GE Nuclear Energy

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

April 29, 1988

MFN No. 42-88

Docket No. STN 50-605

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

Attention: Lester S. Rubenstein, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **Submittal of Responses to Additional Information as Requested in an NRC Letter from Dino C. Scaletti, Dated February 22, 1988**

Dear Mr. Rubenstein:

Enclosed are thirty four (34) copies of the Responses to Additional Information on the Standard Safety Analysis Report (SSAR) for the Advanced Boiling Water Reactor (ABWR). These responses principally pertain to Chapters 4, 5, 6 and 15.

It is intended that GE will amend the SSAR with these responses in June 1988.

Sincerely,

Ricardo Artigas, Manager
Licensing and Consulting Services

Do28
1/1

cc: D. R. Wilkins (GE)
F. A. Ross (DOE)
J. F. Quirk (GE)
D. C. Scaletti (NRC)

CHAPTER 20
QUESTION AND RESPONSE GUIDE

CHAPTER 20

TABLE OF CONTENTS

CHAPTER 20

TABLE OF CONTENTS (Continued)

SECTION 20.1

TABLES

20.1 QUESTION INDEX

This subsection provides an index to each NRC request for additional information (RAI) during its review of the ABWR standard plant. Each NRC question is designated with the NRC branch questions ID number (see Table 20.1-1) followed by the number of the question of the review area for that branch.

For example, question number 210.2 designates the second question of the mechanical engineering branch (EMEB). The index below provides an up-to-date listing in numerical order of each question.

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
EMEB	Mechanical	210.1	5.2.1.2	20.3.1	1
	Engineering	210.2	5.2.1.2	20.3.1	1
EMTB	Inservice	250.1	5.2.4.1	20.3.1	1
	Inspection	250.2	5.2.4.2	20.3.1	1
		250.3	6.6.8	20.3.1	1
	Component	251.1	5.3.1.1	20.3.1	1
	Integrity	251.2	5.3.1.2	20.3.1	1
		251.3	5.3.1.4.4	20.3.1	1
			5.3.1.4.5	20.3.1	1
			5.3.1.4.7	20.3.1	1
			5.3.1.5.2	20.3.1	1
			5.3.1.5.3	20.3.1	1
			5.3.2.1.5	20.3.1	1
		251.4	5.3.1.6.1	20.3.1	1
		251.5	5.3.1.6.3	20.3.1	1
		251.6	5.3.2.1	20.3.1	1
		251.7	5.3.2.1.1	20.3.1	1
			5.3.2.1.2	20.3.1	1
			5.3.2.1.3	20.3.1	1
			5.3.2.1.5	20.3.1	1
		251.8	5.3.3	20.3.1	1
		251.9	5.3.3.1.1.1	20.3.1	1
		251.10	5.3.3.2	20.3.1	1
		251.11	5.3.3.6	20.3.1	1

* *See Table 20.1-1 for abbreviations.*

** *Letter reference of Section 20.4*

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
	Materials Application	252.1	4.5.1.1(1)	20.3.1	1
		252.2	4.5.1.1(2)	20.3.1	1
		252.3	4.5.2.2	20.3.1	1
		252.4	4.5.2.3	20.3.1	1
		252.5	4.5.2.4	20.3.1	1
		252.6	4.5.2.5	20.3.1	1
		252.7	5.2.3.2.2	20.3.1	1
		252.8	5.2.3.2.3	20.3.1	1
		252.9	5.2.3.3.1	20.3.1	1
		252.8	5.2.3.2.3	20.3.1	1
		252.10	5.2.3.4.1.1	20.3.1	1
		252.11	5.2.3.4.2.3	20.3.1	1
ECEB	Chemical Technology	281.1	5.1	20.3.1	1
		281.2	5.2.3.2.2	20.3.1	1
		281.3	5.2.3.2.2	20.3.1	1
		281.4	5.2.3.2.2	20.3.1	1
		281.5	5.2.3.2.2	20.3.1	1
		281.6	5.2.3.2.2.2	20.3.1	1
		281.7	5.2.3.2.2.3(4)	20.3.1	1
		281.8	5.2.3.2.2.3(13)	20.3.1	1
		281.9	6.4.9.2	20.3.1	1
		281.10	Chap. 5	20.3.1	1
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
		470.9	15.7	20.3.1	1
		470.10	15.7	20.3.1	1

TABLE 20.1-1

IDENTIFICATION NUMBERS FOR NRC REVIEW QUESTIONS

Question ID Number	Review Area	Branch	Applicable SRP Sections
100	Miscellaneous	Responsible	None Project Directorate
210	Mechanical Engineering	EMEB	3.2.1, 3.2.2, 3.6.2, BTP-MEB 3-1, 3.9.1, 3.9.2, 3.9.3, 3.9.4, 3.9.5, 3.9.6, 5.2.1.1, 5.2.1.2
220	Structural Engineering	ESGB	3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.8.1, 3.8.2, 3.8.3, 3.8.4, 3.8.5
230	Seismology	ESGB	2.5.2
231	Geology	ESGB	2.5.1, 2.5.3
240	Hydrologic Engineering	ESGB	2.4.1, 2.4.2, 2.4.3, 2.4.4, 2.4.5, 2.4.6, 2.4.7, 2.4.8, 2.4.9, 2.4.10, 2.4.11, 2.4.12, BTP HGEB-1, 2.4.13, 12.4.14
241	Geotechnical Engineering	ESGB	2.5.4, 2.5.5
250	Inservice Inspection	EMTB	5.2.4, 5.4.2.2, 6.6, 10.2.3
251	Component Integrity	EMTB	3.5.1.3, 5.3.1, 5.3.2, BTP MTEB 5-2, 5.4.1.1, 6.2.7
252	Materials Application	EMTB	4.5.1, 4.5.2, 5.2.3, BTP MTEB 5-2, 5.3.3, 5.4.2.1, 6.1.1, 10.3.6
260	Quality Assurance	LQAB	17.1, 17.2
270	Environmental Qualification	SPLB	3.11

TABLE 20.1-1

IDENTIFICATION NUMBERS FOR NRC REVIEW QUESTIONS

Question ID Number	Review Area	Branch	Applicable SRP Sections
100	Miscellaneous	Responsible	None Project Directorate
210	Mechanical Engineering	EMEB	3.2.1, 3.2.2, 3.6.2, BTP-MEB 3-1, 3.9.1, 3.9.2, 3.9.3, 3.9.4, 3.9.5, 3.9.6, 5.2.1.1, 5.2.1.2
220	Structural Engineering	ESGB	3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.8.1, 3.8.2, 3.8.3, 3.8.4, 3.8.5
230	Seismology	ESGB	2.5.2
231	Geology	ESGB	2.5.1, 2.5.3
240	Hydrologic Engineering	ESGB	2.4.1, 2.4.2, 2.4.3, 2.4.4, 2.4.5, 2.4.6, 2.4.7, 2.4.8, 2.4.9, 2.4.10, 2.4.11, 2.4.12, BTP HGEB-1, 2.4.13, 12.4.14
241	Geotechnical Engineering	ESGB	2.5.4, 2.5.5
250	Inservice Inspection	EMTB	5.2.4, 5.4.2.2, 6.6, 10.2.3
251	Component Integrity	EMTB	3.5.1.3, 5.3.1, 5.3.2, BTP MTEB 5-2, 5.4.1.1, 6.2.7
252	Materials Application	EMTB	4.5.1, 4.5.2, 5.2.3, BTP MTEB 5-2, 5.3.3, 5.4.2.1, 6.1.1, 10.3.6
260	Quality Assurance	LQAB	17.1, 17.2
270	Environmental Qualification	SPLB	3.11

TABLE 20.1-1

IDENTIFICATION NUMBERS FOR NRC REVIEW QUESTIONS (Continued)

Question ID Number	Review Area	Branch	Applicable SRP Sections
271	Seismic and Dynamic Load Qualification	EMEB	3.10
280	Fire Protection	ECEB	9 5.1, BTP CMEB 9.5.1
281	Chemical Technology	ECEB	BTP MTEB 5-3, 5.4.8, BTP MTEB 6-1, 6.1.2, 9.3.2, 9.3.4, 9.5.1, BTP CMEB 9.5-1, 10.4.6, 10.4.8
290	Environmental Engineering	ESGB	Environmental Report
310	Regional Impact Analysis	ESGB	Environmental Report
311	Site Analysis	ESGB	2.1.1, 2.1.2, 2.1.3, 2.2.1-2.2.2, 2.2.3, 3.5.1.5, 3.5.1.6
320	Antitrust and Economic Analysis	PTSB	None
410	Auxiliary Systems	SPLB	3.4.1, 3.5.1.1, 3.5.1.2, 3.5.1.4, 3.5.2, 3.6.1, BTP ASB 3-1, 5.2.5, 5.4.11, 6.7, 9.1.1, 9.1.2, 9.1.3, 9.1.4, BTP ASB 9-1, 9.1.5, 9.2.1, 9.2.2, 9.2.4, 9.2.5, BTP ASB 9-2, 9.2.6, 9.3.1, 9.3.3, 9.4.1, 9.4.2, 9.4.3, 9.4.4, 9.4.5, 10.4.5, 10.4.7, BTP ASB 10-2, 10.4.9 BTP ASB 10-1
		SRXB	4.6, 9.3.5
		ECEB	9.2.3

TABLE 20.1-1

IDENTIFICATION NUMBERS FOR NRC REVIEW QUESTIONS (Continued)

Question ID Number	Review Area	Branch	Applicable SRP Sections
420	Instrumentation and Control Systems	SICB	7.1, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7
430	Power Systems	SELB	8.1, 8.2, 8.3.1, 8.3.2, 9.5.3
		SICB	9.5.2
		SPLB	9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8, 10.2, 10.3, 10.4.1, 10.4.4
440	Reactor Systems	SRXB	5.2.2, BTP RSB 5-2, 5.4.6, 5.4.7, BTP RSB 5-1, 5.4.12, 6.3, BTP RSB 6-1, 15.1.1-15.1.4, 15.1.5, 15.2.1-15.2.5, 15.2.6, 15.2.7, 15.2.8, 15.3.1-15.3.2 15.3.3-15.3.4, 15.4.4-15.4.5 15.4.6, 15.5.1-15.5.2, 15.6.1, 15.6.5, 15.8
450	Accident Evaluation	SPLB	6.4, 6.5.3
		PRPB	App. A to 15.4.8, Appendix A to 15.4.9, 15.6.2, 15.6.3, 15.6.4, Appendix A to 15.6.5 Appendix B to 15.6.5 Appendix C to 15.6.5 Appendix D to 15.6.5 15.7.4, 15.7.5
		ECEB	6.5.2, 6.5.4
		SRXB	Appendix A to 15.1.5
451	Meteorology	PRPB	2.3.1, 2.3.2, 2.3.3, 2.3.4, 2.3.5

TABLE 20.1-1

IDENTIFICATION NUMBERS FOR NRC REVIEW QUESTIONS (Continued)

Question ID Number	Review Area	Branch	Applicable SRP Sections
460	Effluent Treatment	SPLB	6.5.1, 10.4.2, 10.4.3, 11.1, 11.2, 11.3, 11.4, BTP ETSB 11-3, BTP ETSB 11-5, 11.5, 15.7.3
470	Radiological Impact	PRPB	Environmental Report
471	Radiation Protection	PRPB	12.1, 12.2, 12.3-12.4, 12.5
480	Containment Systems	SPLB	6.2.1, 6.2.1.1.A, 6.2.1.1.B, 6.2.1.1.C, 6.2.1.2, 6.2.1.3, 6.2.1.4, 6.2.1.5, BTP CSB 6-1, 6.2.2, 6.2.3, BTP CSB 6-3, 6.2.4 BTP CSB 6-4, 6.2.5, BTP CSB 6-2, 6.2.6
490	Fuels	SRXB	4.2
491	Physics	SRXB	4.3, BTP CPB 4.3-1, 15.4.1, 15.4.2, 15.4.3, 15.4.7, 15.4.8, 15.4.9
492	Thermal-Hydraulics	SRXB	4.4
610	Operator Licensing	LHFB	13.2.1
620	Human Factors Engineering	LHFB	18, 18.1, 18.2
630	Licensee Qualifications	LPEB	13.1.1, 13.1.2-13.1.3, 13.4, 13.5.1
		LHFB	13.2.2,
640	Procedures and Systems Review	LHFB	13.5.2, 14.2

TABLE 20.1-1

IDENTIFICATION NUMBERS FOR NRC REVIEW QUESTIONS (Continued)

Question ID Number	Review Area	Branch	Applicable SRP Sections
720	Reliability and Risk Assessment	PRPB	None
730	Generic Issues		None
810	Emergency Planning	PEPB	13.3
910	Safeguards	RSGB	13.6

Abbreviations

ECEB Chemical Engineering Branch
EMEB Mechanical Engineering Branch
EMTB Materials Engineering Branch
ESGB Structural and Geosciences Branch

LHFB Human Factors Assessment Branch
LPEB Performance Evaluation Branch
LQAB Quality Assurance Branch

PEPB Emergency Preparedness Branch
PRPB Radiation Protection Branch
PTSB Policy Development and Technical Support Branch

RSGB Safeguards Branch

SELB Electrical Systems Branch
SICB Instrumentation and Control Systems Branch
SPLB Plant Systems Branch
SRXB Reactor Systems Branch

SECTION 20.2

CONTENTS

20.2 QUESTIONS

This subsection provides an up-to-date chapter-wise listing of the NRC questions. Subsections are numbered (e.g., 20.2.x) in accordance with the questions received for specific chapters.

20.2.1 Chapter 1 Questions

None to date.

20.2.2 Chapter 2 Questions

None to date.

20.2.3 Chapter 3 Questions

None to date.

20.2.4 Chapter 4 Questions

252.1

Subsection 4.5.1.1 (1) should state: "The properties of the materials selected for the control rod drive mechanism must be equivalent to those given in Appendix I to Section III of the ASME Code, or parts A and B of Section II of the ASME Code, or are included in Regulatory Guide 1.85, except that cold-worked austenitic stainless steels should have a 0.2% offset yield strength no greater than 90,000 psi."

252.2

Subsection 4.5.1.1 (2) should state: "All materials for use in this system must be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code."

252.3

Subsection 4.5.2.2: The first sentence should read: "Core support structures are fabricated in accordance with the requirements of ASME Code, Section III, Subsection NG-4000, and the examination and acceptance criteria shown in NG-5000."

252.4

Subsection 4.5.2.3: The following statement should be added to the last sentence of the first paragraph: "The examination will satisfy the requirements of NG-5300."

252.5

Subsection 4.5.2.4 should state: "Furnace sensitized material should not be allowed."

252.6

Subsection 4.5.2.5 should state: "All materials used for reactor internals will be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls will preclude contamination of nickel-based alloys by chloride ions, fluoride ions, or lead."

20.2.5 Chapter 5 Questions

210.1

In Subsection 5.2.1.2, the statement is made that Section 50.55a of 10CFR50 requires NRC staff approval of ASME code cases only for Class 1 components. Revise this statement to be consistent with the current (1987) edition of 10CFR50.55a, which requires staff approval of code cases for ASME Class 1, 2, and 3 components.

210.2

Revise Table 5.2-1 or provide additional tables in Subsection 5.2.1.2 which identify all ASME code cases that will be used in the construction and in-plant operation of all ASME Class 1, 2, and 3 components in the ABWR. All code cases in these tables should be identified by code case number, revision, and title. These tables should include those applicable code cases that are listed either as acceptable or conditionally acceptable in Regulatory Guides 1.84, 1.85, and 1.147. For those code cases listed as conditionally acceptable, verify that the construction of all applicable components will be in compliance with the additional Regulatory Guide conditions.

250.1

Subsection 5.2.4.1 should state that the system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor systems, up to and including:

(1) The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

(2) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.

(3) The reactor coolant system and relief valves.

250.2

Subsection 5.2.4.2 should satisfy the requirements in ASME Code IWA-1500.

251.1

Subsection 5.3.1.1 should state that the materials will comply with the provisions of the ASME Code, Section III, Appendix I, and meet the specification requirements of 10CFR50, Appendix G.

251.2

Subsection 5.3.1.2 should state the specific subsection NB of ASME Code to which the manufacturing and fabrication specifications were alluded.

251.3

Subsections 5.3.1.4.4 and 5.3.1.4.5 should be rewritten; the cross-reference is unacceptable.

Subsections 5.3.1.4.7, 5.3.1.5.2, 5.3.1.6.3, and 5.3.2.1.5: Revision 2 of Regulatory Guide 1.99 should be added in these subsections.

251.4

Subsection 5.3.1.6.1: the third capsule of the vessel surveillance program is designated as a standby; however, according to ASTM 185-82, the capsule should be withdrawn at the end of life. Provide justification for this deviation.

251.5

Subsection 5.3.1.6.3 states that according to estimates of worst-case irradiation effects, the adjusted reference temperature at end-of-life is less than $100^{\circ}F$, and the end-of-life upper shelf energy exceeds 50 ft-lb. Provide the calculation and analysis associated with the estimate.

251.6

Subsection 5.3.2.1 should clarify where Reference 2 is located. Has the NRC staff reviewed and approved Reference 2? If not, the staff needs to review Reference 2 in order to complete the review of this subsection.

251.7

Subsections 5.3.2.1.1, 5.3.2.1.2, 5.3.2.1.3, and 5.3.2.1.5 need to be rewritten. The level of detail must be comparable to that of Standard Review Plan 5.3.2 and Branch Technical Position MTEB 5-2.

252.8

Subsection 5.3.3 cited three GE documents:

(1) GE quality assurance program

(2) "Approved" inspection procedures, and

(3) NEDO-10029.

Has the NRC staff reviewed and approved the above documents? The staff cannot satisfactorily review this subsection without reviewing the above three documents.

251.9

Subsection 5.3.3.1.1.1 discusses the 60-year life of the ABWR reactor vessel. The NRC requirements and calculations on the fracture toughness and material properties are based on a 40-year life. Provide justification for the applicability of NRC's requirements on the 60-year life reactor vessel.

251.10

Subsection 5.3.3.2 should include the following information: neutron fluence, shift in reference temperature RT_{NDT} and upper shelf energy. The staff needs this information to compare to that of predicated values using Regulatory Guide 1.99.

251.11

Subsection 5.3.3.6 should indicate that operating conditions should satisfy the pressure-temperature limits prescribed in Subsection 5.3.2.

252.7

Subsection 5.2.3.2.2 is mostly an academic discussion of BWR water chemistry effect on intergranular stress corrosion cracking (IGSCC) in sensitized stainless steels. The subsection should discuss the actual ABWR water chemistry effects on the IGSCC. The subsection is vague about specific remedies or preventive measures to avoid IGSCC in ABWR. For example, the subsection failed to discuss how much hydrogen is needed for injection into the feedwater system or how the "tight conductivity control" would be implemented.

Also provide references for the "Laboratory studies..." and "available evidence..." that were mentioned in this subsection.

252.8

Subsection 5.2.3.2.3 should state that the requirements of GDC 4, relative to the compatibility of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code and by compliance with the recommendation of Regulatory Guide 1.44.

Specify the "very low limits" of the contaminants in the reactor coolant.

252.9

Subsection 5.2.3.3.1 should clarify where and how was the 45 ft-lb Charpy V value obtained.

The ferritic material used for piping, pumps, and valves should comply with Appendix G, Section G-3100, of ASME Code Section III.

This subsection should indicate that "calibration of instruments and equipment shall meet the requirements of the code, Section III, Paragraph NB-2360."

252.10

Subsection 5.2.3.4.1.1 should be rewritten to include more detailed discussion on avoidance of significant sensitization and on how the ABWR design complies with the NRC regulatory requirements.

252.11

Subsection 5.2.3.4.2.3 states that the ABWR design meets the intent of this Regulatory Guide (1.71) by utilizing the alternate approach given in Section 1.8. We cannot review this subsection because we have not received Section 1.8. In addition, this subsection should be rewritten because it lacks detailed discussion about welder qualification.

281.1

In Section 5.1 (page 5.1-2) the function of the reactor cleanup system filter demineralizer should include the removal of radioactive corrosion and fission products in addition to particulate and dissolved impurities.

281.2

In Subsection 5.2.3.2.2 (page 5.2-7) irradiation-assisted stress corrosion cracking (IASCC) of reactor internal components and its mitigation are not discussed. Present laboratory data and plant experience has shown that IASCC can be initiated even at low conductivity (< 0.3μS/cm) after long exposure to radiation.

281.3

In Subsection 5.2.3.2.2 (pages 5.2-7 and 8) the ABWR Standard Plant design does clearly incorporate hydrogen water chemistry to mitigate IGSCC. Since the plant design life is 60 years, hydrogen water chemistry may be of greater importance in reducing reactor coolant electrochemical corrosion potential to prevent IGSCC as well as IASCC. If hydrogen water chemistry is the referenced ABWR standard design, the following documents should be cited:

EPRI NP-5283-SR-A, *Guidelines for Permanent BWR Hydrogen Water Chemistry Installations* - 1987 Revision.

EPRI NP-4947-SR-LD, *BWR Hydrogen Water Chemistry Guidelines* - 1987 Revision (to be published).

281.4

In Subsection 5.2.3.2.2 (page 5.2-9) the utilization of the General Electric zinc injection passivation (GEZIP) process for radiation buildup control for the ABWR is not discussed. GEZIP was identified as a required design feature in the ABWR presentation to NRC staff.

281.5

In Subsection 5.2.3.2.2 (page 5.2-9) prefilming of stainless steel appears to be a promising method to reduce the buildup rate of activated corrosion products during subsequent plant operation. SIL No. 428 recommends preoperational testing of the recirculation system conducted at temperatures 230°F be done with the dissolved oxygen level controlled to between 200 and 400 ppb. Is control of radiation buildup through preoperational oxygen control being considered for the BWR Standard Plant? Are mechanical polishing and electropolishing of piping internal surfaces also being considered for reducing radiation buildup?

281.6

In Subsection 5.2.3.2.2.2 (page 5.2-9) cobalt 60 is identified as the principle contributor to shutdown radiation levels, especially the recirculation piping system of BWRs. Stellite contributes about 90% of the total cobalt 59 input to the reactor water (EPRI NP-2263, BWR Cobalt Source Identification, February 1982). Since irradiation of cobalt 59 yields cobalt 60, reduction in the source of cobalt 59 is needed to reduce the buildup of shutdown radiation levels. Indicate Stellite surface areas (square feet) in nuclear steam supply system and balance of plant. Provide the criteria for selecting Stellite plant materials for the designed application. Provide evaluation of noncobalt-containing materials whose properties are adequate to replace Stellite in-plant applications.

281.7

Subsection 5.2.3.2.2.3(4) (page 5.2-10) states that control of reactor water oxygen during startup/hot standby may be accomplished by utilizing the de-aeration capabilities of the condenser. In addition, this section states that independent control of control rod drive (CRD) cooling water oxygen concentrations of < 50 ppb during power operation is desirable to protect against IGSCC of CRD materials. Are either one or both of the above dissolved oxygen controls incorporated in the ABWR Standard Plant design?

281.8

In Subsection 5.2.3.2.2.3(13) (page 5.2-11) it states that the main steam line radiation monitor indicates an excessive amount of hydrogen being injected. An explanation of this occurrence should be discussed.

281.10

In the October 1987 ABWR presentation to the NRC staff the design features and/or requirements to improve water chemistry for GE-ABWR were specified. Address each one of these design features and/or requirements listed in Table I in the ABWR Standard Safety Analysis Report.

TABLE I

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

		ABWR Presentation to NRC Staff	ABWR Standard Safety Analysis Report
1 -	Selection of low cobalt materials to minimize radiation buildup	Required Design Feature	Not discussed in Subsection 5.2.3.
2 -	Hydrogen water chemistry to suppress IGSCC	Required Design Feature	Subsection 5.2.3.2.2 references normal water chemistry.
3 -	Zinc injection to minimize radiation buildup	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.2.
4 -	Full flow deep bed condensate system to reduce feedwater impurities	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.

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
5 -	Improved online monitoring instrumentation to assure water quality	Ion chromatography, electrochemical corrosion potential, and crack arrest verification system required design features	Only electrochemical corrosion potential discussed in Subsection 5.2.3.2.2.3.
6 -	Improved corrosion-resistant materials for steam extraction piping to minimize feedwater impurities	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.
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 lead 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.

20.2.6 Chapter 6 Questions

250.3

Subsection 6.6.8 should discuss the augmented inservice inspection for those portions of high energy piping enclosed in guard pipes.

252.12

Subsection 6.1.1.1 should discuss ferritic steel welding in detail. It should also discuss the control of ferrite content in stainless steel weld metal similar to that of Regulatory Guide 1.31.

252.13

Subsections 6.1.1.1.3.1, 6.1.1.1.3.2, and 6.1.1.1.3.5 should be rewritten because the cross-reference is unacceptable.

281.9

Subsection 6.4.4.2 (page 6.4-6) discusses personnel respirator use in the event of toxic gas intrusion into the control room. However, the chlorine detection system is not discussed. Also, any control functions that are automatically triggered by a chlorine detector alarm (closing intake dampers, energizing control room HVAC system recirculation) should be identified.

20.2.7 Chapter 7 Questions

None to date.

20.2.8 Chapter 8 Questions

None to date.

20.2.9 Chapter 9 Questions

None to date.

20.2.10 Chapter 10 Questions

None to date.

20.2.11 Chapter 11 Questions

None to date.

20.2.12 Chapter 12 Questions

None to date.

20.2.13 Chapter 13 Questions

None to date.

20.2.14 Chapter 14 Questions

None to date.

20.2.15 Chapter 15 Questions

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.

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.

470.3

Subsection 15.6.4.5.1.1 of the FSAR 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).

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 would result from using the NRC guidance.

470.5

Provide a discussion of, or reference to, the analysis of the radiological consequences of leakage from engineered safety feature components after a design basis LOCA.

470.6

For the spent fuel cask drop accident, what is the assumed period for decay from the stated power condition? What is the justification for that assumption?

470.7

The tables in Chapter 15 should be checked and revised as appropriate. In several cases the footnotes contain typographical errors related to defining the scientific notation. Table 15.7-12 also appears to contain inappropriate references to Table 15.7-16, rather that Table 15.7-13.

470.8

It is stated that Regulatory Guides 1.3 and 1.45 were used in the calculations of X/Q values. Based on the values presented, it appears as though a Pasquill stability Class F and one meter per second wind speed were assumed, with adjustment for meander per Figure 3 of Regulatory Guide 1.145. If this is not the case, describe the assumptions and justification used in calculating the X/Q values which are used in Chapter 15 dose assessments.

470.9

The SGTS filter efficiencies of 99% for inorganic and organic iodine are higher than the 90% and 70% values, respectively, assumed in Regulatory Guide 1.25 if it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine. Provide a justification for the use of the higher values.

470.10

Dose related factors such as breathing rates, iodine conversion factors and finite versus infinite cloud assumptions for calculating the whole body dose are not stated explicitly, although reference is made to Regulatory Guide 1.25 and another document. State these assumptions explicitly and justify use of any values which deviate from Regulatory Guide 1.25.

20.2.16 Chapter 16 Questions

None to date.

20.2.17 Chapter 17 Questions

None to date.

20.2.18 Chapter 18 Questions

None to date.

20.2.19 Chapter 19 Questions

None to date.

SECTION 20.3

CONTENTS

TABLES

20.3 QUESTIONS/RESPONSES

This subsection provides the responses for each of the NRC questions identified in Sections 20.1 and 20.2. For convenience, each question is repeated here before its corresponding response. These questions/responses are provided in groups corresponding to the NRC Requests for Additional Information (RAI) referenced in Section 20.4. Within each group, the questions/responses are presented in the numerical order of the question numbers.

20.3.1 Response to First RAI-Reference 1

QUESTION 210.1

In Subsection 5.2.1.2, the statement is made that Section 50.55a of 10CFR50 requires NRC staff approval of ASME Code Cases only for Class I components. Revise this statement to be consistent with the current (1987) edition of 10CFR50.55a which requires staff approval of Code Cases for ASME Class I, II, and III components.

RESPONSE 210.1

Response to this question is provided in revised Subsection 5.2.1.2.

QUESTION 210.2

Revise Table 5.2-1 or provide additional tables in Subsection 5.2.1.2 which identifies all ASME Code Cases that will be used in the construction and in-plant operations of all ASME Class I, II, and III components in the ABWR. All Code Cases in these tables should be identified by Code Case number, revision and title. These tables should include those applicable Code Cases that are listed either as acceptable or conditionally acceptable in Regulatory Guides 1.84, 1.85 and 1.147. For those Code Cases listed as conditionally acceptable, verify that the construction of all applicable components will be in compliance with the additional Regulatory Guide conditions.

RESPONSE 210.2

Response to this question is provided in revised Subsection 5.2.1.2 and Table 5.2-1.

QUESTION 250.1

Subsection 5.2.4.1 should state that the system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant systems, up to and including

(A) The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

(B) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.

(C) The reactor coolant system and relief valves.

RESPONSE 250.1

Response to this question is provided in revised Subsection 5.2.4.1.

QUESTION 250.2

Subsection 5.2.4.2 should satisfy the requirements in ASME Code, IWA-1500.

RESPONSE 250.2

Response to this question is provided in revised Subsection 5.2.4.2.

QUESTION 250.3

Subsection 6.6.8 should discuss the augmented inservice inspection for those portions of high energy piping enclosed in guard pipes.

RESPONSE 250.3

Augmented inservice inspection is not required for the ABWR design since there are no guard pipes enclosing high-energy piping between the containment isolation valves.

QUESTION 251.1

Subsection 5.3.1.1 should state that the material will comply with the provisions of the ASME Code, Section III, Appendix I, and meet the specification requirements of 10CFR50, Appendix G.

RESPONSE 251.1

Response to this question is provided in revised Subsection 5.3.1.1

QUESTION 251.2

Subsection 5.3.1.2 should state the specific subsection NB of ASME Code to which the manufacturing and fabrication specifications were alluded.

RESPONSE 251.2

Response to this question is provided in revised Subsection 5.3.1.2.

QUESTION 251.3

Subsections 5.3.1.4.4 and 5.3.1.4.5 should be rewritten; the cross-reference is unacceptable.

Subsections 5.3.1.4.7, 5.3.1.5.2, 5.3.1.6.3, and 5.3.2.1.5; Revision 2 of Regulatory Guide 1.99 should be added in these subsections.

RESPONSE 251.3

Response to the first part of this question is provided in revised Subsections 5.3.1.4.4 and 5.3.1.4.5.

The GE ABWR Licensing Review Bases issued by the NRC on August 7, 1987 specifies a SRP effectivity date of March 30, 1987. Thus, the Regulatory Guides in effect as of that date are applicable to the ABWR. However, rather than providing the specific revision of each Regulatory Guide each time it is noted in the test of the SSAR, GE has chosen to provide the applicable revisions of the Regulatory Guides in SSAR Subsection 1.8.2, which will be provided by June 30, 1988.

QUESTION 251.4

Subsection 5.3.1.6.1: the third capsule of the vessel surveillance program is designated as a standby; however, according to ASTM 185-82, the capsule should be withdrawn at the end of life. Provide justification for this deviation.

RESPONSE 251.4

Response to this question is provided in revised Subsection 5.3.1.6.1

QUESTION 251.5

Subsection 5.3.1.6.3 states that according to estimates of worst-case irradiation effects, the adjusted reference temperature at end-of-life is less that $100^\circ F$, and the end-of-life upper-shelf energy exceeds 50ft-lb. Provide the calculation and analysis associated with the estimate.

RESPONSE 251.5

The calculation and analysis associated with the estimate is provided below:

Calculate RT_{NDT} Shift in Vessel Material

Ref.: February 1986 draft of Regulatory Guide 1.99

A. I <u>Weld Metal</u>

Assume the following maximum values:

P = 0.020%, V = 0.05% Cu = 0.08%
Ni = 1.20% (Max Ni value considered in Regulatory Guide)
$\Delta RT_{NDT\ surface} = [CF]\ f^{(0.28 - 0.10 \log f)}$

Chemistry factor CF = $108^\circ F$

Fluence: $4.0 \times 10^{17}\ n/cm^2$
$f = 4.0 \times 10^{17-19} = 4.0 \times 10^{-2}$
$\Delta RT_{NDT} = 108 \times (4.0 \times 10^{-2})^{(0.28 - 0.10 \log 0.04)}$

$= \underline{27.96}^\circ F$

II Plate: Cu = 0.05%, P = 0.015%, Ni = 0.73% (max)
CF = $31^\circ F$, Fluence $4.0 \times 10^{17}\ n/cm^2$
$\Delta RT_{NDT} = \frac{31}{108} \times 27.96 = 8.03^\circ F$

III Forging: Cu = 0.05%, P = 0.015%, Ni = 1.0% (max)
CF = $31^\circ F$, Fluence $4.0 \times 10^{17}\ n/cm^2$
$\Delta RT_{NDT} = \frac{31}{108} \times 27.96 = 8.03^\circ F$

B. For Fluence 6.0×10^{17} n/cm^2 after 60 years

$f = 6.0 \times 10^{(17-19)} = 0.06$

I **Weld Metal**

$CF = 108^\circ F$

$\Delta RT_{NDT} = 108 \times (0.06)^{(0.28 - 0.10 \log 0.06)}$

$= \underline{34.83}^\circ F$

II, III **Forging and Plate**

$CF = 31^\circ F$

$\Delta RT_{NDT} = 34.83^\circ F \times \frac{31}{108} = 10^\circ F$

$\underline{RT_{NDT}}$ requirements per Vessel Specifications (Maximum Specified Values)

Shell courses, and nozzles	$-20^\circ F$
Weld	$-20^\circ F$

$\underline{RT_{NDT}}$ Shifts per new Regulatory Guide 1.99 (February 1986 Draft)

Material	ABWR Initial RT_{NDT}	Calculated Shift ΔRT at surface 4×10^{17} n/cm^2 after 40 years	Margin = $2\sqrt{\sigma_I 2 - \sigma_\Delta 2}$	Final RT_{NDT} (ART)
Weld	$-20^\circ F$	$27.96^\circ F$	$27.96^\circ F$	$36^\circ F$
Plate or Forging	$-20^\circ F$	$8.03^\circ F$	$8.0^\circ F$	$-4^\circ F$

Final RT_{NDT} = ART = Initial RT_{NDT} + ΔRT + Margin

The above projections are for the 40 year full power basis. The corresponding final RT_{NDT} for 60 years would be $50^\circ F$ for welds and $0^\circ F$ plates and forgings.

DROP IN UPPER SHELF ENERGY

Ref: Regulatory Guide 1.99 Rev 2

Material	% Cu	% Drop per Fig 2 of RG 1.99*
Weld	.08 max	14
Base Metal	.05 max	11

* *Based on cutoff fluence of 10^{18}*

INITIAL VALUE 75 FT-lb

Final Values

Weld	75 x 0.86 = 65
Base	75 x 0.83 = 67

QUESTION 251.6

Subsection 5.3.2.1 should clarify where "Reference 2" is located. Has the NRC staff reviewed and approved Reference 2? If not the staff needs to review Reference 2 in order to complete the review of this subsection.

RESPONSE 251.6

Reference 2, *Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors*, January 1979, (NEDO-21778-A), is an NRC staff approved licensing topical report. This topical report was approved by letter to GE, dated November 13, 1978 according to NUREG-0390 Vol.7, No. 2 (October 15, 1984).

QUESTION 251.7

Subsections 5.3.2.1.1, 5.3.2.1.2, 5.3.2.1.3, and 5.3.2.1.5 need to be rewritten. The level of detail must be comparable to that of Standard Review Plan 5.3.2 and Branch Technical Position MTEB 5-2.

RESPONSE 251.7

Response to this question is provided in revised Subsections 5.3.2.1, 5.3.2.1.1, and 5.3.2.1.5.

QUESTION 251.8

Subsection 5.3.3 cited three GE documents:

(1) GE quality assurance program,

(2) "Approved" inspection procedures, and

(3) NEDO-10029

Has the NRC staff reviewed and approved the above documents? The staff cannot satisfactorily review this subsection without reviewing the above three documents.

RESPONSE 251.8

The GE quality assurance program is contained in topical report NEDO-11029-04A, *GE BWR Quality Assurance Program*, Revision 7, which has been approved by the NRC staff (May 1987).

"Approved inspection procedures" refers to GE approved inspection procedures which govern the manufacturing, fabrication, and testing operations of the reactor vessel fabrication process. These inspection procedures are originated at the time the reactor vessel fabricator is selected, and, as has been the case in the past, the NRC staff will have review opportunities in accordance with 10CFR50 Appendix B.

NEDO-10029, *An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident*, July 1969, was also referenced in Subsection 5.3.3 of GESSAR, Docket No. STN-50-447. This information applies equally well to the ABWR.

QUESTION 251.9

Subsection 5.3.3.1.1.1 discusses the 60-year life of the ABWR reactor vessel. The NRC requirements and calculations on the fracture toughness and material properties are based on a 40-year life. Provide justification for the applicability of the NRC's requirements on the 60-year life reactor vessel.

RESPONSE 251.9

Response to this question is provided in revised Subsection 5.3.3.1.1.1.

QUESTION 251.10

Subsection 5.3.3.2 should include the following information: neutron fluence, shift in reference temperature RT_{NDT}, and upper shelf energy. The staff needs this information to compare to that of predicted values using Regulatory Guide 1.99.

RESPONSE 251.10

Response to this question is provided in revised Subsection 5.3.3.2.

QUESTION 251.11

Subsection 5.3.3.6 should indicate that operating conditions should satisfy the pressure-temperature limits prescribed in Subsection 5.3.2.

RESPONSE 251.11

Response to this question is provided in revised Subsection 5.3.3.6.

QUESTION 252.1

Subsection 4.5.1.1 (1) should state: "The properties of the materials selected for the control rod drive mechanism must be equivalent to those given in Appendix I to Section III of the ASME Code or parts A and B of Section II of the ASME Code or are included in Regulatory Guide 1.85, except that cold-worked austenitic stainless steels should have a 0.2% offset yield strength no greater than 90,000 psi."

RESPONSE 252.1

Response to this question is provided in revised Subsection 4.5.1.1 (1).

QUESTION 252.2

Subsection 4.5.1.1 (2) should state: "All materials for use in this system must be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code."

RESPONSE 252.2

Response to this question is provided in revised Subsection 4.5.1.1 (2).

QUESTION 252.3

Subsection 4.5.2.2: The first sentence should read, "Core support structures are fabricated in accordance with the requirements of ASME Code, Section III, Subsection NG-4000, and the examination and acceptance criteria shown in NG-5000."

RESPONSE 252.3

Response to this question is provided in revised Subsection 4.5.2.2.

QUESTION 252.4

Subsection 4.5.2.3: The following statement should be added to the last sentence of the first paragraph, "The examination will satisfy the requirements of NG-5300."

RESPONSE 252.4

Response to this question is provided in revised Subsection 4.5.2.3.

QUESTION 252.5

Subsection 4.5.2.4 should state: "Furnace sensitized material should not be allowed."

RESPONSE 252.5

Response to this question is provided in revised Subsection 4.5.2.4.

QUESTION 252.6

Subsection 4.5.2.5 should state: "All materials used for reactor internals will be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls will preclude contamination of nickel base alloys by chloride ions, fluoride ions, or lead."

RESPONSE 252.6

Response to this question is provided in revised Subsection 4.5.2.5.

QUESTION 252.7

Subsection 5.2.3.2.2 is mostly an academic discussion of BWR water chemistry effect on intergranular stress corrosion cracking (IGSCC) in sensitized stainless steels. The subsection should discuss the actual ABWR water chemistry effects on IGSCC. The subsection is vague about specific remedies or preventive measures to avoid IGSCC in ABWR. For example, the subsection failed to discuss how much hydrogen is needed for injection into the feedwater system or how the "tight conductivity control" would be implemented.

Also, provide references for the "Laboratory studies..." and "available evidence..." that were mentioned in this subsection.

RESPONSE 252.7

Response to this question is provided in revised Subsection 5.2.3.2.1.

QUESTION 252.8

Subsection 5.2.3.2.3 should state that the requirements of GDC 4, relative to the compatibility of components with environmental conditions, are met by compliance with the applicable provisions of the ASME Code and by compliance with the recommendation of Regulatory Guide 1.44.

Specify the "very low limits" of the contaminants in the reactor coolant.

RESPONSE 252.8

Response to this question is provided in revised Subsection 5.2.3.2.3.

QUESTION 252.9

Subsection 5.2.3.3.1 should clarify where and how the 45 ft-lb Charpy V value was obtained.

The ferritic material used for piping, pumps, and valves should comply with Appendix G, Section G-3100, of ASME Code Section III.

This subsection should indicate: "calibration of instruments and equipment shall meet the requirements of the code, Section III, Paragraph NB-2360."

RESPONSE 252.9

Response to this question is provided in revised Subsection 5.2.3.3.1.

QUESTION 252.10

Subsection 5.2.3.4.1.1 should be rewritten to include more detailed discussion on avoidance of significant sensitization and on how the ABWR design complies with the NRC regulatory requirements.

RESPONSE 252.10

Response to this question is provided in revised Subsection 5.2.3.4.1.1.

QUESTION 252.11

Subsection 5.2.3.4.2.3 states that the ABWR design meets the intent of this Regulatory Guide (1.71) by utilizing the alternate approach given in Section 1.8. We cannot review this subsection because we have not received Section 1.8. In addition, this subsection should be rewritten because it lacks detailed discussion about welder qualification.

RESPONSE 252.11

Response to this question is provided in revised Subsection 5.2.3.4.2.3.

QUESTION 281.1

In Section 5.1 (page 5.1-2) the function of the reactor cleanup system filter demineralizer should include the removal of radioactive corrosion and fission products in addition to particulate and dissolved impurities.

RESPONSE 281.1

Response to this question is provided in revised Subsection y.y.y.y.y.

QUESTION 281.2

In Subsection 5.2.3.2.2 (page 5.2-7) irradiation-assisted stress corrosion cracking (IASCC) of reactor internal components and its mitigation are not discussed. Present laboratory data and plant experience has shown that IASCC can be initiated even at low conductivity (< 0.3μS/cm) after long exposure to radiation.

RESPONSE 281.2

Response to this question is provided in the new Subsection 5.2.3.2.4, *IGSCC Considerations*.

QUESTION 281.3

In Subsection 5.2.3.2.2 (pages 5.2-7 and 8) the ABWR standard plant design does not clearly incorporate hydrogen water chemistry to mitigate IGSCC. Since the plant design life is 60 years, hydrogen water chemistry may be of greater importance in reducing reactor coolant electrochemical corrosion potential to prevent IGSCC as well as IASCC. If hydrogen water chemistry is the referenced ABWR standard design, the following documents should be cited:

EPRI NP-5283-SR-A, *Guidelines for Permanent BWR Hydrogen Water Chemistry Installations* - 1987 Revision.

EPRI NP-4947-SR-LD, *BWR Hydrogen Water Chemistry Guidelines* - 1987 Revision (to be published).

RESPONSE 281.3

Subsection 5.2.3.2.2 will be modified by September 30, 1988 to more clearly discuss hydrogen water chemistry as part of the ABWR standard plant design.

QUESTION 281.4

In Subsection 5.2.3.2.2 (page 5.2-9) the utilization of the General Electric zinc injection passivation (GEZIP) process for radiation buildup control for the ABWR is not discussed. GEZIP was identified as a required design feature in the ABWR presentation to NRC staff.

RESPONSE 281.4

The General Electric zinc injection passivation process (GEZIP) is not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.7.6) that requires the remainder of the plant to meet the water quality requirements of Table 5.2-5.

QUESTION 281.5

In Subsection 5.2.3.2.2 (page 5.2-9) prefilming of stainless steel appears to be a promising method to reduce the buildup rate of activated corrosion products during subsequent plant operation. SIL No. 428 recommends preoperational testing of the recirculation system conducted at temperatures 230°F be done with the dissolved oxygen level controlled to between 200 and 400 ppb. Is control of radiation buildup through preoperational oxygen control being considered for the BWR standard plant? Are mechanical polishing and electropolishing of piping internal surfaces also being considered for reducing radiation buildup?

RESPONSE 281.5

Since the recirculation system piping has been eliminated from the ABWR design, SIL No. 428 does not apply. Preoxidation, mechanical polishing, and electropolishing are not being considered for other ABWR components at this time. However, these methods are available as promising techniques to reduce radiation buildup on all internal stainless steel surfaces.

QUESTION 281.6

In Subsection 5.2.3.2.2.2 (page 5.2-9) cobalt 60 is identified as the principle contributor to shutdown radiation levels, especially the recirculation piping system of BWRs. Stellite contributes about 90% of the total cobalt 59 input to the reactor water (EPRI NP-2263, *BWR Cobalt Source Identification*, February 1982). Since irradiation of cobalt 59 yields cobalt 60, reduction in the source of cobalt 59 is needed to reduce the buildup of shutdown radiation levels. Indicate Stellite surface areas (square feet) in nuclear steam supply system and balance of plant. Provide the criteria for selecting Stellite plant materials for the designed application. Provide evaluation of noncobalt-containing materials whose properties are adequate to replace Stellite in-plant applications.

RESPONSE 281.6

(1) Stellite Surface Area for BWR/6:

Total Nuclear Steam Supply System:	74.39 Sq. Ft.
Total Balance of Plant:	138.0 Sq. Ft.

For ABWR design, the above numbers are greatly reduced. Cobalt-based alloys have been eliminated from fuel assemblies, and control rod blades and drives.

(2) Criteria for Selecting Stellite Materials:

1. Wear resistance

2. Weldability

3. Experience and service history

4. Radiation level in area of application

(3) Evaluation of Noncobalt-containing Material to Replace Stellite:

The major source of cobalt from the reactor core has been Haynes 25 and Stellite 3 (cobalt-based alloys) for pins and rollers, respectively, in BWR control rods. Replacement of the cobalt alloy pins and rollers with noncobalt alloys has been extensively investigated under a joint GE-EPRI program (Project 1331-1). The results of this investigation are documented in the report, EPRI NP-2329, *Project 1331-1, Final Report*, March 1982. The current design noncobalt materials are alloy X-750 for control rod rollers and 13-8 PH for the pins.

QUESTION 281.7

Subsection 5.2.3.2.2.3(4) (page 5.2-10) states that control of reactor water oxygen during startup/hot standby may be accomplished by utilizing the de-aeration capabilities of the condenser. In addition, this section states that independent control of control rod drive (CRD) cooling water oxygen concentrations of < 50 ppb during power operation is desirable to protect against IGSCC of CRD materials. Are either one or both of the above dissolved oxygen controls incorporated in the ABWR standard plant design?

RESPONSE 281.7

In Subsection 5.2.3.2.2.3, control of reactor water oxygen by using the condenser and control of control rod drive water were mentioned as dissolved oxygen control methods. These two plant features are not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.2.5) that requires the remainder of the plant to meet the water quality requirements of Table 5.2-5.

QUESTION 281.8

In Subsection 5.2.3.2.2.3(13) (page 5.2-11) it states that the main steam line radiation monitor indicates an excessive amount of hydrogen being injected. An explanation of this occurrence should be discussed.

RESPONSE 281.8

Subsection 5.2.3.2.2.3(13) will be revised by September 30, 1988 to discuss the effects of excessive hydrogen injection upon the main steam line radiation monitor.

QUESTION 281.9

Subsection 6.4.4.2 (page 6.4-6) discusses personnel respirator use in the event of toxic gas intrusion into the control room. However, the chlorine detection system is not discussed. Also, any control functions that are automatically triggered by a chlorine detector alarm (closing intake dampers, energizing control room HVAC system recirculation) should be identified.

RESPONSE 281.9

Response to this question will be provided upon submittal of control room HVAC system scheduled for December 31, 1988.

QUESTION 281.10

In the October 1987 ABWR presentation to the NRC staff the design features and/or requirements to improve water chemistry for GE-ABWR were specified. Address each one of these design features and/or requirements listed in Table I in the ABWR Standard Safety Analysis Report.

TABLE I

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

		ABWR Presentation to NRC Staff	ABWR Standard Safety Analysis Report
1 -	Selection of low cobalt materials to minimize radiation buildup	Required Design Feature	Not discussed in Subsection 5.2.3.
2 -	Hydrogen water chemistry to suppress IGSCC	Required Design Feature	Subsection 5.2.3.2.2 references normal water chemistry.
3 -	Zinc injection to minimize radiation buildup	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.2.
4 -	Full flow deep bed condensate system to reduce feedwater impurities	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.
5 -	Improved online monitoring instrumentation to assure water quality	Ion chromatography, electrochemical corrosion potential, and crack arrest verification system required design features	Only electrochemical corrosion potential discussed in Subsection 5.2.3.2.2.3.
6 -	Improved corrosion-resistant materials for steam extraction piping to minimize feedwater impurities	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.

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 lead 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

After the first paragraph in Subsection 5.2.3.2.2.2, add the following as a second paragraph:

As a means to reduce cobalt, GE has reduced cobalt content in alloys to be used in high fluence areas such as fuel assemblies and control rods. In addition, cobalt base alloys used for pins and rollers in control rods have been replaced with noncobalt alloys.

Item 2

Subsection 5.2.3.2.2 will be revised by September 30, 1988 to reference the EPRI guidelines for hydrogen water chemistry and for installation of the facilities.

Item 3

Information is being obtained and evaluated from operating plants with GEZIP. However, this feature is not in the Nuclear Island scope.

Item 4

This feature is not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.2.6) that requires the remainder of the plant to meet the water quality requirements in Table 5.2-5.

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

Table 5.2-5 will be revised by September 30, 1988 to include control of ECP.

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

In the ABWR standard plant design, a 2% reactor water cleanup system is provided. By September 30, 1988 Subsection 5.2.3.2.2 will be changed to discuss this.

Item 12

This design feature is not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.2.6) that requires the remainder of the plant to meet the water quality requirements in Table 5.2-5.

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 FSAR 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.

RESPONSE 470.4

The evaluation of the loss of coolant accident (LOCA) involved several assumptions which differ for those outlined in Regulatory Guide 1.3 and SRP 15.6.5. Each assumption is shown in Table 20.1-2 with an associated explanatory paragraph below. In addition, the estimated dose for the two-hour site boundary dose at 300 meters and the LPZ 30-day dose at 800 meters is given in Table 20.3-1 for each assumption.

(1) 1 Hour Release Following Scram. The ABWR incorporates a redundant emergency core cooling system (ECCS) to supply makeup water in the event of a LOCA. The ECCS is sized so that in such an event sufficient water is supplied to insure that core uncovery does not occur. Therefore, the assumptions as to fission product release under Regulatory Guide 1.3 for a LOCA with proper operation of the ECCS are not justified. However, given a potential spectrum of failure of equipment or operator error in conjunction with a LOCA, core uncovery is justified ranging on a time scale of a few tens of minutes for total failure of all systems, to several days for gradual deterioration of equipment. Based upon evaluations of ECCS responses to a wide variety of conditions, it is reasonable to assume that core uncovery would not proceed for a minimum of one hour given the single failure proof design of the system.

(2) Primary Containment Leakage. Following a LOCA case, Regulatory Guide 1.3 stipulated the containment leakage should remain constant for 30 days. Regulatory Guide 1.4 (PWR) permits a reduction by a factor of two 24 hours after a LOCA. Containment leakage is proportional to containment pressure assuming that design leakage is not significantly exceeded. The analysis of containment pressure given in Section 6.2 and long term studies under a variety of conservative assumptions show that the ABWR primary containment pressure is a factor of two below design pressure within 12 hours following a LOCA and decreasing slowly after that. Based upon this type of evaluation, the reduction in leakage by a factor of two 24 hours after a LOCA is justifiable.

(3) Iodine Release Fractions. The release of substantial quantities (>10%) of iodine from the core of a nuclear reactor predicates significant damage to the fuel and the associated fuel assemblies. The only means by which such damage might be sustained is extensive high temperatures leading to fuel melt. Such damage, even though partial such as at TMI, will result in core conditions resulting in the evolution of CsI rather than the I_2 assumed in the regulatory guide (Reference 1). The formation of organic iodides is based upon the release of I_2 and adequate concentrations with organic constituents to form organic iodides (References 2 and 3). Such conditions cannot be reasonably expected since the iodine will be bound as CsI. Therefore, it has been assumed that the formation of iodine species will result in primarily CsI with a minor fraction as is estimated in Reference 1 of organic iodides. Two other points need to be considered. The first is production of organic forms by radiolysis in the suppression pool.

Based upon Reference 4, with pH levels in the wetwell greater than 9, the evolution of iodine species is not expected. The second is consideration of accident situations leading to only minor fuel damage resulting in primarily a fuel gap inventory release. Such a release due primarily to low temperature can be expected to consist of I_2 gas and result in some organic iodide formation. However, such releases are considered under the small line break accident case and control rod drop accident cases.

(4) Suppression Pool Scrubbing. The ability of the suppression pool in the BWR to remove particulate material and elemental iodine has in the past been prohibited under Regulatory Guide 1.3, due, it is thought, to a lack of adequate understanding of the phenomena involved. The ABWR is designed with safety/relief valves and horizontal downcomers integrated into the building to insure that any release of fission product material will be subject to transport to the wetwell via the suppression pool. Over the last several years a preponderance of both empirical and theoretical evidence has been gathered which adequately states the case for suppression pool scrubbing. This has culminated in the development of the GE DECON computer code for evaluation of suppression pool scrubbing and which, when evaluated against the empirical evidence, accurately predicts the empirical results better than any current simulation. Using the DECON code on the conditions expected in the ABWR under LOCA simulation results in overall decontamination factors far in excess of the 100 assumed in the analysis. Therefore, it was considered reasonable in light of the current knowledge to assume a conservative overall decontamination factor for the pool of 100.

(5) MSIV Leakage. The evaluation of potential leakage during a LOCA from the main steam lines has centered on the leakage of the MSIVs and potential for direct release to the environment. This has in the past resulted in the issuance of Regulatory Impact Issue C-8 by the NRC and the use of main steam leakage control systems. Over the past several years, considerable effort has been expended on this subject by the BWR Owners Group and the NRC and has resulted in a series of reports and maintenance procedures for utilities. The ABWR technical specification of MSIV performance recognized potential seating and leakage problems and therefore uses a graduated leakage performance criteria shown in Figure 15.6-2. The evaluation of radionuclide leakage from these valves were then made in accordance with the procedure given in Reference 5, except as noted below.

(a) The primary containment served as a single large repository for fission products from which leakage was derived for the pathways via the reactor building and the MSIVs. Material directly injected from the pressure vessel to the drywell were assumed over a short period of time to cycle into the wetwell. Following pressure suppression, the wetwell airspace would then be considered linked to the drywell airspace via the vacuum breakers. Therefore, a multiple flow path in primary containment was not evaluated.

(b) Flow through each steamline was considered independently at that line flow rate (see Figure 15.6-2). The transport time down each line was considered at a rate three times the plug flow rate specified in Reference 5. This value is a rule of thumb derived from experience in flow through large pipes, and when compared to the results of Reference 6, is similar.

(c) Plateout in the steamlines was not considered for the first 48 hours after the LOCA to allow for line cooling. This was adopted as conservative, based upon the arguments found in Reference 6 (page F-3). The plateout model is found in Reference 5.

(d) The primary controlling factor for MSIV leakage is condenser plateout and leakage. For the condenser, a single mixed volume equal to one-half of the free air volume in the condenser was assumed. Leakage from the condenser assumed that the total in-leakage from the steam lines was non-condensible plus an additional leakage of 100 ft^3 per hour based upon barometric pressure changes from Reference 6 (page F-3). Such a leakage is considered conservative since the leakage to the condenser would primarily be condensible and the barometric pressure change required to cause a 100 ft^3 per hour change could be extremely large (a hurricane). The condenser plateout model used was that found in Reference 5.

(e) Following release from the condenser, the meteorology and the health effects model were those assumed in Regulatory Guide 1.3 and are described in SSAR References 2 through 4.

References for Response 470.4

1. *Technical Basis for Estimating Fission Product Behavior During LWR Accidents,* NUREG-0772, June 1981.

2. Postma, A.K. and Zavadoski, R.W., *Review of Organic Iodide Formation Under Accident Conditions in Water-Cooled Reactors,* WASH-1233, October 1972.

3. Malinauskas, A.P. and Bell, J.T., "The Chemistry of Fission Product Iodine Under Nuclear Reactor Accident Conditions," *Nuclear Safety,* Vol. 28, No. 4, Oct-Dec 1987.

4. Lin, C.C., "Chemical Effects of Gamma Radiation on Iodine in Aqueous Solutions, *Journal of Inorganic and Nuclear Chemistry,* Vol. 42 (1980) 1101-1110.

5. Careway, H.A. et al, *A Technique for Evaluation of BWR MSIV Leakage Contribution to Radiological Dose Rate Calculations,* NEDO-30259, GE, Sept. 1985.

6. Ridgely, J.N. and Wohl, M.L., *Resolution of Generic Issue C-8,* NUREG-1169, Aug. 1986.

QUESTION 470.5

Provide a discussion of, or reference to, the analysis of the radiological consequences of leakage from engineered safety feature components after a design basis LOCA.

RESPONSE 470.5

Leakage from engineered safety features are not specifically analyzed. The total leakage from the primary containment is restricted to 0.5% per day for all leakage except that through the main steam line isolation valves. Leakage from engineered safety features is then included in the 0.5% per day such that all leakage from equipment external to the primary containment shall not result in an airborne release which when combined with the containment leakage shall result in an equivalent release greater than 0.5% per day.

QUESTION 470.6

For the spent fuel cask drop accident, what is the assumed period for decay from the stated power condition? What is the justification for that assumption?

RESPONSE 470.6

Table 15.7-12 has been corrected by changing "core" to "storage" under item I.E. The cask drop accident assumes a 1000 day exposure prior to removal from the core with a radial peaking factor of 1.5. Decay time upon removal from the core is 120 days prior to the accident. This 120 day period was conservatively estimated at one-third of a year, since based upon current practice the minimum time to ship fuel to a long term storage facility is one year (in the case of the GE Morris facility) and 10 years in the case of government storage facilities.

QUESTION 470.7

The tables in Chapter 15 should be checked and revised as appropriate. In several cases the footnotes contain typographical errors related to defining the scientific notation. Table 15.7-12 also appears to contain inappropriate references to Table 15.7-16, rather than Table 15.7-13.

RESPONSE 470.7

The response to this question is provided in revised Tables 15.7-10, 15.7-12, and 15.7-13.

QUESTION 470.8

It is stated that Regulatory Guides 1.3 and 1.145 were used in the calculations of X/Q values. Based on the values presented, it appears as though a Pasquill stability Class F and one meter per second wind speed were assumed, with adjustment for meander per Figure 3 of Regulatory Guide 1.145. If this is not the case, describe the assumptions and justification used in calculating the X/Q values which are used in the Chapter 15 dose assessments.

RESPONSE 470.8

All meteorological calculations in Chapter 15 were made based upon the equations given in Regulatory Guide 1.145 and the tables in Regulatory Guide 1.3.C.2.g(3). Calculations have been encoded into a computer program for routine use and are detailed in report NEDO-20804, *Atmospheric Dispersion CHIQUO2 Function*, Feb. 1979. In all cases, a ground level release was assumed, and where permitted by regulatory guide or SRP, building wake and plume meander accounted for. In addition, the basis for meteorological calculations is found in Appendix B of Reference 2, SSDR Subsection 15.6.7.

QUESTION 470.9

The SGTS filter efficiencies of 99% for inorganic and organic iodine are higher than the 90% and 70% values, respectively, assumed in Regulatory Guide 1.25 if it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine. Provide a justification for the use of the higher values.

RESPONSE 470.9

The ABWR incorporates a 6-inch charcoal bed in the SGTS filter train, and in accordance with Table 2 of Regulatory Guide 1.52 is permitted a removal efficiency for both elemental and organic forms of iodine of 99%.

QUESTION 470.10

Dose related factors such as breathing rates, iodine conversion factors and finite vs. infinite cloud assumptions for calculating the whole body dose are not stated explicitly, although reference is made to Regulatory Guide 1.25 and another document. State these assumptions explicitly and justify the use of any values which deviate from Regulatory Guide 1.25.

RESPONSE 470.10

In all cases except the control room evaluation, a semi-infinite cloud model was used to calculate dose conversion factors. This model was based upon Regulatory Guide 1.3 and Slade's *Meteorology and Atomic Energy - 1968*. A detailed explanation of the model with related factors is found in Appendix C in Reference 2 of SSAR Subsection 15.6.7. In the case of the control room dose, the dose model was a finite cloud model to account for the limited size of the control room and is given in Section 2.5 of Reference 3 of SSAR Subsection 15.6.7.

TABLE 20.3-1

SENSITIVITY STUDY OF PARAMETERS FOR LOCA ANALYSIS

		Site Boundary 24 Hr. Dose at 300 m (REM)		LPZ Dose for 30 Days at 800 m (REM)	
		Thyroid	Whole Body	Thyroid	Whole Body
1.	LOCA Results	1.5	0.62	22.	12.
2.	No Initial 1 Hr. Hold-up	1.5	0.90	22	13
3.	No Pressure Reduction @ 24 Hrs	NC	NC	22	13
4.	Iodine Species Consistent with Regulatory Guide 1.3	10.0	0.64	1700	13
5.	No Suppression Pool Scrubbing	140.	0.92	930	13
6.	No Steamline Plateout	1.5	0.62	23	12
7.	No Steamline Plateout or Hold-up	1.5	0.64	23	12
8.	No Condenser Plateout	2.3	0.62	340	12
9.	No Condenser Plateout or Hold-up	280	41	1300	70

NOTE:

All evaluations are made independently of each other.

4.5 REACTOR MATERIALS

4.5.1 Control Rod Drive System Structural Materials

4.5.1.1 Material Specifications

(1) Material List

The following material listing applies to the control rod drive mechanism supplied for this application. The position indicator and minor non-structural items are omitted.

Q252.1 The properties of the materials selected for the control rod drive mechanism shall be equivalent to those given in Appendix I to Section III of the ASME Code or parts A and B of Section II of the ASME Code or are included in Regulatory Guide 1.85 except that cold worked austenitic stainless steels shall have a 0.2% offset yield strength no greater than 90,000 psi.

(a) Spool Piece Assembly

Spool Piece Housing	ASME 182 Grade F304L
Seal Housing	ASME 182 Grade F304L
Drive Shaft	ASME 479 Grade XM-19 (Hardsurfaced with Colmonoy No. 6)
Ball Bearings	440C
Gland Packing	Asbestos
Gland Packing Spring	Inconel X-750

(b) Ball Spindle

Ball Screw Shaft	ASTM A-564 TP630 (17-4) Condition H-1100
Ball Nut	ASTM A-564 TP630 (17-4) Condition H-1100
Balls	440C
Guide Roller	Stellite No.3
Guide Roller Pin	Haynes Alloy No. 25
Guide Shaft	Stellite No. 6
Guide Shaft Bushing	Stellite No. 12
Separation Spring	Inconel X-750
Separation Magnet	Alnico No. 5

(c) Buffer Mechanism

Buffer Spring	Inconel X-750
Buffer Sleeve	316L (Hardsurfaced with Colmonoy No. 6)
Guide Roller	Stellite No. 3
Guide Roller Pin	Stellite No. 25
Buffer Cone	316L (Hardsurfaced with Stellite No. 6)

(d) Hollow Piston

Piston Tube	XM-19
Piston Head	316L (Hardsurfaced with Stellite No. 3)
Latch	Inconel X-750
Latch Spring	Inconel X-750
Coupling Spud	Inconel X-750

(e) Guide Tube

Guide Tube	316L

(f) Outer Tube Assembly

Outer Tube	XM-19
Flange	ASME SA182 Grade F304L

(g) Miscellaneous Parts

Ball for Check Valve	Haynes Alloy
O-Ring Seal (Between CRD Housing and CRD)	321SS Coated with Teflon
CRD Installation Bolts	ASME SA193 Grade B7

(2) Special Materials

The coupling spud, latch and latch spring, separation spring and gland packing spring are fabricated from Alloy X-750 in the annealed or equalized condition, and aged 20 hours at 1300°F to produce a tensile of 165,000 psi minimum, yield of 105,000 psi minimum, and elongation of 20% minimum. The ball screw shaft and ball nut are ASTM A-564, TP 630 (17-4) (or its equivalent) in condition H-1100 (aged 4 hours at 1100°F), with a tensile of 140,000 psi minimum, yield of 115,000 psi minimum, and elongation of 15% minimum.

These are widely used materials, whose properties are well known. The parts are readily accessible for inspection and replaceable if necessary.

Q252.2 All materials for use in this system shall be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code.

All materials, except SA479 or SA249 Grade XM-19, have been successfully used for the past 15 to 20 years in similar drive mechanisms. Extensive laboratory tests have demonstrated that ASME SA479 or SA249 Grade XM-19 are suitable materials and that they are resistant to stress corrosion in a BWR environment.

No cold-worked austenitic stainless steels with a yield strength greater than 90,000 psi are employed in the Control Rod Drive (CRD) system.

4.5.1.2 Austenitic Stainless Steel Components

(1) Processes, Inspections and Tests

There is a special process employed which subjects selected 300 Series stainless steel components to temperatures in the sensitization range. The drive shaft, buffer sleeve, piston head and buffer are hard surfaced with Colmonoy 6 (or its equivalent). Colmonoy (or its equivalent) hard-surfaced components have performed successfully for the past 15 to 20 years in drive mechanisms. It is normal practice to remove some CRDs at each refueling outage. At this time, the Colmonoy (or its equivalent) hard-surfaced parts are accessible for visual examination. This inspection program is adequate to detect any incipient defects before they could become seri- ous enough to cause operating problems. The degree of conformance to Regulatory Guide 1.44 is presented in Subsection 4.5.2.4.

(2) Control of Delta Ferrite Content

Discussion of this subject and the degree of conformance to Regulatory Guide 1.31 is presented in Subsection 4.5.2.4.

4.5.1.3 Other Materials

These are presented in Subsection 4.5.1.1(2)

4.5.1.4 Cleaning and Cleanliness Control

All the CRD parts listed in Subsection 4.5.1.1 are fabricated under a process specification which limits contaminants in cutting, grinding and tapping coolants and lubricants. It also restricts all other processing materials (marking inks, tape etc.) to those which are completely removable by the applied cleaning process. All contaminants are then required to be removed by the appropriate cleaning process prior to any of the following:

(1) Any processing which increases part temperature above 200°F.

(2) Assembly which results in decrease of accessiblity for cleaning.

(3) Release of parts for shipment.

The specification for packaging and shipping the Control Rod Drive provides the following:

The drive is rinsed in hot deionized water and dried in preparation for shipment. The ends of the drive are then covered with a vapor tight barrier with dessicant. Packaging is designed to protect the drive and prevent damage to the vapor barrier. Audits have indicated satisfactory protection.

Semiannual examination of the humidity indicators of ten percent of the units is required to verify that the units are dry and in satisfactory condition. This inspection shall be performed with a GE-Engineering designated representative present. The position indicator probes are not subject to this inspection.

Site or warehouse storage specifications require inside heated storage comparable to level B of ANSI N45.2.2.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37.

4.5.2 Reactor Internal Materials

4.5.2.1 Material Specifications

Materials used for the Core Support Structure:

Shroud Support - Nickel-Chrome-Iron-Alloy, ASME SB166 or SB168.

Shroud, core plate, and grid - ASME SA240, SA182, SA479, SA312, SA249, or SA213 (all Type 304L or 316L).

Peripheral fuel supports - ASME SA312 Grade Type-304L or 316L.

Core plate and top guide studs, nuts, and sleeves. ASME SA-479 (Type 304, 316, or XM-19) (all parts); or SA-193 Grade B8 (studs); or SA-194 Grade 8 (Type 304) (nuts); or SA-479 (Type 304L or 316L), SA-182 (Grade F304L or F316L), SA-213 (Type 304L, 316 or 316L), SA-249 (Type 304L, 316, or 316L) (sleeves).

Control rod drive housing. ASME SA-312 Grade TP304L or 316L SA-182 Grade F304L or F316L, and ASME SA-351 Type CF3 (Type 304L) or Type CF3M (Type 316L).

Control rod guide tube. ASME SA-351 Type CF3, or SA-358, SA-312, or SA-249 (Type 304L or 316L) or ASME SA-351 Type CF3M (Type 316L).

Orificed fuel support. ASME SA-351 Type CF3 (Type 304L) or CF3M (Type 316L).

Materials employed in shroud head and separator assembly and steam dryer assembly:

All materials are 304L or 316L stainless steel:

Plate, Sheet--ASTM A240 Type 304L or 316L and Strip

Forgings--ASTM A182 Grade 304L

Bars--ASTM A276 Type 316L

Pipe--ASTM A312 Grade TP-304L

Tube--ASTM A269 Grade TP-304L

Castings--ASTM A351 Grade CF8

All core support structures are fabricated from ASME specified materials, and designed accordance with requirements of ASME Code, Section III, Subsection NG. The other reactor internals are noncoded, and they are fabricated from ASTM or ASME specification materials.

4.5.2.2 Controls on Welding

Core support structures are fabricated in accordance with requirements of ASME Code Section III, Subsection NG-4000 and the examination and acceptance criteria shown in NG-5000. Other internals are not required to meet ASME Code requirements. ASME Section IX BPV code requirements are followed in fabrication of core support structures. Q252.3

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products

Wrought seamless tubular products for CRD housings, and peripheral fuel supports, are supplied in accordance with ASME Section III, Class CS, which requires examination of the tubular products by radiographic and/or ultrasonic methods according to paragraph NG-2550, the examination will satisfy the requirements of NG-5000. Q252.4

Wrought seamless tubular products for other internals were supplied in accordance with the applicable ASTM or ASME material specifications. These specifications require a hydrostatic test on each length of tubing.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel - Regulatory Guide Conformance

Q252.5 Cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. Furnance sensitized material shall not be allowed. The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 Ferrite Number (FN) minimum and 8.0 FN average. This ferrite content is considered adequate to prevent any micro-fissuring (Hot Cracking) in austenitic stainless steel welds. This procedure complies with the requirements of Regulatory Guide 1.31.

Proper solution annnealing of the 300 series austenitic stainless steel is verified by testing per ASTM-A262, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels." Welding of austenitic stainless steel parts is performed in accordance with Section IX (Welding and Brazing Qualification) and Section II Part C (Welding Rod Electrode and Filler Metals) of the ASME Boiler and Pressure Vessel Code. Welded austenitic stainless steel assemblies require solution annealing to minimize the possibility of the sensitizing. However, welded assemblies are dispensed from this requirement when there is documentation that welds are not subject to significant sustained loads and assemblies have been free of service failure. Other reasons, in line with Regulatory Guide 1.44, for dispensing with the solution annealing are that assemblies are exposed to reactor coolant during normal operation service which is below 200°F temperature or assemblies are of material of low carbon content (less that 0.025%). These controls are employed in order to comply with the intent of the Regulatory Guide 1.44.

Exposure to contaminant is avoided by carefully controlling all cleaning and processing materials which contract stainless steel during manufacture and construction. Any inadvertent surface contamination is removed to avoid potential detrimental effects.

Special care is exercised to insure removal of surface contaminants prior to any heating operation. Water quality for rinsing, flushing, and testing is controlled and monitored.

The degree of cleanliness obtained by these procedures meets the requirements of Regulatory Guide 1.37.

4.5.2.5 Other Materials

Hardenable martensitic stainless steel and precipitation hardening stainless steels are not used in the reactor internals.

Materials, other than Type-300 stainless steel, employed in reactor internals are:

(1) SA479 Type XM-19 stainless steel;

(2) SB166, 167, and 168, Nickel-Chrome-Iron (Alloy 600); and

(3) SA637 Grade 688 Alloy X-750.

Alloy 600 tubing, plate, and sheet are used in the annealed condition. Bar may be in the annealed or cold-drawn condition.

Alloy X-750 components are fabricated in the annealed or equalized condition and aged when required.

Stellite 6 (or its equivalent) hard surfacing is applied to some austenitic stainless steel castings using the gas tungsten arc welding or plasma arc surfacing processes.

All materials used for reactor internals shall be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls will preclude contamination of nickel base alloys by chloride ions, fluoride ions or lead. Q252.6

All materials, except SA479 Grade XM-19, have been successfully used for the past 15 to 20 years in BWR applications. Extensive laboratory tests have demonstrated that XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

The main steamline flow restrictors of the venturi-type are installed in each main steam nozzle on the reactor vessel inside the primary containment. The restrictors are designed to limit the loss of coolant resulting from a main steamline break inside or outside the primary containment. The coolant loss is limited so that reactor vessel water level remains above the top of the core during the time required for the main steamline isolation valves to close. This action protects the fuel barrier.

Two isolation valves are installed on each main steamline. One is located inside, and the other is located outside the primary containment. If a main steamline break occurs inside the containment, closure of the isolation valve outside the primary containment seals the primary containment itself. The main steamline isolation valves automatically isolate the RCPB when a pipe break occurs outside containment. This action limits the loss of coolant and the release of radioactive materials from the nuclear system.

The RCIC system provides makeup water to the core during a reactor shutdown in which feedwater flow is not available. The system is started automatically upon receipt of a low reactor water level signal or manually by the operator. Water is pumped to the core by a turbine pump driven by reactor steam.

The residual heat removal (RHR) system includes a number of pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. The RHR system allows decay heat to be removed whenever the main heat sink (main condenser) is not available (i.e., hot standby). One mode of RHR operation allows the removal of heat from the primary containment following a LOCA. Another operational mode of the RHR system is low pressure flooder (LPFL).

The LPFL is an engineered safety feature for use during a postulated LOCA. Operation of the LPFL is presented in Section 6.3.

The reactor water cleanup system recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities with their associated corrosion and fission products from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

5.1.1 Schematic Flow Diagrams

Schematic flow diagrams (Figures 5.1-1 and 5.1-2) of the RCS show major components, principal pressures, temperatures, flow rates, and coolant volumes for normal steady-state operating conditions at rated power.

5.1.2 Piping and Instrumentation Diagrams

Piping and instrumentation diagrams covering the systems included within RCS and connected systems are presented as follows:

(1) the nuclear boiler system (Figure 5.1-3);

(2) main steam (Figure 5.1-3a &b);

(3) feedwater (Figure 5.1-3c);

(4) recirculation system (Figure 5.4-4);

(5) reactor core isolation cooling system (Figure 5.4-8);

(6) residual heat removal system (Figure 5.4-10); and

(7) reactor water cleanup system (Figure 5.4-12)

5.1.3 Elevation Drawings

An elevation drawing (Figure 5.1-4) shows the principal dimensions of the reactor and connecting systems in relation to the containment.

SECTION 5.2

CONTENTS (Continued)

SECTION 5.2

CONTENTS (Continued)

SECTION 5.2

CONTENTS (Continued)

SECTION 5.2

CONTENTS (Continued)

281.4
281.7
281.10
item 4 & 11

TABLES

ILLUSTRATIONS

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10CFR50, Section 50.55a

Table 3.2-4 shows the compliance with the rules of 10CFR50, Codes and Standards. Code edition, applicable addenda, and component dates will be in accordance with 10CFR50.55a.

5.2.1.2 Applicable Code Cases

The reactor pressure vessel and appurtenances and the RCPB piping, pumps, and valves will be designed, fabricated, and tested in accordance with the applicable edition of the ASME Code, including addenda that were mandatory at the order date for the applicable components. Section
210.1 50.55a of 10CFR50 requires code case approval for Class 1, 2, and 3 components. These code cases contain requirements or special rules which may be used for the construction of pressure retaining components of Quality Group Classification A,
210.1 B, and C. The various ASME code cases that may be applied to components are listed in Table 5.2-1.

Regulatory Guides 1.84 and 1.85 provide a list of ASME Design and Fabrication code cases that have been generically approved by the Regulatory Staff. Code Cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered active for equipment that has been contractually committed to fabrication prior to the annulment.

210.1

5.2.2 Overpressure Protection

This subsection evaluates systems that protect the RCPB from overpressurization.

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with 10CFR50, Appendix A, General Design Criterion 15. Preoperational and startup instructions are given in Chapter 14.

5.2.2.1.1 Safety Design Bases

The nuclear pressure-relief system has been designed to:

(1) prevent overpressurization of the nuclear system that could lead to the failure of the RCPB;

(2) provide automatic depressurization for small breaks in the nuclear system occurring with maloperation of both the reactor core isolation cooling (RCIC) system and the high pressure core spray (HPCS) system so that the low pressure flooder (LPFL) mode of the residual heat removal (RHR) system can operate to protect the fuel barrier;

(3) permit verification of its operability; and

(4) withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, or faulted conditions.

5.2.2.1.2 Power Generation Design Bases

The nuclear pressure-relief system safety/relief valves (SRVs) have been designed to meet the following power generation bases:

(1) discharge to the containment suppression pool, and

(2) correctly reclose following operation so that maximum operational continuity is obtained.

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure under upset conditions as discussed in Subsection S.2.3 of Reference 1.

The SRV setpoints are listed in Table 5.2-3 and satisfy the ASME code specifications for safety valves because all valves open at less than the nuclear system design pressure of 1250 psig.

The automatic depressurization capability of the nuclear system pressure relief system is evaluated in Section 6.3 and Section 7.3.

The following criteria are used in selection of SRVs:

(1) must meet requirements of ASME Code, Section III;

(2) must qualify for 100% of nameplate capacity credit for the overpressure protection function; and

(3) must meet other performance requirements such as response time, etc., as necessary to provide relief functions.

The SRV discharge piping is designed, installed, and tested in accordance with ASME Code, Section III.

5.2.2.1.4 Safety/Relief Valve Capacity

SRV capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components, up to and including applicable addenda. The essential ASME requirements which are met by this analysis follow.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of two sources: a direct or a flux trip signal. The direct scram trip signal is derived from position switches mounted on the main steamline isolation valves, the turbine stop valves, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 15% travel of full stroke. The pressure switches are actuated when a fast closure of the turbine control valves is initiated. Credit is not taken for the power-operated mode. Credit is only taken for the safety/relief valve capacity which opens by the spring mode of operation direct from inlet pressure.

The rated capacity of the pressure-relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110% of the design pressure (1.10 x 1250 psig = 1375 psig) for events defined in Section 15.2.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

Table 5.2-2 lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

See Appendix A, Subsection A.5.2.2.2.1 of Reference 1.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions.

5.2.2.2.2.1 Operating Conditions

(1) operating power = 4005 MWt (102% of nuclear boiler rated power);

(2) vessel dome pressure $\leq$1040 psig; and

material adjacent to welds in Type 304 and Type 316 stainless steel piping systems has occurred in the past. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures. For the ABWR, IGSCC resistance has been achieved through the use of IGSCC resistant materials such as Type 316 Nuclear Grade stainless
252.7 steel and stabilized nickel-base Alloy 600M and 182M.

Much of the early remedy-development work focused on alternative materials or local stress reduction, but recently the effects of water chemistry parameters on the IGSCC process have received increasing attention. Many important features of the relationship between BWR water chemistry and IGSCC of sensitized stainless steels have been identified.

252.7 Laboratory studies (References 3 and 4) have shown that although IGSCC can occur in simulated BWR startup environments, most IGSCC damage probably occurs during power operation. The normal BWR environment during power operation is ~280 °C water containing dissolved oxygen, hydrogen and small concentrations of ionic and nonionic impurities (conductivity generally below
252.7 0.3 μS/cm at 25°C). It has been well documented that some ionic impurities (notably sulfate and chloride) aggravate IGSCC, and a number of studies have been made of the effects of individual impurity species on IGSCC initiation and
252.7 growth rates (References 3 thru 7). This work clearly shows that IGSCC can occur in water at 280°C with 200 ppb of dissolved oxygen, even at low conductivity (low impurity levels), but the rate of cracking decreases with decreasing impurity content. Although BWR water chemistry guidelines for reactor water cannot prevent IGSCC, maintaining the lowest practically achievable impurity levels will minimize its rate of
252.7 progression (References 5 and 9).

Stress corrosion cracking of ductile materials in aqueous environments often is restricted to specific ranges of corrosion potential*, so a number of studies of impurity effects on IGSCC have been made as a function of either corrosion potential or dissolved oxygen content (the dissolved oxygen content is the major chemical variable in BWR type water that can be used to manipulate the corrosion potential in laboratory tests) (Reference 10).

As the corrosion potential is reduced below the range typical of normal BWR power operation 252.7
(+50 to -50 mV_{SHE}), a region of immunity to IGSCC appears at ~ -230 mV_{SHE}. It is apparent that a combination of corrosion potential (which can be achieved in a BWR by injecting usually <
1 ppm hydrogen into the feedwater) plus tight 252.7
conductivity control (0.2 μS/cm) should permit BWRs to operate in a regime where sensitized stainless steels are immune to IGSCC.

Since the ABWR has no sensitized stainless steel, IGSCC control by hydrogen injection is not required. However, irradiation assisted
stress corrosion cracking (IASCC) can occur in 252.7
highly irradiated annealed stainless steel and nickel-base alloys. Preliminary in-reactor and laboratory studies (Reference 11) have indicated that HWC will be useful in mitigating IASCC.

In-reactor and laboratory evidence also indicates that carbon and low alloy steels also tend to show improved resistance to environmentally assisted cracking with both increasing water purity and decreasing corrosion potential
(Reference 12). 252.7

5.2.3.2.2.1 Fuel Performance Considerations

Nuclear fuel is contained in Zircaloy tubes that constitute the first boundary or primary containment for the highly radioactive species generated by the fission process; therefore, the integrity of the tubes must be ensured. Zircaloy interacts with the coolant water and some coolant impurities. This results in oxidation by the water, increased hydrogen content in the Zircaloy (hydriding), and, often, buildup of a layer of crud on the outside of the tube. Excessive oxidation, hydriding, or crud deposition may lead to a breach of the cladding wall.

Metallic impurities can result in neutron losses and associated economic penalties which increase in proportion to the amount being introduced into the reactor and deposited on the fuel. With respect to iron oxide-type crud deposits, it can be concluded that operation

252.7 **Also called electrochemical corrosion potential or ECP, see Reference 9.*

to have an important influence on IGSCC initiation times for smooth stainless steel specimens in laboratory tests. In addition, pH can serve as a useful diagnostic parameter for interpreting severe water chemistry transients and pH measurements are recommended for this purpose.

(10) Electrochemical Corrosion Potential

The electrochemical corrosion potential (ECP) of a metal is the potential it attains when immersed in a water environment. The ECP is controlled by various oxidizing agents including copper and radiolysis products. At low reactor water conductivities, the ECP of stainless steel should be below -0.23 V_{SHE} to suppress IGSCC.

(11) Feedwater Hydrogen Addition Rate

A direct measurement of the feedwater hydrogen addition rate can be made using the hydrogen addition system flow measurement device and is used to establish the plant-specific hydrogen flow requirements required to satisfy the limit for the ECP of stainless steel (Paragraph 10). Subsequently, the addition rate measurements can be used to help diagnose the origin of unexpected ECP changes.

(12) Recirculation System Water Dissolved Hydrogen

A direct measurement of the dissolved hydrogen content in the reactor water serves as a cross check against the hydrogen gas flow meter in the injection system to confirm the actual presence and magnitude of the hydrogen addition rate.

(13) Main Steam Line Radiation Level

The main steam line radiation monitor reading indicates an excessive amount of hydrogen being injected. Likewise, a decrease in this parameter would be a quick indication of a decrease or stoppage of hydrogen injection.

(14) Constant Extension Rate Test

Constant extension rate tests (CERTs) are accelerated tests that can be completed in a few days, for the determination of the susceptibility to IGSCC. It is useful for verifying IGSCC suppression during initial implementation of hydrogen water chemistry (HWC) or following plant outages that could have had an impact on system chemistry (e.g., condenser repairs during refueling).

(15) Continuous Crack Growth Monitoring Test

This test employs a reversing DC potential drop technique to detect changes in crack length in IGSCC test specimens. The crack growth test can be used for a variety of purposes, including the following:

(a) Initial verification of IGSCC suppression following HWC implementation.

(b) Quantitative assessment of water chemistry transients.

(c) Long-term quantification of the success of the HWC program.

The major impurities in various parts of a BWR under certain operating conditions are listed in Table 5.2-5. The plant systems have been designed to achieve these limits at least 90% of the time. The plant operators are encouraged to achieve better water quality by using good operating practice.

Water quality specifications require that erosion-corrosion resistant low alloy steels are to be used in susceptible steam extraction and drain lines. Stainless steels are considered for baffles, shields, or other areas of severe duty. Provisions are made to add nitrogen gas to extraction steamlines, feedwater heater shells, heater drain tanks, and drain piping to minimize corrosion during layup. Alternatively, the system may be designed to drain while hot so that dry layup can be achieved. 281.10 item 6 & 9

Condenser tubes and tubesheet are required to be made of titanium alloys. 281.10 item 7

281.10 item 10 Water quality specifications for the ABWR require that the condenser is to be designed and erected to minimize tube leakage and to facilitate maintenance. Appropriate features are incorporated to detect leakage and segregate the source. The valves controlling the cooling water to the condenser sections are required to be operable from the control room so that a leaking section can be sealed off quickly.

5.2.3.2.2.4 IGSCC Considerations

Plant experience and laboratory tests indicate that IGSCC can be initiated in solution annealed stainless steel above certain stress levels after losing exposure to radiation.

281.2 Extensive tests have also shown that IGSCC has not occurred at fluence levels below ~$5x10^{20}$ n/cm^2 (E>1MeV) even at high stress levels. Experiments indicate that as fluence increases above this threshold of $5x10^{20}$ n/cm^2, there is a decreasing threshold of sustained stress below which IGSCC has not occurred. (Examination of top guides in two operating plants which have creviced designs has not revealed any IGSCC.)

Reactor core structural components are designed to be below these thresholds of exposure and/or stress to avoid IASCC. In addition, crevices have been eliminated from the top guide design in order to prevent the synergistic interaction with IASCC.

In areas where the $5x10^{20}$ n/cm^2 threshold of irradiation is not practically avoided, the stress level is maintained below the stress threshold. High purity grades of materials are used in control rods to extend their life. Also HWC introduced in the plant design to control IGSCC may also be beneficial in avoiding IASCC.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

The construction materials exposed to the reactor coolant consist of the following:

(1) solution-annealed austenitic stainless steels (both wrought and cast), Types 304, 304L, and 316L;

(2) nickel-based alloy and alloy steel;

(3) carbon steel and low alloy steel;

(4) some 400-series martensitic stainless steel (all tempered at a minimum of 1100°F);

(5) Colmonoy and Stellite hardfacing material (or equivalent)

All of these construction materials are resistant to stress corrosion in the BWR coolant. General corrosion on all materials except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

The requirements of GDC 4 relative to compatability of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code by compliance with the recommendations of Regulatory Guide 1.44. 252.9

Contaminants in the reactor coolant are controlled to very low limits. These controls are implemented by limiting containment levels of elements (such as halogens, S, Pb) to as low as possible in miscelleaneous materials used during fabrication and installation. These materials (such as tapes, penentrants) are completely removed and cleanliness is assumed. 252.9 No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Expected radiolytic products in the BWR coolant have no adverse effects on the construction materials.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation

All non-metallic insulation applied to austenitic stainless steel meets Regulatory Guide 1.36.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Compliance with Code requirements shall be in accordance with the following:

(1) The ferritic materials used for piping, pumps, and valves of the reactor coolant pressure boundary are usually 2-1/2 inches

or less in thickness. Impact testing is performed in accordance with NB-2332 for 252.10 thicknesses of 2-1/2 inches or less. The materials comply with Appendix G, Section G-3100 of ASME Code Section III.

(2) Materials for bolting with nominal diameters exceeding one inch are required to meet both the 25 mils lateral expansion specified in NB-2333 and the 45 ft-lb Charpy V value specified in 10CFR50, Appendix G. The 45 ft-lb requirement stems from the ASME Code where it applies to bolts over 4 inches in diame- 252.10 ter, starting Summer 1973 Addenda. Prior to this, the Code referred to only 2 sizes of bolts ($\leq$ 1 inch and > 1 inch). GE continued the two-size categories, and added the 45 ft-lb as a more conservative requirement.

(3) The reactor vessel complies with the requirements of NB-2331. The reference temperature (RT_{NDT}) is established for all required pressure-retaining materials used in the construction of Class 1 vessels. This includes plates, forgings, weld material, and heat-affected zone. The RT_{NDT} differs from the nil-ductility temperature (NDT) in that in addition to passing the drop test, three Charpy V-Notch specimens (traverse) must exhibit 50 ft-lb absorbed energy and 35 mil lateral expansion at 60°F above the RT_{NDT}. The core beltline material must meet 75 ft-lb absorbed upper shelf energy.

252.10 (4) Calibration of instrument and equipment shall meet the requirements of the ASME Code, Section III, paragraph NB-2360.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Regulatory Guide 1.50: Control of Preheat Temperature Employed for Welding of Low-Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

The use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant-pressure boundary are fabricated from carbon steel materials.

Preheat temperature employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Subsection NA. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

Acceptance Criterion II.3.b(1)(a) of SRP Section 5.2.3 for control of preheat temperature requires that minimum and maximum interpass temperature be specified. While the ABWR control of low-hydrogen electrodes to prevent hydrogen cracking (provided in Subsection 5.2.3.3.4) does not explicitly meet this requirement, the ABWR control will assure that cracking of components made from low-alloy steels does not occur during fabrication. Further, the ABWR control minimizes the possibility of subsequent cracking resulting from hydrogen being retained in the weldment.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

5.2.3.3.2.2 Regulatory Guide 1.34: Control of Electroslag Weld Properties

No electroslag welding is performed on BWR components.

5.2.3.3.2.3 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Welder qualification for areas of limited accessibility is discussed in Subsection 5.2.3.4.2.3.

5.2.3.3.3 Regulatory Guide 1.66: Nondestructive Examination of Tubular Products

Regulatory Guide 1.66 describes a method of implementing requirements acceptable to NRC regarding nondestructive examination requirements of tubular products used in RCPB. This Regulatory Guide was withdrawn on September 28, 1977, by the NRC because the additional requirements imposed by the guide were satisfied by the ASME Code.

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. Additionally, the specification for the tubular products used for CRD housings specified ultrasonic examination to paragraph NB-2550 of ASME Code Section III.

These RCPB components meet 10CFR50 Appendix B requirements and the ASME Code requirements thus assuring adequate control of quality for the products.

5.2.3.3.4 Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes

All low-hydrogen covered welding electrodes are stored in controlled storage areas and only authorized persons are permitted to release and distribute electrodes. Electrodes are received in hermetically sealed canisters. After removal from the sealed containers, electrodes which are not immediately used are placed in storage ovens which are maintained at about 250°F (generally 200°F minimum).

Electrodes are distributed from sealed containers or ovens as required. At the end of each work shift, unused electrodes are returned to the storage ovens. Electrodes which are damaged, wet, or contaminated are discarded. If any electrodes are inadvertently left out of the ovens for more than one shift, they are discarded or reconditioned in accordance with manufacturer instructions.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress/Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization

252.10 When austenitic stainless steels are heated in the temperature range 800°-1800°F, they are considered to become "sensitized" or susceptible to intergranular corrosion. The ABWR design complies with Regulatory Guide 1.44 and with the guide lines of NUREG-0313, to avoid significant sensitization.

Process control are exercised during all stages of component manufacturing and construction to minimize contaminants. Cleanliness controls are applied prior to any elevated temperature treatment. For applications where stainless steel surfaces are exposed to water at temperatures above 200°F low carbon (<0.03%) grade materials are used. For critical applications, nuclear grade materials (carbon content ≤ 0.02%) are used. All materials are supplied in the solution heat treated condition. Special sensitization tests are applied to assure that the material is in the annealed condition. 252.10

During fabrication, any heating operation (except welding) between 800°-1800°F are avoided, unless followed by solution heat treatment. During welding, heat input is controlled. The interpass temperature is also controlled. Where practical, shop welds are solution heat treated. In general, weld filler material used for austenitic stainless steel base metals is Type 308L/3161/3091 with an average of 8% (of Fn) ferrite content.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of causing stress/corrosion cracking of austenitic stainless steel components was avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture, construction, and installation.

Special care was exercised to insure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing was controlled and monitored. Suitable protective packaging was provided for components to maintain cleanliness during shipping and storage.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guides 1.37 and 1.44.

For commitment and revision number, see Section 1.8.

5.2.3.4.1.3 Cold-Worked Austenitic Stainless Steels

Cold work controls are applied for components made of austenitic stainless steel. These materials are used in the cast condition. During fabrication cold work is controlled by applying limits in hardness, bend radii and surface finish on ground surfaces.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

Regulatory Guide 1.31 describes the acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems.

Written welding procedures which are approved by GE are required for all primary pressure boundary welds. These procedures comply with the requirements of Sections III and IX of the ASME Boiler Pressure Vessel Code and applicable NRC Regulatory Guides.

All austenitic stainless steel weld filler materials were required by specification to have a minimum delta ferrite content of 5 FN (ferrite number) determined on undiluted weld pads by magnetic measuring instruments calibrated in accordance with AWS specification A4.2-74.

Delta ferrite measurements are not made on qualification welds. Both the ASME Boiler and Pressure Vessel Code and Regulatory Guide 1.31 specify that ferrite measurements be performed on undiluted weld filler material pads when magnetic instruments are used. There are no requirements for ferrite measurement on qualification welds.

5.2.3.4.2.2 Regulatory Guide 1.34: Electroslag Welds

Electroslag welding was not employed for reactor coolant pressure boundary components.

5.2.3.4.2.3 Regulatory Guide 1.71: Welder Qualification or Areas of Limited Accessibility

Regulatory Guide 1.71 requires that weld fabrication and repair for wrought low-alloy and high-alloy steels or other materials such as static and centrifugal castings and bimetallic joints should comply with fabrication requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code. It also requires additional performance qualifications for welding in areas of limited access.

All ASME Section III welds are fabricated in accordance with the requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code. There are few restrictive welds involved in the fabrication of BWR components. Welder qualification for welds with the most restrictive access is accomplished by mockup welding. Mock-up is examined sectioning and radiography (or UT). 252.11

The Acceptance Criterion II.3.b.(3) of SRP Section 5.2.3 is based on Regulatory Guide 1.71. The ABWR design meets the intent of this regulatory guide by utilizing the alternate approach as follows:

When access to a non-volumetrically examined ASME Section III production weld (1) is less than 12 inches in any direction and (2) allows welding from one access direction only, such weld and repairs to welds in wrought and cast low alloy steels, austenitic stainless steels and high nickel alloys and in any combination of these materials shall comply with the fabrication requirements specified in ASME Boiler and Pressure Vessel Code Section III and with the requirements of Section IX invoked by Section III, supplemented by the following requirements:

(1) The welder performance qualification test assembly required by ASME Section IX shall be welded under simulated access conditions. An acceptable test assembly will provide both a Section IX welder performance qualification required by this Regulatory guide. 252.11

    If the test assembly weld is to be judged by bend tests, a test specimen shall be removed from the location least favorable for the welder. If this test specimen cannot be removed from a location prescribed by Section IX, an additional bend test specimen will be required. If the test assembly weld is to be judged by radiography or UT, the length of weld to be examined shall include the location least favorable for the welder.

Records of the results obtained in welder accessibility qualification shall be as certified by the manufacturer or installer, shall be maintained and shall be made accessible to authorized personnel.

Socket weld with a 2-in. nominal pipe size and under are excluded from the above requirements.

252.11 (2) (a) For accessibility, when more restricted access conditions than qualified will obscure the welder's line of sight to the extent that production welding will require the use of visual aids such as mirrors. The requalification test assembly shall be welded under the more restricted access conditions using the visual aid required for production welding.

(b) GE complies with ASME Section IX.

(3) Surveillance of accessibility qualification requirements will be performed along with normal surveillance of ASME Section IX performance qualification requirements.

5.2.3.4.3 Regulatory Guide 1.66: Nondestructive Examination of Tubular Products

For discussion of compliance with Regulatory Guide 1.66, see Subsection 5.2.3.3.3.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

This subsection discusses the inservice inspection and testing program for the NRC Quality Group A components; i.e., ASME Boiler and Pressure Vessel Code Section III, Class 1, components. It will show how the program meets requirements of Section XI of the ASME Code.

5.2.4.1 System Boundary Subject to Inspection

The system boundary subject to inspection
250.1 includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant systems, up to and including:

(1) The outermost containment isolation valve in system piping that penentrates the primary reactor containment.

(2) The second of two valves normally closed during normal reactor operation in system piping that does not penentrate primary reactor containment.

(3) The reactor coolant system and relief valves.

5.2.4.2 Provisions for Access to the Reactor Coolant Pressure Boundary

5.2.4.2.1 Design and Arrangement of Reactor Coolant Pressure Boundary Components

Accessibility in accordance with ASME Code Section XI, IWA 1500 is provided as described in 250.2
the following paragraphs.

5.2.4.2.2 Reactor Pressure Vessel

Access for examination of the RPV has been provided through provisions incorporated into the design of the vessel, shield wall, and vessel insulation as follows:

(1) The shield wall and vessel insulation behind the shield wall are spaced away from the RPV outside surface. Access ports are located at each reactor pressure vessel nozzle. The annular space between the reactor vessel outside surface and insulation inside surface permits insertion of remotely-operated ultrasonic devices for examination of vessel longitudinal and circumferential welds. Access for insertion of the automated devices is provided through removable insulalation panels at the top of the shield wall.

(2) Access to the reactor pressure circumferential, longitudinal, and nozzle-to-vessel welds above the shield wall is provided through use of removable insulation panels. Either manual or automated examination methods may be employed.

gallon per minute, thus meeting Position C.2 requirements.

By monitoring (1) floor drain sump fillup and pumpout rate, (2) airborne particulates, and (3) air coolers condensate flow rate, Position C.3 is satisfied.

Monitoring of the reactor building cooling water heat exchanger coolant return lines for radiation due to leaks within the RHR, RIP and RWCS heat exchangers (and the fuel pool cooling system heat exchangers) satisfies Position C.4. For system detail, see Subsection 7.6.1.2.

The floor drain sump monitoring, air particulates monitoring, and air cooler condensate monitoring are designed to detect leakage rates of one gpm within one hour, thus meeting Position C.5 requirements.

The fission products monitoring subsystem is qualified for SSE. The containment floor drain sump monitor, air cooler, and condensate flow meter are qualified for OBE, thus meeting Position C.6 requirements.

Leak detection indicators and alarms are provided in the main control room. This satisfies Position C.7 requirements. Procedures and graphs will be provided by the applicant to plant operators for converting the various indicators to a common leakage equivalent, when necessary, thus satisfying the remainder of Position C.7. The leakage detection system is equipped with provisions to permit testing for operability and calibration during the plant operation using the following methods:

(1) simulation of signals into trip units;

(2) comparing channel A to channel B of the same leak detection method (i.e., area temperature monitoring);

(3) operability checked by comparing one method versus another (i.e., sump fillup rate versus pumpout rate and particulate monitoring on air cooler condensate flow versus sump fillup rate); and

(4) continuous monitoring of floor drain sump level and a source of water for calibration and testing is provided.

These satisfy Position C.8 requirements.

Limiting unidentified leakage to the range of 1 to 5 gpm and identified to 25 gpm satisfies Position C.9.

5.2.6 Interfaces

The remainder of plant will meet the water chemistry requirements given in Table 5.2-5.

5.2.7 References

1. *General Electric Standard Application for Reactor Fuel,* (NEDE-24011-P-A, latest approved version).

2. *BWR Normal Water Chemistry Guidelines: 1986 Revision,* EPRI NP-4946-SR, Final Draft, October 17, 1986 (To be published).

3. D.A. Hale, *The Effect of BWR Startup Environments on Crack Growth in Structural Alloys,* Trans. of ASME, vol 108, January 1986.

4. F.P. Ford and M. J. Povich, *The Effect of Oxygen/Temperature Combinations on the Stress Corrosion Susceptibility of Sensitized T-304 Stainless Steel in High Purity Water,* Paper 94 presented at Corrosion 79, Atlanta, GA, March 1979. 281.4

5. *BWR Normal Water Chemistry Guidelines: 1986 Revision,* EPRI NP-4946-SR, October 1987. 281.7 281.10 item 4 & 11

6. B.M. Gordon, *The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature,* Material Performance, NACE, Vol. 19, No. 4, April 1980.

7. W.J. Shack, et al, *Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1983 - September 1984,* NUREG/CR-4287, ANL-85-33, June 1985.

8. D.A. Hale, et al, *BWR Coolant Impurities Program,* EPRI, Palo Alto, CA, Final Report on RP2293-2, to be published.

9. K.S. Brown and G.M. Gordon, *Effects of BWR Coolant Chemistry on the Propensity of IGSCC Initiation and Growth in Creviced Reactor Internals Components*, paper presented at the Third International Symposium of Environmental Degradation of Materials in Nuclear Power Systems, ANS-NACE-TMS/AIME, Traverse City, Michigan, September 1987.

10. B.M. Gordon et al, *EAC Resistance of BWR Materials in HWC*, Proceeding of Second International Symposium Environmental Degration of Materials in Nuclear Power Systems, ANS, LaGrange Park, ILL 1986.

11. B.M. Gordon, *Corrosion and Corrosion Control in BWRs*, NEDE-30637, December 1984.

12. B.M. Gordon et al, *Halogen Water Chemistry for BWRs - Materials Behavior*, EPRI NP-5080, Palo Alto, CA, March 1987.

Table 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

APPLICABLE CODE CASES

Number	Title	Applicable Equipment	Remarks
N-71-15	(1)	Component Support	Accepted per RG 1.85
N-122	(2)	Piping	Accepted per RG 1.84
N-247	(3)	Component Support	Accepted per RG 1.84
N-249-9	(4)	Component Support	Conditionally Accepted per RG 1.85
N-309-1	(5)	Component Support	Accepted per RG 1.84
N-313	(6)	Piping	Accepted per RG 1.84
N-316	(7)	Piping	Accepted per RG 1.84
N-318-3	(8)	Piping	Conditionally Accepted per RG 1.84
N-319	(9)	Piping	Accepted per RG 1.84
N-391	(10)	Piping	Accepted per RG 1.84
N-392	(11)	Piping	Accepted per RG 1.84
N-393	(12)	Piping	Accepted per RG 1.84
N-411-1	(13)	Piping	Conditionally Accepted per RG 1.84
N-414	(14)	Component Support	NEW - Not yet listed
N-430	(15)	Component Support	NEW - Not yet listed
N-433	(16)	Component Support	NEW - Not yet listed
N-451	(17)	Piping	NEW - Not yet listed

210.1

Table 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

APPLICABLE CODE CASES (Continued)

(1) *Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III, Division 1.*

(2) *Stress Indices for Structure Attachments, Class 1, Section III, Division 1.*

(3) *Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Class 1, 2, 3 and MC.*

(4) *Additional Material for Subsection NF, Classes 1, 2, 3 and MC Compo-*

(5) *Identification of Materials for Component Supports, Section III, Division 1.*

(6) *Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1.*

(7) *Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division 1, Class 1, 2, 3.*

(8) *Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1.*

(9) *Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1.*

(10) *Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping. Section III, Division 1.*

(11) *Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1.*

(12) *Repair Welding Structural Steel Rolled Shapes and Plates for Component Supports, Section III, Division 1.*

(13) *Alternative Damping Values for Seismic Analysis of Classes 1, 2, 3 Piping Sections, Section III, Division 1.*

(14) *Tack Welds for Class 1, 2, 3 and MC Components and Piping Supports.*

(15) *Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports.*

(16) *Non-Threaded Fasteners for Section III, Division 1, Class 1,2, and 3 Component and Piping Supports.*

(17) *Alternative Rules for Analysis of Piping Under Seismic Loading Class 1.*

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel and appurtenances are shown in Table 5.2-4 together with the applicable specifications.

251.1 The RPV materials shall comply with the provisions of the ASME Code Section III, Appendix I and meet the specification requirements of 10CFR50, Appendix G.

5.3.1.2 Special Procedures Used for Manufacturing and Fabrication

The reactor pressure vessel is primarily constructed from low alloy, high-strength steel plate and forgings. Plates are ordered to ASME SA-533, TYPE B, Class 1, and forgings to ASME SA-508, Class 3. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low-alloy steels. Materials used in the core beltline region also specify limits of 0.05% maximum copper and 0.015% maximum phosphorous content in the base materials and a 0.08% maximum copper and 0.020% maximum phosphorous content in weld materials.

Studs, nuts, and washers for the main closure flange are ordered to ASME SA-540, Grade B23 or Grade B24. Welding electrodes for low alloy steel are low-hydrogen type ordered to ASME SFA-5.5.

All plate, forgings, and bolting are 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Section III, Division 1.

251.2 Fracture toughness properties are also measured and controlled in accordance with Division 1.

All fabrication of the reactor pressure vessel is performed in accordance with GE-approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates or forgings, and the flanges and nozzles from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples are required for each procedure for major vessel full-penetration welds. Tensile and impact tests are performed to determine the properties of the base metal, heat affected zone, and weld metal.

Submerged arc and manual stick electrode welding processes are employed. Electroslag welding is not applied. Preheat and interpass temperatures employed for welding of low-alloy steel meet or exceed the values given in ASME, Section III, Appendix D. Post weld heat treatment at 1100°F minimum is applied to all low-alloy steel welds.

Radiographic examination is performed on all pressure containing welds in accordance with requirements of ASME, Section III, Subsection NB 5320. In addition, all welds are given a supplemental ultrasonic examination.

The materials, fabrication procedures, and testing methods used in the construction of BWR reactor pressure vessels meet or exceed requirements of ASME Section III, Class 1 vessels.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the reactor pressure vessel are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME, Section III. In addition, the pressure-retaining welds are ultrasonically examined using manual techniques. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME, Section XI, Appendix I. Acceptance standards are equivalent or more restrictive than required by ASME, Section XI.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

5.3.1.4.1 Regulatory Guide 1.31: Control of Stainless Steel Welding

Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2.1.

5.3.1.4.2 Regulatory Guide 1.34: Control of Electroslag Weld Properties

Electroslag welding is not employed for the reactor pressure vessel fabrication.

5.3.1.4.3 Regulatory guide 1.43: Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Reactor pressure vessel specifications require that all low-alloy steel be produced to fine grain practice. The requirements of this regulatory guide are not applicable to BWR vessels.

5.3.1.4.4 Regulatory Guide 1.44: Control of the Use of Sensitized Stainless Steel

251.3 Sensitization of stainless steel is controlled by the use of service proven materials and by use of appropriate design and processing steps including solution heat treatment, corrosion resistant cladding, control of welding heat input, control of heat treatment during fabrication and control of stresses.

5.3.1.4.5 Regulatory Guide 1.50: Control of Preheat Temperature For Welding Low-Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

The use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant-pressure boundary are fabricated from carbon steel materials.

251.3 Preheat temperature employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Appendix D. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until postweld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

Acceptance Criterion II.3.b(1)(a) of SRP Section 5.2.3 for control of preheat temperature requires that minimum and maximum interpass temperature be specified. While the ABWR control of low-hydrogen electrodes to prevent hydrogen cracking (provided in Subsection 5.2.3.3.4) does not explicitly meet this requirements the ABWR control will assure that cracking of components made from low-alloy steels does not occur during fabrication. Further, the ABWR control minimizes the possibility of subsequent cracking resulting from hydrogen being retained in the weldment.

All welds are nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination is performed.

5.3.1.4.6 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in Subsection 5.2.3.4.2.3.

5.3.1.4.7 Regulatory Guide 1.99: Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper shelf energy are made in accordance with the requirements of Regulatory Guide 1.99.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10CFR50, Appendix G

Appendix G of 10CFR50 is interpreted for Class 1 primary coolant pressure boundary component of the ABWR reactor design and complied with as discussed in Subsections 5.3.1.5.2 and 5.3.2. The specific temperature limits operation of the reactor when the core is critical are based on 10CFR50, Appendix G, Paragraph IV, A.3.

5.3.1.5.2 Methods of Compliance

The following items are the interpretations and methods used to comply with 10CFR50, Appendix G.

(1) Material Test Coupons and Test Specimens (GIII-A)

> Test coupons are removed from the location in each product form as specified

in subarticle NB-2220 of the ASME Code, Section III. The heat treatment of the test coupons is performed in accordance with subarticle NB-2210.

It is understood that separately produced test coupons per Subparagraph NB-2223.3 may be used for forgings.

(2) Location and Circulation of Test Specimens (G III-A)

The test specimens are located and oriented per ASME, Section III, Paragraph NB-2322. Transverse Charpy-V impact specimens are used for the testing of plate and forged material other than bolting and bars. Longitudinal specimens are used for bolting and bars.

Both longitudinal and transverse specimens are used to determine the required minimum upper shelf energy level of the core belt line materials.

In regard to 10CFR50, Appendix H, the surveillance test material is selected on the basis of the requirements of ASTM E185-82 and Regulatory Guide 1.99 to provide a conservative adjusted reference temperature for the beltline materials. The weld test plate for the surveillance program specimens has the principal working direction parallel to the weld seam to assure that heat-affected zone specimens are transverse to the principal working direction.

(3) Records and Procedures for Impact Testing (G III-C)

completed vessel. Each in-reactor surveillance capsule contains 36 Charpy V-notch and 6 tensile specimens. The capsule loading consists of 12 Charpy V Specimens each of base metal, weld metal, heat-affected zone material, and 3 tensile specimens each from base metal and weld metal. A set of out-of-reactor baseline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors will be located within the capsules as required by ASTM E 185-82.

Three capsule are provided in accordance with requirements of 10CFR50, Appendix H, since the predicted end of the adjusted reference temperature of the reactor vessel steel is less than 100 ^{o}F.

The following proposed withdrawal schedule is in accordance with ASTM E 185-82.

First capsule: After 6 effective full-power years
Second capsule: After 15 effective full-power years
Third capsule: Schedule determined based on
251.4 results of first two capsules per ASTM E 185-82, Paragraph 7.6.2.

Fracture toughness testing of irradiated capsule specimens will be in accordance with requirements of ASTM E 185-82 as called out for by 10CFR50, Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Transition temperature changes and changes in upper-shelf energy shall be calculated in accordance with the rules of Regulatory Guide 1.99. Reference temperatures shall be established in accordance with 10CFR50, Appendix G, and NB-2330 of the ASME Code.

Since weld material chemistry and fracture toughness data are not available at this time, the limits in the purchase specification were used to estimate worst-case irradiation effects. These estimates show that the adjusted reference temperature at end-of-life is less than 100 ^{o}F, and the end-of-life upper-shelf energy
exceeds 50 ft-lb. (See response to Question 251.5
251.5 for the calculation and analysis associated with this estimate).

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment (Appendix H.II B (2))

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding. Since reactor vessel specifications require that all low-alloy steel pressure vessel boundary materials be produced to fine-grain practice, underclad cracking is of no concern. The capsule holder brackets allow the removal and reinsertion of capsule holders. Although not code parts, these brackets are designed, fabricated, and analyzed to the requirements of ASME Code Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

In areas where brackets (such as the surveillance specimen holder brackets) are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld-deposited cladding or weld-buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight-beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of width equal to at least half the thickness of the part joined. The required stainless steel weld-deposited cladding is similarly examined. The full penetration welds are liquid-penetrant examined. Cladding thickness is required to be at least 1/8 inch. These requirements have been successfully applied to a variety of bracket designs which are attached to weld-deposited stainless steel cladding or weld buildups in many operating BWR reactor pressure vessels.

Inservice inspection examinations of core beltline pressure-retaining welds are performed from the outside surface of the reactor pressure

5.3.2 Pressure/Temperature Limits

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figure 5.3-1 are based on the requirements of 10CFR50, Appendix G.

All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles were evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} +60°F. The maximum throughwall temperature gradient from continuous heating or cooling at 100°F per hour was considered. The safety factors applied were as specified in ASME Code, Appendix G, and Reference 2.

The material for the vessel will be provided with the following requirements of RT_{NDT} as
251.7 determined in accordance with Branch Technical
Position MTEB 5-2: shell and flanges -20°F; nozzles - 20°F and welds - 20°F.

5.3.2.1.1 Temperature Limits for Boltup

Minimum closure flange and fastener temperatures of RT_{NDT} +60°F are required for tensioning at preload condition and during
271.7 detensioning. Thus, the limit is 60°F + (-20°F) = 50°F.

5.3.2.1.2 Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Pressure (measured in the top head) versus temperature (minimum vessel shell and head metal temperature) limits to be observed for the test and operating conditions are specified in Figure 5.3-1. Temperature limits for preservice and inservice tests are shown in Curve A of Figure 5.3-1.

5.3.2.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

Heatup and Cooldown.

Curve B in Figure 5.3-1 specifies limits for non-nuclear heatup and cooldown following a nuclear shutdown.

Reactor Operation

Curve C in Figure 5.3-1 specifies limits applicable for operation whenever the core is critical except for low-level physics tests.

5.3.2.1.4 Reactor Vessel Annealing

Inplace annealing of the reactor vessel, because of radiation embrittlement, is not anticipated to be necessary.

5.3.2.1.5 Predicted Shift in RT_{NDT} and Drop in Upper-Shelf Energy (Appendix G-IV B)

For design purposes the adjusted reference nil ductility temperature and drop in the upper-shelf energy for BWR vessels is predicted using the procedures in Regulatory Guide 1.99.

The calculations (see response to Question 251.5) are based on the specified limits on Phosphorous (0.020%), Vanadium (0.05%), Copper (0.08%) and Nickel (1.2%) in the weld material. In plate material, the limits are Copper (0.05%) and Nickel (0.73%). Forgings will have the same
chemistry as plate but the nickel limit is 1%. 251.7

A surveillance program in accordance with ASTM E 185-82 will be used. The surveillance program will include samples of base metal, weld metal and heat affected zone material. Subsection 5.3.1.6 provides added detail on the surveillance program.

5.3.2.2 Operating Procedures

A comparison of the pressure versus temperature limit in Subsection 5.3.2.1 with intended normal operation procedures of the most severe upset transient shows that those limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established so that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas yields a minimum fluid temperature of 528 °F and a maximum peak pressure of 1215 psig. Scram automatically occurs as a result of this event prior to a possible reduction in fluid temperature to 250°F at a pressure of 930 psig. Per Figure 5.3-1, both the 1215 psig vessel pressure at 528 °F (Curve C) and the 930 psig at 250 °F (Curve B) are within the calculated margin against nonductile failure.

5.3.3 Reactor Vessel Integrity

The reactor vessel material, equipment, and services associated with the reactor vessels and appurtenances would conform to the requirements of the subject purchase documents. Measures to ensure conformance included provisions for source evaluation and selection, objective evidence of quality furnished, inspection at the vendor source and examination of the completed reactor vessels.

GE provides inspection surveillance of the

Design of the reactor vessel and its support system meets Seismic Category I equipment requirements. The materials used in the reactor pressure vessel are listed in Table 5.2-4.

The cylindrical shell and top and bottom heads of the reactor vessel are fabricated of low-alloy steel, the interior of which is clad with stainless steel weld overlay except for the top head and nozzle weld zones.

Inplace annealing of the reactor vessel is not necessary because shifts in transition temperature caused by irradiation during the 60 year life can be accommodated by raising the minimum pressurization temperature, and the predicted value of adjusted reference temperature does not exceed 200 $^{\circ}$F. Radiation embrittlement is not a problem outside of the vessel beltline region because the irradiation in those areas is less than 1 X 10^{18} nvt with neutron energies in excess of 1 MeV. The use of existing methods of predicting embrittlement and operating limits which are based on a 40 year life are considered to be applicable to a 60 year life because the 251.9 age degrading mechanism is irradiation and fatigue duty which are calculated for the 60 year life. Time/temperature effects will either not have any effect or will produce a small beneficial co-annealing.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The vessel top head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal-rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 100 $^{\circ}$F in any one hour period. To detect seal failure, a vent tap is located between the two seal-rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal.

5.3.3.1.1.2 Shroud Support

The shroud support is a circular plate welded to the vessel wall and to a cylinder supported by vertical stilt legs from the bottom head. This support is designed to carry the weight of peripheral fuel elements, neutron sources, core plate, top guide and the steam separators and to laterally support the fuel assemblies and the pump diffusers. Design of the shroud support also accounts for pressure differentials across the shroud support plate, for the restraining effect of components attached to the support, and for earthquake loadings. The shroud support design is specified to meet appropriate ASME Code stress limits.

5.3.3.1.1.3 Protection of Closure Studs

The BWRs do not use borated water for reactivity control during normal operation. This subsection is therefore not applicable.

5.3.3.1.2 Safety Design Basis

The design of the reactor vessel and appurtenances meets the following safety design bases.

(1) The reactor vessel and appurtenances will withstand adverse combinations of loading and forces resulting from operation under abnormal and accident conditions.

(2) To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

(a) impact properties at temperatures related to vessel operation have been specified for materials used in the reactor vessel;

(b) expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design and operational limitations assure that NDT temperature shifts are accounted for in reactor operation; and

(c) operational margins to be observed with regard to the transition temperature are specified for each mode of operation.

5.3.3.1.3 Power Generation Design Bases

The design of the reactor vessel and appurtenances meets the following power generation design bases:

which provide for ease of installation and removal for vessel inservice inspection and maintenance operation. Each insulation unit has lifting fittings attached to facilitate removal. Insulation units attached to the shield wall are not required to be readily removable except around penetrations.

At operating conditions, the insulation on the shield wall and around the refueling bellows has an average maximum heat transfer rate of 65 Btu per hour per square foot of outside insulation surface. The maximum heat transfer rate for insulation on the top head is 60 Btu per hour per square foot. Operating conditions are 550°F for the outside temperature of the reactor vessel and 135°F for the drywell air. The maximum air temperature is 150°F, except for the head area above the bulkhead and refueling seal which has a maximum allowable temperature of 200°F.

5.3.3.1.4.5 Reactor Vessel Nozzles

All piping connected to the reactor vessel nozzles has been designed not to exceed the allowable loads on any nozzle. The vessel top head nozzle is provided with flanges with small groove facings. The drain nozzle is of the full penetration weld design. The feedwater inlet nozzles, core spray inlet nozzles, and ECCS flooding nozzles have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends or extensions made of stainless steel. These safe ends or extensions were welded to the nozzles after the pressure vessel was heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe.

5.3.3.1.4.6 Materials and Inspections

The reactor vessel was designed and fabricated in accordance with the applicable ASME Boiler and Pressure Vessel Code as defined in Subsection 5.2.1. Table 5.2-4 defines the materials and specifications. Subsection 5.3.1.6 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic

The reactor vessel schematic is shown in Figure 5.3-2.

5.3.3.2 Materials of Construction

All material used in the construction of the reactor pressure vessel conform to the requirements of ASME Code, Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low-alloy steel plate and forgings purchased in accordance with ASME Specifications SA-533 Type B, Class 1 and SA-508 Class 3. Special requirements for the low-alloy steel plate employed on the interior surfaces of the vessel consists of austenitic stainless steel weld overlay.

These materials of construction were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long term successful operating experience in reactor service.

The expected peak neutron fluence at the 1/4 t location used for evalution is less than 4 x 10^{17} nvt for 60 years, the calculated shift in RT_{NDT} is 28°F for weld metal and 8°F for base metal and the drop in upper shelf energy is 10 ft-lbs for welds and 8 ft-lbs for base metal. 251.10

5.3.3.3 Fabrication Methods

The reactor pressure vessel is a vertical cylindrical pressure vessel of welded construction fabricated in accordance with ASME Code, Section III, Class 1, requirements. All fabrication of the reactor pressure vessel was performed in accordance with GE-approved drawings, fabrication procedures, and test procedures. The shell and vessel head were made from formed low-alloy steel plates or forgings and the flanges and nozzles from low-alloy steel forgings. Welding performed to join these vessel components was in accordance with procedures qualified to ASME, Section III and IX requirements. Weld test samples were required for each procedure for major-vessel full-penetration welds.

Submerged arc and manual stick electrode welding processes were employed. Electroslag welding was not applied. Preheat and interpass temperatures employed for welding of low-alloy steel met or exceeded the requirements of ASME Section III, Appendix D. Post-weld heat treatment of 1100°F minimum was applied to all low-alloy steel welds.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for an extensive number of years and their service history is rated excellent.

5.3.3.4 Inspection Requirements

All plates, forgings, and bolting were 100% ultrasonically tested and surface examined by magnetic-particle methods or liquid-penetrant methods in accordance with ASME Code, Section III. Welds on the reactor pressure vessel were examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Code, Section III. In addition, the pressure retaining welds were ultrasonically examined using acceptance standards which are required by ASME Code, Section XI.

5.3.3.5 Shipment and Installation

The completed reactor vessel is given a thorough cleaning and examination prior to shipment. The vessel is tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment are in accordance with detailed written procedures.

On arrival at the reactor site the reactor vessel is examined for evidence of any contamination as a result of damage to shipping covers. Measures are taken during installation to assure that vessel integrity is maintained; for example, access controls are applied to personnel entering the vessel, weather protection is provided, and periodic cleanings are performed.

5.3.3.6 Operating Conditions

Procedural controls on plant operation are implemented to hold thermal stresses within acceptable ranges and to meet the pressure/temperature limits of Subsection 5.3.2. The restrictions on coolant temperature are as follows:

251.11

(1) the average rate of change of reac[illegible] coolant temperature during normal h[illegible] and cooldown shall not exceed 1[illegible] during any one hour period;

(2) if the coolant temperature difference between the dome (inferred from P (sat)) and the bottom head drain exceeds 100°F, neither reactor power level nor recirculation pump flow shall be increased.

The limit regarding the normal rate of heatup and cooldown (Item 1) assures that the vessel closure, closure studs, vessel support skirt, control rod drive housing, and stub tube stresses and usage remain within acceptable limits. Vessel temperature limit on recirculating pump operation and power level increase restriction (Item 2) augments the Item 1 limit in further detail by assuring that the vessel bottom head region will not be warmed at an excessive rate caused by rapid sweep-out of cold coolant in the vessel lower head region by recirculating pump operation or natural circulation (cold coolant can accumulate as a result of control drive inleakage and/or low recirculation flow rate during startup or hot standby).

These operational limits when maintained ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these operational limits are exceeded, the reactor vessel has been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained since the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity will be maintained during the most severe postulated transients since all such transients are evaluated in the design of the reactor vessel.

5.3.3.7 Inservice Surveillance

Inservice inspection of the reactor pressure vessel will be in accordance with the requirements of the ASME Boiler and Pr[illegible] Vessel [illegible] Section XI. The vessel will [illegible] examined [illegible] to startup to satisfy the [illegible] operational [illegible] ments of IWB-2000 of ASME Code, [illegible] subsequent inservice inspection will be scheduled and performed in accordance with the requirements of 10CFR50.55a, subparagraph (g) as described in Subsection 5.2.4.

The materials surveillance program will

Table 15.7-10

ISOTOPIC RELEASE TO ENVIRONMENT IN CURIES

ISOTOPE	1-MIN	10-MIN	1-HOUR	2-HOUR
I-131	8.32E-02[a]	7.45E-01	2.61E+00	3.19E+00
I-132	1.07E-01	9.39E-01	3.02E+00	3.52E+00
I-133	8.60E-02	7.69E-01	2.67E+00	3.25E+00
I-134	4.67E-09	3.95E-08	1.12E-07	1.23E-07
I-135	1.41E-02	1.26E-01	4.27E-01	5.13E-01
TOTAL I	2.90E-01	2.58E+00	8.74E+00	1.05E+01
KR-83M	4.43E-01	3.86E+00	1.22E+01	1.40E+01
KR-85M	5.65E+00	5.01E+01	1.68E+02	2.00E+02
KR-85	3.08E+01	2.76E+02	9.68E+02	1.18E+03
KR-87	8.75E-04	7.54E-03	2.27E-02	2.57E-02
KR-88	1.63E+00	1.44E+01	4.71E+01	5.53E+01
XE131M	5.37E+00	4.81E+01	1.69E+02	2.06E+02
XE133M	7.11E+01	6.37E+02	2.23E+03	2.72E+03
XE-133	1.81E+03	1.62E+04	5.70E+04	6.96E+04
XE135M	2.12E+01	1.58E+02	3.11E+02	3.16E+02
XE-135	4.15E+02	3.70E+03	1.27E+04	1.53E+04
TOTAL NG	2.37E+03	2.11E+04	7.36E+04	8.96E+04

[a] 8.32E-02 = 8.32×10^{-2}

470.7

Table 15.7-12

FUEL CASK DROP ACCIDENT PARAMETERS

I	Data and assumptions used to estimate source terms	
	A. Power level of reactor while fuel was in core	4005 MWt
	B. Radial Peaking Factor while fuel was in core	1.5
	C. Fuel Bundles in Cask	18
	D. Fuel Damaged	1116 rods
	E. Minimum time of fuel in storage prior to accident	120 days
	F. Peak linear power density	13.4 kW/ft
	G. Average burn-up of fuel	32,000 MWD/t
	H. Maximum Fuel centerline temperature	3315 F
	I. Fraction of activity released	10% of all isotopes except 30% Kr-85
	J. Time Period for Reactor Building Release	2 hour
	K. Iodine Filter Efficiency	99%
II	Dispersion and Dose Data	
	A. Meteorology	Table 15.7-13
	B. Boundary and LPZ distances	Table 15.7-13
	C. Method of Dose Calculation	Reference 1
	D. Dose conversion Assumptions	Reference 1 and RG 1.109
	E. Activity Inventory/releases	Table 15.7-13 Table 15.7-13
	F. Dose Evaluations	Table 15.7-13

470.6

470.7

Table 15.7-13

CASK DROP ACCIDENT RADIOLOGICAL RESULTS

INVENTORY IN SHIPPING CASK RELEASED TO ENVIRONMENT IN CURIES

	RELEASE TO	
ISOTOPE	REACTOR BUILDING	ENVIRONMENT
I-131	1.08E+01[a]	1.08E-01
KR-85	1.10E+04	1.10E+04
XE131	5.49E+00	5.48E+00
XE-133	1.25E-01	1.25E-01

470.7

METEOROLOGY AND DOSE RESULTS

DISTANCE (M)	χ/Q (SEC/M^3)	THYROID (REM)	WHOLE BODY (REM)
300	1.18E-03	6.57E-02	7.24E-03
500	4.83E-04	2.69E-02	2.96E-03
800	2.19E-04	1.22E-02	1.34E-03
1000	1.77E-04	9.85E-03	1.09E-03
1200	1.48E-04	8.27E-03	9.10E-04
1500	1.19E-04	6.66E-03	7.33E-04
2000	9.01E-05	5.02E-03	5.53E-04
2500	7.22E-05	4.03E-03	4.44E-04
3000	6.02E-05	3.36E-03	3.70E-04
3500	5.16E-05	2.88E-03	3.17E-04

[a] 1.08E+01 = $1.08 \times 10^{+1}$

470.7