessable during the preoperational test phase and that do not show significant intersystem interactions will fall into this category. Visual observations are utilized where vibration is expected to be minimal and the need for a time history record of transient behavior is not anticipated. However, unexpected visual observations or local indications may require that a more sophisticated technique be used. Also, the issue of assessability should be considered. Application of these measurement techniques is detailed in the appropriate testing specification consistent with the guidelines contained in ANSI/ASME OM3. 210.31

3.9.2.1.1.2 Monitoring Requirements

As described in Subsection 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11 all safety-related piping systems will be subjected to steady state and transient vibration measurements. The scope of such testing shall include safety-related instrumentation piping and attached small-bore piping (branch piping). Special attention should be given to piping attached to pumps, compressors, and other rotating or reciprocating equipment. Monitoring location selection considerations should include the proximity of isolation valves, pressure or flow control valves, flow orifices, distribution headers, pumps and other elements where shock or high turbulence may be of concern. Location and orientation of instrumentation and/or measurements will be detailed in the appropriate test specification. Monitored data should include actual deflections and frequencies as well as related system operating conditions. Time duration of data recording should be sufficient to indicate whether the vibration is continuous or transient. Steady state monitoring should be performed at critical conditions such as minimum or maximum flow, or abnormal combinations or configurations of system pumps or valves. Transient monitoring should include anticipated system and total plant operational transients where critical piping or components are expected to show 210.32 210.31

significant response. Steady state conditions and transient events to be monitored will be detailed in the appropriate testing specification consistent with OM3 guidelines.

3.9.2.1.1.3 Test Evaluation and Acceptance Criteria

The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-36000) limits. Acceptable limits are determined after the completion of piping systems stress analysis and are provided in the piping 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. For steady state and transient vibration the pertinent acceptance criteria are usually expressed in terms of maximum allowable displacement/deflection. Visual observation should only be used to confirm the absence of significant levels of vibration and not to determine acceptability of any potentially excessive vibration. Therefore, in some cases other measurement techniques will be required with appropriate quantitative acceptance criteria. 210.31

There are typically two levels of acceptance criteria for allowable vibration displacements/deflections. Level 1 criteria are bounding type criteria associated with safety limits while Level 2 criteria are stricter criteria associated with system or component expectations. For steady state vibration the Level 1 criteria are based on the endurance limit (10,000 psi) to assure no failure from fatigue over the life of the plant. The corresponding Level 2 criteria are based on one half the endurance limit (5,000 psi). For transient vibration the Level 1 criteria are based on either the ASME-III code upset primary stress limit or the applicable snubber load capacity. Level 2 criteria are based on a given tolerance about the expected deflection value.

3.9.2.1.1.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold or termination appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the piping and suspension system should be made in an attempt to identify potential obstructions or improperly operating suspension components. Hangers and snubbers should be positioned such that they can accommodate the expected deflections without bottoming out or extending fully. All signs of damage to piping supports or anchors shall be investigated.

Instrumentation indicating criteria failure shall be checked for proper operation and calibration including comparison with other instrumentation located in the proximity of the excessive vibration. The assumptions used in the calculations that generated the applicable limits should be verified against actual conditions and discrepancies noted should be accounted for in the criteria limits. This may require a reanalysis at actual system conditions. 210.31

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations, then physical corrective actions may be required. This might include identification and reduction or elimination of offending forcing functions, detuning of resonant piping spans by appropriate modifications, addition of bracing, or changes in operating procedures to avoid troublesome conditions. Any such modifications will require retest to verify vibrations have been sufficiently reduced.

3.9.2.1.2 Thermal Expansion Testing

A thermal expansion preoperational and startup testing program performed through the use of visual observation and remote sensors has been established to verify that normal unrestrained thermal movement occurs in specified safety-related high- and moderate-energy piping systems. The purpose of this program is to ensure the following:

(1) the piping system during system heatup and cooldown is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions;

(2) the piping system does shakedown after a few thermal expansion cycles;

(3) the piping system is working in a manner consistent with the assumption of the stress analysis;

(4) there is adequate agreement between calculated values and measured values of displacements; and

(5) there is consistency and repeatability in thermal displacements during heatup and cooldown of the systems.

The general requirements for thermal expansion testing of piping systems are specified in Regulatory Guide 1.68, "Preoperational and Initial Startup Testing Programs for Water-Cooled Power Reactors." More specific requirements are defined in ANSI/ASME OM7 "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems." Detailed test specifications will be prepared in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below. 210.31

3.9.2.1.2.1 Measurement Techniques

Verification of acceptable thermal expansion of specified piping systems can be accomplished by several methods. One method is to physically walkdown the piping system and verify by visual observation that free thermal movement is unrestrained. This might include verification that piping supports such as snubbers and spring hangers are not fully extended or bottomed out and that the piping (including branch lines and instrument lines) and its insulation is not in hard contact with other piping or support structures. Another method would involve local measurements, using a hand held scale or ruler, against a fixed reference or by recording the position of a snubber or spring can. A more precise method would be using permanent or temporary instrumentation that directly measures displacement, such as a lanyard potentiometer, that can be monitored via a remote indicator or recording device. The technique to be used will depend on such factors as the amount of movement predicted and the assessability of the piping.

Measurement of piping temperature is also of importance when evaluating thermal expansion. This may be accomplished either indirectly via the temperature of the process fluid or by direct measurement of the piping wall temperature and such measurements may be obtained either locally or remotely. The choice of technique used shall depend on such considerations as the accuracy required and the assessability of the piping.

3.9.2.1.2.2. Monitoring Requirements 210.31

As described in Subsections 14.2.12.1.51 and 14.2.12.2.10 all safety-related piping shall be included in the thermal expansion testing program. Thermal expansion of specified piping systems should be measured at both the cold and hot extremes of their expected operating conditions. Physical walkdowns and recording of hanger and snubber positions should also be conducted where possible considering assessability and local environmental and radiological conditions in the hot and cold states. Displacements and appropriate piping/process temperatures shall be recorded for those systems and conditions specified. Sufficient time shall have passed before taking such measurements to ensure the piping system is at a steady state condition. In selecting locations for monitoring piping response, consideration shall be given to the maximum responses predicted by the piping analysis. Specific consideration should also be given to the first run of pipe attached to component nozzles and pipe adjacent to structures requiring a controlled gap.

3.9.2.1.2.3 Test Evaluation and Acceptance Criteria

To ensure test data integrity and test safety, criteria have been established to fa-

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3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment (Including Other RBV Induced Loads)

This subsection describes the criteria for dynamic qualification of safety-related mechanical equipment and associated supports, and also describes the qualification testing and/or analysis applicable to the major components on a component by component basis. Seismic and other events that may induce reactor building vibration (RBV)-(see Appendix 3B) are considered. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit (e.g., ECCS pumps). These modules are generally discussed in this subsection and Subsection 3.9.3.2 rather than providing discussion of the separate electrical parts in Section 3.10. Electrical supporting equipment such as control consoles, cabinets, and panels are discussed in Section 3.10.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety function during and after the application of a dynamic load is demonstrated by tests and/or analysis. The analysis is performed in accordance with Section 3.7. 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 equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static bend test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or static bend testing is also used to show there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for other RBV loads*. If a natural frequency lower than 33 Hz in the case of seismic loads and 60 Hz in

* *The 60 Hz frequency cutoff for dynamic analysis of suppression pool dynamic loads is the minimum requirement based on a generic Reference 8, using the missing strain energy method, performed for representative BWR equipment under high-frequency input loadings.*

the case of other RBV induced loads is discovered, dynamic tests and/or mathematical analyses may be used to verify operability and structural integrity at the required dynamic input conditions.

When the equipment is qualified by dynamic test, the response spectrum or time history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic load conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) over the frequency range of interest. Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during the dynamic loading condition.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a static bend test is used to determine spring constant and operational capability at maximum equivalent dynamic load conditions.

3.9.2.2.1.1 Random Vibration Input

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input such as sine beats can be use provided one of the following conditions are met:

(1) the characteristics of the required input motion is dominated by one frequency;

(2) the anticipated response of the equipment is adequately represented by one mode; or

(3) the input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra will envelop the corresponding response spectra of the individual modes.

3.9.2.2.1.2 Application of Input Modes

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2.1.3 Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing

Equipment testing is conducted on prototypes of the equipment to be installed in the plant.

3.9.2.2.2 Qualification of Safety-Related Mechanical Equipment

The following subsections discuss the testing or analytical qualification of the safety-related major mechanical equipment, and other ASME III equipment, including equipment supports.

3.9.2.2.2.1 CRD and CRD Housing

The qualification of the CRD housing (with enclosed CRD) is done analytically, and the stress results of their analysis establish the structural integrity of these components. Preliminary dynamic tests are conducted to verify the operability of the control rod

pletion of preoperational testing, the reactor vessel head and the shroud head are removed, the vessel is drained, and major components are inspected on a selected basis. The inspections cover the shroud, shroud head, core support structures, recirculation internal pumps, the peripheral control rod drive, and incore guide tubes. Access is provided to the reactor lower plenum for these inspections.

The analysis, design and/or equipment that are to be utilized in a facility will comply with Regulatory Guide 1.20 as explained below.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, Quality Standards and Record, Appendix A to 10CFR50 and Section 50.34, Contents of Applications; Technical Information, of 10CFR50. This Regulatory Guide is applicable to the core support structures and other reactor internals.

Vibration testing of reactor internals is performed on all GE-BWR plants. At the time of original issue of Regulatory Guide 1.20, test programs for compliance were instituted for the then designed reactors. The first ABWR plant is considered a prototype and is instrumented and subjected to preoperation and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. Subsequent plants which have internals similar to those of the prototypes are also tested in compliance with the requirements of Regulatory Guide 1.20. GE is committed to confirm satisfactory vibration performance of internals in these plants through preoperational flow testing followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants together with satisfactory operating experience in all BWR plants have established the adequacy of reactor internal designs. GE continues these test programs for the generic plants to verify structural integrity and to establish the margin of safety.

See Subsection 3.9.7.1 for interface requirements of the reactor internals vibration testing program.

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The faulted events that are evaluated are defined in Subsection 3.9.5.2.1. The loads that occur as a result of these events and the analysis performed to determine the response of the reactor internals are as follows:

(1) **Reactor Internal Pressures** - The reactor internal pressure differentials (Figure 3.9-1a) due to assumed break of main steam or feedwater line are determined by analysis as described in Subsection 3.9.5.2.2. In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces during an accident, a comparison is made of 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. Besides the real masses of the RPV and core support structures, account is made for the water inside the RPV.

(2) **External Pressure and Forces on the Reactor Vessel**-An assumed break of the main steam line, the feedwater line or the RHR line at the reactor vessel nozzle results in jet reaction and impingement forces on the vessel and asymmetrical pressurization of the annulus between the reactor vessel and the shield wall. These time-varying pressures are applied to the dynamic model of the reactor vessel system. Except for the nature and locations of the forcing functions, the dynamic model and the dynamic analysis method are identical to those for seismic analysis as described below. The resulting loads on the reactor internals, defined as LOCA loads, are considered as shown in Table 3.9.2.

(3) **Safety/Relief Valve Loads (SRV Loads)**-The discharge of the SRVs result in reactor building vibration (RBV) due to suppression pool dynamics as described in Appendix 3B. The response of the reactor

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internals to the RBV is also determined with dynamic model and dynamic analysis method described below for seismic analysis.

(4) **LOCA Loads**-The Assumed LOCA also results in RBV due to suppression pool dynamics as described in Appendix 3B and the response of the reactor internals are again determined with the dynamic model and dynamic analysis method used for seismic analysis. Various types of LOCA loads are identified on Table 3.9-2.

(5) **Seismic Loads**-The theory, methods, and computer codes used for dynamic analysis of the reactor vessel, internals, attached piping and adjoining structures are described in Section 3.7 and Subsection 3.9.1.2. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and mode shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the resonse-spectrum method. The load on the reactor internals due to faulted event SSE are obtained from this analysis.

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The above loads are considered in combination as defined in Table 3.9-2. The SRV. LOCA (SBL, IBL or LBL) and SSE loads as defined in Table 3.9-2 are all assumed to act in the same direction. The peak colinear responses of the reactor internals to each of these loads are added by the square root of the sum of the squares (SRSS) method. The resultant stresses in the reactor internal structures are directly added with stress resulting from the static and steady state loads in the faulted load combination, including the stress due to peak reactor internal pressure differential during the LOCA. The reactor internals satisfy the stress deformation and fatigue limits as defined in Subsection 3.9.5.3.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

Prior to initiation of the instrumented vibration measurement program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test will be analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and loss of coolant accident (LOCA) analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combination associated with normal operation, postulated accidents, and specified seismic and other reactor building vibration (RBV) events for the design of safety-related ASME Code components (except containment components which are discussed in Section 3.8).

This section discusses the ASME Class 1, 2, and 3 equipment and associated pressure retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME Class 1, 2, and 3 equipment are covered in Subsection 3.9.1.1. Seismic-related loads and dynamic analyses are discussed in Section 3.7. The suppression pool-related RBV loads are described in Appendix 3B. Table 3.9-2 presents the combinations of dynamic events to be considered for the design and analysis of all ABWR ASME Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of each specific equipment are derived from Table 3.9-2 and are contained in the design specifications and/or design reports of the respective equipment. (See Subsection 3.9.7.3 for interface requirements)

Table 3.9-2 also presents the evaluation models and criteria. The predicted loads or stresses and the design or allowable values for the most critical areas of each component are compared in accordance with the applicable code criteria or other limiting criteria. The calculated results meet the limits.

The design life for the ABWR Standard Plant is 60 years. A 60 year design life is a requirement for all major plant components with reasonable expection of meeting this design life. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable, design life not withstanding. The design life requirement allows for refurbihment and repair, as appropriate, to assure the design life of the overall plant is achieved. In effect, essentially all piping systems, components and equipment are designed for a 60 year design life. Many of these components are classified as ASME Class 2 or 3 or Quality Group D. Applicants referencing the ABWR design will identify these ASME Class 2, 3 amd Quality Group D components and provide the analyses required by the ASME Code, Subsection NB. These analysis will include the appropriate operating vibration loads and for the effects of mixing hot and cold fluids.

3.9.3.1.1 Plant Conditions

All events that the plant will or might credibly experience during a reactor year are evaluated to establish design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence as discussed in Subsection 3.9.3.1.1.5) and correlated to service levels for design limits defined in the ASME Boiler and Pressure Vessel Code Section III as shown in Tables 3.9-1 and 3.9-2.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

3.9.3.1.1.2 Upset Condition

An upset condition is any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include system operational transients (SOT) which result from any single operator error or control malfunction, from a fault in a system component requiring its isolation from the system, from a loss of load or power, or from an operating basis earthquake. Hot standby with the main condenser isolated is an upset condition.

3.9.3.4 Component Supports

The design of bolts for component supports is specified in the ASME Code Section III, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

Moreover, on equipment which is to be, or may be, mounted on a concrete support, sufficient holes for anchor bolts are provided to limit the anchor bolt stress to less than 10,000 psi on the nominal bolt area in shear or tension.

Concrete anchor bolts which are used for pipe support base plates will be designed to the applicable factors of safety which are defined in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 1 dated June 21,1979.

3.9.3.4.1 Piping

Supports and their attachments for essential ASME Code Section III, Class 1,2, and 3 piping are designed in accordance with Subsection NF* up to the interface of the building structure. The building structure component supports are designed in accordance with the AISC specification for the Design, Fabrication, and Erection of Structural Steel for buildings. The loading combinations for the various operating conditions correspond to those used for design of the supported pipe. The component loading combinations are discussed in Subsection 3.9.3.1. The stress limits are per ASME III, Subsection NF and Appendix F. Supports are generally designed either by load rating method per paragraph NF-3260 or by the stress limits for linear supports per paragraph NF-3231. The critical buckling loads for the Class 1 piping supports subjected to faulted loads that are more severe than normal, upset and emergency loads, are determined by using the methods discussed in Appendices F and XVII of the Code. To avoid buckling in the piping supports, the allowable loads are limited to two thirds of the determined critical buckling loads.

The design of all supports for non-nuclear piping satisfies the requirements of ANSI B31.1, Paragraphs 120 and 121.

For the major active valves identified in Subsection 3.9.3.2.4, the valve operators are not used as attachment points for piping supports.

The design criteria and dynamic testing requirements for the ASME III piping supports are as follows:

(1) Piping Supports - All piping supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe of equipment after they have been installed. All piping supports are designed in accordance with the rules of Subsection NF of the ASME Code up to the building structure interface as defined in the project design specifications.

(2) Spring Hangers - The operating load on spring hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the operating load at both their hot and cold load settings. Spring hangers provide a specified down travel and up travel in excess of the specified thermal movement.

*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members os ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83 or "Design of Weld Structures" by Omar W. Blodgett, etc.

(3) Snubbers - The operating loads on snubbers are the loads caused by dynamic events (e.g., seismic, RBV due to LOCA and SRV discharge, discharge through a relief valve line or valve closure) during various operating conditions. Snubbers restrain piping against response to the vibratory excitation and to the associated differential movement of the piping system support anchor points. The criteria for locating snubbers and ensuring adequate load capacity, the structural and mechanical performance parameters used for snubbers and the installation and inspection consider- ations for the snubbers are as follows:

(a) Required Load Capacity and Snubber Location

The entire piping system including valves and support system between anchor points is mathematically modeled for complete piping structural analysis. In the dynamic analysis, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each piping components and the forces acting on the snubbers due to all dynamic loading and operating conditions defined in the piping design specification. The forces on snubbers are operating loads for various operating conditions. Those loads are assumed no to exceed the snubber design load capacity for various operating conditions, i.e., design, normal, upset, emergency and faulted.

Snubbers are generally used in situations where dynamic support is required because thermal growth of the piping prohibits the use of rigid supports. The snubber locations and support directions are first decided by estimation so that the stresses in the piping system will have acceptable values. The snubber locations and support directions are refined by performing the dynamic analysis of the piping and support system as described above in order that the piping stresses and support loads meet the Code requirements.

The pipe support design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

(b) Inspection, Testing, Repair and/or Replacement of Snubbers

The pipe support design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period of inspection.

The pipe support design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

The spring constant achieved by the snubber supplier for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and support direction become confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are reconciled.

(c) Snubber Design and Testing

To assure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed by the design specification:

(i) The snubbers are required by the pipe support design specification to be designed in accordance with all of the rules and regulations of the ASME Code Section III, Subsection NF. This design requirement includes analysis for the normal, upset, emergency, and faulted loads. These calculated loads are then compared against the allowable loads to make sure that the stresses are below the code allowable limit.

(ii) The snubbers are tested to insure that they can perform as required during the seismic and other RBV events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The following test requirements are included:

- Snubbers are subjected to force or displacement versus time loading at frequencies within the range of

significant modes of the piping system;

- o Displacements are measured to determine the performance characteristics specified;
- o Tests are conducted at various temperatures to ensure operability over the specified range;
- o Peak test loads in both tension and compression are required to be equal to or higher than the rated load requirements; and
- o The snubbers are tested for various abnormal environmental conditions. Upon completion of the 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.

(d) Snubber Installation Requirements

An installation instruction manual is required by the pipe support design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

(e) Snubber Pre-service Examination

The pre-service examination plan of all snubbers covered by the Chapter 16 technical specifications will be prepared. This examination will be made after snubber installation but not more than 6 months prior to initial system pre-operational testing. The pre-service examination will verify the following:

(i) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

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

(iii) Snubbers are not seized, frozen or jammed.

(iv) Adequate swing clearance is provided to allow snubber movements.

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

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

If the period between the initial pre-service examination and initial system pre-operational tests exceeds 6 months because of unexpected situations, reexamination of Items 1, 4, and 5 will be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements will be repaired or replaced and re-examined in accordance with the above criteria.

(4) Struts - The design load on struts includes those loads caused by dead weight, thermal expansion, seismic forces (i.e., OBE and SSE), other RBV loads,

system anchor displacements, and reaction forces caused by relief valve discharge or valve closure, etc.

Struts are designed in accordance with ASME Code Section III, Subsection NF-3000 to be capable of carrying the design loads for various operating conditions. As in case of snubbers, the forces on struts are obtained from an analysis, which are assured not to exceed the design loads for various operating conditions.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The ABWR RPV support skirt is designed as an ASME Code Class 1 component per the requirements of ASME Code Section III, Subsection NF*. The loading conditions and stress criteria are given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions. The stress level margins assure the adequacy of the RPV support skirt. An analysis for buckling shows that the support skirt complies with Subparagraph F-1332.5 of ASME III, Appendix F, and the loads do not exceed two thirds of the critical buckling strength of the skirt. The permissible skirt loads at any elevation, when simultaneously applied, are limited by the following interaction equation:

$$(P/P_{crit}) + (q/q_{crit}) + (\tau/\tau_{crit}) < (1/S.F.)$$

where:

- q = longitudinal load
- P = external pressure
- τ = transverse shear stress
- S.F. = safety factor
  - = 3.0 for design, testing, service levels A & B
  - = 2.0 for Service Level C
  - = 1.5 for Service Level D.

3.9.3.4.3 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a Safety Class 1 linear type component support in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NF. The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads due to effects such as earthquake, pipe rupture and RBV. The design loading conditions, and stress criteria are given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions.

3.9.3.4.4 Floor-Mounted Major Equipment (Pumps, Heat Exchangers, and RCIC Turbine)

Since the major active valves are supported by piping and not tied to building structures, valve "supports" do not exist (See Subsection 3.9.3.4.1).

The HPCF, RHR, RCIC, SLC, FPCCU, SPCU, and RWCU pumps; RMC, RHR, RWCU, and FPCCU heat exchangers; and RCIC turbine are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the load stresses in the critical support areas are within ASME Code allowables.

Seismic Category I active pump supports are qualified for dynamic (seismic and other RBV) loads by testing when the pump supports

*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83 or "Design of Weld Structures" by Omar W. Blodgett, etc.

together with the pump meet the following test conditions:

(1) simulate actual mounting conditions;

(2) simulate all static and dynamic loadings on the pump;

(3) monitor pump operability during testing;

(4) the normal operation of the pump during and after the test indicates that the supports are adequate (any deflection or deformation of the pump supports which precludes the operability of the pump is not accepted); and

(5) supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Dynamic qualification of component supports by analysis is generally accomplished as follows:

(1) Stresses at all support elements and parts such as pump holddown and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in the ASME Code Section III, Subsection NF.

(2) For normal and upset conditions, the deflections and deformations of the supports are assured to be within the elastic limits, and to not exceed the values permitted by the designer based on design verification tests. This ensures the operability of the pump.

(3) For emergency and faulted plant conditions, the deformations do not exceed the values permitted by the designer to ensure the operability of the pump. Elastic/plastic analysis are performed if the deflections are above the elastic limits.

3.9.3.5 Other ASME III Component Supports

The ASME III component supports and their attachments (other than those discussed in preceding subsection) are designed in accordance with Subsection NF of the ASME Code Section III* up to the interface with the building structure. The building structure component supports are designed in accordance with the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings. The loading combinations for the various operating conditions correspond to those used to design the supported component. The component loading combinations are discussed in Subsection 3.9.3.1. Active component supports are discussed in Subsection 3.9.3.2. The stress limits are per ASME III, Subsection NF and Appendix F. The supports are evaluated for buckling in accordance with ASME III.

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*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83 or "Design of Weld Structures" by Omar W. Blodgett, etc.

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3.9.4 Control Rod Drive System (CRDS)

A control rod drive system CRDS)in an ABWR plant is equipped with an electro-hydraulic fine motion control rod drive (FMCRD) system, which includes the control rod drive (CRD) mechanism, the hydraulic control unit (HCU), the condensate supply system, and power for FMCRD motor, and extends inside RPV to the coupling interface with the control rod blades.

3.9.4.1 Descriptive Information on CRDS

Descriptive information on the CRDs as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2 Applicable CRDS Design Specification

CRDS is designed to meet the functional design criteria outlined in Section 4.6 and con-

sists of the following:

(1) fine motion control rod drive;

(2) hydraulic control unit;

(3) hydraulic power supply (pumps);

(4) electric power supply (for FMCRD motors)

(5) interconnecting piping;

(6) flow and pressure and isolation valves; and

(7) instrumentation and electrical controls.

Those components of the CRDS forming part of the primary pressure boundary are designed according to ASME Code Section III, Class 1 requirements.

The quality group classification of the components of the CRDS is outlined in Table 3.2-1 and they are designed to the codes and standards, per Table 3.2-2, in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRDS components are discussed in the following locations: transients in Subsection 3.9.1.1, faulted conditions in Subsection 3.9.1.4, seismic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The ASME III Code components of the CRDS have been evaluated analytically and the design loading conditions, and stress criteria are as given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses. For the non-Code components, the ASME III Code requirements are used as guidelines and experimental testing is used to determine the CRD performance under all possible conditions as described in Subsection 3.9.4.4.

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of these tests:

(1) development tests;

(2) factory quality control tests;

(3) Five-year maintenance life tests;

(4) 1.5X design life tests;

(5) operational tests;

(6) acceptance tests; and

(7) surveillance tests.

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

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

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

(4) 1.5X Design Life Tests - When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the service life cycles.

3.9.5 Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel (RPV) internals, including core support structures.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are:

(1) Core Support Structures

Shroud;

Shroud support (including the internal pump deck);

driven from underneath by a pump shaft, with the impeller being encircled by a diffuser shroud assembled into the pump deck opening.

The RM section of the RIP is located underneath, and at the periphery of, the RPV bottom head inside a pressure retaining housing termed the motor casing. The motor casing itself is not part of the RM, but is instead a part of and welded into an RPV nozzle (pump nozzle). The motor casing thus comprises part of the reactor coolant pressure boundary and is a Safety Class 1 component.

The principal element of the stretch tube section is a thin-walled tube configured as a hollow bolt fitting around the pump shaft and within the pump nozzle. It has an external lip (bolt head) at its upper end and an external threaded section at this lower end. The stretch tube function is to achieve tight clamping of the IP diffuser to the gasketed, internal-mount end of the RPV pump nozzle, at all extremes of thermal transients and pump operating conditions.

3.9.5.1.2.3 Steam Dryer Assembly

The steam dryer assembly is a non-safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryer removes moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus.

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure which is removable from the reactor pressure vessel as an integral unit. The assembly includes the dryer banks, dryer supply and discharge ducting, drain collecting trough, drain piping, and a skirt which forms a water seal extending below the separator reference zero elevation. Upward and radial movement of the dryer assembly under the action of blowdown and seismic loads are limited by reactor vessel internal stops which are arranged to permit differential expansion growth of the dryer assembly with respect to the reactor pressure vessel. The assembly is arranged for removal from the vessel as an integral unit on a routine basis.

3.9.5.1.2.4 Feedwater Spargers

These are Safety Class 2 components. They are discussed here to describe coolant flow paths in the vessel and their safety function. Each of two feedwater lines is connected to three spargers via three RPV nozzles. One line is utilized by the RCIC system; the other by the RHR shutdown cooling system. During the ECCS mode, the two groups of spargers support diverse type of flooding of the vessel. The RCIC system side supports high pressure flooding and the RHR system side supports low pressure flooding, as required during the ECCS operation.

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger in two halves is fitted to each feedwater nozzle via a tee and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement, with all connections made by full penetration welds. 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 to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feed- water also serves to condense steam in the region above the downcomer annulus and to subcool water flowing to the recirculation internal pumps.

3.9.5.1.2.5 RHR/ECCS Low Pressure Flooder Spargers

These are Safety Class 2 components. The design features of these two spargers of the RHR shutdown cooling system are similar to those of the six feedwater spargers, three of which belonging to one feedwater line support additionally the same RHR (and ECCS) function. During the ECCS mode, these spargers support low pressure flooding of the vessel. The feedwater spargers are described in Subsection 3.9.5.1.2.4.

Two lines of RHR shutdown cooling system enter the reactor vessel through the two diagonally opposite nozzles and connect to the

spargers. The sparger tee inlet is connected to the RPV nozzle safe end by a thermal sleeve arrangement with all connections made by full penetration welds.

3.9.5.1.2.6 ECCS High Pressure Core Flooder Spargers and Piping

210.65 The core flooder spargers and piping are Safety Class 2. The spargers and piping are the means for directing high pressure ECCS flow to the upper end of the core during accident conditions.

Each of two high pressure core flooder (HPCF) system lines enters the reactor vessel through a diagonally opposite nozzle in the same manner as an RHR low pressure flooder line, except that the curved sparger including the connecting tee is routed around the inside of and is supported by the cylindrical portion of the top guide. A flexible coupling is interposed between the sparger tee inlet and the sleeved inlet connector inside the nozzle. The two spargers are supported so as to accommodate thermal expansion.

3.9.5.1.2.7 RPV Vent and Head Spray Assembly

This is designed as a Safety Class 1 component. However, only the nozzle portion of the assembly is a reactor coolant pressure boundary, and the assembly function is not a safety-related operation. The reactor water cleanup return flow to the reactor vessel, via feedwater lines, can be diverted partly to a spray nozzle in the reactor head in preparation for refueling cooldown. The spray maintains saturated conditions in the reactor vessel head volume by condensing stream being generated by the hot reactor vessel walls and internals. The head spray subsystem is designed to rapidly cooldown the reactor vessel head flange region for refueling and to allow installation of steam line plugs before vessel floodup for refueling.

The head vent side of the assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. During shutdown and filling for hydrotesting, steam and noncondensable gases may be vented to the drywell equipment sump while the connection to the steamline is blocked. When draining the vessel during shutdown, air enters the vessel through the vent.

3.9.5.1.2.8 Core and Internal Pump Differential Pressure Lines

These lines comprise the core flow measurement subsystem of the recirculation flow control system (RFCS) and provide two methods of measuring the ABWR core flow rates. The core DP lines (Safety Class 3) and internal pump DP lines (non-safety class) enter the reactor vessel separately through reactor bottom head penetrations. Four pairs of the core DP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation.

Similarly, four pairs of the internal pump DP lines terminate above and below the pump deck and are used to sense the pressure across the pump during normal pump operation. Each pair is routed concentrically through a penetration and upward along a shroud support leg in the lower plenum.

3.9.5.1.2.9 In-Core Guide Tubes and Stabilizers

These are Safety Class 3 components. The guide tubes protect the in-core instrumentation from flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core as well as a path for insertion and withdrawal of the calibration monitors (ATIP, automated traversing incore probe subsystem). The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing to the top of the core plate. The power range detectors for the power range monitoring units and the dry tubes for the startup range neutron monitoring and average power range monitoring (SRNM/APRM) detectors are inserted through the guide tubes.

Two levels of stainless steel stabilizer latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide

3.9.7 Interfaces

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first applicant referencing the ABWR design will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

R.G. 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first applicants docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first applicant referencing the ABWR design will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals. (See Subsection 3.9.2.4 for interface requirements).

3.9.7.2 ASME Class 2 or 3 or Quality Group Components with 60 Year Design Life

Applicants referencing the ABWR design will identify ASME Class 2 or 3 or Quality Group D components that are subjected to loadings which could result in thermal or dynamic fatigue and provide the analyses required by the ASME Code, Subsection NB. These analyses will include the appropriate operating vibration loads and for the effects of mixing hot and cold fluids. (See Subsection 3.9.3.1 for interface requirements).

3.9.7.3 Audit of Design Specification and Design Reports

Applicants referencing the ABWR design will make available to the NRC staff Design Specifications and Design Reports required by the ASME Code for vessels, pumps, valves and piping systems for the purpose of audit.

3.9.8 References

1. *BWR Fuel Channel Mechanical Design and Deflection*, NEDE-21354-P, September 1976.

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

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

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

5. *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, NUREG-0619.

6. *General Electric Environmental Qualification Program*, NEDE-24326-1-P, Proprietary Document, January 1983.

7. *Functional Capability Criteria for Essential Mark II Piping*, NEDO-21985, September 1978, prepared by Battelle Columbus Laboratories for General Electric Company.

8. *Generic Criteria for High Frequency Cutoff of BWR Equipment*, NEDO-25250, Proprietary Document, January 1980.

Table 3.9-7

FATIGUE LIMIT
FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY

Summation of fatigue damage usage following Minor hypotheses(1):

Cumulative Damage in Fatigue	Limit for Service Levels A&B (Normal and Upset Conditions)
Design fatigue cycle usage from analysis using the method of the ASME Code	$\leq$ 1.0

NOTE
(1) Miner, M.A., *Cumulative Damage in Fatigue*, Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

Table 3.9-8

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

STANDBY LIQUID CONTROL SYSTEM

C41-F006 A,B	Injection Valves
C41-F008	Inboard Check Valve

RESIDUAL HEAT REMOVAL SYSTEM

E11-F005 A,B,C	Injection Valve Loops A,B&C
E11-F006 A,B,C	Testable Check Valve A,B&C
E11-F010 A,B,C	Shutdown Cooling Inboard Suction Isolation Valve Loops A,B&C
E11-F011 A,B,C	Shutdown Cooling Outboard Suction Isolation Valve Loops A,B&C

HIGH PRESSURE CORE FLOODER SYSTEM

E22-F003 B,C	Injection Valve Loops B&C
E22-F004 B,C	Testable Check Valve Loops B&C

REACTOR CORE ISOLATION COOLING SYSTEM

E51-F004	Injection Valve
E51-F005	Testable Check Valve

210.49

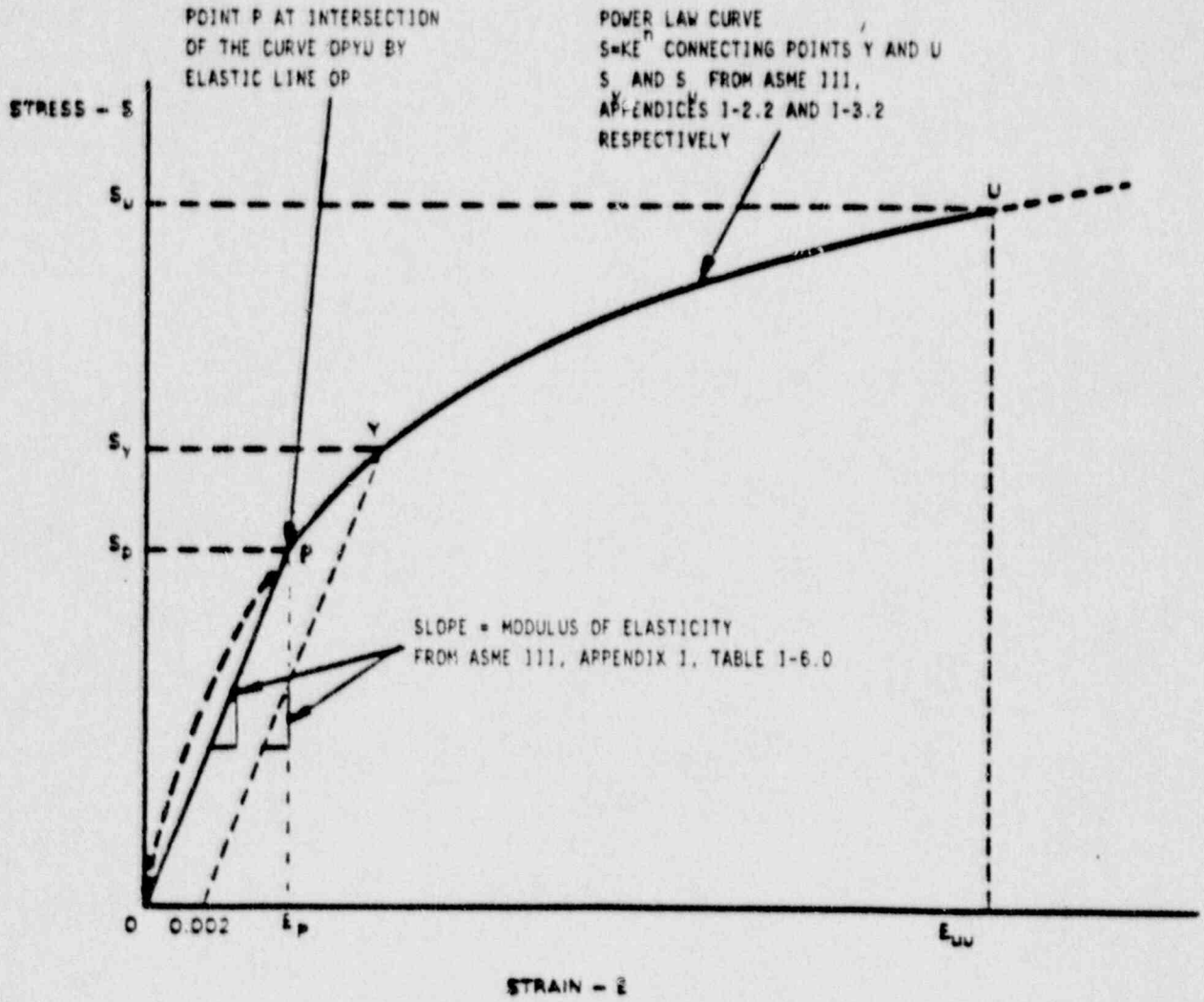

Figure 3.9-6 STRESS-STRAIN CURVE FOR BLOWOUT RESTRAINTS

Table 3G.4-1

Site-Envelope Horizontal OBE loads for R/B and RCCV

X-AXIS (RB 0-180 DEG)

ELEV (M)	REACTOR BUIDING SHEAR (TONS)	REACTOR BUIDING MOMENT (T-M)	RCCV SHEAR (TONS)	RCCV MOMENT (T-M)
44.7		1.93E+04		
	3.95E+03			
33.2		7.67E+04		
	8.29E+03			
26.7		1.46E+05		5.44E+04
	1.35E+04		5.28E+03	
18.5		2.73E+05		8.70E+04
	1.56E+04		1.47E+04	
13.1		3.66E+05		1.08E+05
	2.10E+04		1.63E+04	
7.3		4.84E+05		1.78E+05
	2.10E+04		1.63E+04	
-0.2		4.84E+05		2.13E+05
	2.10E+04		1.63E+04	
-6.7		4.84E+05		2.32E+05
	2.10E+04		1.63E+04	
-13.2		5.64E+05		2.95E+05

Y-AXIS (RB 90-270 DEG)

ELEV (M)	REACTOR BUIDING SHEAR (TONS)	REACTOR BUIDING MOMENT (T-M)	RCCV SHEAR (TONS)	RCCV MOMENT (T-M)
44.7		9.53E+03		
	3.93E+03			
33.2		9.11E+04		
	7.55E+03			
26.7		1.79E+05		4.44E+04
	1.28E+04		3.11E+03	
18.5		2.94E+05		1.12E+05
	1.72E+04		7.80E+03	
13.1		3.79E+05		1.82E+05
	2.11E+04		9.73E+03	
7.3		4.82E+05		2.61E+05
	2.11E+04		9.73E+03	
-0.2		5.09E+05		3.06E+05
	2.23E+04		1.04E+04	
-6.7		5.62E+05		3.32E+05
	2.39E+04		1.11E+04	
-13.2		6.37E+05		3.51E+05

Notes:
1. Elevations are relative to the RPV bottom head.
2. Forces on the RB between EL 33.2M and 18.5M along the x-axis are the sum of maximum forces of the two sticks representing the walls as shown in Fig. 3G.2-1.

Table 3G.4-2

Site-Envelope OBE Loads for Selected Locations

LOCATION	BEAM ELT	MAXIMUM FORCE P(T)	VX(T)	VY(T)	MAXIMUM MOMENT MX(T-M)	MY(T-M)
Shroud Sup't	28	98.0	263.8	286.0	1830.0	2281.0
RPV Skirt	69	577.0	780.3	614.0	5322.1	3909.9
RSW Base	78	458.0	1044.0	853.0	5064.0	4168.0
Pedestal Base	86	2027.0	3343.0	2806.0	72077.0	58671.0

Notes:
1. P is vertical load due to vertical excitation
2. VX and MX are shear and moment due to HOR X excitation
3. VY and MY are shear and moment due to HOR Y excitation

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		420.25	Chap 7	20.3.8	8
		420.26	7.1.2.1.6	20.3.8	8
		420.27	Chap 7	20.3.8	8
		420.28	App 15A	20.3.8	8
		420.29	7.1.1	20.3.8	8
		420.30	7.1.2.2	20.3.8	8
		420.31	7.1.2.3.2	20.3.8	8
		420.32	7.1.2.3.2	20.3.8	8
		420.33	7.1.2.3.2	20.3.8	8
		420.34	7.1.2.3.7	20.3.8	8
		420.35	7.1.2.6.5	20.3.8	8
		420.36	7.1.2.6.6	20.3.8	8
		420.37	7.1.2.6.7	20.3.8	8
		420.38	7.1	20.3.8	8
		420.39	7.1	20.3.8	8
		420.40	7.3.1.1.1.1	20.3.8	8
		420.41	7.3.1.1.1.1	20.3.8	8
		420.42	7.3.1.1.1.1	20.3.8	8
		420.43	7.3.1.1.2	20.3.8	8
		420.44	7.3.1.1.1.3	20.3.8	8
		420.45	7.3.1.1.1.3	20.3.8	8
		420.46	7.3.1.1.1.4	20.3.8	8
		420.47	7.3.1.1.1.4	20.3.8	8
		420.48	7.1.2.1.6	20.3.8	8
		420.49	Chap 7	20.3.8	8
		420.50	7.1	20.3.8	8
		420.51	7.1	20.3.8	8
		420.52	Chap 7	20.3.8	8
		420.53	Chap 7	20.3.8	8
		420.54	Chap 7	20.3.8	8
		420.55	Chap 7	20.3.8	8
		420.56	Chap 7	20.3.8	8
		420.57	Chap 7	20.3.8	8
		420.58	Chap 7	20.3.8	8
		420.59	Chap 7	20.3.8	8
		420.60	7.1.2.2	20.3.8	8
		420.61	7.1.2.2	20.3.8	8
		420.62	7.1.2.10.11	20.3.8	8
		420.63	Chap 7	20.3.8	8
		420.64	Chap 7	20.3.8	8
		420.65	Chap 7	20.3.8	8
		420.66	Chap 7	20.3.8	8
		420.67	Chap 7	20.3.8	8
		420.68	Chap 7	20.3.8	8
		420.69	Chap 7	20.3.11	11

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		420.70	7.1.2.1.6	20.3.8	8
		420.71	7.1.2.1.6	20.3.8	8
		420.72	7.1.2.1.6	20.3.8	8
		420.73	7.1.2.1.6	20.3.8	8
		420.74	7.1.2.1.6	20.3.8	8
		420.75	7.1.2.2	20.3.8	8
		420.76	7.1.2.3.2	20.3.8	8
		420.77	7.1.2.1.4.1	20.3.8	8
		420.78	7.1.2.1.4.1	20.3.8	8
		420.79	7.1.2.1.4.1	20.3.8	8
		420.80	7.1.2.3.1	20.3.8	8
		420.81	7.1.2.3.1	20.3.8	8
		420.82	7.1.2.3.3	20.3.8	8
		420.83	7.1.2.3.4	20.3.8	8
		420.84	App 3I	20.3.8	8
		420.85	Chap 7	20.3.8	8
		420.86	Chap 7	20.3.8	8
		420.87	Chap 7	20.3.8	8
		420.88	Chap 7	20.3.8	8
		420.89	Chap 7	20.3.8	8
		420.90	Chap 7	20.3.8	8
		420.91	Chap 7	20.3.8	8
		420.92	Chap 7	20.3.8	8
		420.93	Chap 7	20.3.8	8
		420.94	Chap 7	20.3.8	8
		420.95	Chap 7	20.3.8	8
		420.96	15A.6	20.3.8	8
		420.97	7.3.1.1.4	20.3.8	8
		420.98	Chap 7	20.3.8	8
		420.99	Chap 7	20.3.8	8
		420.100	Chap 7	20.3.8	8
		420.101	Chap 7	20.3.8	8
		420.102	Chap 7	20.3.8	8
		420.103	Chap 7	20.3.8	8
		420.104	Chap 7	20.3.8	8
		420.105	Chap 7	20.3.8	8
		420.106	Chap 7	20.3.8	8
		420.107	9.3.5.2	20.3.8	8
		420.108	7.1.2.2	20.3.8	8
		420.109	7.1.2.3.1	20.3.8	8
		420.110	7.1.2.3.1	20.3.8	8
		420.111	7.1.2.3.7	20.3.8	8
		420.112	7.1.2.4.3	20.3.8	8
		420.113	7.1.2.6.1.1	20.3.8	8
		420.114	App 7A	20.3.8	8
		420.115	7.3.1.1.1.3	20.3.8	8
		420.116	1.2.2.4.8.1.2	20.3.8	8
		420.117	9.3.5.1.1	20.3.8	8

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		420.118	15.2.4.5.1	20.3.8	8
		420.119	7.4.1.2	20.3.8	8
		420.120	7.3.2.1.2	20.3.8	8
		420.121	7.3.1.2	20.3.8	8
		420.122	15.2.2.2.1.4	20.3.8	8
		420.123	15B.4	20.3.11	11
		420.124	15B.4	20.3.11	11
		420.125	7.4.1.4	20.3.11	11
		420.126	7A.7	20.3.11	11
		420.127	Chap 7	20.3.11	11
		420.128	7A.7	20.2.11	11
		420.129	Chap 7	20.3.11	11
		420.131	19.2.3.4	20.3.11	11
		420.132	19.3.1.3.1	20.3.11	11
		420.133	19.3.1.3.1	20.3.11	11
		420.134	19D.3.4	20.3.11	11
		420.135	19D.6	20.3.11	11
		420.136	App 7A	20.3.11	11
SPLB	Plant Systems	430.1	4.6	20.3.2	2
		430.2	5.2.5	20.3.2	2
		430.3	5.2.5	20.3.2	2
		430.4	5.2.5.4.1	20.3.2	2
		430.5	5.2.5	20.3.2	2
		430.6	5.2.5	20.3.2	2
		430.7	6.2	20.3.2	2
		430.8	6.2	20.3.2	2
		430.9	6.2	20.3.2	2
		430.10	6.2	20.3.2	2
		430.11	6.2	20.3.2	2
		430.12	6.2	20.3.2	2
		430.13	6.2.1.1.3	20.3.2	2
		430.14	6.2	20.3.2	2
		430.15	6.2	20.3.2	2
		430.16	6.2	20.3.2	2
		430.17	6.2.1.2.3	20.3.2	2
		430.18	6.2	20.3.2	2
		430.19	6.2	20.3.2	2
		430.20	6.2	20.3.2	2
		430.21	6.2	20.3.2	2
		430.22	6.2	20.3.2	2
		430.23	6.2	20.3.2	2
		430.24	6.2	20.3.2	2
		430.25	6.2	20.3.2	2
		430.26	6.2	20.3.2	2
		430.27	6.2	20.3.2	2
		430.28	6.2	20.3.2	2
		430.29	6.2.3	20.3.2	2

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		430.30	6.2	20.3.2	2
		430.31	6.2	20.3.2	2
		430.32	6.2	20.3.2	2
		430.33	6.2	20.3.2	2
		430.34	6.2	20.3.2	2
		430.35	6.2	20.3.2	2
		430.36	6.2	20.3.2	2
		430.37	6.2	20.3.2	2
		430.38	6.2	20.3.2	2
		430.39	6.2.4	20.3.2	2
		430.40	6.2	20.3.2	2
		430.41	6.2	20.3.2	2
		430.42	6.2	20.3.2	2
		430.43	6.2	20.3.2	2
		430.44	6.2	20.3.2	2
		430.45	6.2	20.3.2	2
		430.46	6.2	20.3.2	2
		430.47	6.2.5.3	20.3.2	2
		430.48	6.2.6	20.3.2	2
		430.49	6.2.6	20.3.2	2
		430.50	6.2.6	20.3.2	2
		430.51	6.2.6	20.3.2	2
		430.52	6.2.6	20.3.2	2
		430.53	6.2.6	20.3.2	2
		430.54	6.4	20.3.2	2
		430.55	6.5.1	20.3.2	2
		430.56	6.5.3	20.3.2	2
		430.57	6.7	20.3.2	2
		430.58	15.7.3	20.3.2	2
		430.59	10.1	20.3.11	11
		430.60	10.2	20.3.11	11
		430.61	10.2.2.2	20.3.11	11
		430.62	10.2	20.3.11	11
		430.63	10.2.2.4	20.3.11	11
		430.64	10.2.2.4	20.3.11	11
		430.65	10.2	20.3.11	11
		430.66	10.2	20.3.11	11
		430.67	10.3.2.1	20.3.11	11
		430.68	10.3.3	20.3.11	11
		430.69	10.3	20.3.11	11
		430.70	10.3	20.3.11	11
		430.71	10.4.1	20.3.11	11
		430.72	10.4.1	20.3.11	11
		430.73	10.4.1	20.3.11	11
		430.74	10.4.2	20.3.11	11
		430.75	10.4.2	20.3.11	11
		430.76	10.4.2	20.3.11	11
		430.77	10.4.2	20.3.11	11
		430.78	10.4.2	20.3.11	11

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		430.79	10.4.2	20.3.11	11
		430.80	10.4.3	20.3.11	11
		430.81	10.4.3	20.3.11	11
		430.82	10.4.3	20.3.11	11
		430.83	10.4.3	20.3.11	11
		430.84	10.4.4	20.3.11	11
		430.85	10.4.5	20.3.11	11
		430.86	10.4.7	20.3.11	11
		430.87	Chap 10	20.3.11	11
		430.88	Chap 10	20.3.11	11
		430.89	10.4.7	20.3.11	11
		430.90	10.4.7	20.3.11	11
SELS	Power Systems	435.1	8.1.2.1	20.3.8	8
		435.2	Chap 8	20.3.8	8
		435.3	8.1.2.1	20.3.8	8
		435.4	8.2.3	20.3.8	8
		435.5	8.2.3	20.3.8	8
		435.6	8.3.1.1.4.1	20.3.8	8
		435.7	8.3.1.1.4.2.2	20.3.8	8
		435.8	8.3	20.3.8	8
		435.9	8.3.1.1.4.2.3	20.3.8	8
		435.10	8.3.1.1.4.2.4	20.3.8	8
		435.11	8.3.1.1.5.1	20.3.8	8
		435.12	8.3.1.1.5.2	20.3.8	8
		435.13	8.3.1.1.6.4	20.3.8	8
		435.14	8.3.1.1.7	20.3.8	8
		435.15	8.3.1.1.7	20.3.8	8
		435.16	8.3.1.1.7	20.3.8	8
		435.17	8.3.1.1.7	20.3.8	8
		435.18	8.3.1.1.7	20.3.8	8
		435.19	8.3.1.1.7	20.3.8	8
		435.20	8.3.1.1.7	20.3.8	8
		435.21	8.3.1.1.8.2	20.3.8	8
		435.22	8.3.1.1.8.5	20.3.8	8
		435.23	8.3.1.2.1	20.3.8	8
		435.24	8.3.1.2.1	20.3.8	8
		435.25	8.3.1.1.2.3	20.3.8	8
		435.26	8.3.1.2.2	20.3.8	8
		435.27	8.3.1.2.2	20.3.8	8
		435.28	8.3.1.2.4	20.3.8	8
		435.29	8.3.1.3.1	20.3.8	8
		435.30	Chap 8	20.3.8	8
		435.31	8.3.1.4.1.2	20.3.8	8
		435.32	8.3.1.4.2.1	20.3.8	8
		435.33	8.3.1.4.2.2.2	20.3.8	8
		435.34	8.3.1.4.2.2.4	20.3.8	8
		435.35	8.3.1.4.2.3.1	20.3.8	8
		435.36	8.3.1.4.2.3.2	20.3.8	8
		435.37	8.3.2.1	20.3.8	8

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		435.38	8.3.2.1	20.3.8	8
		435.39	8.3.2.1	20.3.8	8
		435.40	8.3.2.1	20.3.8	8
		435.41	8.3.2.1.2	20.3.8	8
		435.42	8.3.2.1.3	20.3.8	8
		435.43	8.3.2.1.3.3	20.3.8	8
		435.44	8.3.2.2.1	20.3.8	8
		435.45	8.3.3.1	20.3.8	8
		435.46	Table 8.3-1,2,3	20.3.8	8
		435.47	Figure 8.3-1	20.3.8	8
		435.48	Figure 8.3-2	20.3.8	8
		435.49	Figure 8.3-3	20.3.8	8
		435.50	Figure 8.3-4	20.3.8	8
		435.51	Figures 8.3-5, 6,7,8	20.3.8	8
		435.52	Figure 8.3-7	20.3.8	8
		435.53	Figure 8.3-8	20.3.8	8
		435.54	Table 3.2-1	20.3.8	8
		435.55	8.2.1.1.8.9	20.3.8	8
		435.56	Chap 8	20.3.8	8
		435.57	Chap 8	20.3.8	8
		435.58	Chap 8	20.3.8	8
		435.59	Chap 8	20.3.8	8
		435.60	8.3.1.2.1	20.3.8	8
		435.61	Chap 8	20.3.8	8
		435.62	Chap 8	20.3.8	8
SRXB	Reactor Systems	440.1	4.6	20.3.2	2
		440.2	4.6.2.3.2.2	20.3.2	2
		440.3	4.6.1.2	20.3.2	2
		440.4	4.6	20.3.2	2
		440.5	4.6	20.3.2	2
		440.6	4.6	20.3.2	2
		440.7	4.6	20.3.2	2
		440.8	4.6	20.3.2	2
		440.9	4.6	20.3.2	2
		440.10	4.6.2.3.1	20.3.2	2
		440.11	4.6	20.3.2	2
		440.12	4.6	20.3.2	2
		440.13	5.2.2	20.3.4	4
		440.14	5.2.2	20.3.4	4
		440.15	5.2.2	20.3.4	4
		440.16	5.2.2	20.3.4	4
		440.18	5.2.2	20.3.4	4
		440.19	5.2.2	20.3.4	4
		440.20	5.2.2	20.3.4	4
		440.21	5.2.2	20.3.4	4
		440.22	5.1	20.3.4	4
		440.23	5.2.2	20.3.4	4
		440.28	1.8	20.3.4	4
		440.29	5.2.2	20.3.4	4

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		440.30	4.6	20.3.4	4
		440.31	4.6	20.3.4	4
		440.17	5.2.2	20.3.4	4
		440.32	4.6	20.3.4	4
		440.33	4.6	20.3.4	4
		440.34	5.4.1	20.3.4	4
		440.35	5.4.1	20.3.4	4
		440.36	5.4.1	20.3.4	4
		440.37	5.4.6	20.3.4	4
		440.38	5.4.6	20.3.4	4
		440.39	5.4.6	20.3.4	4
		440.40	5.4.6	20.3.4	4
		440.41	5.4.6	20.3.4	4
		440.42	5.4.6	20.3.4	4
		440.43	5.4.6	20.3.4	4
		440.44	5.4.6	20.3.4	4
		440.45	5.4.6	20.3.4	4
		440.46	5.4.6	20.3.4	4
		440.47	5.4.6	20.3.4	4
		440.48	5.4.6	20.3.4	4
		440.49	5.4.6	20.3.4	4
		440.50	5.4.6	20.3.4	4
		440.51	5.4.6	20.3.4	4
		440.52	5.4.6	20.3.4	4
		440.53	5.4.6	20.3.4	4
		440.54	5.4.6	20.3.4	4
		440.55	5.4.6	20.3.4	4
		440.56	5.4.6	20.3.4	4
		440.57	5.4.6	20.3.4	4
		440.58	5.4.6	20.3.4	4
		440.59	5.4.7	20.3.4	4
		440.60	5.4.7	20.3.4	4
		440.61	5.4.7	20.3.4	4
		440.62	5.4.7	20.3.4	4
		440.63	5.4.7	20.3.4	4
		440.64	5.4.7	20.3.4	4
		440.65	5.4.7	20.3.4	4
		440.72	5.4.7	20.3.4	4
		440.73	5.4.7	20.3.4	4
		440.74	5.4.7	20.3.4	4
		440.75	6.3	20.3.6	6
		440.76	6.3	20.3.6	6
		440.77	6.3	20.3.6	6
		440.78	6.3	20.3.6	6
		440.79	6.3	20.3.6	6
		440.80	6.3	20.3.6	6
		440.81	6.3	20.3.6	6
		440.82	6.3	20.3.6	6
		440.83	6.3	20.3.6	6

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		440.84	6.3	20.3.6	6
		440.85	6.3	20.3.6	6
		440.86	6.3	20.3.6	6
		440.87	6.3	20.3.6	6
		440.88	6.3	20.3.6	6
		440.89	6.3	20.3.6	6
		440.90	6.3	20.3.6	6
		440.91	6.3	20.3.6	6
		440.92	6.3	20.3.6	6
		440.93	6.3	20.3.6	6
		440.94	6.3	20.3.6	6
		440.95	6.3	20.3.6	6
		440.96	6.3	20.3.6	6
		440.97	6.3	20.3.6	6
		440.98	6.3	20.3.6	6
		440.99	6.3	20.3.6	6
		440.100	6.3	20.3.6	6
		440.101	9.3.5	20.3.6	6
		440.102	9.3.5	20.3.6	6
		440.103	9.3.5	20.3.6	6
		440.104	9.3.5	20.3.6	6
		440.105	9.3.5	20.3.6	6
		440.106	9.3.5	20.3.6	6
		440.107	9.3.5	20.3.6	6
		440.108	Chap 15	20.3.6	6
		440.109	Chap 15	20.3.6	6
		440.110	Chap 15	20.3.6	6
		440.111	Chap 15	20.3.6	6
		440.112	Chap 15	20.3.6	6
		440.113	Chap 15	20.3.6	6
		440.114	Chap 15	20.3.6	6
		440.115	Chap 15	20.3.6	6
		440.116	Chap 15	20.3.6	6
PRPB	Meteorology	451.1	2.0	20.3.3	3
		451.2	2.4	20.3.3	3
SPLB	Effluent Treatment	460.1	11.1	20.3.7	7
		460.2	11.1	20.3.7	7
		460.3	11.1	20.3.7	7
		460.4	11.1	20.3.7	7
		460.5	11.5	20.3.7	7
PRPB	Radiological Report	470.1	15.5.2	20.3.1	1
		470.2	15.6.2	20.3.1	1
		470.3	15.6.4.5.1.1	20.3.1	1
		470.4	15.6.5.5	20.3.1	1
		470.5	15.6.5	20.3.1	1
		470.6	15.7.5	20.3.1	1
		470.7	15.7	20.3.1	1
		470.8	15.7	20.3.1	1

NRC° Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI°° Letter
		470 9	15.7	20.3.1	1
		470.10	15.7	20.3.1	1
PRPB	Radiation Protection	471.1	12.1.1.2	20.3.7	7
		471.2	12.1.2.2.1	20.3.7	7
		471.3	12.1.2.2.2	20.3.7	7
		471.4	12.1.2.3	20.3.7	7
		471.5	12.1.2.3	20.3.7	7
		471.6	12.1.2.3.2	20.3.7	7
		471.7	12.1.2.3.2	20.3.7	7
		471.8	12.2	20.3.7	7
		471.9	11.1	20.3.7	7
		471.10	12.1.2.2.3	20.3.7	7
		471.11	12.3	20.3.7	7
		471.12	11.1	20.3.7	7
		471.13	12.2.2	20.3.7	7
		471.14	12.2.2	20.3.7	7
		471.15	Chap. 12	20.3.7	7
		471.16	Chap. 12	20.3.7	7
		471.17	12.2	20.3.7	7
		471.18	Chap. 12	20.3.7	7
		471.19	12.2	20.3.7	7
		471.20	Chap. 12	20.3.7	7
		471.21	12.3	20.3.7	7
		471.22	Chap. 12	20.3.7	7
		471.23	12.3.1.3	20.3.7	7
		471.24	12.3	20.3.7	7
		471.25	12.3	20.3.7	7
		471.26	12.3	20.3.7	7
		471.27	12.3	20.3.7	7
		471.28	Chap. 12	20.3.7	7
		471.29	12.3	20.3.7	7
		471.30	Chap. 12	20.3.7	7
		471.31	12.2.2.1	20.3.7	7
		471.32	Chap. 12	20.3.7	7
		471.33	12.3	20.3.7	7
		471.34	12.3.5	20.3.7	7
		471.35	Chap. 12	20.3.7	7
		471.36	12.3	20.3.7	7
		471.37	12.3	20.3.7	7
		471.38	12.3.1	20.3.7	7
		471.39	Chap. 12	20.3.7	7
		471.40	12.3	20.3.7	7
		471.41	12.3	20.3.7	7
RES	Probabilistic Risk Assesment	725.1	App. 19D	20.3.9	9
		725.2	App. 19D	20.3.9	9
		725.3	App. 19D	20.3.9	9
		725.3	App. 19D	20.3.9	9
		725.4	App. 19D	20.3.9	9
		725.5	App. 19D	20.3.9	9

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
		725.6	App. 19D	20.3.9	9
		725.7	App. 19D	20.3.9	9
		725.8	App. 19D	20.3.9	9
		725.9	App. 19D	20.3.9	9
		725.10	App. 19D	20.3.9	9
		725.11	App. 19D	20.3.9	9
		725.12	App. 19D	20.3.9	9
		725.13	App. 19D	20.3.9	9
		725.14	App. 19D	20.3.9	9
		725.15	App. 19D	20.3.9	9
		725.16	App. 19D	20.3.9	9
		725.17	App. 19D	20.3.9	9
		725.18	App. 19D	20.3.9	9
		725.19	App. 19D	20.3.9	9
		725.20	App. 19D	20.3.9	9
		725.21	App. 19D	20.3.9	9
		725.22	App. 19D	20.3.9	9
		725.23	App. 19D	20.3.9	9
		725.24	App. 19D	20.3.9	9
		725.25	App. 19D	20.3.9	9
		725.26	App. 19D	20.3.9	9
		725.27	App. 19D	20.3.9	9
		725.28	App. 19D	20.3.9	9
		725.29	App. 19D	20.3.9	9
		725.30	App. 19D	20.3.9	9
		725.31	App. 19D	20.3.9	9
		725.32	App. 19D	20.3.9	9
		725.33	App. 19D	20.3.9	9
		725.34	App. 19D	20.3.9	9
		725.35	App. 19D	20.3.9	9
		725.36	App. 19D	20.3.9	9
		725.37	App. 19D	20.3.9	9
		725.38	App. 19D	20.3.9	9
		725.39	App. 19D	20.3.9	9
		725.40	App. 19D	20.3.9	9
		725.41	App. 19D	20.3.9	9
		725.42	App. 19D	20.3.9	9
		725.43	App. 19D	20.3.9	9
		725.44	App. 19D	20.3.9	9
		725.45	App. 19D	20.3.9	9
		725.46	App. 19D	20.3.9	9
		725.47	App. 19D	20.3.9	9
		725.48	App. 19D	20.3.9	9
		725.49	App. 19D	20.3.9	9
		725.50	App. 19D	20.3.9	9
		725.51	App. 19D	20.3.9	9
		725.52	App. 19D	20.3.9	9
		725.53	App. 19D	20.3.9	9
		725.54	App. 19D	20.3.9	9
		725.55	App. 19D	20.3.9	9
		725.56	App. 19D	20.3.9	9

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
		725.57	App. 19D	20.3.9	9
		725.58	App. 19D	20.3.9	9
		725.59	App. 19D	20.3.9	9
		725.60	App. 19D	20.3.9	9
		725.61	App. 19D	20.3.9	9
RSGB	Safeguards	910.7	13.6.1	20.3.7	7
		910.8	13.6.3.7	20.3.7	7
		910.9	13.6.3	20.3.7	7
		910.10	13.6.3.7	20.3.7	7
		910.11	13.6.3.3	20.3.7	7
		910.12	13.6.3.4	20.3.7	7
		910.13	13.6.3.6	20.3.7	7
		910.14	13.6.3	20.3.7	7
		910.16	13.6	20.3.7	7

420.18

For the proposed use of digital computers, show how the digital system is superior to analog alternatives to implementing the logic. Show how the analyses determined that the realiability of the digital computer based system was better than the reliability of the analog system. (7)

420.19

This section states that automatic self-test is performed sequentially on all four divisions, to minimize common mode effects, and that a complete self-test sequence through all four divisions takes no more than 30 minutes. The original response to Question 19 revised this section. What hardware and software design features are provided to allow sequencing and testing of the four divisions without violating independence/isolation criteria? The revised section appears to allow a common centralized test driver. Illustrate with a block diagram. (7.1.2.1.6.(4))

420.20

Describe the fiber optic links in the safety systems. What signals are multiplexed on each link? Show how the independence criteria in accordance with IEEE Std 603 and IEEE Std 379 is satisfied with the proposed configuration of fiber optic links.(7)

420.21

Describe the safety computer system's interface to any non-safety computer systems and other plant instrumentation. Describe if information transfer from 1E to N-1E computers is via broadcast or handshake. (7)

420.22

Provide a table of conformance to IEEEE 603 and ANSI/IEEE 7-4.3.2. (7)

420.23

Provide a table of conformance to IEEE 384, indicating where credit is taken for isolation or separation, what devices or methods are used, and the basis of isolation device qualification. If specific types of components have not been chosen, provide specification level information including testing acceptance criteria. (7)

420.24

Are any artificial intelligence features provided in the proposed system, whereby probabilistic judgements are made by the system, or whereby the system can "learn" during its operational life? (7)

420.25

Is credit taken in the safety analysis for any rotating memory devices such as disk drives? (7)

420.26

What is the definition of "Safety Associated" as used in SAR Section 7.1.2.1.6? (7.1.2.1.6)

420.27

Specify which parameters are to be triplicated. At what point does the triplication start (flow orifice, sensor?) and end (transmitter, trip logic?). If there is triplication of sensors is there diversity between sensors? (7)

420.29

For those systems where it has not already been done (example 7.1.1.3.5) clarify whether manual or automatic initiation will be used. (7.1.1)

420.30

Define the word "sufficient" used in section (j). (7.1.2.2)

420.31

For section 7.1.2.3.2(1)(c,d,e) and (2)(a) define "sufficient". (7.1.2.3.2)

420.32

The listed design basis should include instrumentation necessary to inform the operator that isolation has been completed and control should provide ability for operator to reset (with adequate safeguards against inadvertently breaking isolation). (7.1.2.3.2)

420.33

Add to 7.1.2.3.2(2)(c)..."without causing plant shutdowns" or reducing safety margins. (7.1.2.3.2)

420.34

For Section 7.1.2.3.7(1)(b) provide a listing of the nonessential parts of the cooling water system which should be isolated. List any nonessential parts for which isolation is not provided. (7.1.2.3.7)

420.35

Is the wetwell to drywell vacuum breaker control manual or automatic? (7.1.2.6.5)

420.36

If the CAMS system is only a monitoring system, why is it not always on instead of waiting for a LOCA to monitor radiation? (7.1.2.6.6)

420.37

What is the immediate safety action required by relief valve leakage and is it automatic? (7.1.2.6.7)

420.38

The table indicates RG 1.151 applies only to safety related display and Non-1E control systems. Section 7.1.2.10.11 refers to other safety systems including RPS and ECCS. Clarify which systems RG 1.151 is to apply to. (Table 7.1-2)

420.39

The table lists few systems for which RG 1.97 is applicable. Address the RG 1.97 for all categories and variables. (Table 7.1-2)

420.40

The HPCF pump is interlocked (7.3.1.1.1.1(3)(c)) with the undervoltage monitor. If the breaker cannot close will it retry and what information is available to the operator if it doesn't close that would indicate an undervoltage problem? (7.3.1.1.1.1)

420.60

Provide examples for section (g) which meet the design bases. (7.1.2.2)

420.61

Explain section (h) further. Does this mean one 480V bus, 4160 bus the generator? Same question at 7.2.3.2(2)(b). (7.1.2.2)

420.62

Provide justification for going to a 2/3 scram instead of 1/3 when one is bypassed. (7.1.2.10.11)

420.63

What are the realiability/availability goals for the reactor protection and engineered safety features systems? (7)

420.64

Describe the reliability model and assumptions used to demonstrate achievement of the reliability goals; this should include a description of the system architecture. (7)

420.65

What methodology is used in determining the system reliability/availability? (7)

420.66

Describe the data validation features in triplicated sensors. (7)

420.67

What testing will be done to demonstrate reliability? What is the specific scope of these tests? (7)

420.68

What is the effect upon the number of spurious trips generated by the RPS if the digital design replaces the previous analog design? Provide comparison. (7)

420.69

Are there any limitations on the ABWR design concerning the use of expert systems? Any limitations on the use of technology not specifically described? The original response does not describe an approach for determining what hardware or software developments which may ocur between design certification and plant operation can be implemented without changes to the design certification and NRC review. (7A)

420.70

Is there any system for in-service testing of the ARI? (7.1.2.1.6)

420.71

Is the CRD scram discharge high water level used as the example of the fifth test valid given that there is no scram discharge volume? (7.1.2.1.6)

420.72

Section (1) of 7.1.2.1.6 states that normal surveillance can identify failures. Discuss whether this system has the capability of transmitting this information to the plant computer so that an immediate alarm can be given in addition to waiting for the scheduled surveillance. (7.1.2.1.6)

420.73

Section (4) notes that the four divisions are tested in sequence. When the thirty minute sequence is complete does the test system start over again or is this an operator initiated test? (7.1.2.1.6)

420.74

Section (5) notes that only one division shall be bypassed at any one time. Describe the interlock protection or administrative controls which assure this. (7.1.2.1.6)

420.75

For section 7.1.2.2(j) clarify that the physical and electrical separation does not preclude the proper environmental qualification of redundant I&C equipment. (7.1.2.2)

420.76

For section 7.1.2.3.2(1)(c,d,e) and (2)(a) define "sufficient". (7.1.2.3.2)

420.77

One of the reasons stated for the utilization of microprocessors for the implementation of instrumentation and logic functions is that less uncertainty exists in the margins between actual safety limits and the limiting safety trips. The margins are stated to be set from experimental data on setpoint drift (see Section 7.1.2.1.4.1) and from quantitative reliability requirements for each system and its components.

Provide the documented bases for this procedure. (7.1.2.1.4.1)

420.78

One of the reasons stated for the utilization of microprocessors for the implementation of instrumentation and logic functions is that less uncertainty exists in the margins between actual safety limits and the limiting safety trips. The margins are stated to be set from experimental data on setpoint drift (see Section 7.1.2.1.4.1) and from quantitative reliability requirements for each system and its components.

Will this precedure be a topical report used as a design tool? (7.1.2.1.4.1)

420.79

One of the reasons stated for the utilization of microprocessors for the implementation of instrumentation and logic functions is that less uncertainty exists in the margins between actual safety limits and the limiting safety trips. The margins are stated to be set from experimental data on setpoint drift (see Section 7.1.2.1.4.1) and from quantitative reliability requirements for each system and its components.

What experimental data has been used to provide inputs to this design approach? (7.1.2.1.4.1)

420.113

Has consideration been given to providing the annunciators with backup diesel or battery power? (Ref. 7.1.2.6.1.1(2)(g)). (7.1.2.6.1.1)

420.114

The copy of Section 7 provided to the staff did not include Appendix 7A nor an indication that it was to be provided later. Provide this section or a schedule for providing it. (7A.1-1)

420.115

In the discussion about torque switches and thermal overloads, there is a reference to Section 3.8.4.2 which is the applicable codes and standards for seismic qualification of the Reactor and Control Buildings. What is the correct reference? (7.3.1.1.1.3(4)(e))

420.119

Are there any other valves which must isolate upon initiation of the SLCS? (7.4.1.2(7))

420.120

List all exemptions to the requirement rather than providing an example. (7.3.2.1.2(3)(c))

420.121

The first paragraph states that pipe break outside containment and feedwater line break are discussed below. The staff could not locate these items. (7.3.1.2(7))

420.122

Is the instrumentation required for the operator to verify bypass valve performance and relief valve operator 1E or N-1E? (15.2.2.2.1.4)

420.123

SSAR 15B.4 describes the essential multiplexing system (EMS) in some detail. SSAR Figure 7A.2-1 states that the design is not limited to this configuration. It is our understanding that the EMS design is still in a preliminary design stage. Is SSAR 15B.4 still accurate and is the design limited to that configuration? (15B.4)

420.124

The FMEA submitted in SSAR 15B.4 is inadequate for a safety evaluation supporting the design certification. The FMEA appears to the staff to be oversimplified with one line item each for component failures and does not address potential software complications. The staff requests clarification of how this FMEA was developed given that the system design has not been finalized. The staff also believes that software failures need to be evaluated. The failure modes investigated should include, as a minimum, stall, runaway, lockup, interruption/restoration, clock and timing faults, counter overflow, missing/corrupt date, and effects of hardware faults on software. (15B.4)

420.125

This section provided additional clarification of the intended use of the remote shutdown system. The degree of independence and isolation from the Safety System Logic and Control (SSLC) and EMS are not clear. Is it intended in the SSAR to take credit for the RSS if there is a total loss of EMS? (7.4.1.4)

420.126

Compared with GESSAR II, the ABWR has significantly reduced the number of input sensors by use of sharing sensors. Provide a bases to why this does not increase potential vulnerability to common mode failures by reducing sensor diversity. (7A-7)

420.127

In general, the applicant should provide a clear presentation of how the ABWR with common software and hardware modules for many functions (including SSLC logic self-test programs) conforms with IEEE 279-1971 and is at least as single failure proof as GESSAR II. The discussion of shared sensors in 7A-7 does not address potential common mode software failures which may be capable of defeating the diverse parameters. Additionally, the applicant should address why diversity of software should not be a requirement to maintain system diversity. (7)

420.128

Will software be used to isolate data? If so, what are the design and qualification criteria that are to be applied? Are there any systems which have non-Class 1E software such as keyboard or display control software that interface with the Class-1E systems? Are there any interface with the Class-1E systems which receive inputs from non-Class-1E systems or other channels of 1E systems. (7A.7)

420.129

List those systems or major components in the I&C design area for which the design is not complete to the "purchase specification" level. (7)

420.130

In response to Question 420.63, a MTBF goal of 100,000 hours (11.4 years) is given for the essential multiplexing system. Is this goal for one channel or the complete system? If this goal is for the complete system, it appears to the staff that the ABWR can expect to loose control at the control room of many of the safety systems (RPS, RHR, ADS) five or six times over the lifetime of the plant. How does this compare with the reliability/availability of multiple ESF systems in the BWR/5 & 6 design (or GESSAR II)?

420.131

Are multiplexer and software failures included in these systems interactions and common cause failures? (19.2.3.4)

420.132

Section 19.3.1.3.1 (b) states that "if core cooling is accomplished without the use of an RHR systems and the suppression pool cooling begins overheating, the suppression pool cooling mode of the RHR will be initiated by the operator." Is any manual action required prior to 30 minutes? (19.3.1.3.1 (b))(Response 420.47)

420.133

Subsection 19.3.1.3.1(c)(i) describes the MSIV closure sequence with the most desirable outcome requiring operator action at 30 seconds to insert rods. If that fails the operator must inhibit ADS valves from opening and initiate SLCS within 10 minutes. THese activities do not appear to be consistent with stated design goal of no operator action for 30 minutes following a transient. Provide a description of how the MSIV closure sequence meets the 30 minute rule (6.3.1.1.1) same question for loss of Offsite Power (LOOP).

420.134

Equipment maintenance or test unavailability are taken from GESSAR PRA and are based upon BWR experience. In the past, I&C has been a large contributor to system downtime. How do these systems (RHR, RCIC) unavailability numbers take into account the new multiplexing and microprocessors? (19D.3.4)

420.135

Provide the justification for Mean Time To Repair (MTTR) of 4 hours for multiplexers and 30 minutes for ESF logic. Inverters and battery chargers have restoration time given in (Table 19A.8) as 48-56 hours. Are the multiplexers designed with all test and maintenance equipment installed? (Table 19D.6-10)

420.136

The staff has reviewed the commitments in the SSAR and has reviewed the available documentation describing the verification and validation plans. To date, the information has been vague, general in nature and lacking in essential detail to demonstrate conformance with ANSI/IEEE 7-4.3.2. Does the applicant intend to enclose the V&V Plan as Appendix B of SSAR Chapter 7 or will the V&V details be left as an interface requirement? The staff requires a formal, structured V&V plan to be in place and implemented early in the software design process. (7A)

20.2.10 Chapter 10 Questions

281.15

In a letter from Thomas E. Murley, NNR, to Ricardo Artigas, G.E., dated August 7, 1987, the staff provided the ABWR licensing review bases as well as the scope and content of the ABWR Standard Safety Analysis Report (SSAR). In Section 8.7, Water Chemistry Guidelines, of the referenced letter, it states that GE has committed to using BWR Owners Group water chemistry guidelines. These guidelines are necessary to maintain proper water chemistry in BWR cooling systems to prevent intergrannular stress corrision cracking of austenitic stainless steel piping and components and to minimize corrision and erosion/corrision-induced pipe wall thinning in single-phase and two-phase high energy carbon steel piping. Water chemistry is also important for the minimization of plant radiation levels due to activated corrision products. Section 10.4.6.3 of the ABWR indicates that the condensate cleanup system complies with Regulatory Guide 1.56. Section 10.4 should indicate that the system meets the guidelines published in:

EPRI NP-4947-SR. BWR Hydrogen Water Chemistry Guidelines 1987 Revision, dated October 1988.

EPRI NP-5283-SR-A,. guidelines for Permanent BWR Hydrogen Water Chemistry-1987 Revision, dated September 1987.

The use of zinc injection as a means of controlling BWR radiation-field build-up should be discussed.

281.16

In Section 10.4.6.3, the ABWR SSAR indicates that the condensate cleanup system removes some radioactive material, activated corrision products and fission products that are carried over from the reactor. More important functions involve removal of condensate leakage to assure meeting BWR Hydrogen Water Chemistry Guidelines. This should be discussed.

281.17

The condensate (Figure 10.4.4) and feedwater (Figure 10.4.7) system diagrams do not indicate the location of the oxygen injection into the condensate system and hydrogen and zinc oxide into the feedwater system. This information should be provided.

281.18

Section 10.4 does not discuss design improvements involving material selection, water chemistry, steam temperatures, piping design and hydrodynamic conditions that are necessary to control erosion/corrision. The EPRI CHECMATE or other erosion/corrision computer codes mat be useful design tools to minimize wall thinning due to erosion/corrision-corrisionn. The ABWR SSAR should discuss design considerations to minimize erosion/corrision and procedures and administrative controls to assure that the structural integrity of single-phase and two-phase high-energy carbon steel piping systems is maintained.

430.59

Provide information on the following figures and tables: (10.1)

(a) Figure 10.1-2. Heat Balance for Guaranteed Reactor Rating

(b) Figure 10.1-3, Heat Balance for Valve-Wide-Open

(c) Table 10.1-1, Summary of Important Design Features and Performance Characteristics of the Steam and Power Conversion System, with regard to:

--- Condensate pumps: total head (ft) and motor hp.

--- Low pressure heaters: Stage pressure (psia) and duty per shell (Btu/hr) for Heaters Nos. 1, 2, 3, and 4.

--- High pressure heaters: Stage pressure (psia) and duty per shell (Btu/hr) for Heaters Nos. 5 and 6.

--- Low pressure turbine exhaust pressure to condenser

430.60

Specify the value for time "T" in Figure 10.2-2. (10.2)

430.61

Provide a description of the bulk hydrogen storage facility mentioned in Section 10.2.2.2. (10.2)

430.62

Provide a description of the speed control unit, the load control unit and the flow control unit of the electro-hydraulic control (EHC) system. Your description should include how they perform their intended functions. Clarify whether the EHC system will fully cut off steam at 103 percent of rated turbine speed. (10.2)

430.63

For turbine overspeed protection system (described in Section 10,2,2,4), the SSAR referred to redundant electrical trip signals. Provide information on the power source associated with each of the trip circuits. (10.2)

430.64

As presented in Section 10.2.2.4 of the ABWR SSAR, the closing time of the extraction nonreturn valves is less than 0.2 seconds, while it is 2 seconds at current BWR plants. Provide additional information on the design of these valves that supports the difference between the above closing time values. (10.2)

430.65

Clarify whether at least one main stop valve, reheat stop valve and reheat intercept valve will be inspected at approximately 3 1/3 years by dismantling them, and whether visual and surface examinations will be conducted for the valve seats, disks and stems (note that the above is an acceptance criterion for SRP Section 10.2). (10.2)

430.66

Identify preoperational and startup tests of the turbine generator in accordance with Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Power Plants," as an interface requirement. (10.2)

430.67

As stated in Section 10.3.2.1, "the four main steam lines are connected to a header upstream of the turbine stop valves...". However, according to Figure 10.3-2a, the main steam header is located downstream of the turbine stop valves. Identify whether the statement or figure is in error and revise the item in error so that the SSAR is consistent. (10.3)

430.68

Provide information on the leakage detection system for steam leakage from the MSSS in the event of a steam line break. Also provide information on the stated "safety feature designed into the MSSS" that will prevent radiation exposures in excess of the limits of 10 CFR Part 100 in the event of a break of a main steam line or any branch line (SSAR Section 10.3.3). (10.3)

430.69

For the following items identified in SSAR Figure 10.3-1: (10.3)

(a) Deaerating steam to condenser
(b) Offgas system
(c) Steam jet air ejectors
(d) Turbine gland sealing system
(e) Reheater
(f) Main steam bypass

Provide the following information:

(a) Maximum steam flow (lbs/hr)
(b) Type of shut-off valve(s)
(c) Size, quality, design code, closure time, actuation mechanism and associated motive power of the valve(s).

430.70

Provide information on the following items: (10.3)

(a) Analysis for steam hammer and relief valve discharge loads issues.
(b) Power source to the solenoid valves for the inboard and outboard main steam isolation valves.
(c) Location of seismic interface restraint (e.g., interface of which buildings?).
(d) Route which the main steam lines, including the branch lines, pass up to the turbine stop valves.
(e) Specific design features provided to protect safety-related portions of the main steam supply system, including the main steam isolation valves, against externally and internally generated missiles and adverse phenomena such as floods, hurricanes and tornadoes.

430.71

Describe provisions for operation of the main condenser with (10.4.1) leaking condenser tubes.

430.72

Describe the permissible cooling water inleakage rate and the allowed time of operation with inleakage. (10.4.1)

430.73

Provide information on the following items: (10.4.1)

(a) Provisions incorporated into the main condenser evacuation system component or tube failure due to steam blowdown from the turbine bypass system.

(b) Worst possible flood level in the applicable buildings due to complete failure of main condenser and provisions for protecting safety-related equipment located in the buildings against such flooding (note that ABWR SSAR Section 3.4 does not discuss the turbine building).

430.74

Discuss how the components of the main steam condenser evacuation system (MCES) conform to the guidelines of Regulatory Guide 1.26, 1.33 and 1.123 with respect to quality group classification and quality assurance programs.

430.75

Provide the design pressure and normal operational absolute pressure for the MCES components that could contain potentially explosive gas mixtures. (10.4.2)

430.76

Identify the radiation monitoring provisions for the mechanical vacuum pump exhaust. Is the exhaust filtered by charcoal adsorber and HEPA filters prior to release?

430.77

Identify the number, location and functions (i.e., recording and annunciating alarm) performed by the hydrogen analyzers. Clarify whether they can withstand a hydrogen detonation. (10.4.2)

430.78

Clarify whether the air ejectors are redundant in the sense that one of them is a standby. (10.4.2)

430.79

Identify the components and portions of the MCES that are designed to withstand a detonation in the system. (10.4.2)

430.80

Discuss how the design of the turbine gland sealing system (TGSS) conforms to the guidelines of Regulatory Guide 1.26 as it relates to the quality group classification for the system, and Regulatory Guide 1.33 and 1.123 as they relate to the quality assurance programs. (10.4.3)

430.81

Provide a description of the exhauster blower provided for the TGSS. (10.4.3)

430.82

ABWR SSAR Subsection 10.4.3.1.2 states that the TGSS exhausts the noncondensable gases to the turbine building equipment vent system; however, Subsection 10.4.3.3 states that the TGSS exhausts the noncondensable gas gases eventually to main vent. Clarify how the TGSS exhausts are monitored. Also, clarify whether the main vent mentioned above is the plant vent referred to in SSAR Section 11.5. (10.4.3)

430.83

What is the source for the auxiliary steam? Justify why an advanced design will use essentially radioactivity free auxiliary steam (see SSAR Section 10.4.3.2.2) as a backup sealing source rather than as normal sealing source. Note that the use of a process steam supply for sealing purpose can result in significant operational radioactivity releases. (10.4.3)

430.84

For turbine bypass system:

(a) Provide figures which delineate the system and its components.
(b) Clarify whether the system includes pressure-reducer assemblies for the bypass valves to reduce steam pressure prior to steam discharge into the condenser. (10.4.4)

430.85

For the circulating water system:

(a) Describe the function of the waterbox fill and drain subsystem mentioned in ABWR Subsection 10.4.5.2.1. Also, describe the "makeup water" shown in SSAR Figure 10.4-3.
(b) Provide the worst possible flood levels that can occur in the applicable plant buildings as a result of circulatory water system failure and indicate how safety-related equipment located in the buildings is protected against such flooding.

430.86

How is the remote manual motor-operated shutoff valve (gate valve F 2820) powered?

430.87

Describe the design features provided to protect the safety-related portion of the condensate and feedwater system from internally generated missiles.

430.88

Provide a summary of the analysis of a postulated high-enegy pipe break for the feedwater piping in the main steam tunnel including the design features provided (e.g., pipe whip restraints) for preventing adverse effects resulting from pipe whip, jet impingement and flooding.

430.89

Provide information on the analysis that shows that the entire feedwater system piping can accommodate water hammer events and the means to prevent water hammer loads due to hydraulic transients.

430.90

Provide detail information on the feedwater control valve and controller design, including the features that ensure the design will be stable and compatible with the system and imposed operating conditions.

SECTION 20.3

ILLUSTRATIONS (Continued)

Figure	Title	Page
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ILLUSTRATIONS (Continued)

Figure	Title	Page
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TABLE I

Comparison of requirements in ABWR standard safety analyses REPORT and ABWR presentation to NRC staff (October 21 and 22, 1987) (continued)

		ABWR Presentation to NRC Staff	ABWR Standard Safety Analysis Report
7 -	Highly corrosion-resistant condenser tubes to minimize leakage into condensate system	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.
8 -	Maintain electrochemical corrosion potential < 0.23 V to suppress IGSCC	Required Design Feature	Not listed in Table 5.2-5.
9 -	Erosion/corrosion-resistant materials in steam extraction and drain lines to minimize failures	Design Feature	Not discussed in Subsection 5.4.9.
10 -	Ease of leak detection in and repair of the main condenser	Design Feature	May be in Subsection 10.4.1 which has not been submitted yet.
11 -	2% Reactor water cleanup system to improve water quality and occupational radiation exposure	Design Feature	Not discussed in Subsection 5.2.3.2.2.
12 -	Full flow recirculation to main condenser from cleanup outlet to reduce feedwater impurities	Design Feature	Not discussed in Subsection 5.2.3.2.2.3.

RESPONSE 281.10

Item 1

Response to Item 1 of this question is provided in revised Subsection 5.2.3.2.2.2.

Item 2

Response to Item 2 of this question is provided in revised Subsection 5.2.3.2.2.

Item 3

The system for adding zinc to the feedwater is discussed in new Section 9.3.11.

Item 4

The system which includes a full flow deep bed condensate treatment system is discussed in revised Subsection 5.2.3.2.2.3 and new Subsection 10.4.6.

Item 5

New and improved water quality monitoring instrumentation is being constantly developed and introduced for use in BWR plants. Several useful instruments have been developed and introduced within the past few years. GE will evaluate the state of the art when a BWR is undergoing detailed design and will incorporate such instruments that are necessary to assure proper water quality.

Item 6

Response to Item 6 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 7

Response to Item 7 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 8

Response to Item 8 of this question is provided in revised Subsection 5.2.3.2.2.2 and Table 5.2-5.

Item 9

Response to Item 9 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 10

Response to Item 10 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 11

Response to Item 11 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 12

The condensate treatment system which includes full flow recirculation to main condenser from condensate cleanup outlet is discussed in revised Subsection 10.4.6.2.

QUESTION 470.1

Subsection 15.6.2 of the ABWR FSAR provides your analysis for the radiological consequences of a failure of small lines carrying primary coolant outside of containment. This analysis only considers the failure of an instrument line with a 1/4-inch flow restricting orifice. Show that this failure scenario provides the most severe radioactive releases of any postulated failure of a small line. Your evaluation should include lines that meet GDC 55 as well as small lines exempt from GDC 55.

RESPONSE 470.1

The analysis for failure of a small line carrying primary coolant was conservatively analyzed as a failure of an instrument line with full flow for a period of two hours. This analysis is deemed conservative for the reason given below.

Of all the lines carrying coolant penetrating the primary containment wall, only the instrument lines are exempt from GDC 55. All other lines use some form of check valve/motor-operated valve combination to stop the flow of primary coolant in the event of a line break. Typically, the motor-operated valves close at the rate of two inches per ten seconds. Considering a two-inch line and assuming that a flow of 175 pounds per second would result in operator action within 60 seconds, the total mass released over the 70 second period would be approximately 12,000 pounds or about one half of the assumed release over two hours from the instrument line. Using this logic and these simplified calculations, it is found that a two-hour instrument line break bounds releases for small lines.

QUESTION 470.2

Provide a justification for your assumption that the plant continues to operate (and therefore no iodine peaking is experienced) during a small line break outside containment (Subsection 15.6.2) accident scenario. Also provide the basis for the assumption that the release duration is only two hours.

RESPONSE 470.2

The analysis for failure of a small line carrying primary coolant was based upon considering the plant remaining at full power for a period of two hours at which time flow was stopped. For conservative purposes, the release was considered instantaneous in the actual computations. These parameters were chosen for conservatism and ease of computation. The actual case of the rupture of an instrument line is described in Chapter 8 of NEDO-21143-1 (Reference 2 of SSAR Subsection 15.6.7) and results in full flow for approximately ten minutes following operator action and gradual depressurization over a five-hour period. The total mass of liquid released is approximately 12,000 pounds or one-half of the assumed release analysis. In addition, iodine spiking is considered on a release per fuel bundle basis. With the spiking term, which is estimated as a 15% initial release following release of the remaining 85% proportional to the depressurization, it is found that the results are similar to those analyzed in Section 15.6 but slightly less conservative.

QUESTION 470.3

Subsection 15.6.4.5.1.1 of the SSAR gives the iodine source term (concentration and isotopic mix) used to analyze the steam-line-break-outside-of-containment accident. The noble gas source term, however, is not addressed. Provide the noble gas source term used. Also the table in Subsection 15.6.4.5.1.1 seems heavily weighted to the shorter lived activities (i.e., I-134). Provide the bases for the isotopic mix used in your analysis (iodine and noble gas).

RESPONSE 470.3

Subsection 15.6.4.5.1.1 states that for case 1 the noble gas source term used was equivalent to an offgas release of 50,000 microCuries per second and 300,000 microCuries per second for case 2. In both cases, the source term is referenced to a 30-minute decay time. The isotopic distribution for such source terms are relatively standard throughout the industry and can be found in Table 2-2 of NUREG-0016. For the iodine isotopes the concentrations are technical specification limits of 0.2 microCuries per gram (case 1) and 4 microCuries per gram (case 2) dose equivalent to I-131. The isotopic breakdown is based upon evaluations of BWR iodine chemistry in the early 1970's and is given in Reference 2 of SSAR Subsection 15.6.7. The breakdown is as follows, and is similar to that found in Table 2-2 of NUREG-0016:

I-131	0.073
I-132	0.71
I-133	0.5
I-134	1.4
I-135	0.73

QUESTION 470.4

Subsection 15.6.5.5 states that the analysis is based on assumptions provided in Regulatory Guide 1.3 except where noted. For all assumptions (e.g., release assumed to occur one hour after accident initiation, the chemical species fractions for iodine, the temporal decrease in primary containment leakage rates, credit for condenser leakage rates, and dose conversion factors) which deviate from NRC guidance such as regulatory guides and ICRP2, provide a detailed description of the justification for the deviation or a reference to another section of the SSAR where the deviations are discussed in detail. Provide a comparison of the dose estimates using these assumptions versus those which result from using the NRC guidance.

QUESTION 430.44

Identify the system lines whose containment isolation requirements are covered by GDC 57 and discuss conformance of the design to the GDC requirements. (6.2)

RESPONSE 430.44

GDC 57 addresses closed loop systems which penetrate the containment but do not communicate with the containment interior. The system lines shown in Table 20.3-5 have been identified and are considered to conform to GDC57 with the valve configuration as shown. The heavy lines denote an extension of the containment boundry.

QUESTION 430.45

For the combustible gas control systems design, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements. (6.2)

RESPONSE 430.45

The combustible gas control systems, consisting of flammability control system (FCS-T49) and atmospheric control system (ACS-T31), are completely within the scope covered by the ABWR SSAR. As such, there are no interfaces with equipment or systems outside the scope of this submittal. Interfaces with systems or equipment within the scope of the SSAR are discussed, as necessary in Subsection 6.2.5.

QUESTION 430.46

According to SRP 6.2.5 specific acceptance criteria related to the concentration of hydrogen or oxygen in the containment atmosphere among others are the following:

(a) The analysis of hydrogen and oxygen production should be based on the parameters listed in Table 1 of Regulatory Guide 1.7 for the purpose of establishing the design basis for combustible control systems.

(b) The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis should be equal to or more conservative than the decay energy model given in Branch Technical Position ASB9-2 in SRP 9.2.5.

Provide justification that the assumptions used in the ABWR in establishing the design basis for the combustible gas control systems are conservative with respect to the criteria a. and b. above. (6.2)

RESPONSE 430.46

The analysis of hydrogen and oxygen production for ABWR combustible gas control design is based on the parameters listed in Table 1 of Regulatory Guide 1.7. The fission product decay energy model used is that presented in SRP 9.2.5 Branch Technical Position ASB 9.2. Therefore, the ABWR design basis for combustible gas control system is conservative and appropriate for design.

QUESTION 430.47

Provide an analysis of the production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident including <u>all</u> applicable information specified in Section 6.2.5.3 of Regulatory Guide 1.70, Revision 3.

RESPONSE 430.47

Figure 6.2-41 provides an analysis of the oxygen and hydrogen concentrations in the primary containment after the design basis accident. Inputs to the analysis are provided in a revised Subsection 6.2.5.3.

QUESTION 430.48

Regarding Containment Type A leakage testing, (6.2.6)

QUESTION 430.48a

Provide the values for P_a and P_t.

RESPONSE 430.48a

P_a approximately 40 psig and $0.5\ P_a < P_t < P_a$.

QUESTION 430.48b

Include the acceptance criterion for L_t during preoperational leakage rate tests, i.e., L_t - L_a (L_{tm}/L_{am}), for the case when L_a (L_{tm}/L_{am}) = 0.7.

RESPONSE 430.48b

Response to this question is provided in revised Subsection 6.2.6.1.1.5.

QUESTION 430.48c

Your acceptance criterion for L_{tm} (SSAR Subsection 6.2.5.1.2.2, Item 1) is at variance with the staff's current practice for acceptance of L_{tm}. Also, it does not comply with the 10 CFR Part 50, Appendix J, Section III, Item A.1.(a) requirement. Therefore, either provide sufficient supporting justification for the exemption from compliance with the above requirement or correct the criterion as appropriate to comply with the requirement. Also, correct the stated acceptance criterion (SSAR Subsection 6.2.6.1.2.2, Item 3) as appropriate to comply with Appendix J, Section III, Item A.6.(b) requirement.

RESPONSE 430.48c

Response to this question is provided in revised Subsection 6.2.6.1.2.2.

QUESTION 430.48d and 430.48e

Regarding ILRT, identify the systems that will not be vented or drained and provide reasons for the same.

RESPONSE 430.48d and 430.48e

Table 3.6-1 lists essential systems that become available to shut down the reactor and mitigate the consequences of a postulated piping failure to acceptable limits. P&IDs and process flow diagrams for these systems are contained in their respective chapters. With regrad to integrated leak rate test (ILRT), the criteria described in Table 3.6-7 is applied to determine which systems will or will not be vented or drained. Provisions for venting/draining affected systems are shown on the F&IDs listed in Table 3.6-7.

QUESTION 430.49

Regarding Type B test, (6.2.6)

QUESTION 430.49a

Clarify how air locks opened during periods when containment integrity is required by plant's Technical Specifications will be tested to comply with Appendix J, Section III, Item D.2.(b).(iii).

RESPONSE 430.49a

Response to this question is provided in revised Subsection 6.2.6.2.3.

QUESTION 430.49b

Provide the frequency for periodic tests of air locks and associated inflatable seals.

RESPONSE 430.49b

Response to this question is provided by revised Subsection 6.2.6.2.3.

QUESTION 430.49c

Provide the acceptance criteria for lock testing and the associated inflatable seal testing.

RESPONSE 430.49c

Response to this question is provided in revised Subsection 6.2.6.2.2.

QUESTION 430.49d

List all containment penetrations subject to Type B tests.

RESPONSE 430.49d

Response to this question is provided in revised Subsection 6.2.6.2.1 and new Table 6.2-8.

QUESTION 430.49e

List all those penetrations to be excluded from Type B testing and the rationale for excluding them.

RESPONSE 430.49e

Response to this question is provided in revised Subsection 6.2.6.2.1 and Table 6.2-8.

QUESTION 430.50

Regarding Type C tests (6.2.6)

QUESTION 430.50a

Correct the statement (Subsection 6.2.6.3.1, Paragraph 1) as appropriate to ensure that the hydraulic Type C tests are performed only on those isolation valves that are qualified for such tests per Appendix J. The current statement implies that these tests are not necessarily restricted to the valves that qualify for such tests.

QUESTION 430.54o

Section 6.4.7.1, External Temperature, "provides design maximum external temperature of 100°F and -10°F. How are these values used in the design and assessments related to the ABWR? What factors, such as insulation, heat generation from control room personnel and equipment and heat losses, are taken into account? Do these values represent "instantaneous" values or are they temporal and/or spatial averages?

RESPONSE 430.54o

These values represent the summer maximum dry bulb air temperature. They are used in sizing the HVAC essential chilled water system chillers and the control room HVAC system.

QUESTION 430.54p

Clarify your position on potential hazardous or toxic gas sources onsite of an ABWR. If applicable, indicate the special features provided in the ABWR design in this regard, to ensure control room habitability.

RESPONSE 430.54p

Response to this question is provided in revised Subsection 6.4.7.3.

QUESTION 430.54q

Identify all the interface requirements for control room habitability systems (e.g., instrumentation for protection against toxic gases in general and chlorine in particular; potential toxic gas release points in the environs).

RESPONSE 430.54q

The ABWR control room habitability system has no interface requirements.

QUESTION 430.55

Regarding ESF Atmosphere Cleanup Systems, (6.5.1)

QUESTION 430.55a

Provide a table listing the compliance status of the Standby Gas Treatment System (SGTS) with each of the regulatory positions specified under C of RG 1.52. Provide justifications for each of those items that do not fully comply with the corresponding requirements. In this context, you may note that the lack of redundancy of the SGTS filter train (the staff considers that filter trains are also active components - See SRP 6.4, Acceptance Criterion II.2.b) is not acceptable. Further, the described sizing of the charcoal adsorbers based on assumed decontamination factors for various chemical forms of iodine in the suppression pool is not acceptable (RG 1.3 assumes a decontamination factor of 1 for all forms of iodine and RG 1.52 requires compliance with the above guide for the design of the adsorber section). Therefore, revise charcoal weight and charcoal iodine loading given in SSAR Table 6.5-1 as appropriate.

RESPONSE 55a

The response to the first part of this question is provided in Appendix 6A. Justification of single filter train is provided in revised Subsection 6.5.1.3.3. The iodine source term is discussed in Subsections 6.5.1.3.3 and 6.5.1.3.4. Tables 6.5-1 and 6.5-2 have been revised.

QUESTION 430.55b

Specify the laboratory test criteria for methyl iodine penetration that will be identified as an interface requirement to be qualified for the adsorber efficiencies for iodine given in SSAR Table 15.6-8. Also, provide the depth of the charcoal beds for the control room emergency system.

RESPONSE 430.55b

The response to the first part of this question is provided in Appendix 6A which assesses compliance against Regulatory Guide 1.52, Positions C.3.i, C.6.a (2) and C.6.a(3).

Control room HVAC charcoal bed depth is discussed in Subsection 9.4.1.1.3.

QUESTION 430.55c

Provide a table listing the compliance status of the instrumentation provided for the SGTS for read out, recording and alarm provisions in the control room with each of the instrumentation items identified in Table 6.5.1-1 of SRP 6.5.1. For partial or non-compliance items, provide justifications.

RESPONSE 430.55c

The response to this question is provided in Appendix 6B.

QUESTION 430.55d

Clarify whether primary containment purging during normal plant operation when required to limit the discharge of contaminants to the environment will always be through the SGTS (See SSAR Section 6.5.1.2.3.3). Clarify whether such a release prior to the purge system isolation has been considered in the LOCA dose analysis.

RESPONSE 430.55d

The response to this question is provided in Subsection 6.5.1.3.6. Note that Subsection 6.5.1.2.3.3 has been renumbered to 6.5.1.2.3.2.

QUESTION 430.55e

Provide the compliance status tables referred to in Items (a) and (c) above for the control room ESF filter trains. (The staff notes that you have committed to discuss control room ESF filter system under Section 9.4.1. However, since evaluation of the control room habitability system cannot be completed until the information identified above is provided, the above information is requested now.)

RESPONSE 430.55e

The response to this question is provided in Subsections 9.4.1.1.6.2 and 9.4.1.1.6.3.

QUESTION 430.55f

Identify the applicable interface requirements for the SGTS and the control room ESF atmosphere cleanup system.

RESPONSE 430.55f

Both the SGTS and the control room ESF atmosphere cleanup system are completely within the scope covered by the ABWR SSAR. As such, there are no interfaces with equipment or systems outside the scope of this submittal. Interfaces with systems or equipment within the scope of the SSAR are discussed, as necessary, in Section 6.4 and Subsections 6.5.1 and 9.4.1.1. Testing requirements are described in the standards and Regulatory Guides referenced in these sections.

QUESTION 430.56

Regarding Fission Product Control Systems and Structures, (6.5.3)

QUESTION 430.56a

Provide the drawdown time for achieving a negative pressure of 0.25 inch water gauge for the secondary containment with respect to the environs during SGTS operation. Clarify whether the unfiltered release of radioactivity to the environs during this time for postulated LOCA has been considered in the LOCA dose analysis. (Note that the unfiltered release need not be considered provided the required negative pressure differential is achieved within 60 seconds from the time of the accident).

RESPONSE 430.56a

The response to this question is provided in Subsection 6.5.1.3.2.

QUESTION 430.56b

Provide justification (See SRP Section 6.5.3, II.4) for the decontamination factor assumed in SSAR Tables 6.5-2 and 15.6-8 for iodine in the suppression pool, correct the elemental, particulate and organic iodine fractions given in tables to be consistent with RG 1.3, and incorporate the correction in the LOCA analysis tables. Alternatively, taking no credit for iodine retention in the suppression pool, revise the LOCA analysis tables. Note that the revision of the LOCA analysis tables (this also includes the control room doses) mentioned above is strictly in relation to the iodine retention factor in the suppression pool (also, there may be need for revision of other parameters (s) given in the tables and these will be identified under the relevant SRP Sections questions).

RESPONSE 430.56b

Table 6.5-2 has been revised to be consistent with the assumptions of Regulatory Guide 1.3

With regard to Table 15.6-8, the LOCA analysis is performed in accordance with paragraph 8.9 of the Licensing Review Bases document. An evaluation of suppression pool scrubbing using the MAAP3B code for LOCA conditions shows a scrubbing factor of 600 to 1000 (Subsection 19E.2.1). Therefore the use of a scrubbing factor of 100 is sufficiently conservative. The variance between the current calculations and the prior evaluation methodologies is found in Table 20.3-1.

QUESTION 430.55c

Identify the applicable interface requirements

RESPONSE 430.56c

The response to this question can be found in the response to Question 430.55f.

20.3.5 Response to Fifth RAI-Reference 5

QUESTION 210.3

In Subsection 3.1.2.1.1.2, "Evaluation Against Criterion 1", a footnote states that "important-to-safety" and "safety-related" are considered equivalent in this SSAR. The staff does not agree with this definition. The staff's position on this issue remains as stated in NRC Generic Letter 84-01, "NRC Use of the Terms "Important to Safety" and "Safety-Related", dated January 5,1984. The staff used this position as guidance in its reviews of applications for operating licenses of nuclear power plants for a number of years prior to the issuance of GL 84-01. During these reviews, the staffs' evaluations of the quality assurance requirements in 10 CFR Part 50, Appendix B generally applied to the narrower class of "Safety-related" equipment as defined in 10 CFR Part 50.49(b)(1). 10 CFR Part 100, Appendix A and in Section 3.2 of this SSAR. This implied that normal industry practice for quality assurance was generally acceptable for most equipment not covered by the "safety-related" definition. However, as pointed out in Generic Letter 84-01, there have been specific situations in the past where the staff has determined that quality assurance requirements beyond normal industry practice were needed for components and equipment in the more broad "important to safety" class.

It is the staff's opinion that a strict interpretation of the ABWR position on this issue could result in an unacceptable classification of structures, systems and components for Table 3.2-1 in this SSAR >

Revise the footnote in Subsection 3.1.2.1.1.2 and the discussion in Section 3.2 to be consistent with the staff's position as stated in Generic Letter 84-01. It should be made clear that the staff's position will not result in a broadening of the staff's review. Rather, it provides the basis which the staff has been using and continues to use as guidance in its reviews of Quality Group Classification for certain components and equipment which are not included in the "safety-related" definition.

RESPONSE 210.3

Subsection 3.1.2.1.1.2 has been revised and the footnote deleted. The ABWR is consistent with the definition of important to safety as defined in 10CFR50, Appendix A, General Design Criteria, as specified in the Nuclear Regulatory Commission's Memorandum and Order dated June 6, 1984 (CLI-84-9). The Commission identified the definition of important to safety as a generic issue to be resolved by rule making. The Commission also stated that in the interim, past practice should be followed.

The past practice for all GE Nuclear Energy BWR design has been that all equipment has been identified as either safety-related or non-safety-related. For certain non-safety-related equipment, the pertinent requirements (including quality assurance requirements) have been specified on a case-by-case basis commensurate with the functional importance of the equipment (e.g., fire protection and radioactive waste treatment systems). All prior GE BWR designs have been licensed on that basis. (This includes Shoreham for which this issue was specifically addressed and resulted in the Commission's June 6, 1984 Memorandum and Order.) The ABWR is consistent with this prior practice as specified by the Commission's Memorandum and order.

QUESTION 210.4

In Subsection 3.2.3 "Safety Classifications", ANSI/ANS 52.1-1983, "Nuclear Safety Criteria for the Design of Stationary BWR Plants" is referenced for the definitions of safety classes. The guidance in this document for components which are not within the scope of Regulatory Guide 1.26 has not been endorsed by the staff. Therefore, the staff does not completely accept ANSI/ANS 52.1 for

the definitions of all safety classes. Questions 210.5, 210.13, 210.15, 210.17, 210.44, and 210.45 are based on this position. To assure that Table 3.2-1 will be consistent with similar tables in recently licensed BWR/6 plants, such as Perry and River Bend, the reference to ANSI/ANS 52.1-1983 should be either eliminated or revised.

RESPONSE 210.4

The safety classification methodology used to classify the ABWR equipment is the same as that used for previous GE BWR designs. The safety class definitions for ABWR while based on AISI/ANS 52.1 - 1983 are consistent with this classification methodology and past designs. ANS adopted the GE BWR safety classification criteria and expanded them into LWR classification criteria. The ANS LWR classification criteria were reviewed by the NRC staff during development and resolutions for all substantive comments were developed at a meeting between the NRC staff and representatives of ANS on October 12, 1982. AISI/ANS 52.1 - 1983 incorporated those resolutions. In addition, AISI/ANS 52.1 - 1983 is specified by the EPRI ALWR Requirements Program.

QUESTION 210.5

In Table 3.2-1, Items B1.7, "Control Rods" and B1.9, "Fuel Assemblies" are classified as Safety Class 3, which is consistent with the criteria in the ANSI/ANS 52.1 - 1983 Standard. As stated in Question 210.4, the staff does not agree with all of the recommendations in that Standard. The staff position is that Control Rods and Fuel Assemblies should be Safety Class 2 and Quality Group B. To be consistent with this position and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classifications of the Control Rods and Fuel Assemblies from Safety Class 3 to 2 and add Quality Group B.

Questions 210.44 and 210.45 provides similar staff positions for Item B1.5 Safety-Related Reactor Internal Structures and Core Support Structures.

RESPONSE 210.5

The safety requirements for non-piping components such as control rods and fuel assemblies are the same whether they are designated Safety Class 2 or Safety Class 3. The Safety Class 3 designation for such items is based on a comprehensive, systematic rational. Designating the items Safety Class 2 instead of Safety Class 3 would not change the safety requirements applied to them but would cause inconsistencies with the rational. To maintain consistency, the Safety Class 3 designation will be retained.

QUESTION 210.6

In Table 3.2.1, Item B2.5 identifies Main Steam Line (MSL) piping from the outermost isolation valve to and including the seismic interface restraint as being Safety Class 1 and Quality Group A. Figure 5.1-3b, "Nuclear Boiler System P&ID, Sheet 2" identifies the same portion of the MSL as Quality Group B. Beyond the seismic interface restraint, the MSL piping is quality Group D, which is not acceptable to the staff. To be acceptable, the MSL should be classified as recommended in Standard Review Plant 3.2.2, "System Quality Group Classification", Appendix A, i.e., Quality Group B from the outermost isolation valve to the turbine stop valve. This staff position is based on the assumption that the ABWR MSL design differs from the BWR/6 design in that it does not contain a shutoff valve in addition to the two containment isolation valves. Revise Table 5.1-3b, Table 3.2-1, Subsection 3.9.3.1.3 and Subsection 5.4.9.3 to be consistent with the above staff position.

RESPONSE 210.6

Table 3.2-1, Subsection 3.9.3.1.3 and Subsection 5.4.9.3 are corrected. The MSL piping beyond the seismic interface restraint will be changed from Quality Group D to Quality Group B at the next revision of Figure 5.1-3b.

QUESTION 210.7

Item B2.5 in Table 3.2-1 does not appear to agree with Figure 5.1-3c, "Nuclear Boiler System P&ID, Sheet 5". Item B2.5 states that piping in the Feedwater (FW) Systems from the outermost isolation valve to and including the seismic interface restraint is Safety Class 1 and Quality Group A. Figure 5.1-3c shows the FW line as Quality Group A up to the first spring closing check valve outside containment (F262A). The FW piping is Quality Group B between valves F262A and F282A and Quality Group D beyond F262A. There does not appear to be a seismic restraint in Figure 5.1-3c. Assuming that the ABWR FW line is similar to the BWR/6 designs, i.e., valve F282A is a shutoff valve in addition to the two containment isolation valves, the Quality Group classification of this line does not appear to be consistent with the guidelines of Standard Review Plan 3.2.2, Appendix B. Revise Table 3.2-1, Figure 5.1-3c and Subsection 5.4.9.3 to be consistent with the staff position on Quality Group in SRP 3.2.2, Appendix B. The transition from Quality Group B to D should be at the seismic interface restraint rather than shutoff valve F282A.

RESPONSE 210.7

Table 3.2-1 and Subsection 5.4.9.3 have been corrected. Figure 5.1-3c will be revised accordingly at its next revision as indicated in Figure 20.3-20.

QUESTION 210.8

In Table 3.2-1, Item B3.1, the primary side recirculating motor cooling system piping is classified as Safety Class 3 and Quality Group C. In Subsection 3.9.3.1.4, this piping is described as being designed to the ASME Code, Section III, Subsection NB 3600, which is comparable to Safety Class 1. In Figure 5.4-4, "Reactor Recirculation System P&ID", this piping is identified as Quality Group A. The staff's position is that this piping should be, as a minimum, Safety Class 1, Quality Group A and meet the requirements of 10 CFR 50, Appendix B from the interface of the piping with the pump motor casing to and including the first pipe support. The remainder of this piping, should be the same Safety Class as the supported piping. Revise Items B3.1 and B3.2 in Table 3.2-1 to be consistent with the staff position.

RESPONSE 210.8

The recirculation motor control system (RMCS) is classified Quality Group C and Safety Class 3 which is in accordance with the requirements of 10CFR50.55a. The RMCS, which is part of the reactor coolant pressure boundary (RCPB), meets 10CFR50.55a(c)(2). Postulated failure of the RMCS piping cannot cause a loss of reactor coolant in excess of normal makeup (CRD return or RCIC flow), and the RMCS is not an engineered safety feature. Thus, in the event of a postulated failure of the RMCS piping during normal operation, the reactor can be shutdown and cooled down in an orderly manner, and reactor coolant makeup can be provided by a normal make up system (e.g., CRD return or RCIC system). Thus, per 10CFR50.55a(c)(2), the RMCS need not be classified Quality Group A or Safety Class 1. Since the RMCS is not an engineered safety feature (e.g., it does not provide emergency reactivity control, emergency core coolant, or primary reactor containment), the system need not be classified Quality Group B or Safety Class 2. The RMCS is classified Quality Group C and Safety Class 3, however, the system is designed and fabricated in accordance with AMSE Boiler and Pressure Vessel Code, Section III, Class 1 criteria as specified in Subsection 3.9.3.1.4 and Figure 5.4-14. A typographical error in Table 3.2-1, Item B3.1, "Piping-primary side, motor cooling", has been corrected.

QUESTION 210.9

In Table 3.2-1, and the classification summary for the Control Rod Drive Mechanism and the Low Pressure Core Flooder System or provide a justification for not including this information. The staff position on the Safety Class of these systems is as stated in Question s 210.5 and 210.45.

RESPONSE 210.9

The classification summary for the control rod drive mechanism has been added to Table 3.2-1. Portions of the control rod drive mechanism that are part of the reactor coolant pressure boundary are Safety Class 1. All other portions of the control rod drive mechanism are Safety Class 3 (see response to Question 210.5). The low pressure core flooder system is a subsystem of the residual

heat removal system and its classification summary is included in Items E1.1 through E1.15 of Table 3.2-1.

QUESTION 210.10

Provide the basis for all Control Rod Drive System valves (Item C1.1 in Table 3.2-1) to be classified as Non-Nuclear Safety and Non-Seismic.

RESPONSE 210.10

Item C1.1 of Table 3.2-1 has been clarified. All valves required to provide the scram function are part of the hydraulic control unit which is Safety Class 2. The hydraulic control unit is Item C1.4 in Table 3.2-1. All other valves do not perform a safety-related function and are Non-Nuclear Safety.

QUESTION 210.11

Provide the basis for portions of piping systems with the outermost isolation valves in the Residual Heat Removal System and the Reactor Core Isolation Cooling System (Items E1.3, E4.1, and E4.6 in Table 3.2-1) to be classified as Safety Class 2 and 3.

RESPONSE 210.11

Portions of piping of the residual heat removal system, high pressure core flooder system, and reactor core isolation cooling system within the outermost isolation valve which are not part of the reactor coolant pressure boundary but are part of an engineered safety feature are Safety Class 2. Examples are the suppression pool suction piping and containment spray piping. Portions of piping of the residual heat removal system, high pressure core flooder system, and the reactor core isolation cooling system which are part of the reactor coolant pressure boundary but are one (1) inch or less in diameter are Safety Class 2. Examples are instrument lines and drain lines. Items B1.5, E1.3, E2.1, E2.2 and E2.5 of Table 3.2-1 have been clarified; Figures 5.4-10a and 6.2-38a will be corrected at their next revision (as indicated in Figures 20.3-20.1 and 20.3-20.2 and Subsections 9.5.1.2.4, 3.9.5.1.2.5, and 3.9.5.1.2.6 have been corrected.

QUESTION 210.12

Items E2.1 and E2.5 in Table 3.2-1 classifies some pumps and valves within the outermost isolation valves in the High Pressure Core Flooder System as Safety Class 2. Provide the basis for this classification.

RESPONSE 210.12

See response to Question 210.11.

QUESTION 210.13

In Table 3.2-1, Item F4.1, "Refueling Equipment Platform Assembly" is classified as Non-Nuclear Safety. To be consistent with the staff position as stated in Question 210.4 and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change this classification to Safety Class 2 and Quality Group B.

RESPONSE 210.13

Item F4.1 of Table 3.2-1, Subsection 9.1.4.1, and Table 9.1-2 have been revised to show that a quality assurance program will be applied to ensure that the design, construction and testing requirements for the refueling equipment platform assembly are met. This quality assurance commitment is similar to the commitments made for other nonsafety-related equipment such as fire protection equipment, radwaste equipment and ATWS equipment.

The Non-Nuclear Safety designation for the refueling equipment platform assembly is consistent with ANSI/ANS-57.1-1980 (that has been endorsed by SRP Section 9.1.1) ANSI/ANS-52.1-1983, the EPRI ALWR Requirements Program and past industry practice. Also, in accordance with past industry practice, the assembly is Seismic Category I to prevent catastrophic collapse onto the reactor core during a seismic event. The consequences of failure of this assembly are within acceptable limits for such an event.

QUESTION 210.14

If a Fuel Transfer System or Tube is applicable to the ABWR, add the Classification Summary for this system under Item F4, "Refueling Equipment" of Table 3.2-1.

RESPONSE 210.14

The ABWR design does not include a fuel transfer system or tube. The refueling arrangement and process for the ABWR is the same as for BWR/3, BWR/4, and BWR/5. The spent fuel storage pool is at the refueling floor level. During refueling the reactor vessel is flooded up to the spent fuel storage pool level, a gate in the spent fuel storage pool is removed connecting it to the flooded reactor vessel, and fuel is transferred underwater via the refueling platform.

QUESTION 210.15

In Table 3.2-1, Items F5.1, "Fuel Storage Racks - New and Spent" and F5.2, "Defective Fuel Storage Container" are classified as Non-Nuclear Safety. Item F5.2 is also classified as Non-Seismic. To be consistent with the staff position as stated in Question 210.4 and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classification of Items F5.1 and F5.2 to Safety Class 3 and Quality Group C. In addition, change the seismic classification of Item F5.2 to Seismic Category I and add "B" in the Quality Assurance column for F5.2.

RESPONSE 210.15

Items F5.1 and F5.2 of Table 3.2-1 and Subsection 9.1.2.1.3 have been revised to show that a quality assurance program will be applied to ensure that the design, construction and testing requirements are met. Since the equipment is not required to prevent or mitigate a design basis event, 10CFR50 Appendix B quality assurance requirements are not required.

The Non-Nuclear Safety designation for fuel storage racks (new and spent) and the defective fuel storage container is consistent with AISI/ANS-57.1-1980 and ANS-57.2/ANSIN210-1976 (that have been endorsed by SPP Sections 9.1.1 and 9.1.2), AISI/ANS-52.1 - 1983, the EPRI ALWR Requirements Program, and past industry practice. The fuel storage racks are Seismic Category I commensurate with their functional importance. The consequences of a credible failure of the racks and container are within acceptable dose limits.

QUESTION 210.16

In Table 3.2-1, the following components in the Reactor Water Cleanup System are correctly classified as Quality Group C, but are also classified as Non-Nuclear Safety:

G1.1 - Vessels
G1.2 - Regenerative Heat Exchanges
G1.3 - Cleanup Recirculation Pump
G1.5 - Pump suction and discharge piping beyond containment isolation valves
G1.8 - Non-regenerative heat exchanger tube inside and piping and valves carrying process water
G1.11 - Filter demineralizer holding pumps, valves and piping

To be consistent with the discussions in Subsections 3.2.2 and 3.2.3 and with the information in Tables 3.2-1 and 3.2-3, the staff is of the opinion that all of the above components should be classified as Safety Class 3 in addition to Quality Group C. Revise Table 3.2-1, Items G1l1, G1.2, G1.3, G1.5, G1.8, and G1.11 to change the Safety Class from "N" to "3" or provide a justification for not doing so.

RESPONSE 210.16

The piping portions of the reactor water cleanup system out to and including the outermost isolation valves are Safety Class 1. In accordance with past practice, the portions of the reactor water cleanup system beyond the outermost isolation valves are Quality Group C, Non-Seismic, Category I, and Non-Nuclear Safety. These latter portions are not part of the reactor coolant pressure boundary or primary reactor containment and do not perform a safety-related function. The consequences of failure of those latter portions are within acceptable dose limits. The classification summary for electrical equipment required for isolation has been added to Table 3.2-1.

QUESTION 210.17

In Table 3.2-1, Items G2.3, "Heat Exchangers", G2.4, "Pumps and Pump Motors", G2.5, "Piping, Valves", and G2.7, "RHR Connections" in the Fuel Pool Cooling and Cleanup System are all classified as Non-Nuclear Safety, which is consistent with the criteria in the ANSI/ANS 52.1 - 1983 Standard. As stated in Question 210.4, the staff does not agree with all of the recommendations in that Standard. The staff position is that all of the above items should be Safety Class 3, Seismic Category 1 and listed under Quality Assurance requirements of 10 CFR 50, Appendix B. Regulatory Positions C.2 in Regulatory Guide 1.26 and C.1 in Regulatory Guide 1.29 includes this position. To be consistent with this position and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classification of Items G2.3, G2.4, G2.5, and G2.7 from Non-Nuclear Safety to Safety Class 3, add Seismic Category I and add "B" under Quality Assurance Requirement.

RESPONSE 210.17

The fuel pool cooling and cleanup system is Non-Nuclear Safety, Quality Group C in accordance with the SRP Section 9.1.3, AISI/ANS 52.1 - 1983, the EPRI ALWR Requirements Program and past industry practice. The spent fuel pool is Safety Class 3, has a Seismic Category I makeup water system and source (i.e., the RHR System) is housed in the Safety Class 3 secondary containment, and has a Safety Class 3 ventilation system. Item G2.7 of Table 3.2-1 has been corrected to show that the RHR Connections are Safety Class 3.

QUESTION 210.18

A staff position is that piping and valves forming part of primary containment boundary should be Seismic Category 1. In Table 3.2-1, piping and valves in the Reactor Building Cooling Water System which form part of the primary containment boundary are classified as Non-Seismic. Revise Table 3.2-1 to add Seismic Category I to the classification of Item P2.1 or provide a justification for not doing so.

RESPONSE 210.18

Item P2.1 of Table 3.2-1 has been corrected.

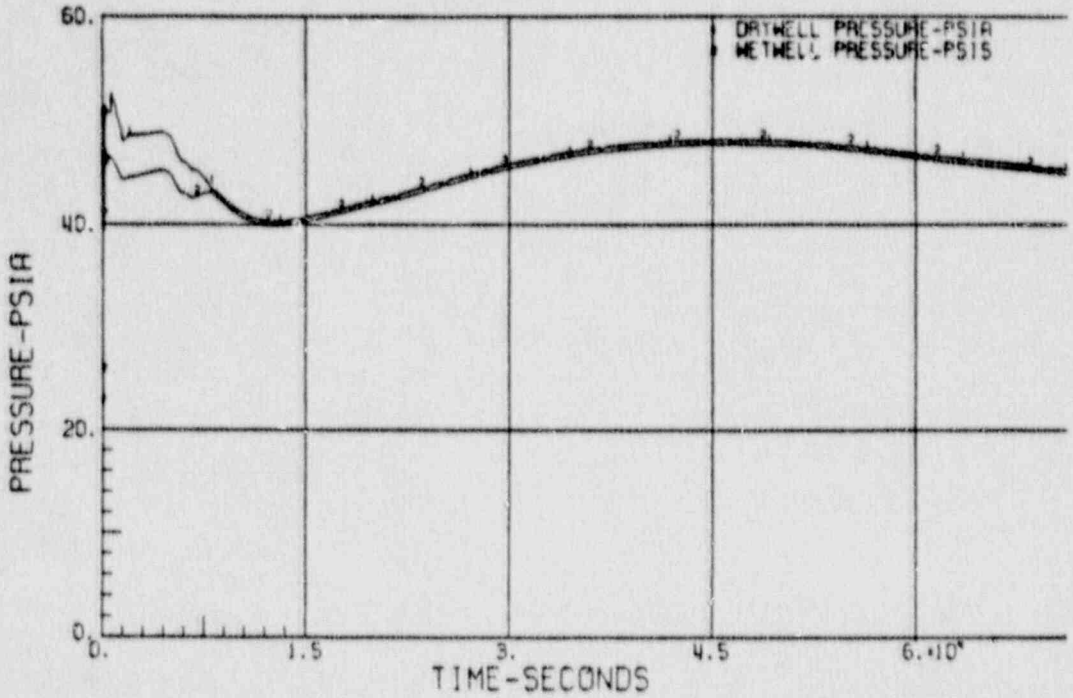

Figure 20.3-41 PRESSURE TIME HISTORY AFTER A FEEDWATER LINE BREAK AVAILABLE ECCS: 1 RHR SYSTEM

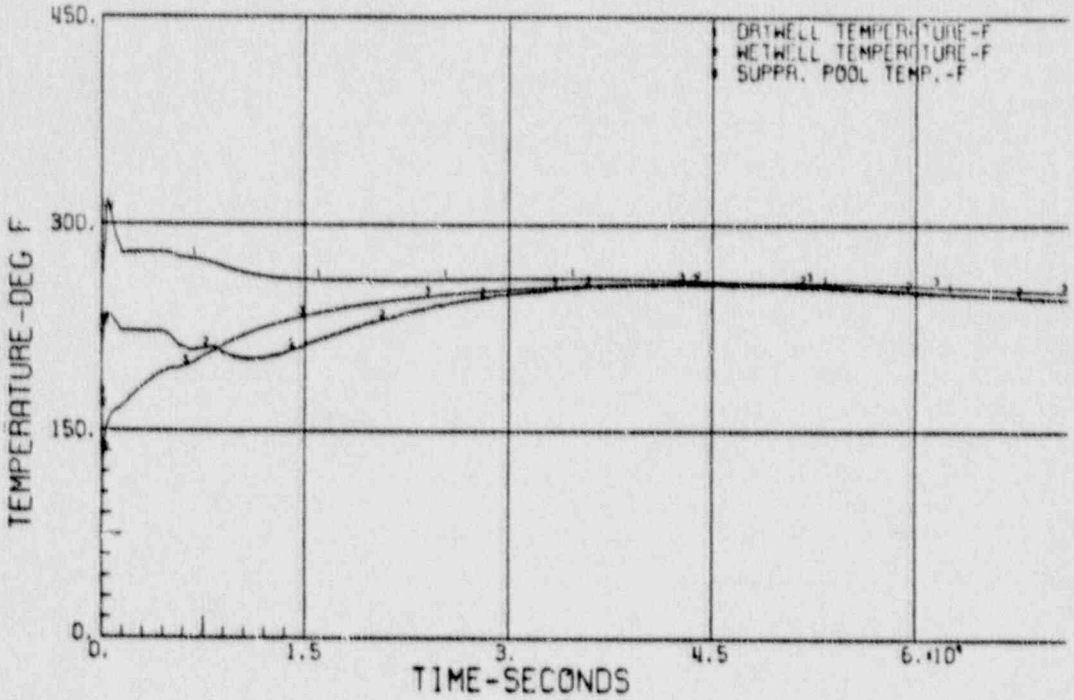

Figure 20.3-42 TEMPERATURE TIME HISTORY AFTER A FEEDWATER BREAK AVAILABLE ECCS: 1 RHR SYSTEM

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11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems are provided to allow determination of the content of radioactive material in various gaseous and liquid process and effluent streams. The design objective and criteria are primarily determined by the system designation of either:

(1) instrumentation systems required for safety, or

(2) instrumentation systems required for plant operation.

11.5.1 Design Bases

11.5.1.1 Design Objectives

11.5.1.1.1 Systems Required for Safety

The main objective of radiation monitoring systems required for safety is to initiate appropriate protective action to limit the potential release of radioactive materials from the reactor vessel and primary and secondary containment if predetermined radiation levels are exceeded in major process/effluent streams. An additional objective is to provide control room personnel with an indication of the radiation levels in the major process/effluent streams plus alarm annunciation if high radiation levels are detected.

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

(1) main steamline RMS;

(2) containment heating, ventilating, and air conditioning (HVAC) radiation monitoring system;

(3) fuel area HVAC radiation monitoring system;

(4) control building HVAC radiation monitoring system; and

(5) standby gas treatment system vent radiation monitoring system.

11.5.1.1.2 Systems Required for Plant Operation

The main objective of operational radiation monitoring systems is to provide operating personnel with measurements of the content of radioactive material in all effluent and important process streams. This allows demonstration of compliance with plant normal operational technical specifications by providing gross radiation level monitoring and collection of halogens and particulates on filters (gaseous effluents) as required by Regulatory Guide 1.21. Additional objectives are to initiate discharge valve isolation on the offgas or liquid radwaste systems if predetermined release rates are exceeded and to provide for sampling at certain radiation monitor locations to allow determination of specific radionuclide content.

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

(1) for gaseous effluent streams

(a) plant vent discharge;

(b) offgas exhaust vent;

(c) radwaste building ventilation; and

(d) turbine building ventilation.

(2) for liquid effluent streams

(a) radwaste effluent radioactivity.

(3) for gaseous process streams

(a) offgas pre-treatment sampling;

(b) offgas post-treatment sampling; and

(c) carbon bed vault gross gamma radiation levels.

(4) for liquid process streams

(a) reactor building closed cooling water intersystem radiation leakage.

11.5.1.2 Design Criteria

11.5.1.2.1 Systems Required for Safety

The design criteria for the main steamline and containment ventilation exhaust plenum monitoring systems are that the systems shall:

(1) withstand the effect of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions;

(2) perform their intended safety functions in the environment resulting from normal and abnormal conditions (e.g., loss of HVAC and isolation events);

(3) meet the reliability, testability, independence, and failure mode requirements of engineered safety features;

(4) provide continuous output on control room panels;

(5) permit checking of the operational availability of each channel during reactor operation with provisions for calibration function and instrument checks;

(6) assure an extremely high probability of accomplishing safety functions in the event of anticipated operational occurrences;

(7) initiate prompt protective action prior to exceeding plant technical specification limits;

(8) provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation;

(9) insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and channel trip;

(10) register full-scale output if radiation detection exceeds full scale; and

(11) have sensitivities and ranges compatible with anticipated radiation levels.

The applicable General Design Criteria of 10CFR50, Appendix A, are 1, 2, 3, 13, 20, 21, 22, 23, 24, and 29. The systems shall meet the design requirements for Safety Class 2, Seismic Category I, systems along with the quality assurance requirements of 10CFR50, Appendix B.

11.5.1.2.2 Systems Required for Plant Operation

The design criteria for operational radiation monitoring systems are that the systems shall:

(1) provide continuous indication of radiation levels in the main control room;

(2) provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation;

(3) insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and discharge valve isolation channel trip;

(4) monitor a sample representative of the bulk stream or volume;

(5) have provisions for calibration, function, and instrumentation checks;

(6) have sensitivities and ranges compatible with anticipated radiation levels; and

(7) register full-scale output if radiation detection exceeds full scale.

The RMS monitoring discharges from the gaseous and liquid radwaste treatment system shall have provisions to alarm and to initiate automatic closure of the waste discharge valve on the affected treatment system prior to exceeding the normal operation limits specified in technical specifications as required by Regulatory Guide 1.21.

radiation monitor and the common two-pen recorder. A 120-Vac local bus supplies the sample panel.

Each radiation monitor has three trip circuits: two upscale (high-high-high and high), and one downscale (low)/inoperative. Each trip is visually displayed on the radiation monitor. The first three trips actuate corresponding main control room annunciators: offgas post treatment high-high-high radiation, offgas post treatment high radiation, and offgas post treatment downscale. A trip circuit on the recorder actuates an offgas post treatment high-high radiation annunciator. High or low sample flow measured at the sample panel actuates a main control room offgas vent pipe sample high-low flow annunciator.

A trip auxiliary unit in the control room takes the high-high-high (HHH) and downscale trip outputs and, if its logic is satisfied, initiates closure of the offgas system discharge and drain valves. The logic is satisfied if either two HHH, on one HHH and one downscale, or two downscale trips occur. The HHH trip setpoints are determined so that valve closure is initiated prior to exceeding technical specification limits. Any one high upscale trip initiates closure of offgas system bypass line valve and permits opening of the treatment line valve.

A vial sampler panel similar to the pre-treatment sampler panel is provided from grab sample collection to allow isotopic analysis and gross monitor calibration.

11.5.2.2.3 Carbon Bed Vault Radiation Monitoring System

The carbon vault is monitored for gross gamma radiation level with a single instrument channel. The channel includes a sensor and converter, an indicator and trip unit, and a locally-mounted auxiliary unit. The indicator and trip unit is located in the main control room. The channel provides for sensing and readout, both local and remote, of gross gamma radiation over a range of six logarithmic decades (1 to 10^6 mR/hr).

The indicator and trip unit has one adjustable upscale trip circuit for alarm and one downscale trip circuit for instrument trouble. The trip circuits are capable of convenient operational verification by means of test signals or through the use of portable gamma sources. Power is supplied from channel A of the containment HVAC radiation monitoring system.

11.5.2.2.4 Plant Vent Discharge Radiation Monitoring System

This system monitors the plant vent discharge for gross radiation level and collects halogen and particulate samples. A representative sample is continuously extracted from the ventilation ducting through an isokinetic probe in accordance with ANSI N13.1 passed through the containment ventilation sample panel for monitoring and sampling, and returned to tne ventilation ducting. The sample panel has a pair of filters (one for particulate collection and one for halogen collection) in parallel (with respect to flow) with a continuous gross radiation detection assembly. The gross radiation detection assembly consists of a shielded chamber, beta-gamma-sensitive GM tubes, and a check source. A radiation monitor in the min control room analyzes and visually displays the measured gross radiation level.

The sample panel shielded chambers can be purged with room air by using two solenoid valves operated from the control room to check detector response to background radiation, thus checking operability of the gross radiation channel.

Power is supplied from 120-Vac instrument Bus J2 for the radiation monitor and recorder and from 120-Vac instrument Bus E2 for the sample panel. The recorder has one pen.

The radiation monitor has three trip circuits: two upscale (high-high and high) and one downscale (low). Each trip is visually displayed on the radiation monitor. These three trips actuate high-high radiation, plant vent discharge high radiation, and plant vent discharge downscale. High or low sample pressure measured at the sample panel actuates a main control room plant vent discharge sample high-low flow annunciator.

Table 11.5-2 presents the gaseous and airborne monitors for the effluent radiation monitoring system.

11.5.2.2.5 Liquid Process and Effluent Monitoring Systems

These systems monitor the gamma radiation levels of liquid process and effluent streams. With the exception of the radwaste system effluent, the streams monitored normally contain only background levels of radioactive materials. Increases in radiation level may be indicative of heat exchanger leakage or equipment malfunction.

Power is supplied from 120-Vac non-divisional buses for the radiation monitors and recorders and from a 120-Vac local bus for the sample panels.

Each radiation monitor has two trip circuits: one upscale (high) and one downscale (low or inoperative). Each trip is visually displayed on the affected radiation monitor. These trips actuate corresponding main control room annunciators: one upscale (high radiation) and the downscale for the affected liquid monitoring channel. Low sample flow measured at the sample panel actuates a control room low-flow annunciator for the affected liquid channel.

For each liquid monitoring location, a continuous sample is extracted from the liquid process pipe, passed through a liquid sample panel which contains a detection assembly for gross radiation monitoring, and returned to the process pipe. The detection assembly consists of a scintillation detector mounted in a shielded sample chamber equipped with a check source. A radiation monitor in the control room analyzes and visually displays the measured gross radiation level.

The sample panel chamber and lines can be drained to allow assessment of background buildup. The panel measures and indicates sample line flow. A solenoid-operated check source operated from the control room can be used to check operability of the channel.

11.5.2.2.5.1 Radwaste Effluent Radiation Monitoring System

This system monitors the radioactivity in the radwaste effluent prior to its discharge.

Liquid waste can be discharged from the sample tanks containing liquids that have been processed through one or more treatment systems such as evaporation, filtration, and ion exchange. Prior to the discharge from the tank, the liquid in the tank is sampled and analyzed in the laboratory. Based upon this analysis

discharge is permitted at a specified release rate and dilution rate.

The radiation monitor has four trip circuits. This high-high upscale trip on the radwaste effluent radiation monitor is used to initiate closure of the radwaste system discharge valve and simultaneously actuate an alarm in the radwaste control room.

The high upscale trip and the low downscale trip actuate annunciators in the main control room. A low flow switch in the sample line also actuates an annunciator in the main control room. Table 11.5-3 presents the liquid monitors for the process radiation monitoring system.

11.5.2.2.5.2 Reactor Building Cooling Water Radiation Monitoring System

This system consists of three channels: one for each loop for monitioring intersystem radiation leakage into the rector building cooling water system.

11.5.2.2.5.3 Deleted

11.5.2.2.5.4 Deleted

11.5.2.2.6 Radwaste Building HVAC Radiation Monitoring System

This system monitors the radwaste building ventilation discharge, including radwaste storage tank vents, for gross radiation level. The system consists of two redundant instrument subsystems which are physically and electrically separate from each other. Each subsystem consists of four channels (A, B, C, and D) and each channel has a local detector, a converter, and a main control room radiation monitor. Power is supplied to one subsystem of four channels by the 120-Vac instrument Bus J1 and to the other subsystem by the 120-Vac instrument Bus J2.

Each radiation monitor provides two trip circuits: one for upscale (high) radiation or an inoperative circuit and one for downscale. An upscale/inoperative trip of the channel A or B radiation monitors initiates the closing of the radwaste tank and pump room isolation valves and exhausts the air from the radwaste working areas. An upscale/inoperative trip of channel C initiates the closing of the oil separator room zone supply valves, the closing of the radwaste exhaust fan inlet damper, the stopping of the radwaste exhaust fan, and the startup of the air clean up fan.

The same trips on the other four main control room radiation monitors of the redundant system initiate the actuation of the same valves, damper, and fan.

High radiation and downscale main control room annunciators are actuated by the trip signals from the monitors.

Each main control room radiation monitor visually displays the radiation level. In addition, each annunciator supplies an output signal to the computer.

11.5.2.2.7 Offgas Vent Radiation Monitoring System

This system monitors the offgas vent discharge for gross radiation level and collects halogen and particulate samples. The system is identical to the plant vent radiation monitoring system with corresponding annunciators.

11.5.3 Effluent Monitoring and Sampling

All potentially radioactive effluent materials are monitored for radioactivity in accordance with Criterion 64 of General Design Criteria, 10CFR50, Appendix A, as follows:

(1) liquid releases are monitored for gross gamma radioactivity;

(2) solid wastes are monitored for gross gamma radioactivity; and

(3) gaseous releases are monitored for gross gamma radioactivity.

11.5.3.1 Basis for Monitor Location Selection

Monitor locations are selected to assure that all effluent materials comply with regulatory requirements as covered in Regulatory Guide 1.21.

11.5.3.2 Expected Radiation Levels

Expected radiation levels are in the ranges listed in Tables 11.5-2 and 11.5-3.

11.5.3.3 Instrumentation

Radiation monitors used are listed in Table 11.5-1.

Grab samples are analyzed to identify and quantify the specific radionuclides in effluents and wastes. The results from the sample analysis are used to establish relationships between the gross gamma monitor readings and concentrations or release rates of radionuclides in continuous effluent releases.

11.5.3.4 Setpoints

Setpoints for actuation of automatic control features initiating actuation of isolation valves, dampers or diversion valves are specified in the plant technical specifications.

Setpoints are listed in Table 11.5-1.

11.5.4. Process Monitoring and Sampling

11.5.4.1 Implementation of General Design Criterion 60

All potentially significant radioactive discharge paths are equipped with a control system to automatically isolate the discharge on indication of a high radiation level. These include:

(1) offgas post-treatment;

(2) containment HVAC; and

(3) liquid radwaste effluent.

The effluent isolation functions for each monitor are given in Table 11.5-1.

11.5.4.2 Implementation of General Design Criteria 64

Radiation levels in radioactive and potentially radioactive process streams are monitored by the following process monitors:

(1) main steamline;

(2) offgas pretreatment and post-treatment;

(3) carbon bed vault; and

(4) reactor building cooling water.

11.5.4.3 Basis for Monitor Location Selection

Monitor locations are selected to assure compliance with Regulatory Guide 1.21 in that sample points are located where there is a minimum of disturbance due to fittings and other physical characteristics of the equipment and components. Sample nozzles are inserted into the flow or liquid volume to ensure sampling the bulk volume of pipes and tanks. In the case of both liquid and gas flow, care is taken to assure that individual samples are actually representative of the effluent mixture. A more detailed discussion is given in ANSI N13.1.

11.5.4.4 Expected Radiation Levels

Expected radiation levels are listed in Tables 11.5-2 and 11.5-3.

11.5.4.5 Instrumentation

Radiation monitors used are listed in Table 11.5-1.

Grab samples are analyzed to identify and quantify the specific radionuclides in process streams. The results from the sample analysis are used to establish relationships between the gross gamma monitor readings and concentration and radionuclides in the process streams.

11.5.4.6 Setpoints

Set points are listed in Table 11.5-1.

11.5.5 Calibration and Maintenance

11.5.5.1 Inspection and Tests

During reactor operation, daily checks of system operability are made by observing channel behavior. At periodic intervals during reactor operation, the detector response of each monitor provided with a remotely positioned check source will be recorded together with the instrument background count rate to ensure proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely positioned check source will have its response checked with portable check source. A record will be maintained showing the background radiation level and the detector response.

The system has electronic testing and calibrating equipment which permits channel testing without relocating or dismounting channel components. An internal trip test circuit adjustable over the fuel range of the readout meter is used for testing. Each channel is tested at least semiannually prior to performing a calibration check. Verification of valve operation, ventilation diversion, or other trip function will be done at this time if it can be done without jeopardizing plant safety. The test will be documented.

The following monitors have alarm trip circuits which can be tested by using test signals or portable gamma sources:

(1) main steamline;

(2) containment HVAC;

(3) fuel area HVAC;

(4) offgas pretreatment; and

(5) carbon bed vault.

The following monitors include built-in check sources and purge systems which can be operated from the main control room:

(1) offgas post-treatment;

(2) plant vent discharge;

(3) radwaste building; and

(4) offgas vent.

11.5.5.2 Calibration

The continuous radiation monitor calibration is according to certified National Bureau of Standards of commercial radionuclide standards, and is accurate to at least + or - 15%. The source-detector geometry during primary calibration is identical to the sample-detector geometry in actual use. Secondary standards which were counted in reproducible geometry during the primary calibration are supplied with each continuous monitor for calibration after installation. Each continuous monitor is calibrated usually during plant operation or during the refueling outage if the detector is not readily accessible. A calibration can also be performed by using liquid or gaseous radionuclide standards or by analyzing particulate iodine or gaseous grab samples with laboratory instruments.

The offgas pretreatment monitor shall respond to a gross gamma signal obtained from the periodic analyses of grab samples. The readout units shall be mR/hr per mCi/sec.

The following monitors shall respond to a gross gamma signal obtained from the periodic analyses of grab samples to read the rate in counts/min

(1) offgas post-treatment;

(2) plant vent discharge;

(3) radwaste building vent;

(4) fuel area vent;

(5) offgas vent;

(6) radwaste effluent; and

(7) reactor building cooling water.

The following monitors shall be calibrated to read the gross gamma dose rate in mR/hr:

(1) main steamline;

(2) containment HVAC;

(3) fuel area HVAC;

(4) carbon bed vault; and

(5) control building HVAC.

11.5.5.3 Maintenance

The channel detector, electronics, and recorder are serviced and maintained on an annual basis or in accordance with manufacturers recommendations to ensure reliable operations. Such maintenance includes cleaning, lubrication, and assurance of free movement of the recorder in addition to the replacement or adjustment of any components required after performing a test or calibration check. If any work is performed which would affect the calibration, a recalibration is performed at the completion of the work.

11.5.5.4 Audits and Verifications

Out of Standard Plant scope.

QUESTION 210.46

Portions of the stress, deformation and buckling limits for safety class reactor internals which are listed in Tables 3.9-4, 3.9-5 and 3.9-6 requires additional review by the staff. If either Equation b in Table 3.9-4, Equations e, f, and g in Table 3.9-5 or Equation c in Table 3.9-6 will be used in the design of safety class reactor internals for the ABWR, provide a commitment in each of these tables that supporting data will be provided to the staff for review.

RESPONSE 210.46

A commitment to provide supporting data for NRC review has been added as requested.

QUESTION 210.47

The information in Subsection 3.9.6 infers that only ASME Class 1, 2 and 3 pumps and valves will be included in the inservice testing (IST) program for the ABWR. It is the staff's position as stated in Standard Review Plan, Sections 3.9.6.II.1 and 3.9.6.II.2 that all pumps and valves which are considered as safety-related should be included in the IST program even if they are not categorized as ASME Class 1, 2 or 3. Revise Subsection 3.9.6 to agree with this position.

RESPONSE 210.47

Response to this question is provided in revised Subsection 3.9.6.

QUESTION 210.48

The first paragraph in Subsection 3.9.6 states that accessibility for inservice testing of applicable pumps and valves is provided in the plant design. However, the second paragraph and Subsection 3.9.6.3 infers that relief from ASME Section XI inservice testing will be submitted for some pumps and valves.

RESPONSE 210.48

GE is aware that there may be a need for exceptions in cases where access or configuration prohibits some types of NDE or where operability of certain equipment must be tested indirectly rather than by direct operation (based on the type of equipment). However, it is not GE's intention to take deliberate exception to Section XI requirements on the basis of inappropriate design. Therefore, Subsection 3.9.6.3 has been removed.

QUESTION 210.49

In Subsection 3.9.6, "Inservice Testing of Pumps and Valves," provide a commitment to perform periodic leak testing of all pressure isolation valves in accordance with the applicable sections of the technical Specifications for recently licensed BWR/6 plants. Normally, this information includes a list of all pressure isolation valves which will be leak tested. If such a list is not available for the ABWR, a commitment to provide the list of valves as a part of the ABWR Technical Specifications will be acceptable.

RESPONSE 210.49

Response to this question is provided in revised Subsection 3.9.6.