UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of General Electric Company Standard Plant	)))	Docket No. STN 50-447

AMENDMENT NO. 18 TO APPLICATION
FOR REVIEW OF 238 NUCLEAR ISLAND GENERAL ELECTRIC
STANDARD SAFETY ANALYSIS REPORT (GESSAR II)

General Electric Company, applicant in the above captioned proceeding, hereby files Amendment No. 18 to the 238 Nuclear Island General Electric Standard Safety Analysis Report (GESSAR II).

Amendment No. 18 further amends GESSAR II by:

1. Incorporation of the following General Electric submittals:

   a. April 28, 1983 letter pertaining to post-accident monitoring instrumentation (Appendix 1D).

   b. June 15, 1983 letter pertaining to Question 410.24 of the Commission's August 25, 1982 information request and a summary of the measures taken to avoid intergranular stress corrosion cracking.

   c. June 28, 1983 letter pertaining to a revised response to the CP/ML Rule (Appendix 1G).

2. Updating the GESSAR II/FSAR interface tables of Subsection 1.9.1.

3. Clarifying portions of the text where obvious discrepancies exist.

Respectfully submitted,

General Electric Company

By: Glenn Sherwood
Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss

On this 8th day of July in the year 1983, before me, Karen S. Vogelhuber, Notary Public, personally appeared Glenn G. Sherwood, personally proved to me on the basis of satisfaction evidence to be the person whose name is subscribed to this instrument, and acknowledged that he executed it.

OFFICIAL SEAL
KAREN S. VOGELHUBER
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My Commission Expires Dec. 21, 1984

Karen S. Vogelhuber
NOTARY PUBLIC, STATE OF CALIFORNIA

I07072*-1

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General Electric Company

By: s/Glenn G. Sherwood
Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss

On this 8th day of July in the year 1983, before me, Karen S. Vogelhuber, Notary Public, personally appeared Glenn G. Sherwood, personally proved to me on the basis of satisfaction evidence to be the person whose name is subscribed to this instrument, and acknowledged that he executed it.

By: s/Karen S. Vogelhuber
Notary Public - California
Santa Clara County
My Commission Expires
December 21, 1984
175 Curtner Avenue
San Jose, CA 95125

I07072*-2

INSTRUCTIONS FOR FILING AMENDMENT NO. 18

Remove and insert the pages listed below. Dashes (----) in the remove or insert column indicate no action required.

Remove		Insert
	Chapter 1	
1.5-13,1.5-15/1.5-16, 1.8.117-1,1.9-4.1-19, 1.9-4.1-20,1.9-4.3-3, 1.9-4.3-3a, and 1.9-4.3-4		1.5-13,1.5-15/1.5-16, 1.5-17,1.5-18,1.8.117-1, 1.9-4.1-19,1.9-4.1-20, 1.9-4.3-3,1.9-4.3-4, and 1.9-4.3-4a
	Appendix 1D	
1D.1-1,1D.2-5,1D.2-14, 1D.2-15, and 1D.2-16		1D.1-1,1D.2-5,1D.2-5a, 1D.2-12a,1D.2-14,1D.2-15, 1D.2-16,1D.4-0, and 1D.4-0a through 1D.4-0e
	Appendix 1G	
1G-v/1G-vi,1G.12-1/1G.12-2, and 1G.21-2		1G-v/1G-vi,1G.12-1/1G.12-2, 1G.12-3/1G.12-4,1G.21-2, and 1G.21-2a
	Chapter 3	
3.2-2,3.2-16,3.2-31,3.2-34, 3.2-38,3.2-57/3.2-58,3.6-17, 3.9-36e, and 3.9-38		3.2-2,3.2-16,3.2-31,3.2-34, 3.2-38,3.2-57,3.2-58,3.6-17, 3.9-37, and 3.9-38
	Chapter 4	
4.2-1b,4.2-1c,4.2-3,4.2-5, and 4.2-7/4.2-8		4.2-1b,4.2-1c,4.2-3,4.2-5, and 4.2-7/4.2-8
	Chapter 5	
5.1-12,5.2-v/5.2-vi,5.2-ix/ 5.2-x,5.2-7,5.2-25,5.2-103, and 5.2-104		5.1-12,5.2-v/5.2-vi,5.2-ix/ 5.2-x,5.2-7,5.2-25,5.2-103, and 5.2-104

Remove	Insert
Chapter 19	
19.3.9.24-2	19.3.9.24-2 and 19.3.9.24-3/ 19.3.9.24-4

Table 1.5-2

COMMITMENT ITEMS

No.	Additional Technical Information	Reference Where Item Discussed
1.	Preop Piping Vibration Test Program	Subsection 3.9.2
2.	Reactor Internals Preop Vibration Test Program Results	Subsection 3.9.2
3.	Dynamic Analysis of Reactor Internals and Piping	Section 3.9
4.	Seismic Qualification of Class IE Electrical Equipment	Subsection 3.9.2.2 & 3.10
5.	Environmental Qualification of Class IE Electrical Equipment	Section 3.11
6.	Electrical Isolation Devices Test Program Results	Chapter 7
7.	Fuel Experience Update - NEDO-10505	See Note A
8.	Fuel Surveillance Program Results	See Note B
9.	Fuel Assembly Components Stress Report	See Note C
10.	Fuel Assembly Pressure and Temperature Capability	See Note D
11.	Fuel Assembly Dynamic Analysis	See Note C
12.	Fuel Assembly Analysis Method for Creep-Rupture	See Note E
13.	Fuel Assembly Design Limit for Instability	See Note D
14.	Fuel Channel Deformation Analysis Methods	See Note F
15.	Fuel Assembly Stress Limits	See Note D
16.	Fuel Rod 0.060 Inch Deflection Justification	See Note D
17.	Gadolinia Rods Performance Experience	See Note G
18.	Process Computer Performance Evaluation Accuracy Update	See Note H
19.	Lattice Physics Methods Verification	See Note I
20.	Boiling Water Reactor Simulator Verification	See Note J
21.	Void and Doppler Reactivity Coefficients	See Note K
22.	Full Power Scram Reactivity Function	See Note L
23.	Feedwater Flow Rate Uncertainty Justification	See Note M
24.	Resolution of Feedwater Nozzle Design and Verification	See Note N
25.	Description of WHAM Code and Loads on Internals During LOCA	See Note O
26.	Safety/Relief Valve Surveillance Program Details	Section 1E.3
27.	Update PGCC LTR NEDO-10466	See Note P
28.	Analytical Methods of Plant Transient Evaluation	Chapter 15
29.	ATWS	Section 15.8
30.	Test Program for Safety/Relief Valve Solenoids	See Note R
31.	Fire Protection for PGCC	See Note P Appendix 9A
32.	Primary Coolant Pump Seals Leakage Characteristics	See Note S
33.	Large Scale Mark III Test	Appendix 3B
34.	Environmental Design of Isolation Valves and Safety Related Equipment	See Note T
35.	Post LOCA Manual Operator Actions	See Note U
36.	Instrument and Control Systems	Chapter 7
37.	HPCS Onsite Electrical Systems	Chapter 8
38.	Fire Protection for Nuclear Island Conformance	Appendix 9A

Development and Verification Test Programs

No.	Item	Reference
1.	Fuel Surveillance Program	See Note B
2.	Safety Relief Valve Surveillance Program	Subsection 5.2.2
3.	Core Spray Distribution	See Note V
4.	Fast Scram Design Verification	See Note W
5.	Feedwater Nozzle Design Verification	See Note N
6.	Long Term Pipe Replacement Program	See Note Z
7.	Instrumentation for Vibration and Loose Parts Detection	X
8.	Pressure Suppression Design Verification	Y
9.	Suppression Pool Dynamics	Appendix 3B
10.	Evaluate Effects of Relief Valve Blow-Down	Appendix 3B and Chapter 15

Table 1.5-2

FDA COMMITMENT ITEMS (Continued)

NOTES

A. (1) Information Report, NEDO-20922-P, Experience with BWR Fuel through September 1974, 7/17/75

(2) Information Report, NEDO-21660-P, Experience with BWR Fuel through December 1976, 12/14/77

B. (1) Letter R. Engel to D. Ross, dated 7/11/77

(2) Letter G. G. Sherwood to D. Ross, dated 4/7/78

C. Licensing Topical Report, NEDO-21175, BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake and Loss-of-Coolant Accident Loadings

D. Licensing Topical Report, NEDO-20948, BWR/6 Fuel Design, 2/20/76

E. Licensing Topical Report, NEDE-20606-A, Creep Collapse Analysis at BWR Fuel Using Safe-Colaps Model, Approved by NRC 8/76.

F. Licensing Topical Report, NEDO-21354, BWR Fuel Channel Mechanical Design and Deflections

G. Licensing Topical Report, NEDO-20943, Urania-Gadolinia Nuclear Fuel Physical and Irradiation Characteristics and Material Properties

H. Licensing Topical Report, NEDO-20340, Process Computer Performance Evaluation Accuracy, 7/17/74

I. Licensing Topical Report, NEDO-20939-A, Lattice Physics Methods Verification, Approved by NRC 9/22/76.

J. Licensing Topical Report, NEDO-20946-A, BWR Simulator Methods Verification, Approved by NRC 9/22/76

K. Licensing Topical Report, NEDO-20964, Generation of Void and Doppler Reactivity Feedback for Application to BWR Design, 2/13/76

L. Appendix A of the Odyn Report

M. Letter G. G. Sherwood to E. Case, dated 5/1/78

Table 1.5-2

FDA COMMITMENT ITEMS (Continued)

NOTES (Continued)

N. Information Report NEDO-21821-A Boiling Water Reactor Feedwater Nozzle/Sparger Final Report.

O. Licensing Topical Report, NEDO-24048, Evaluation of Acoustic Pressure Loads on BWR/6 Internal Components, 12/1/77

P. Licensing Topical Report, NEDO-10466-A, Power Generation Control Complex Design Criteria and Safety Evaluation, Approved by NRC 7/31/78

Q. Not used.]

R. (1) Letter G. G. Sherwood to E. Case, dated 2/18/78

(2) Letter S. Varga to G. G. Sherwood, MFN-183-78

(3) Information Report NEDO-23978, ADS Solenoid Valve Reliability Demonstration

S. Licensing Topical Report, NEDO-24083, Recirculation Pump Shaft Seal Leakage Analysis, 12/12/78

T. Letter G. G. Sherwood To N. Denton, dated, 10/11/78

U. Letter G. G. Sherwood to H. Denton dated 3/22/79

V. (1) NEDO-10846, Boiling Water Reactor Core Spray Distribution

(2) NEDO-20566-3 Effect of Steam Environment on BWR Core Spray Distribution

W. NEDO-24142, Fast Scram Control Rod Drive Qualification Program

X. Letter, G. G. Sherwood to E. Case, dated 3/8/78

Y. Licensing Topical Report, NEDO-20533, The GE Mark III Pressure Suppression Containment System Analytical Model, approved by NRC 8/14/75 (Amended 6/30/78)

Table 1.5-2

FDA COMMITMENT ITEMS (Continued)

NOTES (Continued)

Z. The GESSAR II design complies with Regulatory Guide 1.44, Rev. 0, and with the requirements of NUREG-0313, Rev. 1. Regulatory Guide 1.44 addresses 10CFR50, Appendix A, GDCs 1 and 4, and Appendix B, i.e., requirements for components important to safety shall be designed in accordance with appropriate codes and to accommodate the effects of and to be compatible with the environmental conditions associated with reactor operations. Experience with operating reactors has demonstrated that certain wrought austenitic stainless steels, when welded, are sensitized to the degree that they are susceptible to stress corrosion cracking in reactor coolant water environments. With selection of alloys and control of thermomechanical processing, intergranular stress corrosion cracking (IGSCC) of reactor coolant pressure boundary components can be avoided.

All austenitic stainless steel material that is fabricated into components which see temperatures in reactor environments greater than 200°F is purchased as low carbon grade or nuclear grade and in the solution annealed condition, in accordance with the applicable ASTM and ASME specifications.

Cooling rates from solution annealing heat treatment temperatures are required to be rapid enough to prevent sensitization. Resistance to IGSCC is verified using ASTM A262, Practice A methods.

Material changes have been made to minimize the possibility of IGSCC. All welded wrought austenitic stainless steel in the reactor coolant pressure boundary is low carbon nuclear

Table 1.5-2

FDA COMMITMENT ITEMS (Continued)

NOTES (Continued)

Z. (Continued)

grade 316LN with 0.02% maximum carbon content and nitrogen control for strength. There is no piping which is service sensitive or nonconforming as defined in NUREG-0313, Rev. 1. All other applications of stainless steel (Types 304 and 316) are of the L grade (0.03% maximum carbon content).

Welding heat input controls are required for all stainless steel welds. For machine, automatic and manual welding, interpass temperatures are restricted to 350°F maximum for all stainless steel welds. High heat welding processes such as block welding and electroslag welding are not permitted. All weld filler metal consumable inserts and castings are required by specification to have a minimum of 5% ferrite.

The above practices avoid IGSCC and are reflected in Subsections 1.8.44, 4.5.1, 4.5.2, and 6.1.1.

1.8.117 Regulatory Guide 1.117, Revision 1, Dated April 1978

Title: Tornado Design Classification

This guide describes a method for identifying those structures, systems, and components of light-water-cooled reactors that should be designed to withstand the effects of the design basis tornado (Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants) including tornado missiles and remain functional.

Evaluation

The BWR/6 Mark III Standard Plant Tornado Design Classification of structures, systems, and components including their foundations and supports that are designed or protected to withstand the effects of a design basis tornado per Regulatory Guide 1.76 (including tornado missiles) without loss of capability to perform their safety function) are those listed in the Appendix of Regulatory Guide 1.117 except for portions that are not within the Nuclear Island scope.

The analysis of structures, shields, and barriers indicates tornado winds could damage non-Category I structures. However, ability to shutdown the reactor, integrity of the containment, and capability of the essential Heat Removal Systems are not impaired. All Seismic Category I systems are protected by being housed in tornado-resistant structures. Collapse of non-Category I towers or stacks will not endanger Category I structures since plant arrangement provides sufficient distance between them.

1.8.117 Regulatory Guide 1.117, Revision 1, Dated April 1978 (Continued)

The specific plant Applicant is responsible for the design and protection of GE out-of-scope-of-supply structures, systems, and components.

Table 1.9-1

CHAPTER 1
GESSAR II/FSAR INTERFACES (CONTINUED)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	INTERFACE CATEGORY
1.127	Alternative Hydrogen Control System	Provide a costs and benefits comparison of the alternative systems considered for a Hydrogen Control System. For the selected system, provide design description, function layout, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of 10CFR50.34.	1G.12-1	1G.12	3
1.128	Long-Term Training Upgrade	Establish a training program which addresses the concerns related to Item I.A.4.2 of NUREG 0718.	1G.13-1	1G.13	3
1.129	Long-Term Program of Upgrading of Procedures	Establish a program for integrating and expanding current efforts to improve plant procedures.	1G.14-1	1G.14	3
1.130	Hydrogen Control System	Provide a Hydrogen Control System capable of handling equivalent of a 100% active fuel-clad metal water reaction.	1G.21-2	1G.21	3
1.131	Purging	Provide performance information of purge valves	1G.27-1	1G.27	3

Table 1.9-1

CHAPTER 1
GESSAR II/FSAR INTERFACES (CONTINUED)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	INTERFACE CATEGORY
1.132	Upgrade License Emergency Support Facility	Provide Technical Support Center, an Onsite Operational Support Center and a near site Emergency Operations Facility.	1G.37-1	1G.37	3
1.133	In-Plant Radiation Monitoring	Provide monitoring of in-plant radiation and airborne radioactivity for routine and accident conditions.	1G.39-1	1G.39	3
1.134	Feedback of Operating, Design and Construction Experience	Provide administrative procedure for evaluating operating, design and construction experience and ensure applicable important industry experience is provided to other plants.	1G.41-1	1G.41	3
1.135	Expansion of QA List	Ensure that the Quality Assurance list required by Criterion II, App. B. 10CFR50, includes all structures, systems, and components important to safety.	1G.42-1	1G.42	3
1.136	Containment Penetration	Provide details of containment penetration arrangement.	1G.44-1	1G.44	5
1.137	Containment Integrity	Provide containment vessel design capability of 45 psig for Service Level C.	1G.45-3	1G.45	3

Table 1.9-3

CHAPTER 3
GESSAR II/FSAR INTERFACES (Continued)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
3.15	Seismic Category I Structure Descriptions	Describe other seismic Category I structures not within the Nuclear Island.	3.8-98	3.8.4.1.8		1
3.16	Codes, Standards and Specifications	Provide applicable codes, standards and specifications for other seismic Category I structures not within Nuclear Island.	3.8-102	3.8.4.2.5		1
3.17	Loads and Load Combinations	Provide loads and load combinations appropriate to Seismic Category I structures not within Nuclear Island	3.8-114	3.8.4.3.8		1
3.18	Design and Analysis Procedures	Provide design and analysis procedures for seismic Category I structures now within Nuclear Island	3.8-126a	3.8.4.4.8		1
3.19 (Deleted)						

Table 1.9-3

CHAPTER 3
GESSAR II/GSAR INTERFACES (Continued)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
3.20	Materials Criteria	Provide structural acceptance criteria for Seismic Category I structures not within Nuclear Island.	3.8-128	3.8.4.5.3		1
3.21	Materials, QA and Special Construction Techniques	Describe materials, quality control, and special construction techiques for Seismic Category I structures not within Nuclear Island.	3.8-130	3.8.4.6.7		1
3.22	Testing and Inservice Inspection	Describe testing and inservice inspection requirements for Seismic Category I structures not within Nuclear Island.	3.8-131	3.8.4.7.4		1
3.23	Verification of Foundation Soil	Provide verification, through Applicant's soils consulting engineer, that the Nuclear Island foundation soil meets the soil requirements of Appendix 3A, subsection 3.8.6 and 3.7.	3.8-142	3.8.6.2		3
3.24	Containment Vessel Analysis Summary	Provide a summary of the containment vessel analysis.	3.8-146	Table 3.8-3		1
3.24.1	NUREG-0800 Compliance on Computer Programs	Update computer programs and indicate method of verification and the version used.	3.9-1a	3.9.1.2	3.151[MEB (DSER) Item No. 39]	3
3.24.2	Pump and Motor Vendor Computer Programs	Provide a list of the pump and motor vendor computer programs along with statement of verification approach.	3.9-1b	3.9.1.2	3.151[MEB (DSER) Item No. 39]	2
3.24.3	Program Verification	Provide method of verification and version of PDG1 used.	3.9-24	3.9.1.2.6.3.1.2		5

Table 1.9-3

CHAPTER 3
GESSAR II/FSAR INTERFACES (Continued)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
3.24.4	Inservice Testing of Pumps and Valves	Provide details of the pump and valve inservice testing program, including test schedules and frequencies. Also, applications for written relief from Section XI Addendum requirements, pursuant to 10CFR50, Section 55 a(g) (6) (i).	3.9-125 and 3.9-126	3.9.6		3
3.25	Major Safety-Related Mechanical Components	Provide equipment loading conditions, stress criteria, limiting stress types, allowable and calculated stresses on a component-by-component basis.	3.9-177 3.9-179 through 3.9-185 3.9-187 through 3.9-190 3.9-191 through 3.9-192	Table 3.9-11 (2) Table 3.9-11 (3) Table 3.9-11 (4) Table 3.9-11 (5)		2,4

APPENDIX 1D

ASSESSMENT OF REGULATORY GUIDE 1.97, REVISION 2 AGAINST GESSAR II DESIGN

1D.1 SUMMARY

This report provides an assessment of the 238 Nuclear Island design described by GESSAR II against the guidelines of Regulatory Guide 1.97, Revision 2, dated December 1980, "Instrumentation for Light Water Cooling Nuclear Power Plant to Assess Plant and Environmental Conditions During and Following an Accident." This assessment is summarized in Table 1D-0.

1D.1.1 Introduction

Regulatory Guide 1.97 describes a method acceptable to the NRC staff for compliance with the Commission's regulation to provide instrumentation to monitor plant variables in systems during and following an accident at light water cooled nuclear power plants. Although the intent of the Regulatory Guide has been met, several exceptions have been made to the Regulatory Guide as written. These exceptions and their bases are discussed further in this report. These exceptions are viewed as acceptable means to implement the guide's intent.

All variables identified by the regulatory guide are identified in this report. Table 1D-1 summarizes these variables. However, since some variables are outside of the 238 Nuclear Island scope, they are left for the Applicant to assess.

1D.1.1 Introduction (Continued)

Since there are other modifications being made to address post-TMI changes (Appendix 1A), the assessment makes use of the existence of these changes and does not repeat a description of them here.

1D.1.2 Objective

This assessment is conducted in three stages. The first stage develops an implementation position which defines the specific requirements against which each variable is assessed. The implementation position is included as Attachment A and is discussed in Subsection 1D.2.1.

The second stage of the assessment defines the Type A variables. These variables are identified from a review of design basis accidents in Chapter 15 and the Emergency Procedure Guidelines. A more detailed discussion of the Type A variables is given in Subsection 1D.2.2.

The third stage of the assessment is a review of each variable shown on Table 1D-1. For this review, statements are provided to either justify the current design or to provide a conceptual definition of changes necessary to comply with the regulatory guide. These discussions may be found in Subsection 1D.2.3.

1D.1.3 Conclusions

Table 1D-2 summarizes the results of the assessment. The degree of compliance is shown by the information in the table and the accompanying remarks.

Table 1D-2 shows that the 238 Nuclear Island, as modified by the recommendations under review as a result of this assessment, fully complies with the intent of Regulatory Guide 1.97, Revision 2.

1D.2.3.3 Reactor Coolant Boron Concentration

Boron concentration in the reactor water is determined by analysis of a reactor water sample obtained from the post-accident sample station (refer to Subsection 1D.2.3.38). Recommended boron analysis procedures are included with the operation and maintenance manual supplied with the sample station. Actual procedures are left for the applicant to provide.

1D.2.3.4 RPV Water Level Indication

The existing post-accident water level indication system consists of three wide-range instruments (calibrated for full pressure) covering the range -160" to +60" and two fuel zone instruments (calibrated for atmospheric pressure) covering the range -326" to -116". Narrow-range indication is not included in the assessment.

The RPV water level indication system is the primary variable indicating the availability of adequate core cooling and is considered acceptable, without diverse methods of indication, provided adequate redundancy and unambiguity are provided over the entire range of interest.

The range for coolant level in the reactor specified on Table 1D-1 of the implementation position has been modified to require indication only "above normal reactor water level" rather than the "centerline of main steam line" as specified in the regulatory guide. This change was made because the high reactor water level transient has been shown to be not a safety concern (Subsection 1A.23) especially considering the existence of improved high reactor water level trips on HPCS. Furthermore, since the presence of reactor water on the normal range assures adequate core cooling, higher ranges of indication are not needed for this purpose. Although Contingencies 5 and 6 of the Emergency Procedure Guidelines discusses flooding of the main steam lines for alternate shutdown cooling and instrument reference legs refill, these

1D.2.3.4 RPV Water Level Indication (Continued)

actions are not needed to achieve adequate core cooling. Other instrumentation is used to carry out high water level flooding procedure (see Attachment A).

1D.2.3.4 RPV Water Level Indication (Continued)

The requirement in Table 1D-1 of the regulatory guide for BWR core thermocouples has been replaced with a requirement that the reactor water level indication should extend from below the core plate to above the pressure vessel Level 1 as adequate indication of core cooling. Through work by General Electric and BWR Owners' Group (References 2 and 3), the NRC staff has recently retracted the requirement for BWR core thermocouples (Reference 4), but they have requested that an investigation continue to identify an acceptable diverse means of indicating adequate core cooling. This investigation is still in progress. Work by General Electric and the BWR Owners' Group (Reference 3) has provided the NRC staff with extensive information on the relationship between the reactor water level and adequate cooling of the core. These studies have shown that as long as at least one of the water delivery systems is available and flowing to the reactor pressure vessel, that adequate core cooling exists. Indication of these flows is already required by the Regulatory Guide. Consequently, it is General Electric's view that no instrumentation other than RPV level indication is required to assure indication of adequate core cooling.

Subsection C.1.3.1.b states that additional instrumentation should be provided to allow the operator to determine the actual conditions of the plant "when failure of one accident monitoring channel results in information ambiguity." Three independent channels of indication are considered acceptable to meet this provision.

During a small-break accident which results in increasing drywell temperatures, the accuracy of the RPV water level indication varies significantly. This effect is described in detail in Reference 2.

A small drywell break could lead to ambiguity in all instrument ranges either because the redundant channels would not agree (if one failed) or because of increasing drywell temperatures and its

1D.2.3.12 Primary Containment Pressure

Also, 35 hours are available before reaching the top of the indicating range (Table 18.2-3a), which provides sufficient time to open MSIVs or re-establish containment cooling.

1D.2.3.12 Primary Containment Pressure (Continued)

Two channels of pressure instrumentation are adequate in case of a single failure because diverse indications to monitor containment integrity are provided.

No modifications are needed for this instrumentation. However, the instruments including displays should be included in Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.13 Drywell/Containment Hydrogen Concentration

Drywell and containment hydrogen concentration may be determined by analysis of samples obtained from the post-accident sample station (Subsection 1D.2.3.38). On-line instrumentation is the responsibility of the Applicant to provide.

1D.2.3.14 Secondary Containment Area Radiation

Area radiation levels in the secondary containment are defined in Section 12.3. The specific location and range of area radiation monitors are the responsibility of the Applicant to provide.

1D.2.3.15 Secondary Containment Noble Gas Effluent

Applicant to provide.

1D.2.3.16 Containment Noble Gas Effluent

Applicant to provide.

1D.2.3.17 Suppression Pool Temperature

Two temperature sensors per quadrant of the suppression pool are provided in the 238 Nuclear Island design with control room

1D.2.3.17 Suppression Pool Temperature (Continued)

indication and recording. The instruments are discussed in Subsections 7.6.1.11 and 7.6.1.12.

The sensors are located in the upper third of the suppression pool and thus provide a conservative indication of suppression pool temperature for use on the Emergency Procedure Guidelines.

No modifications are needed to these instruments. The instruments including the displays should be included in Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.18 Drywell Air Temperature

Two Class 1E channels of drywell temperature indication are provided on the control room BOP benchboard (P870) as shown in Figure 9.4-5.

The range of the display (up to 400°F) is adequate to carry out the functions prescribed by the Emergency Procedure Guidelines (Reference 1). Highest post-LOCA drywell temperature is approximately 340°F for a main steam line break (Figure 6.2-12).

No modifications to the design are needed. The instruments including displays should be included on Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.19 Coolant Radiation

No instrumentation is provided in the current design to monitor radioactivity levels in the primary coolant and no changes to the plant design are planned.

The specified range for the potential instrument (1/2 Technical Specification Limits (TSL) to 100 times TSL) suggests that the

1D.2.3.19 Coolant Radiation (Continued)

purpose of this instrument is to assess coolant radiation level during routine plant operation. The current design provides sampling capability for reactor coolant as described in Subsection 9.3.2 and provides offgas and mainstream line process radiation measurement as discussed in Section 11.5 for detection of fuel cladding breaches.]

The value for the technical specification limit has not been established by the staff in standard technical specifications for BWR/6. Subsection 16.3/4.4.5, however, indicates that the TSL is 2 μCi/g Iodine-131 equivalent. On-line reactor coolant monitoring of this level of coolant activity may be impractical during normal operation because of the additional contributions to the detector from other isotopes such as circulating Nitrogen-16 or Cobalt-60 deposited on reactor coolant piping.

Furthermore, should a significant breach of the fuel cladding occur the expected levels of iodine in the reactor coolant would far exceed the upper range specified by the regulatory guide for this instrument. The samples provided by the post-accident sample system (Subsection 1D.2.3.38) will provide quantification of the coolant radioactivity level. Post-Accident Sample Station identifies gross gamma indication (Figure 1AB.1-1a). Under these conditions, an on-line monitor would serve no useful purpose.

1D.2.3.20 Coolant Gamma Sample

The radioactivity content of the reactor water is determined by analysis of a reactor water sample obtained from the post-accident sample station (Subsection 1D.2.3.38). Recommended procedures to determine the gross activity in the coolant are included with the operation and maintenance manual supplied with the sample station. Actual procedures are the responsibility of the Applicant.

1D.2.3.21 MSIV Leakage Control System Pressure

The current design uses a Class 1E positive leakage control system as described in Section 6.7. Proper system function is monitored and recorded by air system flows rather than system pressure as specified by the regulatory guide. System isolation automatically occurs on high flow or low differential pressure between the RPV and the pressurized lines.

The flow monitoring instrumentation is considered adequate to meet the intent of the regulatory guide to indicate proper system function. No changes are planned for this system.

1D.2.3.22 RHR System Flows

The RHR system Loops A and B serve a variety of functions among them being low pressure coolant injection, containment spray, suppression pool cooling, and shutdown cooling. Loop C is only used for LPCI mode of operation. The valving arrangements (refer to Figure 5.4-12a) required to achieve these different functions of the RHR System occur downstream of the flow element and flow transmitter which is used to indicate RHR System flow. This instrument channel therefore provides the operator with flow indication during any of these operating modes for the RHR System.

From an operational point of view, proper functioning of the containment spray mode of the RHR System is provided by the containment temperature (Subsection 1D.2.3.11) or containment pressure instrumentation (Subsection 1D.2.3.12). Should the containment spray mode be used, it is anticipated that the operators would only initiate flow long enough to decrease these containment parameters at which time flow in the containment spray mode would be terminated. Thus, the primary indicator of proper containment spray mode operation is the containment pressure or temperature indication rather than the RHR System flow.

Table 1D-0

SUMMARY OF ASSESSMENT OF REGULATORY GUIDE 1.97 AGAINST GESSAR II DESIGN

Variable	Deviation	Justification	GESSAR II Reference	Remarks
1. Neutron Flux	New system not yet approved by NRC	• Will meet Reg. Guide 1.97	1D.2.3.1	-
2. MSIV LCS Pressure Relief	MSIV LCS flow considered as alternate variable (FRS R607, FRS 627)	• Indication of high flow (MSIV excess leakage) or zero flow indicates system malfunction.	1D.2.3.21	-
3. RHR Heat Exchanger Temperature	RHR Service Water flow (FI R602A, B) considered an alternate variable (Figure 9.2-1a)	• Meets all Category 2 requirements • Positive indication of heat removal capability	1D.2.3.23	-
4. SLC Flow	SLC pressure considered an alternate variable (PI R600)	• Indicates proper system function - High pressure indicates flow blockage - Erratic or low pressure indicates line break • Back-up information available - Squib valve position - Neutron flux - SLC tank level	1D.2.3.25	Qualification of indicator required. Change to "Essential MPL classification

Table 1D-0

SUMMARY OF ASSESSMENT OF REGULATORY GUIDE 1.97 AGAINST GESSAR II DESIGN (Continued)

Variable	Deviation	Justification	GESSAR II Reference	Remarks
5. SGTS Common Vent Flow Rate	Damper position is an alternate variable	• Indicator in control room (Figure 7A.3-9A) • Design flow rate may be used for release assessment if isokinetic flows are unavailable	1D.2.3.34	-
6. LPCI Flow	For LPCI Loops A and B injection valve position and RHR flow is alternate variable (Figure 5.4-12b)	• Valve position sensors are qualified; indication in control room • RPV water level is a back-up variable.	1D.2.3.22	-
7. Containment Spray Flow	Containment spray valve position and RHR flow is alternate variable	• Valve position sensors are qualified indications in control room • Containment pressure indication is back-up variable	1D.2.3.22	-
8. RPV Level	Alternate range for Category I indication is to top of wide range zone (Note: Deviation assumes enhanced level instrument (ELI) to meet the intent of RG 1.97)	• Existing shutdown range is adequate to main steam lines (meets Category 3)	1D.2.3.4	-

Table 1D-0

SUMMARY OF ASSESSMENT OF REGULATORY GUIDE 1.97 AGAINST GESSAR II DESIGN (Continued)

Variable	Deviation	Justification	GESSAR II Reference	Remarks
9. Incore Temperature	RPV water level measurement satisfies requirements for core cooling indication	• ELI provides unambiguous indication of approach to and existence of ICC • EPGs provide operator actions to prevent ICC • ELIs Third Division eliminates need for diversity. • ECCS flow indication provides indication of recovery from ICC (supported by NEDO-24708-A analysis)	1D.2.3.4	-
10. Containment Pressure	Existing range and method of indication adequate	• 35 hours available before reading top of indicating range - time available to open MSIVs or re-establish containment cooling (Table 8.2-3a) • Trend information is useful, but not essential to follow EPG.	1D.2.3.12	Change design to 0-60 psig range. Change pressure indicator to recorders. (RG 1.97, Rev. 2 requires all Category 1 channels to be recorded.

Table 1D-0

SUMMARY OF ASSESSMENT OF REGULATORY GUIDE 1.97 AGAINST GESSAR II DESIGN (Continued)

Variable	Deviation	Justification	GESSAR II Reference	Remarks
11. Suppression Pool Water Level	The existing range is adequate for expected situations	• Low End of Scale - Upper pool dump initiates at 18'5" (raises water level) - High suppression pool level causes auto transfer to pumps to containment to pool (20'5") - Lower tap risks failing following SRV action or vent clearing High End of Scale - Drywell sump level indicates weir flow - ADS causes ≅ 5' pool rise (within scale of indication)	1D.2.3.9	-
12. Drywell Air Temperature	The lower range is acceptable	Highest post-LOCA drywell temperature is ≈340°F for a main steam line break. (Figure 6.2-12)	1D.2.3.18	-
13. Coolant Radiation	Post-Accident Sample System provides adequate information	• Interference by N-16 and Co-60 may make monitoring of TSL impractical or ambiguous	1D.2.3.19	-

Table 1D-0

SUMMARY OF ASSESSMENT OF REGULATORY GUIDE 1.97 AGAINST GESSAR II DESIGN (Continued)

Variable	Deviation	Justification	GESSAR II Reference	Remarks
13. Coolant Radiation (cont)		• Significant fuel release would exceed 100 TSL (100 TSL is not a public health risk) • Post-accident sample station identifies gross gamma indication (Figure 1AB.1-1a) - indicates radiation of circulating sample prior to sample		
14. Post-Accident Samples	Sump samples are not needed	• Sumps are isolated (not sources of release) • RPV or suppression liquid sample provides indication of extent of core damage • Process radiation monitors are used for release assessment • Analysis of sumps would be ambiguous	1D.2.3.38	-

Table 1D-0

SUMMARY OF ASSESSMENT OF REGULATORY GUIDE 1.97 AGAINST GESSAR II DESIGN (Continued)

Variable	Deviation	Justification	GESSAR II Reference	Remarks
15. Drywell Sump Level	Category 3 (commercial, single channel) requirements are acceptable	• Sumps isolated by accident • Back-up indication to drywell pressure and radiation level • Early indication only	1D.2.3.6	Add Category 3 differential pressure wide range instrumentation
16. SLC Tank Level	Present design is adequate	• Back-up variable to SLC pressure • Air supply reliability is assured until isolated air supply bleeds off.		

APPENDIX 1G

LIST OF TABLES

LIST OF ILLUSTRATIONS

1G.12 EVALUATION OF ALTERNATIVE HYDROGEN CONTROL SYSTEMS [Item (1) (xii)]

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f) (2) (ix) of 10CFR50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f) (2) (ix) of 10CFR50.34.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

Response

(A) GE has conducted evaluations of the various hydrogen control concepts for the GESSAR II design. These concepts included distributed ignition, catalytic burners and post-accident inerting with carbon-dioxide or halon. Of the concepts considered, only distributed ignition (igniters) and carbon-dioxide post-accident inerting appear to be viable alternatives. The costs and benefits of these two alternatives are summarized in Table 1G.12-1.

(B) The Applicant will provide the analyses and test data to verify compliance with the requirements of 10CFR50.34(f)(2)(ix).

(C) The Applicant will provide the design descriptions of equipment, function, and layout.

Table 1G.12-1

COMPARISON OF HYDROGEN CONTROL ALTERNATIVES

Item	Igniters	Post-Accident Inerting
Description	Distributed ignition systems controlled burn at low H_2 concentration	Liquid CO_2 discharged into containment air-space (prevents combustion)
Cost (Order of Magnitude)	$1 Million*	$10 Million*
R&D Concerns	Flammability, mixing, pressure response	Possible partial inerting flammability characteristics. Mixing, effects on electronic equipment
R&D Programs	Underway EPRI/NRC	None planned
Pros	• Minor impact of inadvertent operation • Low cost • Minimum design impact • Lower containment pressures	• No heat loads • No dependence on H_2 generation rate • Minor impact on existing equipment • AC power not required for inerting
Cons	• Potential for large equipment qualification program • Assurance of combustion at low concentrations • Sensitize to hydrogen generation rate and containment entry point • Requires active heat removal	• Inadvertent actuation has potential adverse impact on plant operation • High containment pressure • High cost • Some redesign of containment piping to accommodate • Potential adverse effects from low temperatures during injection

*These costs do not include the cost of corresponding equipment qualification programs. Inclusion of equipment qualification costs could result in nearly equal total costs for the first Applicant referencing GESSAR II.

1G.21 HYDROGEN CONTROL SYSTEM PRELIMINARY DESIGN [Item (2) (ix)]

NRC Position

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1) (xii) of 10CFR50.34(f) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8)

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

1G.21 HYDROGEN CONTROL SYSTEM PRELIMINARY DESIGN [Item (2) (ix)] (Continued)

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Response

The Applicant will provide a Hydrogen Control System capable of handling hydrogen generated by the equivalent of a 100% active fuel-clad metal water reaction. Detailed descriptions of the selection of the hydrogen generation event, the progression of the event, and the methodology used in evaluating the hydrogen generation rate during the hydrogen generation event and contained in the April 1982 BWR/6 Mark III Hydrogen Control Owners' Group report.*

The Hydrogen Control System shall provide with reasonable assurance that:

(1) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel-clad metal water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

*S. S. Dua, et al., "BWR/6 Mark III Hydrogen Control Owners' Group Report on Hydrogen Control: Accident Scenarios, Hydrogen Generation Rates and Equipment Requirements," General Electric Company Report, April 1982.

1G.21 HYDROGEN CONTROL SYSTEM PRELIMINARY DESIGN [Item (2) (ix)] (Continued)

Response (Continued)

(2) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(3) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are considered important to safety because they perform safety actions required to avoid or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify structures, components, and systems according to the importance of the safety function they perform. In addition, design requirements are placed upon such equipment to assure the proper performance of safety actions when required.

3.2.1 Seismic Classification

Plant structures, systems, and components important to safety are designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional if they are necessary to assure:

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

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

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

Plant structures, systems, and components, including their foundations and supports, that must remain functional in the event of a safe shutdown earthquake are designated as Seismic Category I as indicated in Table 3.2-1.

Structures, components, equipment, and systems designated as Safety Class 1, Safety Class 2, or Safety Class 3 (see Subsection 3.2.3 for a discussion of safety classes) are classified as

3.2.1 Seismic Classification (Continued)

Seismic Category I except those portions of the radioactive waste treatment handling and disposal systems whose postulated simultaneous failure would not result in conservatively calculated potential offsite exposures comparable to the guideline exposures of 10CFR100.

All Seismic Category I structures, systems, and components are designed to withstand, without loss of function, the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads.

The seismic classifications indicated in Table 3.2-1 and shown in Figure 3.2-2 meet the requirements of NRC Regulatory Guide 1.29 except as otherwise noted in the table and as discussed in Subsection 1.8.29.]

3.2.2 System Quality Group Classifications

System quality group classifications as defined in NRC Regulatory Guide 1.26 have been determined for each water, steam, or radioactive waste containing component of those applicable fluid systems relied upon to:

(1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary;

(2) permit shutdown of the reactor and maintain it in the safe shutdown conditions; and

(3) contain radioactive material.

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
II (Continued)						
8. Piping - other beyond outermost isolation valves						
a. RPV heat vent	3	D	C	B	I	
b. Main steam drains	2	A	B	B	I	
9. Piping - instrumentation beyond outermost isolation valves	2/Other	D	B/D	B/N/A	I/N/A	(g)
10. Safety/relief valves	1	D	A	B	I	
11. Valves - main steam and feedwater isolation valves	1	C,D	A	B	I	
12. Valves, other - isolation valves and within outermost isolation valves						
a. RPV heat vent valves	1	D	A	B	I	(g)
b. Main steam drain valve	1	D,A	A	B	I	(g)
c. Main steam - first valve downstream of isolating valves	2	A	A	B	I	(g)
13. Valves - instrumentation beyond outermost isolation valves	2/Other	A	B/D	B/N/A	I/N/A	(g)
14. Mechanical modules - instrumentation with safety function	2	C	N/A	B	I	
15. Electrical modules with safety function	2	C	N/A	B	I	(i)

3.116

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
II (Continued)						
16. Cable with safety function	2	C,D,A,X	N/A	B	I	
III Reactor Recirculation System						
1. Piping	1/2	D	A/B	B	I	(g)
2. Pipe suspension - recirculation line	1	D	A	B	I	
3. Pipe restraints - recirculation line	other	D	N/A	N/A	N/A	
4. Pumps	1	D	A	B	I	
5. Valves	1/2	D	A/B	B	I	(g)
6. Motor - pump	other	D	N/A	B	I	
7. Cable with safety function	2	X,C,A,X	N/A	B	I	
8. LFMG Set	Other	T	N/A	N/A	N/A	
IV CRD Hydraulic System						
1. Valves - scram discharge volume lines	2	C	B	B	I	(g)
2. Valves insert and withdraw lines	2	C	B	B	I	(j)
	3.121		3.121	3.121		

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XXIX (Continued)						
4. Branch lines of MFL between the second isolation valve and the shutoff valve from the branch point at MFL to and including the first valve in the branch line	2	A	B	B	I	(u)
5. Main steamline piping between the MSL shutoff valve and the turbine main stop valve	Other	T	D	N/A	N/A	(u,v)
6. Turbine bypass piping	Other	T	D	N/A	N/A	(u)
7. Branch lines of the MSL between the MSL shutoff valve and the turbine main stop valve	Other	T	D	N/A	N/A	(u,v)
8. Turbine stop valve, turbine bypass valves, and the main steam leads from the turbine control valve to the turbine casing	Other	T	D	N/A	N/A	
9. Feedwater system components beyond the MFL shutoff valve	Other	T	D	N/A	N/A	
10. Turbine generator	Other	T	N/A	N/A	N/A	
11. Condenser	Other	T	N/A	N/A	N/A	

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XXIX (Continued)						
12. Air ejector equipment	Other	T	N/A	N/A	N/A	
13. Turbine gland sealing system components	Other	T	D	N/A	N/A	
XXX Offgas System						
1. Tanks	Other	T	D	N/A	N/A	(p)
2. Heat exchangers	Other	T	D	N/A	N/A	(p)
3. Piping	Other	T	D	N/A	N/A	(p, q, s)
4. Pumps	Other	T	D	N/A	N/A	(p, s)
5. Valves - flow control	Other	T	D	N/A	N/A	(p, q, s)
6. Valves - other	Other	T	D	N/A	N/A	(p, q, s)
7. Mechanical modules with safety function	Other	T,A	D	N/A	N/A	(p, q, s)
8. Pressure vessels	Other	T,A	D	N/A	N/A	(p)
9. Recombiners	Other	T	D	N/A	N/A	(p)

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XXXI Standby Gas Treatment System						
1. Filters	2	R	N/A	B	I	
2. Valves - ductwork	2	A,R,C	N/A	B	I	
3. Cable with safety function	2	R,A,C,X	N/A	B	I	
4. Fans and motors	2	R	N/A	B	I	
XXXII NI Chilled Water Systems						
1. Control Building	3	D,C,A	C	B	I	
2. Electrical switch gear	3	A	C	B	I	
3. Other buildings	Other	A,R,W	D	N/A	N/A	
XXXIII HPCS Service Water System						
This system is included under group/MPL XXXIV/P41.						
XXXIV Essential Service Water System						
1. Piping	2,3	O,A,C	B/C	B	I	(g)
2. Pumps	3	P	C	B	I	
3. Pump motors	3	P	N/A	B	I	

3.116

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XXXIV (Continued)						
4. Valves - isolation	2	C,A	B	B	I	
5. All valves that are not isolation valves	3	A,C,R,S	C	B	I	
6. Electrical modules with safety function	3	A	N/A	B	I	
7. Cable with safety function	3	A,O,P,X	N/A	B	I	
8. Nonessential portions	Other	A,C,R,W	D	N/A	N/A	
XXXV Closed Cooling Water System						
1. Piping and valves forming part of primary containment boundary	2	A,C,R	B	B	I	
2. All piping and valves not forming part of primary containment boundary	Other	A,C,D,R,W	D	N/A	N/A	
XXXVI Condensate and Demineralized Water Storage and Transfer						
1. Piping and valves forming part of the containment boundary	2	C	B	B	I	
2. Condensate storage tank	Other	O	D	N/A	N/A	(w)

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XLI Water Positive Seal System						
1. Piping	2	A,C	B	B	I	
2. Valves	2	A,C	B	B	I	
3. Tank	2	A	B	B	I	
XLII Air Positive Seal System						
1. Accumulator	2	A,R	B	B	I	
2. Piping	2	A	B	B	I	
3. Valves	2	A	B	B	I	
4. Compressor	2	A	B	B	I	
5. Compressor motors	2	A	B	B	I	
6. Electrical modules	2	A	B	B	I	
7. Cables	2	A	N/A	B	I	
XLIII Breathing Air System						
1. Piping and valves	Other	C,A	D	N/A	N/A	
2. Air purifier and other equipment	Other	C,A	D	N/A	N/A	

Table 3.2-1

EQUIPMENT CLASSIFICATION (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Comments
XLIV Plant Electrical Systems						
(Applicant to Supply)						
XLV Auxiliary AC Power System						
1. All components with safety function	2/3	A,C,X	N/A	B	I	(g)
XLVI Diesel Generator Systems						
1. Fuel oil storage and transfer system	3	S,O	C	B	I	(y)
2. Cooling water system	3	S	C	B	I	(y)
3. Starting air tanks receivers, piping from and including check valve and downstream piping and valves	3	S	C	B	I	(y)
4. Starting air compressor and motors	Other	S	N/A	N/A	N/A	
5. Lubrication system	3	S	C	B	I	(y)
6. Combustion air intake and exhaust system	3	S,O	C	B	I	

3.116

3.118

Table 3.2-3

SUMMARY OF MINIMUM SAFETY CLASS DESIGN REQUIREMENTS

Design Requirements	Safety Class 1	2	3	Other
Quality Group Classification[a]	A	B	C	D
Quality Assurance Requirement[b]	B	B	B	N/A
Seismic Category[c]	I	I	I/N/A	N/A
Electrical Classification[d]	1E	1E	1E	N/A

NOTES

a. Equipment containing radioactive material, water, or steam shall be constructed in accordance with the indicated quality group listed in Table 3.2-1, and the code class defined in Table 3.2-2. A quality group is not applicable (N/A) for equipment not containing radioactive material, water, or steam.

b. B - The equipment shall be constructed in accordance with the quality assurance requirements of 10CFR50 Appendix B as delineated in Chapter 17.

N/A - The equipment shall be constructed in accordance with the quality assurance requirements consistent with accepted practice for steam power generating stations.

c. I - The equipment of these safety classes shall be constructed in accordance with the seismic requirements for the safe shutdown earthquake as described in Section 3.7.

N/A - The seismic requirements for the safe shutdown earthquake are not applicable to the equipment of this classification.

Table 3.2-3

SUMMARY OF MINIMUM SAFETY CLASS DESIGN REQUIREMENTS (Continued)

NOTES (Continued)

d. All electrical equipment with Safety Class 1, 2 or 3 shall have an electrical classification of 1E. An electrical classification is not applicable (N/A) to equipment of Safety Class Other.

3.6.2.1.4.2 Piping in Containment Penetration Areas (Continued)

(b) The cumulative usage factor must be less than 0.1

(c) If the calculated maximum stress range of Equation 10 exceeds 2.4Sm, then the stress ranges calculated by both Equation 12 and 13 of NB-3653 do not exceed 2.4Sm. 3.127

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

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

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

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

3.6.2.1.4.2 Piping in Containment Penetration Areas (Continued)

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

(3) The piping runs are straight.

(4) Welded attachments for pipe supports or other purposes were avoided unless the detailed stress analyses or tests were performed to demonstrate compliance with the stress limits given in items (1) and (2). 3.128

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

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

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

determining and modifying resonant sections, or changing operating conditions. After corrective action is taken, additional testing shall be performed to determine if the vibrations have been sufficiently reduced to satisfy the acceptance criteria.

3.160

3.9.2.1.2.2 Preoperational Thermal Expansion and Dynamic Testing

Preoperational thermal expansion and dynamic testing is provided in Subsection 14.2.12.1.75.

3.9.2.1.3 Thermal Expansion Testing of Main Steam and Recirculation Piping]

A thermal expansion pre-operational and startup testing program performed through the use of potentiometer sensors has been established to verify that normal thermal movement occurs in the piping systems. The main purpose of this program is to ensure the following:

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

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

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

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

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

Limits of thermal expansion displacements are established prior to start of piping testing to which the actual measured displacements are compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with the predictions and is therefore acceptable. Two levels of limits of displacements would be established to check the systems as explained in Subsection 3.9.2.1.5.

4.2.1 Design Bases (Continued)

Acceptance Criterion II.A.3.(e) of SRP Section 4.2 describes analytical procedures for the determination of fuel assembly structural deformation. The GESSAR II fuel assembly structural analysis is described in Topical Report NEDE-21175-3-P. In this report, each major fuel assembly component part is shown to be functionally adequate to withstand the separate and combined peak loadings from the dynamic and LOCA blowdown events without experiencing structural failure.

4.2.2 Description and Design Drawings

See Appendix A, Subsection A.4.2.2 of Reference 1.

Acceptance Criterion II.B of SRP Section 4.2 lists design parameters and drainage to be included in the fuel system description. The GESSAR II fuel system description, given in Reference 1, does not include all of the design parameters listed in Acceptance Criterion II.B. However, sufficient information is given to provide a reasonably accurate representation of the GESSAR II fuel system, satisfying the intent of the SRP.

4.2.2.1 Control Rods

The control rods perform the duel function of power shaping and reactivity control. A design drawing of the control blade is seen in Figure 4.2-1 and 2. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods. Control rod displacement tends to counterbalance steam void effects at the top of the core and results in significant power flattening.

The control rod consists of a sheathed cruciform array of stainless steel tubes filled with boron-carbide powder. The control rods are 9.868 in. in maximum total span and are separated uniformly throughout the core on a nominal 12-in. pitch. Each control rod is surrounded by four fuel assemblies.

4.2.2.1 Control Rods (Continued)

The main structural member of a control is made of Type-304 and/or 316L stainless steel and consists of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, bottom casting, and center post are welded into a single skeletal structure.

4.2.2.2 Velocity Limiter (Continued)

The velocity limiter is in the form of two nearly mated, conical elements that act as a large clearance piston inside the control rod guide tube. The lower conical element is separated from the upper conical element by four radial spacers 90 degrees apart and is at a 15-degree angle relative to the upper conical element, with the peripheral separation less than the central separation.

The hydraulic drag forces on a control rod are proportional to approximately the square of the rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke, the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod drag during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction.

Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the conical elements and discharged through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 3.11 ft/sec.

4.2.3 Design Evaluation

See Appendix A, Subsection A.4.2.3 of Reference 1.

Acceptance Criterion II.C.3(a) of SRP Section 4.2 lists phenomenological models to be included in fuel system thermal calculations. The GESSAR II fuel thermal model does not include the use of approved gadolinia fuel properties. However, as discussed with the NRC staff, the General Electric Company does not license material properties for design analyses but, rather, maintains these analyses up-to-date. To fulfill the quality control obligations under 10CFR50, Appendix B, the latest property values are incorporated into design applications only after they are qualified in the design code. An improved fuel rod thermal-mechanical design code has recently been developed and qualified which includes the revised gadolinia fuel thermal conductivity relations. The results of the fuel centerline melting analysis using this improved fuel rod design code verify that gadolinia fuel melting is not expected to occur during normal steady-state operation or during the largest whole core anticipated operational transient.

The Reference 1 amendment incorporating the application of the above thermal-mechanical design code is currently under review by the Core Performance Branch. Since GESSAR II references the "latest approved revision" of Reference 1, this issue will be resolved when the Reference 1 amendment is approved.

Acceptance Criterion II.C.3.(d) of SRP Section 4.2 describes acceptance criteria for evaluation of fuel assembly structural response to externally applied forces.

An analysis has been performed (NEDE-21175-3) to show that the GESSAR II fuel meets structural requirements (including lift-off) similar to those of Appendix A of Section 4.2 of the SRP (NUREG-0800). That analysis is currently under review by the NRC staff. Because previous generic analytical methods presented in earlier versions of NEDE-21175 have been approved by the NRC staff (Letter from O. P. Parr (NRC), May 17, 1979) and because favorable sample results were also presented in Amendment 2 of NEDE-21175, the new GE analysis is expected to be approved.

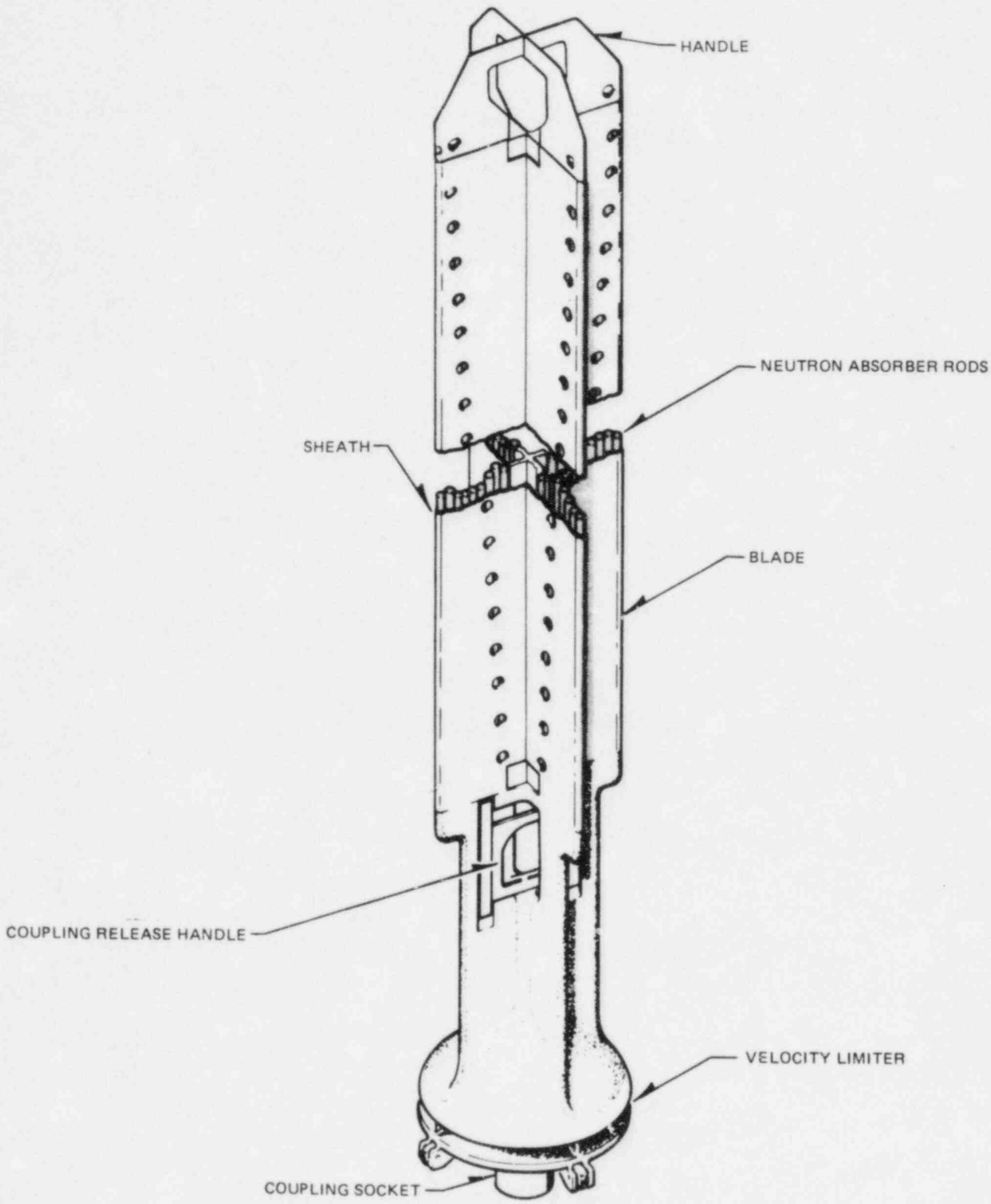


Figure 4.2-1. Control Rod Assembly

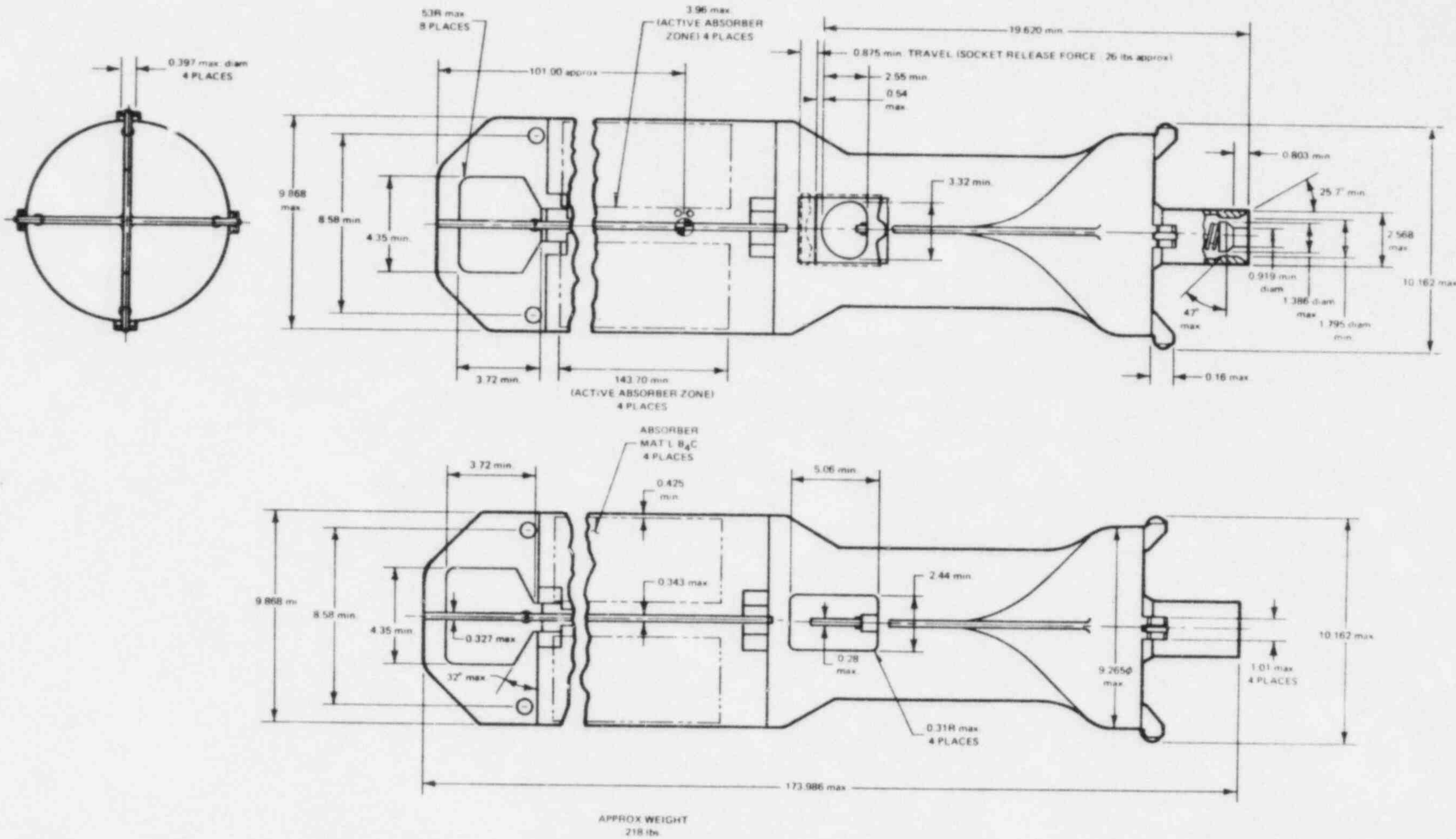


Figure 4.2-2. Control Rod Information Diagram

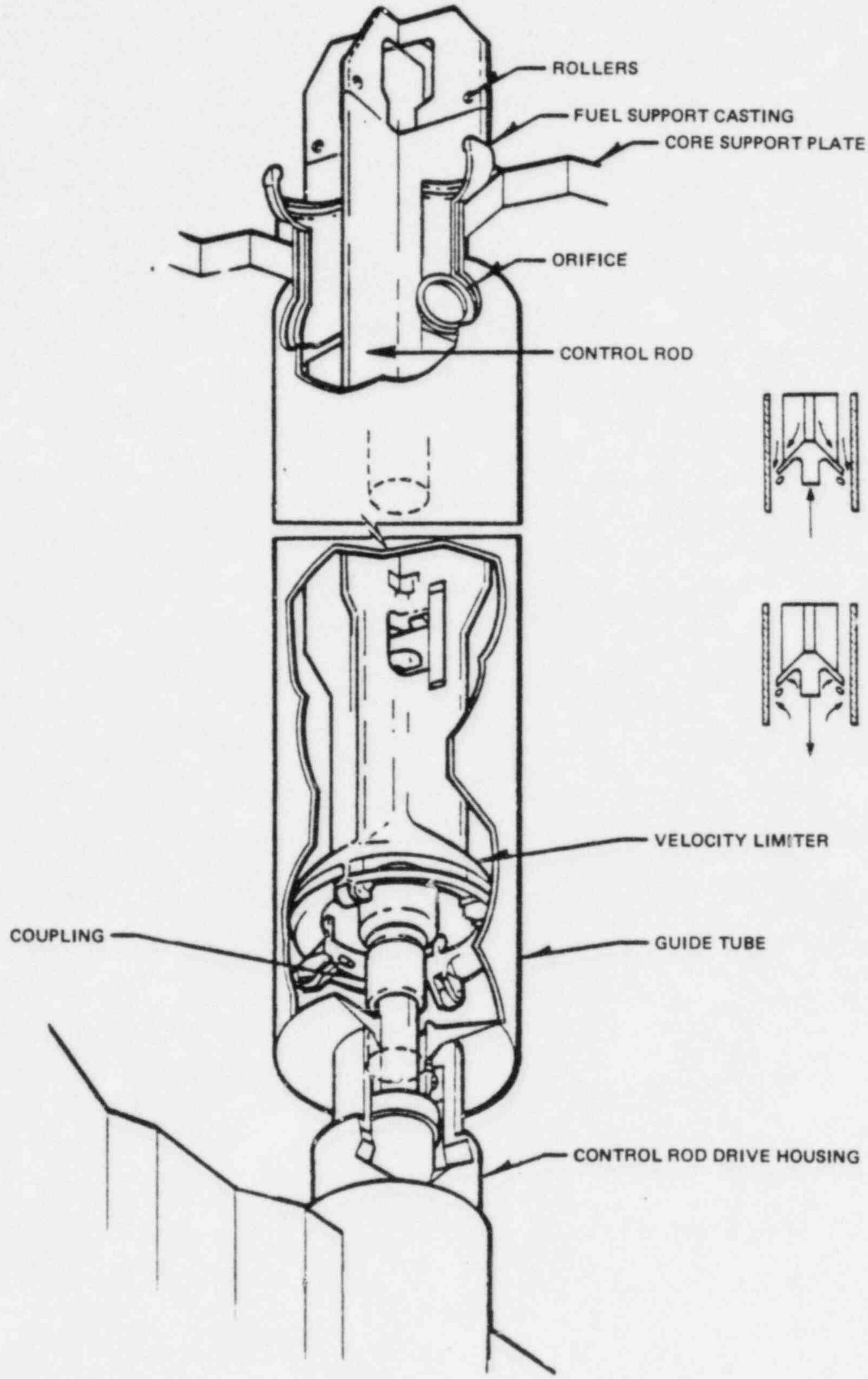
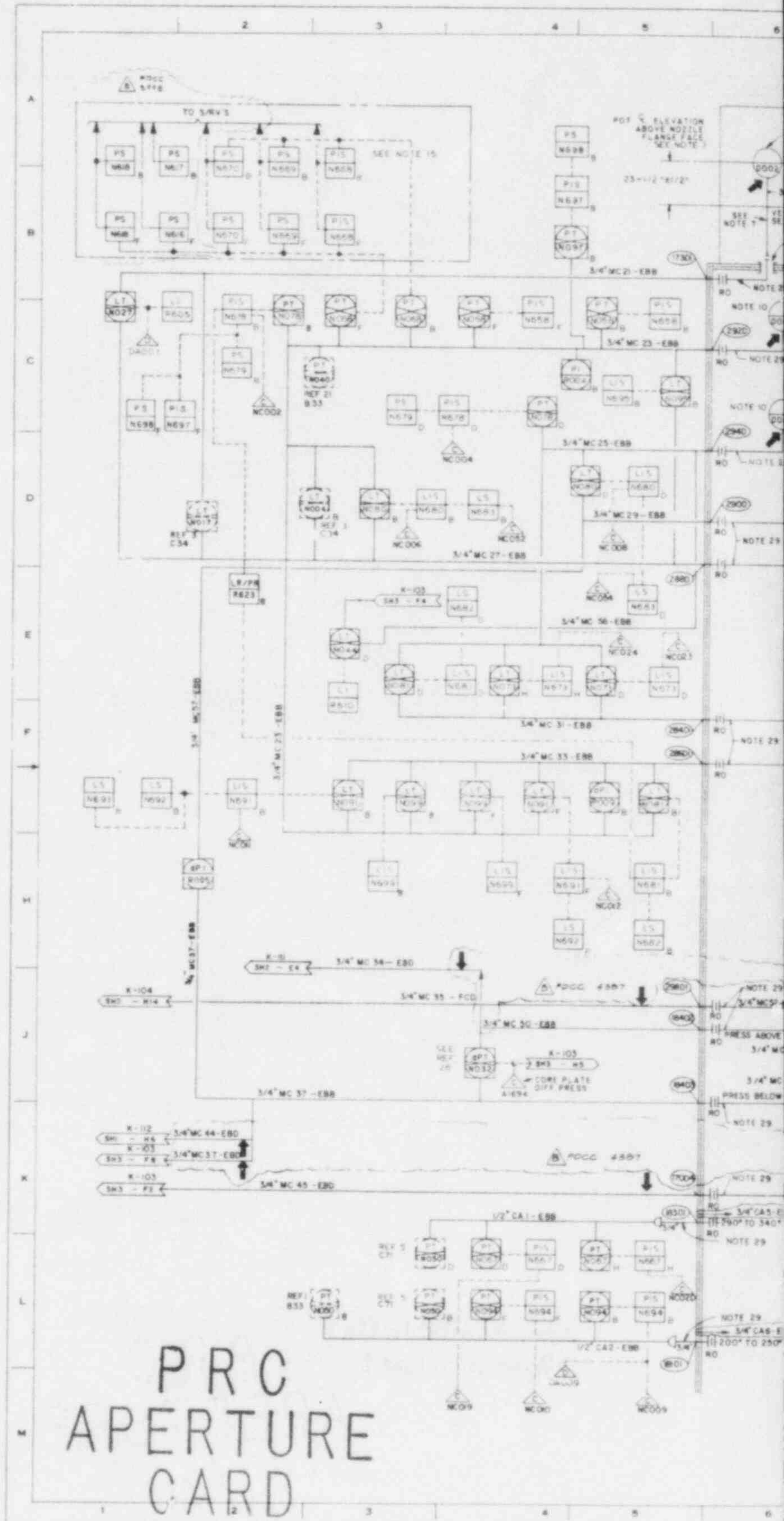


Figure 4.2-3. Control Rod Velocity Limiter



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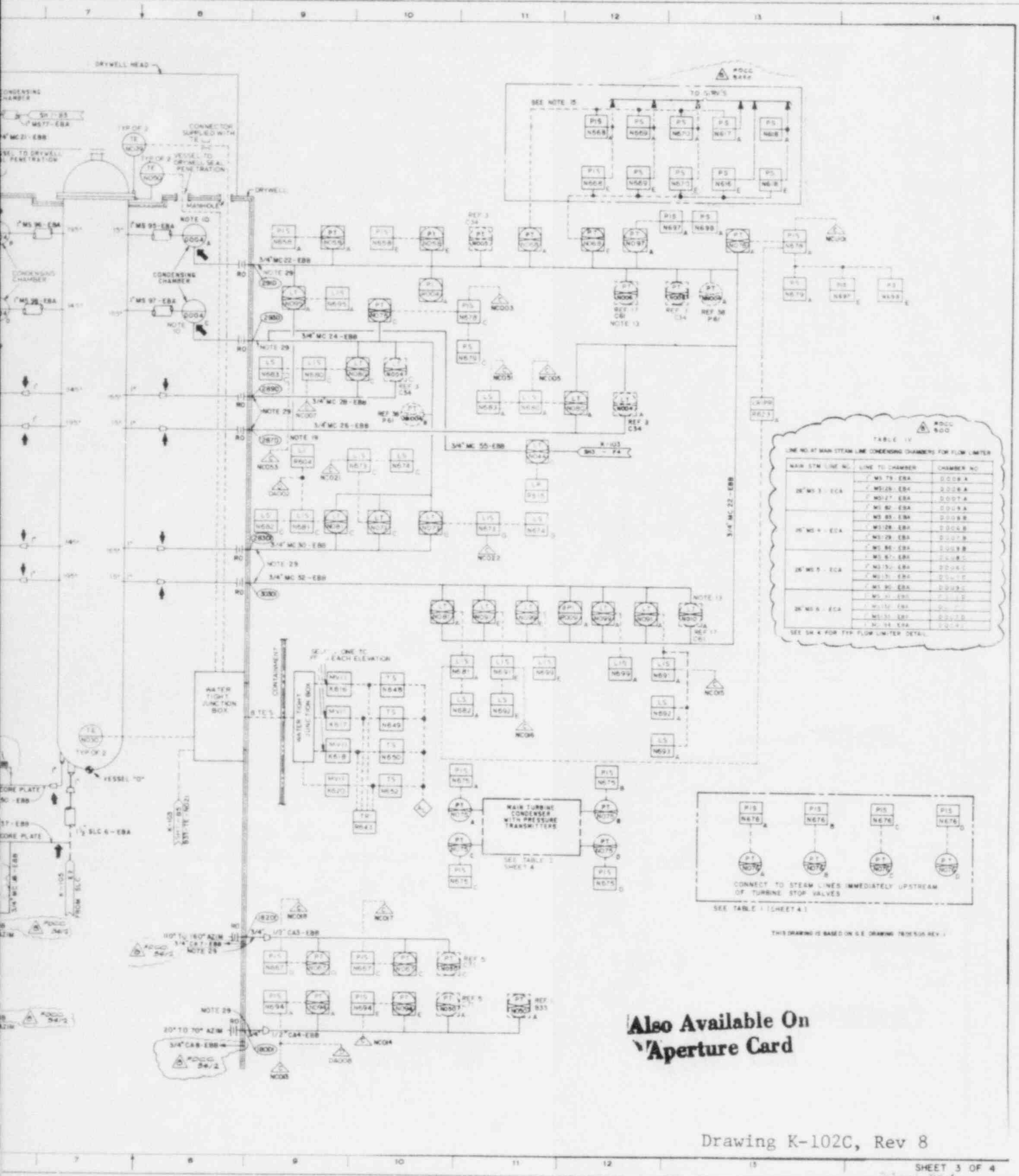

Also Available On Aperture Card

Drawing K-102C, Rev 8

Figure 5.1-3c. Nuclear Boiler Sys
P&I Flow Diagram

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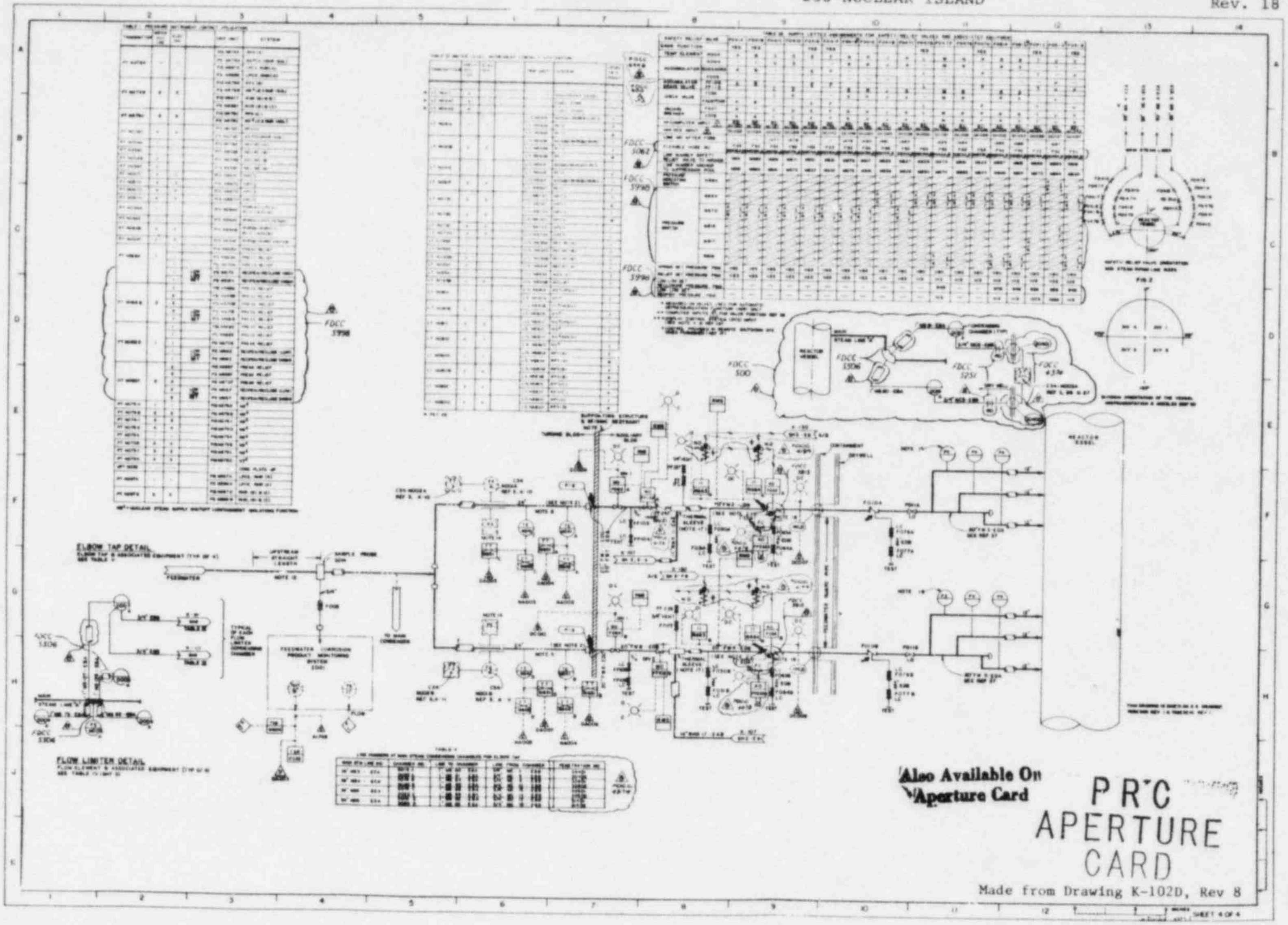

Figure 5.1-3d. Nuclear Boiler Sys
P&I Flow Diagram

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CONTENTS (Continued)

SECTION 5.2

ILLUSTRATIONS

5.2.2.2.2.1 Operating Conditions

(1) operating power = 3729 MWt (104.2% of nuclear boiler rated power);

(2) vessel dome pressure $\leq$ 1045 psig; and

(3) steamflow = 16.71×10^6 lb/hr (105% of nuclear boiler rated steamflow).

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

See Appendix A, Subsection A.5.2.2.2.2.2 of Reference 3.

5.2.2.2.2.3 Safety/Relief Valve Transient Analysis Specification

(1) Simulated valve groups:

power-actuated relief mode - 4 groups
spring-action safety mode - 5 groups

(2) opening pressure setpoint (maximum safety limit):

power-actuated relief mode -	group 1	1125 psig
	group 2	1135 psig
	group 3	1145 psig
	group 4	1155 psig
spring-action safety mode -	group 1	1175 psig
	group 2	1185 psig
	group 3	1195 psig
	group 4	1205 psig
	group 5	1215 psig

(3) reclosure pressure setpoint (% of opening setpoint) both modes:

maximum safety limit (used in analysis)	98
minimum operational limit	89

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically the assumed

5.2.3.2.2 BWR Chemistry of Reactor Coolant (Continued)

Several investigations have shown that in neutral solutions some oxygen is required to cause stress/corrosion cracking of stainless steel. In the absence of oxygen no cracking occurs. One investigation of the chloride oxygen relationship (Reference 2) showed that, at high chloride concentration, little oxygen is required to cause stress/corrosion cracking of stainless steel, and, at high oxygen concentration, little chloride is required to cause cracking. These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and, therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

The water quality requirements are further supported by General Electric stress corrosion test data summarized as follows.

(1) Type 304 stainless steel specimens were exposed in a flowing loop operating at 537°F. The water contained 1.5 ppm chloride and 1.2 ppm oxygen at pH 7. Test specimens were bent-beam strips stressed over their yield strength. After 2100 hours exposure, no cracking or failures occurred.

(2) Welded Type-304 stainless steel specimens were exposed in a refreshed autoclave operating at 550°F. The water contained 0.5 ppm chloride and 1.5 ppm oxygen at pH 7. Uniaxial tensile test specimens were stressed at 125% of their 550°F yield strength. No cracking or failures occurred at 15,000 hours exposure.

When conductivity is in its normal range, pH, chloride, and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, chloride

5.2.3.2.2 BWR Chemistry of Reactor Coolant (Continued)

measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, where no additives are used and where near neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the operator with a warning to investigate and remedy the condition before reactor water limits are reached. Methods available to the operator for correcting the off-standard condition include operation of the reactor water cleanup system, reducing the input of impurities, and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature-dependent corrosion rates and provide time for the cleanup system to re-establish the purity of the reactor coolant.

The following is a summary and description of BWR water chemistry for various plant conditions.

(1) Normal Plant Operation

The BWR system water chemistry is described by following the system cycle (Figure 5.2-13 and Table 5.2-6).

For normal operation starting with the condenser/ hotwell, condensate water is processed through a condensate treatment system. This process consists of filtration and demineralization resulting in effluent water quality represented in Table 5.2-6.

Figure 5.2-5 Deleted

Figure 5.2-6 Deleted

19.3.9.24 QUESTION/RESPONSE 9.24 (410.24)

QUESTION 9.24

Verify that the information provided in Section 9.1.3 of your FSAR is based on the new high density spent fuel pool storage capacity. Provide additional information regarding the spent fuel decay heat load for the maximum, normal and abnormal heat loads as discussed in Items 1.d and 1.h of the review procedures in Section 9.1.3 of the SRP. (9.1.3)

RESPONSE 9.24

Paragraph 9.1.3.1.2(4) for the Power Generation Design Basis states that the heat load is the sum of (1) the 37 percent core batch just removed at the last 18-month equilibrium fuel cycle, with 4-year exposure, and (2) the 37 percent core batch from the previous refueling outage. Therefore, the heat load is a function of two 37 percent batches, which means that the entire heat capacity of the fuel storage pool does not enter the design. The fresh core supplies about 90 percent of the heat load and the aged core fraction supplies the other 10 percent of the design load. The density of the fuel racks would change the heat load calculation only if all of the potential batches stored within the pool were used toward the total design value. Even under these conditions, the design value would be only slightly affected.

Paragraph 9.1.3.2 describes that the above design core load for heat capacity is based upon maintaining 125°F in the pool. This is the system design maximum load and temperature combination. However, if conditions exist as described in Paragraph 9.1.3.3, wherein up to a full reactor core is

19.3.9.24 QUESTION/RESPONSE 9.24 (410.24) (Continued)

placed into the pool, instead of the 37 percent batch, the pool may go to 150°F. But adding the RHR cooling capacity will keep the temperature at a maximum of 125°F.

Item 1.h(ii) states that the normal maximum spent fuel heat load is set as one refueling load at equilibrium conditions after 150 hours of decay, with one refueling load after 1 year of decay, and 140°F pool temperature. The GESSAR II design basis is more conservative in that the refueling load is assumed at 112 hours of decay and the maximum pool temperature is set at 125°F. The shorter fresh batch decay time adds to the total heat load sum of the two batches.

Item 1.h(iii) states that the Spent Fuel Pool Cooling System will have capacity for a full core at equilibrium and one refueling load, at 36 days, for a total of 1-1/3 core fraction. Item 1.h(iv) further adds 1/3 core for pool capacity over 1-1/3 batches. If RHR cooling capacity is included in the Spent Fuel Pool Cooling System, then the cooling capacity is more than adequate to meet these criteria. The two fuel pool heat exchangers cover only the normal maximum, reflecting item 1.h(ii), and the RHR covers any additional load while the reactor is open.

The failure of one of the two active pumps or heat exchangers will reduce the capacity of the Spent Fuel Pool Cooling System. The amount of cooling required by the pool is a function both of the amount of core placed in the pool at the last refueling and of the time for decay of that core fraction. If the decay heat exceeds the removal capacity, the RHR System shall be employed to maintain fuel pool temperature. During this time of RHR use, the reactor will not be restarted following the refueling shutdown that placed the spent fuel in the pool. When the decay heat is

19.3.9.24 QUESTION/RESPONSE 9.24 (410.24) (Continued)

less than the capacity of the Fuel Pool Cooling System, either by reduction due to time or by increased system cooling capacity, the use of the RHR will be stopped and the reactor startup will be allowed.