UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of)

General Electric Company) Docket No. STN 50-447

Standard Plant)

AMENDMENT NO. 8 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. 8 to the 238 Nuclear Island General Electric Standard Safety Analysis Report (GESSAR II).

Amendment No. 8 consists of two parts, a non-proprietary portion and a portion considered by the General Electric Company to be proprietary. The pages considered to be proprietary are so marked and are transmitted under separate cover.

Amendment No. 8 further amends GESSAR II by furnishing responses to the following questions requested by Enclosure 1 to the Commission's GESSAR II acceptance review letter dated December 3, 1981:

1. Geotechnical (Questions 241.1, 241.3-241.9, 241.11-241.14, 241.17-241.21, 241.23 and 241.24)

2. Radiation Sources (Question 471.2)

3. Failure Modes and Effects Analysis (Questions 280.1, 410.3, 410.8, 420.3, 420.7 and 480.1)

Amendment No. 8 also amends GESSAR II by:

1. Providing a statement in Appendix 3B (Containment Loads) that commits the Applicant to address the NRC draft acceptance criteria for LOCA-related Mark III containment pool dynamic loads.

Respectfully submitted,

General Electric

by: R Villa
Rudolph Villa, Manager
BWR Standardization

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

On this 18 day of October in the year 1982, before me, Karen S. Vogelhuber, Notary Public, personally appeared Rudolph Villa, personally proved to me on the basis of satisfactory 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

I10086-1-2

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of)

General Electric Company) Docket No. STN 50-447

Standard Plant)

AMENDMENT NO. 8 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. 8 to the 238 Nuclear Island General Electric Standard Safety Analysis Report (GESSAR II).

Amendment No. 8 consists of two parts, a non-proprietary portion and a portion considered by the General Electric Company to be proprietary. The pages considered to be proprietary are so marked and are transmitted under separate cover.

Amendment No. 8 further amends GESSAR II by furnishing responses to the following questions requested by Enclosure 1 to the Commission's GESSAR II acceptance review letter dated December 3, 1981:

1. Geotechnical (Questions 241.1, 241.3-241.9, 241.11-241.14, 241.17-241.21, 241.23 and 241.24)

2. Radiation Sources (Question 471.2)

3. Failure Modes and Effects Analysis (Questions 280.1, 410.3, 410.8, 420.3, 420.7 and 480.1)

Amendment No. 8 also amends GESSAR II by:

1. Providing a statement in Appendix 3B (Containment Loads) that commits the Applicant to address the NRC draft acceptance criteria for LOCA-related Mark III containment pool dynamic loads.

Respectfully submitted,

General Electric

by: s/R. Villa
Rudolph Villa, Manager
BWR Standardization

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

On this 18 day of October in the year 1982, before me, Karen S. Vogelhuber, Notary Public, personally appeared Rudolph Villa, personally proved to me on the basis of satisfactory evidence to be the person whose name is subscribed to this instrument, and acknowledged that he executed it.

s/K. S. Vogelhuber
NOTARY PUBLIC, STATE OF CALIFORNIA
Santa Clara County
My Commission Expires
December 21, 1984
175 Curtner Avenue
San Jose, CA 95125

GESSAR II
238 NUCLEAR ISLAND

INSTRUCTIONS FOR FILING AMENDMENT NO. 8

Appendix 3I pages which were omitted from Amendment 6 are included here.

Remove and insert the pages listed below.

Remove	Insert
Chapter 1	
1.1-1	1.1-1
Appendix 1A	
1A.16-1/1.16-2, 1A.17-1/1A.17-2, and 1A.30-3/1A.30-4	1A.16-1/1A.16-2, 1A.17-1/1A.17-2, and 1A.30-1/1A.30-4
Chapter 3	
3.7-ix/3.7-x, 3.7-6, 3.7-139, 3.8-136, 3.8-141, 3.8-142, 3.8-154, 3.8-291, and 3.8-292	3.7-ix/3.7-x, 3.7-6, 3.7-139, 3.9-90a, 3.8-136, 3.8-136a, 3.8-141, 3.8-142, 3.8-142a, 3.8-154, 3.8-291, and 3.8-292
Appendix 3B	
3B-1 and 3B-2	3B-1, 3B-2, and 3B-2a
Appendix 3I	
3I.1-1 through 3I.1-6, 3I.2-1 through 3I.2-4, 3I.5-1 through 3I.5-6, 3I.6-1 through 3I.6-6, 3I.7-1 through 3I.7-4, and 3I.8-1 through 3I.8-36	3I.1-2, 3I.2-3, 3I.5-2, 3I.8-11, 3I.8-21, and 3I.8-24
Chapter 6	
6.2-76 and 6.2-78	6.2-76 and 6.2-78
Chapter 7	
7.2-141 and 7.3-168	7.2-41 and 7.3-168
Chapter 9	
9.1-63a and 9.3-39	9.1-63a, 9.3-39, and 9.5-10b

Remove	Insert
Chapter 12	
12.2-iii, 12.2-iv, 12.2-10, 12.2-26, 12.2-27, 12.2-65, 12.2-67, and 12.2-75/12.7-76	12.2-iii, 12.2-iv, 12.2-10, 12.2-26, 12.2-27, 12.2-65, 12.2-67, and 12.2-75/12.2-76
Appendix 15C	
Appendix 15C Title Page	New Appendix 15C
Chapter 18	
Chapter 18 Title Page	Chapter 18 Title Page
Chapter 19	
19.1.2-1/19.1.2-2, 19.1.3-1, 19.1.6-1/19.1.6-2, 19.1.7-1/19.1.7-2, 19.1.9-1/19.1.9-2, 19.1.12-1/ 19.1.12-2, 19.3.2.1-1/19.3.2.1-2, 19.3.3.12-1/19.3.3.12-2, 19.3.3.13-1/19.3.3.13-2, 19.3.3.14-1/19.3.3.14-2, 19.3.3.15-1/19.3.3.15-2, 19.3.3.16-1/19.3.3.16-2, 19.3.3.17-1/19.3.3.17-2, 19.3.3.18-1/19.3.3.18-2, 19.3.3.20-1/19.3.3.20-2, 19.3.3.21-1/19.3.3.21-2, 19.3.3.22-1/19.3.3.22-2, 19.3.3.23-1/19.3.3.23-2, 19.3.3.26-1/19.3.3.26-2, 19.3.3.27-1/19.3.3.27-2, 19.3.3.28-1/19.3.3.28-2, 19.3.3.29-1/19.3.3.29-2, 19.3.3.30-1/19.3.3.30-2, 19.3.6.1-1/19.3.6.1-2, 19.3.7.3-1/19.3.7.3-2, 19.3.9.1-1/19.3.9.1-2, 19.3.9.4-1/19.3.9.4-2, 19.3.9.9-1/19.3.9.9-2, 19.3.7.7-1/19.3.7.7-2, and 19.3.12.2-1/19.3.12.2-2	19.1.2-1/19.1.2-2, 19.1.3-1, 19.1.6-1/19.1.6-2, 19.1.7-1/19.1.7-2, 19.1.9-1/19.1.9-2, 19.1.12-1/ 19.1.12-2, 19.3.2.1-1/19.3.2.1-2, 19.3.3.12-1/19.3.3.12-2, 19.3.3.13-1/19.3.3.13-2, 19.3.3.14-1/19.3.3.14-2, 19.3.3.15-1/19.3.3.15-2, 19.3.3.16-1/19.3.3.16-2, 19.3.3.17-1/19.3.3.17-2, 19.3.3.18-1/19.3.3.18-2, 19.3.3.20-1/19.3.3.20-2, 19.3.3.21-1/19.3.3.21-2, 19.3.3.22-1/19.3.3.22-2, 19.3.3.23-1/19.3.3.23-2, 19.3.3.26-1/19.3.3.26-2, 19.3.3.27-1/19.3.3.27-2, 19.3.3.28-1/19.3.3.28-2, 19.3.3.29-1/19.3.3.29-2, 19.3.3.30-1/19.3.3.30-2, 19.3.6.1-1/19.3.6.1-2, 19.3.7.3-1/19.3.7.3-2, 19.3.9.1-1/19.3.9.1-2, 19.3.9.4-1/19.3.9.4-2, 19.3.9.9-1/19.3.9.9-2, 19.3.7.7-1/19.3.7.7-2, and 19.3.12.2-1/19.3.12.2-2

1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

The General Electric Standard Safety Analysis Report, GESSAR II, is written in accordance with Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 3, November 1978). For consistency with NUREG-0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reports, Revision 0, July 1981), GESSAR II includes Section 15.8 which addresses anticipated transients without scram and Chapter 18 which addresses human factors. Finally, GESSAR II contains Chapter 19 to serve as a question and response guide.

The GESSAR II response to TMI-related matters is contained in Appendix 1A. The assessment of unresolved safety issues is given in Appendix 1B. Appendix 1C gives the GESSAR II response to the NRC additional guidance provided in the Commissions' GESSAR II acceptance review letter, dated December 9, 1981. Appendix 1D provides an assessment of GESSAR II against Regulatory Guide 1.97, Revision 2.

1.1.1 Type of License Required

This General Electric Standard Safety Analysis Report (GESSAR) is submitted in support of the application for a construction permit and facility operating license for the Nuclear Island portion of a nuclear powered electric generating plant. The Nuclear Island (sometimes referred to as Reactor Island) consists of all buildings which are dedicated exclusively or primarily to housing systems and equipment related to the nuclear system. Under the concept presented herein, there are seven such buildings that comprise the Nuclear Island. These are:

(1) Reactor Building (including shield building and containment);

(2) Fuel Building;

(3) Auxiliary Building;

(4) Diesel Generator Buildings;

(5) Control Building; and

(6) Radwaste Building.

The only major system related to the nuclear system that is not housed in one of the seven buildings is the Offgas System which is more appropriately housed in the turbine building since it is physically associated with the condenser air ejectors.

1A.16 CONTROL ROOM DESIGN REVIEW (NUREG-0737 Item I.D.1)

NRC Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Response

The response to this requirement is provided in Chapter 18.

1A.17 PLANT SAFETY PARAMETER DISPLAY CONSOLE (NUREG-0737 Item I.D.2)

NRC Position

In accordance with Task Action Plan 1.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Response

The response to this requirement is provided in Chapter 18.

1A.30 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION (NUREG-0737 Item II.F.1) (Cont'd)

Response

The response to this requirement is provided in Appendix 1D.

ILLUSTRATIONS (Continued)

3.7.1.4 Supporting Media for Seismic Category I Structures (Continued)

(2) Auxiliary Building - 34 ft, 0 in.

(3) Fuel Building - 35 ft, 2 in.

(4) Control Building - 10 ft, 2 in.

(5) Radwaste Building (substructure only is Seismic Category I) - 41 ft, 0 in.

(6) Diesel Generator Buildings - 7 ft, 4 in.

All of the above buildings have independent foundations. In all cases the maximum value of embedment is used for the dynamic analysis to determine seismic soil-structure interaction effects. The foundation support materials withstand the pressures imposed by appropriate loading combinations without failure. The total structural height of each building is described in Subsection 3.8.2 through 3.8.4. For details of the structural foundations refer to Subsection 3.8.5. The Nuclear Island is designed for range of soil conditions given in Appendix 3A.

3.7.1.4.1 Soil-Structure Interaction

When a structure is supported on a flexible foundation, the soil-structure interaction is taken into account by coupling the structural model with the soil medium. The base mat is considered to be rigid. A finite-element representation is used for the following supporting medium conditions:

(1) rock foundations;

3.7.1.4.1 Soil-Structure Interaction (Continued)

(2) deep soil with uniform dynamic properties where the embedment is less than 15% of the smaller horizontal dimensions of the foundation mat; and

(3) all other supporting medium conditions.

3.7.1.4.2 Finite Element Representation

The damping ratio curve of the foundation soil included in the finite-element analysis is given in Subsection 3A.2.2 of 3.27
Appendix 3A. Radiation damping is treated by using energy absorbing boundaries to ensure that there is negligible reflection of energy from this boundary back into the structure.

A comprehensive program consisting of 15 cases was used in the Reactor Building finite-element soil-structure interaction study. The cases cover a wide range of site conditions with lower average and upper bound soil moduli. Two of the 15 cases are in the vertical direction.

Detailed methodology and results of the soil-structure interaction analysis are provided in Appendix 3A.

3.7.2 <u>Seismic System Analysis</u>

This subsection applies only to the design of Seismic Category I structures and the reactor pressure vessel (RPV). Subsection 3.7.3 applies to all Seismic Category I piping systems and equipment.

3.7.2.1 Seismic Analysis Methods

Analysis of Seismic Category I structures and the RPV is accomplished using the response spectrum or time-history approach.

Figure 3.7-21

Deleted

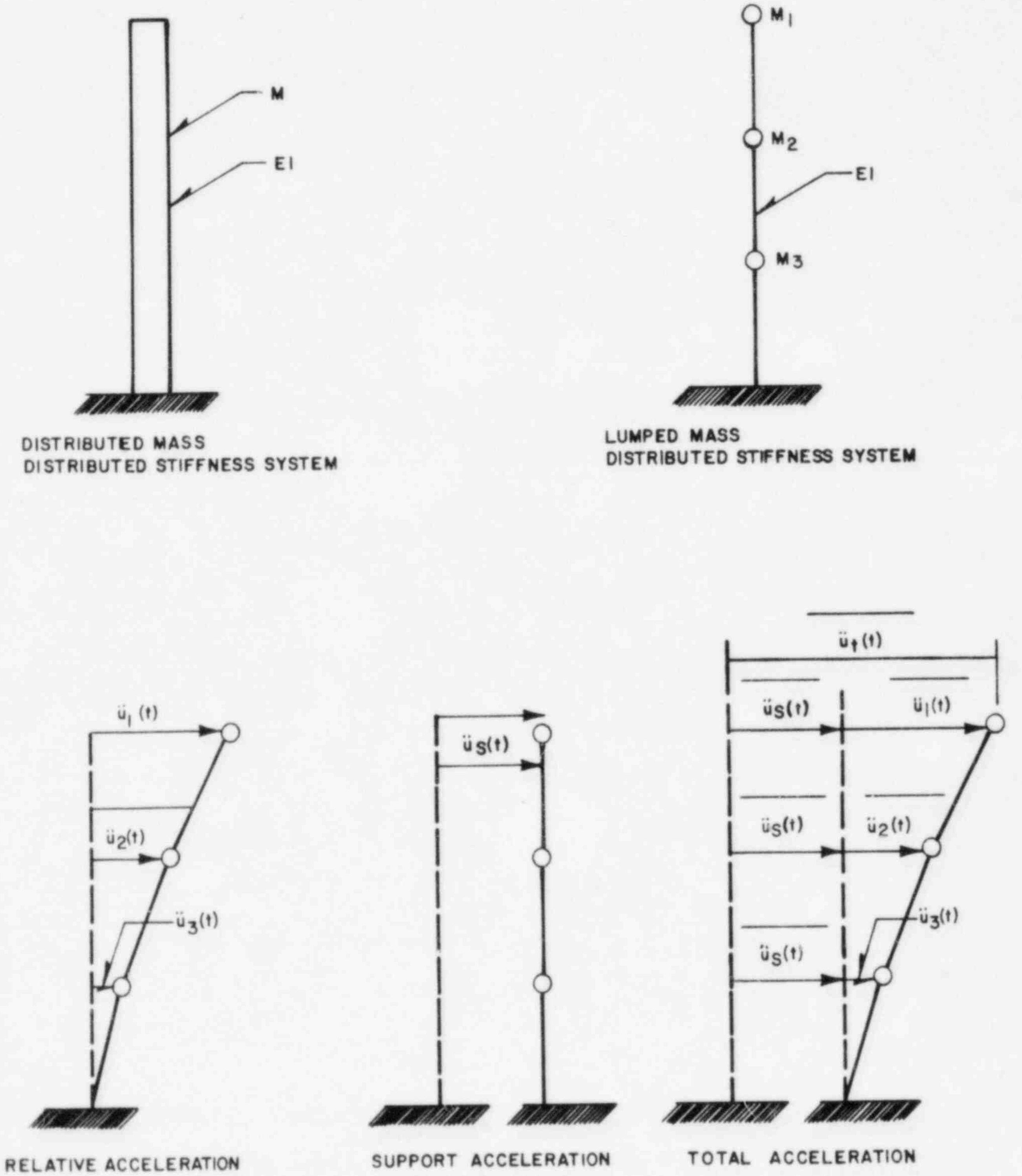


Figure 3.7-22. Seismic System Analytical Model

3.8.4 Other Seismic Category I Structures (Continued)

The Nuclear Island does not contain seismic Category I pipelines buried in soil. Refer to Section 1.9 for pipelines that may be buried in soil; these pipelines are the responsibility of the Applicant. 3.16

3.8.5.1.6 Diesel Generator Buildings Foundations (Continued)

below grade under trenches. The building is 109 ft, 0 in., by 80 ft, 6 in. The mat is physically separated from the Auxiliary Building by a 3-inch gap filled with a compressible material to minimize seismic interaction. The Diesel Generator Building, Division 2 and 3, foundation details are shown in Figures 3.8-69 through 3.8-72.

3.8.5.2 Applicable Codes, Standards and Specifications

3.8.5.2.1 Reactor Building Foundation

See Subsection 3.8.3.2.1 except that item (3), ASME Code Subsection NE, is omitted.

3.8.5.2.2 Auxiliary, Fuel, Control, Radwaste, and Diesel Generator Buildings Foundations

The applicable regulations, codes, standards, and specifications are discussed in Subsection 3.8.4.2 for these structures.

3.8.5.3 Loads and Load Combinations

3.8.5.3.1 Reactor Building Foundation

See Subsection 3.8.3.3.1 except that reference to 500 cycles of temperature variation in Subsection 3.8.3.3.1.2 is omitted and Load B is added in all load combinations except test where

B = uplift force due to displacement of ground water by the structure. The normal water level is at a level two feet below grade. The design basis flood elevation is one foot below the plant finished grade.

3.8.5.3.2 Auxiliary, Fuel, Control, Radwaste, and Diesel Generator Buildings Foundations

The foundation loads and load combinations for these structures are discussed in Subsection 3.8.4.3.

3.8.5.4 Design and Analysis Procedures

The soil and structural settlements have been calculated at various points of each building. These values are included in the piping design specification for each individual piping system. In each specification, there is a table listing load combination requirements and stress limits. The differential settlement has been listed as a loading condition. The piping systems are designed for this settlement. The acceptance criteria are specified in the form of stress limits. They are in accordance with the appropriate sections of the ASME Code which is identified in the piping specifications. 3.17

3.8.5.4.1 Reactor Building Foundation

The design of the Reactor Building foundation is concerned primarily with determining shear and moments in the reinforced concrete and determining the interaction of the substructure with the underlying foundation. For a reactor building foundation supported on soil or rock, the pertinent aspects in the design are to maintain the bearing pressures within allowable limits, particularly due to overturning forces, and to ensure that there is adequate frictional force to prevent sliding of the structure when subjected to lateral loads.

The design loads considered for analysis of the base slab foundation are the worst resulting forces from superstructures due to static and dynamic load combinations and such loads directly applied on the base slab as dead, live, seismic, hydrostatic, internal pressure, and temperature loads. The post-LOCA flooding condition has

3.8.5.4.1 Reactor Building Foundation (Continued)

been considered in conjunction with OBE seismic loads as a design in the factored load category.

Considering all design loads and subgrade stiffness, the finite-element computer program, NASTRAN, is employed for analysis of the Reactor Building base slab to obtain the design forces in the base slab and the stresses in the subgrade. The model is shown in Figure 3.8-73. The summary of design is presented in Figure 3.8-74.

The subgrade stiffness used in NASTRAN has been represented by spring constants. The spring constant represents a linear relation between applied load and displacement of the foundation which implies a linear stress-strain relation for the soil. Formulas for circular footings on the surface of the elastic half-space were used to compute the spring constants. The formulas were taken from "Vibrations of Soils and Foundations" by F. E. Rickart, Jr., J. R. Hall, and R. D. Woods, Table 10-13. The total spring constants were then distributed to each node in proportion to its tributary area. The individual springs were represented in the NASTRAN model. 3.18

In the foundation analysis, the seismic forces for various soil conditions have been enveloped and conservatively applied to the foundation.

As for the effects of different stiffnesses for the various soil conditions, the Applicant will show that for this particular site and subgrade stiffness, the foundation design is acceptable.

3.8.6.2 Soil Properties

The relationship between foundation pressure and soil properties is the critical factor in building stability, foundation design, and relative displacement between adjacent buildings. Supporting soil requirements of the Nuclear Island structures are provided in this subsection.

The maximum soil bearing pressures are calculated by assuming (1) the soil has zero tensile strength, and (2) the soil pressure in the soil bearing area varies linearly with distance from the point of minimum compressive stress to maximum compressive stress for the case of no foundation uplift and from the point of zero to maximum compressive stress. The maximum static soil bearing is calculated by summing the pressure due to the dead weight of the structure and the foundation mat. The maximum dynamic soil bearing pressure is similarly calculated but includes the square root of the sum of the squares of the maximum pressures due to earthquake motions in each of the three perpendicular directions. The maximum soil bearing pressures of the Nuclear Island structures are given in Table 3.8-10.

The ultimate and residual soil settlements and soil settlement profile are given in Table 3.8-11. Refer to Figure 3.8-88 for orientation and location of points.

The ultimate and residual soil settlements were calculated for the worst soil marginally suitable for a power plant. (See Table 3.8-11 for definitions.) The soil properties used were a shear wave velocity of 1000 fps, subgrade modulus of 150 lb/in.^{3}, initial void ratio 0.8, compression index 0.1, medium to strong clay. A 300 ft layer over the bedrock was assumed. We consider the calculations to represent an upper bound, applicable to a range of site conditions. 3.21

3.8.6.2 Soil Properties (Continued)

The orientation and location of settlement points are provided in Figure 3.8-89. The purpose of the points being shown in Figure 3.8-89 is to show differential settlement between points and/or buildings. 3.21

See Subsection 3.8.5.4 for interpretation of results and limiting criteria.

The maximum base rotational displacement for the Nuclear Island structures is given in Table 3.8-12. Refer to Figure 3.8-89 for orientation of X and Y axis. The calculation procedures are described in Subsection 3.7.2.14.

The maximum lateral earth pressure loads are given in Figure 3.8-90 for the following conditions:

(1) normal operation;
(2) construction condition;
(3) normal operation plus design basis flood; and
(4) normal operation plus OBE.

The required soil properties have been determined in accordance with Appendix 3A.

It is the responsibility of the Applicant's soils consulting engineer to verify that the nuclear island foundation soil meets the soil requirements of Appendix 3A, Subsection 3.8.6, and Section 3.7 to ensure that no slip failure occurs in the various 3.22
soils and soils profiles and that displacements between adjacent buildings do not exceed the design bases presented in Table 3.8-11. Actual Applicant site-unique soils report soil properties and soil profiles shall be compared with the soil properties used in Appendix 3A and the eight requirements of Appendix 3A to assure compliance to fall within the envelope for the

3.8.6.2 Soil Properties (Continued)

GESSAR II range of soil conditions. The maximum Applicant site-unique actual soil bearing pressures of the nuclear island structures shall be calculated and compared to Table 3.8-10. Any larger soil pressures shall be justified. 3.22

3.8.6.3 Loads for Potential Hazards in the Plant Vicinity

The design of the Nuclear Island structures assumed no loads resulting from man-made hazards and accidents such as potential explosions and associated missiles in the vicinity of the plants potential aircraft impacts, etc. If such loads are present, a case by case analysis must be made.

Table 3.8-10

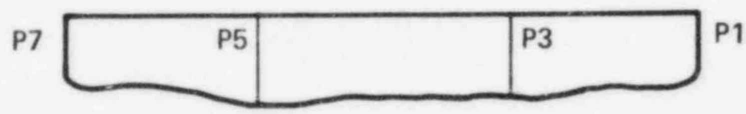
MAXIMUM SOIL BEARING PRESSURE

Building	Calculated Maximum Soil Pressure $\frac{kip}{sq\ ft}$ Static	Dynamic
Reactor Building	11.9	119.0
Auxiliary Building	5.6	47.2
Fuel Building	5.2	45.1
Control Building	2.8	29.6
Radwaste Building	4.2	27.7
Diesel Generator Building, Division 1	2.8	45.7
Diesel Generator Building Divisions 2 and 3	2.7	37.4

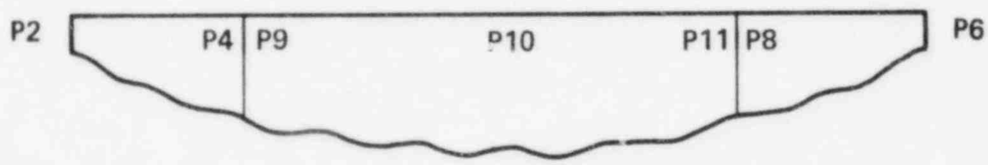
Table 3.8-11

VALUES OF ULTIMATE AND RESIDUAL SOIL SETTLEMENTS

3.21



Soil Settlement Profile Through Section A-A



Soil Settlement Profile Through Section G-G

Point[1]	Ultimate Settlement (inches)	Design[2,3] Residual Settlement, Residual (inches)
1,3,5,7	3.46	2.0
2,6	4.82	2.5
4,8	7.33	4.0
9,11	8.07	4.5
10	10.73	5.5

NOTES

1. This table is to be used in conjunction with Figure 3.8-89.

2. The residual settlement is that portion of the total ultimate settlement which will occur after the start-up. These are the values to be used to compute design differential settlements.

3. For any arbitrary span the settlement can be scaled down linearly between nearest specified points.

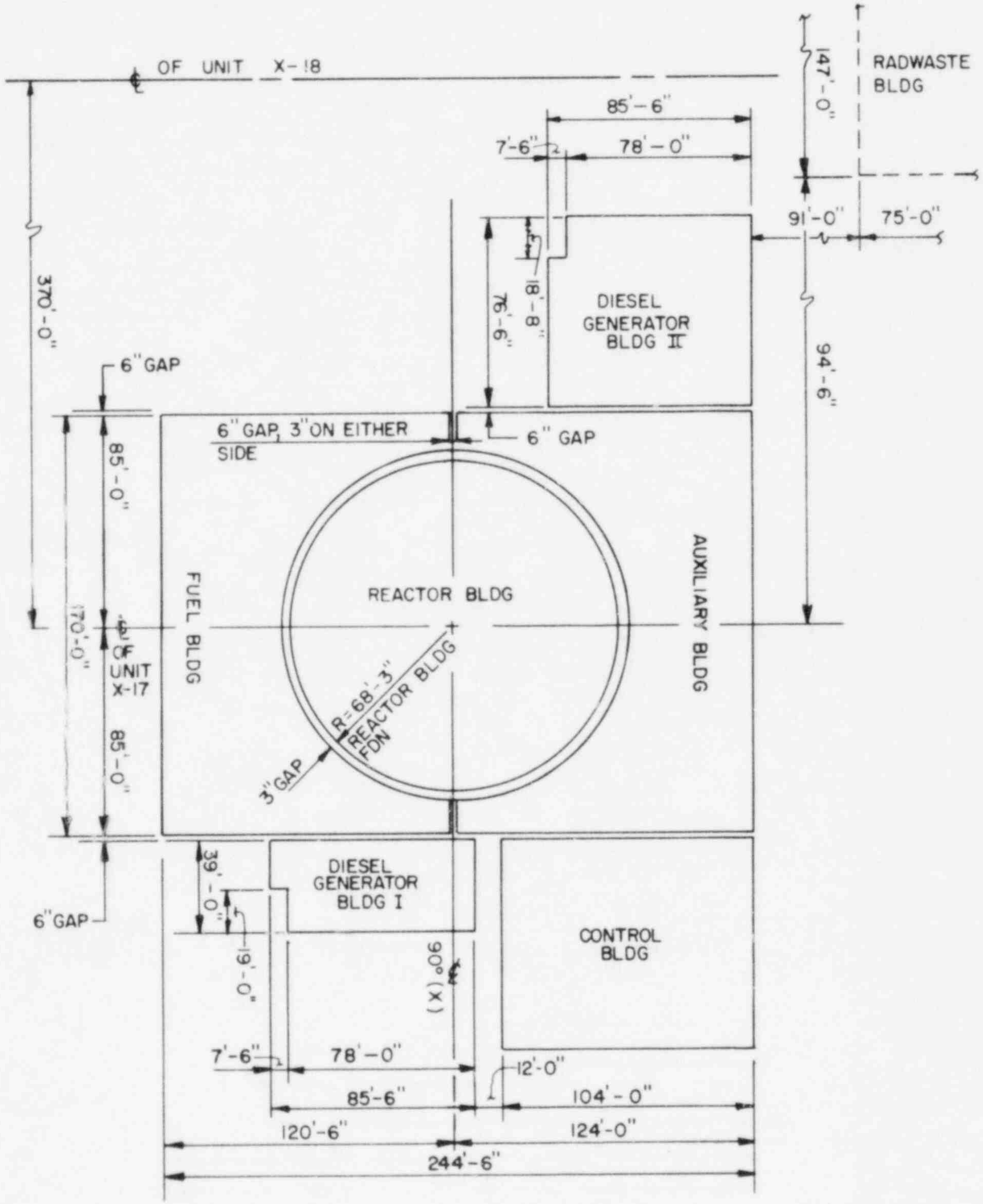


Figure 3.8-88. Foundation Outlines in Reactor Island

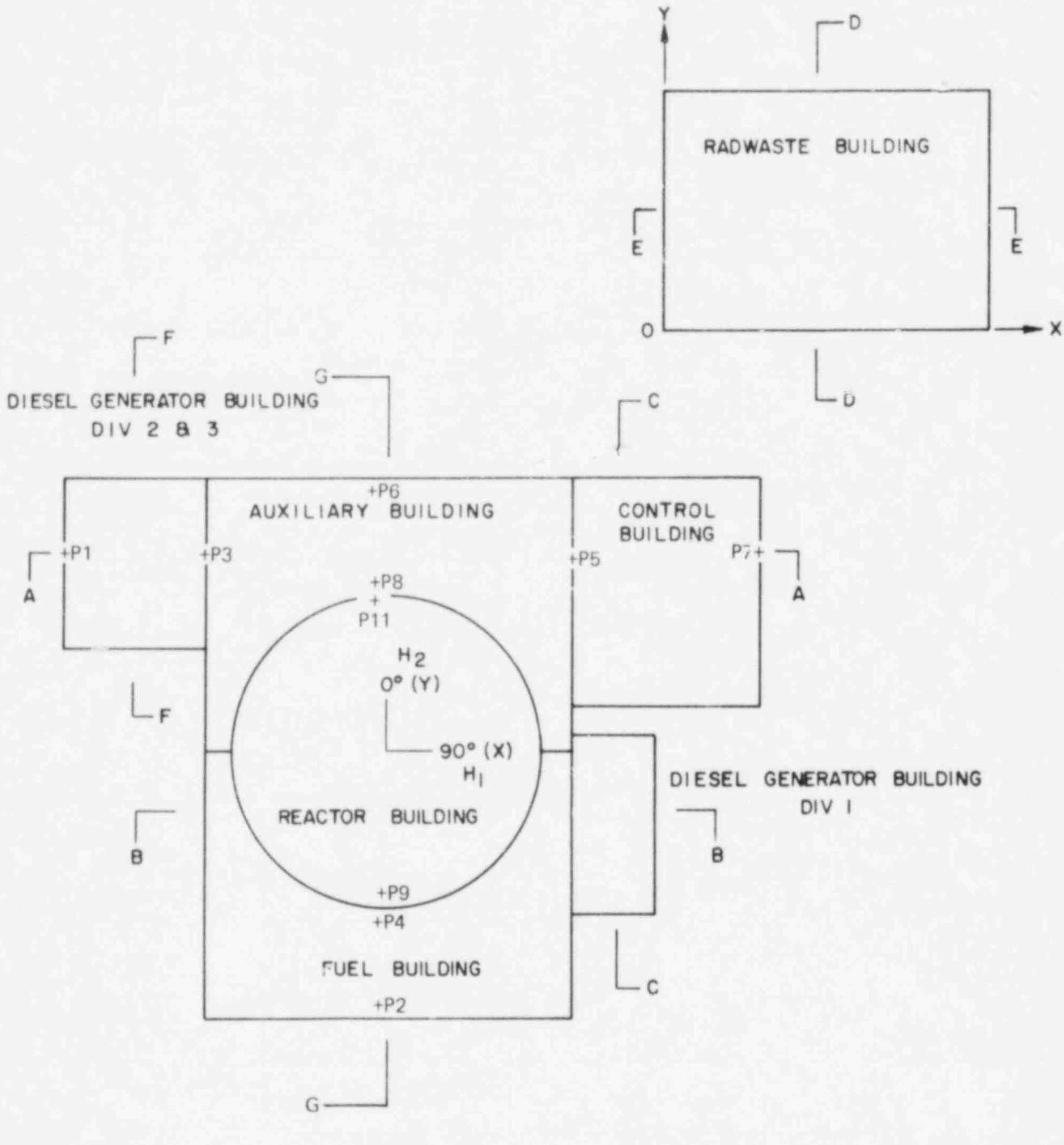


Figure 3.8-89. General Plant Arrangement

3.21

APPENDIX 3B
CONTAINMENT LOADS

This appendix provides the thermal-hydraulic dynamic loading methodology for the General Electric Company (GE) Mark III pressure suppression containment system during a loss-of-coolant accident (LOCA), safety/relief valve (SRV) discharge and related dynamic events. Complete numerical information is provided for the GE Mark III Reference (238 Standard) Plant. Information is also provided for other GE Mark III Standard Plants. This information and guidance is provided to assist the Applicant in evaluating the design conditions for the various structures which form its containment system.

The NRC draft acceptance criteria for LOCA-related Mark III containment pool dynamic loads* will be addressed by the Applicant.

3B.1 INTRODUCTION

GE has concluded the confirmatory test program for the Mark III containment configuration. These tests support and confirm the pressure suppression loads that result from the postulated LOCA and from SRV operation. The confirmatory program includes a series of scaled multivent tests that demonstrate no significant vent interaction effects for the LOCA process. Also included is an evaluation of the full-scale Caorso SRV tests[1,2] as described in Attachment A.

During a LOCA and events such as SRV actuation, the structures forming the containment system and other structures within the Reactor Building experience dynamic phenomena. This appendix provides the numerical information on the dynamic loads that these phenomena impose on the Mark III containment system structures.

*Enclosure to NRC letter R. L. Tedesco to G. G. Sherwood (GE), "Draft Acceptance Criteria for Pool Dynamic Loads," April 5, 1982.

3B.1 INTRODUCTION (Continued)

The loading information is based on either observed test data or conservatively calculated peak values. The LOCA loading combinations are presented in the form of bar charts for each of the containment system structures. In addition to defining the timing of the LOCA related loads, the bar charts identify other loading conditions such as seismic accelerations, dead-weight, etc. For each bar on the chart, reference is made to the section of the appendix where specific discussion of the load is presented.

To provide a better understanding of the various dynamic loads and their interrelationships, Section 3B.2 contains a qualitative description of sequential events for a wide range of postulated accidents. The air-clearing loading phenomena associated with the actuation of a SRV are also described.

3B.1.1 Confirmatory Testing

Impact and impingement load specifications for small structures affected by suppression pool swell are based on the results of the Pressure Suppression Test Facility (PSTF) air tests conducted in March 1974[3]. The intent of these tests was to provide conservative design data. It was recognized that the data base would require extension beyond that provided by the air tests and, to achieve this, additional impact tests for both small and large structures were included in the PSTF schedule. These tests involved measurement of pool swell impact forces on a variety of targets representative of small structures found in the Mark III containment annulus and are discussed in Attachment B.

This appendix relies on a large experimental test data base from the PSTF program. See Table 3B-1 for a summary of these tests. The scaling of the large-scale and 1/3-area scale PSTF precludes direct application to the prototype Mark III. Conservative interpretation of these tests results employing dimensional similitude

3B.1.1 Confirmatory Testing (Continued)

scaling relationship is used to arrive at specified design loads for Mark III (Attachment B).

As previously mentioned, evaluation of full-scale Caorso SRV tests is included in Attachment A. The evaluation shows that the SRV load design values are adequately conservative.

3I.1 PRODUCT PERFORMANCE QUALIFICATION SPECIFICATION GUIDELINES

PROPRIETARY INFORMATION - provided under separate cover

3I.1 PRODUCT PERFORMANCE QUALIFICATION SPECIFICATION GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.1 PRODUCT PERFORMANCE QUALIFICATION SPECIFICATION GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.1 PRODUCT PERFORMANCE QUALIFICATION SPECIFICATION GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.1 PRODUCT PERFORMANCE QUALIFICATION SPECIFICATION GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.1 PRODUCT PERFORMANCE QUALIFICATION SPECIFICATION GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.2 PRODUCT ANALYSIS REPORT GUIDELINES

PROPRIETARY INFORMATION - provided under separate cover

31.2 PRODUCT ANALYSIS REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.2 PRODUCT ANALYSIS REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.2 PRODUCT ANALYSIS REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.5 PRETEST EVALUATION

PROPRIETARY INFORMATION - provided under separate cover

31.5 PRETEST EVALUATION (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.5 PRETEST EVALUATION (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.5 PRETEST EVALUATION (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.5 PRETEST EVALUATION (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.5 PRETEST EVALUATION (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.6 TEST PLAN AND PROCEDURES GUIDELINES

PROPRIETARY INFORMATION - provided under separate cover

31.6 TEST PLAN AND PROCEDURES GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.6 TEST PLAN AND PROCEDURES GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.6 TEST PLAN AND PROCEDURES GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.6 TEST PLAN AND PROCEDURES GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.6 TEST PLAN AND PROCEDURES GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.7 TEST REPORT GUIDELINES

PROPRIETARY INFORMATION - provided under separate cover

31.7 TEST REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.7 TEST REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.7 TEST REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

3I.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

31.8 ENVIRONMENTAL QUALIFICATION REPORT GUIDELINES (Continued)

PROPRIETARY INFORMATION - provided under separate cover

6.2.1.7 Instrumentation Requirements (Continued)

drywell and containment-to-shield annulus differential pressure and suppression pool level as inputs to the ESF systems. Suppression pool temperature monitoring, drywell and RWCU Room temperature monitoring and the Ventilation Exhaust Radiation Monitoring System are discussed in Section 7.6. The display instrumentation for all containment parameters, including the number of channels, recording of parameters, instrument range and accuracy and post-accident monitoring equipment is discussed in Section 7.5.

6.2.2 Containment Heat Removal System

6.2.2.1 Design Basis

The Containment Heat Removal System, consisting of the Containment Cooling System, is an integral part of the RHR system. The purpose of this system is to prevent excessive containment temperatures and pressures, thus maintaining containment integrity following a LOCA. To fulfill this purpose, the containment cooling system meets the following safety design bases:

(1) The system shall limit the long-term bulk temperature of the suppression pool to 185°F without spray operation when considering the energy additions to the containment following a LOCA. These energy additions, as a function of time, are provided in the previous section.

(2) The single-failure criterion shall apply to the system.

(3) The system shall be designed to safety grade requirements including the capability to perform its function following a Safe Shutdown Earthquake.

6.2.2.1 Design Basis (Continued)

(4) The system shall maintain operation during those environmental conditions imposed by the LOCA.

(5) Each active component of the system shall be testable during normal operation of the nuclear power plant.

6.2.2.2 Containment Cooling System Design

The Containment Cooling System is an integral part of the RHR system. Water is drawn from the suppression pool, pumped through one or both RHR heat exchangers and delivered to the suppression pool or to the containment spray header. Water from the Essential Service Water (ESW) System is pumped through the heat exchanger tube side to exchange heat with the processed water. Two cooling loops are provided, each being mechanically and electrically separate from the other to achieve redundancy. A piping and instrumentation (P&I) diagram is provided in Section 5.4. The process diagram, including the process data, is provided for all design operating modes and conditions.

All portions of the Containment Cooling System are designed to withstand operating loads and loads resulting from natural phenomena. All operating components can be tested during normal plant operation so that reliability can be assured. Construction codes and standards are covered in Subsection 5.4.7.

The Containment Cooling System is started manually or automatically in the case of containment sprays. The LPCI mode is automatically initiated from ECCS signals and the RHR System realigned for containment cooling by the plant operator after the reactor vessel water level has been recovered (Subsection 6.2.1). The RHR pumps are already operating. Containment cooling is initiated in loop A or B by manually starting the ESW pump, closing the heat exchanger bypass valve, opening the service water valve at the heat

6.2.2.2 Containment Cooling System Design (Continued)

exchangers, closing the LPCI injection valve and opening the pool return valve. In the event that a single failure has occurred, and the action which the plant operator is taking does not result in system initiation, then the operator will place the other totally redundant system into operation by following the same initiation procedure. If the operator chooses to utilize the containment spray, he must close the LPCI injection valves and open the spray valves. The containment spray mode is also initiated automatically on high containment pressure, with an interlock to delay initiation until 10 min after a high drywell pressure signal. Automatic initiation is provided to protect the containment in the event of suppression pool bypass leakage as is described in Section 6.2.1.1.5.4.

Preoperational tests are performed to verify individual component operation, individual logic element operation and system operation up to the containment spray spargers. A sample of the sparger nozzles is bench tested for flow rate versus pressure drop to evaluate the original hydraulic calculations. Finally, the spargers are tested by air and visually inspected to verify that all nozzles are clear. (See Subsection 5.4.7.4 for further discussion of preoperational testing.)

6.2.2.3 Design Evaluation of the Containment Cooling System

In the event of the postulated LOCA, the short-term energy release from the Reactor Primary System will be dumped to the suppression pool. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The Containment Cooling System will remove this energy which is released into the primary containment system, thus resulting in acceptable suppression pool temperatures and containment pressures.

6.2.2.3 Design Evaluation of the Containment Cooling System (Continued)

In order to evaluate the adequacy of the RHR System, the following sequence of events is assumed to occur:

(1) With the reactor initially operating at 102% of rated power, a LOCA occurs.

(2) A loss of offsite power occurs and one emergency diesel fails to start and remains out of service during the entire transient. This is the worst single failure.

(3) Only three ECCS pumps are activated and operated as a result of there being no offsite power and minimum onsite power. (Section 6.3 describes the ECCS equipment.)

(4) After 30 min, it is assumed that the plant operators activate one RHR heat exchanger in order to start containment heat removal. Once containment cooling has been established, no further operator actions are required.

General compliance or alternate approach assessment for Regulatory Guide 1.1 may be found in Subsection 6.3.2.2.

General compliance for Regulatory Guide 1.26 may be found in Subsection 3.2.2.

Failure modes and effects analyses for the RHR and ESW Systems are provided in Appendix 15C.

6.2.2.3.1 Summary of Containment Cooling Analysis

When calculating the long-term, post-LOCA pool temperature transient, it is assumed that the initial suppression pool temperature

6.2.2.3.1 Summary of Containment Cooling Analysis (Continued)

and the RHR service water temperature are at their maximum values. This assumption maximizes the heat sink temperature to which the containment heat is rejected and thus maximizes the containment

7.2.1.2.I Design Bases (Continued)

will maintain a scram signal condition at the control rod drive system terminals until the trip channels have returned to their normal operating range and the seal-in is manually reset by operator action. Thus, once a trip signal is present long enough to initiate a scram and the seal-ins, the protective action will go to completion.

7.2.1.3 Final System Drawings

The final RPS drawings are processed at two different levels relative to this document.

First, all the necessary system and subsystem level instrument and electrical diagrams (IEDs) and channel logic diagrams are provided in this section.

Second, detailed circuit, component design elements, and cabinet and panel layout drawings (or similar finite detail design diagrams) are being provided under separate cover by reference in Section 1.7. This documentation is complementary to discussions and drawings included in this chapter.

A direct comparison of the subject documents verifies this observation. A list of drawings supplied under separate cover is provided in Section 1.7.

7.2.2 Conformance Analysis

This subsection presents an analysis of how the various functional requirements and the specific regulatory requirements of the RPS design bases are satisfied. A failure modes and effects analysis for the RPS will be provided by the Applicant using procedures outlined in Appendix 15C. 7.3

7.2.2 Conformance Analysis (Continued)

7.2.2.1 Conformance to Design Bases Requirements

A. Design Bases

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier. Chapter 15, Accident Analysis, identifies and evaluates events that jeopardize the fuel barrier. The methods of assessing barrier damage and radioactive material releases along with the methods by which abnormal events are sought and identified are presented in that chapter.

Design bases require that the precision and reliability of the initiation of reactor scrams be sufficient to prevent or limit fuel damage.

Table 7.2-1 provides a listing of the sensors selected to initiate reactor scrams and delineates the range for each sensor. Setpoints, accuracy and response time can be found in Chapter 16. This information establishes the precision of the RPS variable sensors.

The selection of scram trip settings has been developed through analytical modeling, experience, historical use of initial setpoints and adoption of new variables and setpoints as experience was gained. The initial setpoint selection method provided for settings which were sufficiently above the normal operating levels (to preclude the possibilities of spurious scrams or difficulties in operation) but low enough to protect the fuel. As additional information became available or systems were changed, additional scram variables were provided using the above method for initial setpoint selection. The selected scram settings are analyzed to verify that they are conservative and that the fuel and fuel

7.3.1.2.G Design Basis Information (Continued)

To protect the ESF systems in the event of a postulated fire, the redundant portions of the systems are generally separated by fire barriers. If a fire were to occur within one of the sections of a main control room panel or in the area of one of the local panels, the ESF systems functions would not be prevented by the fire. The use of separation and fire barriers ensures that, even though some portion of the systems may be affected, the ESF systems will continue to provide the required protective action.

Plant fire protection system is discussed in Subsection 9.5.1 and Appendix 9.5.1.A. Licensing Topical Report NEDO-10466-A discusses the details of the fire protection system.

4. LOCA

The following ESF system instrument taps and sensing lines are located inside the drywell and terminate outside the drywell. They could be subjected to the effects of a design basis loss-of-coolant accident (LOCA):

- Reactor vessel pressure
- Reactor vessel water level
- Drywell pressure

These items have been environmentally qualified to remain functional during and following a LOCA as discussed in Section 3.11 and indicated in Table 3.11-1.

7.3.1.2.G Design Basis Information (Continued)

H. Minimum Performance Requirements

The instrumentation and control for the various systems described in this section shall, as a minimum, initiate safety action in a sufficient number of systems and subsystems to accomplish timely initiation of any required safety function under conditions of a single design basis event with its consequential damages and a single failure together with its consequential damages.

Trip points are within the operating range of instruments with full allowance for instrument error, drift and setting error.

7.3.1.3 Final System Drawings

The final system drawings, including Process and Instrumentation Diagrams, Functional Control Diagrams and Logic Diagrams, Instrument and Electrical Drawings, and Elementary Diagrams, have been provided for the ESF.

Full-size logic, schematic, electrical interconnection will be supplied under separate cover as the regulations allow. A list of the drawings is provided in Section 1.7. P&IDs are provided within the FSAR in Chapters 5, 6, and 9, and are referenced where appropriate in Chapter 7. Elementary diagrams are seen in Appendix 7A. All other diagrams are included in Chapter 7 as appropriate.

7.3.2 Analysis

Failure modes and effects analyses for ESF Systems are provided in Appendix 15C. 7.7

7.3.2.1 Emergency Core Cooling Systems Instrumentation and Controls

9.1.4.3 Safety Evaluation of Fuel-Handling System (Continued)

in order to remove the grapple from the water for servicing and for storage.

The grapple has two independent hooks, each operated by an air cylinder. Engagement is indicated to the operator. Interlocks prevent grapple disengagement until a "slack cable" signal from the lifting cables indicates that the fuel assembly is seated. The slack cable indication is also used to determine if a fuel bundle is lodged in a position other than its normal, seated position in the core.

In addition to the main hoist on the trolley, there is an auxiliary hoist on the trolley and another hoist on its own monorail. These three hoists are precluded from operating simultaneously because control power is available to only one of them at a time. The two auxiliary hoists have load cells with interlocks which prevent the hoists from moving anything as heavy as a fuel bundle.

The two auxiliary hoists have electrical interlocks which prevent the lifting of their loads higher than 8 ft underwater. Adjustable mechanical jam-stops on the cables back up these interlocks.

The spent fuel handling crane is a low profile crane which is limited in travel so it cannot carry a shipping cask over stored fuel. Also, its height is limited such that the cask cannot be lifted up on the operating floor. Thus the cask cannot roll into the fuel pool if accidently dropped.

Further, a series of watertight gates are provided such that the cask left never exceeds the 30-ft drop design criteria. The cask is moved to the loading area and gated off and the loading pool filled with water. Only then is the fuel storage pool connected to the cask loading pool and the fuel transfer begun. When the 9.5

9.1.4.3 Safety Evaluation of Fuel-Handling System (Continued)

cask is loaded, the fuel storage pool is gated closed and the cask removal procedure reversed. A decontamination area is provided. 9.5

Applicant will describe any deviations to this arrangement.

In summary, the fuel-handling system complies with General Design Criteria 2, 3, 4, 5, 61, 62 and 63, and applicable portions of 10CFR50.

A failure modes and effects analysis for the Reactor and Fuel Servicing/Inclined Fuel Transfer System is given in Appendix 15C. 9.4

The safety evaluation of the new and spent fuel storage is presented in Subsections 9.1.1.3 and 9.1.2.3.

9.3.5.3 Safety Evaluation (Continued)

APCSB 3-1 and MEB 3-1

Since the SLC System is located within its own compartment inside the containment, it is adequately protected from flooding, tornadoes and internally externally generated missiles. SLC System equipment is protected from pipe break by providing adequate distance between the seismic and nonseismic SLC System equipment, where such protection is necessary. In addition, appropriate distance is provided between the SLC System and other high energy piping systems. Barriers have been considered to assure SLC System protection from pipe break (Section 3.6).

It should be noted that the SLC System is not required to provide a safety function during any postulated pipe break events. This system is only required under an extremely low probability event, where all of the control rods are assumed to be inoperable while the reactor is at normal full power operation. Therefore, the protection provided is considered over and above that required to meet the intent of APCSB 3-1 and MEB 3-1.

This system is used in special plant capability demonstration events cited in Appendix A of Chapter 15; specifically, Events 46 and 48, which are extremely low probability nondesign basis postulated incidents. The analyses given there are to demonstrate additional plant safety consideration far beyond reasonable and conservative assumptions.

A failure mode and effects analysis for the SLC System is provided in Appendix 15C. 9.6

9.3.5.4 Testing and Inspection Requirements

Operational testing of the SLC System is performed in at least two parts to avoid inadvertently injecting boron into the reactor.

With the valves to the reactor and from the storage tank closed and the valves to and from the test tank opened, condensate water in the test tank can be recirculated by locally starting either pump.

During a refueling or maintenance outage, the injection portion of the system can be functionally tested by valving the suction line to the test tank and actuating the system from the control room. System operation is indicated in the control room.

After functional tests, the injection valve shear plugs and explosive charges must be replaced and all the valves returned to their normal positions as indicated in Figure 9.3-5.

After closing a local locked-open valve to the reactor, leakage through the injection valves can be detected by opening valves at a test connection in the line between the drywell checkvalves. Position indicator lights in the control room indicate that the local valve is closed for tests or open and ready for operation. Leakage from the reactor through the first checkvalve can be detected by opening the same test connection in the line between the checkvalves when the reactor is pressurized.

The test tank contains condensate water for approximately 3 min of pump operation. Condensate water from the Makeup System or the Condensate Storage System is available for refilling or flushing the system.

9.5.1.3 Safety Evaluation (Fire Hazard Analysis - Appendix 9A) (Continued)

The basis of the overall plant design with respect to the effects of fire is to assume that all functions are lost for equipment, including electrical cables, located within a fire area experiencing a fire. Redundant equipment is provided in other fire areas. A fire area by fire area treatment for the fire hazard analysis evaluates the compliance of the design to this requirement for redundancy. Compliance is confirmed. Therefore, the most serious consequence of a fire is that it may incapacitate one safety or safe shutdown division. This is consistent with the single failure design criteria used throughout the plant. Regardless of the location of a fire, sufficient operable equipment is assured for use in safely shutting the plant down.

The fire hazard analysis assumes that the function of a piece of equipment may be lost if the equipment is either involved in fire 9.1
fighting activities or subjected to fire suppression agents and confirms that redundant equipment out of the fire area is available. This redundant equipment is capable of performing the required safety or shutdown function. The basis of the design is not to assume a questionable limit on damage within a given fire area but to provide redundant equipment elsewhere.

As described in Appendix 9A (Section 9A.4.1.15, for example), the fire detection systems are Class A, and therefore are tolerant of single failures. The fire suppression systems are designed such that there are two suppression systems available to any given area. Areas covered by sprinklers or CO_2 systems are also covered by the manual hose system. Areas covered by manual hose systems only may be reached from at least two hose stations. Standpipes are fed from two directions.

Design of the fire protection supply system to the Nuclear Island is the responsibility of the Applicant.

9.5.1.3 Safety Evaluation (Fire Hazard Analysis - Appendix 9A) (Continued)

A failure modes and effects analysis (FMEA) for the wet standpipe fire protection system is provided in Appendix 15C. A FMEA for the carbon dioxide fire protection system will be provided by the Applicant utilizing the FMEA procedures given in Appendix 15C. 9.1

SECTION 12.2

TABLES

12.2

TABLES (Continued)

ILLUSTRATIONS

12.2

12.2.1.2.6.3 Radioactive Sources in the Gaseous Radwaste System (Continued)

been evaluated for several possible operating modes. In all cases, a 1-yr operating time has been used to accumulate the decay activities. This is sufficient time for most isotopes to reach equilibrium.

12.2.1.2.6.4 Radioactive Sources in the Solid Radwaste System

The solid radwaste system provides the capability for solidifying and packaging waste from the other radwaste systems (Subsection 11.4.2). The wastes are not solidified separately by type or source. The final waste is placed in a steel container. The expected average radioactivity content of the solid waste per container is given in Table 12.2-15.

12.2.1.2.6.5 Radioactive Sources in the Fuel Pool Cleanup System

The radiation source data used in the shield design of the Fuel Pool Cleanup (FPCCU) System filter demineralizer system is given in Table 12.2-16.

12.2.1.2.6.6 Radioactive Sources in the Suppression Pool Cleanup System

The radiation source data used in the shield of the Suppression Pool Cleanup (SPCU) System is given in Table 12.2-17.

12.2.1.2.7 Radioactive Sources in Piping and Main Steam Systems

12.2.1.2.7.1 Radioactive Sources in Main Steam System

All radioactive materials in the Main Steam System result from radioactive sources carried over from the reactor during plant operation. In most of the components carrying live steam, the

12.2.1.2.7.1 Radioactive Sources in Main Steam System (Continued)

source is dominated by Nitrogen-16. In components where N-16 has decayed, the other activities carried by the steam become significant. During plant shutdown, there is a residual activity resulting from prior plant operations. These data will be provided by the Applicant.

12.2.1.2.7.2 Radioactive Crud in Piping and Steam Systems

The inside surfaces of the piping and all reactor and power systems components become coated with activated corrosion products, commonly called crud. The quantity of crud on the components is dependent on a number of factors, including power history, water quality and fuel experience. The piping and components carrying reactor water are coated with higher levels of crud than piping and components carrying steam. Figure 12.2-2 shows the data used in the design of this plant to characterize crud accumulation in Recirculation System Piping. Crud levels in steam piping are estimated to be about 1% of those in the recirculation piping. 12.2

12.2.1.2.8 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in Subsection 12.2.1.2.1.1.4 (Table 12.2-3) in terms of MeV/sec/W. The design calculation is carried out for a mean element for an appropriate decay time.

12.2.1.2.9 Other Radioactive Sources

12.2.1.2.9.1 Reactor Startup Source

The reactor startup source is shipped to the site in a special cask designed for shielding. The source is transferred under water while in the cask and loaded into beryllium containers. This is then loaded into the reactor while remaining under water. The

Table 12.2-5

FAST NEUTRON FLUXES AND GAMMA RAY DOSE
AROUND THE REACTOR VESSEL*

Location	Fast Neutron Flux ($\geq$1.0 MeV) (n/cm^2-sec)	Gamma Ray Dose Rate (rads/hr)
Core - at Peak Axial Location	2.7 E+8	2.0 E+4
Above Top Head	1.0 E+2	2.2 E+1
Below Bottom Head	<0.1	1.7 E-1

*The data presented in this table are calculations of the primary and secondary radiation from the core vessel. In power plant applications, the neutron fluxes above and below the core will include contributions from radiation exiting the vessel at mid-plane and being scattered to these locations.

Table 12.2-6

FISSION PRODUCT GAMMA SOURCE STRENGTH IN RHR
HEAT EXCHANGER SHELL (MeV/sec)

Energy Bounds (MeV)	Gamma-Ray Source (MeV/sec)
6.0	
	1.2 + 10
4.0	
	1.8 + 10
3.0	
	4.8 + 10
2.6	
	3.4 + 11
2.2	
	4.5 + 11
1.8	
	6.0 + 11
1.4	
	9.9 + 11
0.9	
	2.1 + 12
0.4	
	5.3 + 11
0.1	

12.2

Table 12.2-7

FISSION PRODUCT INVENTORY IN THE RHR HEAT EXCHANGER SHELL
3, 2 HOURS AFTER SHUTDOWN (Ci)

Isotope	Activity
Kr-85m	3.4E + 00
Kr-85	2.8E - 01
Kr-87	1.9E + 00
Kr-88	6.9E + 00
Rb-88	1.2E + 01
Rb-89	5.0E - 03
Sr-89	4.4E - 02
Te-131m	3.1E - 04
Te-131	1.4E - 02
I-131	1.4E + 01
Xe-131m	1.8E - 01
Te-132	5.1E - 03
I-132	1.2E + 01
Te-133m	1.9E - 04
Te-133	1.0E - 02
I-133	3.1E + 01
Xe-133m	8.6E - 01
Xe-133	3.0E + 01
Te-134	1.5E - 04
I-134	4.0E + 00
I-135	2.4E + 01
Xe-135m	5.1E + 00
Xe-135	2.9E + 01
Cs-137	3.7E - 04
Ba-137m	2.0E - 01
Xe-138	2.9E - 03
Cs-138	1.0E + 00

12.2

Table 12.2-8

REACTOR COOLANT CONCENTRATION EQUILIBRIUM VALUES
ENTERING RCIC TURBINE (μCi/g)

Isotope	Concentration	Isotope	Concentration
N-13	6.6E-03	Kr-88	1.1E-02
N-16	4.8E-01	Kr-89	1.9E-04
N-17	1.5E-02	Xe-131m	3.7E-05
F-18	4.0E-03	Xe-133m	1.9E-04
O-19	6.9E-01	Xe-133	4.7E-03
Na-24	2.3E-06	Xe-135m	6.0E-03
Mn-56	[illegible]E-04	Xe-135	1.4E-02
Co-58	5.9E-06	Xe-137	7.3E-04
Br-83	5.1E-04	Xe-138	1.7E-02
Br-84	6.7E-04	Sr-91	3.4E-04
I-131	4.3E-04	Sr-92	3.4E-04
I-132	4.7E-03	Tc-99	1.6E-04
I-133	3.1E-03	Tc-101	5.3E-05
I-134	6.8E-03	Cs-138	1.4E-02
I-135	4.8E-03	Ba-139	2.8E-03
Kr-83m	2.1E-03	Ba-140	1.9E-05
Kr-85m	3.4E-03	Ba-141	5.8E-04
Kr-85	1.2E-05	Ba-142	1.3E-04
Kr-87	9.7E-03	Np-239	2.5E-04

Table 12.2-16

RADIOACTIVE SOURCES IN THE FUEL POOL FILTER DEMINERALIZED SYSTEM

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	Ci	Isotope	Ci	Isotope	Ci	Isotope	Ci
Br-83	0.	Sr-89	2.8E+00	Zr-95	3.8E-02	Na-24	1.4E-04
Br-84	0.	Sr-90	2.4E-01	Zr-97	5.6E-06	P-32	1.5E-02
Br-85	0.	Sr-91	3.8E-05	Nb-95	1.2E-01	Cr-51	4.8E-01
I-131	1.1E+01	Sr-92	0.	Ru-103	1.7E-02	Mn-54	4.8E-02
I-132	2.1E+00	Y-90	2.4E-01	Ru-106	2.7E-03	Mn-56	0.
I-133	1.3E-01	Y-91M	3.8E-05	Rh-103M	1.7E-02	Co-58	5.5E-00
I-134	0.	Mo-99	2.2E+00	Rh-106	2.76-03	Co-60	6.0E-01
I-135	3.4E-07	Tc-99M	1.5E-08	La-140	6.3E-00	Fe-59	8.6E-02
		Tc-101	0.	Ce-141	3.4E-02	Ni-65	0.
		Te-129M	2.7E-01	Ce-143	3.6E-04	Zn-65	2.4E-03
TOTAL	1.3E+01	Te-132	2.0E+00	Ce-144	3.6E-02	Zn-69M	9.7E-07
		Cs-134	1.7E-01	Pr-143	2.4E-02	Ag-110M	7.3E-02
		Cs-136	6.6E-02	Nd-147	8.1E-03	W-187	6.6E-03
		Cs-137	2.6E-01				
		Cs-138	0.				
		Ba-137	2.6E-01	TOTAL	6.6E+00	TOTAL	6.8E-00
		Ba-139	0.				
		Ba-140	5.5E-00				
		Ba-141	0.				
		Ba-142	0.				
		Np-239	1.6E+01				
		TOTAL	3.0E+01				

Source Volume = 480 gallons

Total Curies = 56

Table 12.2-17

RADIOACTIVE SOURCES IN THE SUPPRESSION POOL CLEANUP SYSTEM
(ACTIVITY IN THE BACKWASH TANK)(Ci)

Isotope	Activity	Isotope	Activity
85Br	1.6E-3	88Rb	7.6E-1
87Br	2.9E-4	89Rb	7.6E-1
88Rb	2.2E-1	89Sr	2.1E-2
89Rb	2.0E-1	131Te	1.2E-6
89Sr	1.5E-2	131I	1.7E-0
131I	1.1E-1	134Te	1.9E-6
132Te	1.8E-5	134I	6.4E-1
132I	1.9E-2	135I	1.5E-0
133I	8.9E-2	137Cs	4.9E-3
134Im	6.6E-4	137Bam	2.4E-3
134I	2.8E-2	138Cs	3.4E+1
135I	4.7E-2		
136Im	1.4E-4	Total	1.2E+2
136I	1.2E-3		
137I	6.6E-6		
137Cs	4.8E-3		
137Bam	2.3E-3		
138Cs	1.9E-1		
Total	9.3E-1		

Table 12.2-18 (Deleted)

12.2

Table 12.2-19

RADIOACTIVE SOURCES IN CONTROL ROD DRIVE SYSTEM

Control Rod Drive Radiation Survey Data

	Gamma Dose Measured at Contact MR/hr			
	Before Cleaning		After Cleaning	
Component	Maximum	Average	Maximum	Average
Spud	10,000	600	500	110
Filter	23,000	3,500	20,000	300
Collet Housing	3,000	1,800	4,000	700
Outer Cylinder	1,200	60	80	40
Strainer	8,000	1,800	1,000	500
Flange	1,000	200	400	150

Control Blade Principal Isotopes

Isotope	Curies (135 GWd/Te 7-Days Cooled) Ci/Blade
Cr51	1.4E5
Mn54	9.1E3
Fe55	1.6E5
Co58m	7.7E3
Co58	8.8E3
Co60	1.1E5
Ni63	5.0E3
Total	4.4E5

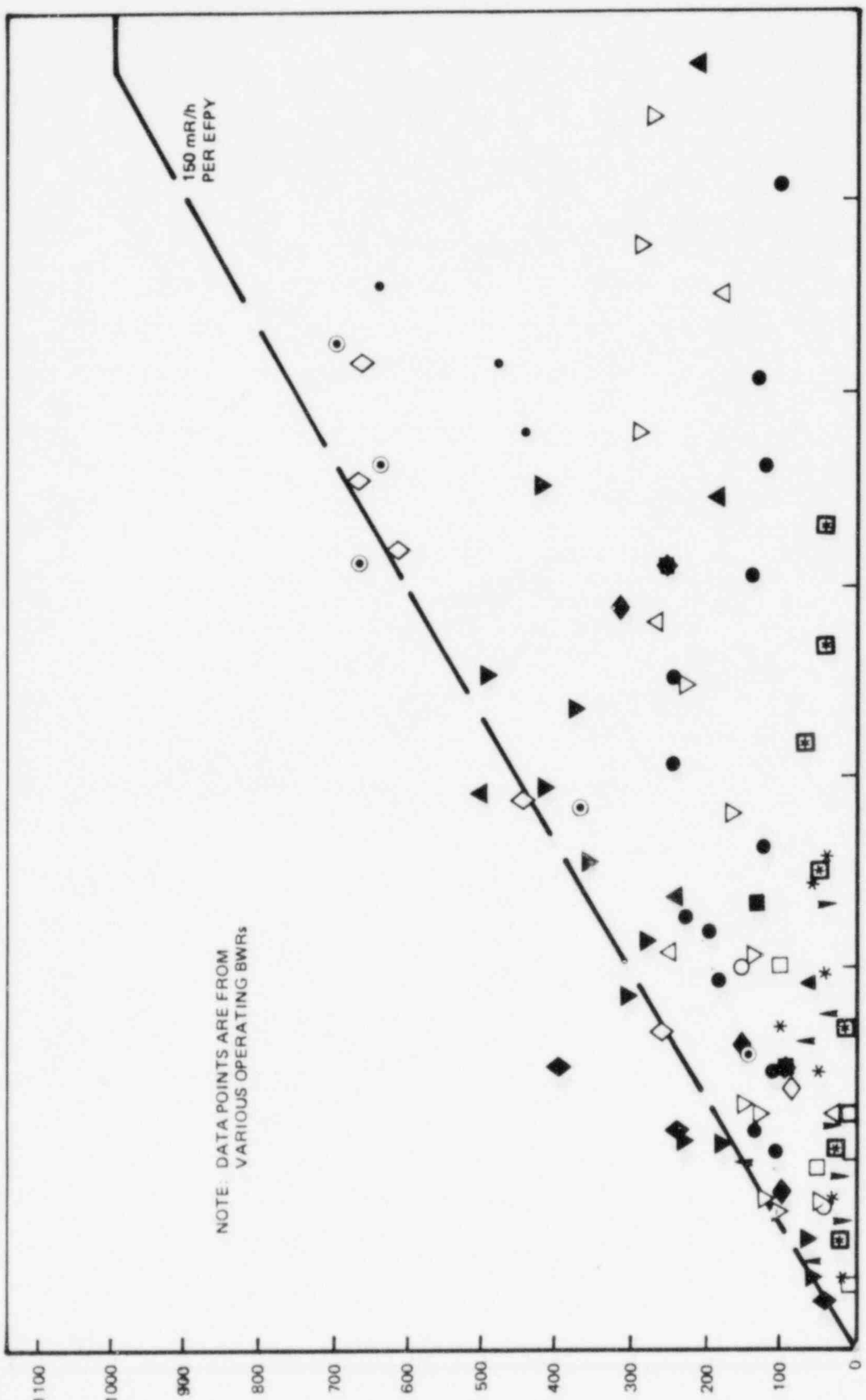


Figure 12.2-2. Buildup of Dose Rate Due to Radioactive Crud in Recirculation Piping

12.2

APPENDIX 15C

FAILURE MODES AND EFFECTS ANALYSIS

APPENDIX 15C

FAILURE MODES AND EFFECTS ANALYSIS

CONTENTS

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

15C.0 GENERAL

Regulatory Guide 1.70 requires failure modes and effects analysis (FMEA) to be performed on selected subsystems of Chapters 6, 7, and 9. This appendix provides the FMEAs for 20 systems and identifies 12 system FMEAs for the applicant to provide. In addition, several interfacing systems are identified in the 20 FMEAs as requiring FMEAs to be provided by the Applicant.

15C.0.1 Introduction

A FMEA is a qualitative failure study of the system design. A FMEA tabulates the effects of single component failures and represents a disciplined qualitative evaluation of the system from a reliability point of view. Every active mechanical and active or passive electrical component of the system is analyzed and evaluated for all conceivable failure modes with respect to its overall potential effect on safety. Those components and associated failure modes having an overall potential effect on safety are listed and described along with the cause(s) of failure. The analysis does not imply that any or all of the listed failure modes will occur; however, the estimated probability of occurrence is noted.

15C.0.2 Assumptions

Where applicable, each FMEA was developed utilizing the following assumptions:

(1) All testing, inspection, and maintenance are properly performed on a routine basis.

(2) All system and associated components are placed in the correct readiness state following any maintenance, checkout test, or periodic system surveillance test.

15C.0.2 Assumptions (Continued)

(3) Component failures are considered primarily for system post-accident operating modes. Non-post-accident modes of operation (normal operation, shutdown, etc.) are included only if a potential effect on safety exists. In such cases, the specific mode of operation is always identified.

(4) Operator action or error is excluded except for operator failure to follow a procedure in order to accomplish a necessary safety-related function. A single operator error or omission is considered in such a case.

(5) Common mode equipment failures and common causes (flooding, fire, tornado, seismic event, etc.) are not considered.

15C.0.3 FMEA Procedures and Task Scope

The procedures for preparation of the FMEAs are provided in Attachment A. The FMEA task scope is given in Attachment B.

15C.0.4 FMEA Format

The data contained in a FMEA form (Figure 15C-1) is listed as follows:

(1) Item. This column identifies each system component by identification number and description. Location of the device on the P&ID or Elementary Diagram is referenced as well as the device(s) it controls, when applicable.

(2) Mode of Failure. This column identifies all conceivable modes of failure for each component listed.

(3) Specific Local Effect of Failure. This column describes the specific local effect of each identified failure mode.

15C.0.4 FMEA Format (Continued)

(4) How Detected. This column describes the means for detecting each failure mode. Automatic and redundant detection provisions will be indicated when applicable.

(5) Overall Effect on Safety-Potential Effect. The safety-related functions and safety-related supporting functions were examined for each identified failure mode and operating condition to determine if the failure mode can in any way prevent the system or any other system from performing its assigned safety-related functions, on demand.

(6) Overall Effect on Safety-Compensation Provision. All existing compensating provisions are described here whenever a potential effect on safety is identified.

(7) Overall Effect on Safety-P (Probability). The annual probability of an event where a safety-related function is degraded or inhibited when it is required P is designated as follows:

Negligible (N)	- When probability is negligible, (i.e., $P < 10^{-7}$)
Low (L)	- When P is in the range, $10^{-7} \leq P < 10^{-4}$
Medium (M)	- When P is in the range, $10^{-4} \leq P < 10^{-2}$
High (H)	- When P is in the range, $10^{-2} \leq P < 1.0$

(8) Overall Effect on Safety-S (Significance). The significance of the loss of safety-related function on demand. The assignment of the significance is judgmental and is a function of the consequences of the loss or degradation of the safety-related function. S is designated as follows:

15C.0.4 FMEA Format (Continued)

None (N) - Radiological exposure to equipment, operators or public which does not lead to technical specification violation(s).

Small (S) - Radiological exposure to equipment but not operators or public which causes technical specification violation(s).

Medium (M) - Radiological exposure to equipment and operators but not public which causes technical specification violation(s).

Large (L) - Radiological exposure to equipment, operators and public which causes technical specification violation(s).

(9) Cause of Failure. This column describes the most likely cause(s) of failure for each component failure mode listed. Also the Probability (P) of each failure cause is indicated assuming the failure mode has occurred. The sum of the probabilities for all causes for a single failure mode should equal 1.0.

(10) Comments. This column indicates the failure mode classification (Subsection 15C.0.5) and provides other comments that are appropriate to the FMEA.

15C.0.5 Classification of Failure Modes

Each failure mode is classified within the following matrix based on the probability of its occurrence and its significance. Failure modes classified H, K, L, N, O and P are unacceptable and may require engineering action.

15C.0.5 Classification of Failure Modes (Continued)

Probability	Negligible (N)	Low (L)	Medium (M)	High (H)
Significance				
None (N)	A	B	C	D
Small (S)	E	F	G	H
Medium (M)	I	J	K	L
Large (L)	M	N	O	P

15C.0.6 System-Defining Documents

The system-defining documents (electrical, instrumentation, and control drawings, and piping and instrumentation diagrams) utilized in conducting the FMEAs are listed under each of the individual FMEAs. These documents are annotated versions of the corresponding documents listed in Table 1.7-1. However, some of the Table 1.7-1 documents were revised (updated) after the FMEAs were completed. In each case the impact of the document update(s) was assessed and it was determined that the FMEA results were still valid.

15C.0.7 Abbreviations

Abbreviations used in this appendix are defined in Table 15C.0-1.

Table 15C.0-1

ABBREVIATIONS

AB	Auxiliary Building
AC	Alternating Current
ACLD	AC Load Driver
ACU	Analog Comparator Trip Unit
ACUI	Analog Comparator Unit Interface
AO	Air Operated
AOV	Air-Operated Valve
BOP	Balance of Plant
BRKR	Breaker
CAOI	Computer Annunciator Opto Isolator
CCW	Closed Cooling Water
CDS	Condensate Distribution System
CIOI	Control and Indication Opto System
CRD	Control Rod Drive
DC	Direct Current
DCLD	DC Load Driver
DFS	Differential Flow Switch
DISCH	Discharge
DIV	Division
DRW	Dirty Radwaste
DW	Demineralized Water
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
ESW	Essential Service Water
FC	Failed Closed
FO	Failed Open
FPCCU	Fuel Pool Cooling and Cleanup
FT	Flow Transmitter
HDCS	High Voltage Level Input Digital Signal Conditioner
HI	High
HPCS	High Pressure Core Spray
HPCS SW	HPCS Service Water

Table 15C.0-1

ABBREVIATIONS (Continued)

HVAC	Heating, Ventilating, and Air Conditioning
HX	Heat Exchanger
H_2	Hydrogen
INBD	Inboard
IND	Indicator
INJ	Injection
ISO	Isolation
LC	Locked Closed
LIS	Level Indicating Switch
LLOI	Logic to Logic Optical Isolator
LO	Locked Open or Low
LOCA	Loss-of-Coolant Accident
LOPP	Loss of Preferred Power
LPCS	Low Pressure Core Spray
LS	Limit Switch
LT	Light or Level Transmitter
LVL	Level
MCC	Motor Control Center
MECH	Mechanical
MO	Motor Operated
MOV	Motor-Operated Valve
NA	Not Applicable
NC	Normally Closed
ND	Normally Deenergized
NE	Normally Energized
NO	Normally Opened
NSSSS	Nuclear Steam Supply Shutoff System
OPN	Open
OUTBD	Outboard
O_2	Oxygen
PI	Pressure Indicator
PLOI	Motor to Logic Optical Isolation
POS	Position

Table 15C.0-1

ABBREVIATIONS (Continued)

PRESS	Pressure
PWR	Power
RCIC	Reactor Core Isolation Cooling
RCTD	Resistor Capacitor Time Delay
RHR	Residual Heat Removal
RI	Reactor Island
RMS	Remote Manual Switch
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSTS	Remote Shutdown Transfer Switch
RV	Relief Valve
RW	Radioactive Waste
RWCU	Reactor Water Cleanup
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLME	System Logic Memory
SLTD	System Logic Time Delay
S/R	Safety Relief
SRU	Signal Resistor Unit
SUPP	Suppression
SYS	System
TI	Temperature Indicating
TLOI	Trip 4. Logic Opto Isolator
TYP	Typical
VL	Valve
VPI	Valve Position Indicating
WPS	Water Positive Seal

ITEM	MODE OF FAILURE	SPECIFIC LOCAL EFFECT OF FAILURE	HOW DETECTED	OVERALL EFFECT ON SAFETY				CAUSE OF FAILURE MODE	P	COMMENTS
				POTENTIAL EFFECT	COMP PROVISIONS	P	S			

Figure 15C.0-1. Failure Modes and Effects Analysis Form

15C.1 NUCLEAR BOILER SYSTEM (B21)

Applicant to provide

15C.2 STANDBY LIQUID CONTROL SYSTEM (C41)

15C.2.1 Scope

The FMEA covers all the active components depicted on the Standby Liquid Control System (SLCS) P&ID and all the mechanical, pneumatic, and electrical control devices necessary to permit the SLCS to perform its safety-related functions on demand.

15C.2.2 System Defining Documents

(1) P&ID, Figure 15C.2-1

(2) Elementary Diagram, Figures 15C.2-2a through e

(3) Simplified Block Diagram, Figure 15C.2-3

15C.2.3 System Safety-Related Functions

(1) Shuts down the reactor by injecting a neutron-absorbing solution into the primary reactor coolant sufficient to bring the reactor from full-rated power conditions to cold subcritical conditions, without control rod movement. This is a redundant backup shutdown system to the Control Rod Drive System. This system is only considered for normal shutdown, that is, not LOCA or transient conditions.

(2) Provides isolation signals to the Main Steam Supply Shutoff System to close Reactor Water Cleanup System (RWCUS) inboard and outboard isolation valves.

15C.2.4 Safety-Related Supporting System Functions

(1) AC & DC Power Distribution System, ESF (Divisions 1 and 2) - Provides electrical power to operate the SLCS.

(2) Control Building HVAC Systems - Provides suitable environment for SLCS instrumentation and controls.

15C.2.5 Initiating Events or Signals

Failure of control rod insertion on demand.

15C.2.6 Operator Actions Required

The system is to be manually initiated by the control room operator.

15C.2.7 System Description

The SLCS is classified as a safety system. It consists of a storage tank, a test tank, two pumps, two explosive-type valves and other valves, piping instrumentation and controls necessary to prepare and inject a neutron absorbing solution into the reactor and to test the system. The neutron absorber is pumped from the storage tank through explosive-type valves into the reactor vessel by remote manual operation of the pumps and valves. The SLCS is a redundant, independent, backup system for the control rod drive system. Its objective is to assure reactor shutdown from full power operation to cold subcritical, without control rod movement, by mixing a neutron absorber with the primary reactor coolant. The SLCS is not required to scram the reactor or to be a backup scram system for the reactor.

15C.2.8 FMEA Exclusions

The list of components or subsystems excluded because they do not perform a safety-related or safety-related supporting function, hence, they have no effect on safety, are as follows:

(1) P&ID Figure 15C.2-1
   (a) Pressure Regulators PCV-FF111, Instrumentation PI-R003, PR-N004
   (b) PI-R600, RO-D001, LI-R601, LI-R001, FIC-R004, TE-N006, TIC-R002
   (c) TS-N003, Electric Heaters, Sparger, Light and Annunciator

(2) Elementary Diagram Figures 15C.2-2a through e
Figure 15C.2-2e (Reference information only)

15C.2.9 Analysis and Results

The results of the FMEA given in Table 15C.2-1 are summarized as follows.

Number of line items analyzed	26
Number of failure modes analyzed	39
Number classified "A"	28
Number classified "B"	11

This FMEA verifies that the failure of check valves F006/F007 to open during SLCS operation mode will prevent the flow of neutron absorber into the reactor pressure vessel which results in failure of the standby liquid control function. However, the operation of the SLCS is required when both Division 1 and Division 2 control rod drive systems fail. Therefore, the failure of SLCS is the third failure which is beyond the scope of this analysis. This FMEA also verifies that the no single failure of any other active mechanical component or any active or passive electrical component of the Standby Liquid Control System will prevent this or any other system from performing its safety functions on demand.

Table 15C.2-1

STANDBY LIQUID CONTROL SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.2-1. Standby Liquid Control System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.2-2. Standby Liquid Control System Elementary Diagram (FMEA)

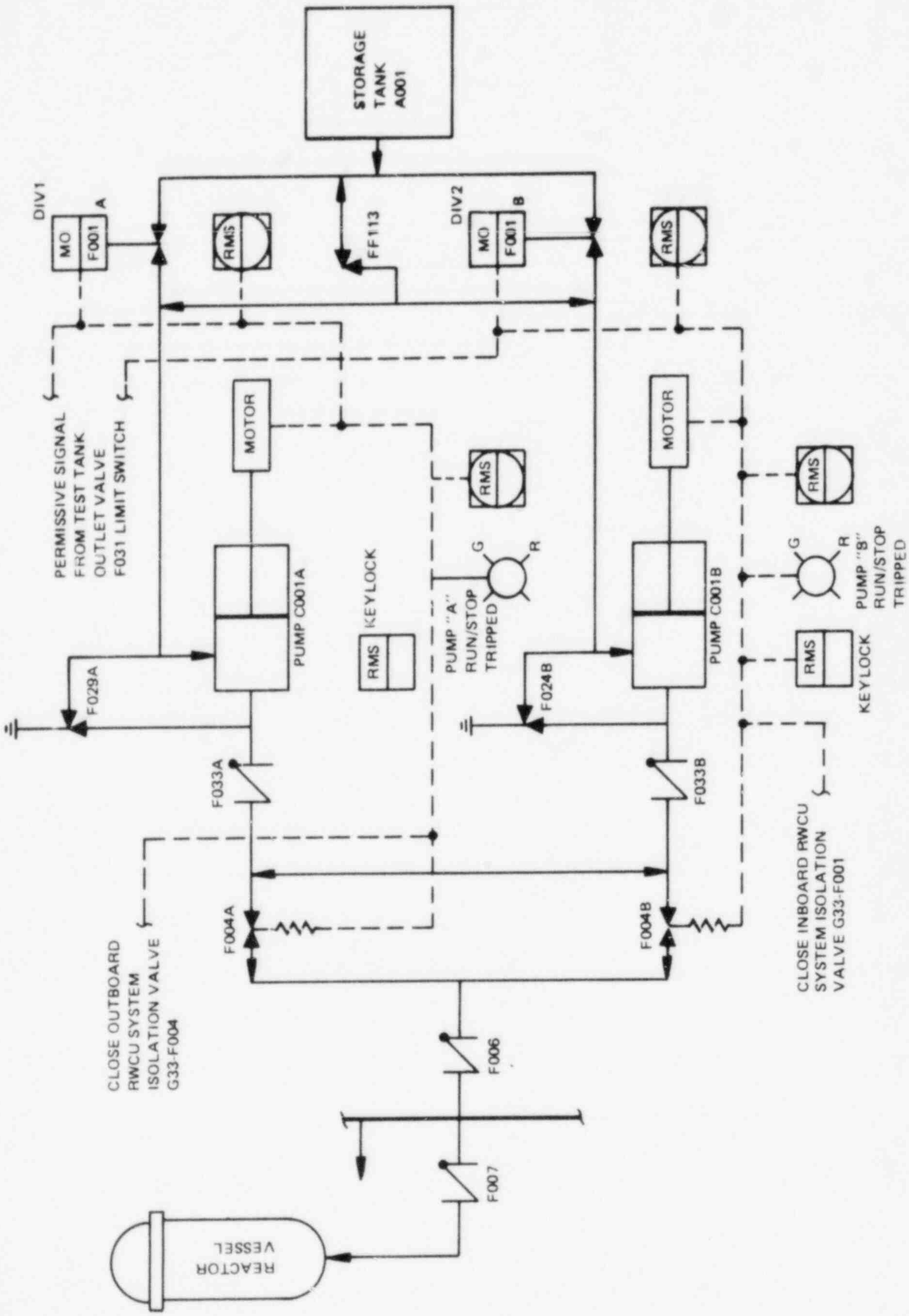


Figure 15C.2-3. Standby Liquid Control System Simplified Block Diagram

15C.3 REMOTE SHUTDOWN SYSTEM (C51)

Applicant to provide

15C.4 REACTOR PROTECTION SYSTEM (C71)

Applicant to provide

15C.5 RESIDUAL HEAT REMOVAL SYSTEM (E12)

(To be provided in December 1982)

15C.6 LOW PRESSURE CORE SPRAY SYSTEM (E21)

15C.6.1 Scope

The FMEA covers all the active components shown on the Low Pressure Core Spray (LPCS) System P&ID and all mechanical, pneumatic, and electrical devices necessary to permit the LPCS System to perform its safety-related functions on demand.

The mechanical devices include pneumatic, electric motor and manually operated valves, relief valves, pressure and fluid flow transducers, and water pumps. The pneumatic controls include air-operated valves. The electrical controls include the switches for the system, valve, water pump, and manual panel control; the electrical transducers; signal transmitters; trip units, converters; isolators; logic; and device load drivers.

15C.6.2 System Defining Documents

(1) P&ID, Figure 15C.6-1.

(2) Elementary Diagram, Figures 15C.6-2a through j.

(3) Simplified Block Diagram, Figure 15C.6-3.

15C.6.3 System Safety-Related Functions

(1) Provides reactor core cooling following a loss-of-coolant accident (LOCA) when proper reactor pressure conditions exist. The LPCS System, operating in conjunction with the Automatic Depressurization System (ADS), High Pressure Core Spray (HPCS) System, and Low Pressure Coolant Injection (LPCI) System, provides adequate reactor core cooling to prevent fuel cladding temperatures from exceeding limits for all design basis LOCAs.

15C.6.3 System Safety-Related Functions (Continued)

(2) Maintains primary containment integrity by providing system isolation valves.

(3) Provides the Automatic Depressurization System with redundant Division 1 reactor low water level and high drywell pressure signals.

(4) Provides the Residual Heat Removal System "A" with Division 1 low reactor pressure, high drywell pressure and LOCA signals.

(5) Provides the Standby Gas Treatment System with a Division 1 LOCA signal.

(6) Provides the Reactor Core Isolation Cooling System with a Division 1 high drywell pressure signal.

15C.6.4 Safety-Related Supporting System Functions

(1) Electric Power, ESF (Division 1) - Provides electric power to electrical components.

(2) Main Steam System (Division 1 only) - Provides redundant reactor water level, drywell pressure, and reactor pressure signals to the LPCS system for generation of Division 1 LOCA and LPCS System initiation signals.

(3) Primary Containment System - Provides suppression pool water for the LPCS System pumps.

(4) Control Building HVAC - Provides a suitable environment for the system logic and controls.

15C.6.4 Safety-Related Supporting System Functions (Continued)

(5) Auxiliary Building ECCS Pressure Control - Provides a suitable environment for the LPCS System equipment (pump and motor assembly).

15C.6.5 Initiating Events or Signals

(1) LOCA event - Initiates main pump startup, closes the test valve, if open, and opens the minimum flow valve.

(2) LOCA event plus proper (low) reactor pressure - Opens the water injection valve and closes the minimum flow valve.

15C.6.6 Operator Actions Required

None.

15C.6.7 System Description

The LPCS System is a water spray loop consisting of a core spray pump, a sparger ring, spray nozzles, and the necessary piping, valves, and instrumentation for control. The core spray pump takes suction from the suppression pool and sprays the water through the sparger ring into the plenum chamber above the core.

The LPCS System operating in conjunction with the Automatic Depressurization System (ADS), High Pressure Core Spray (HPCS) System, and the Low Pressure Coolant Injection (LPCI), a subsystem of the Residual Heat Removal System, provides adequate reactor cooling to prevent excessive fuel cladding temperatures.

The LPCS System consists of a centrifugal pump, valves, a spray sparger which is located above the reactor vessel core, the

15C.6.7 System Description (Continued)

necessary piping to convey the water from the suppression pool to the sparger, and the associated controls and instrumentation.

The discharge line to the reactor has two isolation valves; one, located inside the drywell, is a testable check valve with an air operator and the other, located outside the containment, is a motor-operated injection valve that is automatically opened following a LOCA and after the reactor pressure has been reduced sufficiently to permit the LPCS System to provide core cooling water.

A main pump discharge bypass line is used to protect the main pump from overheating whenever the injection and test valves are closed. This bypass line which consists of a minimum flow valve and associated orifice and pipe which connects the downstream side of the main pump to the suppression pool is used primarily to bypass the reactor water injection path at LPCS System startup (LOCA signal/condition) of the main (core spray) pump until the reactor vessel pressure is reduced sufficiently to permit opening the injection valve. The minimum flow valve is automatically closed and the injection valve is automatically opened when the reactor vessel pressure is sufficiently reduced.

A test line is used to test the system capability. The "test" valve is automatically closed upon receipt of a LOCA signal if it is open.

Two safety relief valves are placed in the line to relieve (protect) the pipe if and when the reactor pressure should "leak" through the injection valves and the pump and pump discharge check valve.

15C.6.7 System Description (Continued)

To provide a quick reactor vessel water spray injection response, the injection valve is quick opening and the main pump reaches full rated flow rapidly. To enhance the water spray injection the pipe line downstream of the main pump is kept filled with water by using a water fill pump. This also eliminates water hammer on initial core spray injection.

15C.6.8 FMEA Exclusions

The components or subsystems excluded because they do not perform a safety-related or safety-related supporting function (hence, having no effect on safety) are as follows:

(1) P&ID, Figure 15C.6-1

(a) Instrumentation and test point: PI-R001, PI-R002, RO-D003, TX-NN001

(b) Check valve: F009

(c) Manual drain valves: F004, F027, F028, F035, FF103, FF104, FF105, FF106, FF108, FF113, FF116, FF123, FF124, FF125

(2) Elementary Diagram, Figures 15C.6-2a through j

Figures 15C.6-2a through c (reference information only)

Figure 15C.6-2d

(a) All logic that "drives" annunciator lights

(b) Pressure sensor and associated logic that drives an annunciator light

15C.6.8 FMEA Exclusions (Continued)

Figure 15C.6-2e

(a) Valve position, pump condition, and status indicating lights and associated drive logic

(b) Annunciator drive logic

Figure 15C.6-2f

(a) All status lights and associated drive logic or interlock switch

(b) Computer printouts and associated COIs or AISO

Figures 15C.6-2g and h

All (status lights and associated drive logic and interlock switches)

Figure 15C.6-2i

(a) Reference information (Table 1)

(b) Pump discharge flow indicator and XMTR

(c) Status lights and associated interlock switches (F007)

Figure 15C.6-2j

(a) Reference information, i.e., MOV and annunciation schemes and relay information

15C.6.9 Analysis and Results

The results of the FMEAs given in Table 15C.6-1 are summarized as follows:

Number of line items analyzed	48
Number of failure modes analyzed	84
Number classified "A"	21
Number classified "B"	59
Number classified "C"	4

This FMEA verifies that the LPCS System has no single failure that would cause a degradation of its safety-related performance nor cause a degradation of performance to any safety-related supporting system. Hence, the LPCS System performs as designed.

Table 15C.6-1

LOW PRESSURE CORE SPRAY SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.6-1. Low Pressure Core Spray System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.6-2a through j. Low Pressure Core Spray System Elementary Diagram (FMEA)

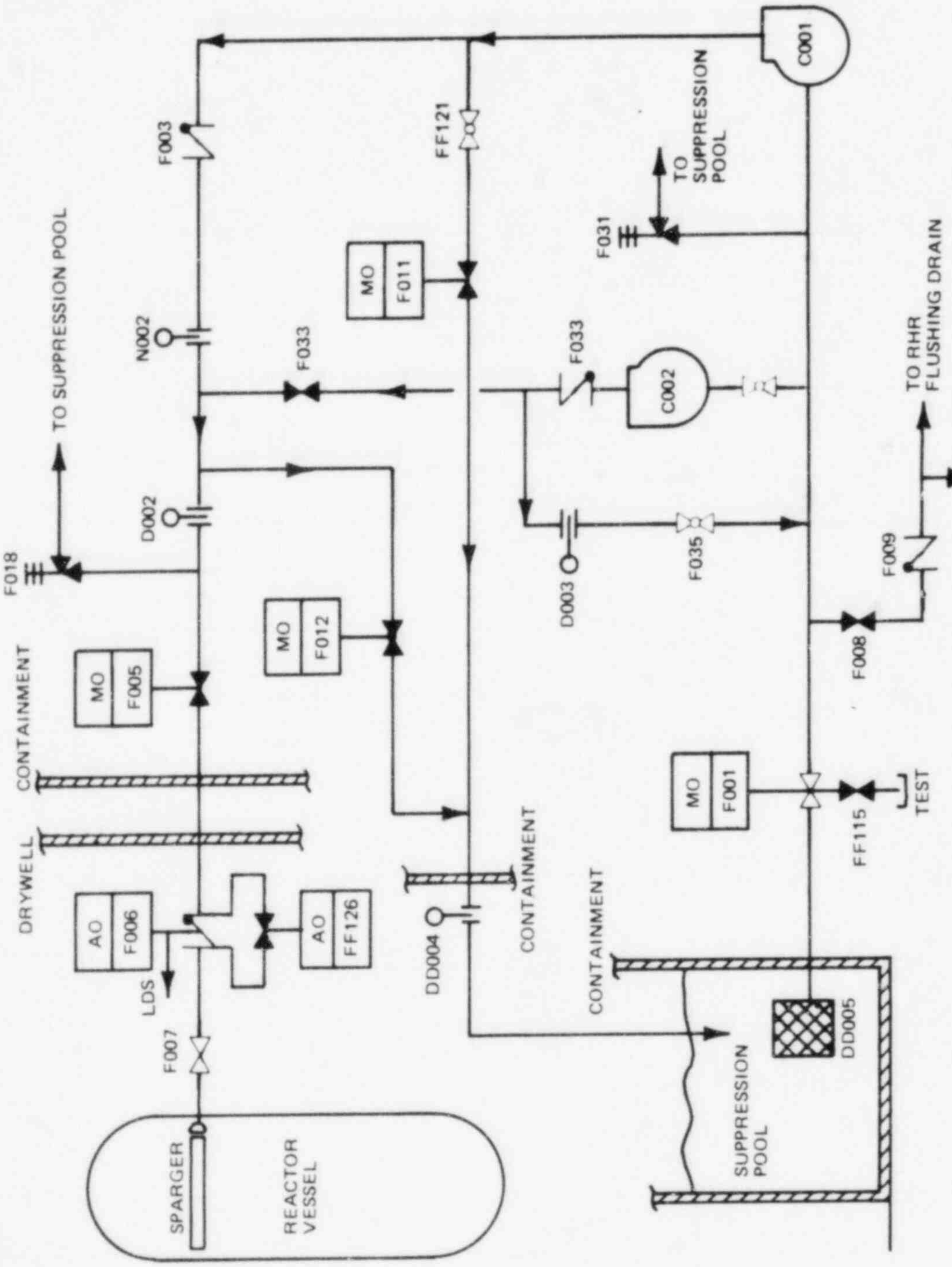


Figure 15C.6-3. Low Pressure Core Spray System Simplified Block Diagram

15C.7 HIGH PRESSURE CORE SPRAY SYSTEM (E22)

15C.7.1 Scope

This FMEA covers all the active components shown on the HPCS System P&ID, and all the mechanical, pneumatic, and electrical devices necessary to permit the HPCS System to perform its safety-related functions on demand.

The mechanical devices include pneumatic, electric motor and manually operated valves, relief valves, check valves, strainers and water pumps. The pneumatic devices include air actuators. The electrical controls include switches, signal transmitters, trip units, optical isolators, logic devices, load drivers, motor control centers and electric motors.

15C.7.2 System Defining Documents

(1) P&ID, Figures 15C.7-1a and b

(2) Elementary Diagram, Figures 15C.7-2a through m

(3) Simplified Block Diagram, Figure 15C.7-3

15C.7.3 System Safety-Related Functions

(1) Provides reactor core cooling following a loss-of-coolant accident (LOCA). The HPCS System operating in conjunction with the Automatic Depressurization System (ADS), Low Pressure Core Spray (LPCS) System and Low Pressure Coolant Injection (LPCI) System provide adequate reactor core cooling to prevent fuel cladding temperatures from exceeding limits for all design basis LOCAs.

15C.7.3 System Safety-Related Functions (Continued)

(2) Supplies makeup water to the reactor vessel in the event of reactor isolation assuming failure of the Reactor Core Isolation Cooling (RCIC) System.

(3) Maintains Primary Containment integrity by providing containment isolation valves.

(4) Provides "HPCS Initiation" signal to the Division 1, 2 and 3 Diesels and Auxiliaries System for an automatic start of the Division 3 Diesel Generator and for blocking of its protective devices.

15C.7.4 Safety-Related Supporting System Functions

(1) AC & DC Power Distribution System, ESF Electrical Buses (Division 3) - Provides electrical power to operate the HPCS System.

(2) Main Steam (Nuclear Boiler) System (Divisions 3 and 4) - Provides low reactor water level and high drywell pressure signals for system initiation, as well as high reactor water level signal to automatically shut off the HPCS System by closing the injection valve.

(3) Auxiliary Building ECCS Area Pressure Control System - Provides suitable environment for the equipment located in the HPCS room.

(4) Demineralized Water and Condensate Distribution System - Provides preferred water source for the HPCS System.

(5) Primary Containment System - Provides alternate water source (suppression pool) for the HPCS System.

15C.7.4 Safety-Related Supporting System Functions (Continued)

(6) Control Building HVAC System - Provides suitable environment for the HPCS components located in the Control Building.

15C.7.5 Initiating Events or Signals

Low reactor water level and high drywell pressure (LOCA) signals from the Main Steam System will result in the automatic initiation of the HPCS System.

15C.7.6 Operator Actions Required

None.

15C.7.7 System Description

The HPCS System consists of pumps, valves, piping, and controls required to deliver coolant to the reactor vessel in the unlikely event a LOCA occurs. The HPCS System takes water from a condensate storage header and delivers it to the reactor vessel through a spray sparger located above the core. Upon depletion of the condensate or high water level in the suppression pool, the HPCS System control logic automatically switches to the suppression pool as the source for HPCS System water. The HPCS is one of three redundant coolant delivery systems of the ECCS network.

The HPCS System supplies sufficient flow following the reactor scram to depressurize the reactor vessel in the event of a LOCA. Following depressurization and depletion of the coolant inventory to below the top of the core, the HPCS System delivers sufficient water spray, through a sparger ring with nozzles located in the plenum chamber above the core, to prevent excessive fuel cladding temperatures.

15C.7.7 System Description (Continued)

If the pipe break is small enough that the reactor inventory is not depleted to below the top of the core following depressurization, the HPCS System supplies makeup coolant.

The HPCS System supplies makeup water to the reactor vessel in the event of reactor isolation and failure of the Reactor Core Isolation Cooling System (RCIC). The makeup water is required to maintain sufficient reactor water inventory since steam generation will continue at a reduced rate, even though the reactor has scrammed, due to the core fission product decay heat.

The HPCS System has its own independent AC & DC electric power sources (Division 3). These sources are not shared by any other ECC System.

The HPCS System is capable of automatic startup upon receipt of an initiation signal. The automatic initiation signals include reactor vessel low water level and high drywell pressure as specified in the data sheet, both utilizing one-out-of-two twice logic. Manual control is possible after automatic initiation.

In addition to the automatic operational features of the system, provisions are included for remote-manual startup, operation, and shutdown.

15C.7.8 FMEA Exclusions

(1) P&ID Figures 15C.7-1a and b

15C.7.8 FMEA Exclusions (Continued)

The following mechanical components were excluded:

Figure 15C.7-1a

(a)	Valves F008, F009, F030, F032, FF101, FF102, FF103, FF107, FF108, FF109, FF110, FF209 and FF210	Normally closed, passive valves which are part of double or triple pressure barrier (one or two valves plus cap or blind flange).
(b)	Valves F026, F033 and F034	Passive, no effect on safety.
(c)	Restriction orifice D003	Because of bore size plugging is not credible failure.
(d)	PI-R001 and TX-NN001	Not required for safety functions.

Figure 15C.7-1b

(a)	Valves F003, F017, F021, F022, F031, and FF113 through FF123	Normally closed, passive valves which are part of double or triple pressure barrier (one or two valves plus cap or blind flange).
(b)	Valves F019, F025 and F036	Passive, no effect on safety.
(c)	Restriction orifices DD006, DD004 and DD002	Because of bore size plugging is not credible failure.

15C.7.8 FMEA Exclusions (Continued)

(d) PI-R002 — Not Required for safety functions.

(2) Elementary Diagram Figures 15C.7-2a through m

Figures 15C.7-2a, b, c and m

Reference information only, all excluded.

Figure 15C.7-2d

All components generating or processing signals used for annunciators are excluded - not required for safety functions.

Figure 15C.7-2e

Status lights and their ILDs are excluded - not required for safety functions.

Figure 15C.7-2f

Status lights and components for annunciation only are excluded - not required for safety functions.

Figure 15C.7-2g

Switches, status ligh[illegible]DSC and ILDs are excluded - not required for safety functions.

Figure 15C.7-2h and i

All excluded (components for annunciation on[illegible] and reference information) - not required for safety functions.

Figure 15C.7-2j

All components are for status monitoring and not required for safety functions - all excluded.

Figure 15C.7-2k

All excluded (components for computer input and indication) - not required for safety functions.

Figure 15C.7-2ℓ

Shown components were considered as part of their respective "Driver" items.

15C.7.9 Analysis and Results

The results of the FMEA given in Table 15C.7-1 are summarized as follows:

Number of line items analyzed	71
Number of failure modes analyzed	130
Number classified "A"	28
Number classified "B"	102

Some single failures identified in this FMEA prevent the HPCS System from performing its safety functions. However, no HPCS System component failure can affect the performance in the balance of the ECCS Network and prevent the ECCS Network from performing its safety functions on demand.

Table 15C.7-1

HIGH PRESSURE CORE SPRAY SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.7-1a and b. High Pressure Core Spray System (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.7-2 a through m. High Pressure Core Spray System Elementary Diagram (FMEA)

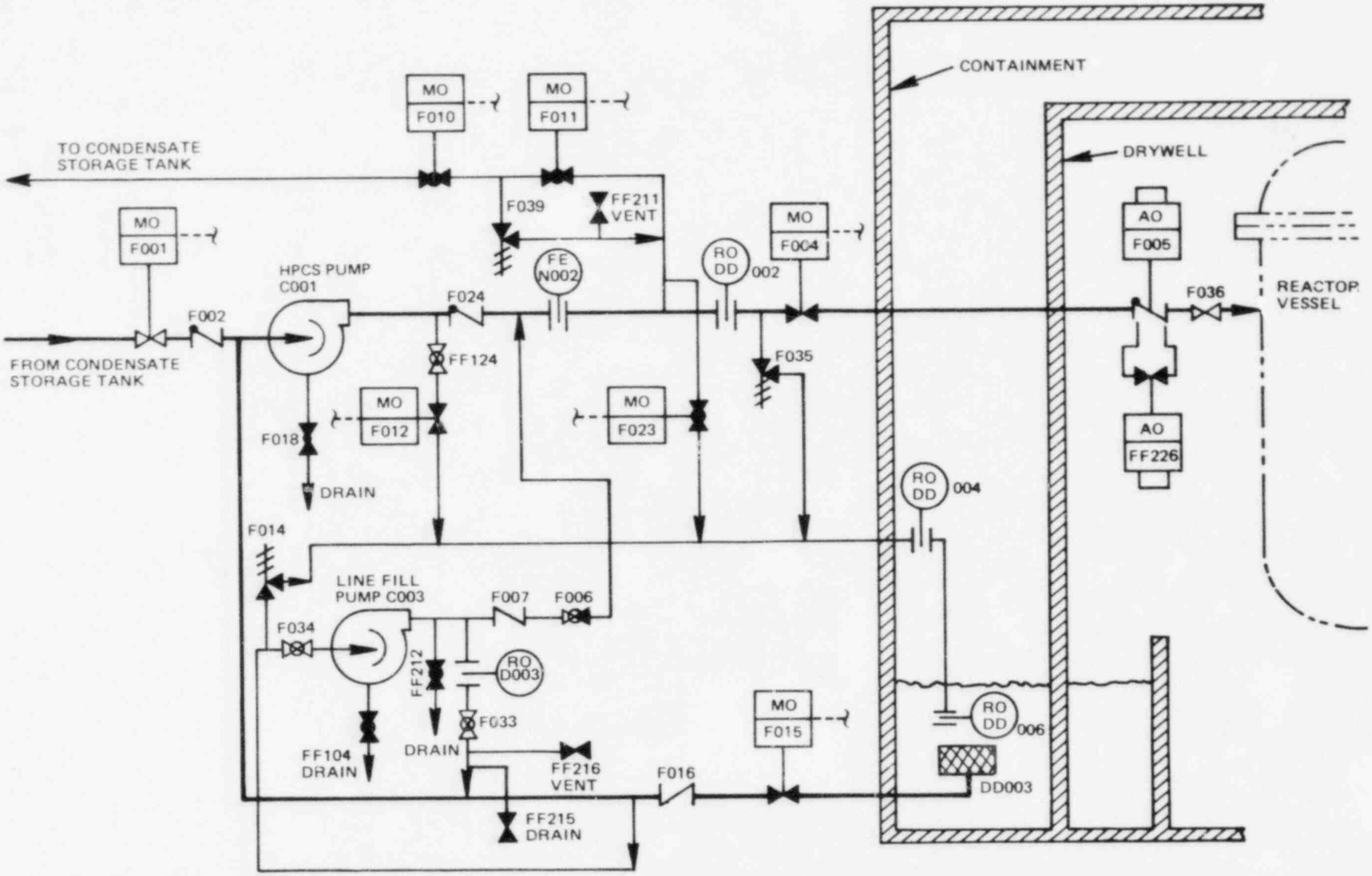


Figure 15C.7-3. High Pressure Core Spray System Simplified Block Diagram

15C.8 MAIN STEAM POSITIVE LEAKAGE CONTROL SYSTEM (E32)

(To be provided in December 1982)

15C.9 REACTOR CORE ISOLATION COOLING SYSTEM

Applicant to provide

15C.10 REACTOR AND FUEL SERVICING/INCLINED FUEL TRANSFER SYSTEM (F42)

15C.10.1 Scope

The FMEA covers all of the active components depicted on the P&ID and all mechanical, pneumatic, and electrical control devices necessary to permit the Reactor and Fuel Servicing/Inclined Fuel Transfer System (RFS/IFTS) perform its safety-related functions on demand.

15C.10.2 System Defining Documents

(1) P&ID, Figure 15C.10-1

(2) Elementary Diagram, Figures 15C.10-2a through z

(3) Simplified Block Diagram, Figure 15C.10-3

15C.10.3 System Safety-Related Functions

(1) Maintains primary containment integrity by installation of the blind flange at the upper end of the transfer tube downstream of the shutoff valve.

(2) The RFS/IFTS shall not fail in such a manner as to cause or interface with other systems in such a way as to cause:

(a) Loss of safety shutdown capability

(b) Exceed dose limits to operating personnel

(c) Exceed dose limits to public

(d) Loss of primary pressure boundary

(3) Safety interlocks are provided to prevent:

(a) Opening the transfer tube bottom valve when the flap valve is open and vice versa to prevent upper pool drainage.

(b) Raising the transfer carriage above the fill/drain position without the transfer tube being filled with water.

(c) Lowering or raising the transfer carriage inside the transfer tube while bottom, flap and fill valves are shut and incapable of operation and the transfer tube drained to the level of the drain line.

(d) Inadvertent cycling of upender hydraulic control valves giving unexpected movement of the upender while loading or unloading fuel.

(e) Lowering of the transfer carriage into the tube bottom valve or raising of the transfer carriage into the tube flap valve.

(f) Access being afforded to the areas containing hazardous radiation levels during fuel transfer operations.

15C.10.4 Safety-Related Supporting System Functions

None.

15C.10.5 Initiating Events or Signals

None.

15C.10.6 Operator Actions and Supporting Systems

Blind flange shall be installed by the operator at the upper end of the transfer tube downstream of the shutoff valve F002 to maintain primary containment integrity.

15C.10.7 System Description

The IFTS is used for the underwater transfer of the fuel assemblies, control rods, or other irradiated items, between the Fuel Building transfer pool and the Reactor Building fuel transfer pool.

The transfer tube provides a sealable, enclosed path for the carrier which is lowered and raised by means of a winch assembly. The position of the carrier in the tube is known by a calibrated position system.

The transfer tube includes a valve at both ends, and is provided with a containment isolation assembly. The valve at the upper end consists of a flap with actuating cylinder mounted on a sheave box. Below the sheave pipe, a shutoff valve provides a means to block the reactor building pool water above the containment isolation assembly. The containment isolation assembly consists of two pipe spools separated by a removable blank flange. When the transfer tube system is placed in operation, the containment isolation assembly is disassembled and the blank flange is replaced with an open-faced gasket, and reassembled. At the intersection of the two transfer tube sections, the drain line branch and the "tube-drained" liquid level sensors are installed. The lower end of the transfer tube terminates with a hydraulically actuated gate valve (bottom valve).

An upender is located on both ends of the transfer tube. The upenders are mounted on pivot arms which permit them to be raised to a vertical position for loading and unloading the carrier. The

15C.10.7 <u>System Description</u> (Continued)

upenders are similar in construction and perform identical functions. Hydraulic power supplies are provided to operate valves and rotate the carrier from its inclined position to the vertical and back again for loading and unloading the fuel. IFTS is operating as follows: A transfer is considered as starting at the lower terminus of the inclined tube while bottom and drain valves are open and flap fill valves are closed. Fuel is lowered into the carrier which is then rotated to the same angle as the inclined tube. The carrier is now permitted to ascend to the tube fill/drain position. At this position bottom and drain valves are closed and fill valve is open and when the tube is full of water the flap valve is opened. If the upper upender is in the inclined position the fuel transfer is permitted to ascend to the upper terminus. The carrier is rotated to the vertical position and fuel exchange is made. With both upenders inclined and the valves in the same condition required for the upward transfer the carrier is permitted to descend to the tube fill/drain position. The flap and fill valves are then closed and drain valve is opened. When the tube has drained to the fill/drain position the bottom valve is opened and the carrier is permitted to descend to the lower terminus. Fuel is placed in and withdrawn from the carrier at each terminus by a refueling machine which incorporates a telescoping grapple.

The control system is operating on a semi-automatic basis with provision made for manual override of certain functions. The system operator has the means to start and stop the control, to raise or lower the load, and rotate the upenders to the vertical and inclined positions. The control panel incorporates a locked door behind which switches provide for manual operation of the bottom valve, the drain valve, the fill valve, and the flap valve. These switches, while permitting actuation of the individual valves, are not bypassing the interlocks which provide for operating the

15C.10.7 System Description (Continued)

valves in the correct sequence. Dual channel logic which provides a backup sensor for each required sensor is used to provide the redundancy necessary for the system to function safely and with the least amount of interruption. The failure of any channel to perform its intended function causes an alarm which identifies the failed channel. It is possible to switch to the monitoring channel to continue operation. However, it is possible to reset the system each time a transfer is made with any channel failed.

15C.10.8 FMEA Exclusions

The IFTS is designated as nuclear non-safety related. However, failure modes of the components depicted on the P&ID and elementary diagrams were analyzed and evaluated to determine what effect the failure would have on the IFTS to allow performance of its safety-related function. The following paragraphs list components excluded because they do not perform a safety-related or safety-related supporting function; hence, they have no effect on safety.

(1) P&ID Figure 15C.10-1

The following mechanical components were excluded. Electromechanical, electrical and control components are addressed on elementary diagram.

Blind flange - passive component.
Upper and lower upenders with hydraulic operators and supply - components not essential for safety.

(2) Elementary Diagram Figures 15C.10-2a through z

Figures 15C.10-2a, b, c and e

Reference information only.

15C.10.8 FMEA Exclusions (Continued)

Figure 15C.10-2d

All items, except master power control, are not essential for safety.

Figures 15C.10-2f through j

All items are not essential for safety.

Figure 15C.10-2k

Indicating lights, switches S18 and S24, and valve F42-F001 proximity sensor are not essential for safety.

Figures 15C.10-2ℓ and m

All items are not essential for safety.

Figures 15C.10-2n and o

Hydraulic power supplies motors and associated controls, accumulator solenoid relief valve and associated controls are not essential for safety.

Figures 15C.10-2p and q

All items are not essential for safety.

Figure 15C.10-2r

Position indicating lights and associated relays are not essential for safety.

15C.10.8 FMEA Exclusions (Continued)

Figure 15C.10-2s

Position indicating lights are not essential for safety.

Figures 15C.10-2t through z

Reference information only.

Programmable controller shown on the Elementary Diagram is not essential for safety.

15C.10.9 Analysis and Results

The results of the FMEA given in Table 15C.10-1 are summarized as follows:

Number of line items analyzed	23
Number of failure modes analyzed	57
Number classified "A"	29
Number classified "B"	14
Number classified "C"	12
Number classified "I"	2

The RFS/IFTS is not a safety system. However, this system provides safety-related functions as to maintain primary containment integrity and prevent public, operating personnel and equipment from radioactive exposure. This analysis has verified that no single failure of an active mechanical component or active or passive electrical component in this system prevents this or any other system from performing its safety function on demand.

Table 15C.10-1

REACTOR AND FUEL SERVICING/INCLINED FUEL TRANSFER SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.10-1. Reactor and Fuel Servicing/Inclined Fuel Transfer System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.10-2a through z. Reactor and Fuel Servicing/ Inclined Fuel Transfer System Elementary Diagram (FMEA)

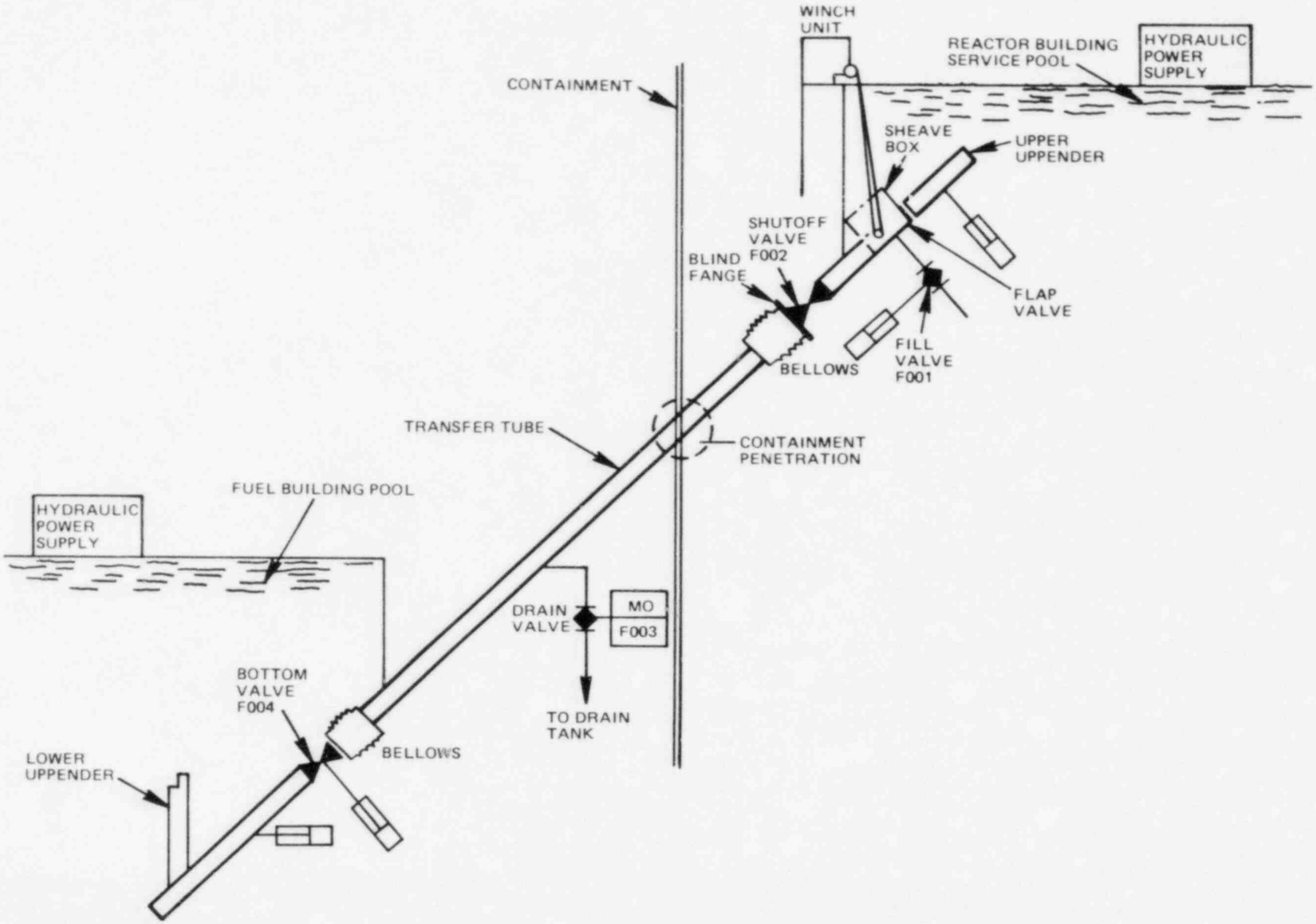


Figure 15C.10-3. Reactor and Fuel Servicing/Inclined Fuel Transfer System Simplified Block Diagram

15C.11 REACTOR WATER CLEANUP SYSTEM (G33/G36)

15C.11.1 Scope

The FMEA covers all the active mechanical components depicted on the P&ID and all the mechanical, pneumatic, and electrical control devices necessary to permit the Reactor Water Cleanup and Filter Demineralizer (RWCU and FD) System to perform its safety-related functions on demand.

The mechanical devices considered in the FMEA include pneumatic, electric motor and manually operated valves, check valves, pressure relief valves, cleanup filters/demineralizers, filters, tanks, heat exchangers, and water pumps. The pneumatic controls include air operator valves. The electrical control devices include load drivers, switches, motor control centers, electric motors, and solenoids.

15C.11.2 System Defining Documents

(1) P&ID, Reactor Water Cleanup System, Figures 15C.11-1a and b

(2) P&ID, Filter Demineralizer System, Figures 15C.11-2a and b

(3) Elementary Diagram, Nuclear Steam Shutoff System, Figures 15C.11-3a through d

(4) Elementary Diagram, Reactor Water Cleanup System, Figures 14C.11-4a through f

(5) Elementary Diagram, Filter Demineralizer System RWCU, Figures 15C.11-5a through q

(6) Simplified Block Diagram, Figure 15C.11-6

15C.11.3 System Safety-Related Functions

(1) Ensures containment, drywell, and reactor pressure boundary integrity and effectiveness of the Standby Liquid Control System by closing the system isolation valves upon receiving isolation signal from the Main Steam (Nuclear Boiler) System.

(2) Provides flow signal from flow elements to the Leak Detection and Isolation System.

(3) Non-safety-related portions of the Reactor Water Cleanup System shall not fail in such a manner as to cause or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown

(b) Exceed dose limits to operating personnel

(c) Exceed dose limits to public

(d) Loss of primary pressure boundary

15C.11.4 Safety-Related Supporting System Functions

(1) AC Distribution System ESF Electrical Buses Division 1, 2, and 3 - Provides electric power to isolation valves motor operator.

(2) Main Steam (Nuclear Boiler) System - Provides Division 1 and 2 isolation signals for closing the RWCU System isolation valves.

(3) Water Positive Seal (WPS) Isolation Valve Leakage Control System - Supplies Division 1 and 2 sealing water to ensure containment integrity in post-LOCA operation.

15C.11.4 Safety-Related Supporting System Functions (Continued)

(4) Control Building HVAC (Divisions 1 and 2) - Provides suitable environment for system valve and control logic.

(5) Leak Detection and Isolation System (Divisions 1 and 2) - Provides sensing and monitoring instrumentation for leak detection and generates a signal in case of a leak [this signal goes to the Main Steam (Nuclear Boiler) System].

15C.11.5 Initiating Events or Signals

(1) Either Division 1 or 2 isolation signals from the NSSS System will close the RWCU System isolation valves.

(a) Actuation of the Standby Liquid Control (SLC) pump for boron injection

(b) High RWCU differential flow

(c) High steam tunnel temperature

(d) High ambient temperature within the RWCU pump room

(e) Low reactor vessel water level

(2) Division 1 isolation signal from the NSSS will close the RWCU System isolation valve.

(a) High reactor water temperature exiting the nonregenerative heat exchanger

15C.11.6 Operator Actions Required

None.

15C.11.7 System Description

The RWCU and FD System is classified as a Primary Power Generation System but not considered as an Essential Safety System. The primary function of the RWCU and FD System is to maintain reactor primary coolant water quality within specified limits by removing soluble and insoluble impurities during all modes of operation. The system also provides a means for reducing the secondary source of beta and gamma radiation resulting from activated corrosion and fission products in the reactor primary system.

Water is removed from the reactor through the reactor recirculation pump suction line and returned through the feedwater line via the Residual Heat Removal System. Under normal operation, the water is removed at reactor temperature and pressure and pumped through regenerative and nonregenerative heat exchangers where it is cooled, and then through the filter-demineralizer units. The flow continues through the shell side of the regenerative heat exchanger where it is heated before returning to the reactor.

The Reactor Pressure Vessel (RPV) water level is controlled by routing primary coolant to the Main Condenser/Radwaste via the RWCU and FD System during all modes of reactor operation.

The RWCU and FD System provides circulation to minimize thermal gradients within the recirculation piping and RPV during periods when the reactor is hot and the main reactor recirculation pumps are not available for service.

15C.11.8 FMEA Exclusions

(1) P&ID, Figures 15C.11-1a and b and Figures 15C.11-2a and b

The following mechanical components are excluded from the FMEA sheets. They do not perform a safety-related function or a safety-related supporting function; hence, they have no effect on safety. Electromechanical, electrical, and control components are addressed in Elementary Diagrams.

(a) P&ID Figures 15C.11-1a and b

Figure 15C.11-1a

Except for isolation valves F001 and F004, all components are passive devices with no impact on safety and are excluded.

Figure 15C.11-1b

Except for isolation valves F028, F034, F039, F040, F053, and F054, all components are passive devices with no impact on safety and are excluded.

(b) P&ID Figures 15C.11-2a and b

Figure 15C.11-2a

Except for isolation valves FF203, FF235, and FF238, all components are passive devices with no impact on safety and are excluded.

15C.11.8 FMEA Exclusions (Continued)

Figure 15C.11-2b

All components are passive devices with no impact on safety and are excluded.

(2) Elementary Diagrams - Figures 15C.11-3a through d, Figures 15C.11-4a through f, and Figures 15C.11-5a through q.

The following electrical/electronic components are excluded from the FMEA sheets. Some components are excluded because they do not perform a safety-related function or a safety-related supporting function; hence, they have no effect on safety. Other components are excluded because they belong to different systems other than RWCU and FD System and are analyzed in FMEAs on their systems.

(a) Elementary Diagram (Figure 15C.11-3a through d)

Figure 15C.11-3a

B5 and D5 are analyzed in FMEA on Residual Heat Removal System.

L5, D12, F12, J12, and K12 are analyzed in FMEA on Main Steam (Nuclear Boiler) System.

All isolation valve indicator lights are not required for safety functions.

15C.11.8 FMEA Exclusions (Continued)

Figure 15C.11-3b

D4 and J4 are analyzed in FMEA on Residual Heat Removal System.

F4 is analyzed in FMEA on Main Steam (Nuclear Boiler) System.

K4 is analyzed in FMEA on Fuel Pool Cooling and Cleanup System.

All isolation valve indicator lights are not required for safety functions.

Figure 15C.11-3c

B5, B12, C12, F12, and J12 are analyzed in FMEA on Residual Heat Removal System.

J5 and L5 are analyzed in FMEA on Fuel Pool Cooling and Cleanup System.

E12 and K12 are analyzed in FMEA on Reactor Recirculation System.

All isolation valve indicator lights are not required for safety functions.

Figure 15C.11-3d

C5, E5, C12, and E12 are analyzed in FMEA on Leak Detection System.

15C.11.8 FMEA Exclusions (Continued)

All isolation valve indicator lights are not required for safety functions.

(b) Elementary Diagram, Figures 15C.11-4a through f

All components are not required for safety functions.

(c) Elementary Diagram, Figures 15C.11-5a through q

All components are not required for safety functions.

15C.11.9 Analysis and Results

The results of the FMEA given in Table 15C.11-1 are summarized as follows:

Number of line items analyzed	22
Number of failure modes analyzed	38
Number classified "A"	16
Number classified "B"	22

This FMEA verifies that no single failure of active mechanical and active or passive electrical component in the RWCU and FD System will prevent this system or any other system from performing their safety-related functions on demand.

Table 15C.11-1

REACTOR WATER CLEANUP SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.11-1a and b. Reactor Water Cleanup System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.11-2a and b. Filter Demineralizer System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.11-3a through d. Nuclear Steam Supply Shutoff System Elementary Diagram (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.11-4a through f. Reactor Water Cleanup System Elementary Diagram (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.11-5a through q. Filter Demineralizer System RWCU Elementary Diagram (FMEA)

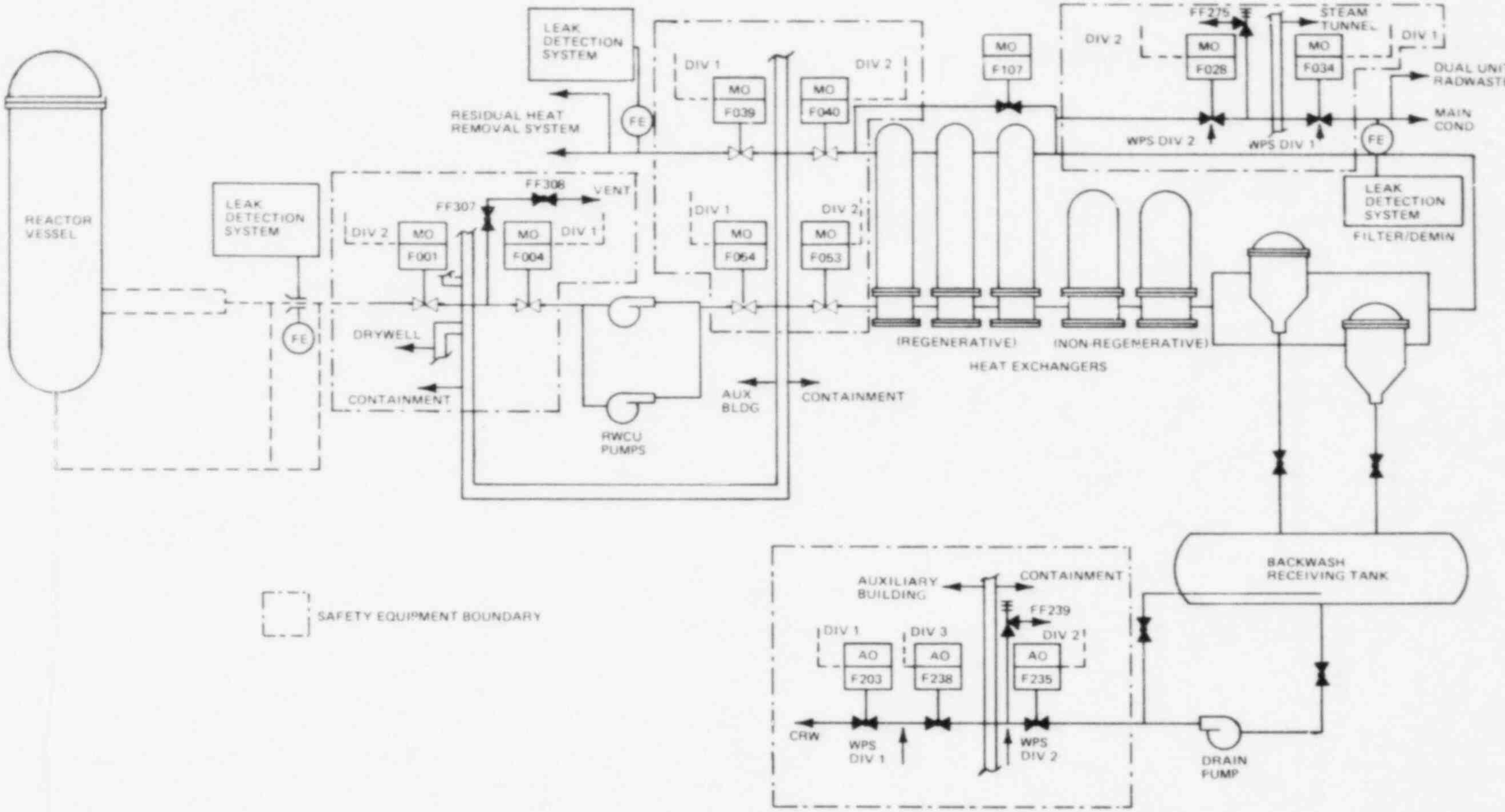


Figure 15C.11-6. Reactor Water Cleanup System Simplified Block Diagram

15C.12 STANDBY GAS TREATMENT SYSTEM (P38)

Applicant to provide

15C.13 ESSENTIAL SERVICE WATER SYSTEM (P41)

15C.13.1 Scope

The FMEA covers all active components depicted on the system P&ID and the mechanical and electrical control devices necessary to permit the Essential Service Water (ESW) System to perform its safety-related functions.

The supply header pumps are BOP equipment and are not included in this FMEA. The mechanical devices analyzed include manual, motor operated, check, and safety/relief valves and flow measuring instruments.

15C.13.2 System Defining Documents

(1) P&ID, Figures 15C.13-1a and b

(2) Elementary Diagram, Essential Service Water System, Figures 15C.13-2a through j

(3) Elementary Diagram, Residual Heat Removal System, Figures 15C.13-3a through d

(4) Simplified Block Diagram, 15C.13-5a and b

15C.13.3 System Safety-Related Functions

(1) The ESW System performs cooling functions for Divisions 1 and 2 essential safety-related equipment, as shown in Table 15C.13-1. Two completely independent trains, mechanically and electrically separated, are provided.

(2) Provides water leak detection and isolation capability for ensuring the integrity of the essential safety portion of the ESW System by separating the essential part of the service water system from the nonessential part (which is a subsystem) during a LOCA, or, in case of a leak, in the nonessential portion.

(3) Provides isolation valves on ESW line penetrating primary containment for ensuring primary containment isolation.

(4) Provides secondary containment boundary integrity by means of loop seals on ESW System non-safety lines penetrating secondary containment boundary.

(5) Provides a qualified source of water for the fuel storage pool emergency makeup.

(6) Provides a qualified source of water for the Water Positive Seal (WPS) System.

(7) The non-nuclear safety-related portions of the ESW System shall not fail in such a manner as to cause or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown capability

(b) Exceed dose limits to the operating personnel

(c) Exceed dose limits to the public

(d) Loss of primary pressure boundary

15C.13.4 Safety-Related Supporting System Functions

(1) AC & DC Power Distribution System (Divisions 1 and 2) - Provides electrical power for each divisional instrumentation, controls and MO valve actuators.

(2) Residual Heat Removal (RHR) System (Divisions 1 and 2) - Provides isolation signals for system containment isolation valves.

(3) BOP Pumping Station (Divisions 1 and 2) - Provides a pressurized source of water supply for the ESW System.

15C.13.5 Initiating Events or Signals

(1) A LOCA event will generate an ECCS "Initiation" signal which is fed to the ESW via the RHR-A Auto/Manual Initiation signal path for Division 1 isolation valves and via RHR-B/C Auto/Manual Initiation signal path for Division 2 isolation valves.

(2) A high differential flow rate between supply and return lines to the non-safety portion of the ESW will also isolate the nonessential portion of the ESW.

15C.13.6 Operator Actions Required

Following a LOCA, when the H_2 mixing function is required, the operator must establish ESW flow to the mixing blower by opening two MO valves via RMS and one LC manually operated valve.

15C.13.7 System Description

The ESW System consists of two separate systems, Division 1 and Division 2. The Balance of Plant (BOP) portion of the service water systems will be provided by the applicant, including water sources and supply pumps. The BOP portion is not included in this analysis.

The service water system distributes cooling water under all operating modes and during shutdown to remove heat from plant auxiliaries. It serves equipment used for normal plant operation and normal or emergency reactor shutdown, as well as those auxiliaries whose operation is desired but not essential to safe shutdown.

A listing of the equipment (subsystems) cooled by the ESW is shown in Table 15C.13-1 depicting both essential and nonessential equipment and the division to which the equipment belongs. Equipment appearing in only one division is serviced by that division of the

ESW. Table 15C.13-1 also shows whether cooling water flows during normal or post-accident operation or under both conditions.

The ESW System is designed to perform its required cooling function following a postulated loss-of-coolant accident (LOCA), assuming a single active mechanical failure or single active or passive electrical failure. In order to meet this requirement, the ESW System provides two complete trains mechanically and electrically separated. In case of a failure which disables Division 1 or 2, the other division operating in conjunction with the HPCS SW system will meet plant safe shutdown requirements, including a LOCA or a loss of offsite power, or both. ESW Division 1 equipment is supplied from Division 1 electric power and control, and ESW Division 2 from Division 2 power and control.

The nonessential parts of the ESW System are not required for safe shutdown or accident mitigation and, hence, are not safety systems. Isolation valves separate the ESW System from the nonessential subsystem during a LOCA, in order to assure the integrity and safety functions of the safety-related parts of the system. Nonessential parts of the ESW System should be operated during all other modes, including the emergency shutdown following loss of preferred power (LOPP).

Instrumentation is provided to detect significant leakage in the nonessential subsystem. The water flow is measured in both entrance and exit pipes. Any significant leakage shows up as a difference between the two flow measurements. A differential flow switch detects leakage and isolates this subsystem, thus assuring continued operability of the essential services.

During normal operation, service water flows through all the Division 1 and 2 equipment except the RHR heat exchangers.

During all plant operating modes, at least one service water pump will be normally operating in both Division 1 and 2. Therefore, if a LOCA occurs, the ESW Systems required to shut down the plant safely will already be in operation.

In the event that the break in the primary coolant boundary cannot be isolated, the ESW System provides a means of flooding (through the RHR system via F094 & F096) the reactor, drywell, and containment with water permitting fuel removal and cleanup of the plant. This is a post-accident recovery program function and is not a safety function.

Connections to the Process Radiation Monitoring System are provided on the RHR exchangers discharge and Division 1 and 2 return headers to detect and alarm radioactive contamination resulting from a tube leak in one of the RHR exchangers, CCW exchangers or fuel pool exchangers.

Isolation valves for RHR heat exchangers, and nonessential service water subsystems on Divisions 1 and 2 are provided with remote manual switches and indication on the remote shutdown panel.

All equipment utilizes either globe or butterfly valves to give the capability for manual control. These valves are accessible downstream of the equipment for regulation of flow through the equipment or for balancing the circuits. The isolation valves to the nonessential service water system are automatically and remote-manually operated.

Pressure taps or indicators at equipment are provided to enable the operator to adjust the differential pressure across each heat exchanger or cooler and also to allow leak checking. Maximum pressure drop for individual pieces of equipment is specified in the plant requirements documents.

Locally mounted temperature indicators or test wells are furnished on the equipment cooling water discharge lines to enable verification of specified heat removal during plant operation.

15C.13.8 FMEA Exclusions

Failure modes of all the components depicted on the P&ID and elementary diagrams were analyzed and evaluated to determine what effect the failure would have on the ESW System as well as any interfacing safety-related or safety-related supporting system. The following components or subsystems are excluded from the FMEA since the failure of any one of these does not affect the safety-related function(s) of the essential service water.

(1) P & ID, Figures 15C.13-1a and b

The following mechanical components were excluded (electromechanical, electrical and control components are addressed on elementary diagrams):

Figure 15C.13-1a

FF031, FF058A, FF059A, FF067, FF068, FF069A, FF070A, FF073A FF141, FF143, FF148, FF180, FF183, FF186, FF197, FF198, FF203A, FF204A, FF207A, FF237, FF238, FF245A, FF246A, FF247A, FF248, FF249, FF250, FF260	Passive, manually operated vent, drain and test valves NO or NC performing no safety function.
FF001A, FF002A, FF003A, FF004A, FF005A, FF006A, FF007A, FF013A, E12-F014A, FF026A, FF027A, FF028, FF030, FF033, FF035, FF036A, FF038A, FF042A, FF044A, FF045A, E12-F094, E12-F096, FF100A, FF101A, FF102A, FF103A, FF104A, FF131A, FF132A, FF133A, FF137, FF139, FF145, FF146, FF184, FF185	Passive, manually operated in-line valves NO or NC set at a fixed position and not changed during normal operation or after a LOCA event.

Figure 15C.13-1b

FF032, FF058A, FF059B, FF069B, FF070B, FF073B & C, FF074, E12-F095, FF124, FF125, FF126, FF127, FF128, FF129, FF142, FF144, FF152, FF161, FF162, FF163, FF164, FF168, FF173, FF174, FF175, FF176, FF177, FF178, FF179, FF188, FF196, FF199, FF200, FF201, FF202, FF203B, FF204B, FF207B, FF210, FF212, FF213, FF214, FF241, FF242, FF245B, FF246B, FF247B, FF251, FF252, FF253, FF254, FF257, FF258, FF259, FF269	Passive, manually operated vent, drain and test valves NO or NC performing no safety function.
FF001B, FF002B, FF003B, FF004B FF005B, FF006B, FF007B, FF013B, E12-F014B, FF026B, FF027B, FF036B, FF038B, FF042B, FF044B&C, FF045B&C, FF046, FF048, E12-F094, E12-F096, FF100B, FF101B, FF102B, FF103B, FF104B, FF131B, FF132B, FF133B, FF138, FF140, FF149, FF150, FF153, FF154, FF155, FF156, FF157, FF158, FF159, FF160, FF165, FF166	Passive manually operated in-line valves NO or NC set at a fixed position and not changed during normal operation or after a LOCA event.

The following valves are excluded from the FMEA due to the passive nature of their operation during normal plant operation or following a LOCA event. These components are activated during abnormal conditions.

Figure 15C.13-1a

FF077A, FF085A, FF086A, FF147, FF187	Passive relief valve not required for safety function.

Figure 15C.13-1a (Continued)

FF215, FF126, FF217, FF218	Passive vacuum breaker is required only following a pipe break.
FF130A	Passive check valve.

Figure 15C.13-1b

FF077B, FF085B&C FF086B, FF151, FF167	Passive relief valve not required for safety function.
FF130B	Passive check valve.

The following safety equipment are cooled by the ESW System but are not part of this system.

Figure 15C.13-1a

E12-B001A&C, E12-C002A, G41-BB001A, P45-ZZ001A, R43-S001A, X63-BB002A, X63-BB010A, X63-BB0011A, X63-BB015, X73-ACU03, X73-ACU05, X73-BB003, X73-BB004, X73-BB006	Essential equipment, not part of ESW System.

Figure 15C.13-1b

E12-B001B&D, E12-C002B&C, G41-BB001B, P45-ZZ001B, R43-S001B, T41-CC008A&B, V41-ACU02, X63-BB002B, X63-BB010B, X63-BB011B,	Essential equipment, not part of ESW System.

Figure 15C.13-1b (Continued)

X73-ACU04, X73-BB007,
X73-BB008, X73-BB017

All of the following instrumentation are excluded.

Figure 15C.13-1a

E12-TE-N003A E12-TE-N005A E12-FE-N006A E12-FT-N007A E12-FI-R602A	This instrumentation, part of RHR System.
TX-NN001A, FE-NN002A, TX-NN004A, TX-NN013A, TX-NN014, TX-NN015, FE-NN019A, FE-NN021A, TX-NN023A, TX-NN024A, TI-RR001A, FI-RR004A, TI-RR005A, PI-RR006A, TI-RR008A, PI-RR009A, dPI-RR010A, TI-RR011A, TI-RR021A, TI-RR022A, dPI-RR028, TI-RR029, dPI-RR034, TI-RR035	Local instrumentation only, not required for safety function.

Figure 15C.13-1b

E12-TE-N003B E12-TE-N005B E12-FE-N006B E12-FT-N007B E12-FI-R602B	This instrumentation, part of RHR System.

Figure 15C.13-1b

TX-NN001B, FE-NN002B, TX-NN004B, TX-NN012, TX-NN013B, FE-NN019B, FE-NN021B, TX-NN023B, TX-NN024B, TX-NN025A&B, TX-NN026A&B, TI-RR001B, FI-RR004B, TI-RR005B, PI-RR006B TI-RR008B, PI-RR009B, dPI-RR010B, TI-RR011B, TI-RR021B, TI-RR022B, dPI-RR030, TI-RR031, dPI-RR032, TI-RR033	Local instrumentation only, not required for safety function.

(2) Elementary Diagram (Essential Service Water System), Figures 15C.13-2a through j

Figure 15C.13-2a

All. (Reference Material only)

Figures 15C.13-2b and d

(a) Indicating lights, associated resistors, ILDs, PLOIs, CIOIs, SLMEs, and logic (inverters). (Lights are not required for safety function.)

(b) Annunciator/computer printout, associated SLMEs, PLOIs, TLOIs, CAOIs, SLTDs and logic (AND/OR gates inverters, etc.) (Annunciator and printout are not required for safety function.)

(c) PT-NN016A (B), PI-RR600A (B), and associated SRUs. (Instrumentation is not required for safety function.)

Figures 15C.13-2c, e and f

All. (Status lights and alarm logic are not required for safety function.)

Figure 15C.13-2g

All. (Non-safety components)

Figure 15C.13-2h

All. (Annunciator is not required for safety function.)

Figure 15C.13-2i

Valves FF015 A&B and FF049A&B and all associated components. (Nondivisional component is not required for safety function.)

Figure 15C.13-2j

All. (MOV test control and valve overload on power loss annunciator information only.)

(3) Elementary Diagram (Residual Heat Removal System), Figures 15C.13-3a through d

Figures 15C.13-3a and c

Valve E12-F024 A&B and all associated components. (Safety equipment, but not a part of this system.)

Figures 15C.13-3b and d

a. Valves E12-F027 A&B, E12-F028 A&B and E12-F064A, B&C Flow Instrumentation E12-N052C and all associated components. (Safety equipment, but nor a part of this system.)

b. Indicating lights. (Lights are not required for safety function.)

c. Containment spray initiation and associated ACLDs. (Safety function, but not a part of this system.)

15C.13.9 Analysis and Results

The FMEA forms consist essentially of three parts. The first addresses the mechanical devices and the mechanical aspects of the valves and the second part addresses the electrical-logic devices, logic, and electrical control of the motor-operated valves. The devices, generally, are by system flow of cooling water per Figures 15C.13-19 and 15C.13-20. The electrical-logic devices follow the mechanical order and are grouped as indicated on the elementary diagram sheets by location coordinates at the center of the grouping. The third section addresses operator actions and safety-related supporting systems.

The results of the FMEA, given in Table 15C.13-2, are summarized as follows:

Number of line items analyzed	30
Number of failure modes analyzed	57
Number classified "A"	11
Number classified "B"	34
Number classified "C"	12

Data in parentheses in the FMEA sheets are for corresponding equipment in another division, i.e., Division 1 equipment is described while the corresponding Division 2 nomenclature and location is contained in parentheses.

This FMEA verifies that no single failure of an active mechanical and active or passive electrical component in the ESW System will prevent the accomplishment of its safety-related functions.

Failures within the nonessential portions of the ESW System will not cause loss of safe shutdown capability or loss of primary pressure boundary and will not cause radiation exceeding dose limits to operating personnel or the public.

Table 15C.13-1

EQUIPMENT COOLED BY ESSENTIAL
SERVICE WATER SYSTEM

	Divisional Equipment	Service Water Flow Division 1 Normal Oper	Division 1 Post Accident	Division 2 Normal Oper	Division 2 Post Accident
	Essential				
1	Fuel Pool Cooling Hx	Yes	Yes	Yes	Yes
2	Diesel Generator Hx	Yes	Yes	Yes	Yes
3	LPCS Pump Room Cooler	Yes	Yes	NA	NA
4	FPCCU Pump Room Cooler	Yes	Yes	Yes	Yes
5	RHR Pump Room Cooler, A or B	Yes(A)	Yes(A)	Yes(B)	Yes(B)
6	RHR Pump Seals, A or B	Yes(A)	Yes(A)	Yes(B)	Yes(B)
7	H_2 Mix Blowing Sys Air and Oil Coolers	No	Yes	No	Yes
8	RCIC Pump Room Cooler	Yes	Yes	NA	NA
9	Control Bldg Chiller	Yes	Yes	Yes	Yes
10	SGTS Room Cooler	Yes	Yes	Yes	Yes
11	RHR Hx	No	Yes	No	Yes
12	RHR Pump Room Cooler, C	NA	NA	No	Yes
13	RHR Pump Seal, C	NA	NA	No	Yes
14	Shield Ann Exh Fan Room Cooler	Yes	Yes	Yes	Yes
15	AB Self-Cont AC Unit	Yes	Yes	Yes	Yes
16	Remote Shutdown Area Air Unit (when required)	Yes	Yes	NA	NA
17	Seal Air Compressor Cooler	Yes	Yes	Yes	Yes
18	Seal Air Compressor Room Cooler	Yes	Yes	Yes	Yes
	Nonessential				
19	RW Bldg Self-Cont AC Unit	NA	NA	Yes	No
20	Radwaste Evap Condenser	Yes	No	Yes	No
21	RI CCW Hx	Yes	No	Yes	No
22	Steam Tunnel Cooler	Yes	No	Yes	No
23	RI Chiller	Yes	No	Yes	No
24	DW Chiller	Yes	No	Yes	No

Table 15C.13-2

ESSENTIAL SERVICE WATER SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.13-1a and b. Essential Service Water System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.13-2a through j. Essential Service Water System Elementary Diagram (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.13-3a through d. Residual Heat Removal System Elementary Diagram (FMEA)

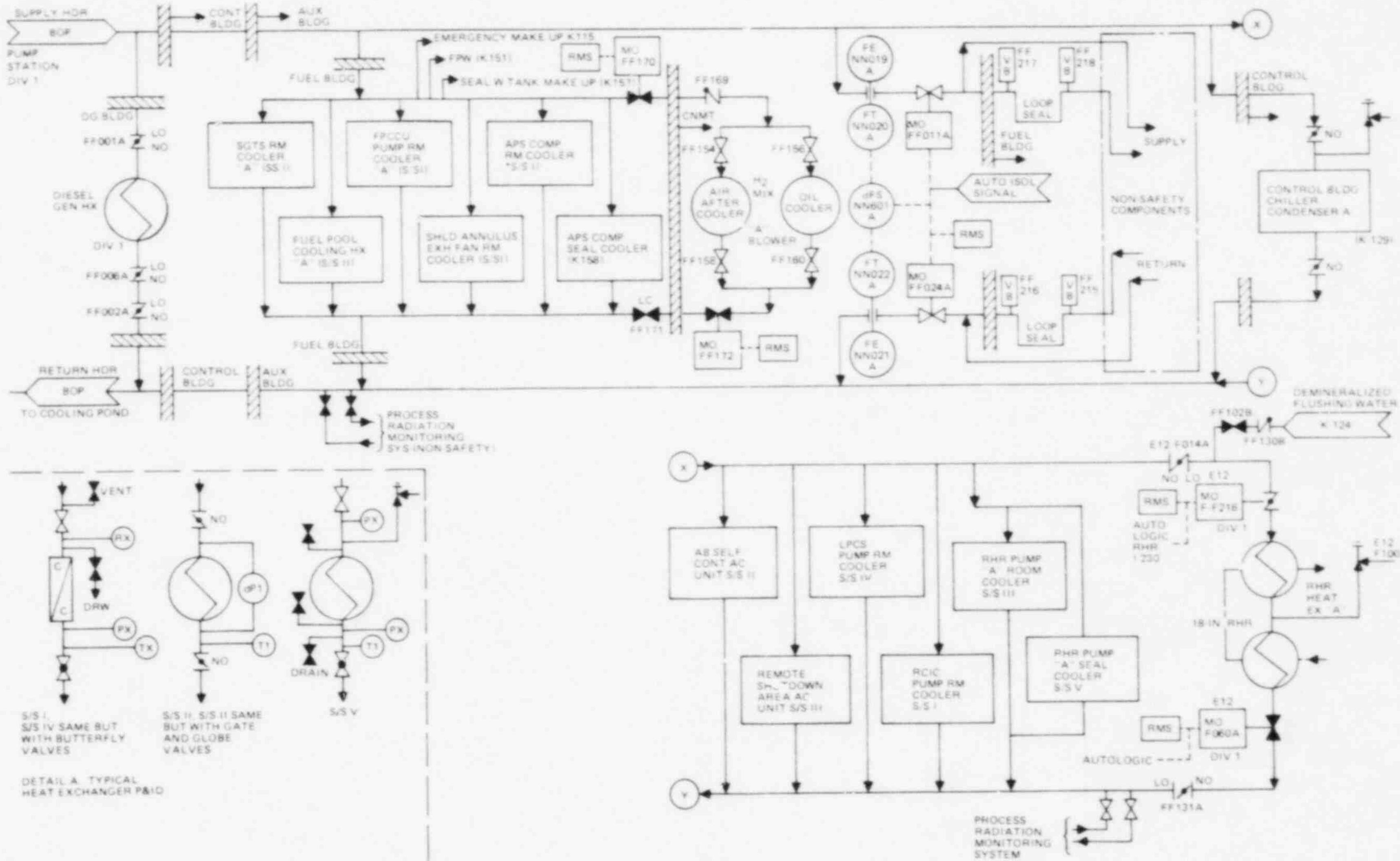


Figure 15C.13-4a. ESWS-Division 1 Simplified Block Diagram

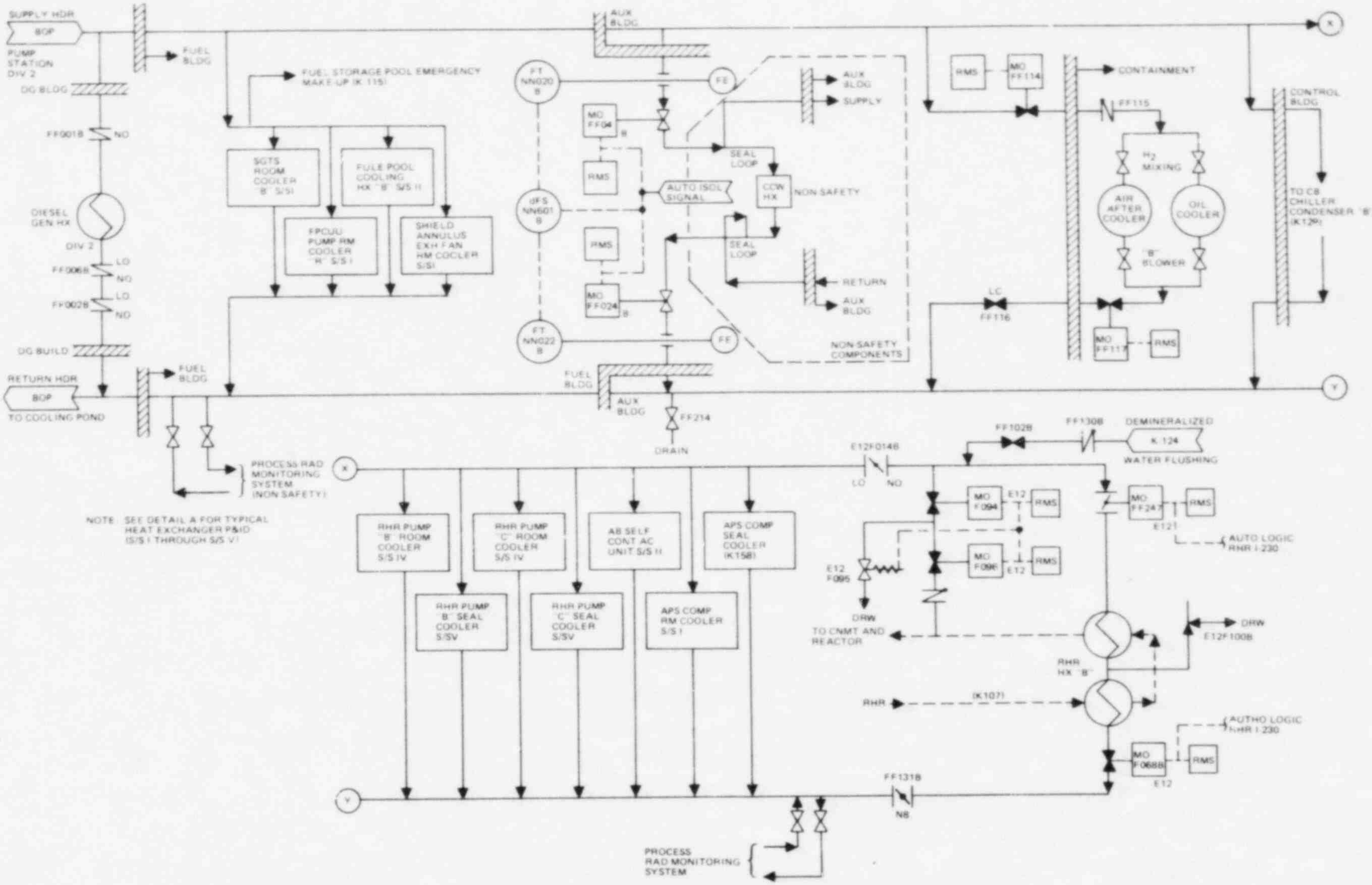


Figure 15C.13-4b. ESWS-Division 2 Simplified Block Diagram

15C.14 CONTROL BUILDING CHILLED WATER SYSTEM (P45)

(To be provided in December 1982)

15C.15 CONDENSATE AND DEMINERALIZED WATER DISTRIBUTION (P46)

15C.15.1 Scope

This FMEA covers the Demineralized Water and Condensate Distribution Systems (DW & CDS) active components shown on the system P & ID and all mechanical and electrical devices necessary to permit the system to perform its safety-related functions on demand.

Mechanical devices include air-operated and manually operated valves. The electrical controls of the system include the manual switches, transducers, signal transmitters, inverters, isolators, logic and device load drivers.

15C.15.2 System Defining Documents

(1) P & ID, Figures 15C.15-1a through c

(2) Elementary Diagram, Figures 15C.15-2a through d

(3) Simplified Block Diagram, Figure 15C.15-3

15C.15.3 System Safety-Related Functions

(1) Division 1 and 2 isolation valves ensure primary containment isolation on receipt of an isolation signal.

(2) Division 2 isolation valves ensure secondary containment isolation on receipt of an isolation signal.

(3) Normally locked closed manual valve in series with a check valve provides drywell integrity.

(4) System provides storage provisions of a 7,000-gallon surge volume of condensate in the piping to the RCIC and HPCS

Systems which allows for automatic switchover to the suppression pool upon loss of water supply from the condensate storage tank.

(5) Non-safety parts of the DW & CDS shall not fail in such a manner as to cause or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown capability

(b) Exceed dose limits to the operating personnel

(c) Exceed dose limits to the public

(d) Loss of primary pressure boundary

15C.15.4 Safety-Related Supporting System Functions

(1) Division 1 and 2 Engineered Safety Feature (ESF) Buses - Provides electrical power for the electrically operated components.

(2) Main Steam System provides Division 1 and 2 - LOCA signals for closing primary containment isolation valves.

(3) Control Building HVAC - Provides suitable environment for the system logic and control equipment.

(4) Water Positive Seal Isolation Valve Leakage Control System - Provides water positive seal for system isolation valves during and after LOCA.

(5) Standby Gas Treatment System - Provides a secondary containment isolation signal (through the Auxiliary Building ECCS Area Pressure Control System).

15C.15.5 Initiating Events or Signals

(1) LOCA Signal: Division 1 and 2 LOCA signals initiate closing of the primary containment isolation valves.

(2) LOCA Signal: Division 2 LOCA signals initiate closing of the secondary containment isolation valves.

15C.15.6 Operator Actions Required

None.

15C.15.7 System Description

The Demineralized Water and Condensate Distribution System consists of necessary piping, manually operated valves, pump and solenoid valves, air-operated isolation valves, and associated instrumentation and controls. Figure 15C.15-3 is a simplified block diagram of the Demineralized Water and Condensate Distribution System. The Demineralized Water Distribution System provides reactor quality water for preoperational tests, startup, and normal operation of several systems in the nuclear island. The Condensate Distribution System provides process water for the RCIC, HPCS, and CRD Systems. The Condensate Distribution System also provides makeup water for plant equipment and flushing water for various systems as well as preferred source of water for fire protection in the containment building. The Reactor Island demineralized water booster pump provides normal water makeup to the Closed Cooling Water (CCW) expansion tank at 101 feet elevation and CRD maintenance area test water at the required pressure of 200 psig. A 7,000-gallon surge volume is provided in the piping from the condensate tank to the RCIC and HPCS systems for automatic switchover to the suppression pool upon loss of water supply from the condensate storage tank. The automatic switchover is achieved through a level alarm provided with the 7,000-gallon surge volume.

Two air-operated isolation valves are provided for both demineralized water and condensate pipe penetrations through primary and secondary containments. The condensate line to the drywell is provided with a normally locked closed manual isolation valve in series with a check valve. This line is used only during drywell maintenance.

Normally open air-operated isolation valves receive isolation signals from their respective divisions. The Water Positive Seal Isolation Valve Leakage Control System provides a water positive seal for isolation valves during and after a LOCA condition.

Water loop seals are provided for any pipe penetrating the secondary containment boundary (Fuel Building, ECCS Room, etc.). The BOP portion of the system has the following features. The minimum water level of the condensate storage tank is 150,000 gallons and the takeoff below this level is for the RCIC and HPCS system pumps only. Demineralized water and condensate water quality is continuously monitored by conductivity and pH measuring devices which indicate and alarm on the local and control room panels. Also, the silica content is continuously monitored and recorded.

15C.15.8 FMEA Exclusions

The components or subsystem which are excluded from this FMEA because they do not perform a safety-related or safety-related supporting function and have no effect on safety are as follows:

(1) P & ID Figures 15C.15-1a through c

The following mechanical components were excluded: electromechanical, electrical and control components are addressed on elementary diagrams.

The following in-line valves are normally closed (NC) and are set at the closed position before normal plant operation. Thereafter, the valve position is not changed either during normal operation or after a LOCA event and are considered passive devices. Therefore, they are excluded from the FMEA analysis.

Figure 15C.15-1a

FF011, FF012, FF021, FF022, FF023, FF036, FF037, FF038, FF045, FF046, FF084, FF088, FF094, FF095, FF096, FF097, FF098, FF099, FF106, FF107, FF108, FF109, FF110, FF111, FF112, FF113, FF114, FF115, FF116, FF117, FF118, FF119, FF120, FF154, FF155, FF188, FF189, FF190, FF191, FF192, FF193, FF195, FF196, FF197, FF198, FF199, FF400, FF412, FF451, FF470, FF471, FF472, FF473, FF474, FF475, FF476,	Passive, NC in-line valves
FF033	Passive, NC, LC in-line valve

Figure 15C.15-1b

FF050, FF051, FF060, FF067, FF073, FF074, FF089, FF100, FF101, FF102, FF103, FF104, FF122, FF123, FF124, FF125, FF127, FF128, FF129, FF131, FF133, FF134, FF135, FF136, FF137, FF138, FF139, FF140, FF142, FF143, FF144, FF147, FF149, FF150, FF151, FF152, FF156, FF157, FF166, FF171, FF172, FF173, FF174, FF175, FF176, FF177, FF178, FF180,	Passive, NC in-line valve

Figure 15C.15-1b (Continued)

FF181, FF185, FF186, FF406,
FF408, FF410, FF411, FF415,
C11-FF210, C11-FF211

Figure 15C.15-1c

FF227, FF249, FF250, FF251, — Passive NC in-line valves
FF253, FF254, FF257, FF260,
FF262, FF263, FF264, FF266,
FF268, FF270, FF273, FF274,
FF276, FF279, FF280, FF281,
FF282, FF283, FF284, FF285,
FF290, FF291, FF293, FF297

The following vent, drain and test valves are normally closed (NC) during normal plant operation or after a LOCA event. Therefore, these valves are considered passive devices and excluded from the FMEA analysis.

Figure 15C.15-1a

FF077, FF425, FF246, FF427, — Passive NC vent valve
FF429, FF430, FF434, FF435,
FF453, FF454, FF456, FF464,
FF466, FF468, FF481, FF483,
FF485, FF490, FF492, FF493,
FF496, FF501, FF502, FF503,
FF504, FF508

FF075, FF076, FF421, FF424, — Passive, NC drain valve
FF428, FF431, FF432, FF433,
FF436, FF439, FF441, FF461,
FF462, FF463, FF465, FF467,
FF469, FF482, FF489, FF507

Figure 15C.15-1a (Continued)

FF034, FF053, FF056, FF061, FF063, FF065	Passive, NC and capped test valve

Figure 15C.15-1b

FF484, FF486, FF487, FF494, FF495, FF497, FF498, FF499, FF509, FF510	Passive, NC vent valve
FF409, FF500, FF505	Passive, NC drain valve

Figure 15C.15-1c

FF292, FF294, FF296, FF300, FF301, FF302, FF304, FF306, FF309, FF313, FF318	Passive, NC vent valve
FF295, FF298, FF299, FF305	Passive, NC drain valve

The following in-line valves are normally open (NO) and are set at a fixed open position before normal plant operation to establish proper DW & CDS flow. Thereafter, the valve position is not changed either during normal operation or after a LOCA event and are considered passive devices. Therefore, they are excluded from the FMEA analysis.

Figure 15C.15-1a

FF009, FF013, FF014, FF016, FF017, FF019, FF024, FF025, FF026, FF028, FF029, FF030, FF031, FF032, FF048, FF049, FF057, FF058, FF082, FF083, FF086, FF092, FF093, FF153,	Passive, NO in-line valve

Figure 15C.15-1a (Continued)

FF159, FF160, FF161, FF194, FF421, FF452	
FF072, FF090	Passive, NO, LO in-line valve

Figure 15C.15-1b

FF040, FF041, FF042, FF043, FF044, FF066, FF068, FF069, FF070, FF121, FF126, FF130, FF141, FF145, FF158, FF162, FF401, FF402, FF404, FF405, FF416, FF457	Passive, NO in-line valve

Figure 15C.15-1c

FF201, FF204, FF205, FF206,
FF207, FF209, FF210, FF211,
FF212, FF213, FF214, FF215,
FF217, FF218, FF220, FF223,
FF224, FF225, FF226, FF227,
FF228, FF229, FF233, FF238,
FF247, FF248, FF252, FF255,
FF256, FF258, FF259, FF261,
FF265, FF269, FF271, FF272,
FF286, FF288, FF289, FF315,
FF316, FF317

The following drain valve is normally open (NO) during normal plant operation or after a LOCA event. Therefore, this valve is considered a passive device and excluded from the FMEA analysis.

Figure 15C.15-1a

FF440	Passive, NO drain valve

The following valves are excluded from FMEA analysis due to the passive nature of their operation during normal plant operation or subsequent to a LOCA event. These components are activated in case of an abnormal condition.

Figure 15C.15-1a

FF450	Passive, relief valve not required for safety function
FF437, FF438, FF445, FF446, FF447, FF448	Passive, vacuum break is required only following a pipe break
FF071	Passive check valve

Figure 15C.15-1b

CC11-FF216	Passive, relief valve not required for safety function
FF443, FF444	Passive, vacuum breaker is required only following a pipe break
FF403	Passive check valve

The following is excluded as a non-safety component.

Figure 15C.15-1b

P46-CC001	Passive, RI Demineralized Water Booster Pump, not required for safety function

The following instruments were excluded.

Figure 15C.15-1a

E22-LT-N054 C&G E22-LIS-N654 C&G	Safety instrumentation covered as part of HPCS System
E51-LT-NO35 A&E E51-LIS-N635 A&E	Safety instrumentation covered as part of RCIC System
TI-RR001, TI-RR002, PI-RR003, PI-RR004, PI-RR005, PI-RR012	Local instrumentation only

Figure 15C.15-1b

PI-RR010	Local instrumentation only

Figure 15C.15-1c

TI-RR006, PI-RR007, TI-RR008, PI-RR009	Local instrumentation only

(2) Elementary Diagram, Figures 15C.15-2a through d

Figure 15C.15-2a

All (Reference Material only)

Figures 15C.15-2b and c

(a) Indicating lights and associated resistor and switches (Lights not required for safety function)

(b) Computer signals and associated CAOI (Computer signals not required for safety function)

Figure 15C.15-2d

All (MCC and motor information only)

15C.15.9 Analysis and Results

The results of the FMEA given in Table 15C.15-1 are summarized as follows:

Number of line items analyzed	34
Number of failure modes analyzed	63
Number classified "B"	62
Number classified "N"	1

This FMEA verifies that, with the exception noted below*, the Demineralized Water and Condensate Distribution System will

*In the event check valve FF085 "fails to open," analysis is required to evaluate the effect of the RCIC/HPCS pumps on the surge volume header system. Lack of condensate supply and continued pump discharge will produce a partial vacuum resulting in degraded pump performance (lower NPSH), delayed switchover to the suppression pool and possible header collapse. The Applicant will provide this analysis.

perform its safety-related function on demand and it will not degrade nor prohibit any other system from performing its safety-related or safety-related supporting function. Further, the FMEA verified that no single failure within the Demineralized Water and Condensate Distribution System results in a loss of a safety function.

Table 15C.15-1

DEMINERALIZED WATER AND CONDENSATE SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.15-1a through c. Condensate and Demineralized Water Distribution P & ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.15-2a through d. Condensate and Demineralized Distribution Elementary Diagram (FMEA)

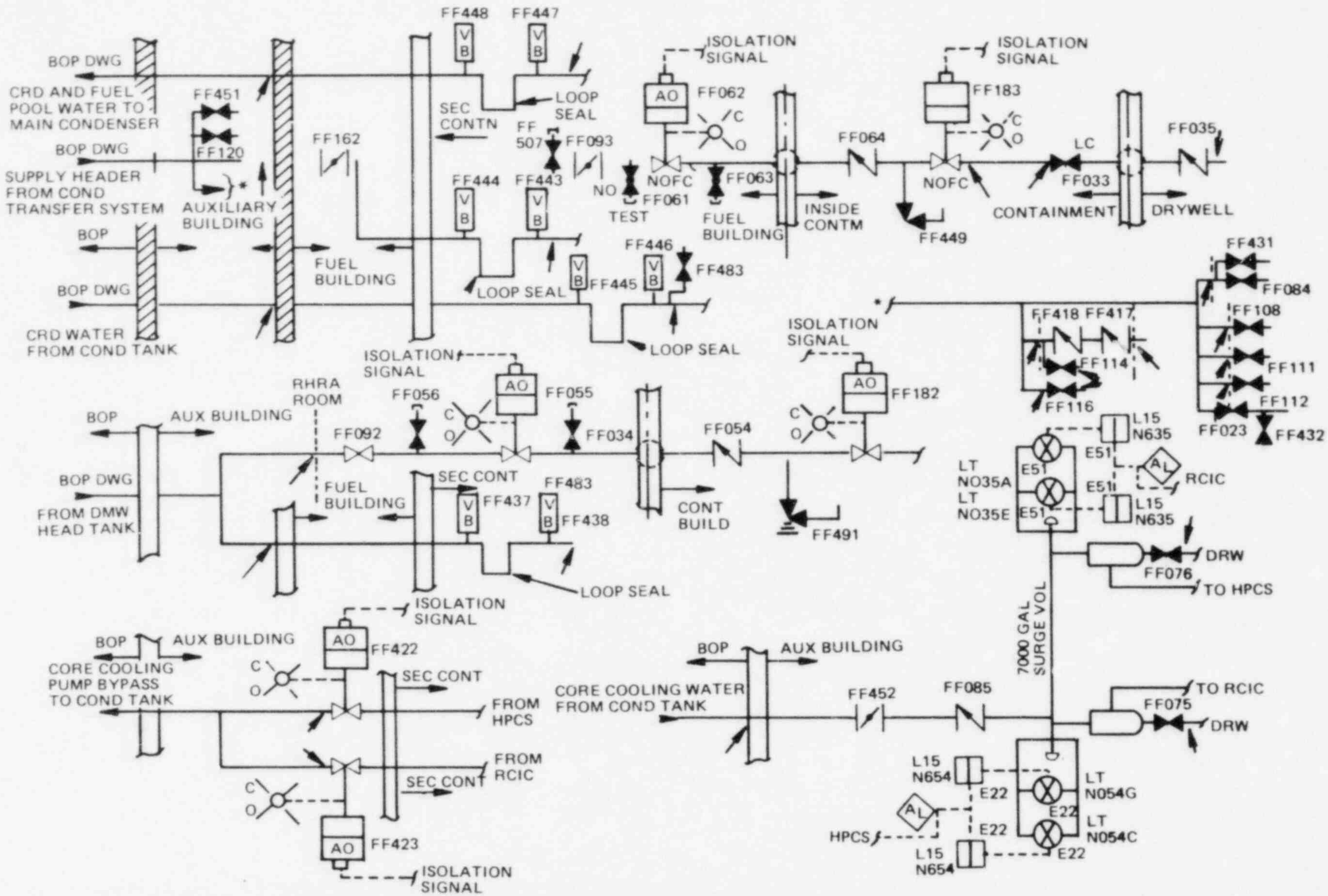


Figure 15C.15-3. Condensate and Demineralized Water Distribution Simplified Block Diagram

15C.16 SUPPRESSION POOL MAKEUP SYSTEM (P50)

(To be provided in December 1982)

15C.17 INSTRUMENT AIR DISTRIBUTION SYSTEM (P52)

15C.17.1 Scope

The FMEA covers all the active mechanical components depicted on the system P&ID, and all the mechanical, pneumatic, and electrical control devices necessary to permit the Instrument Air Distribution System (IADS) to perform its safety-related functions on demand.

The mechanical devices considered in the FMEA include pneumatic, electric motor and manually operated valves, and an air filter. The pneumatic controls include air-operated valves. The electrical control devices include load drivers, solenoids, system logic memories, control and indication opto isolators, and switches.

15C.17.2 System Defining Documents

(1) P&ID, Figures 15C.17-1a through e

(2) Elementary Diagram, Figures 15C.17-2a through e

(3) Simplified Block Diagram, Figures 15C.17-3a through c

15C.17.3 System Safety-Related Functions

(1) Division 1 and 2 isolation valves ensure the primary containment isolation and the drywell integrity.

(2) Division 1 and 2 isolation valves ensure the integrity of the secondary containment.

(3) The balance of the Instrument Air Distribution System is non-nuclear safety-related and does not perform any

15C.17.3 System Safety-Related Functions (Continued)

safety-related function. This system shall not fail in such a manner as to cause or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown capability

(b) Exceed dose limits to operating personnel

(c) Exceed dose limits to public

(d) Loss of primary pressure boundary

15C.17.4 Safety-Related Supporting System Functions

(1) AC & DC Power Distribution System Engineered Safety Feature (ESF) Electrical Buses (Divisions 1 and 2) - Provides electrical power for closing the containment isolation valves.

(2) Main Steam (Nuclear Boiler) System (Divisions 1 and 2) - Provides LOCA signals for closing the primary containment isolation and drywell integrity valves.

(3) Standby Gas Treatment System (SGTS) (Divisions 1 and 2) - Provides LOCA signals for closing the secondary containment isolation valves.

(4) Air Positive Seal Isolation Valve Leakage Control System (Divisions 1 and 2) - Provides backup leakage control for the primary containment isolation valves.

(5) Control Building HVAC System (Divisions 1 and 2) - Provides suitable environment for valve control logic and associate instrumentation.

15C.17.5 Initiating Events or Signals

(1) Division 1 and 2 LOCA signals for closing the primary containment and drywell isolation valves.

(2) Division 1 and 2 LOCA signals for closing the secondary containment isolation valves.

15C.17.6 Operator Actions Required

None.

15C.17.7 System Description

The Instrument Air Distribution System provides dry, oil-free compressed air for valve actuators including those for the main steam isolation valves, non-ADS relief valves, for non-safety-related instrument control functions and for general instruments and valve services within the Reactor Island. It also provides non-safety air supply to primary containment and drywell personnel air lock inflatable seals.

The air supply to the main steam isolation valves and the relief function of the safety relief valves of the Nuclear Boiler System is considered non-safety related since the valves are provided with air accumulators and check valves to isolate the air supply from the main header on loss of compressed air. This will provide the necessary safety functions in spite of loss of the instrument air system. The accumulators and check valves are safety-related components of the nuclear boiler system.

15C.17.8 FMEA Exclusions

(1) P&ID, Figures 15C.17-1a through e

The following mechanical components are excluded from the FMEA sheets because they do not perform a safety-related function or a safety-related supporting function; hence, they have no effect on safety. Electromechanical and electrical components are addressed on elementary diagrams.

Figure 15C.17-1a

Except for valves FF062, FF063, FF064, and FF065, all valves shown on this sheet are passive manual valves normally closed or normally open for maintenance or test operation.

Figure 15C.17-1b

Except for air filter DD001 and valves FF038, FF040, FF042, and FF043, all valves shown on this sheet are passive manual valves normally closed or normally open for maintenance or test operation.

Figure 15C.17-1c

Except for valves FF066, FF067, FF068, and FF069, all valves shown on this sheet are passive manual valves normally closed or normally open for maintenance or test operation.

15C.17.8 FMEA Exclusions (Continued)

Figure 15C.17-1d

All valves shown on this sheet are passive manual valves normally closed or normally open for maintenance or test operation.

Figure 15C.17-1e

All valves shown on this sheet are passive manual valves normally closed or normally open for maintenance or test operation.

(2) Elementary Diagram, Figures 15C.17-2a through e

The following electrical/logic components or subsystems are excluded from the FMEA sheets because they do not perform a safety-related function or a safety-related supporting function; hence, they have no effect on safety.

Figure 15C.17-2a

Reference information; all components are excluded.

Figure 15C.17-2b

(a) ILDs and associated indicating lights

(b) PLOIs, CAOIs, and computer printout logic

(c) Voltage dropping resistors and associated valves position indicating lights

15C.17.8 FMEA Exclusions (Continued)

Figure 15C.17-2c

(a) ILDs and associated indicating lights

(b) PLOIs, CAOIs and computer printout logic

(c) Voltage dropping resistors and associated valve position indicating lights

Figure 15C.17-2d

All components on this sheet are excluded because they are for reference only.

Figure 15C.17-2e

All components on this sheet are excluded because they are shown for wiring information and they are for reference only.

15C.17.9 Analysis and Results

The results of the FMEA given in Table 15C.17-1 are summarized as follows:

Number of line items analyzed	33
Number of failure modes analyzed	61
Number classified "A"	60
Number classified "B"	1

This FMEA verifies that no single failure of an active mechanical or active and passive electrical component in the Instrument Air Distribution System (IADS) will prevent this system or any other system to perform its safety functions on demand.

Table 15C.17-1

INSTRUMENT AIR DISTRIBUTION SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.17-1a through e. Instrument Air Distribution System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.17-2a through e. Instrument Air Distribution System Elementary Diagram (FMEA)

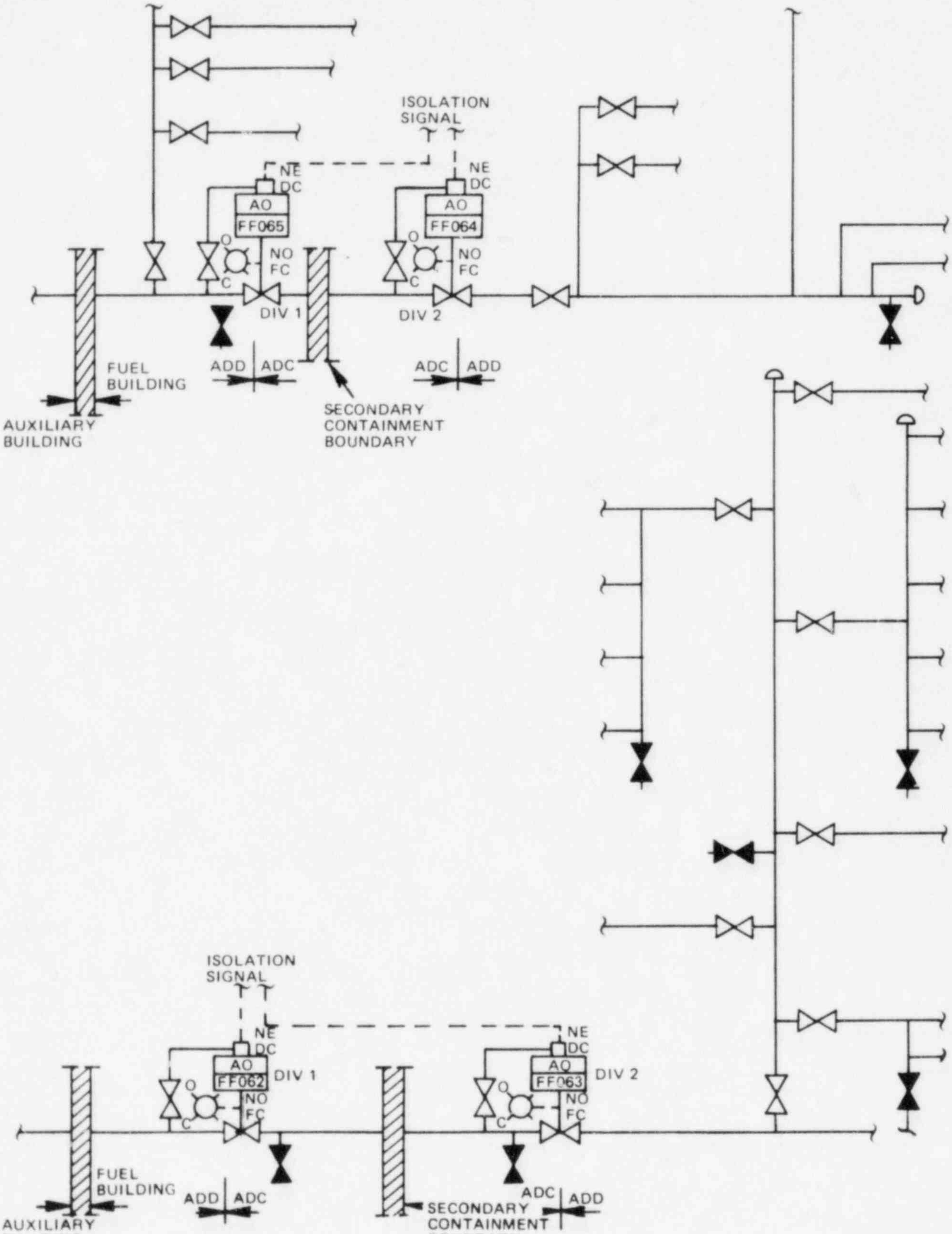


Figure 15C.17-3a. Instrument Air Distribution System Simplified Block Diagram

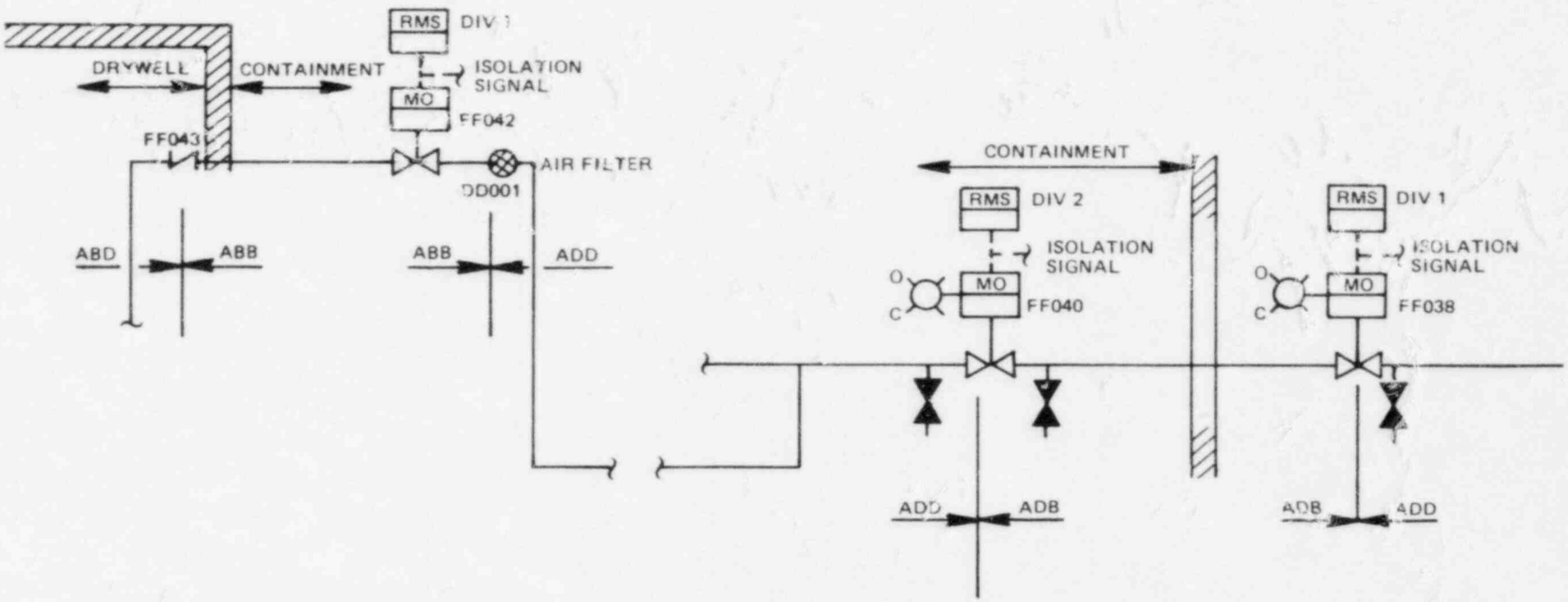


Figure 15C.17-3b. Instrument Air Distrubiton System Simplified Block Diagram

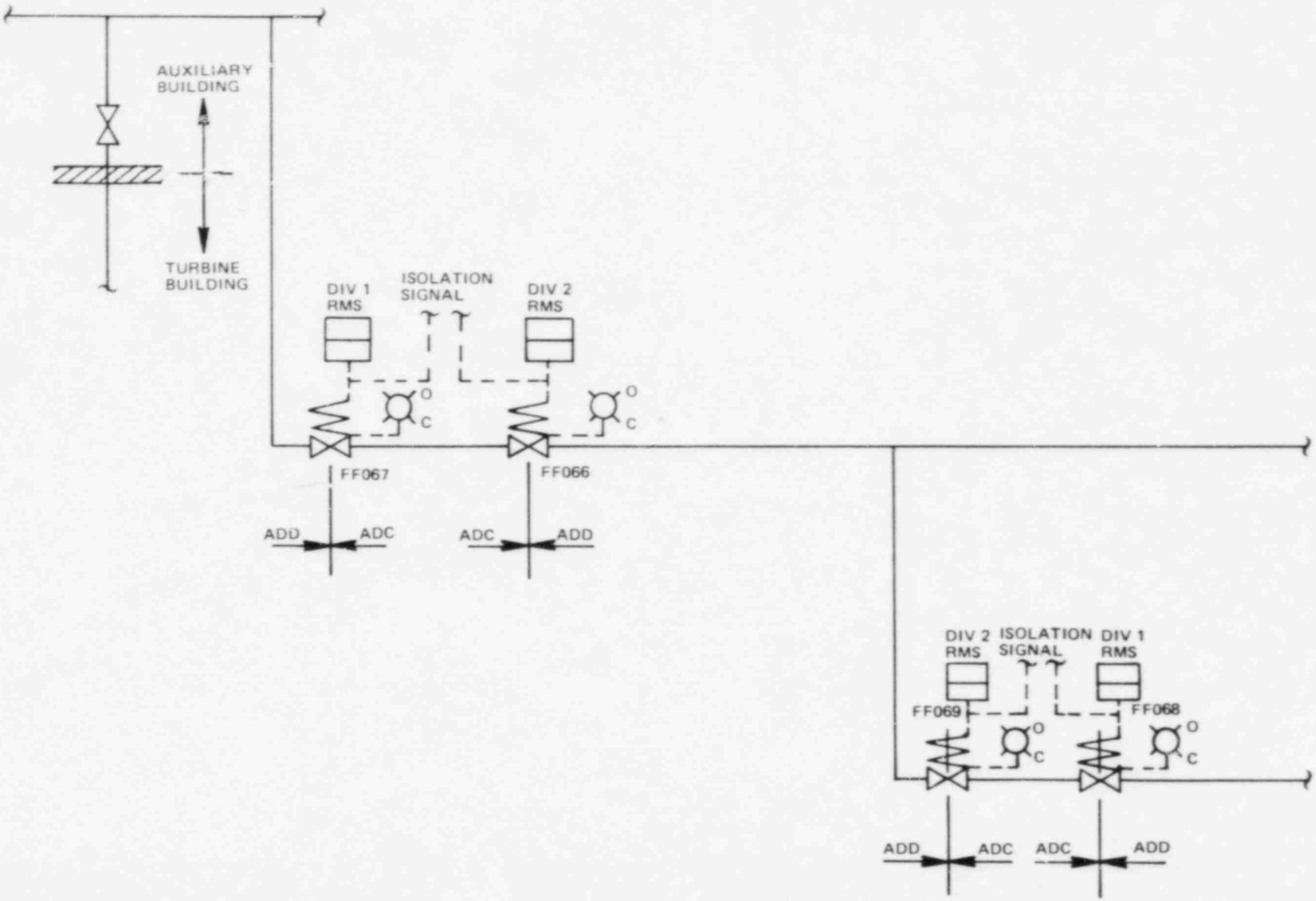


Figure 15C.17-3c. Instrument Air Distribution System Simplified Block Diagram

15C.18 SERVICE AIR DISTRIBUTION SYSTEM (P52)

15C.8.1 Scope

This analysis covers the Service Air Distribution System active components shown on the system P&ID and all mechanical and electrical devices necessary to permit the system to perform its safety-related functions on demand.

The mechanical devices include motorized and manually operated valves. The electrical controls of the system include electric motors, manual switches, transducers, signal transmitters, inverters, isolators, logic and device load drivers.

15C.8.2 System Defining Documents

(1) P&ID, Figure 15C.18-1

(2) Elementary Diagram, Figures 15C.18-2a through d

(3) Simplified Block Diagram, Figure 15C.18-3

15C.18.3 System Safety-Related Functions

(1) Division 1 and 2 isolation valves in this system ensure the primary containment isolation.

(2) Division 1 and 2 isolation valves in this system ensure the secondary containment integrity.

(3) Normally locked closed manual valve in series with check valve provides drywell integrity.

(4) Provides "Isolation Valve FF010 Closed" signal for initiation of the portion of the Air Positive Seal Isolation

15C.18.3 System Safety-Related Functions (Continued)

Valve Leakage Control System which supplies sealing air to the valve.

(5) The balance of the Service Air Distribution System is non-nuclear safety-related and does not perform any safety-related function. This system shall not fail in such a manner as to cause or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown capability

(b) Exceed dose limits to operating personnel

(c) Exceed dose limits to public

(d) Loss of primary pressure boundary

15C.18.4 Safety-Related Supporting System Functions

(1) AC & DC Power Distribution System, ESF buses (Division 1 and 2) - Provides electrical power for the divisional isolation valves.

(2) Air Positive Seal Isolation Valve Leakage Control System - Provides leakage control for the isolation valves.

(3) Main Steam System - Provides Division 1 and 2 LOCA signals for closing primary containment isolation valves.

(4) Standby Gas Treatment System - Provides Division 1 and 2 isolation signals for closing the secondary containment isolation valves.

15C.18.5 Initiating Events or Signals

(1) Division 1 and 2 LOCA signals initiate closing of the primary containment isolation valves.

(2) Division 1 and 2 LOCA signals initiate closing of the secondary containment isolation valves.

15C.18.6 Operator Actions Required

None.

15C.18.7 System Description

The Service Air Distribution System is designed to provide compressed air for non-safety functions within the Nuclear Island. The Service Air Distribution System consists of necessary piping, manually operated valves, motor-operated isolation valves, instrumentation and controls. It provides compressed air for tank sparging, filter demineralizer backwashing, refueling equipment, jib crane, fuel pool gate seals and other services requiring air of lower quality than that provided by the instrument air distribution system at a nominal pressure of about 110 psig. The air supply is provided by BOP.

Two motor-operated isolation valves are provided for the service air pipe penetrations through primary and secondary containments. The service air line to the drywell is provided with a normally locked closed manual isolation valve in series with a check valve.

Normally open, motor-operated isolation valves receive isolation signal from their respective division.

15C.18.8 FMEA Exclusions

(1) The components or subsystems excluded because they do not perform a safety-related or safety-related supporting function and have no effect on safety are as follows:

P&ID, Figure 15C.18-1

All manually operated valves (either normally open or closed)	Non-safety related, passive
All temperature or pressure indicators	Non-safety related
Isolation valve FF015	Passive mechanical component (locked closed)

Elementary Diagram, Figures 15C.18-2a through d

Figure 15C.18-2a	All excluded (reference information and non-safety instrumentation)
Figure 15C.18-2b and c	Logic circuits for computer inputs and components for position indication

15C.18.9 Analysis and Results

The results of the FMEA given in Table 15C.18-1 are summarized as follows:

Number of line items analyzed	30
Number of failure modes analyzed	42
Number classified "A"	1
Number classified "B"	41

This FMEA verifies that no single failure in the Service Air Distribution Systems results in a loss of a safety function.

Table 15C.18-1

SERVICE AIR DISTRIBUTION SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.18-1. Service Air Distribution System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.18-2a through d. Service Air Distribution System Elementary Diagram (FMEA)

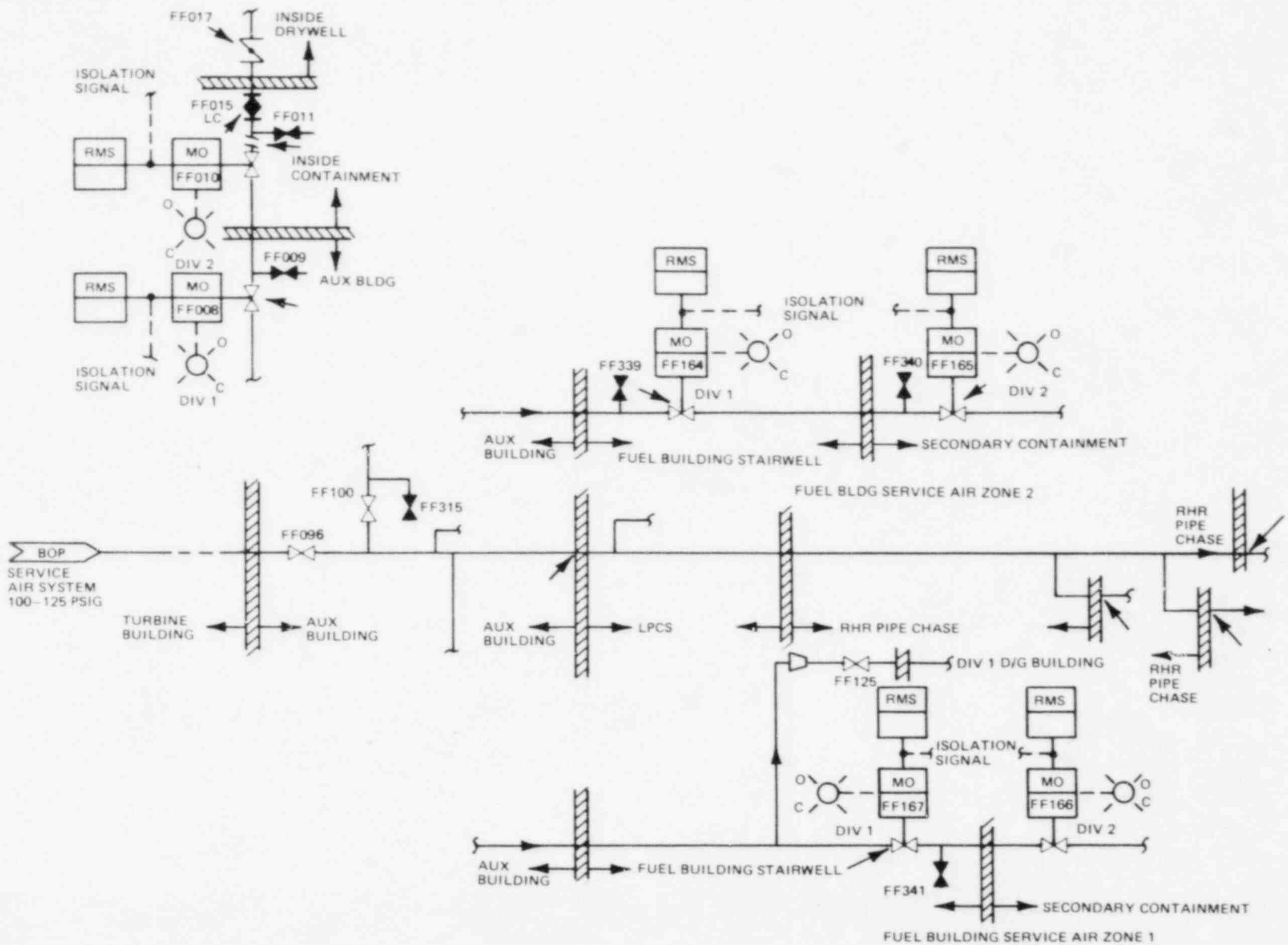


Figure 15C.18-3. Service Air Distribution System Simplified Block Diagram

15C.19 PNEUMATIC SYSTEM (P53)

(To be provided in December 1982)

15C.20 CLEAN RADWASTE DRAIN SYSTEM (P55)

15C.20.1 Scope

The FMEA covers all the active components depicted on the P&ID, and all the mechanical, pneumatic, and electrical control devices necessary to permit the Clean Radioactive Waste (CRW) Drain System to perform its safety-related functions on demand.

The mechanical devices considered in the FMEA include electric motor and manually operated valves, relief valves, check valves, and water pumps. The electrical control devices include switches, signal transmitters, trip units, converters, isolators, logic and device load drivers.

15C.20.2 System Defining Documents

(1) P&ID, Figures 15C.20-1a through c

(2) Elementary Diagram, Figures 15C.20-2a through n

(3) Simplified Block Diagram, Figure 15C.20-3

15C.20.3 System Safety-Related Functions

(1) Ensures primary containment integrity by closing the primary containment isolation valves on receipt of an isolation signal.

(2) The balance of the CRW System is a non-safety system and does not perform any nuclear safety-related functions. This system shall not fail in such a manner as to cause or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown capability

(b) Exceed dose limits to operating personnel

(c) Exceed dose limits to public

(d) Loss of primary pressure boundary

15C.20.4 Safety-Related Supporting System Functions

(1) Main Steam System (Divisions 1 and 2) - Provides isolation (LOCA) signals on low reactor water level and/or high drywell pressure.

(2) AC & DC Power Distribution System, Engineered Safety Feature (ESF) Electrical Buses (Divisions 1 and 2) - Provides electrical power to operate and control the system isolation valves.

(3) Water Positive Seal (WPS) System - Provides water sealing for the isolation valves following a LOCA event.

(4) Control Building HVAC System - Provides suitable environment for valve control logic and associated instrumentation.

15C.20.5 Initiating Events or Signals

(1) Primary Containment Isolation - The Main Steam System provides isolation signals to automatically close the system isolation valves.

15C.20.6 Operator Actions Required

None.

15C.20.7 System Description

The Clean Radioactive Waste (CRW) Drain System consists of individual equipment drain subsystems which collect, monitor and transfer low conductivity liquid wastes within a particular building area. There are provisions in the system to collect samples at various points to determine the radioactivity and/or chemical composition of the liquids.

Each of these CRW subsystems serves to individually collect, monitor, and transfer all the low conductivity liquid waste solution from all sources within the specified building or building area served by the particular subsystem. Each subsystem has its individual collection sump which provides sufficient collection and surge capacity for the liquids obtained from all CRW sources within the subsystem.

The CRW System consists of clean liquid collected from individual sources of high purity, low conductivity waste solutions with varying radioactivity content. The CRW wastes generally have a conductivity less than 1 μmho/cm. The CRW consists of drains from piping and equipment containing high quality water (e.g., primary system, condensate, feedwater). The CRW liquids are normally directly piped from specific equipment or piping drains in a closed piping system so they remain clean and do not pick up dirt or foreign matter from floors or the atmosphere.

The drywell equipment drain sump has a sensitivity of detecting steam leakage of 50 percent of anticipated background leakage. The sump alarm setpoint has an adjustable range up to 25 gpm. The drywell equipment drain sump collects only identified leakages piped from equipment. This drywell equipment drain sump receives liquid from the following sources, condensate drainage from primary recirculating pump seal leakoffs, reactor vessel head flange vent-drain, valve packing leakoffs, upper-containment pool seals

15C.20.7 System Description (Continued)

and other equipment. Sump volume increases in excess of background leakage are indicative of reactor coolant leakage.

The inner packing leakage on valves in 2-in. and larger lines are piped to the equipment drain sump after passing through leak detection system equipment. The valve seal leakoff applies to all power-operated valves located inside the drywell for the following systems; nuclear boiler, reactor recirculation, reactor water cleanup, high and low pressure core spray, RCIC, RHR and feedwater.

The containment equipment drain sump collects drainage from various equipment located in the containment area. The sump instrumentation is capable of monitoring leakage from 0 to 30 gpm. A leakage limit slightly below 25 gpm is annunciated in the main control room.

The drywell, containment and RCIC pump room equipment drain sumps have instruments and controls for detecting leakage from sources entering the sumps and determining/controlling sump liquid levels. The instruments and controls are part of the Leak Detection System located on the leak detection panel in the main control room. Pumping cycles which become too lengthy or too frequent indicate high leakage rates from one or more of the components in the area draining to the sump.

The containment and drywell CRW sumps and RCIC CRW sump have instrumentation that permits the detection of leakage and provides an alarm for high leakage rates. The instrumentation includes two timers, one for pump fillup, the other for pumpout, pump running time meters and event counters. Recorders are provided to record the level. This provides a method of determining changes in inlet flow rates into the sump and hence provides a means of detecting increases in sources which leak into the sump and are part of the Leak Detection System.

15C.20.7 System Description (Continued)

The Fuel Building, Radwaste Building, and RCIC pump room sumps have liquid level detection and recording instrumentation. The containment and drywell equipment drain sumps each have provisions for measuring their sump liquid temperature and automatically recirculating the sump contents through a drain cooler to cool the sump contents if the temperature rises to 150°F.

Each CRW subsystem sump contains sump pumps, which are controlled by sump level instruments and controls, and serve to automatically transfer the CRW sump liquids through pump discharge lines to the radwaste system.

A drywell equipment drain cooler and a containment equipment drain cooler are provided to cool the contents of the drywell and containment DRW drain sumps. Both of these coolers are shell and tube heat exchangers each with a design heat duty of 500,000 Btu/hr.

The CRW has three motor-operated isolation valves and two check valves to provide isolation integrity of the drywell and containment. An isolation signal from two divisions of either low reactor water or high drywell pressure will automatically close the motor-operated isolation valves. Closure of the isolation valves automatically shuts down the two drywell equipment drain sump pumps and the two containment equipment drain sump pumps.

The CRW System provides secondary containment integrity by using a liquid seal loop in the sump pump discharge line from the Fuel Building equipment drain sumps "A" and "B" and a water seal loop in a 3-in. drain line to the RCIC Pump Room equipment drain sump.

Each sump is provided with dual sump pumps of identical design. The sump pumps are automatically operated by sump level switches. One sump pump starts automatically when the sump liquid reaches

15C.20.7 System Description (Continued)

the high level. The second sump pump is started if the sump liquid reaches the high-high level. Both sump pumps are stopped when sump liquid reaches the low level. An alternator automatically alternates operation of the two sump pumps.

15C.20.8 FMEA Exclusions

(1) P&ID, Figures 15C.20-1a through c

The following mechanical components are excluded from the FMEA sheets because they do not perform a safety-related function or a safety-related supporting function; hence, they have no effect on safety. Electromechanical and electrical components are addressed on elementary diagrams.

Figure 15C.20-1a

All components are excluded except manual valves FF057 through FF060 at H6.

Figure 15C.20-1b

(a)	CC012A and B and CC014A and B	Pumps
(b)	FF017A and B, FF005A and B, FF122, FF056	Check Valves
(c)	FF041A and B, FF038A and B, FF018A and B, FF006A and B, FF015, FF074, FF086 through FF088, FF073, FF061, FF089, FF090	Gate Valves

15C.20.8 FMEA Exclusions (Continued)

(d)	FF008	Sample Valve
(e)	FF010, FF013, FF014, FF024 and FF034	Test Valves
(f)	FF023 and FF016	Recirculation Valves
(g)	FF022	Discharge Control Valve
(h)	BB001, BB002	Drain Coolers
(i)	NN026, NN027, NN028, NN034, NN035, NN036, NN043, NN044, NN045, NN046, RR016A and B, RR018, RR021A and B, RRO, RR031A and B, RR032A and B, RR033A and B, RR034A and B, RR035, R605, R606, R607, R610, RR040	Instrumentation

Figure 15C.20-1c

All components are excluded except vacuum relief valves FF062 and FF063 at A2.

(2) Elementary Diagram, Figures 15C.20-2a through h

The following electrical/logic components or subsystems are excluded from the FMEA sheets because they do not perform a safety-related function or a safety-related supporting function; hence, they have no effect on safety.

15C.20.8 FMEA Exclusions (Continued)

Figure 15C.20-2a

Reference information, all components are excluded.

Figure 15C.20-2b

(a) ILDs and associated indicating lights

(b) Fault logic, CAOIs, and computer printout

(c) Voltage dropping resistors and associated valve position indicating lights

Figure 15C.20-2c through n

Analysis of these figures verifies that components shown on these drawings do not perform a safety-related function and have no effect on a safety-related function or on a safety-related supporting function.

15C.20.9 Analysis and Results

The results of the FMEA given in Table 15C.20-1 are summarized as follows:

Number of line items analyzed	20
Number of failure modes analyzed	32
Number classified "A"	23
Number classified "B"	9

This FMEA verifies that no single failure of an active mechanical or active and passive electrical component in the CRW System will prevent this system or any other system from performing its safety functions on demand.

Table 15C.20-1

CLEAN RADWASTE DRAIN SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.20-1. Clean Radwaste Drain System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.20-2a through n. Clean Radwaste Drain System Elementary Diagram (FMEA)

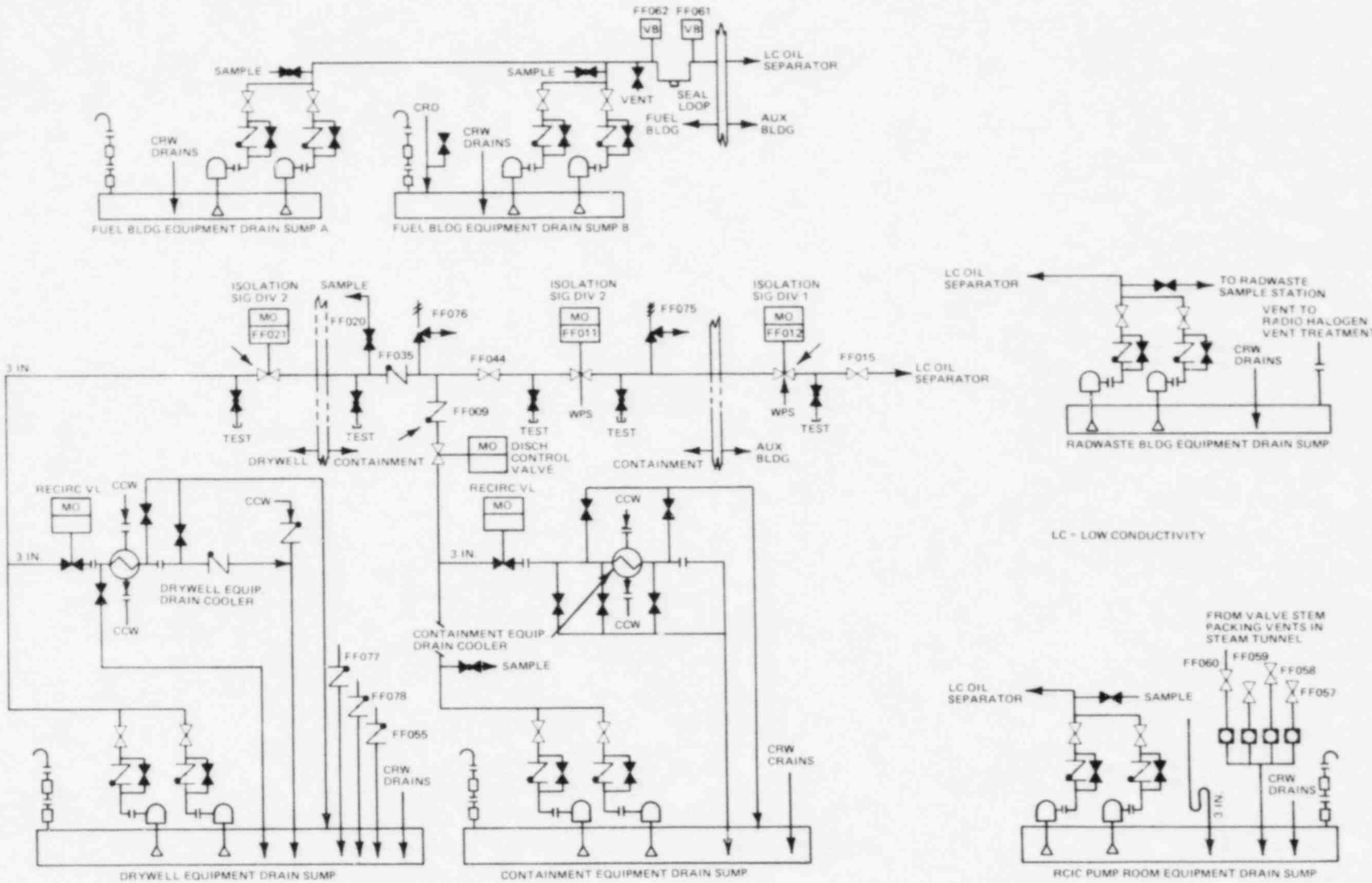


Figure 15C.20-3. Clean Radwaste Drain System Simplified Block Diagram

15C.21 ESSENTIAL SERVICE WATER SYSTEM (P41)

Applicant to provide

15C.22 CONTAINMENT COOLING PRESSURE CONTROL AND PURGE (T41)

Applicant to provide

15C.23 DRYWELL COOLING SYSTEM (T41)

15C.12.1 Scope

The Drywell Cooling System is a non-safety-related system; therefore, this FMEA covers system safety-related functions only on the system level.

15C.23.2 System Defining Documents

(1) P&ID, Figure 15C.23-1

(2) Elementary Diagram, Figures 15C.23-2a through i

(3) Simplified Block Diagram, Figure 15C.23-4

15C.23.3 System Safety-Related Functions

The Drywell Cooling System is a non-safety system and does not perform any safety-related functions. This system shall not fail in such a manner to cause or interfere with other systems in such a way as to cause:

(1) Loss of safe shutdown capability

(2) Exceed dose limits to the operating personnel

(3) Exceed dose limits to the public

(4) Loss of primary pressure boundary

15C.23.4 Safety-Related Supporting System Functions

None.

15C.23.5 Initiating Events or Signals

None.

15C.23.6 Operator Actions Required

Operator must take appropriate corrective action(s) upon receiving a high drywell temperature alarm signal to protect safety-related equipment located in the drywell area.

15C.23.7 System Description

The Drywell Cooling System which uses 100 percent recirculated air is designed to achieve four things: (1) control drywell space atmosphere for any operational transient after which a rapid return to power production is expected; (2) maintain drywell temperature during normal operations; (3) prevent hot spots from occurring in the area of the drywell; and (4) prevent the drywell temperature from rising during operational transients such that the drywell pressure does not rise to the LOCA trip point.

The system consists of six (6) fan-coil units which are divided into two (2) fan-coil unit sections. Two units of each section are in the operate mode during normal operation with one unit in the standby mode. Three fan-coil units have the cooling capacity to maintain the drywell temperature during the shutdown mode. The fan-coil units are automatically stopped during a LOCA.

The fan-coil units are manually started. Temperature sensors in the fan-coil unit inlet and outlet air stream and area sensors provide readout of the temperatures as well as control of the chilled water 3-way valve and an alarm on any air temperatures above 160°F.

Air flow and temperatures are also measured on the air supplied to the Reactor Pressure Vessel (RPV) skirt area. Low air flow or temperature as well as high temperature in either branch will cause an alarm.

The fan-coil units will automatically restart on loss of power except during a LOCA.

15C.23.8 FMEA Exclusions

None.

15C.23.9 Analysis and Results

The results of the FMEA given in Table 15C.23-1, are summarized as follows:

Number of line items analyzed	1
Number of failure modes analyzed	3
Number classified "D"	3

This FMEA verifies that no failure of the Drywell Cooling System will result in the performance degradation or failure of any other safety-related or safety-related supporting system to provide its designed safety-related functions.

Table 15C.23-1

DRYWELL COOLING SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.23-1. Drywell Cooling System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.23-2. Drywell Cooling System
Elementary Diagram (FMEA)

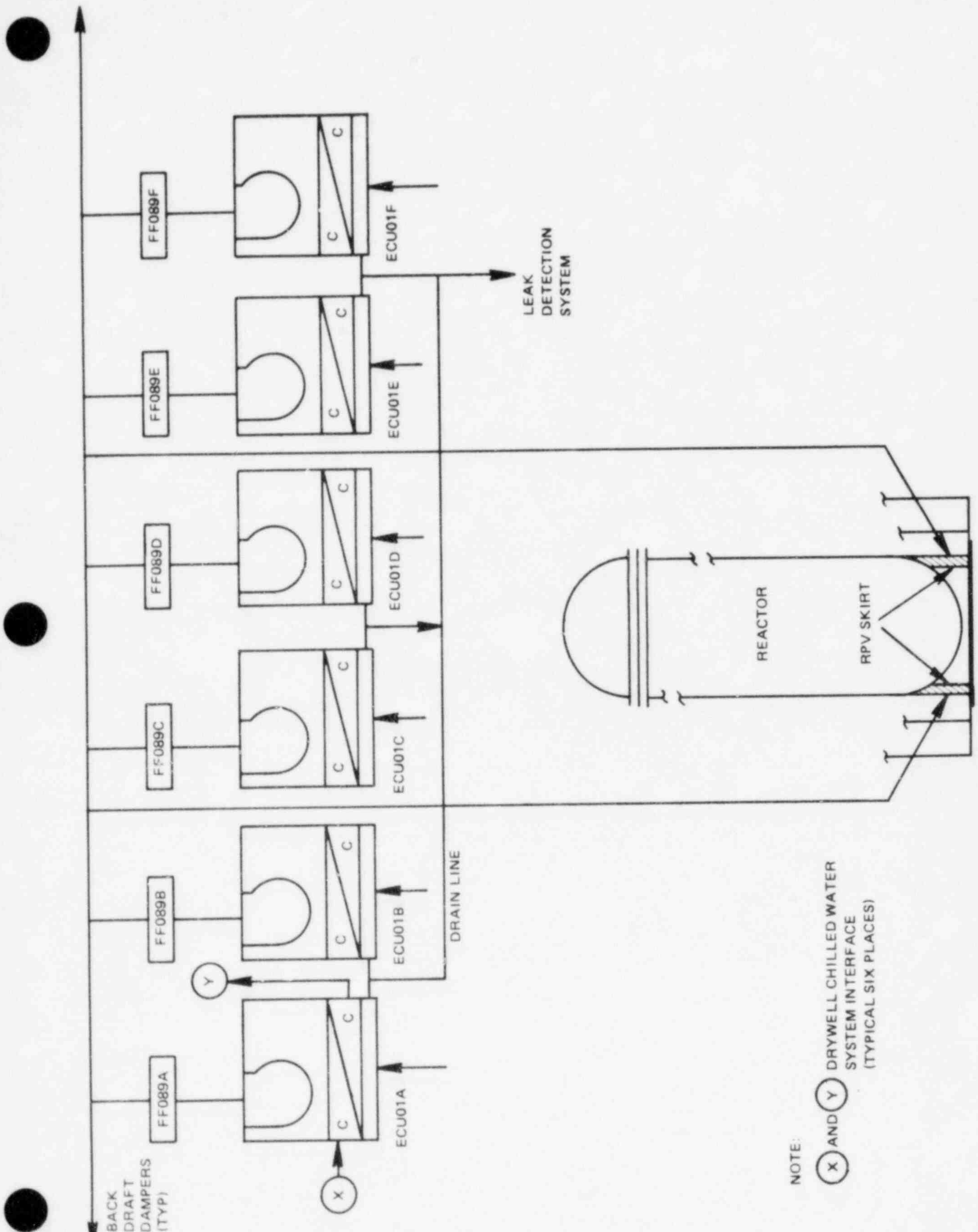


Figure 15C.23-3. Drywell Cooling System Simplified Block Diagram

15C.24 SHIELD ANNULUS RETURN/EXHAUST SYSTEM AND PLANT VENTILATION (T41)

(To be provided in December 1982)

15C.25 HYDROGEN MIXING, DRYWELL VACUUM RELIEF AND CONTAINMENT VACUUM RELIEF (T41)

Applicant to provide

15C.26 HYDROGEN RECOMBINER SYSTEM (T49)

15C.26.1 FMEA Task Scope

This FMEA covers the components of the Hydrogen Recombiner System which are necessary to permit the hydrogen recombiners to perform their safety-related functions. The only mechanical component is the recombiner itself; the electrical components include the power supply panel and the control panel.

15C.26.2 System Defining Documents

(1) Elementary Diagram, Figures 15C.26-1a through c

(2) Simplified Block Diagram, Figure 15C.26-2

15C.26.3 System Safety-Related Functions

Prevents accumulation of an explosive concentration of hydrogen within the containment building following a loss-of-coolant accident (LOCA). Hydrogen is recombined under controlled conditions to maintain H_2 concentrations below safe limits.

15C.26.4 Safety-Related Supporting System Functions

(1) AC & DC Power Distribution System (Divisions 1 and 2) ESF Buses - Provides electrical power for operation and control of the Hydrogen Recombiners.

(2) Containment Atmosphere Monitoring System - Provides the hydrogen concentration in the containment atmosphere information to the operator.

(3) Hydrogen Mixing System - Equalizes hydrogen concentrations in the drywell and containment prior to and during the hydrogen recombiner operation.

(4) Auxiliary Building Electrical Areas HVAC System (Divisions 1 and 2) - Provides suitable ambient conditions for the Hydrogen Recombiner Control and Power Supply Panels.

15C.26.5 Initiating Events or Signals

Hydrogen generation inside the containment following an initiating event requires the operator to initiate operation of the Hydrogen Recombiners when the hydrogen concentration in the containment atmosphere reaches a predetermined level.

15C.26.6 Operator Actions Required

After an initiating event, the operator will manually initiate the operation of the hydrogen recombiners when the hydrogen concentration in the containment atmosphere reaches a predetermined level.

15C.26.7 System Description

The Hydrogen Recombiner System offers maximum reliability in combustible gas control (hydrogen) in the containment following a LOCA. It consists of two redundant, completely independent, identical subsystems, each capable of providing the required hydrogen removal capacity. Each subsystem consists of:

(a) Hydrogen Recombiner unit located within plant containment building

(b) Power supply panel

(c) Control panel

The b and c components are located outside the containment in an area which is accessible to operator following a LOCA. Figure 15C.23-3 is a simplified block diagram of the Hydrogen Recombiner System.

Hydrogen Recombiner Units

The hydrogen recombiners are of a "thermal type" utilizing electric resistance heaters. The gas stream entering the hydrogen recombiner(s) when it is placed into operation will be a mixture of nitrogen, oxygen, hydrogen, steam, and noble gases. Hydrogen and oxygen will be combined without the possibility of propagation of the reaction upstream of the hydrogen recombiner. Each recombiner unit is effectively a constant volume machine with minimum flow of 100 cfm. It has a minimum hydrogen removal rate equivalent to removal efficiency of 98 percent at a flow rate of 100 scfm with a process gas hydrogen concentration of 4 percent. It is free from spontaneous combustion and/or detonation for all modes of operation specified. The recombiner unit consists of an outer structure made from Type-300 series stainless steel, inner structure made from Inconel 600, four banks (60 units each) of vertically stacked electric heaters sheathed with Incoloy 800 and operated below their rated power densities. This system is capable of attaining recombination conditions in less than 4 hours. There are three Chromel-Alumel thermocouples installed on the unit for periodic functional testing of the system. They are not needed during post-accident operation or control.

Power Supply Panel

Each power supply panel is located outside the containment and supplies power to the recombiner. It consists of 75 kW 3-phase transformer, solid state power controller, auxiliary transformer, magnetic contactor and other control circuitry that is needed to convert 3-phase delta input power to 3-phase 4-wire wye output and to control output power level to the heater banks.

Control Panel

This panel is also located outside the containment and is used to initiate and control the power supply and to read out temperatures from the three test thermocouples located in the recombiner units. Instruments and controls mounted on this panel include a power meter, thermocouple readout, power control potentiometer, on/off switch, and power available indicating lamp.

15C.26.8 FMEA Exclusions

The following components are excluded because they do not perform any safety-related function.

(1) Elementary Diagram, Figures 15C.23-1a through c

Figure 15C.23-1a

All	General information

Figures 15C.23-1b and c

All	Components for status indicating and annunciation only

15C.26.9 Analysis and Results

The results of the FMEA given in Table 15C.23 are summarized as follows:

Number of line items analyzed	8
Number of failure modes analyzed	10
Number classified "B"	10

Hydrogen recombiners are designed to recombine the effluent hydrogen and oxygen in the containment atmosphere through thermal reaction. Two redundant and independent 100-percent capacity hydrogen recombiners are provided for this function. Any single failure postulated in this FMEA will, in the worst case, disable one hydrogen recombiner, but the second unit will perform the hydrogen recombining function as needed.

It is therefore concluded that no single failure in the hydrogen recombiner system will prevent the system from performing its safety-related functions on demand or prevent any other system from providing its designed safety-related functions.

Table 15C.26-1

HYDROGEN RECOMBINER SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.26-1a through c. Hydrogen Recombiner System Elementary Diagram (FMEA)

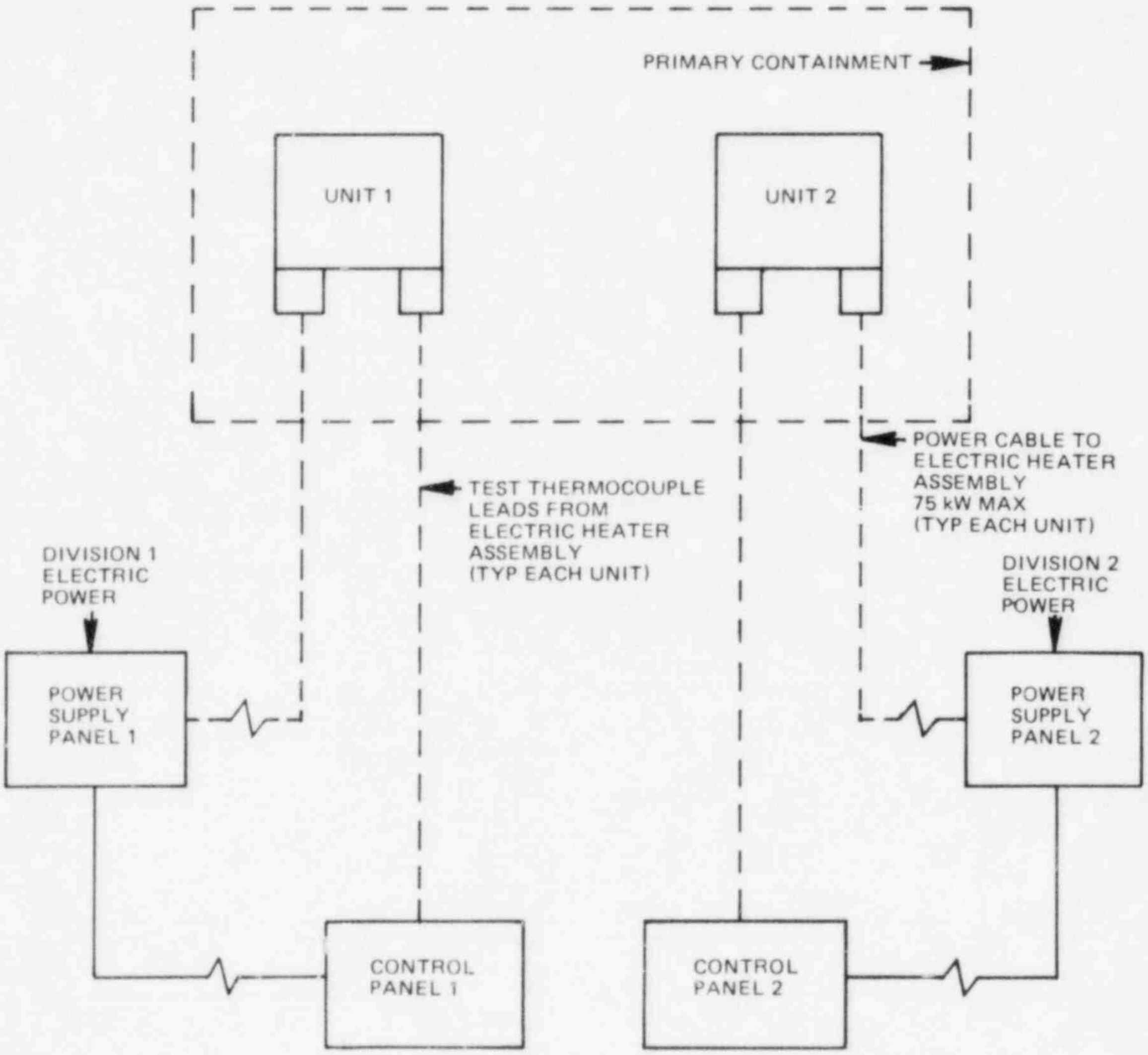


Figure 15C.23-2. Hydrogen Recombiner System Simplified Block Diagram

15C.27 WET STANDPIPE FIRE PROTECTION SYSTEM (X43)

15C.27.1 Scope

The FMEA covers all the active mechanical components depicted on the P&ID and all the mechanical, pneumatic, and electrical control devices necessary to permit the Wet Standpipe Fire Protection (WSFP) System to perform its safety-related functions on demand.

The mechanical devices include electric motor-operated valves and a check valve. The electrical components include switches, optical isolators, logic components, device load drivers, motor control centers and electric motors.

15C.27.2 System Defining Documents

(1) P&ID, Figures 15C.27-1a and b

(2) Elementary Diagram, Figures 15C.27-2a through c

(3) Simplified Block Diagram, Figures 15C.27-3a and b

15C.27.3 System Safety-Related Functions

(1) The WSFB System maintains the integrity of the primary containment by closing the system isolation valves on receipt of an isolation signal and the drywell integrity by a normally closed isolation valve in series with a check valve.

(2) The WSFPS contains water seal loops to maintain the integrity of the secondary containment in the event of an accident.

(3) The balance of the WSFPS is not a nuclear safety-related system and does not perform any safety-related function. It shall

not fail in such a manner as to cause, or interfere with other systems in such a way as to cause:

(a) Loss of safe shutdown capability

(b) Exceed dose limits to operating personnel

(c) Exceed dose limits to the public

(d) Loss of primary pressure boundary

15C.27.4 Safety-Related Supporting System Functions

(1) Main Steam (Nuclear Boiler) System (Divisions 1 and 2) - Provides containment isolation signals.

(2) AC & DC Power Distribution System, ESF Electrical Buses (Divisions 1 and 2) - Provides electrical power for the isolation valves operation and control.

(3) Water Positive Seal Isolation Valve Leakage Control System (Divisions 1 and 2) - Supplies sealing water to containment isolation valves.

15C.27.5 Initiating Events or Signals

(1) Divisions 1 and 2 primary containment isolation signals. In the event of a LOCA, the WSFPS function inside containment shall be automatically terminated to accomplish containment and drywell isolation. Valves FF128, FF131, FF133 close upon receipt of an isolation signal.

15C.27.6 Operator Actions Required

None.

15C.27.7 System Description

The WSFP System consists of wet standpipe (WSP) and fire hose systems, and sprinkler systems.

The BOP fire water loop around the nuclear island provides two 8-in. supply mains for the Auxiliary and Fuel Building standpipe system. The mains are located on opposite sides of the nuclear island and are interconnected within the nuclear island. They are routed within the building to two risers. One riser is located in the Zone 1 stairwell and the other in the Zone 2 stairwell. Locked open block valves are provided in each riser as well as the interconnecting pipe.

The standpipe system for the Control Building is fed from the BOP fire water loop by a 6-in. supply main. The standpipe system for the Radwaste Building is fed from a 6-in. supply main which comes from the BOP fire water loop. Each standpipe supply main is provided with siamese fire department connections and check valves. Nuclear island, Control Building, and Radwaste Building standpipe supply mains have pressure gages and flow alarms located at or near the entry point within the building.

A hose station is located outside the main entry door to each diesel generator room. The water line to each diesel generator room includes a normally closed post indicator valve. The piping between the hose station and the post indicator valve is provided with an automatic ball drip to keep the line free of water in order to prevent freezing.

The wet standpipe system supplies the water spray system of the Standby Gas Treatment units, Radwaste Building filtration unit, and Control Building outdoor air cleanup unit. The spray system provides heat removal and fire inhibition for the charcoal absorber section of the units.

A wet sprinkler system is provided for those areas of the Nuclear Island buildings which may contain combustible materials. The sprinkler systems are supplied from the wet standpipe risers and include an alarm check valve and locked open block valve. System design conforms to the requirements of NFPA13 for Group 1, ordinary hazard occupancies.

A wet standpipe system is provided within containment and supplied with water. During normal plant operation the containment standpipe is connected to the condensate distribution system through a branch dedicated for fire protection only. Pressure in the system is boosted to the required level by an automatically initiated operation of the BOP condensate transfer pumps if any connected hose valve is open. The condensate branch includes alarm check valve and locked open block valve upstream of the first hose reel takeoff. When condensate is not available, the containment standpipe shall be supplied with fire water. The supply line is connected to the Auxiliary and Fuel Buildings wet standpipe system and is provided with flow alarm, normally open block valve and two normally closed containment isolation valves.

The Essential Service Water System is connected to the wet standpipe system to provide manual fire fighting capability in areas within hose reach of equipment required for safe plant shutdown in the event of an SSE. The following buildings are affected:

Control Building
Divisions 1, 2 and 3 Diesel Generator Buildings
Fuel and Auxiliary Buildings - up to elevation 11 ft - 0 in.
Reactor Building - up to elevation 59 ft - 7 in.

To limit the hose valves outlet pressure to 100 psig, the pressure restricting hose valves must be properly set on the basis of the BOP fire water pressure at each hose station.

15C.27.8 FMEA Exclusions

The following components were excluded from the analysis.

(1) P&ID Figure 15C.27-1a

All vacuum breakers	Passive components
All check and alarm check valves	Passive components with no safety function
All drain, vent and test valves	Normally closed passive valves with no safety function
All automatic ball drip valves	Passive components with no safety function
All butterfly valves	Passive components, normally locked open or closed
All pressure indicators	Not required for safety function
Gate valve FF009	Passive, with no safety function
Sprinklers, hose valves and hose reels	Passive, with no safety function

(2) P&ID Figure 15C.27-1b

All vacuum breakers	Passive components
All check and alarm check valves	Passive components with no safety function
All drain, vent and test valves	Normally closed passive valves with no safety function
All automatic ball drip valves	Passive components with no safety function

All butterfly valves	Passive components, normally locked open or closed
All pressure indicators and flow switches	Not required for safety function
Sprinklers, hose valves and hose reels	Passive, with no safety function
Valve FF414	Normally closed passive valve

(3) Elementary Diagram Figure 15C.27-2a
All items for reference only.

(4) Elementary Diagram Figure 15C.27-2b

ILDS (B-5) and (B-6), status lights S5A (C-2), S6A (C-2), S1A (E-3), S2A (E-3), S3 (H-2), S4 (H-2), and CAOI (F-5).

(5) Elementary Diagram Figure 15C.27-2c

ILDs (B-5) and (B-6), status lights S5B (C-2), S6B (C-2), S1B (E-2), S2B (E-2), and CAOI (F-6).

15C.27.9 Analysis and Results

The results of the FMEA given in Table 15C.27-1 are summarized as follows:

Number of line items analyzed	28
Number of failure modes analyzed	53
Number classified "A"	10
Number classified "B"	43

The WSFP is capable of performing its safety functions per referenced design documents. The water seal loops are passive devices, and therefore are excluded from this analysis. Hence, this FMEA has verified that a single failure within the system will not degrade nor prohibit this or any other system from performing its safety-related functions.

Table 15C.27-1

WET STANDPIPE FIRE PROTECTION SYSTEM FMEA

GE PROPRIETARY - provided under separate cover

GE PROPRIETARY - provided under separate cover

Figure 15C.27-1a and b. Wet Standpipe Fire Protection System P&ID (FMEA)

GE PROPRIETARY - provided under separate cover

Figure 15C.27-2a through c. Wet Standpipe Fire Protection System Elementary Diagram (FMEA)

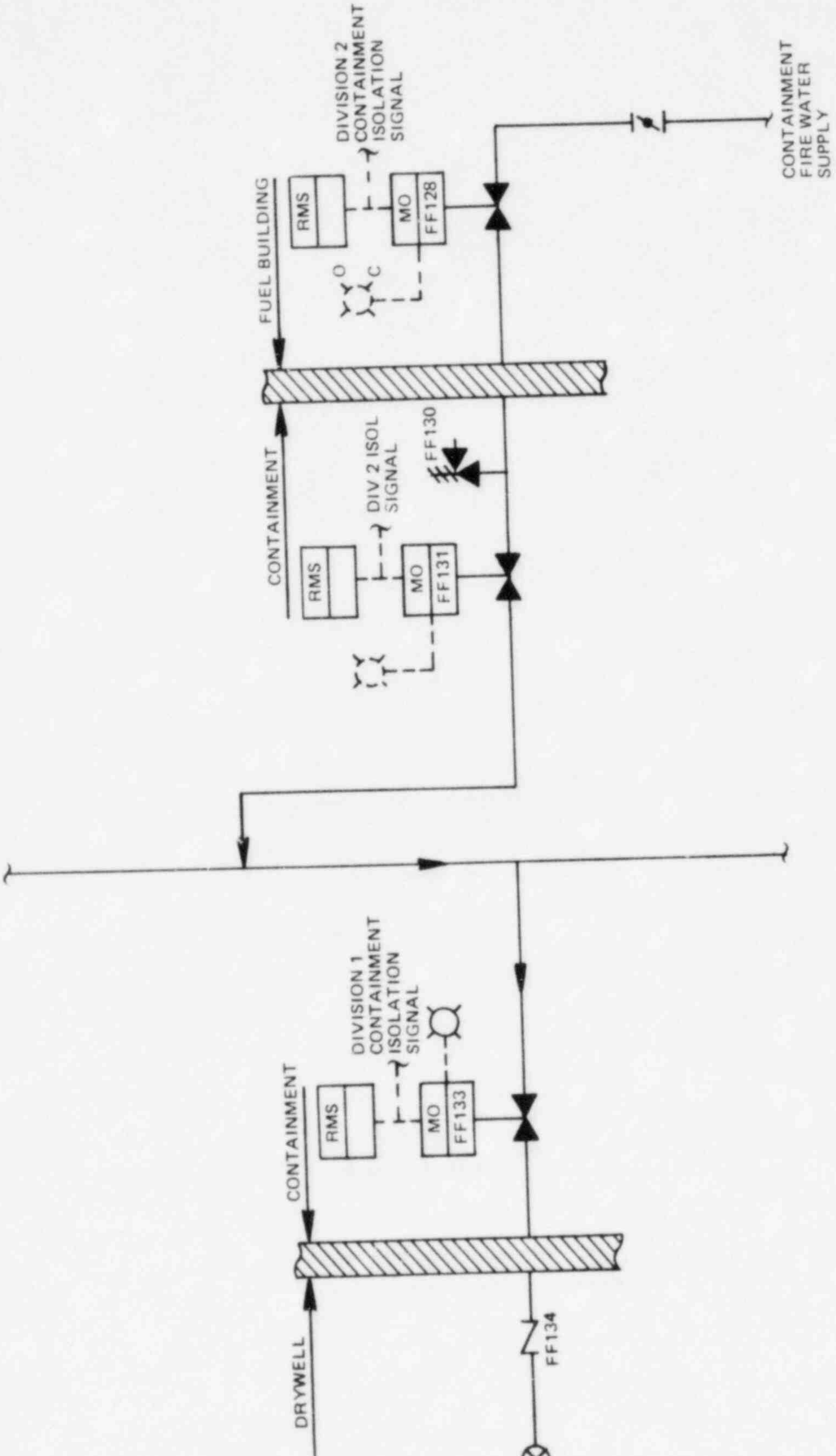


Figure 15C.27-3a. Wet Standpipe Fire Protection System Simplified Block Diagram

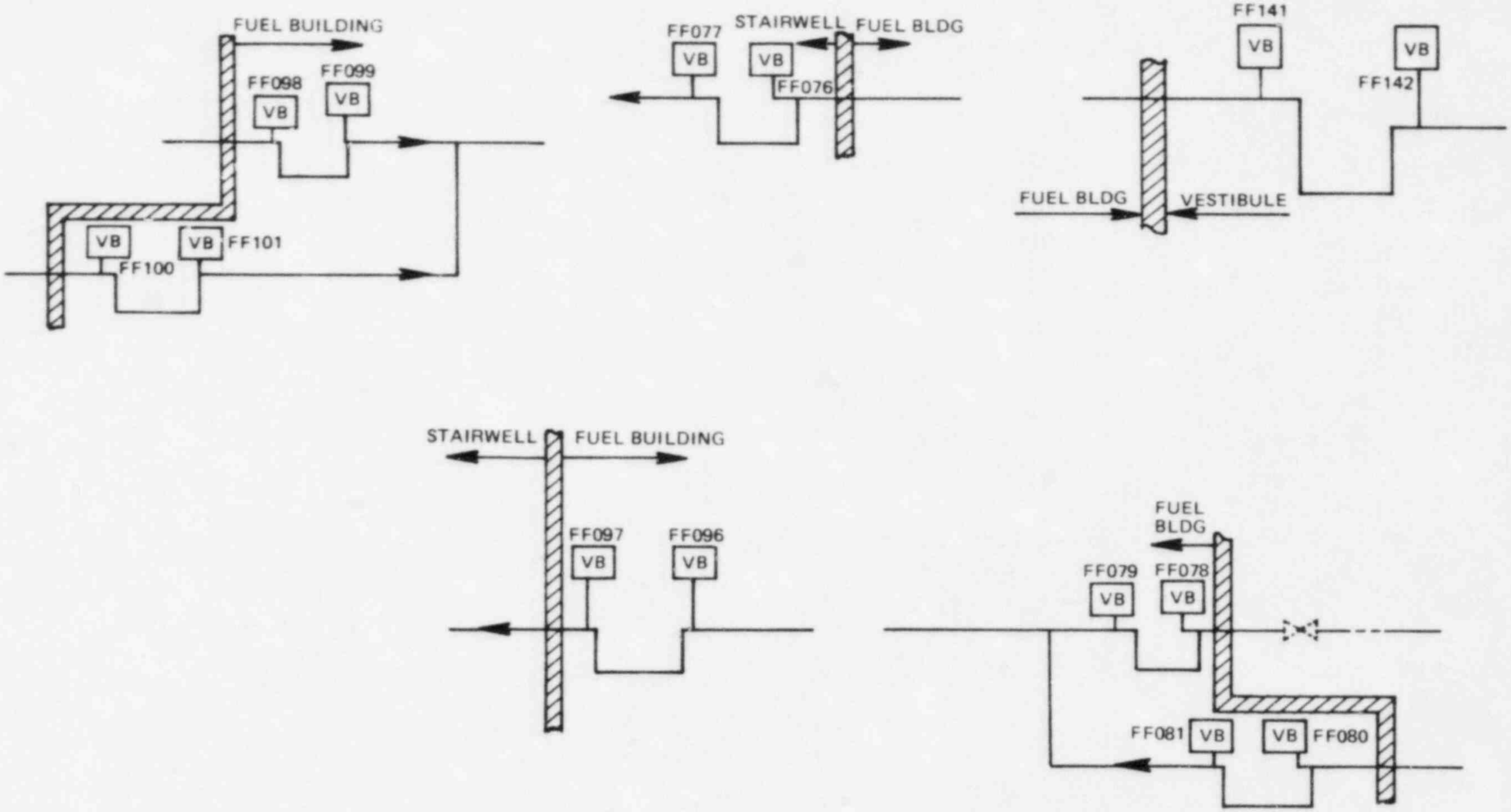


Figure 15C.27-3b. Wet Standpipe Fire Protection System Simplified Block Diagram

15C.28 FUEL BUILDING HVAC (X63)

Applicant to provide

15C.29 AUXILIARY BUILDING ECCS AREA PRESSURE CONTROL SYSTEM (X73)

Applicant to provide

15C.30 AUXILIARY BUILDING ELECTRICAL AND ELEVATOR TOWER HVAC (X73)

Applicant to provide

15C.31 CONTROL BUILDING HVAC (X93)

(To be provided in December 1982)

15C.32 CARBON DIOXIDE FIRE PROTECTION SYSTEM (XA5)

Applicant to provide

ATTACHMENTS

TO

APPENDIX 15C

ATTACHMENT A

TO

APPENDIX 15C

PROCEDURES FOR PREPARATION OF FAILURE
MODES AND EFFECTS ANALYSIS

ATTACHMENT A is PROPRIETARY and is provided under separate cover.

ATTACHMENT B
TO
APPENDIX 15C

FAILURE MODES AND EFFECTS
ANALYSIS TASK SCOPE

Attachment B is PROPRIETARY and is provided under separate cover.

CHAPTER 18

HUMAN FACTORS

(Chapter 18 to be provided
in November 1982)

19.1.2 Chapter 2 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	241.1	2.1	Subsection 19.3.2.1	8

Chapter 2 - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

19.1.3 Chapter 3 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	210.1	3.1	Subsections 3.6.1.1.4, 3.6.2.2.1, 3.6.2.3.1 and 3.6.2.3.2.2	4
	210.2	3.2	Tables 3.9-11 and 3.9-12	
	220.1	3.3	Section 3.7.2.6	
	220.2	3.4	Subsection 19.3.3.3	
	220.3	3.5	Subsection 19.3.3.5	
	220.4	3.6	Table 3.8-3 and Subsection 19.3.3.6	
	220.5	3.7	Subsection 3.8.2.5	
	220.6	3.8	Subsection 19.3.3.8	
	220.7	3.9	Subsection 19.3.3.9	
	220.8	3.10	Subsection 19.3.3.10	
	241.2	3.11	Subsection 19.3.3.11	4
	241.3	3.12	Subsection 19.3.3.12	8
	241.4	3.13	Subsection 19.3.3.13	8
	241.5	3.14	Subsection 19.3.3.14	8
	241.6	3.15	Subsection 19.3.3.15	8
	241.7	3.16	Subsection 3.8.4	8
	241.8	3.17	Subsection 3.8.5.4	8
	241.9	3.18	Subsection 3.8.5.4.1	8
	241.10	3.19	Subsection 19.3.3.19	4
	241.11	3.20	Subsection 19.3.3.20	8
	241.12	3.21	Subsection 3.8.6.2	8
	241.13	3.22	Subsection 3.8.6.2	8
	241.14	3.23	Subsection 19.3.3.23	8
	241.15	3.24	Subsection 3A.1.2	4
	241.16	3.25	Subsection 3A.1.2	4
	241.17	3.26	Subsection 19.3.3.26	8
	241.18	3.27	Subsection 19.3.3.27	8
	241.19	3.28	Subsection 19.3.3.28	8
	241.20	3.29	Subsection 19.3.3.29	8
	241.21	3.30	Subsection 19.3.3.30	8
	241.22	3.31	Subsection 19.3.3.31	5
	241.23	3.32	Subsection 19.3.3.32	8
	241.24	3.33	Subsection 19.3.3.33	8
	241.25	3.34	Subsection 19.3.3.34	5
	241.26	3.35	Subsection 3A.5.2	5
Note 2	251.1	3.36	Subsection 3.5.1.3	4

*Geotechnical

19.1.3 Chapter 3 - Question/Response Index (Continued)

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	270.1	3.37	Tables 3.11-2 through 3.11-9	5
	270.2	3.38	Subsection 3.11.4	5
	270.3	3.39	Subsection 3.11.2.1.3	5
	270.4	3.40	Subsection 3.11.2.1.1	5
Note 2	371.1	3.41	Table 3.10-1	4

**Environmental Qualification

Chapter - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

3. See Section 3BO.1 for Appendix 3B Question/Response Index.

19.1.6 Chapter 6 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	480.1	6.1		8

Chapter 6 - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

19.1.7 Chapter 7 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	420.1	7.1	Subsection 19.3.7.1	4
Note 2	420.2	7.2	Subsection 19.3.7.2	4
Note 2	420.3	7.3	Subsection 7.2.2	8
Note 2	420.4	7.4	Subsection 19.3.7.3	4
Note 2	420.5	7.5	Subsection 19.3.7.5	4
Note 2	420.6	7.6	Subsection 19.3.7.6	4
Note 2	420.7	7.7	Subsection 7.3.2	8
Note 2	420.8	7.8	Subsection 7.1.2.4	4
Note 2	420.9	7.9	Subsection 19.3.7.9	4
Note 2	420.10	7.10	Subsection 7.4.1.2	4
Note 2	420.11	7.11	Table 7.5-1, Subsection 19.3.7.11	4
Note 2	420.12	7.12	Subsection 19.3.7.12	4

Chapter 7 - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

19.1.9 Chapter 9 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	280.1	9.1	Subsection 9.5.1.3 (FMEA portion in 9/82)	8
	410.1	9.2	Subsection 9.1.2.3.2	4
	410.2	9.3	Subsections 9.1.4.2.2 and 19.3.9.3	8
	410.3	9.4	Subsection 9.1.4.3	
	410.4	9.5	Subsection 9.1.4.3	4
	410.5	9.6	Subsection 9.1.4.5.1	4
	410.6	9.7	Subsection 9.2.1.3.1	4
	410.7	9.8	Subsection 9.2.6	8
	410.8	9.9	Subsection 9.3.5.3	
	410.9	9.10	Subsection 9.4.2.1	4
	410.10	9.11	Subsection 9.4.3.1	4
	430.2	9.12	Subsection 19.3.9.12	4
Note 2	430.3	9.13	Subsection 9.5.8.1	4

Chapter 9 - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

19.1.12 Chapter 12 - Question/Response Index

NRC Transmittal	NRC Question Number	GESSAR II Question Number	Disposition[1]	GESSAR II Revision Number
Note 2	471.1	12.1	Subsection 12.2.1.1	4
Note 2	471.2	12.2	Subsection 12.2.1.2.7.2; Tables 12.2-6, 12.2-7 and 12.2-16; and Figure 12.2-2.	8
Note 2	471.3	12.3	Subsection 19.3.12.3 (Subsections 12.3.2.3 and 12.3.4)	4
Note 2	471.4	12.4	Subsection 12.3.3	4

Chapter 12 - Question/Response Index Notes

1. Subsections shown in parentheses reference the corresponding Chapter 19 subsection which details the answer to the question.

2. Darrell G. Eisenhut to Glenn G. Sherwood, "Acceptance Review of Application for Final Design Approval for 238 Nuclear Island," December 9, 1981.

19.3.2 Chapter 2 - Responses

19.3.2.1 QUESTION/RESPONSE 2.1 (241.1)

QUESTION 2.1

Provide details of the broad spectrum of foundation conditions that have been used to arrive at the design loads. Discuss the procedure used to select the foundation conditions and the method of computing the design loads. Present the information in the appropriate sections of the SSAR. (2.5.1)

RESPONSE 2.1

The details of the range of foundation conditions used to arrive at the design loads were provided during the review of PDA GESSAR. The procedures used to select the foundations and the method of computing the design loads were also provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic design.

19.3.3.12 QUESTION/RESPONSE 3.12 (241.3)

QUESTION 3.12

Explain how the finite-element representation has been used to model all the supporting medium conditions. Provide appropriate figures, if necessary. (3.7.1.4.1)

RESPONSE 3.12

Explanation of finite-element model for the supporting medium conditions was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic design.

19.3.3.13 QUESTION/RESPONSE 3.13 (241.4)

QUESTION 3.13

Explain and justify the use of a single curve representative of sandy soil properties for representing other sands, clays and silty soils that may be encountered at various sites. Provide this information in Subsection 3.7.1.3 of the SSAR. (3.7.1.4.2)

RESPONSE 3.13

The explanation and justification of a single-curve representative of sandy soil properties was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic analysis.

19.3.3.14 QUESTION RESPONSE 3.14 (241.5)

QUESTION 3.14

Based on a review of the range of soil properties used in the GESSAR II seismic analysis, we do not find an adequate basis to agree that the "uncertainties in soil properties and frequencies are adequately accounted for in the envelope design." We believe that the envelope design will meet all the requirements of a specific design based on site specific geotechnical parameters, and the parameters will be based on state-of-the-art soil exploration and seismic analysis for a specific site. (3.8)

RESPONSE 3.14

The uncertainties in soil properties and frequencies were reviewed during PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic analysis.

19.3.3.15 QUESTION/RESPONSE 3.15 (241.6)

QUESTION 3.15

Describe in detail the procedure followed to arrive at the soil properties used in the model for LOCA/SRV loads analysis. (3.8.2.3.9)

RESPONSE 3.15

The procedure used to arrive at the soil properties was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis and methods procedures of GESSAR provide an acceptable basis for system and subsystem seismic analysis.

19.3.3.16 QUESTION/RESPONSE 3.16 (241.7)

QUESTION 3.16

Provide a plan and profile of Category I pipelines that will be buried in soil. (3.8.4)

RESPONSE 3.16

Response to this question is provided in Subsection 3.8.4.

19.3.3.17 QUESTION/RESPONSE 3.17 (241.8)

QUESTION 3.17

Discuss how the effects of soil and structural settlement are considered, and state the acceptance criteria for the proposed values. (3.8.5.4)

RESPONSE 3.17

Response to this question is provided in Subsection 3.8.5.4.]

19.3.3.18 QUESTION/RESPONSE 3.18 (241.9)

QUESTION 3.18

Describe in detail the procedure used for calculating the subgrade stiffness used in NASTRAN. Discuss the applicability of the stiffness values to various sites with different soil conditions. (3.8.5.4.1)

RESPONSE 3.18

Response to this question is provided in Subsection 3.8.5.4.1.]

19.3.3.20 QUESTION/RESPONSE 3.20 (241.11)

QUESTION 3.20

The factors of safety against sliding given in Figure 3.8-78 for the reactor building, auxiliary building and control building are below those of the Standard Review Plan (Subsection 3.8.5). Justify the use of such low factors of safety. (3.8.5.4.1)

RESPONSE 3.20

This question has been replaced by NRC question 220.43.*

*Transmitted by NRC letter from T. M. Novak to Glenn G. Sherwood (GE), "Request for Additional Information Regarding the General Electric Application for an FDA for a standardized Nuclear Island (GESSAR II), August 25, 1982.

19.3.3.21 QUESTION/RESPONSE 3.21 (241.12)

QUESTION 3.21

Describe how the ultimate and residual soil settlements were calculated, and discuss the applicability of these computations to a range of site conditions. Provide the required orientation, location and purpose of settlement points on Figure 3.8-91. Explain how settlement values will be interpreted, and establish limiting criteria. (3.8.6.2)

RESPONSE 3.21

Response to this question is provided in Subsection 3.8.6.2. (Note: Figure 3.8-91 of the pre-docket version of GESSAR II is Figure 3.8-88 in the docket version.)

19.3.3.22 QUESTION/RESPONSE 3.22 (241.13)

QUESTION 3.22

Clarify the meaning of the last sentence of Subsection 3.8.6.2, which states that the actual soil properties will be compared with the required soil properties in the building design stress reports. Describe how the required soil properties will be determined and how comparisons will be made. (3.8.6.2)

RESPONSE 3.22

Response to this question is provided in Subsection 3.8.6.2.]

19.3.3.23 QUESTION/RESPONSE 3.23 (241.14)

QUESTION 3.23

The staff does not agree that site-unique seismic analysis or review by regulatory agencies should not be required. It is the position of the staff that the applicant must demonstrate that, based on actual geotechnical site parameters and the state-of-the-art at the time of submission of an FSAR, the seismic analysis results given in GESSAR II envelope the results of the seismic analysis for actual sites. (3A.1.2)

RESPONSE 3.23

The staff did not require a site-unique analysis for Phipps Bends (Subsection 3.7.2 of NUREG-0101) which referenced the GESSAR SER (NUREG-75/110). The staff concluded that the Phipps Bend application contained sufficient information that assured that the six GESSAR conditions (to ensure seismic design adequacy) are met. Since GESSAR II added two additional qualifying conditions that provide even more conservatism, site-unique seismic analysis should not be required.

19.3.3.26 QUESTION/RESPONSE 3.26 (241.17)

QUESTION 3.26

You state that in your analysis no special restrictions were provided for parameters, such as variation of water table, material density, material composition or soil profile. Discuss how your analysis is applicable to many potential sites in light of this lack of parameter variation. In the seismic analysis for layered soil sites, not only the range of parameters for soil properties is important, but also the sequence in which the soil layering exists. Explain how soil layering is accounted for in your analysis. The water table elevation not only affects strain dependent material properties for sandy soils, but also the compressional wave velocity for the soil-structural interaction analysis due to vertical excitation. Justify in detail your approach to these items. (3A.1.2)

RESPONSE 3.26

No special restrictions were provided for these parameters since the study described in Appendix 3A of the GESSAR PDA showed that they did not affect overall results. Thus, the GESSAR II analysis is applicable to many potential sites. Condition 7 addresses layered soil sites with parameters which have very abrupt variations with depth. In these instances, analysis with site-unique properties will be performed to confirm the applicability of the generic analysis.

19.3.3.27 QUESTION/RESPONSE 3.27 (241.18)

QUESTION 3.27

Your description of the soil damping curve used in the analysis is different than that presented in Subsection 3.7.1.4.2. The two descriptions should be consistent. Describe how the damping properties for clays, tills, or other materials have been accounted for. Also, indicate how you plan to justify your damping curve on a site-specific application. (3A.2.2)

RESPONSE 3.27

The soil damping curve, Figure 3.7-21, referenced in Subsection 3.7.1.4.2 has been deleted. An appropriate reference to the soil damping curve of Appendix 3A has been added to this subsection.

The basis for the soil damping curve was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic analysis.

19.3.3.28 QUESTION/RESPONSE 3.28 (241.19)

QUESTION 3.28

Most of the soil profiles, other than fixed base conditions, that you have analyzed are 75 ft. deep. What is the bases for assuming this profile represents a wide range of soil profile conditions? (3A.2.2)

RESPONSE 3.28

The bases for the soil depth range was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic analysis.

19.3.3.29 QUESTION/RESPONSE 3.29 (241.20)

QUESTION 3.29

Your selection of shear wave velocity profiles does not seem to include a wide range of soil profiles. Your lower bound soil properties are very stiff below the foundation elevation of the reactor building. Other profiles are much stiffer and many of these are close to being representative of rock sites. Please justify your selections. (3A.2.2)

RESPONSE 3.29

Justification of the shear wave velocity profiles was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem seismic analysis.

19.3.3.30 QUESTION/RESPONSE 3.30 (241.21)

QUESTION 3.30

Figure 3A-5 shows shear wave velocity profiles to a depth of 300 ft. Most of your profiles, other than fixed base, are 75 ft deep. One profile is 150 ft. deep. What values of shear wave velocities were used below the depth of the analyzed profile, and what was the basis for selecting these values? (3A.2.2)

RESPONSE 3.30

Justification of the shear wave velocity profiles was provided during the review of PDA GESSAR. The staff concluded in the GESSAR SER (Subsection 3.7.3 of NUREG-75/110) that the seismic analysis methods and procedures of GESSAR provide an acceptable basis for system and subsystem analysis.

19.3.6 Chapter 6 - Responses

19.3.6.1 QUESTION/RESPONSE 6.1 (480.1)

QUESTION 6.1

Per the requirements of Regulatory Guide 1.70, Section 6.2.2.3, provide a failure mode and effects analysis of the containment heat removal systems. (6.2.2.3)

RESPONSE 6.1

Response to this question is included in Subsection 6.2.2.3.]

19.3.7 Chapter 7 - Responses

19.3.7.1 QUESTION/RESPONSE 7.1 (420.1)

QUESTION 7.1

Throughout Chapter 7.0 you refer to Figures 7A.X.X (See Table 7.8-1 in Section 7.8, Page 7.8-3 for an example). Please clarify these references, since these figures cannot be found in the SSAR. (7.0)

RESPONSE 7.1

These figures, the I&C Elementary Diagrams, were included at docketing.

19.3.7.3 QUESTION/RESPONSE 7.3 (420.3)

QUESTION 7.3

As per Regulatory Guide 1.70, include a failure mode and effects analysis. (7.2.2)

RESPONSE 7.3

Response to this question is included in Subsection 7.2.2.]

19.3.9 Chapter 9 - Responses

19.3.9.1 QUESTION/RESPONSE 9.1 (280.1)

QUESTION 9.1

As per Regulatory Guide 1.70, a failure mode and effects analysis should be provided that demonstrates that operation of the fire protection system in areas containing engineered safety features would not produce an unsafe condition or preclude safe shutdown. The effects of firefighting activities and fire suppression agents on safety systems should be discussed. An analysis of the fire detection and protection system with regard to design features to withstand the effects of single failures should be included. (9.5.1.3)

RESPONSE 9.1

The response to this question is provided in Subsection 9.5.1.3.]

19.3.9.4 QUESTION/RESPONSE 9.4 (410.3)

QUESTION 9.4

As per Regulatory Guide 1.70, the results of a failure mode and effects analysis should be presented to demonstrate that the individual subsystems and components, including controls and interlocks, are designed to meet the single-failure criterion without compromising the capability of the system to perform its safety function. (9.1.4.3)

RESPONSE 9.4

The response to this question is provided in Subsection 9.1.4.3.]

19.3.9.9 QUESTION/RESPONSE 9.9 (410.8)

QUESTION 9.9

The results of a failure mode and effects analysis should be presented to demonstrate that the system can meet the single-failure criterion without compromising the shutdown capability of the system. (The reference to Section 15A.6.6 is not adequate.) (9.3.5.3)

RESPONSE 9.9

The response to this question is provided in Subsection 9.3.5.3.]

19.3.7.7 QUESTION/RESPONSE 7.7 (420.7)

QUESTION 7.7

As per Regulatory Guide 1.70, include the failure mode and effects analyses. (7.3.2)

RESPONSE 7.7

Response to this question is included in Subsection 7.3.2.

19.3.12.2 QUESTION/RESPONSE 12.2 (471.2)

QUESTION 12.2

Provide the missing information in Tables 12.2-6, 12.2-7, 12.2-16 and 12.2-18. (12.2.1.2)

RESPONSE 12.2

Response to this question is provided in Subsection 12.2.1.2.7.2. (Note: Table 12.2-18 is replaced by Figure 12.2-2).