OFFSHORE POWER SYSTEMS

DOCKET NO. STN 50-437

INSTRUCTIONS FOR ENTERING AMENDMENT NO. 28

IN THE PLANT DESIGN REPORT

1. Replace Master Table of Contents, pages vii, viii in the front of each of the eight volumes of the Plant Design Report

2. Remove and insert pages in the text of the Plant Design Report in accordance with the following tabulation.

Remove Page(s)		Insert Page(s)
	Chapter 1	
1.1-1, 2		1.1-1, 2
1.1-5, 6		1.1-5, 6
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3.8-8d, 8d(a)		3.8-8d, 8d(a)
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	Chapter 8	
8.1-5, 6		8.1-5, 6
8.1-7, 8		8.1-7, 8
	Chapter 12	
12.1-17, 17a		12.1-17, 17a
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13.1-1 thru 13.1-24		13.1-1 thru 13.1-36
Figure 13.1-1		Figure 13.1-1

Remove Page(s)		Insert Page(s)
	Chapter 17	
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Figure 17.1-2		Figure 17.1-2
	Appendix B	
B-v		B-v
---		B.19.1-1
---		B.19.2-1 thru 7

3. Add a new Appendix C in the back of Volume 8 consisting of an index tab, pages C-i, C-ii and pages C-1 through C-160.

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Plant Design Report is submitted to support the application of Offshore Power Systems, herein referred to as the Applicant, for a Manufacturing License for eight floating nuclear power plants.

Manufacture of the eight floating nuclear plants (FNP's) is subject to utility delivery requirements at the time a unit is ordered. Under present manufacturing plans the first FNP will be ready for delivery no earlier than 1991. The remaining seven FNPs will be available for delivery at a maximum rate of one per year beginning in 1992. 28

The Floating Nuclear Plant concept, incorporating centralized manufacture and flexibility in siting, makes possible a significant reduction in the overall schedule for nuclear power plant installation. This is possible through generic licensing of the plant independently of site licensing, as provided in Appendix M to 10 CFR 50, issued November 2, 1973.

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To permit generic review of the plant, the Applicant has included in Chapter 2 of this report a description of those plant-site interfaces which define an envelope of acceptable site parameters. Sites having characteristics falling within this envelope will be compatible with the plant design bases. Site envelope information contained in this report will enable owners to select acceptable sites. In addition, expertise gained by the Applicant through the conduct of environmental and siting feasibility studies can be made available to assist owners in locating acceptable siting areas. Adequate flexibility with respect to siting has been incorporated in the plant design such that a spectrum of sites will be acceptable. 15

Each Owner's application will include information pertaining to his specific site, plant staffing and training, quality assurance program, and other unique aspects. The Owner's application will contain sufficient information to demonstrate that the characteristics of the site at which the reactor is 15 to be operated falls within the postulated site parameters specified in the manufacturing license.

Manufacture of plants on order will continue while site-related surveys, investigations, and licensing procedures are being implemented by each owner.

1.1.3.2 Power Output

Each unit will be rated at a net core power output of 3411 MWt plus 14 MWt net of heat from non-reactor sources. The Engineering Safety Features design rating for each unit will be 3579 MWt.

Although the license application is for 3411 MWt net core per unit, all safety systems, including the containment and engineered safety features, will be designed and evaluated for operation at the ultimate power level, 3579 MWt. This power is used in the analysis and evaluation of the major structures, systems, and components of the plant which bear significantly on the acceptability of a site. The thermal-hydraulic and nuclear aspects of the core have been evaluated on the basis of a core thermal output of 3411 MWt.

1.1.4 SCHEDULE FOR COMPLETION AND COMMERCIAL OPERATION

Manufacture of the eight floating nuclear plants will be completed during the period commencing no earlier than July, 1991 and ending no later than July, 1999, with manufacture of the first plant in the previously prepared 28
manufacturing facility to begin no earlier than 1985. Assuming each owner obtains the necessary permits and licenses in a timely manner, plant commercial operation should follow completion of manufacture by no more than eighteen months.

1.1.5 ORGANIZATION OF CONTENTS

1.1.5.1 Subdivision

This report is organized into 17 chapters, each of which consists of a number of sections that are numerically identified by two numerals separated by a decimal (e.g., 3.4 is the fourth section of chapter three). Sections are further sub-divided into subsections that are numerically identified by three numerals separated by decimals (e.g., 3.4.1). All further subdivisions numerically identified in the manner described above are paragraphs (e.g., 3.4.1.1, 3.4.1.1.1, and 3.4.1.1.1.1).

1.1.5.2 Standard Format

This report has been written to comply with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" as issued by the Atomic Energy Commision in February, 1972.

This report uses the same chapters, sections, subsections, and paragraph headings used in the standard format. Where information has been presented that is not specifically requested by the standard format and this information is identified numerically (chapter, section, subsection, or paragraph), this information is presented under the appropriate general heading as a subdivision following all subdivisions containing information specifically requested by the standard format. (For example, subsection 1.1.5 is not requested in the standard format. Since it apparently belonged in section 1.1, it was placed after the four subsections containing information requested by the standard format).

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Structural Member		μ
2.	Steel	
	Members subjected to flexure, compression and shear	10
	Members with non-ductile failure mode such as column buckling	1.3

d. Dynamic increase factors (DIF) appropriate for the strain rates and design temperatures involved are applied to static material strengths of steel and concrete for purposes of determining section strength but shall not exceed the following:

Material		DIF
1.	Reinforcing Steel		25
	40 ksi yield strength	1.20
	50 ksi yield strength	1.15
	60 ksi yield strength	1.10
2.	Concrete	
	Axial and flexural compression	1.25
	Shear	1.10
3.	Structural Steel	
	40 ksi or less yield strength	1.20
	50 ksi yield strength	1.15
	60 ksi yield strength	1.10

The above dynamic increase factors are established from References 1, 2, 5, 6, 7 and 8.

5. Design of steel members for cyclic loading due to wave motion will be based on Appendix B of the AISC Specification for the Design, Fabrication and Ere^tion of Structural Steel for Buildings.

Design of concrete members for cyclic loading due to wave motion will be based on ACI Committee Report 215, "Consideration for Design of Concrete Structures Subjected to Fatigue Loadings", ACI Journal, March 1974.

6. For shell type steel structures where buckling is a design consideration, the basic allowable stress S of Table 3.8.1-1B is established from the critical buckling stress divided by an overall factor of safety of 1.92 (per AISC column formulas). The critical buckling stress is determined by the classical linear bifurcation analysis of the shell structure reduced by margins which reflect the difference between theoretical and actual buckling capacities. This is the same approach as the buckling criteria used for the design of the steel containment shell as described in Appendix 3F.

7. Design of steel embedments will be in accordance with Appendix B of ACI-349 (1979 Supplement). This Appendix was published as a proposed addition to the code in the ACI Journal, August 1978 and was adopted by ACI in July 1979. Design of the steel embedment will account for base plate flexibility. Load factors and capacity reduction facotrs will be those identified in ACI-349-76 as modified by Regulatory Guide 1.142 and Appendix B. 28

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3.8.1.1.6 Structural Analysis and Design

Category I Structures are analyzed and designed for the loads described in paragraph 3.8.1.1.4.

Seismic loads are determined by dynamic analysis. The ANSYS Computer program is used for multidegree of freedom models. This program is described in Appendix 3A.

3.8.1.1.7 Materials and Quality Control

Refer to subsection 3.8.3.

3.8.1.2 Category II Structures

3.8.1.2.1 General Description

Category II Structures include the turbine area, power transmission area, and service areas.

The turbine area is structural steel. It houses the turbine generator and its associated auxiliary equipment and is approximately 277 feet in length and 156 feet in width.

LOCATION OF INFORMATION RELATING TO DIVISION 1 REGULATORY GUIDES (CONT)

Regulatory Guide Number	Title	Location of Information
1.55	Concrete Placement in Category I Structures	3.8.3
1.56	Maintenance of Water Purity in Boiling Water Reactors	Note 1
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	3.8.2.6.2
1.58 (Rev. 1)	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	Note 4 28
1.59 (Rev. 2)	Design Basis Floods for Nuclear Power Plants	Note 17
1.60 (Rev. 1)	Design Response Spectra for Seismic Design of Nuclear Power Plants	3.7
1.61	Damping Values for Seismic Design of Nuclear Power Plants	3.7
1.62	Manual Initiation of Protective Actions	7.1.9
1.63 (Rev. 2)	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	Note 4
1.64 (Rev. 1)	Quality Assurance Program Requirements for the Design of Nuclear Power Plants	Note 4
1.65	Materials and Inspections for Reactor Vessel Closure Studs	Note 7
1.66	Nondestructive Examination of Tubular Products	Note 8
1.67	Installation of Overpressure Protection Devices	5.2.1, 3.9.2.5
1.68	Preoperational and Initial Start-up Test Programs for Water-Cooled Power Reactors	14
1.68.1	Preoperational and Initial Start-up Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	Note 1
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	Note 2
1.69	Concrete Radiation Shields for Nuclear Power Plants	3.8.1.1.3

LOCATION OF INFORMATION RELATING TO DIVISION 1 REGULATORY GUIDES (CONT)

Regulatory Guide Number	Title	Location of Information	
1.70 and 1.70.x Series	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	Note 16	17
1.71	Welder Qualification for Areas of Limited Accessibility	Note 4	
1.72	Spray Pond Plastic Pipe	Note 1	
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	8.1.4	
1.74	Quality Assurance Terms and Definitions	Note 4	
1.75 (Rev. 1)	Physical Independence of Electric Systems	8.1.4	17
1.76	Design Basis Tornado for Nuclear Power Plants	3.3	
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	Note 12	
1.78	Assumptions for Evaluating the Habitability of A Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	6.5.4	
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	6.3.4.2	
1.80	Preoperational Testing of Instrument Air Systems	Note 13	
1.81 (Rev. 1)	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	Note 1	17
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems	6.2.2.7	
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	Note 9	
1.84 (Rev. 2)	Code Case Acceptability - ASME Section III Design and Fabrication	Note 4	17

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Regulatory Guide Number	Title	Location of Information
1.85 (Rev. 2)	Code Case Acceptability - ASME Section III Materials	Note 4
1.86	Termination of Operating Licenses for Nuclear Reactors	Note 2
1.87 (Rev. 1)	Construction Criteria for Class I Components in Elevated Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)	Note 1
1.88	Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records	Note 4
1.89	Qualification of Class IE Equipment for Nuclear Power Plants	8.1.4
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	Note 1
1.91 (Rev. 1)	Evaluation of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plant Sites	2.9 Appendix 2A
1.92	Combination of Modes and Spatial Components in Seismic Response Analysis	3.7
1.93	Availability of Electric Power Sources	Note 14
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	Note 4
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	6.4.5
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	Note 1

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LOCATION OF INFORMATION RELATING TO DIVISION 1 REGULATORY GUIDES (CONT)

Regulatory Guide Number	Title	Location of Information
1.97 (Rev. 1)	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident	Note 18
1.99 (Rev. 1)	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	Note 24
1.101 (Rev. 1)	Emergency Planning for Nuclear Power Plants	Note 2
1.102 (Rev. 1)	Flood Protection for Nuclear Power Plants	Note 17
1.105 (Rev. 1)	Instrument Setpoints	Note 19
1.108 (Rev. 1)	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	Note 2
1.114 (Rev. 1)	Guidance on Being Operator at the Controls of a Nuclear Power Plant	Note 2
1.115 (Rev. 1)	Protection Against Low-Trajectory Turbine Missiles	Note 20, 3.5.4
1.116	Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems.	Note 4
1.117 (Rev. 1)	Tornado Design Classification	3.5.2, 3.5.3
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	Note 21
1.123 (Rev. 1)	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	Note 4
1.124 (Rev. 1)	Service Limits and Loading Combinations for Class 1 Linear Type Component Supports	Note 22
1.127 (Rev. 1)	Inspection of Water Control Structures Associated with Nuclear Power Plants	Note 2
1.130	Design Limits and Loading Combinations for Class 1 Plate and Shell Type Component Supports	Note 23

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LOCATION OF INFORMATION RELATING TO DIVISION 1 REGULATORY GUIDES (CONT.)

Regulatory Guide Number	Title	Location of Information	
1.144 (Rev. 1)	Auditing of Quality Assurance Programs for Nuclear Power Plants	Note 4	28
1.146	Qualifications of Quality Assurance Program Audit Personnel for Nuclear Power Plants	Note 4	

8.1.4 DESIGN CRITERIA

The design bases, criteria, safety guides, standards, and other documents that are implemented in the design of this unit are:

1. Industry Manufacturing Standards.

   a. ANSI C37 Switchgear

   b. ANSI C50 Rotating Electrical Machinery

   c. ANSI C57 Transformers, Regulators, and Reactors

   d. IPCEA P-46-426 Power Cable Ampacities

   e. NEMA SG3-1965 Low Voltage Power Circuit Breakers

   f. NEMA S64-1968 AC High Voltage Circuit Breakers

   g. NEMA SG5-1967 Power Switching Assemblies

   h. NEMA SG5-1966 Power Switching Equipment

   i. NEMA R12-1971 Battery Charger - General Purpose & Communications

   j. NEMA IB-1-1971 Defintiions for Lead Acid Storage Batteries

k. NEMA TRI-1968 Transformers, Regulators and Reactors

l. NEMA TR-P3-1970 Guide for Preparation of Specifications for Large Power Transformers

m. NEMA MGI-1967 Motors and Generators

2. The power supply for the reactor protection system and the safety systems will be in accord with Criteria 17 and 18 of 10 CFR 50, Appendix A.

3. Additional design criteria are as follows:

   a. IEEE 279-1971 - Criteria for Nuclear Power Plant Protection Systems.

   b. IEEE 288-1969 - Guide for Induction Motor Protection.

   c. IEEE 308-1971 - Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations.

   d. IEEE 317-1976 - Electrical Penetration Assembly in Containment Structures for Nuclear Fueled Power Generating Stations.

   e. IEEE 323-1974, "IEEE Standard for Qualifying Class IE Equipment 28
   for Nuclear Power Generation Stations," and NUREG-0558, "Interim

Staff Position on Environmental Qualifications of Safety-Related Electrical Equipment." Class IE Equipment supplied by the Applicant will be qualified in accordance with these requirements. Class IE 28
NSSS equipment supplied by Westinghouse will be qualified in accordance with generic agreements between Westinghouse and NRC.

f. IEEE 334-1971 - Trial Use Guide for Type Tests of Continuous-Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations.

g. IEEE 336-1971 - Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During The Construction of Nuclear Power Generating Stations.

h. IEEE 344-1971 - Guide for Seismic Qualifications of Class 1 Electrical Equipment for Nuclear Power Generating Stations.

i. IEEE 383-1974 - Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations.

j. IEEE 387-1972 - Trial Use Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations.

k. IEEE 379-1972 - Trial Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems.

4. The following Guides will be used as Design Criteria in establishing the plant design:

   a. AEC Regulatory Guide 1.6 - Independence between Redundant Standby (Onsite) power Sources and between their Distribution Systems.

   b. AEC Regulatory Guide 1.9 - Selection of Diesel Generator Set Capacity for Standby Power Supplies.*

   c. AEC Regulatory Guide 1.22 - Periodic Testing of Protection System Actuation Function. Refer to Section 8.3.1.2.1 for application to Power System. 9

   d. AEC Regulatory Guide 1.30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment.

   e. AEC Regulatory Guide 1.32 - Use of IEEE STD 308-1974, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations."

* The diesel-electric generating system fully meets the intent of Regulatory Guide 1.9 except that calculation of estimated electric loads given in Table 8.3-1 and 8.3-2 of this report is based on a motor efficiency of 92%. This efficiency is based on the average efficiency obtained on similar motors applied on other installations.

the purpose of shield design) to contain the coolant activity (based on 1% failed fuel assumption) as presented in Table 11.1-2 of RESAR 3.

Pumps	Heat Exchangers	Tanks
Charging pumps	Seal water	Floor Drain
Waste holdup tank pumps	Letdown	Waste holdup
Waste evaporator feed pumps Floor Drain tank pump	Moderating	Waste evap. condensate
Waste evaporator condensate	Letdown Chiller	Chemical drain
Chemical drain tank	Letdown reheat	

The volume control tank and recycle holdup tanks also contain radioactive sources and must be shielded. Sources used to design the shielding for the volume control tank are presented in Table 11.1-3 of RESAR 3. Sources for the recycle holdup tanks are given in Table 12.1-2A. The liquid sources in Table 12.1-2A are applicable for the recycle evaporator feed pump. For the cask decontamination tank and the laundry and hot shower tank, past plant experience was used as a basis for determining sources for use in designing shielding. It was assumed, based on plant experience, that the contact dose rate for each tank would not exceed 100 mr/hr and that the source consisted of photons with an energy of 1 mev. Calculations were performed showing that a tank full of water with a specific activity of 0.2 μC/cc (emitting a single 1 mev gamma per desintegration) would give a | 28
contact dose rate of 100 mr/hr.

The above activities were also employed to design shielding for the pumps associated with each tank respectively. 24

For the refueling water storage tank, detailed conservative analyses (in conjunction with operating plant data) showed that a water contact dose rate of <15 mr/hr could be maintained by operating cleanup systems during refueling. 24

12.1.3.3 Auxiliary System Sources

Specific sources for the following equipment have not been calculated: demineralizers, evaporators, filters, hydrogen recombiners, concentrates holding tank and pumps, and the spent resin storage tank. Preliminary shield thicknesses for this equipment were estimated based on operating experience at other power plants. For the Waste Gas Decay Tanks, the sources given in Table 11.3-6 of RESAR 3 were used.

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CHAPTER 13

CONDUCT OF OPERATIONS

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CHAPTER 13

CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

13.1.1 CORPORATE ORGANIZATION

13.1.1.1 Corporate Functions, Responsibilities, and Authorities

The major organizations engaged in the design, construction, quality assurance, and pre-operational testing are:

Offshore Power Systems - Jacksonville, Florida

Westinghouse Electric Corporation - Pittsburgh, Pennsylvania

The functions, responsibilities, and authorities of these organizations are described in Chapter 1 with their quality assurance programs described in Chapter 17. Applicant's technical qualifications are described in detail in paragraph 13.1.1.2.

13.1.1.2 Applicant's Organization

The responsibility for the design, manufacturing, testing, and quality assurance function is that of the President of Offshore Power Systems, who has delegated those responsibilities to the Director, Power Systems | 28

Technology; Director, Marine Design and Director, Operations. Figure 13.1-1 shows the applicant's functional organization.

13.1.1.2.1 Power Systems Technology 28

Power Systems Technology is headed by the Director, Power Systems Technology and 28
includes the following divisions: Electrical and Control Engineering, Structural
Engineering, Mechanical Engineering, Nuclear Engineering and Quality Assurance. 28
The educational background and total years of engineering or science experience are as follows:

No. Baccalaureate Degrees	27	
No. Master Degrees	19	
No. Doctorate Degrees	4	
No. Non-Degreed Technical Personnel	5	
Total Technical Personnel	55	28
Total Man-Years of Technical Experience	967	

The forecasted peak level of staffing during the final design/manufacturing phase is approximately 130 technical personnel. This estimate is based in part on the Applicant's actual level of staffing during the FNP preliminary design phase (reported in Appendix C).

1. <u>Electrical and Control Engineering</u> | 28

Electrical and Control Engineering Division is responsible for electrical and control design of the total plant including participation by Engineering in selection of suppliers. Specific responsibilities of this division include:

Electrical Power Systems Engineering

a. Select arrangement and determine size of all electrical distribution systems including interface with owner on connection to the utility grid.

b. Approve location and arrangement of all electrical power equipment in the plant, including cable tray and conduit for Power and Control Systems.

c. Determine ratings of all electrical equipment including diesel generators.

d. Prepare equipment and system specifications for all power equipment including diesel generator, power, control and instrumentation cable 28 and establish the quality level and acceptance standards for the equipment.

e. Select cable insulation, size and quantity.

f. Provide physical requirements to Design and Drafting for separation 28 of Redundant Power Systems Equipment and Cable Trays.

g. Approve all Electrical Power Systems Drawings including all Cable Trays.

h. Provide program and approve Cable Schedule.

i. Specify lighting, grounding and lightning protection system requirements. Approve lighting fixture selection.

j. Provide technical liason with customer and suppliers.

k. Provide operation manuals for all Electric Power Equipment as well as testing and operations requirements for all Electrical Power Distribution Systems.

l. Support installation and test operations in the Applicant's manufacturing facility.

m. Evaluate test results.

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Control Systems Engineering

a. Interpret the techn I & C requirements established by Mechanical Engineering. 28

b. Prepare logic diagrams for use in preparation of schematic diagrams for plant control and protection circuits.

c. Prepare instrument list and data sheets.

d. Prepare process computer input/output list.

e. Approve arrangement and detail layout of Electrical Building Control Module.

f. Prepare instrument block diagrams and specify control and instrumentation equipment.

g. Provide technical liason with customer and suppliers.

h. Provide operation manuals for all I & C equipment provided and testing requirements for I & C aspects of all plant systems.

i. Provide separation requirements for all I & C Equipment and Systems.

j. Support installation and test operations in the Offshore Power Systems manufacturing facility.

k. Evaluate test results.

2. Structural Engineering

The Structural Engineering Division is responsible for the organization and direction of the functions required to develop the overall structural design and construction specifications for the Floating Nuclear Plants. This division is also responsible for the structural analysis of structures, components and piping and support systems. Specific responsibilities include: 28

a. Classify structures with respect to consequence of failure.

b. Design the platform hull.

c. Design, analyze and procure material for all concrete structures of the plant and all steel structures. 28

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d. Establish quality level of structures on drawings or specifications and establish the acceptance criteria.

e. Prepare structural specifications defining design, material, manufacturing, fabrication, quality level, evaluation requirements and acceptance standards.

f. Perform necessary analysis and specify testing to insure fulfillment of design requirements.

g. Establish a weight analysis program to continuously monitor the weight and stability of the Floating Nuclear Plant during all stages of design, manufacturing and outfitting.

h. Interface with customers and suppliers in matters involving plant structures.

i. Provide technical guidance and approval for the development of detail working drawings. 28

j. Specify and coordinate requirements for towing.

k. Provide technical assistance to other departments for lifts of subassemblies, blocking of platform during dry dock manufacturing and ballasting during manufacturing.

l. Prepare and distribute technical reports and manuals.

m. Perform static and dynamic analyses of plant structures in order to provide design loads for individual structural elements and components.

n. Perform static and dynamic analyses of piping, ducting and cable tray systems including their supports to demonstrate adequacy for all specified conditions.

o. Specify pipe rupture locations and perform analyses to evaluate the effects of pipe rupture.

p. Perform static and dynamic analyses of the containment vessel to demonstrate adequacy for all specified conditions. 28

q. Analyze and specify the motion of the plant due to environmental loads.

r. Review supplier's analyses and tests or perform independent calculations to verify that equipment is qualified for the specified environmental motion criteria.

s. Establish and administer noise and vibration control programs.

3. Mechanical Engineering

The Mechanical Engineering Division is responsible for plant systems configuration and specification, mechanical component specification, carrying out design analyses, and participation in supplier selection. The Mechanical Engineering Division includes the following functions:

Heating, Ventilation, Air Conditioning and Support Systems Engineering

This function includes all HVAC systems in the plant and support systems such as fire protection, domestic water, platform trim and sanitary sewer. Specific responsibilities include:

a. Establish systems concepts.

b. Classify all components with respect to consequence of failure.

c. Issue systems specifications, operating, maintenance and emergency procedures. Prepare system test specification and acceptance criteria.

d. Approve location of mechanical equipment and system layouts.

e. Specify certain equipment and packaged systems, prepare procurement packages, and interface with supplier regarding all technical matters.

f. For certain equipment and packaged systems, analyze supplier's bids, prepare technical evaluation of the bids, approve supplier selection and place and follow purchase order through Purchasing.

g. Review and approve equipment specifications and bid evaluations. 28

h. Support installation and test operations in the applicant's manufacturing facility.

i. Evaluate system test results.

j. Coordinate plant design with insurance requirements. 28

Nuclear Systems Engineering

This function includes integration of the NSSS into the plant and the design of auxiliary systems related to the NSSS, such as radioactive waste treatment systems, fuel and mechanical handling systems, containment safeguards systems (including the ice condenser) and auxiliary fluid systems, such as essential service water and component cooling water. Specific responsibilities include:

a. Establish systems concepts.

b. Issue system specifications and system operating, maintenance and emergency procedures. Prepare system test specifications and acceptance criteria.

c. Classify all components with respect to consequence of failure.

d. Specify the NSSS and interface with the supplier regarding all technical matters.

e. Specify certain equipment and packaged systems, prepare procurement packages, and interface with suppliers regarding all technical matters.

f. For certain equipment and packaged systems, analyze suppliers' bids, prepare technical evaluation of the bids, approve supplier selection, and place and follow purchase order through Purchasing.

g. Review and approve equipment specifications and bid evaluations. | 28

h. Support installation and test operations in the Applicant's manufacturing facility.

i. Evaluate System Test results.

Power Conversion Systems Engineering

This function includes integration of the turbine-generator into the plant and the design of all systems associated with the power conversion cycle. Specific responsibilities include:

a. Establish system concepts.

b. Issue system specification and system operating, maintenance and emergency procedures. Prepare system test specifications and acceptance criteria.

c. Classify all components with respect to consequence of failure.

d. Approve location of mechanical equipment and system layout.

e. Specify the turbine-generator and interface with the supplier regarding all technical matters.

f. Specify certain equipment and packaged systems, prepare procurement package and interface with suppliers regarding all technical matters.

g. For certain equipment and packaged systems analyze suppliers' bids, prepare technical evaluation of the bids, approve supplier selection and place and follow purchase orders through Purchasing.

h. Review and approve equipment specifications and bid evaluations. 28

i. Support installation and test operations in the Applicant's manufacturing facility.

j. Evaluate system test results.

Mechanical Component Engineering

This function includes the definition of requirements for mechanical equipment, specifying mechanical equipment, procurement and production coordination of mechanical equipment. Specific responsibilities and authorities include:

a. Prepare mechanical equipment specifications (defining design, material, manufacturing quality level and test, evaluation requirements and acceptance criteria) using applicable codes and standards.

b. Prepare inquiry and procurement packages for components.

c. For those items covered in (b) above, analyze suppliers' bids, prepare technical evaluation of the bids, approve supplier selection and place and follow purchase order through Purchasing.

d. Interface with the suppliers regarding all technical matters for those items specified in (b) above.

e. Prepare mechanical component maintenance instructions.

f. Support supplier design and manufacture of mechanical components.

g. Support component installation and test operations in the Applicant's manufacturing facility.

Materials and Process Engineering 28

This function provides support to the various engineering disciplines, Operations, Purchasing and Product Assurance. Specific responsibilities include:

a. Recommend and approve materials and processes used in the plant and platform hull.

b. Provide consulting service on materials and materials processing.

c. Develop processes and prepare specifications for the process including the acceptance criteria.

d. Establish specifications for materials procurement including acceptance criteria.

e. Approve procedures for materials processing such as heat treatment, welding, brazing, cladding, and for materials testing such as tensile testing, impact testing, and metallography.

f. Evaluate candidate materials and processes.

g. Establish materials design standards.

h. Establish and follow material related qualification test programs and evaluate test results.

i. Participate in review activities on matters concerning material quality.

4. Nuclear Engineering

The Nuclear Engineering Division is responsible for radiological analyses, shielding design, systems and safety analysis and licensing engineering. Nuclear Engineering is also responsible for the organization and direction of environmental programs related to floating nuclear plant manufacture and operation (generically). Specific responsibilities include the following: 28

a. Preparation of material for submittal to regulatory agencies.

b. Liaison with USNRC and USCG.

c. Establish source terms for radiological analyses.

d. Selection of shielding materials and thicknesses to establish acceptable radiation levels throughout the plant.

e. Determination of radiation exposures external to the plant from direct radiation and radioactivity released from the plant. 28

f. Determination of occupational radiation exposures internal to the plant.

g. Containment pressure analysis.

h. Evaluation of accidents external to the plant and accidents within the plant to confirm plant safety.

i. Establishing safety criteria for plant design and confirming that plant design conforms to these criteria.

j. Assure that designs are compatible with environmental regulations and standards. 28

k. Maintain liaison and coordination with applicable federal, state, local agencies and non-governmental environmental groups.

l. Work with utilities and their consultants and obtain assistance as required on environmental problems.

5. Product Assurance

The Product Assurance function is temporarily assigned to the Director, Power Systems Technology (for Quality Assurance) and the Director, Operations (for Quality Control) until the Applicant enters the final plant design and manufacturing phases. At such time the Product Assurance Department, reporting directly to the President, will be re-established (this is shown on Figure 13.1-1). The ultimate organization of the Product Assurance function and associated responsibilities are detailed in Chapter 17; this is the basic Product Assurance structure which was in effect during the preliminary design phase (which is now essentially complete).

The Product Assurance function is presently staffed by three professionals 28
with 81 man-years total experience and three technicians with 23 man-years total experience. Anticipated peak staffing of the Product Assurance Department, during final plant design and manufacture is 69 Quality Assurance personnel and 236 Quality Control personnel. This estimate is based in part on the Applicant's actual level of staffing during the Floating Nuclear Plant preliminary design phase (reported in Appendix C).

13.1.1.2.2 Marine Design

Marine Design is headed by the Director, Marine Design and includes the following divisions with Floating Nuclear Plant responsibility: Design and Drafting and Engineering Administration. The educational background and total years of engineering or science experience are as follows:

No. Baccalaureate Degrees	5
No. Master Degrees	1
No. Doctorate Degrees	0
No. Non-Degreed Technical Personnel	68
Total Technical Personnel	74
Total Man-Years of Technical Experience	1073

28

The forecasted peak level of staffing during the final design/manufacturing phase is 340 technical personnel. This estimate is based in part on the Applicant's actual level of staffing during the FNP preliminary design phase, (reported in Appendix C).

1. <u>Design and Drafting</u>

The Design and Drafting Division is responsible for the planning, organization and direction of the functions required to produce the drawings 28
required for the engineering and construction of the plant. Design and Drafting is also responsible for plant modeling.

The following specific responsibilities are assigned to the Design and Drafting Division:

a. Schedule all drawings to insure orderly flow of design information to Operations.

b. Interface with Operations to establish the boundaries, content and sequence of drawings to be developed to suit the Manufacturing Plan.

c. Insure the physical fit of all components, systems and structures which makeup the plant. This includes plant modeling activities.

d. During design and construction of the plant maintain liaison with Operations.

e. Review of supplier's drawings as they relate to the physical fit of the plant.

f. Preparation of the Plant Arrangement Drawings and responsibility for control and coordination of the plant configuration including the relationships of major buildings with each other and the placement of equipment within buildings.

g. Preparation of drawings and obtaining all signatures required for review, comment and approval of these drawings.

h. Design and specify the architectural and outfitting aspects of the plant. 28

6. Engineering Administration

The Engineering Administration Division is responsible for Configuration Control, Engineering Planning and Scheduling and Material Administration. These functions are discussed below:

Engineering Planning and Scheduling

This function includes planning, scheduling and reporting the progress of plant design.

Material Administration

This function includes the control and scheduling of procurement for non-stock plant material and components.

Configuration Control

Configuration Control includes the following functions:

a. Documentation Planning and Release - Develop and maintain a comprehensive listing of Engineering documentation that defines the configuration of the standard plant. 28

Control the release of documentation to identify the correct application and issue of documents that are used in manufacturing or that are required to be delivered to the customer for each plant.

b. Engineering Change Control - Assure that all changes proposed against released configuration documents are thoroughly defined and evaluated.

Verify that approved changes are incorporated into the appropriate documentation.

13.1.1.2.3 Operations

Operations is headed by the Director of Operations. Included within Operations are the following functional responsibilities: Manufacturing, Assembly, Manufacturing Control, Manufacturing Engineering, Facility Planning and Facility Development. The educational background and total years of experience of the present staff are as follows: 28

No. Baccalaureate Degrees	17
No. Master Degrees	1
No. Doctorate Degrees	0
No. Non-Degreed Technical Personnel	21
Total Technical Personnel	39
Total Man-Years of Technical Experience	593

28

The forecasted level of staffing during the final design/manufacturing phase is 270 technical personnel. This estimate is based in part on the Applicant's actual level of staffing during the FNP preliminary design phase, (reported in Appendix C).

1. Manufacturing

The manufacturing function is responsible for the manufacture, assembly, and testing of components preparatory to their assembly into the plant. To accomplish this, raw material, supplier-furnished components, drawings, specifications, schedules, and controls criteria are distributed to the following manufacturing activities:

Steel

Steel manufacturing receives raw material and/or prefabricated panels required to fabricate platform and hull structures and the plant superstructure. The raw material will be processed by bending, rolling, cutting, and welding to produce structural subassemblies and modules. These modules will be pre-outfitted to the maximum degree feasible and delivered to assembly at the graving dock.

Concrete

Concrete manufacturing receives raw materials, ready-mixed concrete, reinforcing bar, concrete forms, and places concrete.

Pipe

Pipe manufacturing receives pipe and tubing (both raw and prefabricated), prepares piping details, and fabricates assemblies which are installed in modules or delivered to Assembly as applicable.

Mechanical

Mechanical manufacturing receives components from suppliers and fabricates these components into subassemblies of varying complexity. These subassemblies are then delivered to Assembly.

Electrical

Electrical manufacturing receives electrical components and assemblies. These are assembled into completed systems or subsystems. Completed systems are either integrated into modules or delivered to Assembly as applicable.

2. Assembly

The assembly function is coordinated on a project management basis and includes the integration of all parts, components and materials to produce completed plants. Parts, components and materials received from manufacturing and from suppliers are utilized to perform the required erection, installation, systems integration, testing and grooming operations. Responsibility for each plant undergoing assembly will be vested in an assembly management team responsible for that plant from the start of assembly in the graving dock through preparation for towing. 28

3. Manufacturing Control

The manufacturing control function includes detailed planning; scheduling; materials receiving, storage, issue and handling; and manpower allocation.

4. Manufacturing Engineering

The manufacturing engineering function includes long range planning, process development and refinement, process methods and performance standards. 28 Included are improvement of man-machine relationships, and review and use of new technological developments. Manufacturing Engineering also interfaces with engineering disciplines to ensure product producibility. 28

5. Manufacturing Services

The manufacturing services function includes welding research/procedure qualification, materials evaluation, coatings technology and mechanical testing, concrete testing, chemical analysis and metrology.

6. Facility

The facility function receives data on individual processes required to manufacture plants and integrates them into a total process expressed as a site layout and detailed facility layouts. Facility requirements are formulated as specifications for buildings, structures, utilities, site improvements, and as technical specifications for process equipment. Planning for proper safe and healthy working conditions for a positive impact on the overall environment, and for an efficient facility maintenance system are included. The Facility function also administers facility construction contracts and provides for the maintenance and security of the Manufacturing Facility.

13.1.1.3 Interfaces with Contractors, Suppliers, and Owners

The Applicant is responsible for design, manufacture, and hot functional testing of totally integrated nuclear power plants mounted on floating platforms. Because the plants float, are standardized, and are manufactured in a shipyard like facility, rather than erected as at a land site, the interfaces and divisions of responsibility normally encountered in site erection

are different and sometimes non-existent. For example, the Architect-Engineer, and various contractors necessary for site erection become functional organizations within Offshore Power Systems. Refer to paragraph 13.1.1.2 for Offshore Power Systems' organizational responsibilities and authorities. Since the complete plant is sold to the owner (Purchasing Utility) interfaces with the owner require description.

13.1.1.3.1 Interfaces with Owners

Offshore Power Systems Projects Management is responsible for contract interpretation and contract customer contacts.

A project manager is assigned to each purchasing utility (owner) and is responsible for contract administration providing liaison between the owner and Engineering, Planning and Scheduling, Operations, and Product Assurance. Even though liaison between Offshore Power Systems and the owner is the responsibility of Project Management, working relationships exist between related organizations of the owner and Offshore Power Systems to expedite the flow of technical information. For example, the owner's Quality Assurance Representative works directly with Offshore Power Systems' Product Assurance Management and Engineers. If problems arise affecting price or schedule, it is brought to the attention of the cognizant Project Manager for resolution.

The owner's Project Organization may be based at its home office or at Offshore Power Systems.

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13.1.1.3.2 Interfaces with Suppliers

Offshore Power Systems' Product Assurance is responsible for the quality assurance program. The program is designed to assure that Offshore Power Systems, Westinghouse Electric Corporation, and other major suppliers comply with the requirement to 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants".

Division of Responsibility

The Applicant is responsible for the design and manufacture of the Floating Nuclear Plant. Major equipment for the plant is being supplied by the following Divisions of Westinghouse Electric Corporation:

Water Reactor Divisions - Nuclear Steam Supply System

Steam Turbine Division - Turbine Generators

13.1.1.4 Qualifications of Key Personnel

The qualifications of the Applicant's key management personnel with overall new responsibility for plant design and manufacture are presented in Table 13.1-1. 28

13.1.2 OPERATING ORGANIZATION

The owner has the sole responsibility for the operation of the plant, therefore, these responsibilities will be discussed in the owner's application.

TABLE 13.1-1

SUMMARY OF EXPERIENCE

A.R. COLLIER - PRESIDENT, OFFSHORE POWER SYSTEMS

Education

B.Ch.E.	Chemical Engineering, Syracuse University
Certificate	Harvard Business School

Summary of Experience

1979-Present — President
OFFSHORE POWER SYSTEMS

Mr. Collier has overall responsibility for the design and manufacture of Floating Nuclear Plants, including Quality Assurance and Quality Control.

1972-1979 — Vice President, Engineering
OFFSHORE POWER SYSTEMS

In this capacity, Mr. Collier was responsible for all engineering activities related to the Floating Nuclear Plant, including the mechanical, structural, nuclear, electrical and control systems, and licensing disciplines.

1971-1972 — Engineering Manager, Special Project Division 28
WESTINGHOUSE ELECTRIC CORPORATION

Mr. Collier's responsibilities included all technical engineering activities of the Engineering Department in the Special Project Division. This organization was the precursor to Offshore Power Systems.

1970-1971 — Manager, Plant Application
WESTINGHOUSE ELECTRIC CORPORATION - NUCLEAR ENERGY SYSTEMS
PWR SYSTEMS DIVISION

Mr. Collier was responsible for technical support of pressurized water reactor marketing activities.

1969-1970 — Manager, Licensing and Reliability
WESTINGHOUSE ELECTRIC CORPORATION - NUCLEAR ENERGY SYSTEMS
PWR SYSTEMS DIVISION

Responsibilities included all Atomic Energy Commission licensing activities and quality assurance functions for Westinghouse nuclear plant projects.

1967-1969	Manager, Commonwealth Edison and American Electric Power Projects WESTINGHOUSE ELECTRIC CORPORATION - NUCLEAR ENERGY SYSTEMS Mr. Collier was responsible for overall project management of the Zion and D.C. Cook Nuclear Steam Supply Systems.
1963-1967	Manager, Fluid Systems WESTINGHOUSE ELECTRIC CORPORATION - ATOMIC POWER DIVISION Responsible for the design and development of reactor plant fluid systems.
1957-1963	Senior Engineer WESTINGHOUSE ELECTRIC CORPORATION - ATOMIC POWER DIVISION Responsibilities included participation in the engineering and supervision of design and development efforts related to reactor system's design.
1956-1957	Engineer FOSTER WHEELER CORPORATION Responsibilities included participation in development efforts on a homogeneous reactor design and associated on-site fuel processing.
1954-1956	Engineer E. I. DUPONT de NEMOURS & COMPANY, INC. Responsibilities included studies of operating limitations on a production reactor due to core heat transfer characteristics. 28

Professional Affiliations

American Nuclear Society

TABLE 13.1-1

SUMMARY OF EXPERIENCE

P. B. HAGA - DIRECTOR, POWER SYSTEMS TECHNOLOGY

Education

B.S.	Electrical Engineering, Virginia Polytechnic Institute
M.L.	Mathematics, University of Pittsburgh
Certificate	Nuclear Engineering, Oak Ridge School of Reactor Technology
Certificate	Executive Management, Pennsylvania State University

Summary of Experience

1979-Present
Director, Power Systems Technology
OFFSHORE POWER SYSTEMS

Mr. Haga is currently responsible for the management of engineering design and analysis of power plant including mechanical, nuclear, structural, electrical and control systems, licensing and quality assurance. 28

1978-1979
Director, Plant Analysis and Licensing
OFFSHORE POWER SYSTEMS

Mr. Haga's primary responsibilities included the management of all Floating Nuclear Plant licensing and systems analyses including structural, safety and fluid systems. In addition, Mr. Haga directed all quality assurance and quality control activities and policies for the Floating Nuclear Plant.

1972-1978
Chief Engineer, Mechanical and Nuclear Engineering
OFFSHORE POWER SYSTEMS

While acting in this capacity, Mr. Haga was responsible for the engineering design and analysis of the Floating Nuclear Plant reactor and turbine plant fluid systems including shielding and radiation analysis, system safety evaluation, licensing, HVAC and mechanical components design.

1971-1972
Manager, Mechanical and Nuclear Engineering
WESTINGHOUSE ELECTRIC CORPORATION - SPECIAL PROJECT DIVISION

Responsibilities included design and analysis of nuclear and turbine plant fluid systems including shielding systems, licensing, plant layout and safety evaluations.

1968-1971 Manager, Plant and Systems Development
WESTINGHOUSE ELECTRIC CORPORATION - NUCLEAR ENERGY SYSTEMS, PWR SYSTEMS DIVISION

Responsible for the engineering of new reactor fluid, shielding, steam and structural systems for pressurized water-reactor plants.

1959-1968 Manager, Reactor and Steam Systems Engineering
WESTINGHOUSE ELECTRIC CORPORATION - ATOMIC POWER DIVISION

While employed with Westinghouse - Atomic Power Division, Mr. Haga acted in various supervisory and management positions with responsibilities for the engineering of reactor systems including fluid, shielding, steam and structural systems. This work also included Westinghouse Turn-Key Projects.

1952-1959 Supervisory Engineer
WESTINGHOUSE ELECTRIC CORPORATION - BELGIAN THERMAL REACTOR, YANKEE REACTOR - ATOMIC POWER DIVISION 28

Responsibilities included participation in the engineering and supervision of design of both the Belgian Thermal Reactor (BR-3) and the Yankee Reactor. Earlier responsibilities included participation in reactor plant feasibility studies and preparation of proposals to the electric utility industry.

Professional Affiliations

American Nuclear Society

TABLE 13.1-1

SUMMARY OF EXPERIENCE

R. A. THOMAS - DIRECTOR, MARINE DESIGN

Education

B.S.	Mechanical Engineering, Virginia Polytechnic Institute
Certificate	Advanced Management, Emory University

Summary of Experience

1979-Present — Director, Marine Design
OFFSHORE POWER SYSTEMS

Mr. Thomas is currently responsible for the production of all Floating Nuclear Plant engineering and working drawings and for planning and scheduling services to the Power Systems Technology Function.

1978-1979 — Director, Design Engineering
OFFSHORE POWER SYSTEMS

Mr. Thomas' primary responsibilities included the management of engineering design tasks for the Floating Nuclear Plant, including electrical, structural and fluid systems. 28

1974-1978 — Chief Engineer, Design
OFFSHORE POWER SYSTEMS

While Chief Engineer, Design, Mr. Thomas was responsible for the scheduling and production of all drawings for the Floating Nuclear Plant. He also acted as the primary interface between Engineering and Operations for the Floating Nuclear Plant.

1972-1974 — Manager, Program Control
OFFSHORE POWER SYSTEMS

Responsible for development of information systems for planning, measuring and controlling progress of work on the Floating Nuclear Plant.

1968-1972 Assignments in Marine Engineering and Design
NEWPORT NEWS SHIPBUILDING

At Newport News Shipbuilding, Mr. Thomas worked in all phases of Marine Design. He started in preparation of working drawings, did stress and fluid flow analyses, and was in design supervision for piping systems on a variety of U.S. Naval and commercial ships including submarines, aircraft carriers, and cargo ships. His experience also includes project management on several conceptual design programs for the U.S. Navy and Maritime Administration. 28

Professional Affiliations

Society of Naval Architects and Marine Engineers
American Society of Naval Engineers

SUMMARY OF EXPERIENCE

D. T. VAN LIERE - DIRECTOR, OPERATIONS

Education

B.S.	Marine Engineering and Naval Architecture University of Michigan

Summary of Experience

1978-Present	Director, Operations OFFSHORE POWER SYSTEMS
	Resonsible for the management of Facility Engineering, Construction and Maintenance, Manufacturing Engineering, Manufacturing and Material Control, and Laboratory Services.
1976-1978	Manager, Manufacturing Engineering OFFSHORE POWER SYSTEMS
	Responsible for the management of Manufacturing Engineering and the Manufacturing and Construction Planning including interfacing with Engineering, Design, Procurement and Project Assurance. 28
1973-1975	Director of Production Control & Manpower Planning NEWPORT NEWS SHIPBUILDING
	Directed the overall production scheduling, budgeting, labor and material progressing for many nuclear and non-nuclear ships.
1972-1973	Trades Administrator NEWPORT NEWS SHIPBUILDING
	Managed all waterfront trades (about 13000) working on nuclear and non-nuclear ships.
1966-1972	Manager of the Machinery Division NEWPORT NEWS SHIPBUILDING
	Managed all of the Pipefitting (shop and ship) Machinery Installation, and Insulation application in the building and repair of nuclear powered and non-nuclear powered ships such as Navy aircraft carriers, cruisers, submarines and commercial cargo ships to the appropriate quality and productive requirements.

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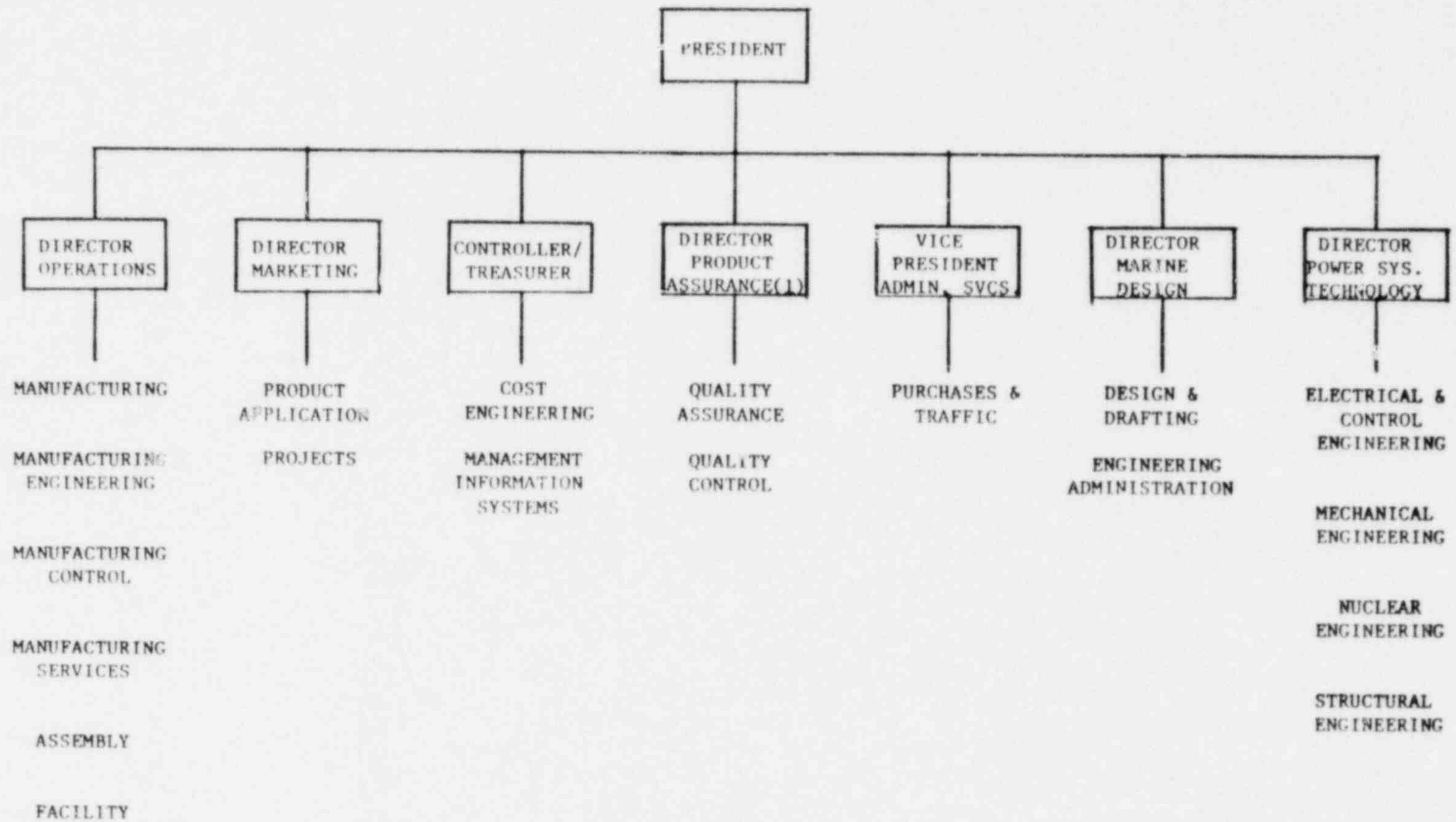
1959-1966	Various Supervisory positions in the Machinery Division NEWPORT NEWS SHIPBUILDING Managed various aspects of nuclear submarine construction and testing, determined facility needs, and developed certain processes.
1957-1959	Project Planner ELECTRIC BOAT DIVISION OF GENERAL DYNAMICS Planned and scheduled the replacement of the reactor plant on a nuclear submarine.
1954-1957	Engineering Duty Officer U. S. NAVY Ship Superintendent and Type Desk Officer at Mare Island Naval Shipyard and the Ship Repair Facility at Subic Bay, R.P.I., responsible for overhaul and repair work on all kinds of conventionally powered Navy ships. 28

Professional Affiliations

The Society of Naval Architects and Marine Engineers
The American Society of Mechanical Engineers



NOTES

(1) WILL BE RE-ESTABLISHED PRIOR TO THE FINAL DESIGN AND MANUFACTURING PHASES.

OFFSHORE POWER SYSTEMS
FLOATING NUCLEAR PLANT
FUNCTIONAL ORGANIZATION CHART

FIGURE 13.1-1

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CHAPTER 17

QUALITY ASSURANCE PROGRAM

17.0 INTRODUCTION

Applicant has established and will execute an effective Quality Assurance Program for Floating Nuclear Plants which meets the requirements of Title 10 CFR 50, Appendix B, "Quality Assurance Program Requirements for Nuclear Power Plants and Fuel Reprocessing Plants" and ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants." In addition the Quality Assurance Program will meet the special quality requirements of Branch Technical Position ETSB 11-1 (Rev. 1) applicable to radioactive waste systems.

Management Commitment to Quality

Offshore Power Systems top management is committed by attitude and practice to the successful implementation of the quality assurance program. The following describes major areas of management participation. 28

Quality Program Support

Offshore Power Systems management by adoption of the quality assurance policy and the approval of implementing procedures and instructions

recognizes and agressively supports the quality assurance program as an important tool in assuring that plant design and construction are as required.

Implementing the Quality Assurance Program

Various techniques are used to convey the importance of the Quality Assurance Program to all levels of management and employes performing activities affecting quality. This is accomplished by the following.

New employes, soon after reporting for work, participate in a new employe indoctrination program conducted by upper management. Among other subjects discussed, the importance of their personal commitment and attitude to performance of work assignments within the requirements of the Offshore Power Systems quality program is stressed. Employes are advised that they are a significant part of the Offshore Power Systems quality team.

Offshore Power Systems employes at all levels of activity are kept current with change to the state of the art appropriate to their work assignments by established company training programs, regulatory agency published documents, industrial codes and standards reviews, other technical publications, and by seminars conducted within OPS, and by other recognized professional groups.

Deficiencies, when identified, will be documented and presented to the the Director, Product Assurance, and the President. Resolution of deficiencies are documented and approved by the next higher level of management than that reviewed.

While major quality assurance problems are made known to management by issued reports, the President's Staff meetings and functional director staff meetings address Quality Assurance Program status and adequacy on a 28
continual basis. Additionally, upper management will meet with quality assurance management as required to resolve major quality problems that have been identified.

The Quality Assurance Program provides the necessary systematic and administrative controls to provide checks to assure that activities which affect quality and safety-related functions during design, procurement, fabrication, handling, shipment, erection, installation, inspection and testing are performed in accordance with established requirements.

Quality levels will be established for all FNP items commensurate with their importance to safety, complexity, intended function, or other engineering considerations. The functional elements of the Quality Assurance Program from design through procurement, manufacturing, installation, inspection and testing will be applied to those items covered by 10CFR50, Appendix A and to those designated Safety Class 1, 2, and 3 per 28
ANSI N18.2 and per Regulatory Guide 1.29.

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

17.1.1 ORGANIZATION

Applicant has established a quality organization to assure that activities which affect the quality and reliability of the Floating Nuclear Plants are systematically performed and verified in a manner commensurate with their importance to plant safety and reliability. The Quality Organization functions are listed in Figure 17.1-1.

The President of Offshore Power Systems has the overall responsibility for the design construction and quality of Floating Nuclear Plants. The President has established the Product Assurance Function and assigned to the Director of Product Assurance the responsibility for establishing, implementing and maintaining the Applicant's quality assurance program as outlined below.

The director of Product Assurance is free from all other duties not related 28
to Product Quality and his full attention is dedicated to assuring that an effective Quality Assurance program is being implemented. As a member of the President's Staff, he has effective direct communication channels with the other senior managers.

17.1.1.1 PRODUCT ASSURANCE

The Director of Product Assurance as the designated individual responsible for the overall quality program, has the authority to stop unsatisfactory

work or control further processing, delivery or installation of nonconforming material. The Director of Product Assurance has direct access to all levels of management without undue influence and responsibility for schedules and costs.

The Product Assurance function will be closely involved in work scheduling and designated quality personnel will attend status meetings to assure that QA/QC procedures and activities are adequate and are scheduled in a timely manner. The Product Assurance Director determines the staffing and qualification level required to assure there is adequate coverage relative to procedural and process controls, inspection controls, and auditing.

On a periodic basis, the Product Assurance Director will evaluate the staffing of the QA & AC organizations and based on projected workloads and schedules will assure that an adequate number of qualified quality personnel are assigned. The Director of Product Assurance, as a member of the President's Staff, may modify the staffing level as required. 28

Figure 17.1-2 further explains the quality organization relationship between Quality Assurance and Quality Control organizations relative to procurement, design, manufacturing and utility organizations.

The responsibility for the overall quality program is retained and exercised by the Applicant through the Product Assurance function. Offshore Power Systems does not delegate any major portion of its quality program to other organizations.

The overall responsibility for approval of quality manuals lies with the Director of Product Assurance.

28

The qualifications for the Director of Product Assurance meet the requirements of ANSI/ANS 3.1 - 1978 (Section 4.4.5) and as a minimum are as follows:

- o A degree in an engineering or related discipline.
- o A minimum of ten years experience in manufacturing, engineering or quality assurance, preferably in the nuclear field.
- o Management skills and supervisory experience in the above areas for a minimum of five years.

The Product Assurance organization is comprised of the Quality Assurance and Reliability Division and the Quality Control Division. See Figure 17.1-1.

The Product Assurance Director has delegated responsibility for stopwork authority to the Quality Assurance and the Quality Control Manager in writing. These individuals are free from cost/schedule pressures and may stop unsatisfactory work, control further processing, control delivery, or stop installation of nonconforming material. 28

The following paragraphs show the assignment of responsibilities and authorities that control the quality assurance program.

17.1.1.1.1 QUALITY ASSURANCE & RELIABILITY

The Manager of Quality Assurance & Reliability is responsible for the organization, planning and implementation of the Applicant's quality assurance program. In carrying out these responsibilities the Manager of Quality Assurance & Reliability has the organizational freedom to:

1. Identify quality problems.

2. Initiate action that results in solutions.

3. Verify implementation of solutions.

4. Stop unsatisfactory work or control further processing, delivery or installation of nonconforming material.

The authorities and responsibilities of Quality Assurance and Reliability are listed below:

1. Policies and Procedures - Establish the policies and procedures for the phases of quality assurance by which the various groups operate to insure that Floating Nuclear Plants meet contractural and regulatory requirements.

2. Drawing and Specification Review - Review and approve drawings and specifications used for procurement and manufacturing to insure that necessary conformance limits, test methods, identification, quality and

regulatory requirements are included. Identify quality problems encountered on similar previous designs and recommend changes or additions when necessary. Assist or direct development of special test specifications or quality standards when necessary to define the article and implement its use.

3. Procurement Quality Requirements - Prepare procurement quality requirements for purchased items defining quality requirements for source inspection, supplier documentation process qualification, and personnel certification.

4. Surveillance Planning - Plan surveillance operations to be conducted by supplier surveillance. Issue Quality Assurance plans consistent with the criticality or complexity of the item being procured.

5. Supplier Submittal Review - Review supplier test, inspection and manufacturing plans prior to use, to insure that characteristics which influence quality are adequately controlled and inspected. Indicate which of these controls or inspections are to be hold points in the purchase order.

6. Material Review Board - Act as chairman of the Material Review Board providing technical disposition of nonconfroming material. Insure that dispositions of the review board are properly documented, clear to personnel required to implement them, and that sufficient technical justification exists to support the dispositions.

7. Design Review - Actively paeticipate in, monitor and document Design Review Meetings that are scheduled by the Applicant's Design Review Board.

8. Plant Design Report - Prepare the Product Assurance input to the Plant Design Report, defining the program for controlling the quality of nuclear safety related systems, components and structures.

9. Supplier Surveys - Implement the supplier quality evaluation program including surveys of potential suppliers to determine capabilities of their quality control system. Assist purchasing in maintaining the approved suppliers list.

10. Supplier Indoctrination - Advise and instruct supplier in the use of quality forms and procedures in compliance with purchase order requirements. Assist supplier in the resolution of quality problems and, when required, advise cognizant Applicant personnel of possible problem areas.

11. Corrective Action - Identify and report conditions at suppliers' and Applicant's facilities which are adverse to quality; initiate investigation of the causes by responsible agencies for development of corrective measures to prevent recurrence; evaluate and follow resultant corrective actions to assure effectiveness.

12. Supplier Rating System - Maintain the quality assurance supplier rating system which evaluates the performance of Applicants' suppliers.

13. Failure-Mode Analysis - Review reliability analysis of the various systems or equipment which determine failure modes and advise other engineering divisions as to weak points, when required.

14. Data Analysis - Analyze field reports, nonconforming material documents, inspection records, and supplier surveillance reports for potential or recurring problems which can indicate development of trends. Initiate appropriate corrective action to prevent recurrence of problems.

15. Quality Assurance Audit Programs - Establish a quality audit program for Applicant and administer the system within the facility and at supplier facilities. Includes performing of audits, resolving deficiencies with affected divisions, and reporting results to executive management.

16. Quality Data System - Coordinate the administration of the Applicant's Quality Data System for the storage and retrieval of completed quality assurance records.

17. Training Program - Establish policies and procedures for assuring that training programs are established and maintained to qualify and certify personnel performing activities affecting quality such as: NDE examiners, inspectors, testers, auditors, welders, and other special process personnel.

18. Verification of Special Process Controls - Verify that special processes are adequately controlled, i.e., qualification of personnel, equipment and procedures as defined in Section 17.1.9.

19. <u>Quality Standards</u> - Establish standards to assure uniformity of quality in documentation, workmanship, and other repetitive activities affecting quality.

17.1.1.1.2 <u>QUALITY CONTROL</u>

Quality Control is responsible for the organization, planning and implemen-
tion of the quality control program. In carrying out these responsibili- 28
ties, the Manager of Quality Control has the authority and organizational
freedom to:

1. Identify quality problems.

2. Stop work pending resolution of material, fabrication and/or processing problems or deficiencies.

3. Initiate, recommend or provide solutions to quality problems through designated channels.

4. Verify implementation of solutions to identified quality problems.

The authorities and responsibilities of Quality Control are:

1. Develop, issue and maintain quality control and Nondestructive Examination (NDE) procedures and quality control and NDE instructions for the performance of Quality Control functions.

2. Establish quality control manning levels and personnel training and qualification programs.

3. Maintain and operate the quality control records system.

4. Determine product acceptance through the performance of receiving inspection on materials, components, and equipment and in-process and final inspection during manufacturing and assembly operations.

5. Provide source surveillance and inspection for parts and materials, including surveillance of manufacturing operations, process qualification, mechanical, electrical, and nondestructive tests, material storage, and any other surveillance points.

6. Implement and control a system for the identification of inspection status of product and materials.

7. Identify nonconformance materials and, when practical, assure their segregation.

8. Participate in the preliminary dispositioning of nonconforming material.

9. Provide surveillance of the Metrology Program to assure adequate control over the calibration and repair of inspection tools and instruments.

10. Develop and specify the necessary inspection, measuring, and test equipment for performance of Quality Control functions.

11. Witness functional tests and review test data for compliance with requirements.

12. Review, approve and provide quality input prior to release for manufacture for thos Process Sheets having a significant effect on operation or safety.

13. Monitor the Applicant's Materials Testing Laboratory to ensure compliance to approved procedures and instructions.

14. Maintain a program for calibration and control of nondestructive testing equipment.

15. Perform nondestructive examinations in accordance with established procedures and instructions with qualified personnel to insure that materials and workmanship comply with specification and procedure requirements.

The details of the Quality Assurance and Reliability and the Quality Control organizations are described in the procedure on organization in the Quality Assurance Manual.

17.1.1.2 PRODUCT ASSURANCE INTERFACES

17.1.1.2.1 INTERFACES WITHIN THE APPLICANT'S ORGANIZATION

Product Assurance interfaces with engineering and operations functions from the design definition stage through manufacturing and testing, and identifies the extent the quality program is to be applied to specific structures, systems, and components. A graded "quality level" approach is used to determine the degree of controls required based on importance to safety and plant reliability. These controls are applied to design, procurement, document control, inspection, tests, special processes, records, audits and other applicable areas as required. 28

17.1.1.2.2 INTERFACE WITH THE UTILITY OWNER

The overall responsibility for maintaining project interface with the Owner is vested in Project Management. In addition to this formal project interface, quality assurance representatives of the Owner may interface directly with the Manager of Quality Assurance and Reliability and the Manager of Quality Control or their designated representatives to provide a flow of information and communication.

When the quality assurance requirements of the contract are in question or need resolution, the formal interface is through the Applicant's Project Manager and the Owner's designated contract representative.

Figure 17.1-2 further explains the quality organizational relationship of the Quality Assurance and Reliability and the Quality Control organizations to procurement, design, manufacturing and utility organizations.

17.1.1.2.3 INTERFACE WITH SUPPLIER AND SUBCONTRACTORS

The responsibility for maintaining formal interface with suppliers and subcontractors is assigned to the Purchases & Traffic Division of Administrative Services. Direct technical discussions and liaison between representatives of suppliers and subcontractors and with the cognizant Quality Assurance and Reliability representatives are permitted but all written or formal communcation is through Purchases and Traffic.

The management of Quality Control or the Applicant's source inspectors interface with suppliers on matters relating to product acceptance.

17.1.1.2.4 INTERFACES WITH REGULATORY AGENCIES

The Manager of Quality Assurance and Reliability or his designated representative coordinates quality assurance activities with the Nuclear Regulatory Commission and the United States Coast Guard. The Manager of Quality Control or his designated representatives interface direct with Coast Guard inspection personnel on matters concerning product acceptance. The interface relative to reporting significant deficiencies to the Nuclear Regulatory Commission is through the Director of Product Assurance.

17.1.1.2.5 INTERFACE WITH ASME AND AUTHORIZED INSPECTION AGENCY

The Manager of Quality Assurance and Reliability is responsible for obtaining the necessary "Certificates of Authorization" issued by the American Society of Mechanical Engineers to perform work in accordance with the ASME Boiler and Pressure Vessel Code.

The Manager of Quality Assurance and Reliability is responsible for maintaining the contract with the Authorized Inspection Agency. Direct technical discussions between the Authorized Nuclear Inspector and representatives of Quality Assurance and Reliability, for matters of Code interpretation and clarification, are permitted. 28

17.1.2 QUALITY ASSURANCE PROGRAM

17.1.2.1 Policy

It is the policy of Offshore Power Systems to establish and maintain a quality program that will result in Offshore Power Systems providing products and services which are safe, and which meet customer needs and specification requirements. The Offshore Power Systems quality program conforms in all respects to such government and industry quality standards as agreed by contract. 28

The quality program encompasses all activities of the Offshore Power Systems divisions, from initial order and product development to installation and service of delivered product. Each function has specific

responsibilities for its segment of the overall Offshore Power Systems quality program. Continuing efforts to improve quality performance and minimize failures are a part of the quality program.

17.1.2.2 Quality Documents

Applicant's quality assurance program relates to each of the eighteen criteria of 10 CFR 50, Appendix B and is expanded or implemented by the following basic documents: 28

1. Quality Assurance Manual - This manual contains policies and procedures which define the applicant's quality assurance program. It also contains the Quality and Reliability Procedures (QRP's) which give the overall quality instructions to Applicants' employees, NRC, and Owners as to how the Applicant complies with the eighteen criteria of 10 CFR 50, Appendix B. Table 17.1-1 contains a list of the QRP's currently being used to implement the quality assurance program.

2. QRS-100, Quality Requirements for Suppliers - This standard establishes the manner in which the supplier is to work with the Applicant in meeting the quality requirements of 10CFR50, Appendix B, Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants or ASME Code, Section III, Division 1 NA-4000, Quality Assurance, as appropriate. The degree of applicability of the above referenced quality requirements is detailed in the Procurement Quality Requirements attached to or contained in the purchase order. 28

3. Quality Control Procedures & Instructions - Documents which prescribe the framework and details for controlling product integrity from procurement through processing to final test, acceptance, and record transferral. Through these procedures and instructions, the Quality Control Division implements those elements of the Quality Assurance Program which pertain to: source and in-house inspections; all nondestructive qualifications and examinations; surveillance of internal processing, storage, equipment calibration, and material/equipment testing programs; control of production inspection planning; identification of inspection status; control & disposition of nonconforming material; and the collection, maintenance, and protection of quality records.

4. Metrology Procedures: Procedures will be established which describe the methods used for maintenance, calibration, and control of measuring, and test equipment, used for the measurement, inspection and monitoring of structures, systems and components. These procedures are developed by the metrology section which is part of the operations function. Prior to issue, they are reviewed for quality requirements by Product Assurance and concurrence is documented. 28

17.1.2.3 Indoctrination, Training and Qualification.

Product Assurance reviews and documents concurrence for quality related indoctrination, training and qualification activities which are described in departmental procedures.

(a) Personnel performing quality related activities are instructed as to the purpose, scope and implementation of the quality related manuals, instructions and procedures.

(b) Personnel verifying activities affecting quality are trained and qualified in the principals, techniques, and requirements of the activity being verified.

(c) Formal training and qualification programs are documented. This documentation includes objective, program content, attendees, and the date of attendance.

(d) Personnel performing and verifying activities affecting quality are given proficiency tests based on developed acceptance criteria to determine if individuals are properly trained. 28

(e) Qualifications certificates or records clearly indicate the specific functions personnel are qualified to perform and the criteria used to qualify them.

(f) Continued personnel proficiency is accomplished by periodic retraining, reexamining, and or recertifying as determined by management.

(g) The overall training program provisions satisfy Regulatory Guide 1.58, Rev. 1.

The Applicant's management schedules audits of the performance of the Product Assurance Function on a calendar year not to exceed eighteen month elapsed time schedule, by qualified personnel, to assure conformance to the requirements of the quality assurance program. The other functions, divisions and departments of the Applicant's organization are audited by the Quality Assurance & Reliability Division. These audits assess the scope, status, implementation and effectiveness of the quality assurance program to assure that the program is adequate and complies with 10 CFR Part 50 Appendix B criteria.

The responsibility for defining the quality assurance program is vested in the Quality Assurance and Reliability Division of Product Assurance. The applicable quality procedures are reviewed and approved by the Director of Product Assurance. Changes to the program are handled in the same manner as the original issue.

The introduction to the Quality Assurance Manual contains a policy statement from the President of the Applicant's company. This statement communicates to organizations and individuals that the quality assurance program and documents referenced therein are mandatory quality requirements which are to be adhered to by affected personnel.

The safety-related structures, systems, and components controlled by the quality assurance program are identified in Section 3.2.2 of this Plant Design Report.

Quality levels for the nuclear safety related equipment are shown in Figure 17.1-3.

The principal Contractor for the Nuclear Steam Supply System is Westinghouse Electric Corporation. The Contractor's quality assurance program is contained in "Westinghouse Nuclear Energy Systems Divisions Quality Assurance Plant", WCAP-8370. The Applicant will utilize the latest WCAP-8370 as approved by the NRC to evaluate the performance of the Contractor during the life of the contract. Incoming employees are given an indoctrination into the requirements of the quality assurance program. Manuals of other related functions, such as Purchasing, Engineering, and Manufacturing are included in the indoctrination as required. Training is conducted as necessary for personnel performing various qualityrelated activities in those activities. This training may be conducted both within the Applicant's facilities and at regularly conducted schools.

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TABLE 17.1-1

IMPLEMENTATION OF 10 CFR 50, APPENDIX B

PROCEDURE	CRITERION
Organization	I
Product Assurance Organization	I
Product Assurance Interfaces	I
Quality Assurance Program	II
Preparation of Quality and Reliability Procedures	II
Quality Levels	II
Indoctrination & Training Program for Product Assurance Personnel	II
Qualification and Certification of Auditors	II
Quality Assurance Memorandums	II
Design Control	III
Drawing & Specification Review	III
Design Review	III
Procurement Document Control	IV
Procurement Quality Requirements	IV
Instructions, Procedures & Drawings	V
Quality Assurance Plans for Procured Items	V
Document Control	VI
Drawing & Specification Release & Control	VI
Procedure & Document Release & Control	VI
Control of Purchased Material, Equipment and Services	VII
Supplier Quality Survey & Evaluation	VII
Approval Requests	VII
Identification & Control of Materials, Parts & Components	VIII
Materials Identification & Marking	VIII
Control of Special Processes	IX
NDE Personnel Certification	IX
Inspection Program	X
Test Control	XI
Test Procedure Review	XI

The procedures used by the Applicant's engineering department for the preparation of the Plant Design Report include an organized development of the design information, distribution of the report material to those having interest and/or interface responsibility, and conclude with formal review sessions before being included in the report. 28

For the Floating Nuclear Plant the Applicant is the manufacturer; because of this, the transfer of records is simplified. The Nuclear Steam Supply System supplier will furnish the records for each of the components to the Applicant at the Applicant's manufacturing facility. The same policy will be followed by each of the other suppliers. These components will be integrated into the plant by the Applicant. Records pertaining to the quality of various systems and components will be transferred to the Owner by the Applicant on a regular systematic basis as part of the contract between the Owner and the Applicant.

The final transfer of records will be at a mutually agreed time between Applicant and Owner but prior to the time the Floating Nuclear Plant is made ready for operation.

Testing conducted by OPS is described in Chapter 14. This testing will be conducted in accordance with the OPS Quality Assurance Program. Pre-operational tests conducted by the Owner will be conducted in accordance with the Owner's Quality Assurance Program.

The Applicant and his suppliers, where applicable, will structure their quality assurance programs in accordance with the guidelines provided in Applicable Regulatory Guides and ANSI Standards. The position of the Applicant relative to the applicable Regulatory Guides (and ANSI Standards invoked by Regulatory Guides) are defined in Appendix 3D of this report. In addition, the Applicant will comply with the requirements of Regulatory Guide 1.116, "QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems," and Regulatory Guide 1.123, "QA Requirements for Control of Procurement of Items and Services for Nuclear Power Plants."

The Applicant's suppliers are advised of applicable Regulatory Guides and ANSI Standards by mean of appropriate procurement documents such as: drawings, equipment specifications, material specifications and the purchase order.

Disputes between Product Assurance personnel and personnel of other organizations which cannot be resolved will be referred to higher management of the organizations involved for resolution.

17.1.3 DESIGN CONTROL

17.1.3.1 Program Requirements

The Engineering Divisions of Offshore Power Systems have the responsibility and the authority for defining and implementing the Offshore Power Systems design control program. The Engineering Policies & Procedures (EPP) Manual contains procedures which define the technical and management responsibilities for the design activities of the Mechanical & Nuclear Engineering, Electrical & Control, Naval Architecture, Structural, Technical Services, and Design Divisions of Engineering.

The Engineering Divisions plan, control, execute, verify, and document the program for the control of internal and external design activities of Offshore Power Systems

17.1.3.2 Design Input Requirements

Design inputs, such as performance specifications, quality, reliability and safety requirements, design bases, customer requirements, regulatory requirements, and codes and standards are identified prior to implementing specific design processes. The design inputs are specified, timely reviewed, approved, and documented in sufficient detail to permit the

design and its verification to proceed in accordance with the procedures contained in the EPP Manual.

The EPP Manual contains procedures which require the use of applicable industry and government standards or specifications, material or prototype hardware testing programs, or design reviews to assure the suitability of the material, hardware or processes prior to their selection.

For commercial "off-the-shelf" items where specific nuclear application
controls cannot be imposed in a practicable manner, Product Assurance will
develop verification requirements which will provide necessary assurance
that the item is acceptable for the prescribed use and before its incorpor- 28
ation into the product. Selection of new or unique materials, parts,
equipment and processes for safety-related structures, systems, or compo-
nents is also verified and concurred in by Product Assurance.

17.1.3.3 <u>Design Processes</u>

Design activities are planned, prescribed, controlled, and executed by the Engineering Divisions in compliance with written procedures. The EPP Manual contains procedures for implementing design activities such as translating design requirements into technical descriptions, drawings, or specifications, methods for performing and documenting design or analysis calculations, seismic or other induced stress evaluations, accident and pipe rupture analyses, material compatibility and environmental control, inservice inspection, and safety classification.

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Engineering drawings, specifications, procedures, and other design documents are reviewed for the inclusion and proper application of technical, regulatory, and quality requirements in accordance with the applicable EPP's and Quality & Reliability Procedures (QRP's).

17.1.3.4 Interface Control

Design interfaces and channels for communication of technical information, both internal and external to Offshore Power Systems, are prescribed, controlled, and documented by adherence to the procedures contained in the corporation and function manuals.

17.1.3.5 Design Verification

Engineering drawings, specifications, procedures, and other design documents that are generated by, or submitted for approval to Offshore Power Systems by suppliers, are systematically reviewed and approved by the involved Engineering Divisions, Quality Assurance & Reliability, and other interfacing activities. All such design documents are reviewed prior to their issue as a controlled document to provide documented assurance that:

1. The design is considered to be technically adequate and that a check of dimensional accuracy and completeness of the design drawing and specification has been made.

2. The design processes are performed by authorized design activities.

3. The design conforms to established engineering, regulatory, quality and safety criteria.

4. The design poses no design interface problems.

5. The design and all its changes are documented and traceable to design input requirements.

6. Deviations from established quality standards are minimized.

7. The intended service requirements are met.

In addition to the requirements listed above the Quality Assurance and Reliability Division performs a documented review that assures that: 28

1. Design characteristics are stated in a manner which can be controlled, inspected, and tested for verification of product compliance.

2. Inspection and test criteria are adequately identified on drawings or related documents.

This review is conducted by knowledgeable individuals qualified in Quality Assurance/Quality Control techniques. These individuals will assure that the documents have been prepared, reviewed and approved in accordance with 28 written procedures. In addition, they will assure that necessary quality

requirements relative to inspection, test, acceptance, and the extent of documented results are specified in the document.

The design adequacy of concepts, parameters, components, structures, and systems that are considered by the Offshore Power Systems Design Review Board to be vital to the safety, performance, or reliability of the FNP is independently verified at one or more appropriate stages of the design in accordance with the procedures for design verification contained in the EPP Manual. The design is checked and confirmed to be adequate by one or more of the following verification methods: alternate or simplified calculations, design reviews, or by the performance of a suitable qualification testing program, by personnel other than the original "designer" or his immediate supervisor. The reviewers and their position, authority, and responsibilities are identified and controlled by the procedures for design verification contained in the EPP Manual.

The Applicant's Engineering procedures specify that when a design is to be verified by a suitable qualification test program, a prototype unit will be tested under adverse design conditions.

17.1.3.6 Document Control

The EPP and Quality Assurance (QA) Manuals contain procedures for the preparation, coordination, approval, issue, distribution, control, change, and maintenance of design documents such as drawings, specifications, procedures, instructions, design reviews, and other design records.

17.1.3.7 Design Change Control

Changes to design documents, including those resulting from manufacturing changes, are subjected to the same design controls and approvals that were applicable to the original design. The EPP Manual and the Offshore Power Systems Policies and Procedures Manual (PPM) contain procedures for design and configuration control to assure that the reason for the change is justified and documented, the impact of the change is considered, the required action is documented and implemented, and the affected organizations are informed in a timely manner.

17.1.3.8 Corrective Action

Errors or deficiencies that may adversely affect the performance or the safety aspects of the design parameters, items, components, structures, or systems of the FNP are documented, and cognizant management informed, in compliance with the applicable EPP and QRP procedures. Corrective ation is implemented in a timely manner.

17.1.3.9 Records

Design documents and records which provide legible, identifiable, traceable, and retrievable evidence that the design, and the verification thereof, was performed in accordance with the procedures contained in the EPP and QA Manuals, are collected, stored, and maintained in accordance with the requirements of ANSI N45.2.9 and the EPP Manual.

The retained documentation includes the final design documents and sufficiently detailed records of the important steps and changes made during the design process, including the source of design requirements and inputs, which support the final design.

17.1.3.10 Audits

The QA Manual contains procedures for implementing a comprehensive system of planned and documented audits for verification of compliance with all aspects of the Offshore Power Systems design control program.

17.1.4 Procurement Document Control

As described in Section 17.1.3, design specification and drawings that are the basis of a procurement action are reviewed and approved by the affected organizations.

The procurement purchasing requisition is reviewed and approved by Quality Assurance and Reliability to assure that applicable quality requirements are clearly defined. Purchase orders are also reviewed by Quality Assurance to assure that the applicable procurement quality requires (PQR) are properly incorporated. This review will include a determination that quality requirements are correctly stated, the items are inspectable, controlled manufacture is possible, adequate acceptance/rejection criteria are stated and the overall document meets the quality program requirements. 28

Documented evidence of the review and approval of procurement documents are maintained by the originator of the document in files that identify the responses of the affected orgnizations and are available for verification.

Quality assurance document QRS-100, "Quality Requirements for Suppliers" establishes the manner in which the supplier is to work with the Applicant in meeting the quality requirements of 10CFR50, Appendix B, Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants or ASME Code, Section III, Division 1 NA-4000, Quality Assurance, as appropriate. The degree of applicability of the above referenced quality requirements is detailed in the Procurement Quality Requirements attached to or contained in the Purchase Order.

The PQR requires subcontractors/suppliers to provide a quality assurance program that meets the quality level of the procured item. Product Assurance reviews Quality Assurance programs and QA Plans submitted by suppliers and determines their acceptance for the type of item to be procured. 28

Procurement documents are reviewed by designated organizations in accordance with established documented procedures that are implemented by the division in their procedure manuals to assure that procurement documents contain, or reference the following:

1. The applicable design basis technical requirements, including regulatory requirements, component and material identification, drawings, specifications, codes and industrial standards, including their revision status, tests and inspection requirements and special process instructions for such activities as fabrication, cleaning, erecting, packaging, handling, shipping and storage.

2. The requirements which identify the documentation to be prepared, maintained, submitted, and made available to the buying agent for review and/or approval, such as drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, materials, chemicals and physical test results.

3. The requirements for retention, control, and maintenance of records.

4. The procuring agency's right of access to suppliers' facilities and records for source inspection and audit.

Changes and/or revisions to procurement documents are subject to the same review and approval requirements as the original document.

Procurement documents for spare or renewal parts are subject to the same review and approval requirements in accordance with established documented procedures to determine similarity, compatibility, and inclusion of the quality assurance requirements and acceptance criteria of the original part.

17.1.5 Instructions, Procedures and Drawings

Quality related procedures required to accomplish the objectives of the Quality Program are reviewed by Product Assurance to assure that the procedure is consistent with Quality Program commitments, and Corporate policies. In addition, Quality-related procedures are controlled documents, made mandatory for the specified activity by policy statements and are signed and approved for use by the responsible manager.

Product Assurance documents its review and concurrence with Quality-related procedures. 28

Specific procedures have been established to control nonconforming material. These procedures describe methods of identifying, segregating, reviewing, and dispositioning nonconforming materials, components, equipment and certain services performed which would affect plant safety. These procedures identify authorized organizations to participate in a Materials Review Board and provide methods for notifying affected organizations. Product Assurance reviewes these procedures and documents concurrence.

Activities affecting quality, such as quantitative and qualitative criteria, are defined by documented specifications, drawings, plans, procedures, and instructions. Methods for their issue, revision and control are delineated in manuals appropriate to their application. Implementation and compliance to these requirements and to the quality assurance program are assured by documented Quality Assurance and Quality Control procedures, plans and instructions.

Table 17.1-2, 10CFR50, Appendix B Implementation Manuals and Procedures Matrix, identifies those Offshore Power Systems manuals and documents which contain policies, procedures and instructions applicable to specific criteria of 10CFR50, Appendix B.

17.1.6 Document Control

The Applicant has initiated an Engineering Configuration Control function to control the release and accountability of engineering documents.

Manufacturing, inspection and test documents will also be controlled in accordance with written procedures to assure that each discipline has the latest issue at the location where the activity is performed.

Documents that are used in design, manufacturing, test and inspection of safety related materials, structures and components are formally reviewed and approved in accordance with established documented procedures as described in Section 17.1.3. These document control procedures are reviewed by Product Assurance and concurrence is documented. Changes to documents are reviewed and approved to the same procedure requirements as the original document. 28

The minutes of the Configuration Control Board are distributed to the departments responsible for the prompt inclusion of changes into Work Orders, Purchase Orders, Instructions, Procedures, drawings and other appropriate documents associated with the change.

TABLE 17.1-2

10 CFR 50 APPENDIX B IMPLEMENTATION MANUALS AND PROCEDURES MATRIX

OPS DOCUMENT	CRITERIA OF 10 CFR 50, APPENDIX B																	
	I	II	III	IV	V	VI	VII	VIII	IX	X	XI	XII	XIII	XIV	XV	XVI	XVII	XVIII
QUALITY ASSURANCE MANUAL	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
QUALITY REQUIREMENTS FOR SUPPLIERS (QRS-100)	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
QUALITY CONTROL PROCEDURES MANUAL	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
NONDESTRUCTIVE EXAMINATION MANUAL	X	X			X	X	X	X	X	X	X	X	X	X	X		X	
OPERATIONS POLICY & PROCEDURES MANUAL	X	X			X	X		X	X		X	X	X	X			X	
POLICIES AND PROCEDURES MANUAL	X		X	X	X	X											X	
ENGINEERING POLICIES AND PROCEDURES MANUAL	X		X	X	X	X	X	X	X		X		X				X	
PURCHASING MANUAL				X	X	X	X						X				X	

Procedures will be developed to establish a method for preparation of "as-built drawings" and related documentation. These as-built documents will reflect actual plant design and will be prepared in a timely manner. 28

The Floating Nuclear Plant has a controlled computer list of the current approval status of documents to preclude the inadvertent use of obsolete or superseded documents. The following documents are typical of those controlled by this list:

Design technical specifications.

Engineering drawings.

Manufacturing drawings.

Process specifications.

Test specifications.

Each of the following manuals incorporate a procedure which describes control of the manual.

Quality Assurance Manual

Quality Control Procedures Manual

Engineering Policies and Procedures Manual

Purchasing Manual

17.1.7 Control of Purchased Material, Equipment and Services

Suppliers are required to furnish materials, systems and components which are manufactured in compliance with 10CFR50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, and ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants. Supplier surveys are documented on standard supplier survey forms which are reviewed and approved by Quality Assurance and Reliability management. Suppliers that have been conditionally approved are controlled; deficient items have to be scheduled for correction and corrective action approved prior to award of contract. Completed supplier survey reports are filed in Quality Assurance and Reliabilitv as quality assurance records.

17.1.7.1 Supplier Surveys and Evaluation

Prior to the award of an order an evaluation of the supplier's capability to perform the required work are conducted by the Applicant. This evaluation may consist of either a review of known facts about the supplier's capabilities or a survey of the supplier's manufacturing facilities conducted by a team consisting of representatives from Product Assurance, Purchasing, and Engineering. Surveys assess the ability of the supplier to perform in accordance with each of the eighteen criteria of 10 CFR 50 Appendix B or ANSI N45.2 which are applicable to the suppliers product for its prescribed use in the Floating Nuclear Plant. 28

Suppliers who are judged acceptable by all three of the evaluating organizations are added to Purchasing's list of approved suppliers. [illegible] material,

equipment or services may be procured from a supplier who is not on the list.

During fabrication, inspection, testing, and shipment of equipment, materials, and components by suppliers, Product Assurance will verify conformance to purchase order requirements.

Supplier Inspection procedures, instructions, and checklists are reviewed by Product Assurance and concurrence documented to assure that quality requirements as appropriate are met when applicable. These procedures, instructions, or checklists shall include as appropriate:

(a) Characteristics/activities to be inspected.

(b) Inspection method description.

(c) Identification of the individual or groups responsible for performing the inspection and qualification level.

(d) Acceptance/rejection criteria.

(e) Identification of required procedures, drawings, specifications and revisions used.

(f) Name of inspector performing the inspection and the results of the inspection. 28

(g) Required measuring and test equipment including accuracy. 28

17.1.7.2 Procurement Quality Requirements

After the award of an order, the supplier is subject to surveillance by the Applicant as indicated on the Procurement Quality Requirements (PQR) which is a part of the purchase order. The PQR specifies in writing the processes or characteristics to be witnessed, inspected or verified and accepted including holdpoints when required. In addition, the extent of documentation, i.e., CMTR, inspection reports, etc., required is specified in the PQR. The PQR requires the supplier to submit procedures, practices, processes, and inspection plans to the Applicant at specified intervals throughout the life of the order. The review of these procedures, practices, and processes will indicate the points at which the Applicant will witness and verify certain processes or characteristics. 28

17.1.7.3 Supplier Surveillance

Surveillance of Suppliers is conducted by the Applicant's quality representatives in accordance with written instructions which describe the method of surveillance and the extent of documentation required, to assure that the supplier complies with the quality requirements specified in the PQR and purchase order. 28

17.1.7.4 Supplier Audits

Evaluations are conducted by the Applicant to assure the effectiveness of the control of quality by suppliers on a calendar year not to exceed 18 month elapsed time interval consistent with the importance complexity and quantity of the item being supplied. These evaluations are conducted in accordance with established written procedures as defined in the Quality Assurance Manual. As a result of the evaluations, identified suppliers will be scheduled for quality program audits. Audits are conducted by the Applicant in accordance with established written procedures to:

1. Provide an objective evaluation of compliance with established quality requirements, methods and procedures.

2. Determine adequacy of Quality Program performance.

3. Verify implementation of recommended corrective action. When an audit reveals an unsatisfactory condition, the supplier is sent a Request for Corrective Action to which he must respond within a specified time. The details of the procedure for obtaining corrective action are explained in Section 17.1.16 of this Plant Design Report.

17.1.7.5 Receiving Inspection

Receiving inspection is conducted by Quality Control in accordance with established written instructions to assure the following: 28

1. The material, equipment, or component is properly identified and corresponds with the receiving documentation and the purchase order.

2. The material, component, or equipment and required acceptance records satisfy the receiving inspection instructions prior to release for installation or use. 28

3. Completion of any required inspections which may have been waived at the supplier's facility.

4. Accumulation of all specified inspection, test and other records (such as certificates of conformance, heat treat records, CMTR, etc.) at the Offshore Power Systems facility prior to releasing the item for installation or use. 28

Incoming material inspected by Quality Control and found acceptable is released to materials services, prior to fabrication, erection, assembly, or controlled storage.

Certificates of conformance submitted with material, parts or components are periodically evaluated by the following means:

1. Audits of the supplier to review the background data for the certificate.

2. Independent inspection of the item upon receipt by the applicant.

3. Tests of material to verify the results reported by the supplier.

17.1.7.6 Nonconforming Material

When material and equipment or required records for same is determined to be nonconforming, the nonconformity will be documented, and when the physical size of the material permits, the material will be segregated in a holding area until disposition of the nonconformity has been completed. Nonconforming material will be identified by a tag until disposition is made. Details pertaining to nonconforming material are specified in the Quality and Reliability Procedures and Sections 17.1.15.

17.1.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

The Applicant has established procedures to asure that the identification and traceability of items is maintained throughout manufacturing, assembly, and delivery of the Floating Nuclear Plant.

QRS-100 Quality Requirements for Suppliers, requires suppliers to establish means to maintain identity and traceability of parts and components throughout manufacturing and delivery.

17.1.8.1 Part Identification Numbers

Identification of material, parts, and components is verified and documented by Quality Control prior to release for fabrication, assembly, shipping or installation at the Applicant's facility. 28

When identification and control of parts and components is required for items produced by a supplier, a control number is assigned by the Applicant in the procurement document(s). The procedure used by the suppliers to assign control numbers to the item is reviewed for adequacy by the Applicant and audited during surveillance activities for compliance. When identification and controls are required, the control number remains with the materials throughout manufacturing and can be traced to associated documents, such as drawings, specifications, purchase orders, manufacturing and inspection documents, and deviation reports. 28 28

17.1.8.2 Marking

Process specifications have been established for marking nuclear safety-related components by methods that will not result in any harmful contaminations, sharp discontinuities or metallurgical changes. Location of the marking is identified on the drawing so as not to affect critical surfaces or damage the part or component.

Procedures will be established for the verification of the correct identification of materials, parts and components prior to release for assembly, installation or shipping by surveillance activities at suppliers and by Quality Control functions at the Applicant's manufacturing facility.

17.1.9 CONTROL OF SPECIAL PROCESSES

Written procedures will be established by the Applicant for accomplishment of special processes. These procedures will specify personnel qualifications, required equipment and specific acceptance criteria and performance parameters. The Applicant's Quality Control organization verifies the recorded evidence and documents the results, at witness points, examination points, and during in-process inspection, and assures that qualified personnel were used. 28

Special processes include but are not limited to: welding, cleaning, heat treating, and nondestructive testing. Other controlled processes effecting safety or reliability, as specified in Engineering Design documents, will be controlled and accomplished in accordance with applicable codes, standards, specifications, criteria, or other special requirements.

17.1.9.1 Special Processes at Suppliers

Suppliers are required to submit to the Applicant, for approval, the procedures for qualification and certification of special process personnel as well as the procedures for the performance of special processes. The Applicant, where applicable, requires that performance procedures be in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

Welders, where required, will be qualified and certified in accordance with Section III & IX of the ASME Code and nondestructive test personnel will be

qualified and certified to the requirements of Section V of the ASME Code and SNT-TC-1A, "Recommended Practice for Nondestructive Testing Personnel

Qualification and Certification," published by the American Society for Nondestructive Testing. Other special processes such as plating, heat treating and cleaning will require qualification in accordance with specifications developed by the Applicant. Special processes, performed by 28
suppliers will be audited and verified by Product Assurance personnel.

17.1.9.2 Special Processes at Applicant's Facilities

The Applicant will prepare procedures for the performance of special processes which are in accordance with the applicable sections of the ASME Code. The operators and inspectors will be qualified and certified to the applicable sections of the ASME Code.

The Applicant will audit the suppliers to assure their continued compliance with the provisions of the applicable codes, specifications and standards. The Quality Assurance and Reliability organization will audit the Operations, Manufacturing and Quality Control divisions to assure their compliance to applicable codes, specifications and standards.

17.1.10 INSPECTION - QUALITY CONTROL

The Applicant has implemented an inspection program that establishes the 28
independence of Quality Control inspection personnel from the individuals or group performing the activity being inspected and operates independently

of cost or schedules. Quality Control Inspection personnel report to the Quality Control Manager, who in turn reports to the Director of Product Assurance. Quality Control inspection activities are audited by Quality Assurance. 28

Inspection results are documented, evaluated, and product acceptance determined by Quality Control personnel who have been trained and qualified.

17.1.10.1 Procedures, Instructions and Checklists

Inspection procedures, instructions, and checklist are reviewed and approved by the appropriate manager within the Product Assurance organization, and concurrence documented to assure appropriate quality requirements are specified. These procedures, instructions, or checklists shall include as appropriate

(a) Characteristics/activities to be inspected.

(b) Inspection method description. 28

(c) Identification of the individual or group responsible for performing the inspection and their qualification level.

(d) Acceptance/rejection criteria.

(e) Identification of required procedures, drawings, specifications, and revisions used.

(f) Name of inspector performing the inspection and the results of the inspection.

(g) Required measuring and test equipment including accuracy. 28

Procedures, instructions, and checklists are provided to identify the methods of inspection and the characteristics to be inspected. For purchased items this inspection information is utilized by the Applicant's Quality field representatives in accordance with established procedures. Within the Applicant's manufacturing facility this inspection information is maintained for use at specified inspection locations.

The work order package, along with the necessary drawings and specifications, will be available at the inspection location prior to the scheduled inspection operation.

Product Assurance reviews the work order and inserts required inspection, witness and hold points on the work order prior to issue, and assures that appropriate quality requirements have been included on the work order. The sequence and time for performance of required inspections along with the specific type of test or inspection is determined by review of the current specification, code or standard to which the item will be fabricated. The sequence of required inspections will not be changed until the revised work order has been reviewed and concurred with by Quality Assurance. 28

Quality Control Inspection Instructions (Q.C.I.I.) and Inspection Data Reports (I.D.R.) transmit to the inspection personnel applicable information on the specification and drawings including acceptance criteria, the method of inspection, when required, and assigns use of special inspection tools and gauges.

Inspections of modifications, repairs and replacement items which are made after inital inspection will be performed in accordance with the original design and inspection requirements or acceptable alternatives per approved procedures which will verify acceptability. Documentation of all modifications, repairs and replacements will be maintained.

17.1.10.2 Responsibility

Quality Control Engineer

The Quality Control Engineer, in addition to preparation of the original Q.C.I.I.'s and I.D.R.'s will assure that inspections of modifications, repairs, and replaced items made after the initial inspection are performed per the original requirements or approved alternatives.

Quality Control Inspectors

The Quality Control Division will be staffed with inspectors and NDE specialists who satisfy the minimum education and work experience requirements specified in the Applicant's position descriptions. In addition, Quality Control Inspectors including nondestructive examination personnel engaged in inspection and nondestructive examination activities governed by Regulatory Codes and Standards will be qualified and certified as Level I, 28

II, and III categories in accordance with ANSI N45.2.6, "Qualifications of Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants."

A written personnel qualification and certification program has been established for Quality Control Inspection personnel. Training, examination qualification and certification data are documented. Quality personnel are periodically evaluated in accordance with written procedures to determine continued proficiency in the methods certified. This re-certification is documented. 28

The Quality Control Inspector will, upon acceptance of an inspection characteristic or other verification function, indicate the acceptance status of the item by stamp, or signature and date on the applicable work order. When rejection results from the inspection activity, a Quality Deviation Report is prepared and the work order noted with the report number which indicates follow-up action is required.

Quality Control will initiate a Quality Deviation Report for those non-conformities that cannot be corrected by rework of the item within the parameters of the specification or drawing.

The Quality Control Inspector is responsible for assuring that inspection tools and gauges that he uses are maintained in calibration and to take prompt action for the re-certification of inpsection tools and gauges found to be out of calibration. When inspection, measuring and test equipment

are found to be out of calibration, an evaluation will be made and documented of the validity of previous inspection or test results and of the acceptability of items previously inspected or tested. The Quality Control Inspector will perform process control audits where inspection of the product is not possible.

17.1.11 TEST CONTROL

The Applicant will establish an overall test program to demonstrate that safety-related structures, systems, and components will perform satisfactorily within the design acceptance limits. This program includes "proof" tests prior to installation and pre-operational tests. Test procedures provide criteria for determining accuracy requirements of test equipment. Timing of tests and under what conditions the test must be performed are determined by reviewing the applicable specifications, code or standard to which the item has been fabricated. 28

17.1.11.1 Test Requirements

Engineering documents, for example, "Systems Specifications" and "Equipment Specifications", list and describe the tests necessary to demonstrate compliance with specification requirements and design accuracy.

17.1.11.2 Test Procedures

Test procedures, written by the Applicant or his suppliers and incorporating the Engineering specification test requirements, are reviewed,

concurred with and the review documented by the Applicant's product assurance organization.

The sequence of required test will not be changed until the revised test procedure has been reviewed and concurred with by the original organizations reviewing the document including product assurance. Quality and Reliability Procedures have been written which describe how test procedures are to be reviews and list those general requirements which are to be included in the test procedure. The general requirements which are to be included where applicable are as follows: 28

1. The test procedure is identified by a procedure number.

2. The Engineering specification documents are referenced.

3. The instruments and equipment required for test are identified by name, model, range and accuracy.

4. Checks to assure instruments and controls including installed equipment are properly calibrated.

5. Checks to assure that the items to be tested have been prepared, inspected and are in proper condition and state of completeness and all prerequisites have been met prior to initiation of the test. 28

6. The conditions are specified that must be maintained during the test, including ambient conditions, environmental conditions, and safety

precautions to be observed to prevent personnel injury or damage to test items or instruments.

7. The requirements for test personnel are specified such as training, qualification, certification or licensing.

8. A detailed description of the test methods to be used.

9. Applicable schematics of the test setup, e.g., electrical, pneumatic and hydraulic.

10. A test schedule which specifies the required sequence of tests.

11. The acceptance criteria such as the variables to be measured and the accuracy or tolerances required as determined from applicable design documents.

12. A list of mandatory inspection hold points, where applicable, for witness by the Applicant, owner or regulatory agency.

13. Sample data sheets to be used in recording the test data. The data sheets should include identification to the part number, serial number, test procedure, and work order or purchase order.

14. Filing and record retention requirements. These requirements shall specify who is responsible for filing, where the records are to be

filed, for how long, and what the record dispostion requirements are, upon completion of the filing period.

15. A list of responsibilities which include:

    a. Who performs the test.
    b. Who witnesses the test.
    c. Who records test data.
    d. Who evaluates the test data.
    e. Who approves the test data.
    f. Who files the test data.

16. Data sheets/reports which record test parameters and results are completed and signed by responsible individuals within the test organization. Quality Control personnel will evaluate the data sheets, verify selected data, and document that the required test and associated documents have been properly completed and meet the acceptance rejection criteria. 28

Inspections and tests of modifications, repairs and replacement items will be performed to verify acceptability in accordance with those design, inspection and test requirements to which they were previously evaluated or with acceptable alternatives per approved procedures. Documentation of all modifications, repairs, and replacement actions will be controlled under the Applicant's Quality Assurance Records Program (Reference 17.1.1.17).

17.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

17.1.12.1 Control

Quality Control is responsible for the identification, calibration, control and documentation of NDE and radiation safety equipment.

All other inspection equipment will be controlled by Operations. The method and frequency of calibration and the methods of issue and use control for each inspection device shall be defined by written procedures. Methods and frequencies of calibration shall be based on the type of equipment, stability characteristics, required accuracy, degree of usage and other conditions affection measurement control. All inspection equipment will be verified as being within calibration limits prior to assignment. Inspection personnel will verify that inspection equipment is within calibration due dates prior to performing an inspection operation.

17.1.12.2 Receipt and Initial Calibration

All incoming gaging devices shall be initially certified in accordance with calibration procedures specified in written procedures, assigned a control number and identified accordingly before being placed in use.

17.1.12.3 Calibration Status

Inspection devices found acceptable shall be labeled or color coded to show the date on which new calibration is due.

17.1.12.4 Standards

All Measuring and Test Equipment (M&TE) will be calibrated by using certified measurement standards whose accuracy is:

1. Traceable to the National Bureau of Standards OR

2. Obtained from independent reproducible standards derived from accepted values of natural physical constants OR

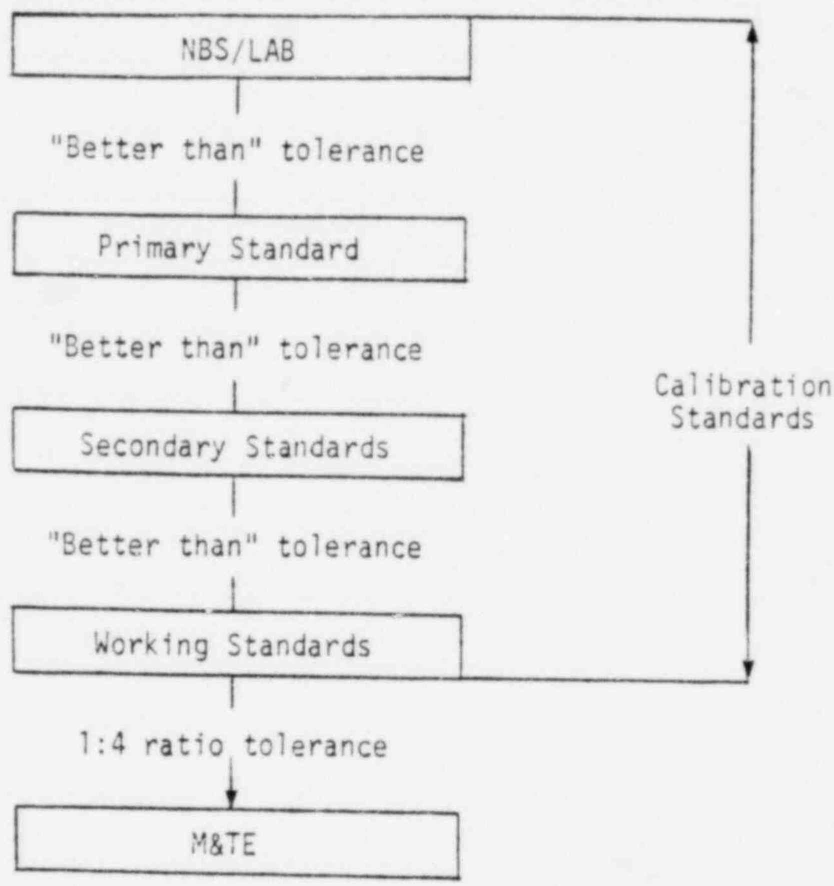
3. Derived from the ratic type of self calibration techniques.

Calibration standards will have an accuracy, range, and stability which are adequate to verify that the equipment being calibrated is within the required tolerance. The tolerance of higher level calibration standards will be better than the tolerance of lower level calibration standards being calibrated.

Measuring and test equipment (MT&E) used for measuring, gauging, testing, inspection or control to determine compliance with design, specifications, or other technical requirements will be calibrated against standards having tolerances not greater than one-fourth the required tolerance of the M&TE being calibrated.

M&TE which is used both for calibration and measuring or test will follow the one-fourth tolerance requirement. In situations where it is impracti-

cal to comply with the above, calibration equipment of a lesser accuracy is allowed providing (1) the justification and basis are documented and authorized by responsible management and (2) the equipment being calibrated can be shown to be within the required tolerance after calibration. The following calibration diagram is provided for illustration:



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17.1.12.5 Identification

Where practical each measuring device shall be marked with a permanent identification control number. Special measuring devices with unique configurations may be identified by other means.

17.1.12.6 Records

Lists of all measurement and test equipment and calibration standards shall be maintained.

Lists shall include the following information:

1. Control Number

2. Description

3. Manufacturer

4. Purchase Order Number

5. Date Received

6. Equipment Location

An individual Calibration Record Card shall be maintained for each device and calibration standard listed.

The required information and calibration history of each device shall be recorded on the Calibration Record Card. This shall include but not be limited to: The date of last caliabration, by whom calibrated, and the date on which new calibration is due.

FNP instruments and gages are not included in the inspection equipment calibration program, but are calibrated and/or adjusted as part of the test program in compliance with engineering test specifications.

17.1.13 HANDLING, STORAGE AND SHIPPING

17.1.13.1 Requirements

The Applicant and his suppliers, within the scope of their work, will establish specifications, procedures and special instruction to control the handling, cleaning, preservation, shipping and storage of material,

components and equipment in accordance with applicable design and procurement requirements to prevent damage, loss or deterioration. When required, special protective environments and/or maintenance procedures will be specified for shipment and storage. Product Assurance will review and document concurrence with these procedures and instructions. 28

17.1.13.2 Compliance

1. At Suppliers

When surveillance inspection is required determination of compliance with the applicable control documents is the responsibility of the Applicant's Quality Surveillance representative. Written Quality Procedures will describe the procedures, check sheets and documentation necessary to assure that the required operations are performed.

2. At Offshore Power Systems

The Applicant's Material Services Department is responsible for receiving and storing all material and equipment in accordance with specification, code and customer requirements. The Applicant's Quality Control Department is responsible for verifying that the quality of material and equipment is maintained from the time of receipt, through assembly and preoperational testing to final delivery of the Floating Nuclear Plant. Qualified personnel will implement written Quality Control Procedures which describe the procedures, check sheets and documentation necessary to assure that the required operations are performed.

17.1.13.3 Audits

Compliance with the program is verified by scheduled Quality Assurance audits as defined in the Applicant's Quality Assurance Audit Program referenced in Section 17.1.18.

17.1.14 INSPECTION, TEST, AND OPERATING STATUS

The Applicant's quality assurance program provides written procedures to assure that inspection, test and operating status of systems, structures, components and materials are evident at all times throughout manufacturing and installation. In this section the word "material" will apply to any material, item, component, system or structure of the Floating Nuclear Plant. Product Assurance reviews and documents concurrence with these written procedures. 28

17.1.14.1 Status Identification

Markings, such as stamps, tags, seals, decals, routing cards or other suitable means to indicate inspection, test or operating status will be applied to or accompany the material. At Offshore Power Systems, the Work Order will accompany the material at all times during the processing and inspection of product and will be the primary control for indicating the inspection, test, and operating status of materials. The Work Order describes the step-by-step sequence of operations to be performed as well as indicating the process and inspection status.

17.1.14.2 Material Awaiting Inspection

Materials received from suppliers are identified and processed by Receiving. The accompanying data and material receipt, for materials requiring Quality Control inspection as specified in the purchase order, are forwarded to Quality Control inspection. Quality Control inspects the material in accordance with written procedures.

Other materials awaiting inspection, which are in the process of manufacture or assembly, are identified as described in Section 17.1.14.1.

17.1.14.3 Accepted Material

Accepted material will be identified by an inspector's signature or stamp on the work order or material receiving document in the space provided for acceptance status. An acceptance tag, sticker, or decal will be attached to or accompany the material after final acceptance. The material receiving report serves the same function as the work order for materials received from suppliers. 28

17.1.14.4 Rejected Material

Rejected or nonconforming material will be identified with a Quality Hold Tag in accordance with written "Quality Control Procedures."

17.1.14.5 Operating Status

Switches, valves or other items which require control of their operating status will be identified with an appropriate tag in accordance with written procedures. Product Assurance reviews and documents concurrence with these written procedures. 28

17.1.14.6 Tag Removal

Tags markings, labels and stamps applied by Quality Control Inspection or operations personnel may not be removed by anyone except authorized personnel. Procedures describing the application of tags will also 28 describe those activities or personnel authorized to remove them. Product Assurance reviews and documents concurrence with these written procedures.

17.1.14.7 Supplier Compliance

The Applicant's Quality Surveillance representative will assure that the inspection status of Offshore Power Systems' material in suppliers' plants is evident at all times throughout the manufacturing cycle as required by contract.

17.1.14.8 Applicant Compliance

The Inspection Department of the Quality Control Division is responsible for assuring that the inspection and operating status of material at

Offshore Power Systems is evident at all times throughout the manufacturing, assembly and test cycle. Procedures for implementing this requirement are defined in written "Quality Control Procedures."

17.1.14.9 Stamp Control

Procedures for the control of issuance, use and withdrawal of inspection stamps will be issued by the Quality Control Division.

17.1.14.10 Inspection Points

1. At Suppliers

Suppliers of safety related equipment are required to submit their manufacturing and inspection plans to the Applicant's Quality Assurance and Reliability Division for review, approval, and insertion of the Applicant's inspection, verification or hold points prior to the performance of work via an Approval Request (A.R.) form. This assures that appropriate operations, inspections and tests are performed and that the Applicant's hold points are established. Deviations or bypassing of these approved plans require resubmittal and approval by personnel from the organization that signed the original A.R. The issue and control of this document is defined in a written Quality and Reliability Procedure (QRP) as well as the "Quality Requirements for Suppliers" document which is part of the procurement document package.

2. At Offshore Power Systems

The "Work Order Process Sheet" which is reviewed and approved by the Applicant's Quality Control Division prior to the performance of work, defines the operations, inspections and tests required to perform the task and to assure compliance to specification requirements. The bypassing of any of these operations, inspections or tests is controlled by Quality Control per written procedures in accordance with the quality assurance program.

17.1.14.11 Audits

Quality Assurance will perform schedules audits of the Applicant's facility to assure that the inspection, test and operating status of systems, structures, components and materials is performed in accordance with governing procedures.

17.1.15 NONCONFORMING MATERIALS, PARTS OR COMPONENTS

Applicant will establish and implement a procedure for the control of materials, parts and components which do not conform to the specification or purchase order requirements.

17.1.15.1 Nonconformity at Suppliers

Suppliers are required to have documented procedures for the control of items which do not conform to requirements. These procedures shall include measures for identification, documentation, segregation, disposition, and notification to affected organizations. Product Assurance will review and document concurrence with these procedures. The supplier shall have an established procedure for the review and analysis of discrepant material. The material review system will include review and approval, by Product Assurance of all nonconformities which are dispositioned "Use As Is," "Repair,", or any changes to the technical requirements of the purchase order. Tentatively approved changes which could affect Utility Owner interchangeability or interfaces are coordinated with Utility Owner representative. The supplier accomplishes this review and approval by submitting an Offshore Power Systems' Quality Assurance and Reliability "Deviation Notice" to the Applicant's quality Assurance and Reliability organization through Purchasing. The "Deviation Notice" is a controlled multi-copy document which provides tear-off copies for each affected organization providing input to the "Deviation Notice." Quality Assurance and Reliability meets with Engineering, and, when required, with the Utility Owner representative to arrive at a disposition of the nonconformity or deviation.

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The dispositioned supplier copy of the "Deviation Notice" is returned to the supplier through Purchasing. A Quality representative will audit the supplier to assure compliance to the requirements of the disposition. Quality Assurance and Reliability maintains a system for logging, filing

and distributing Deviation Notices to affected organizations for necessary corrective action or follow-up. A copy of all dispositioned Deviation Notices is provided to the Utility Owner's Quality representative.

17.1.15.2 Nonconformity at Manufacturing Facility

Within the Applicant's facility, quality Control is responsible for identifying and reporting nonconformities. Until any nonconforming material has been dispositioned as acceptable, Quality Control will identify and control segregation of such material to prevent unauthorized use. Quality Control will also conduct a preliminary review of the nonconformity to determine the nature and severity of the nonconformity. Those "Rework" nonconformities which can be made to conform to the specification or drawing requirements with additional work and abvious "Scrap" decisions are dispositioned directly by Quality Control. Nonconformities that required a "Use As Is" or "Repair" disposition are forwarded to Quality Assurance and Reliability for disposition in accordace with the Quality and Reliability procedure on nonconforming material. Tentatively approved "Use As Is" or "Repair" disposition which could affect Utility Owner interchangeability or interface are coordinated with the Utility Owner representative. When the disposition has been completed, the Nonconformity Report is returned to Quality Control for action to implement the dispostion. Quality Assurance and Reliability audits Quality Control to verify proper disposition of nonconformities. Copies of all "Use As Is" and "Repair" nonconformity dispositions are provided to the Utility Owner's Quality Representative.

17.1.15.3 Nonconformity Records 28

Reports of nonconformities dispositioned "use as is" or "repair" become a part of the permanent quality records for the part, component ~ system as defined in Section 17.1.17. The quality Assurance and Reliability Division will periodically use the records to perform quality and reliability data analysis. These analyses will consist of the review and analysis of field reports, nonconforming material documents, inspection records, audit reports and supplier surveillance reports for potential or recurring problems which could indicated the development of trends. Preventive action or corrective action programs will be implemented to prevent occurrence or recurrence of discrepancies.

17.1.16 CORRECTIVE ACTION

Product Assurance will establish and implement procedures to assure that 28
conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance are properly identified and corrected. Product Assurance will verify that corrective action has been specified and assure that it is carried out in a timely manner.

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Product Assurance Reviews and documents concurrence for all procedures related to corrective action.

17.1.16.1 Corrective Action at Suppliers

When the supplier's performance is judged by the Applicant to be of substandard quality, a Request for Corrective Action will be used by the Applicant to inform the supplier of its deficiencies. The supplier will respond to this notice by indicating the action he intends taking to correct immediate deficiences and to prevent recurrence of the deficiency. The Applicant will review the supplier's proposed corrective action and notify the supplier of his acceptance or rejection of the proposal. Disregard for or unsatisfactory resolution of a Request for Corrective Action may be cause for termination of the purchase order and removal from Purchasing's list of approved suppliers.

The supplier is required to have a system for obtaining corrective action from his subcontractors. This system will be approved and audited by the Applicant.

17.1.16.2 Internal Corrective Action by the Applicant

With the applicant's organization, Product Assurance has overall responsibility for implementation of an effective corrective action program.

Internal Corrective Action Requests will be issued by Product Assurance to the responsible department(s) based upon:

a. Continual recurrence of similar nonconformances.

b. Analysis of quality data revealing adverse trends related to specific processes.

c. Noncompliance with established procedures or ineffective resolution of audit results.

The manager of the affected department must investigate the reported deficiency for identification of the responsible cause(s) and development of action to be taken to preclude recurrence of the adverse condition.

The proposed corrective action will be evaluted for acceptability by the requestor. Unsatisfactory corrective action replies will result in an additional request directed to a higher level of management within the responsible department. Following agreement as to the adequacy of corrective action, the requestor will perform a follow-up review to verify that the action has been implemented as stated and is effective in preventing further problems.

17.1.16.3 Reporting and Records

Corrective Action documents which describe the significant condition adverse to quality, the cause of the conditions and the corrective action taken to preclude repetition are distributed to cognizant levels of management for review and assessment. As a minimum the documents are distributed to the initiator, the appropriate levels of Product Assurance management, the appropriate levels of the affected organization management and Purchasing (when supplier are involved).

17.1.17 QUALITY ASSURANCE RECORDS

The Applicant's Quality Assurance Records Program will be developed in accordance with Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records", as stated in Appendix 3D of this report.

17.1.17.1 Record Catagories

The Quality Assurance and Reliability Procedure (QRP) will define two categories of records, Lifetime and Nonpermanent. An index will be included as part of the procedure which lists the type of records, its category and its retention time.

1. Lifetime Records

   Lifetime Quality Assurance Records are required to be maintained by or for the plant owner for the life of the particular item while it is installed in the plant or stored for future use. Lifetime quality Assurance Records will be transferred to the owner on a regular systematic basis as part of the contract between the Owner and the Applicant. The final transfer of records will be at a mutually agreed time between the Applicant and the Owner but prior to the time the Floating Nuclear Plant is made ready for operation.

2. Nonpermanent Records

Nonpermanent records are required to show evidence that an activity was performed in accordance with the applicable requirements but need not be retained for the life of the item. Records classified as nonpermanent will be retained for at least the minimum period of time as specified in the index. After this time these records will be disposed of by or with concurrence of the owner.

17.1.17.2 Record Keeping Requirements

Transmittal, retention and maintenance requirements for records as specified in codes or standards will be reference in applicable control documents.

17.1.17.3 Types of Quality Assurance Records

Quality Assurance Records will provide documentary evidence of the quality and will include operating logs, results of review, inspections, tests, audits, and monitoring of work performance and material analysis; the qualification of personnel, procedures, and equipment; other documentation such as drawings, specifications, procurement documents, calibration procedures, calibration reports, and nonconforming and corrective action reports. Inspection and test records will contain the following:

1. A description of the type of operation.

2. Evidence of completing and/or verifying a manufacturing, inspection or test operation.

3. The results of the inspection or test.

4. Information related to nonconformances.

5. Inspector or data recorder.

6. Acceptability.

17.1.17.4 Filing Requirements

1. Identification

Quality Assurance Records will provide sufficient information to permit identification of the record with the item, items, or activity to which it aoplies.

2. Filing, Storage, and Security

The quality assurance record files will be stored in predetermined locations as necessary to meet the requirements of applicable codes, standards and regulatory agencies.

The Applicant complies with the requirements of Reg. Guide 1.88 and ANSI N45.2.9-1974 for the storage of records. The system is designed to utilize the concept of maintenance of duplicat records stored in a separate remote location.

Where working documents are not duplicated they will be stored in a fire-protected area that complies with features cited in ANSI N45.2.9-1974.

3. Availability and Retrieval

To assure their availability, specific submittal plans will be established for supplier quality assurance records in the purchase order or contract.

Quality assurance records maintained by a supplier at his facility or other location will be accessible to the applicant or the owner for the period defined in the purchase order or contract.

The Applicant's storage system will provide for the accurate retrieval of information without undue delay. A list will designate those personnel who will have access to the files.

17.1.17.5 Audits

Scheduled Quality Assurance Audits will be conducted to assure compliance to the Quality Assurance Records Program.

17.1.18 AUDITS

The Quality Assurance Audit Program is controlled by written procedures, contained in the Quality Assurance Manual, which defines requirements and guidelines for scheduling, Planning, conducting and documenting audits to assure compliance to the Quality Assurance Program. The Applicant's audit program complies with ANSI N45.2.12 - 1977 with the exception that the program provides for audits to be conducted each calendar year with a maximum of 18 months elapsed time.

17.1.18.1 Audit Schedules

The Manager of Quality Assurance and Reliability of his designated representative is responsible for establishing random and scheduled audits. Internal and supplier audit schedule sheets are used to plan comprehensive audit systems which assure that audits are performed in a sequential and systematic manner to assure compliance with the requirements of an activity or contract. Audits of major supplier may include audits of their suppliers to assure compliance to the quality assurance program. Audits are scheduled and initiated early enough to assure effective quality assurance during the design, procurement and contracting process.

Audit schedules are based on the following considerations:

1. Compliance and determination of effectiveness of the quality assurance program.

2. Implementation and effective of procedures or practices.

3. Significant change in the management of an activity.

4. New location of an activity.

5. Deficiencies attributable to an activity.

6. Implementation and effectiveness of corrective actions.

7. Length of time since last audit.

17.1.18.2 Audit Teams

The Manager of Quality Assurance and Reliability or his designated representative selects audit teams and appoints team leaders. Verification of conformance to established requirements will be performed by Quality Assurance Personnel who do not have direct responsibility for the catagory of work being verified. 28

Selection of team members is based upon audit experience, particular expertise in the area being audited, and general capability to supplement

the audit effort in some way, (for example, a unique technical knowledge in the area being audited, such as a technical consultant with an expert knowledge of the audit topic). The team leader is responsible for the briefing of team members, assignment of tasks, performance and reporting of the audit and corrective action followup.

17.1.18.3 Audit Notification

The activity being audited is normall notified by the audit team leader or, in the case of supplier audits by the cognizant buyer, at least five working days before the proposed audit to: define the general scope and purpose of the audit, establish an audit schedule, assure that appropriate personnel from the organizations are available at the time of the audit, and when possible, to obtain operation manuals and other documents pertaining to the activity which may be reviewed prior to performance of the audit.

17.1.18.4 Audit Preparation

The team leader obtains and distributes to the applicable team members available documents pertaining to the activity being audited. Checklists are developed in accordance with written procedures for: Supplier quality assurance program audits, products and process audits, and internal quality assurance program audits which include audits of Engineering, Purchasing, Material Control, Manufacturing and Quality Control. These audit procedures and check sheets are designed to measure, by objective evaluation of quality related practices, procedures and instructions, that the quality

assurance program is implemented and effective. They also provide measures for evaluating work areas, activities, processes, products, documents and records. A written procedure also defines requirements for Offshore Power Systems' participation and documentation of audits of Offshore Power Systesm quality assurance program by outside organizations, such as consultants, customers, NRC and the ASME. 28

Prior to each audit the audit team meets to establish the audit agenda. The audit procedures describe the minimum requirements to be included in the agenda.

17.1.18.5 Audit Performance

The audit team conducts the audit in accordance with the agenda. Additions are made to the agenda if audit findings indicts such to be necessary to assure the activity is meeting the quality and technical requirements as stated in written procedural requirements. If procedures are not available or are suspected to be inadequate, further investagation is pursued to determine by evaluation of objective evidence that audit objectives are satisfied.

17.1.18.6 Exit Conference

Upon completion of the audit, the audit team holds an exit conference with the manager of the audited activity to discusss the audit findings. The audit team proposes recommendations for corrective action of deficiences and obtain a tentative schedule for completion of corrective action.

17.1.18.7 Audit Report

Within a designated period following an audit, the team leader issues a formal audit report. Copies of the audit report are sent to the management having responsibility in the area audited to assure correction of deficient areas revealed by the audit. Supplier deficiencies discovered as a result of audits are reported to the supplier via formal letter from Purchasing.

17.1.18.8 Corrective Action Follow-up

Audit deficiencies are recorded on Audit Deficiency Report (ADR) forms.

Audit Deficiency Report form(s) are submitted to both OPS management and Supplier personnel who have the responsibility within their organization for supplying corrective action. The corrective action reply is recorded on the form and submitted to the auditor by the reply due date. Documentation which substantiates the corrective action is attached to the Audit Deficiency Report form. The auditor evaluates the reply and either concurs by signing and dating the response, or re-submits the form to the audited organization for futher corrective action.

The Lead Auditor assigns a team memeber to perform the internal OPS follow-up audit to assure implementation of corrective action as stated on the Audit Deficiency Report. Supplier follow-up audits may be performed by a member of the audit team. A resident or itinerant quality representative may, at the request of the Lead Auditor, perform follow-up verification of

corrective action implementation. When the corrective action has been verified, the Lead Auditor signs the Audit Deficiency Report indicating acceptance and close out.

17.1.18.9 Filing and Distribution

Audit reports are filed and distributed per directions included in audit procedures.

17.1.18.10 Quality Assurance Audit Log

An audit log is maintained in the quality assurance audit file to provide a ready reference for quality assurance audits performed by the Applicant or in the Applicant's facility. The quality assurance audit log contains the following information: the audit report identification number, the name of the audited activity or supplier, the date the audit was completed, and the corrective action completion date.

17.1.18.11 Audit Analysis

Audit data is analyzed by quality assurance to determine trends in quality problems and continued areasof noncompliance. The results are used to determine the effectiveness of the quality program and the necessity for reaudit of deficient areas. Reports of audit analysis are presented to Management for review and assessment. 28

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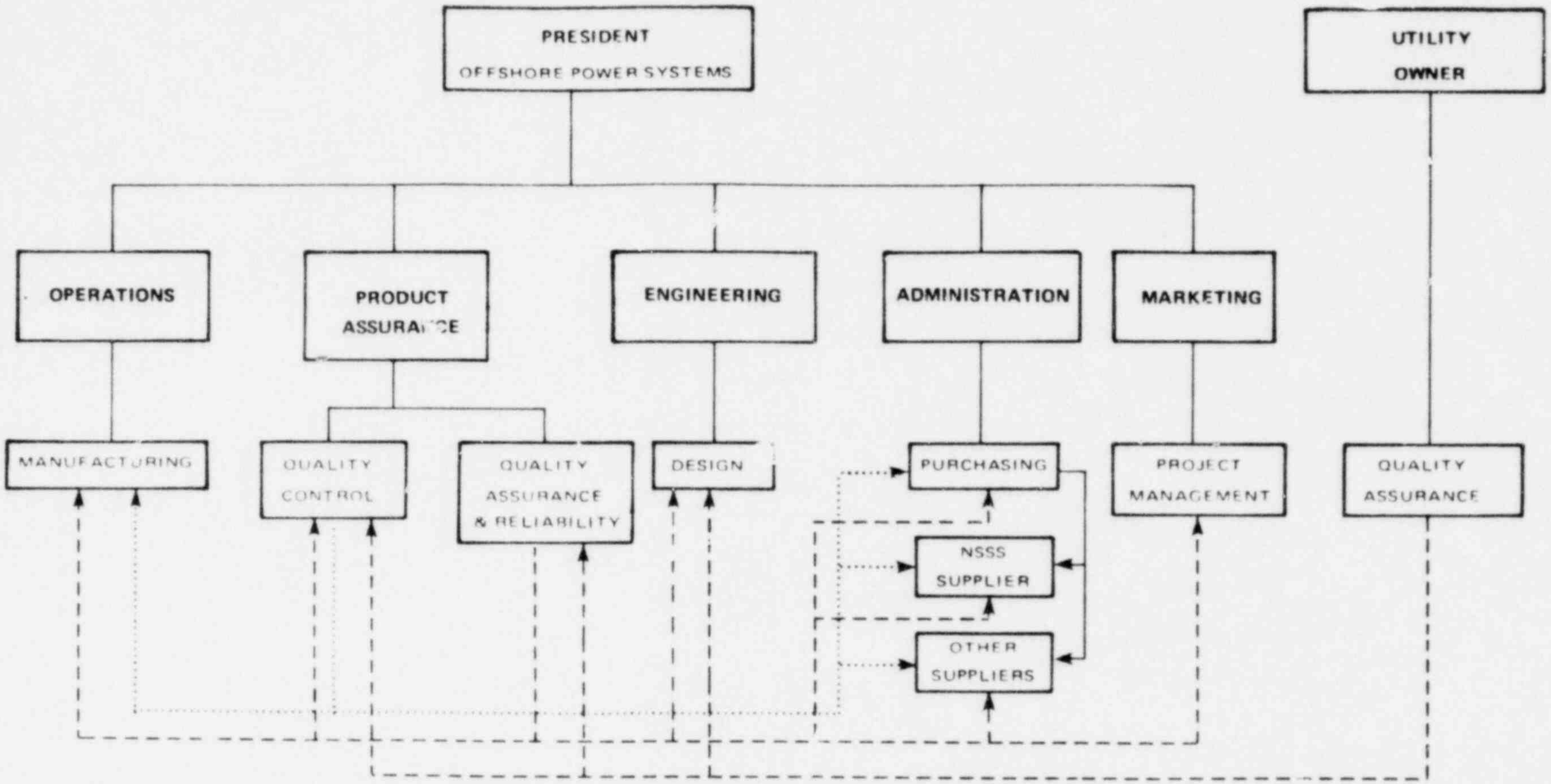
QUALITY ORGANIZATION RELATIONSHIP



——— DIRECT RESPONSIBILITY

— — — QUALITY ASSURANCE PROGRAM DIRECTION AND AUDIT RESPONSIBILITY

········ PRODUCT QUALITY VERIFICATION RESPONSIBILITY

* WHEN SUPPLIER NOTIFIES OPS PURCHASING OF PLANNED INSPECTION/PRODUCT RELEASE DATES, PURCHASING WILL COORDINATE DIRECTLY WITH THE QUALITY CONTROL ACTIVITY.

QUALITY ORGANIZATION RELATIONSHIPS

FIGURE 17.1-2

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APPENDIX B

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ENCLOSURE OCT 17 1979

REACTOR ANALYSIS SECTION, ANALYSIS BRANCH

221.1 (4.4) Provide the radial pressure gradient in the upper and lower plenums and at the core inlet and outlet for steady state and transient conditions for each allowable loop configuration. Provide an explanation of how the radial pressure gradients are included in the thermal-hydraulic design calculations. Discuss and support by calculations, the differences in hot channel pressure drop, flow, enthalpy rise and minimum DNBR relative to the assumption of a uniform pressure at the core boundaries.

221.2 (4.4) Provide an explanation of how the effects on the core flow and pressure drop of possible crud deposits are included in the thermal-hydraulic design.

Provide a description of the instrumentation available which would alert the reactor operator to an abnormal core flow or core pressure drop during steady-state operation.

221.3 (4.4) Provide a commitment to address the following aspects of rod bowing in the FSAR for plants referencing FNP (1-8):

1. to fully define the gap closure rate for prototypical bundles;

2. to determine by appropriate experiments the DNB effects that bounds the effect of gap closure;

3. to include the effects of rod bowing in the final design and safety analysis calculations.

221.4 (4.4) Floating Nuclear Plants (1-8) used the HYDNA code to describe the effect of open channel flow on thermal-hydraulic flow instability. Information supplied by Westinghouse has been insufficient to support a conclusion that the DYDNA code conservatively predicts the onset of flow instability in the core. To support such a conclusion, either (1) provide a complete description of the HYNDA code and its use in the analysis, or (2) provide a discussion excluding the HYDNA code which supports the contention that the core is thermal-hydraulically stable.

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July 15, 1981

APPLICANT'S RESPONSES

NRC Question No.	Response
221.1	DNB analyses are based on uniform inlet velocity and exit pressure distributions. Data from several 1/7 scale model tests and THINC analyses of various inlet flow distributions have led to a conservative design basis of 5% reduction in flow to the hot assembly. Section 5.6 of Reference 1 presents analyses which verify the adequacy of the design assumption of a uniform exit pressure distribution.

The effect of core outlet radial pressure gradients on DNB analysis has been shown to be negligible in four-loop 193 assembly cores. An analysis was performed which assumes a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and 0 psi at the core periphery for four-loop and three-loop operation. The results of these analyses showed that there was no effect on the minimum DNBR (to three significant figures) of this radial pressure gradient on four-loop or three-loop operation.

In performing this analysis the hot assembly was assumed to be in the center of the core where the greatest flow reduction near the core outlet will occur due to the radial pressure gradient. In addition, an axial power distribution extremely peaked to the top of the core (+30% axial offset) was assumed. This axial power distribution is more severe than would be expected during plant operation.

Thus, the use of a uniform upper plenum pressure distribution in thermal-hydraulic design is acceptable.

Reference:

(1) Hockreiter, L.E., and Chelemer, H., "Application of the THINK-IV Program to PWR Design", WCAP-8504 (Proprietary) and WCAP-8155, September 1973.

221.2 Operating experience to date has indicated that a flow resistance allowance for possible crud deposition is not required. There has been no detectable long-term flow reduction reported at any plant. Inspection of the inside surfaces of steam generator tubes removed from operating plants has confirmed that there is no significant surface deposition that would affect system flow. Although all of the coolant piping surfaces have not been inspected, the small piping friction contribution to the total system resistance and the lack of significant deposition on piping near steam generator nozzles support the conclusion that an allowance for piping deposition is not necessary. The effect of crud enters into the calculation of core pressure drop through the fuel rod frictional component by use of a surface roughness factor. Present analyses utilize a surface roughness value which is a factor of three greater than the best estimate obtained from crud measurements from several operating Westinghouse reactors.

Instrumentation available to alert the operator to abnormal core flow or core pressure drop is as follows:

1. Primary flow indication is provided by the RCS flow meters. There are 3/loop and read from 0-100%. Any significant flow reduction would appear on these meters.

2. There are several methods that could be used to infer flow. They are:

   a. With rods in the automatic control mode, reduced flow would result in lower core power as rods drove in attempting to maintain Tavg.

   b. With rods in the manual control mode, reduced flow would result in higher Tavg.

   c. RCP amps reading higher or lower than normal, could indicate abnormal flow (pump or motor malfunction primarily).

   d. Significant flow reductions in a particular core quadrant could be indicated by a power mismatch between the various power range detectors.

   e. Sustained local flow stoppages could be detected by incore flux maps and core exit thermocouples.

3. There are alarms that would alert the operator to low RCS flow as indicated by the RCS flow meters, high RCS temperatures, abnormal RCP and motor temperatures and RCP trip.

4. Reactor trips are generated by low RCS flow, low RCP bus voltage and frequency, and high temperature.

221.3 DNB analyses (which will be reported during the final FNP design approval phase) will be performed such that generic DNBR margins described in the "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (Revision 1) February 16, 1977" will be available for offsetting rod bow penalties. The appropriate rod bow penalty and any operating restriction in the technical specifications, if required, will be addressed prior to the issuance of an Operating License to the owner of the first FNP.

221.4 Boiling flows may be susceptible to thermohydrodynamic instabilities[1]. These instabilities are undesirable in reactors since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermohydraulic design was developed such that operation under Condition 1 or 11 events does not lead to thermohydrodynamic instabilities.

Two specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg or flow excursion type of static instability and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs[1] when the slope off the reactor coolant system pressure drop-flow rate curve

$$\left.\frac{\partial \Delta P}{\partial \Delta G}\right|_{INTERNAL}$$

becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve

$$\left.\frac{\partial \Delta P}{\partial \Delta G}\right|_{EXTERNAL}$$

The criterion for stability is thus

$$\left.\frac{\partial \Delta P}{\partial \Delta G}\right|_{INTERNAL} > \left.\frac{\partial \Delta P}{\partial \Delta G}\right|_{EXTERNAL}$$

The Westinghouse pump head curve has a negative slope

$$\left.\frac{\partial \Delta P}{\partial \Delta G}\right|_{EXTERNAL} < 0$$

whereas the reactor coolant system pressure drop-flow curve has a positive slope

$$\left.\frac{\partial \Delta P}{\partial \Delta G}\right|_{INTERNAL} > 0$$

over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody.[2] Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single phase region and causes quality or void perturbations in the two-phase regions which travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, the two phase pressure drop perturbation feeds back to the single phase region. These resulting perturbations can be either attenuated or self sustained.

A simple method has been developed by Ishii[3] for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method had been used to assess the stability of typical Westinghouse reactor designs[4,5,6] including Virgil C. Summer, under Condition I and II operation. The results indicate that a large margin to density wave instability exists, e.g., increases on the order of 200% of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii[3] to Westinghouse reactor designs is conservative due to the parallel open channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high power density. There is also energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions. Flow stability tests[7] have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross connected at several locations. The cross connections were such that the resistance to channel-to-channel cross flow and enthalpy perturbations would be greater than that which would exist in a PWR core which has a relatively low resistance to cross flow.

Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at low pressures relative to the Westinghouse PWR operating pressures. Kao, Morgan and Parker[8] analyzed parallel closed channel stability experiments simulating a reactor core flow. These experiements were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many Westinghouse rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundle.

In summary, it is concluded that thermohydrodynamic instabilities will not occur under Condition I and II of operation for Westinghouse PWR reactor designs. A large power margin, greater than doubling rated power, exists

to predicted inception of such instabilities. Analysis has been performed which shows that minor plant to plant differences in Westinghouse reactor designs such as fuel assembly arrays, core power to flow ratios, fuel assembly length, etc. will not result in gross deterioration of the above power margins.

References:

(1) J. A. Boure, A. E. Bergles, and L. S. Tong, "Review of Two-Phase Flow Instability," Nucl. Engr. Design 25 (1973), pp. 165-192.

(2) R. T. Lahey and F. J. Moody, "The Thermal Hydraulics of a Boiling Water Reactor," American Nuclear Society, 1977.

(3) P. Saha, M. Ishii, and N. Zuber, "An Experimental Investigation of the Thermally Induced Flow Oscillations in Two-Phase Systems," Journal of Heat Transfer, Nov. 1976, pp. 616-622.

(4) Virgin C. Summer FSAR, Docket #50-395.

(5) Byron/Braidwood FSAR, Docket #50-456.

(6) South Texas FSAR, Docket #50-498.

(7) S. Kakac, T. N. Veziroglu, K. Akyuzlu, O. Berkni, "Sustained and Transient Boiling Flow Instabilities in a Cross-Connected Four-Parallel Channel Upflow System," Proc. of 5th International Heat Transfer Conference, Toyko, Sept. 3-7, 1974.

(8) H. S. Kao, C. D. Morgan, and W. B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," Trans. ANS, Vol. 16, 1973, pp. 212-213.

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APPENDIX C

RESPONSES TO POST-TMI REQUIREMENTS

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NOTE: Numbers in parentheses following the subject are the applicable requirement numbers from NUREG-0178.

APPENDIX C

RESPONSES TO POST-TMI REQUIREMENTS

Appendix C provides the Offshore Power Systems' (the Applicant's) responses to the post-TMI requirements for pending Construction Permit and Manufacturing License Applications. These requirements were issued initially for comment[(1)] in the form of a proposed paragraph (e) to be added to 10CFR 50.34. The basis for the technical requirements set out in the proposed rule making is NUREG-0718[(2)].

In preparation for a Commissioner meeting on May 27, 1981, the Staff made a number of changes to the text of the proposed rule. These changes, which reflect the latest thinking of both the Staff and the Commissioners, is contained in the public document SECY-81-20D[(3)].

At the time of filing Amendment 28, a final rule had not yet been adopted by the Commission. However, during the May 27, 1981 meeting, the Commissioners instructed the Staff to conduct their review of the pending Construction Permit and Manufacturing License applications on the basis of the rule most recently proposed by the Staff for Commission approval. The responses contained in their appendix, therefore, address the proposed rule as it appears in SECY-81-20D. Each response is preceded by a restatement of the relevant section of the proposed rule. The alpha-numeric designator (for example II.B.8) appearing at the end of such each restatement is the section of NUREG-0718, Appendix B, from which the technical content of the rule is drawn.

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(1) Federal Register (46FR18034), March 23, 1981.

(2) USNRC Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License, NUREG-0718, March 1981.

(3) Memorandum from W. J. Dircks to the Commissioners, SECY-81-20D, May 18, 1981, Enclosure 3.

Appendix C was added to the Plant Design Report in Amendment 28. As a result of staff review of Amendment C in preliminary form, a number of text changes were made by the applicant prior to formally filing Amendment 28. In order to highlight changes from the preliminary version, identifying bars have been placed in the margin of Appendix C pages only where such changes were made rather than on the entire page. Changes, if any, which are made to Amendment 28 will be highlighted in the usual manner.

REGULATION 10CFR50.34(e)(1)(i)

Subject: Probablistic Risk Assessment

To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility:

Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)

OFFSHORE POWER SYSTEMS RESPONSE

Introduction

Following is a description of the risk/reliability program for the Floating Nuclear Plant in response to NUREG-0718 including an outline of the program scope, methodology, schedule and applications of results. The objective of the program will be to identify improvements in the reliability of core and containment heat removal functions as are significant and practical and do not impact excessively on the plant. Alternative core and containment heat removal systems will be considered in the study.

A preliminary outline of the report is given as Table C-1.

Project Scope

The FNP risk/reliability program will be similar in scope to the Interim Reliability Evaluation Program (IREP) being performed by the NRC on several operating plants. Individual accident sequences and their probabilities will be analyzed to identify the initiating events and plant system/component failures which are dominant contributors to the potential for core damage. The initiating events to be analyzed will include LOCA's (small and large breaks), transient events, and other accident initiators as determined during the initial phase of the study. As a minimum, the systems listed in Table C-2 will be analyzed to determine if system modifications are appropriate and could significantly reduce overall plant risk. Special consideration will be given to interdependencies of support

systems for any scenarios which could lead to significant release of radioactivity to the environment. Other systems are likely to be added to Table C-2 as the core and containment cooling systems are evaluated with respect to interactions and dependencies.

Program Organization and Responsibilities

The approach to be used in the program will employ the event tree/fault tree methodology similar to that used in WASH 1400 and other comprehensive plant risk studies. OPS is responsible for the PRA study and will ensure that the study will be performed by engineers who are qualified and experienced in risk assessment methodology. Prior to decisions affecting plant design, OPS may have a third party conduct a peer review of the study. OPS will retain ultimate authority and responsibility for implementing any design improvements as a result of this study.

The major tasks involved in the program are discussed below.

Initiating Event Selection

A list will be established of initiating events which, together with system failures, have the potential for causing core damage. This will be accomplished through a screening of the accidents and transients, identified in PDR Chapter 15, WASH 1400, and in other studies, to identify the basic set of initiating events requiring operation of the key safety systems for core protection and release mitigation. The frequency of these initiating events will be estimated based on available data including WASH 1400, EPRI NP-801, and, pertinent plant-specific information. Failures that could occur during cold shutdown, during severe natural phenomena and during fires will also be considered. Natural initiating events to be considered are those listed in Table C-3.

Event Tree Development

For each type of initiating event, an event tree will be constructed, identifying the systems required to mitigate the event and the expected effect on ability to maintain core and containment integrity given success or failure of each system involved. The full event tree will be reduced to reflect system interdependencies and required sequences of operation. Note

that the event trees will address certain non-safety systems such as offsite power and the power conversion system.

System Failure Modes and Effects Analysis (FMEA)

For each safety system involved, a FMEA will be conducted to identify and tabulate component and common cause failures and their effect on system operability for each initiating event. The FMEA will provide documentation of the basis for inclusion or exclusion of specific failure modes in the system fault tree analysis. Failure modes will include mechanical and electrical faults, operator error, maintenance or testing outages, etc. Particular attention will be paid to potential common cause failures which could disable multiple components. Common cause failure mechanisms to be investigated include environmental factors, operator or maintenance errors, passive failures and system interactions.

System Fault Tree Analysis

Using the FMEA as input, fault trees will be constructed for each safety system identifying the failures (basic events) and their logical combinations which will result in system unavailability (top event). The fault tree will be analyzed to determine the minimal cut sets and failure combinations which are the dominant contributors to system unavailability. Using the appropriate component failure data, a quantitative assessment of overall system unavailability and of the dominant cut sets will be performed.

Data Base Development

A component failure data base for use in system fault tree analysis will be developed from recognized reference sources including WASH 1400 and IEEE 500. In addition, prototype-specific failure data will be requested from vendors of selected components (e.g., diesel generators). The data base will identify the types of components and estimated median failure rates on demand and, where appropriate, per hour of continuous operation. Error ranges will be assigned to each median value to reflect the uncertainty in the data base. The data base development task will include methodologies to adjust failure data to account for varying testing and surveillance strategies. Test and maintenance unavailability contributions will be

included based on preliminary technical specifications and typical nuclear plant operating and maintenance procedures.

Human error rates will be estimated for required or corrective actions by control room operators and for maintenance or testing operations which are included as failure modes in the system fault trees. Available human error and performance data, including those provided by NUREG/CR 1278, will be used.

Containment Response Analyses

Based on studies of core melt phenomenology and containment transient calculations which evaluate the interrelated physical processes taking place within containment, containment event trees will be constructed for specific categories of core melt event sequences. These plant event categories will cover all inputs from the plant event tree analysis. The event tree technique permits tracking of the containment response to a degraded core accident. Through careful definition of each node, the containment event tree will be applicable to all core melt accident sequences. Additionally, the framework is provided to evaluate the relative importance to risk of related physical phenomena (such as hydrogen burning) and to evaluate the relative merits of various preventive and mitigative features.

Accident Sequence Probabilities

The unavailability of each system will be calculated by inputting the appropriate failure rate data into the system fault tree analysis. The various accident sequences, as represented by the branches on the event trees will then be quantified by inputting the system failure probabilities determined from the quantitative fault tree analysis. Each individual accident sequence will be classified according to release category and the total probability of a given release category will be obtained by the summation of all accident sequence probabilities assigned to that category.

Fission Product Release Category Identification

The release categories of WASH-1400 will be examined and modified as necessary to produce a set of release categories appropriate for the ice condenser containment and associated containment failure modes. Specific

values for release of radioactivity to the environment will be identified for each release category.

Uncertainty Analysis

Quantitative results will be reported in terms of point values of a probability distribution function, including expected (mean) or median (50th percentile) value and upper (90th percentile) and lower (10th percentile) uncertainty bounds. These point values will be determined based on a propagation of component failure data, including error ranges, through the fault trees and event trees. The uncertainty propagation will be performed using standard statistical distribution functions (e.g. log-normal) or numerical (e.g. Monte Carlo) techniques.

Sensitivity Analysis

The results of the study will be reviewed to determine the relative importance to risk of the various accident sequences and to identify those which are the dominant contributors. Within those sequences, the significant system and component failure modes will be determined. Comparisons with existing risk studies, including WASH 1400, will be made to identify and explain any significant differences.

The sensitivity of the results to assumptions regarding component or common cause failures will be evaluated by varying the assumed failure rates and determining the resultant effect on system failure rates and overall results.

Schedule

OPS will perform the risk/reliability analysis on a time scale such that results from the evaluation can be factored into the design, specification and fabrication of the core and containment cooling system and these support systems. The plant risk/reliability program outlined above will be completed within two years after receipt of the Manufacturing License. Since significant additional design work on the Floating Nuclear Plant is not anticipated during the next few years, results of the analysis can readily be factored into the plant design, component and system specifications and fabrication for the systems cited above.

Application of Results to Final Design

There are currently no established regulatory requirements or acceptance criteria for judging the acceptability of quantitative system reliability analyses. Thus the need for implementing changes in design or in operating, testing, or maintenance procedures to achieve improvement in system reliability will be based on judgmental criteria which are not directly related to licensing requirements. These acceptance criteria will be established during the program and will include both quantitative and qualitative assessments of potential design changes taking into account impact on plant cost, schedule and availability.

Following completion of the base line reliability analysis, the results will be reviewed and various options available for improvement in reliability will be evaluated with respect to the established acceptance criteria. Recommendations will be made regarding changes in the design or in recommended plant procedures and the reliability analysis will be revised to reflect those recommendations selected for implementation.

Routine design changes will also be evaluated on an ongoing basis. A determination will be made regarding the effect of any proposed design change on the reliability analysis results. If the change is expected to affect reliability, the reliability analysis will be revised and the results reviewed for acceptability and need for further modifications as described above. In this manner, the Risk/Reliability Program will be kept current with respect to design modifications and a mechanism will be in place to evaluate reliability-related changes for acceptability as the design is finalized.

Results of the study will be utilized in component selection, specification and testing by assisting in identifying those areas where additional QA may be needed. The results of the reliability program, including all calculations will be subject to review anad verification in accordance with normal QA program procedures. Furthermore, the results of the study will be used to identify improvements in the maintenance, procedures, operator training, operating feedback and for reducing system interaction effects.

TABLE C-1

Outline of Reliability Analysis Report

I. INTRODUCTION

II. SUMMARY

III. METHODOLOGY OVERVIEW

A. Event Trees

B. Fault Trees

C. Quantification of Accident Sequences

D. Containment Failure Analyses

E. Fission Product Release Analyses

F. Treatment of Uncertainties

IV. SYSTEM DESCRIPTIONS

A. Performance Requirements

B. Actuation

C. Environment Considerations

D. Dependency Diagrams for Support Systems

V. CORE MELT PROBABILITIES

A. Dominant Sequences

B. Dominant Cut-Sets

VI. PLANT MODIFICATIONS THAT ADDRESS DOMINANT SEQUENCES

A. Improvement in Reliability Expected

B. How Factored into Design, Equipment purchase, Fabrication, Procedures, Operation, etc.

C. Basis for Not Implementing More Reliable Alternatives

TABLE C-1 (cont'd)

VII. FISSION PRODUCT RELEASE ANALYSIS

A. Release Groups

B. Containment Failure Probabilities

C. Fission Product Release Fractions

D. Total Radioactive Release from Containment to Environment for the Various Release Groups

VIII. APPENDICES (DETAILS OF STUDY)

TABLE C-2

Systems On Which Reliability Evaluations Are To Be Performed

1. Auxiliary Feedwater
2. Essential Raw Water
3. Residual Heat Removal
4. Diesel Generators (including support systems)
5. Upper Head Injection
6. Essential Service Water
7. Containment Spray
8. Component Cooling Water
9. Safety Injection
10. Safeguards Compartment Ventilation
11. Control Building HVAC
12. Ice Condenser
13. Electric Power (1E Power supplies, buses & breakers)
14. Protection System
15. Radiation Monitoring (1E channels)
16. Annulus Air Filtration
17. Emergency Instrument Air
19. Containment Isolation

TABLE C-3

Initiating Events for Risk Study

1. LOCA
2. Transients
3. Steam/feedwater line breaks
4. Steam generator tube rupture
5. Failures during cold shutdown operation
6. Fire
7. Earthquakes*
8. Explosions and missiles, internal and external*
9. Floods, tsunamis*
10. Tornadoes, hurricanes*
11. Electrical failure

*Using best available methodology where practical should be consistent with IEEE/ANS effort if feasible.

TABLE C-4

FNP Risks/Reliability Program Elements and Program Milestones

```
                                         Schedule (Months After M.L.)
Program Element                    0         6         12        18       21

Initiating Event Selection         |----|

Event Tree Development               |-----|

Systems Failures Modes &             |-----------|
  Effects Analyses (FMEA)

System Fault Tree Analysis               |-------------|

Data Base Development              |---------|

Containment Response Analysis            |-------------|

Accident Sequence Probabilities                     |--------|

Fission Product Release                             |-------|
  Category Identification

Uncertainty Analysis                                    |--------|

Sensitivity Analysis                                    |-----------|

Overall Results Assembly                                    |----------|

Report Preparation                                                     |---|
```

REGULATION 10CFR50.34(e)(1)(ii)

Subject: Auxiliary Feedwater Evaluation

To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility:

Perform an evaluation of the proposed auxiliary feedwater systems (AFWS), to include (applicable to PWR's only): (II.E.1.1)

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.
(B) A design review of AFWS.
(C) An evaluation of AFWS flow design bases and criteria.

OFFSHORE POWER SYSTEMS RESPONSE

Prior to the TMI accident Offshore Power Systems had performed a preliminary reliability analysis of the Floating Nuclear Plant's Auxiliary Feedwater (AFW) System. This analysis utilized the same component failure rate data base as the Staff's generic AFW System evaluation contained in NUREG-0611 and NUREG-0635, and investigated the following three accident scenarios:

a) Loss of main feedwater with offsite power available.
b) Loss of main feedwater combined with loss of offsite power.
c) Loss of main feedwater combined with total loss of AC power.

For the first two scenarios above, the unreliability of the Floating Nuclear Plant AFW System was found to be in the range of 10^{-5} to 10^{-4} failures per demand, and for the total loss of AC case in the range of 10^{-2} failures per demand. The difference in reliability between the first two cases and the last one, is due to the fact that during total loss of AC power, only the steam driven train is available, and thus no credit can be taken for the redundancies available in the diesel driven trains.

Overall, our review concluded that the Floating Nuclear Plant AFW System has above average reliability, as compared to the AFW systems already

examined by the Staff. Nevertheless, Offshore Power Systems will re-evaluate the reliability of its AFW System using event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss of main feedwater transient conditions, with particular emphasis being given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages. The results of this evaluation will be submitted in appropriate detail within two years of the issuance of a Manufacturing License. (See the response to 10CFR50.34(e)(1)(i).

The Floating Nuclear Plants Auxiliary Feedwater System is designed in accordance with the requirements of Standard Review Plan Section 10.4.9. However, a deterministic review of the system in accordance with this plan will be carried out and submitted to the Staff within two years of the issue of a Manufacturing License.

The AFW system flow design bases and design criteria have been carefully derived during the design evolution by consideration of the following safety-related functions of the system in the Floating Nuclear Plant.

a) The AFW system provides feedwater to the steam generators to remove residual heat from the core and prevent release of reactor coolant through the pressurizer safety valves in the following situations:

   o Loss of offsite power
   o Loss of normal feedwater
   o Malfunction of the Condensate Feedwater System
   o Major secondary system pipe rupture
   o Steam generator tube rupture
   o Control room evacuation
   o Sinking emergency

b) The AFW supplements the ECCS flow in removing core residual heat in the event of a small break LOCA.

c) The AFW System is utilized in cooling the reactor coolant down to the cut-in point of the Residual Heat Removal System for the sequences listed in a) above.

d) The AFW System maintains the plant at hot shutdown conditions during control room evacuation and extended loss-of-offsite power.

As part of the final design process, Offshore Power Systems will re-evaluate the above requirements, verify the corresponding AFW system functions, and submit detailed results to the Staff. This will be done within two years of issue of the Manufacturing License.

As noted below, the present design of the FNP Auxiliary Feedwater System generally satisfies the recommendations contained in the staff position paper entitled, "NTCP Acceptance Criteria, Task II.E.1.1, Auxiliary Feedwater (AFW) System Evaluation." The one possible exception is the suction piping arrangement, which will be examined as a part of the probabilistic risk assessment required by 10CFR50.34(e)(1)(i). (See item b, below).

a) As discussed in Section 10.4.6.7.4 of the Plant Design Report, the four motor driven auxiliary feedwater pumps automatically start on lo-lo level in any steam generator, loss of main feed pump, safety injection signal, or loss of offsite AC power. The turbine driven pump starts automatically on lo-lo level in any two steam generators or loss of offsite power. Automatic initiation signals and circuits for the Auxiliary Feedwater System are Class 1E and can be tested on-line. Manual capability to actuate the Auxiliary Feedwater System is provided in such a manner that no single failure will result in loss of the system function. No single failure of the automatic acutation circuitry will prevent manual actuation of the Auxiliary Feedwater System from the Control Room.

b) The arrangement of AFW suction piping (including isolation valves) will be examined as a part of the probabilistic risk assessment required by 10CFR50.34(e)(1)(i). Changes in suction line design which are found to

contribute significantly to system availability will be made during FNP final design. Offshore Power Systems expects that staff recommendation (4)(b) will be implemented on this basis. However, Offshore Power Systems prefers to await the study results before proposing detailed changes to the AFW suction piping arrangements.

c) Operation of the turbine-driven AFW subsystem is initiated, monitored and controlled by instrumentation which will continue to operate for a period in excess of two hours following coincident loss of both the offsite and onsite AC power supplies.

d) The AFW system is housed in areas which are protected from tornado missile damage in accordance with Regulatory Guide 1.117. Further, the AFW system is designed to seismic Category 1 requirements in accordance with Regulatory Guide 1.29. Therefore, additional pump suction protection from tornado or earthquake damage is not required.

e) Redundant level indications and low-level alarm functions will be provided for the AFW storage tanks. The low level alarm point will be set so as to provide a minimum of 20 minutes warning to loss of suction, assuming operation of the turbine-driven pump (the pump with the highest flow rate). AFW storage tank level and alarm will be provided in the control room.

REGULATION 10CFR50.34(e)(1)(iii)

Subject: RCP Seal Damage

To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility:

Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

OFFSHORE POWER SYSTEMS RESPONSE

For the FNP, reactor coolant pump seal injection is provided by the CVC charging pumps. In the event of a loss-of-offsite power (LOOP), a "B" or blackout signal is generated. The "B" signal automatically starts the charging pumps on redundant Class 1-E buses to provide RCP seal injection flow at 10 seconds after the LOOP. In addition, the CCW pumps are loaded on at 20 seconds after the LOOP to provide thermal barrier cooling. Therefore, for LOOP alone, the reactor coolant pump seals are adequately provided with both seal injection and thermal barrier cooling, either of which will preclude seal damage and a subsequent increase in seal leakage.

In the event of a small-break LOCA (with pre-existing LOOP) operating loads, including the CCW pumps, are stripped from the emergency buses. In addition, the charging pumps are tripped (without restart) and containment isolation valves in the seal injection lines are shut. If containment pressure increases to the Hi-Hi setpoint, the "P" signal is generated and containment isolation valves are shut in the thermal barrier cooling water lines. Thus, in the present design, operator action is required to protect the RCP seals following a small-break LOCA. As a result, Offshore Power Systems will complete one of the following courses of action within two years after issuance of the manufacturing license.

1. Modify the plant design to provide automatic restoration of either seal injection or thermal barrier cooling flow, or

2. Demonstrate that adequate time is available without seal failure for the operator to restore either seal injection or thermal barrier cooling flow, or

3. Perform an evaluation of the potential for and impact of RCP seal damage following a small break LOCA with loss of offsite power.

REGULATION 10CFR50.34(e)(1)(iv)

Subject: LOCA from PORV Failure

To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility:

Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)

OFFSHORE POWER SYSTEMS RESPONSE

An analysis has been performed by Westinghouse and reported in WCAP-9804 dated February 1981, which estimates the probability associated with a small break loss-of-coolant accident (LOCA) caused by a stuck open power operated relief valve (PORV). The WCAP concludes that a significant reduction in the frequency of a small break LOCA, due to a stuck open PORV, has been achieved by plant modifications made subsequent to TMI. Specifically, the probability is estimated to be 2×10^{-6} per reactor year for Westinghouse plants. Since the probability of a small-break LOCA from all causes is approximately 1×10^{-3} per year, PORV failure is not a significant contributor. Based on historical plant data, no failures of a PORV to close have occurred in domestic Westinghouse plants. The FNP incorporates PORVs supplied by Westinghouse; the piping layout and valve setpoints are those recommended by Westinghouse. Therefore, the results reported in WCAP-9804 are applicable to the FNP.

In addition, the consequences of a small break LOCA caused by a transient-related PORV opening and failure to reclose has been analyzed in WCAP-9601 dated June 1979. Even with conservative licensing basis assumptions, no core damage is expected for this compound transient as reported in WCAP-9601.

REGULATION 10CFR50.34(e)(2)(ii)

Subject: Plant Procedure Improvement

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (I.C.9)

OFFSHORE POWER SYSTEMS RESPONSE

Each plant owner will be responsible to the NRC for the preparation and updating of plant operating procedures. Offshore Power Systems will assist plant owners in discharging this responsibility by serving as a clearinghouse for important information derived from Floating Nuclear Plant in-service experience, design developments and experience gained during testing.

REGULATION 10CFR50.34(e)(2)(iii)

Subject: Control Room Design

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems will provide a control room design that reflects state-of-the-art human factor principles by the application of design criteria which will assure the ability of the control room operators to prevent anticipated transients from developing into accidents and to cope with accidents should they occur. The control room will include an advanced design, computer based CRT display room. The control room design will be provided to the NRC prior to committing to fabrication of the control room panels.

The FNP Control Room design bases, criteria and general functional requirements are presently adequate for meeting all of the concerns identified after the TMI accident. A summary of the relevant design bases, criteria, functional specifications/descriptions for the FNP Control Room is presented in the attached excerpts from the Control Room Specification. Further description of safety system status monitoring (including description of the light and alarm sequences for control board modules) is presented in the response to 10CFR50.34(e)(2)(v). The design bases, criteria, and functional specifications for the FNP Control Room meet or exceed Draft 1C of IEEE-566 and Draft 3 of IEEE-567. OPS does not expect problems in implementing forseeable future requirements, such as the human factor requirements to be published as NUREG 0700, because:

1. OPS practice for control room design evolution includes a formal, ongoing multi-discipline review process (including the use of a full-scale mockup). Any significant changes will be implemented with

that process. Final design review will use the methods being developed by Sandia and NRC.

2. The design concept is flexible (modular construction and modular display software will be used).

3. The status of detailed design is such that schedule constraints are not expected.

EXCERPTS FROM THE FNP CONTROL ROOM SYSTEM SPECIFICATIONS

1.0 Control Room Design Bases and Criteria

1.1 Equipment Design Basis

1.1.1 Classification:

The control room panels, their foundations and supporting structures are classified as follows:

Panels that have Class 1E equipment or circuits, are classified as Class 1E, Seismic Category I.

Panels that house no Class 1E equipment or circuits are classified as Class N1E Seismic Category II.

The classification, 1E or N1E, of the individual display and control devices and their associated wiring are as designated in their respective schematic connection diagrams. The FNP Instrument List will provide a composite listing of all the devices mounted on the panels and include, among other information, the classification.

1.1.2 Environment:

The main control panels and all equipment mounted therein shall be designed to operate under the following atmospheric conditions:

Temperature	$5^{o}C$ to $50^{o}C$ ($40-122^{o}F$)
Pressure	1 Atm
Humidity	10 - 90% RH

NOTE: The Control Building Air Conditioning System will maintain the control room at $75\pm 2^{o}F$ and a relative humidity of 40 - 70% under all postulated plant conditions.

All equipment associated with the main control room panels are located within the control module and will receive less than 2 RAD over the design life of the FNP.

1.1.3 Criteria:

The basic criteria for the design, fabrication and testing of the main control panels are derived from the application of IEEE 279 and IEEE 384 to the Class 1E equipment and circuits contained therein.

The main control panels must provide sufficient support and physical protection to its Class 1E equipment and circuits to enable them to perform their essential functional requirements before, during, and after motion conditions, up to and including design basis motion conditions.

The panels shall be so designed that, at the frequencies and accelerations of the floor resulting from design basis motions, they do not amplify the forces beyond the level at which the equipment contained therein is qualified to function properly. To meet this requirement, the panel shall be designed with sufficient rigidity so that no natural frequencies or resonances can exist at a frequency less than (later) Hz. Welded stiffeners, diagonal braces and thick plate skin shall be used singly or in any combination that will satisfy this rigidity requirement. The panel design shall be seismically qualified in accordance with the requirements of IEEE 344.

The panel design shall include provisions for securely mounting the board to its supports. A dynamically equivalent support shall be used in the seismic testing of the panel.

Plug-in or slide mounted equipment shall be provided with mechanical constraints, if needed to maintain positional integrity.

All equipment within, attached to, or adjacent to, the panel shall be mounted such that the structural failure of this equipment cannot damage Class 1E equipment or circuits.

The internal structural design of the main control panel shall provide for the physical separation of redundant Class 1E circuits and equipment, as required by IEEE 384 so that no single credible event can prevent the proper functioning of any Class 1E system. The required separation shall be achieved by an adequate air space or a fire-retardant barrier between redundant Class 1E circuits and equipment. The circuit wiring shall be supported in a manner that will assure maintenance of the air space throughout the design life of the panel. The required separation shall be maintained from the point of entry of the circuit into the panel to the final termination on the surface mounted devices. Non-Class 1E circuits and equipment shall likewise be separated from all Class 1E equipment and circuits.

Inherent flame-retardant characteristics and properties shall be a major consideration in the selection of materials for use in the main control room panels. The structural framework and surfaces of the panel shall be fabricated of steel or aluminum stock. Any nonmetallic

components and devices should be manufactured from self-extinguishing material as defined by ASTM Std. D635-1972.

Paints or other applied surface preparations should contribute only nominally to the total combustible potential of materials or components in or on the panel. Consideration should be given to the release of toxic or corrosive gases and dense smoke and their effects upon personnel and equipment.

1.1.4 Electrical Design:

In accordance with IEEE-279, components and modules shall be of a quality that is consistent with minimum maintenance requirements and low failure rates*. Quality levels shall be achieved through the specification of requirements known to promote high quality.

All control and instrument wiring shall have sufficient mechanical strength, current capacity, thermal rating and insulation characteristics to meet the circuit and installation requirements established by plant design.

Wire and cable insulation shall be flame retardant with self-extinguishing nonpropagating characteristics. Consideration shall be given to the potential release of toxic or corrosive gases and dense smoke and their possible effects upon personnel and equipment. All wire and cable installed within the main control panels must be capable of meeting the flame test requirements of IEEE 383.

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*Mean Time Between Failures: 20,000 hr.
Mean Time To Repair: 30 min.

1.2 Operator Interface Criteria

1.2.1 General:

The primary criterion for the FNP operator interface is that it shall provide to the operator the information and control facilities that he needs to safely and efficiently operate the plant under normal and upset conditions and present them in such a manner as to enhance his understanding of the plant status and reduce the probability of an operational error. Systems analysis as outlined below will be the basis for the design of the control room to meet the intent of the elements of Appendix B of NUREG 0659.

1.2.2 Information and Control Requirements:

The determination of which information and controls are to be provided in the control room begins with the responsible OPS process system engineer, in close cooperation with the corresponding control system engineer. This process is formalized by evolution of the following controlled documents:

1. Process System Specifications
2. Instrument Block Diagrams
3. Control Logic Diagrams
4. Schematic Connection Diagrams

Throughout this design process, each system is analyzed from an operating point of view using sequence analysis techniques. The control systems engineering group is responsible for the systems integration as well as for the application of control and display hardware.

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In the process system specifications, the information and control requirements are defined in functional terms. These functional requirements are further defined in the Instrument Block Diagrams and Control Logic Diagrams and then converted to hardware requirements (i.e. indicators, lights, switches, etc.) in Schematic Connection Diagrams. The I&C hardware requirements for all the FNP systems are consolidated into a single document -- the FNP Instrument List. This document is a computerized list of all the I&C hardware provided on the FNP and includes sufficent information to completely describe each item. Included in the bank of information is the mounting location for each item.

1.2.3 Arrangement Requirements:

The configuration and arrangement of the control center panels and the placement of indication and control devices on the panels shall be based on the following:

The FNP control center shall be designed to enable a single operator to safely control the plant under all operating conditions. Provisions shall also be made for accommodating additional operating personnel during periods of high activity when it is desirable to relieve the burden of the lead operator.

In multi-operator situations, the following organizations will be assumed:

The lead operator will be singularly responsible for the safe conduct of control center operations. The second operator will be assigned a subordinate role and shall take action only at the direction of the lead operator or in accordance with written procedures authorizing specific independent action. The subordinate operator will make

reports to the lead operator (1) prior to the initiation of independent action (2) when difficulty is encountered in the performance of an assigned or independent action (3) periodically on the status of long term assignments and (4) whenever an abnormal situation is noticed. The lead operator will likewise be responsible for keeping the assistant operator(s) informed of the plant status.

Functional Areas -

The control center shall be subdivided into distinct operating areas. The functional requirements defined for each area determine the major criteria for the allocation of display and control devices within the control center. These operating areas are designed to provide for a separation of safety-related system and auxiliary and supporting devices from those required by the operators to monitor and control the plant under normal conditions. This method of device allocation affords a reduction in number of displays that the operator must observe under normal conditions and consequently, a reduction in the probability of misinterpretation and erroneous action.

The operating areas to be incorporated in the FNP control center design are as follows:

Normal Operations Area

The normal operations area is the primary control location for the Floating Nuclear Plant under hot non-upset conditions. The display and control devices located in this area will be the minimum required for the operator to perform the following:

1. to assess the status of the plant and its systems at any time

2. to be alerted to abnormal situations and changes in plant status

3. to maintain the plant in a safe hot shutdown condition

4. to maneuver the plant from a hot shutdown condition to full power operation

5. to manually initiate safety systems (on a system level)

Safety System Operating Area

This area provides the facilities required by the operation to:

1. quickly assess that safety systems are performing their required safety functions

2. monitor long term course of the accident

3. determine when conditions exist that require specific manual actions, to take such action and monitor the results

4. perform safety system functional testing

5. determine the availability status of protection and safety systems at any time during panel operation

Infrequent Operations Area

This area is allocated to display and control devices needed to perform auxiliary or supporting functions that are required infrequently. (An example would be the

devices that are used only during heatup, cooldown, cold shutdown and refueling.)

Historical Records Area

This area is allocated to the devices which are required to provide hard copies of computer stored data.

Area Arrangement

The arrangement of the operating areas within the control center will be consistent with the following criteria:

1. The normal operations area will be centrally located and provide the operator with surveillance and access capability to other operating areas.

2. The safety systems operating and the infrequent operations area shall be directly accessible and visible from the normal operations area and not be in a separate enclosure.

3. The historical records area shall be located apart from the operating areas.

4. The supervisor's office shall be located as to give him a visual command of al. control center activities.

1.2.4 Human Engineering Requirements:

A human factor review of the control room design will be performed using industry and NRC-developed guidelines, including NUREG-0700. The scope of this review will include the control room design, including the control room arrangement and environment, and the main control panel layouts. The design will be evaluated for conformance with

the design criteria, and any resulting modifications will also be reviewed. This human factors review will be performed by individuals experienced in operations, systems analysis, human factors engineering, architectural engineering and control room design.

The following human engineering considerations shall be factored into the design of the FNP Control Center.

Anthropometric Considerations -

The control panels will be designed to permit 5 to 95 percentile (in height) operators to read or reach all indicators and controls from a standing position in front of the auxiliary panels and from a seated or standing position in front of the consoles. The 5th percentile operator is 5'4" tall and the 95th percentile operator is 6'4" tall.

Task Analysis -

The assignment of controls and displays to the functional areas and their placement within the areas will be based on the operators need for the devices in the performance of his assigned tasks.

Each control and display device will be analyzed to determine:

1. the operating modes during which the operator needs the device,

2. how often the operator uses the device when it is needed,

3. in the case of controls, how fast does the plant respond to a control manipulation, and

4. in the event of a malfunction, how fast must the operator take corrective action.

Other Considerations -

During the last few years a number of human engineering reviews of existing nuclear power plant control panels have been conducted by the Electric Power Research Institute, W Research Laboratories and others. The results of these studies shall be used to develop a checklist for the FNP design to ensure that the typical deficiencies noted in Table C-5 are avoided.

1.2.5 Panel Mock-Up:

A full-scale control center mock-up will be constructed and used to evaluate the human engineering aspects of the FNP design. The evaluation will include "walk-throughs" of the FNP operating procedures to ensure that no operational problems are overlooked.

2.0 FUNCTIONAL DESCRIPTION

Figure C-1 shows the preliminary FNP control room layout and the locations of the various functional areas. The correlation of equipment with these areas is as follows:

(1) NORMAL OPERATIONS AREA -- - UNIT CONTROL CONSOLE
(2) SAFETY SYSTEMS OPERATING AREA - SAFETY CENTER
(3) INFREQUENT OPERATION AREA - SECONDARY CONTROL CENTER
(4) HISTORICAL RECORDS AREA - COMPUTER OPERATORS CONSOLE

2.1 Unit Control Console:

The Unit Control Console (UCC), shown in Figure C-2, is a compact modularized console from which all normal plant operation is conducted. It includes all the displays and controls necessary to bring the unit from hot shutdown to rated power (and back to hot shutdown) and for controlling and monitoring load changing operations. The UCC design permits one-man operation while providing space for two.

The UCC provides three computer generated visual displays (CRT's). These displays, together with their associated keyboards, provide the operator with all the information he needs to assess the status of the plant and its systems at any time. The center CRT contains a process overview in which all key parameters are continuously updated in a single display. The left hand CRT contains an alarm display showing the status of all points in alarm. The right hand CRT is used to display parameters for any selected system in detail.

The remainder of the UCC is arranged in stations that are dedicated to those portions of the following systems used during normal operation:

1. Rod Control System and Rod Position System
2. Nuclear Instrumentation System

3. Reactor Coolant System
4. Chemical and Volume Control System
5. Feedwater and Condensate Systems and Auxiliary Feedwater
6. Main Steam System
7. Turbine Generator System
8. Generator Circuit Breakers and Synchronizing

The UCC will also include safety system manual actuation controls and any permissives and blocks required for normal operation.

2.2 Auxiliary Panels

2.2.1 Safety Center:

The Safety Center, shown in Figure C-3, provides for the monitoring, control and testing of the FNP protection and engineered safety features systems. The panel is arranged in stations that present a logical flow of information to the operator. The left most station provides the displays and controls associated with the Reactor Protection System (SSPS) and those displays and recorders required for Post-Accident Monitoring. The stations located immediately to the right contain the component level displays and controls for the ESF systems. These include:

1. Upper Head Injection
2. Safety Injection
3. Containment Spray
4. Residual Heat Removal
5. Essential Service Water
6. Essential Raw Water

Also included in this area are the CRT and keyboard and system level binary status displays required by the Protection and Engineered Safety Feature Availability and Test System.

To the right of the ESF stations will be located the indicators and controls associated with 1E support systems including:

1. Air Conditioning Systems
2. Hydrogen Recombiner Systems
3. Containment Isolation Valves (those not used during normal control)
4. Ice Condenser Systems

2.2.2 Secondary Control Center:

The Secondary Control Center, shown in Figure C-3, contains all of the required control and indicators that are not located on the UCC or the Safety Center. The arrangement of the systems generally follows the order in which they are used in bringing the unit to power operation. The controls and indicators located here are primarily those that are used only during refueling, heatup, cooldown and maintaining cold shutdown.

2.2.3 Component Arrangement:

The controls and indicators required for the operation of each individual system will be integrated into a common work station. The arrangement of components within the work station will follow the placement of the controlled components in the actual process system. For complicated systems or those used infrequently, a graphic display will be provided above the work station. A typical example of this arrangement is shown in Figure C-4.

2.2.4 Annunciators:

Annunciators will be located along the upper sloping portion of the safety center panels. These will be conventional hardwired alarm points and will be used as backups in the unlikely event that the computer generated alarm display is inoperative. This use of annunciators will be restricted to alarming only those fault conditions that could affect the ability to reach and maintain a safe shutdown condition or those required to meet regulatory requirements.

2.3 Historical Records Area

The Historical Records Area, shown in Figure C-1, is centered at the computer operators console. At this console an operator will be able to obtain a hard copy of CRT displays, computer calculations, test results, etc.

3.0 EQUIPMENT DESCRIPTION

3.1 Unit Control Console

The Unit Control Console (UCC) will be a free standing sit-stand console as defined by IEEE-27. Figure C-2 shows the general size and shape of the console.

The UCC will be provided with front and rear removable panels for access to internally mounted equipment and cable terminations Equipment will be mounted within the panel with a view towards maximum accessibility for testing and maintenance.

The console design will provide for the entry of cables through bottom access holes centered below vertical wire-ways housing terminal blocks and connectors. Five sets of vertical wire-ways will be consistent with the requirements of IEEE-384 for cable spread rooms.

Horizontal raceways will be provided for supporting cable along the length of the console. Five raceways will be provided, one for each set of vertical wire-ways. Cable access from the wire-ways to the raceways will be provided only between those of the same division. The spacing of the raceways will be consistent with the requirement of IEEE-384 for panel internals.

Cable runs from the raceways to the panel mounted equipment will be by the most direct route consistent with the following division separation requirements.

1. From the raceways to a distance of one foot from the panel surface, cables of different divisions will be maintained at least six inches apart.

2. Within one foot of the panel surface the division separation may be reduced to one and one-half inch when, due to equipment proximity, six inches cannot be maintained.

The reduced separation requirement at the panel surface is provided to allow for cases when, due to operational considerations, it is desirable to mount equipment belonging to different divisions at adjacent locations. The justification for the one and one-half inch separation will be provided through analysis and testing. The analysis will show that no single credible event, with the exception of an internally generated fire, could prevent the proper functioning of a 1E system. The testing will demonstrate that, with the low energy circuits used in the panel, an internally generated fire that could affect redundant divisions, is incredible.

3.2 Back Panels:

The back panels will be free-standing duplex benchboards as defined by IEEE-27. The panels will be provided with rear access doors and removable front panels.

Provisions for cable entry and routing are similar to those described in section 3.1.

3.3 Instrument Modules:

The majority of the discrete display and control functions required on the panels will be accomplished using a modular system of instrumentation. The modules are all of the same height and their widths are multiples of a fixed modular dimension. The basic modules include: an indicator module; and auto/manual module; a pushbutton module; and a recorder module.

A typical indication module, shown in Figure C-5, has two vertical displays. Each of the displays will provide one percent reading accuracy and will be scaled in engineering units.

The auto/manual module, shown in Figure C-6, contains four backlighted pushbuttons and an edgewise indication. The auto/manual module together with an indicator module will be used to perform auto-manual control functions. In this application, one of the displays will be used for the measured variable and the other for the set point.

The pushbutton control module, shown in Figure C-7, is the primary binary control and indicating means. The module can contain up to six backlighted pushbutton operators and each button can be split to display two messages.

A recorder module will be used whenever a hard copy record of a process variable is required and the plant computer cannot be used to provide it. The recorder module will be four module widths wide and will be available in 1, 2 or 3 pen configurations.

All of the modules will be removable from the front of the panel and further, can be removed with the circuit active without affecting the state of the controlled component or parameter.

TABLE C-5

TYPICAL HUMAN ENGINEERING DEFICIENCIES

Reading Indications

Recommended viewing distance exceeded

Meter design causes glare and improper viewing angle

Control design obscures position setting

Reaching Controls

Functional reach exceeded

Working posture leads to accidental activation

Activating Controls

Inconsistent direction of movement relationships between control and associated display

Violation of operator expectation of direction of movement of control

Nomenclature that violates operator expectation

Use of same nomenclature for different functions

Different nomenclature used for functionally identical controls

Interpreting Coding

Use of the same color for more than one function

Color - function associations that violate operator expectation

Functionally identical controls color-coded differently

Inconsistent use of illumination coding

Interpreting Alarms

No differentiation of the severity of alarms

Nuisance alarms

TABLE C-5 (CONT'D)

Locating Components

No delineation between major control systems

Side by side location of functionally unrelated controls that are identical in appearance

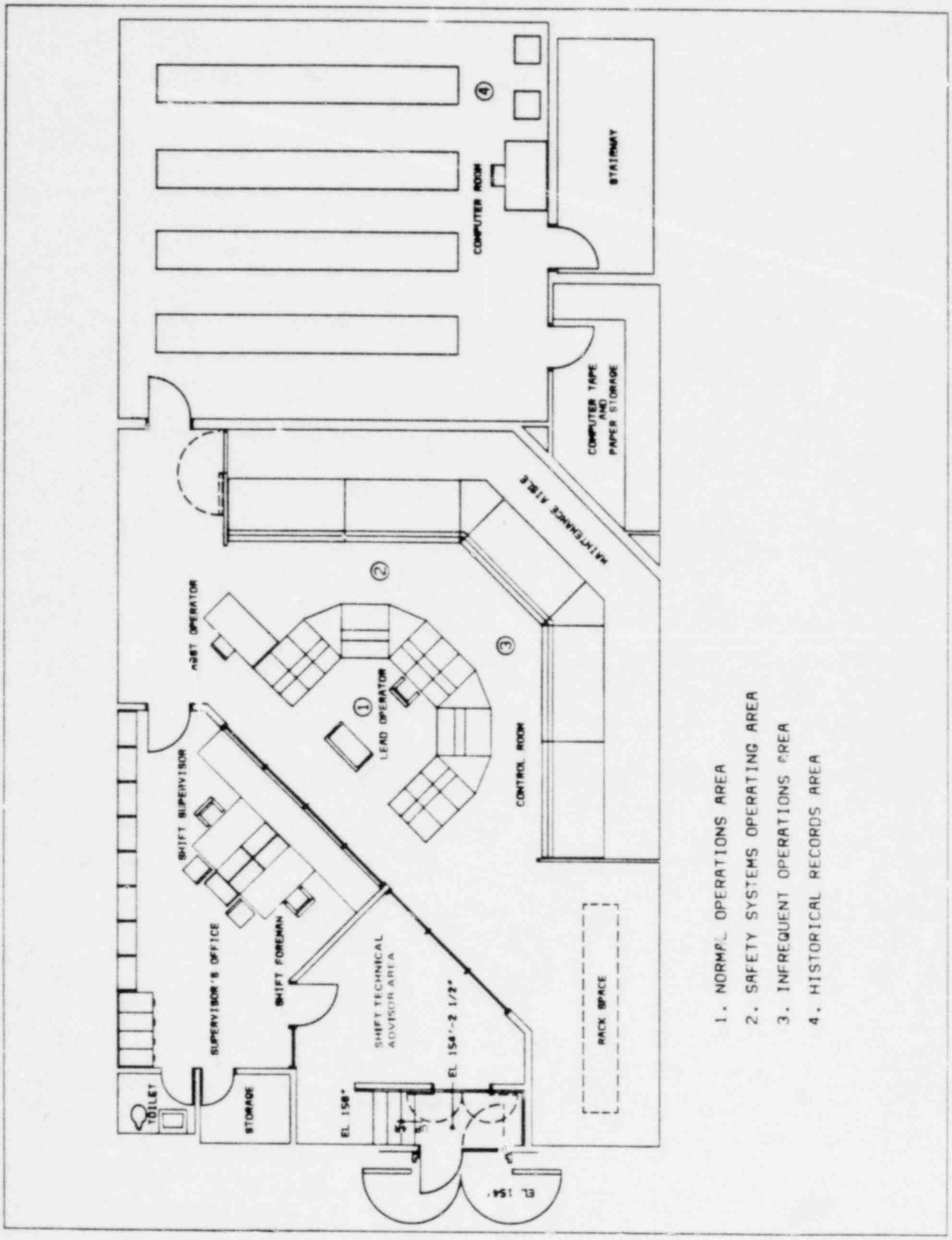
Incompatible arrangements of associated displays and controls

Illogical arrangement of related controls

Inconsistent location of the same type of control

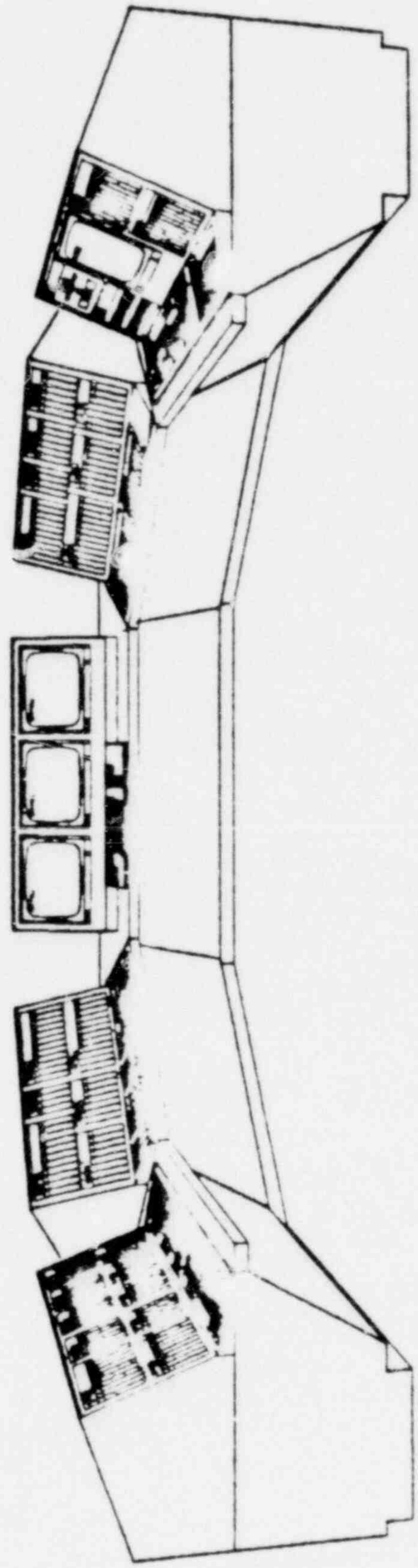
Performing Sequential or Simultaneous Operations

Spatial separation of controls that must be used together



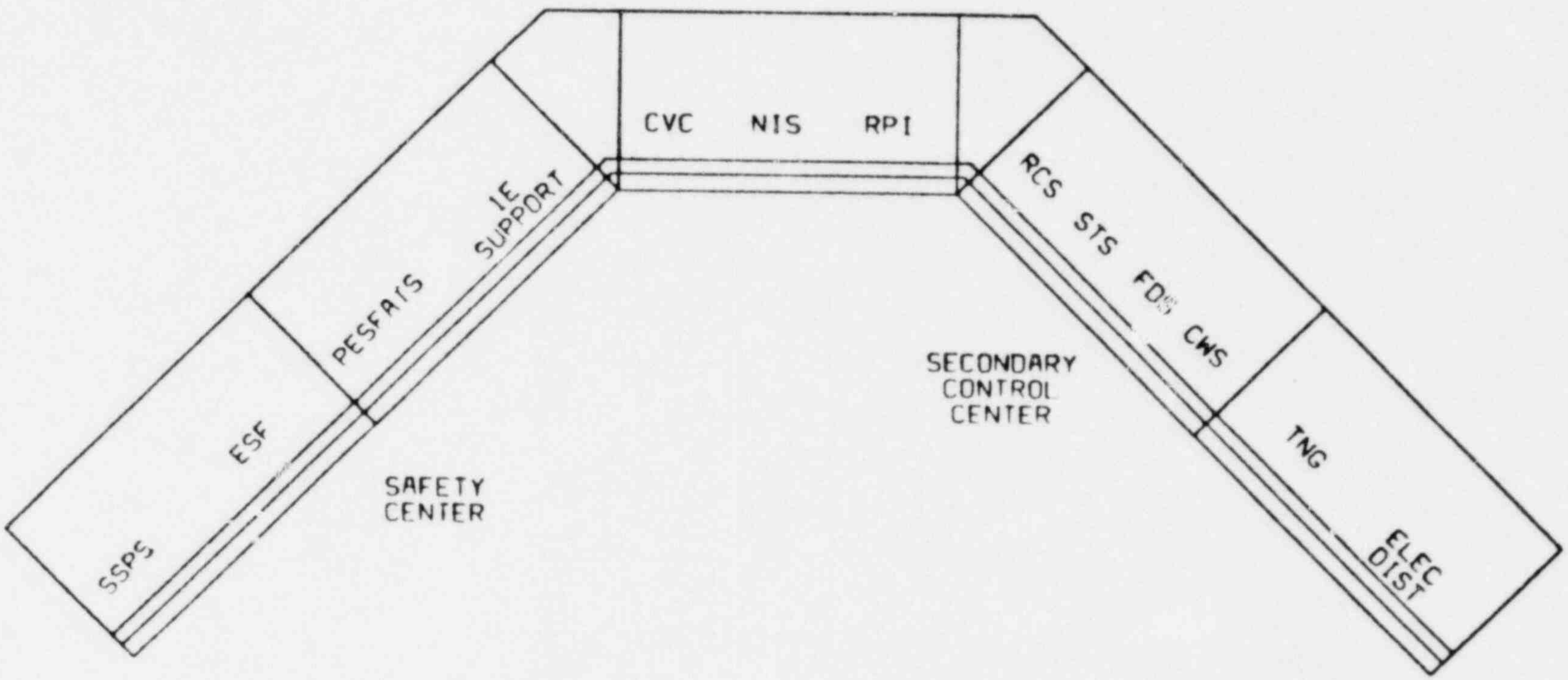
CONTROL ROOM ARRANGEMENT

FIGURE C-1



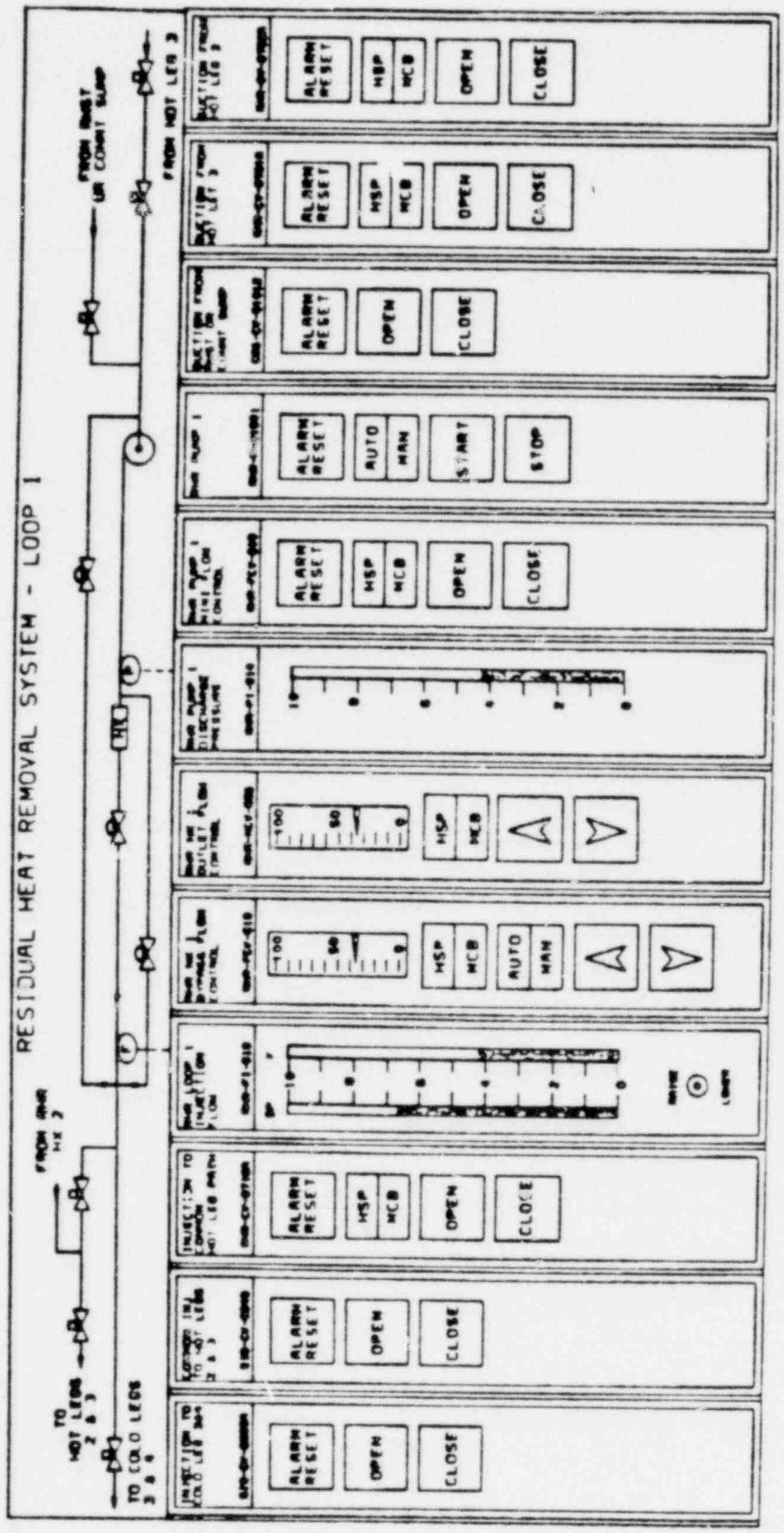
UNIT CONTROL CONSOLE

FIGURE C-2



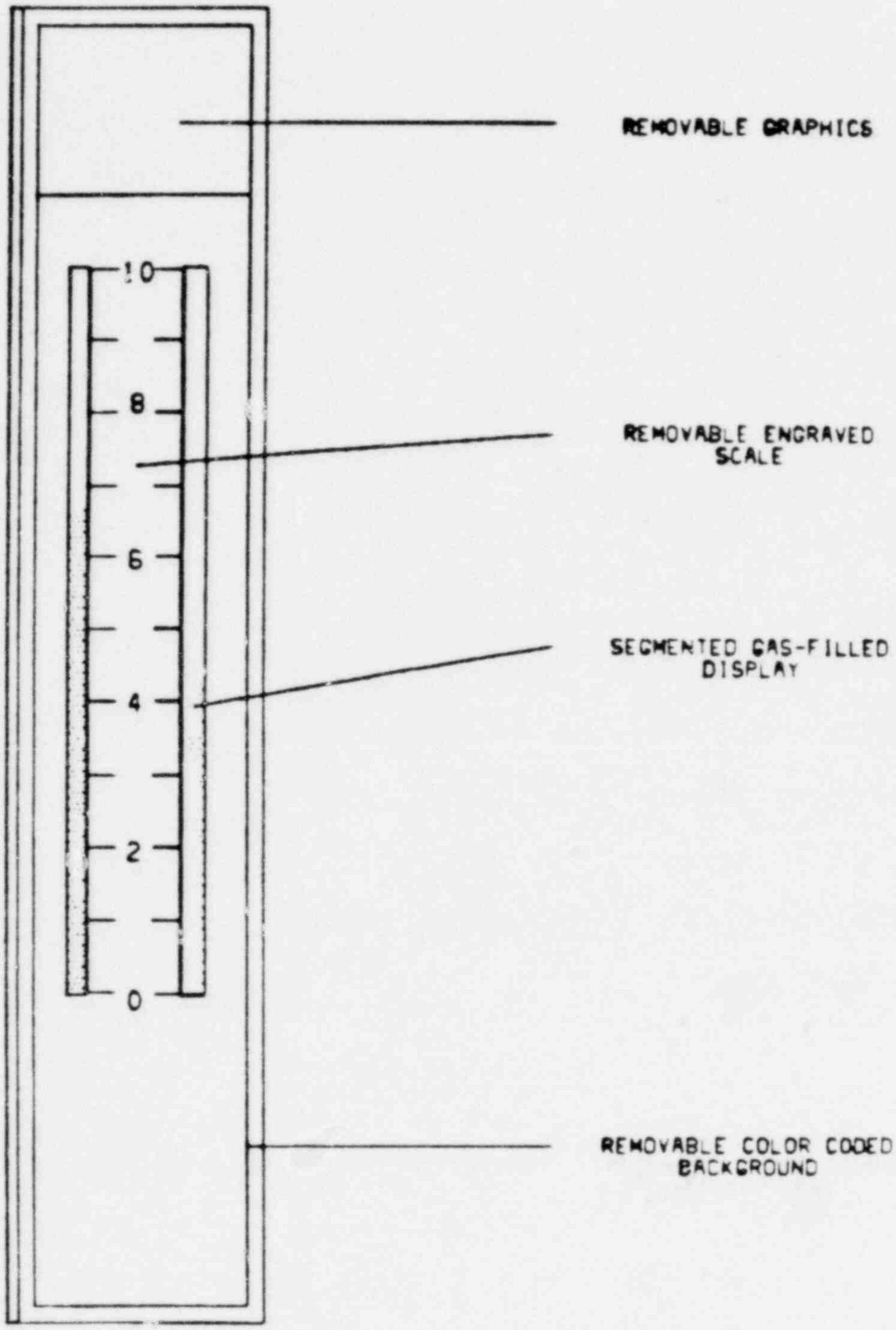
BACKPANEL PLAN

FIGURE C-3



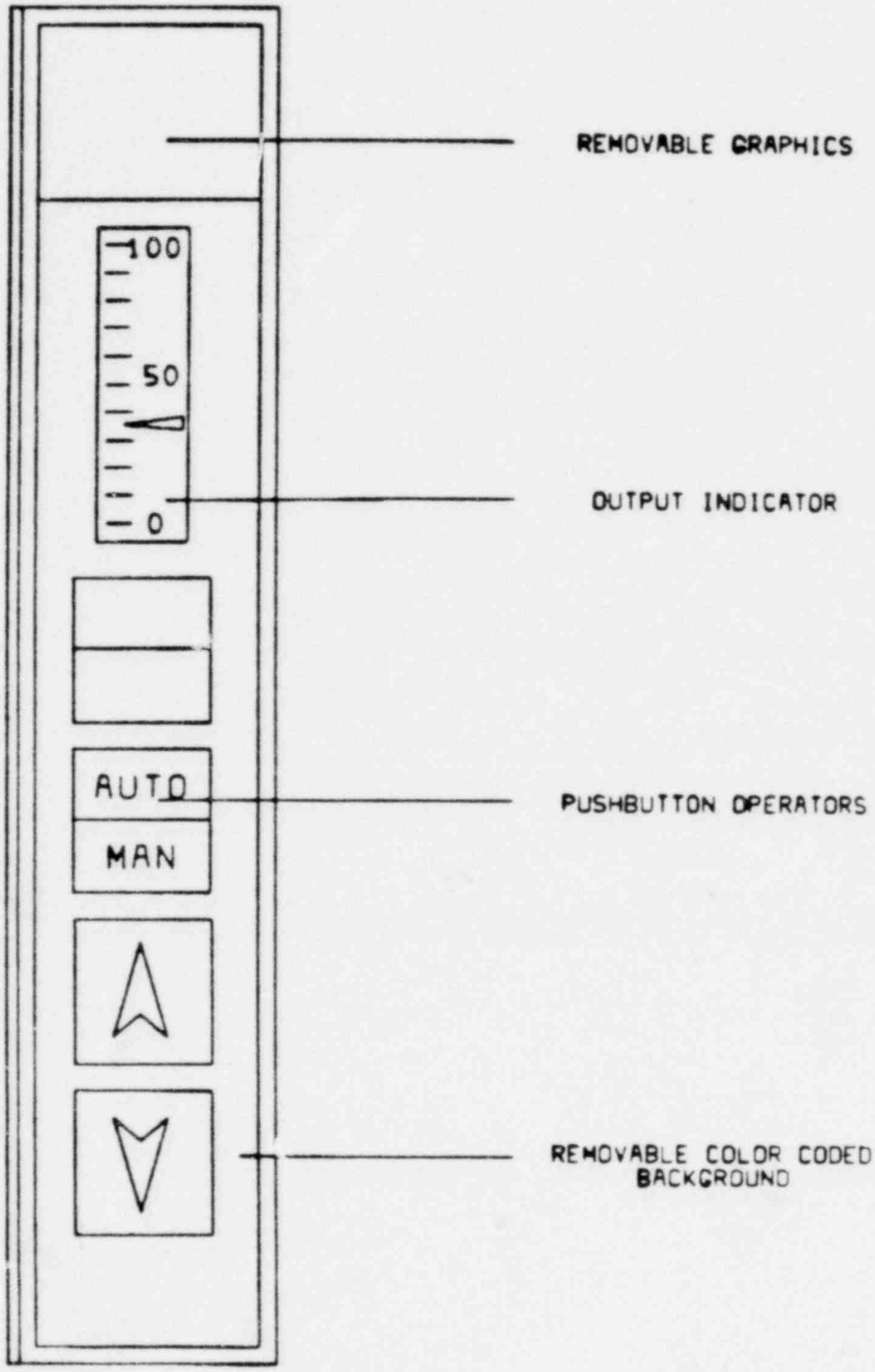
TYPICAL MODULE ARRANGEMENT

FIGURE C-4



TYPICAL INDICATION MODULE

FIGURE C-5



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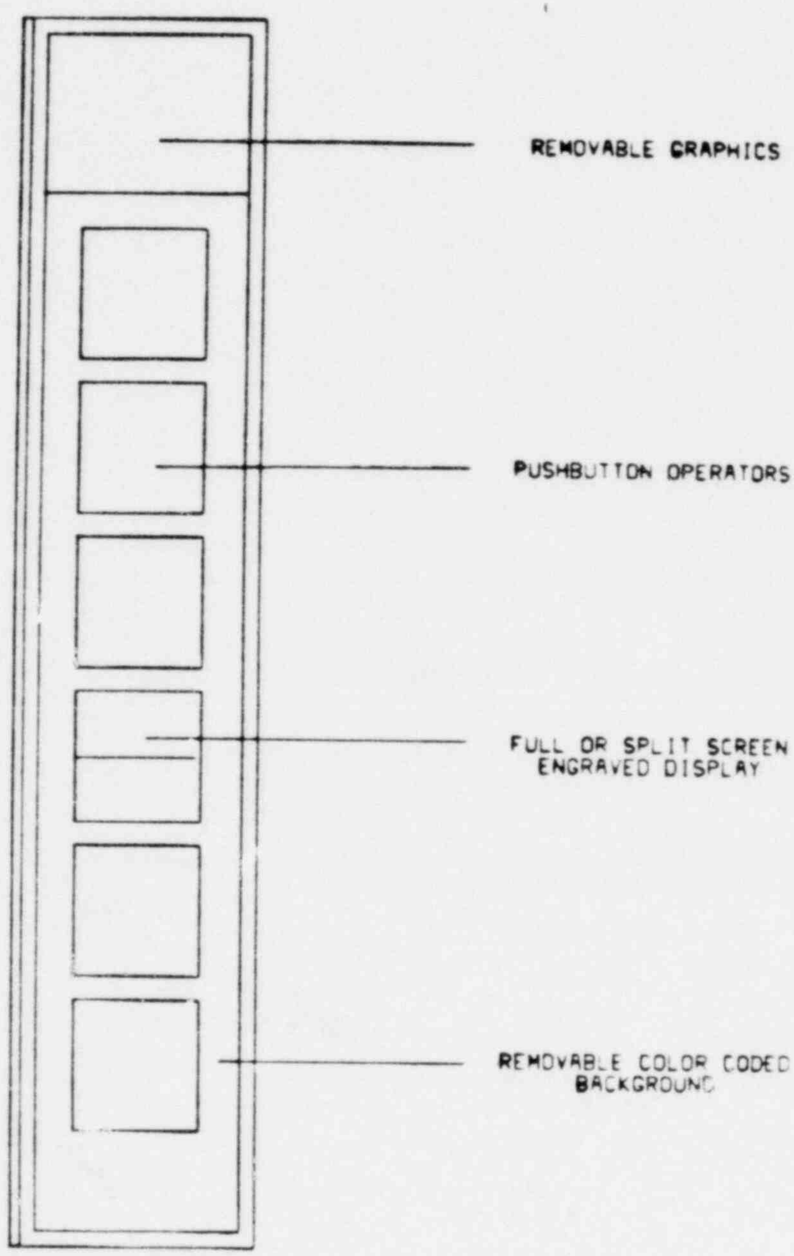
TYPICAL AUTO/MANUAL MODULE

FIGURE C-6



TYPICAL PUSHBUTTON CONTROL MODULE

FIGURE C-7

REGULATION 10CFR50.34(e)(2)(iv)

Subject: Safety Parameter Display

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and is capable of indicating when process limits are being approached or exceeded. (I.D.2)

OFFSHORE POWER SYSTEMS RESPONSE

The FNP will be equipped with a safety parameter display system (SPDS).

The system will have the capability of continuously displaying data of the safety status of the plant and of displaying the full range of important plant parameters and data trends on demand. The system will also indicate when plant parameters are approaching or exceeding process limits.

The SPDS will be designed to the following criteria:

1. Input parameters will be selected and the parameter data processed in such a way as to yield concise, reliable indication of the status of the following safety functions: core cooling, reactor coolant subcooling, reactivity control, control of primary coolant inventory, coolant temperature and pressure control, and containment of radioactivity.

2. The display (output) format will be designed in accordance with human engineering principles so as to provide visability and ease of information interpretation.

3. Diversity, redundancy, and error-checking will be utilized to assure reliable safety status indication.

4. Display devices will be prominantly located on the Safety Center and output information will also be available for display on a CRT in the Unit Control Console. (See Figures C-1, C-2 and C-3 for the basic Control Room and panel arrangements). Provisions for duplicate information display in the Emergency Operations Facility (Plant Owners Scope) and the Technical Support Center will be provided.

5. The design provisions for safety-related display instrumentation described in Section 7.5 of the PDR will be utilized so as to complement the SPDS.

6. The design of the SPDS will be in conformance with the guidance of NUREG-0696, February 1981.

REGULATION 10CFR50.34(e)(2)(v)

Subject: Bypassed and Inoperable Status Indication

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)

OFFSHORE POWER SYSTEMS RESPONSE

Since as early as 1974, Offshore Power Systems has been committed to total conformance with Regulatory Guide 1.47 (Plant Design Report Section 7.1.7). The design includes automatic indication of the bypassed and operable status of safety systems. To the extent practical, inputs to the Safety System Status Monitoring system will be direct measurements of the desired variables. The Floating Nuclear Plant presently includes significant provisions for status monitoring; these are outlined in the following paragraphs.

Assurance of proper operation and/or positioning of safety-related equipment (including equipment in engineered safety features supporting systems) during all operating activities is provided by:

Main Control Board (MCB) Display Features: including position/status indicating lights, position/status disagreement indication, availability indication, and system level bypass indication. These features which meet or exceed Regulatory Guide 1.47, are as follows:

Position/Status Indicating Lights (Backlit Pushbutton) (PIL)
Backlit red (open) and green (closed) pushbuttons indicate actual valve position from limit switches on the valve. The pushbutton is part of the MCB module for that valve.

Backlit red (on) and green (off) pushbuttons indicate breaker or contactor status from appropriate auxiliary contacts. The pushbutton is part of the MCB module for that component (pump, fan, etc.)

These position/status signals are also inputs (through isolation devices) to the Plant Computer Systems.

Valve Position Indicating Lights (Lights Only) (PIL*)
This valve position signal is also an input to the Plant Computer Systems.

Position/Status Disagreement Light/Alarm (Backlit Pushbutton) (PDL)
A backlit alarm indication/acknowledgement pushbutton (normally extinguished) flashes in conjunction with an audible alarm if the equipment fails to achieve the last position or state commanded. In addition, the commanded position/status indicating light flashes. This backlit pushbutton is part of the MCB module for that equipment.

Both of these flashing lights are acknowledged by this pushbutton, changing the alarm indication pushbutton from flashing to steady, and the commanded PIL from flashing to extinguished. The steady alarm indication light is not extinguished until the commanded and the actual equipment state are in agreement.

Availability Light/Alarm (Same Backlit Pushbutton as PDL above) (AVL)
If the equipment is removed from service (i.e., if motive power is unavailable or locked out) either deliberately or due to failure, the backlit alarm indication/acknowledgement pushbutton (the same device actuated by the PDL) flashes in conjunction with an audible alarm.

For equipment removed from service, this alarm signal is also an input (through an isolation device) to the Plant Computer System. The Plant Computer System flashes a system level display (BYP) on the MCB indicating that the appropriate system ESF train is bypassed.

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*indicates "Lights Only", see Table C-3.

System Level Bypass Indication (BYP)

An engraved backlit window, prominently displayed to the operator, is provided for each division of each major Safety Subsystem (e.g., SIS, RHR). This window flashes whenever any of the following conditions (within the scope of the window) indicates a bypass of a protective action:

a) Motive power unavailable to an ESF actuation device (for example, an MOV, power unavailable to the reversing contactor), due to deliberate bypass or circuit failure. This condition is derived from "AVL" signal.(AVL/BYP)

b) Valve positioned so as to create a bypass of a protective action. This condition is derived from actual valve position. (PIL/BYP)

c) Window activated manually by operator from MCB, responding to information received through administrative control. (ADM/BYP)

If two redundant divisions of any subsystem are concurrently placed in a bypass mode (due to any of the above inputs), the second division window would flash and an audible alarm would occur. Acknowledgement of the first division level bypass causes the first window to change from flashing to steady, until the bypass is cleared. Acknowledgement of the second (concurrent) division level bypass silences the audible alarm, but leaves the second window flashing until one of the bypass conditions is cleared.

The plant computer systems perform the combination and sequence logic that is required to control the system level bypass indication windows. The position/status inputs to the computer that are derived from Class IE control circuits are isolated in accordance with Regulatory Guide 1.75.

The bypass indication system meets or exceeds the requirements of Regulatory Guide 1.47. Additional design criteria for the bypass indication system are provided in Section 7.5.1 of the PDR.

System Level Monitor Indication (MON)

An engraved, backlit window, prominently displayed to the operator, is provided for each division of each major safety subsystem (e.g., SIS, CSS). This window flashes, in conjunction with its corresponding PDL light(s), whenever any equipment (within the scope of the window) has failed to respond to an ESF signal.

Control Circuit Design Features: In addition to the display features described above, circuit design features are provided to assure proper alignment of equipment. These features include assignment of control priorities to ESF signals and selection of failure modes. These control features are described below.

Control Priority Assignment (CP)

While the equipment is in service (i.e., while motive power is available to it), its control priorities are assigned such that ESF signals will always override non-ESF signals (with the exception of electrical and mechanical circuit protection features which must override ESF signals in order to prevent component damage).

Failure Mode of Actuation Device (FM)

Removal of an air operated or solenoid operated valve from service (i.e., removing motive power) will cause the valve to move to the safe position.

Administrative Control Input (Manual) to Bypass Indication System (ADM)

The system level bypass indication (BYP) can be manually input by the operator through administrative control. Computer software supplements plant administrative controls by tracking these manual inputs (together with non-manual inputs), determining the system level effects, and providing appropriate displays.

Table C-6 illustrates the specific application of these design features to the generic types of FNP equipment that could be incorrectly operated. The

table indicates which of the FNP control and display design features provide direct defense against:

a) The effects of mispositioned circuit breakers or contactors

b) The effects of mispositioned valves, <u>or</u>

c) Undetected mispositioning of equipment for various conditions of plant operation and for various types of equipment.

Considcred in the table are:

a) The nature of the safety system bypass (deliberate vs. inadvertant)

b) The plant operating mode (periodic test, maintenance, etc.)

c) The engineered safety features systems mode (standby vs. active)

d) The type of safety equipment (circuit breaker, motor operated valve, hand operated valve, etc.)

Table C-6 applies only to FNP components which are important to safety. The table does not include other types of design features (e.g., process alarms) that in some cases would further enhance safety.

Additional Control Room design information is contained in the response to 10CFR50.34(e)(2)(iii).

TABLE C-6, SHEET 1

DESIGN FEATURES THAT VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

TYPE OF SAFETY-RELATED EQUIPMENT / NATURE OF BYPASS AND ESF STATUS	CKT BRKRS, OR CONTACTORS OPERABLE FROM MCB	VALVES OPERABLE FROM MCB: ACTUATED BY ESF SIGNAL: NOT ALIGNED IN "SAFE" POSITION AT POWER		VALVES OPERABLE FROM MCB: NOT ACTUATED BY ESF SIGNAL: ALIGNED IN "SAFE" POSITION AT POWER		HAND OPERATED VALVES (NOTE 3): OPERATED PERIODICALLY AT POWER (> ONCE/YR)	OPERATED INFREQUENTLY AT POWER (≤ 1/YR)	OPERATED AT STARTUP/ SHUTDOWN ONLY	OPERATED AT REFUELING ONLY
		MOV's	AOV's	MOV's	AOV's				
DELIBERATE BYPASS: ESF IN STANDBY ·On-Line Periodic Test of ESF	CP	CP	CP	PIL/BYP PIL ADM/BYP	PIL/BYP PIL ADM/BYP	(Note 1) PIL*/BYP PIL* ADM/BYP	ADM/BYP		
·On-Line Non-Routine Maintenance (ESF Equipment Temporarily Removed From Service For Repair)	AVL/BYP ADM/BYP	AVL/BYP ADM/BYP	FM AVL/BYP ADM/BYP	PIL/BYP AVL/BYP PIL ADM/BYP	FM PIL/BYP AVL/BYP PIL ADM/BYP	N/A	ADM/BYP		
INADVERTENT BYPASS: ESF IN STANDBY, PLANT AT POWER ·Plant Operator Error (Note 2)	CP	CP	CP	PIL/BYP PIL	PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*
·Field Operator Error: Hand Repositioning	N/A	CP	CP	PDL PIL/BYP PIL	PDL PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*
·Field Operator Error: Removal from Service	AVL/BYP ADM/BYP	AVL/BYP	FM AVL/BYP PDL	PIL/BYP AVL/BYP PIL	FM PIL/BYP AVL/BYP PIL	N/A	N/A	N/A	N/A
·Loss of Act. Equip. Motive Power	UNDER-VOLTAGE ALARM AVL/BYP ADM/BYP	AVL/BYP	FM AVL/BYP PDL	AVL/BYP PIL/BYP PIL	FM AVL/BYP PIL/BYP PIL	N/A	N/A	N/A	N/A
·Loss of Act. Device Control Power	AVL/BYP ADM/BYP	AVL/BYP	FM AVL/BYP PDL	AVL/BYP PIL/BYP PIL	FM AVL/BYP PIL/BYP PIL	N/A	N/A	N/A	N/A
·Valve "Drifts" From Position	N/A	CP	CP	PDL PIL/BYP PIL	PDL PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*
BYPASS: ESF ACTIVE	PDL/MON PIL	PDL/MON PIL	PDL/MON PIL	PIL/BYP PIL	PIL/BYP PIL	(Note 1) PIL*/BYP PIL*	(Note 1)	(Note 1) PIL*/BYP PIL*	(Note 1) PIL*/BYP PIL*

LEGEND:
- CP — CONTROL PRIORITY
- FM — FAILURE MODE
- PIL — POSITION INDICATION LIGHT (BACKLIT PUSHBUTTON)
- ADM/BYP — ADMINISTRATIVELY CONTROLLED LIGHT
- AVL/BYP — AVAILABILITY LIGHT
- PDL/BYP — POSITION DISAGREEMENT LIGHT
- PIL/BYP — POSITION INDICATION LIGHT

(ADM/BYP, AVL/BYP, PDL/BYP, PIL/BYP:) INPUTS TO SYSTEM LEVEL BYPASS (BYP) INDICATION

- PDL/MON — POSITION DISAGREEMENT INPUT TO SYSTEM LEVEL MONITOR (MON) INDICATION
- * — LIGHTS ONLY

See Sheet 2 for Notes

TABLE C-6, Sheet 2

NOTE 1. Valves are locked in safe position, and are under administrative control.

NOTE 2. "Operator error" includes failure to recognize a valve that is left improperly positioned (for power operation) following startup.

NOTE 3. Safety-related hand operated process valves that have the capability of significantly degrading a protective action if left mispositioned are subject to the following criteria:

a) If normally operated more frequently than once per year with the plant at power, shall be locked in the safe position under administrative control. In addition, remote position indication shall be provided.

b) If normally operated at startup, shutdown and/or refueling, shall have the provisions of paragraph a).

c) If only operated for non-routine maintenance or repair (e.g., to isolate a pump or heat exchanger for repair) with the plant at power, shall be locked in the safe position under administrative control.

NOTE 4. This table includes only those control and display features that provide direct defense against these conditions, recognizing that others of these features might be provided for a particular component, but would be less relevant.

NOTE 5. Where more than one design feature provides defense, the most prominent one is listed first.

REGULATION 10CFR50.34(e)(2)(vi)

Subject: RCS Vents

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide the capability of high point venting of non-condensible gases from the reactor coolant system and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

OFFSHORE POWER SYSTEMS RESPONSE

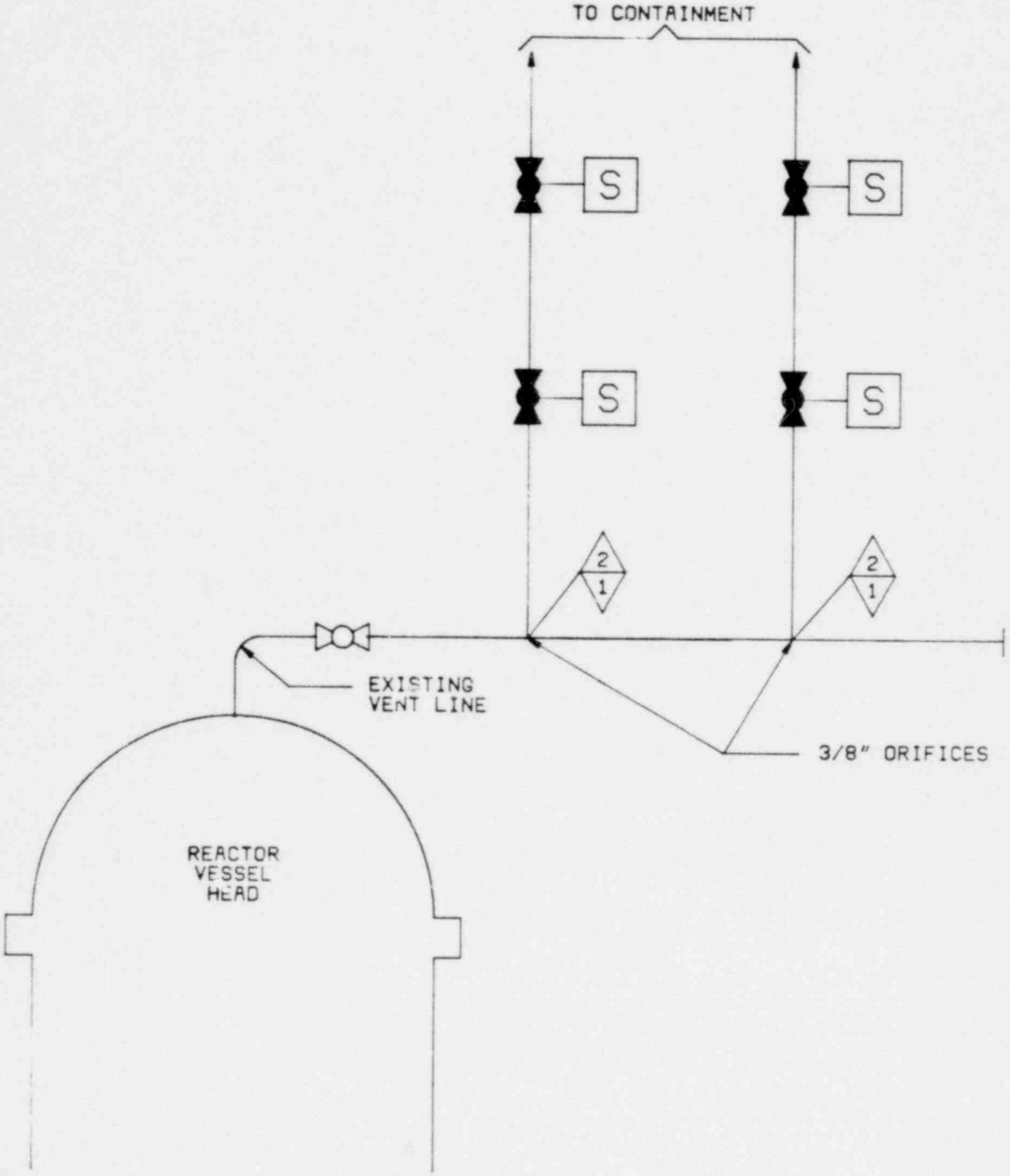
The FNP will include reactor vessel head and pressurizer vent systems which are designed to remove gases from the reactor coolant system via remote manual operations from the control room. The reactor vessel head and pressurizer vent systems are completely independent systems which provide the capability of venting separately the reactor vessel head and the pressurizer. The reactor vessel head and pressurizer vent systems will discharge into a well ventilated area of the containment in order to ensure adequate dilution of combustible gases.

The reactor vessel head vent system flow diagram is shown in Figure C-8. The system arrangement provides for venting the reactor vessel head by using only safety grade equipment. The system mainly consists of 1-inch vent piping with four Safety Class 2 "fail closed" isolation valves. To eliminate potential downtime due to isolation valve seat leakage, the system utilizes all normally closed valves. The isolation valves are powered from redundant Class 1-E buses. The system is designed such that any single active failure will not prevent vessel gas venting nor prevent venting isolation. The system is capable of being dismantled with relative ease for refueling, and provides the necessary manual venting functions during vessel filling operations. All piping and equipment between the orifice and the discharge point is Safety Class 2.

The pressurizer vent system flow diagram is shown in Figure C-9. The system arrangement provides for venting the pressurizer by using only safety grade equipment. The system consists of 1-inch vent piping with four Safety Class 2 "failed closed" isolation valves. To eliminate potential downtime due to isolation valve seat leakage, the system utilizes normally closed valves. The isolation valves are powered from redundant Class 1-E buses. The system is designed such that any single active failure will not prevent pressurizer venting nor prevent venting isolation.

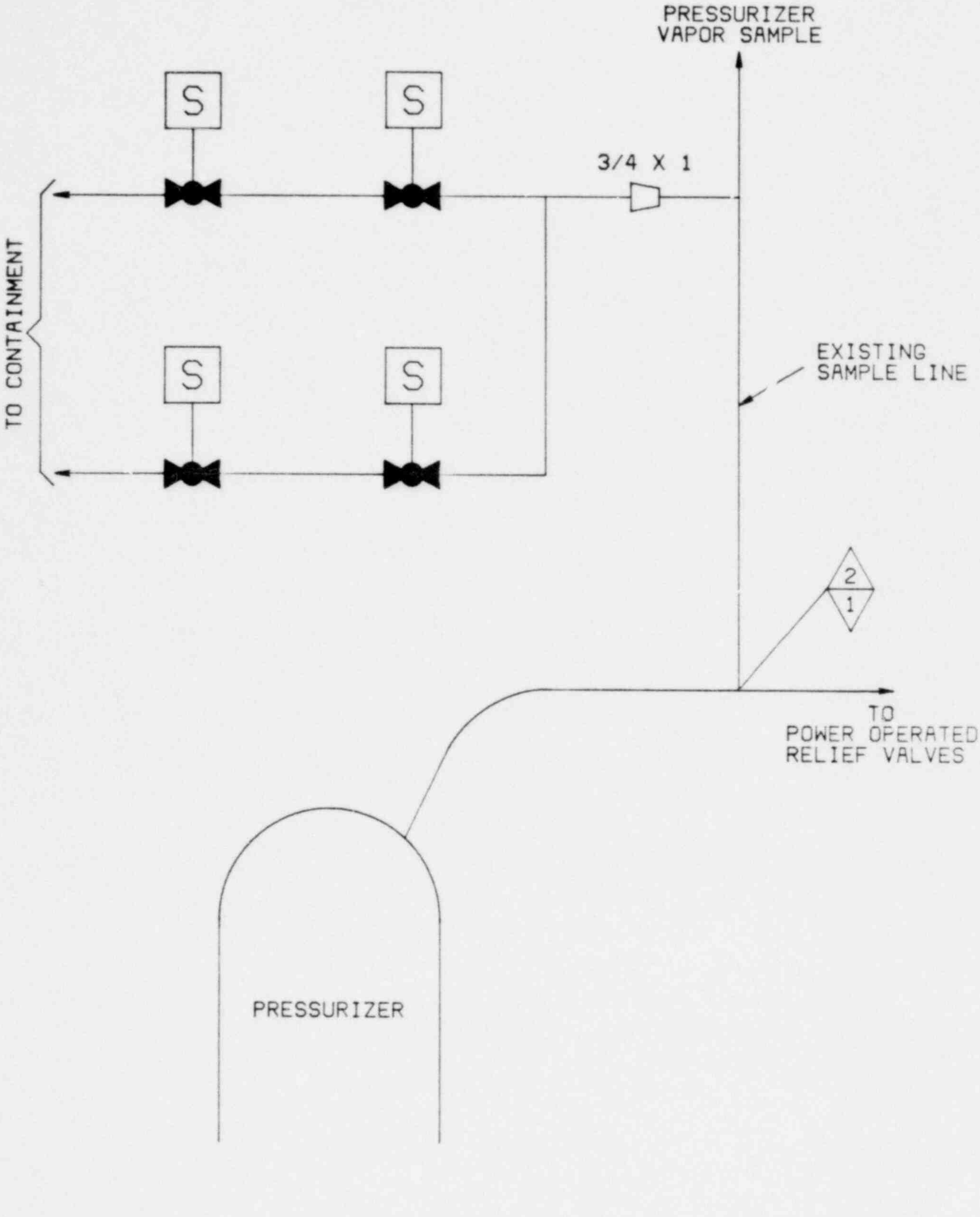
The system connnects to the Nuclear Safety Class 2 pressurizer vapor sample line, which is normally filled with steam. The pressurizer venting system is of Nuclear Safety Class 2 up to the discharge point to the containment.

The contribution of these vent systems to the probability and consequences of small-break LOCA will be evaluated during the probabilistic risk assessment discussed in the response to 10CFR50.34(e)(1)(i).



REACTOR VESSEL
HEAD VENT SYSTEM

FIGURE C-8



PRESSURIZER VENT SYSTEM

FIGURE C-9

REGULATION 10CFR50.34(e)(2)(vii)

Subject: Radiation Design Review

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment (II.B.2)

OFFSHORE POWER SYSTEMS RESPONSE

Post-accident release of radioactivity, as defined in Regulatory Guide 1.4, has been used to derive source terms for the existing design of the FNP shielding around equipment in systems that may contain highly radioactive fluids or gases as a result of accidents. The existing design includes provision for access to emergency coolant recirculation equipment for maintenance following a loss of coolant accident, since long term post-accident operation of this equipment must be assured. Following is a more detailed summary discussion of the current FNP post-accident design basis and design features. As part of the detailed final design of the FNP, a comprehensive design review will be conducted in accordance with NUREG-0737 to insure that shielding for systems which may contain highly radioactive fluids or gases following an accident is adequate to meet dose rate criteria for vital areas, or potentially vital areas, on the plant. Should the additional review so indicate, design modifications will be implemented to permit adequate post accident access or to protect safety equipment from the radiation environment.

Preliminary review of the vital areas on the plant indicates compliance with NUREG-0737. These vital areas are: the Control Room, the Technical Support Center, ERA (OSC) and the Post-Accident Sampling Station and Radiochemical/Chemical Analysis Station.

The FNP control room has been designed to meet General Design Criterion 19, assuming continuous occupancy over 30 days following a Design Basis accident. The average whole body gamma dose rate in the control room is <5 mrem/hr., which is less than that required by item 3a of NUREG-0737. The detailed dose analysis for the FNP control room is given in Section 6.5 of the Plant Design Report. The FNP onsite Technical Support Center (See the response to 10CFR50.34(e)(2)(xxv) which will be located adjacent to the Control Room, is provided with the same degree of radiological protection as the control room. The post-accident sampling area (see the response to 10CFR50.34(e)(2)(viii)), which will be located between the control room and the shield building, will also be provided with radiological protection such that GDC 19 criteria will be met.

On the FNP, controls for actuation of the post-LOCA hydrogen control system, containment isolation reset controls, manual ECCS alignment, motor center controls, vital instrument panels and emergency power supply actuation are all located inside of the Control Building.

Most of the systems which normally interface with the Reactor Coolant System (either directly or indirectly) are isolated from the Reactor Coolant System following an accident in which significant quantities of radioactivity are released. (Release of radioactivity is considered potentially significant if concentrations in the reactor coolant are greater than those associated with 1% failed fuel under normal operating conditions.) Those systems which are isolated from the reactor coolant system are the following:

1. Gaseous Waste Treatment System (WTG)
2. Sampling System (SSR)
3. Chemical and Volume Control System (CVC)
4. Boron Recycle System (BRS), and
5. Liquid Waste Treatment System (WTL)[1]

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(1) In the existing FNP design, the Safeguards Area sumps are drained to the Liquid Waste Treatment System. The design will be changed such that, following an accident, liquids collected in these sumps will be pumped back to the containment sump. (See the response to 10CFR50.34 (e)(2)(xxvi).

The only systems interfacing with reactor coolant which are not isolated are:

1. Safety Injection System (SIS) (for initial coolant injection), including upper head injection,
2. Residual Heat Removal Sytem (RHR) (for coolant recirculation), and
3. Containment Spray System (CSS) (for spray injection and recirculation)
4. Post-Accident Sampling System (PASS)

Systems potentially containing post-accident gaseous sources, which are not isolated, are the Annulus Filtration System (AFS) and the Containment Atmosphere Sampling System.

These three systems (piping and components) are located within four, separate, shielded safeguards compartments in the FNP.

Shielding thicknesses for spaces in which these systems are located were calculated employing a source derived in accordance with Regulatory Guide 1.4. The source term includes 50% of the core equilibrium halogen inventory and 1% of all other fission products uniformly mixed in the containment sump water inventory. Table C-7 lists the core fission product inventory for predominant isotopes. Noble gases are not included in the fluid sources used for design of shielding for these spaces, an assumption which is justified for recirculated depressurized cooling water. The sources employed are documented in Table 12.1.4 of the PDR. Radiation sources for post-accident sampling system piping include 100% of the noble gases, 50% of the halogens and 1% of all others.

The criterion for shielding of the systems in the safeguards compartments is that the dose in potentially occupied areas outside the shield walls shall not exceed 3 Rem for an 8 hour exposure beginning at 24 hours after an accident. This dose criterion (<3 Rem for an 8 hour exposure one day after the accident) is the post-accident shield design criterion for all post-accident work locations on the plant except for the vital areas previously discussed. In addition, the FNP is designed with an Emergency

TABLE C-7

CORE FISSION PRODUCT INVENTORY FOR
PREDOMINANT ISOTOPES (CURIES)
(BASED ON 3565 MWT)

Isotope	Core Inventory (Curies)	Isotope	Core Inventory (Curies)
Sr-89	9.2(7)	CS-134	2.1(7)
Sr-90	6.1(6)	Te-135	1.7(8)
Y-90	6.4(6)	I-135	1.6(8)
Y-91	1.2(8)	I-136	7.0(7)
Zr-95	1.7(8)	CS-136	5.8(6)
Nb-95	1.7(8)	CS-137	8.6(6)
MO-99	1.8(8)	Ba-140	1.8(8)
Ru-103	1.4(8)	La-140	1.8(8)
Rh-103M	1.4(8)	Ce-141	1.7(8)
Rh-105	6.7(7)	Ce-143	1.5(8)
Ru-106	5.1(7)	Ce-144	1.1(8)
Rh-106	7.6(7)	Pr-143	1.4(8)
Ag-110M	3.5(5)	Pr-144	1.1(8)
Ag-111	4.3(6)	Pm-147	9.0(6)
Cd-113M	1.0(3)	Kr-83M	7.7(6)
Cd-115M	6.2(4)	Kr-85M	2.4(7)
Cd-115	8.8(5)	Kr-85	9.0(5)
Sn-123	9.4(5)	Kr-87	4.6(7)
Sn-125	1.5(6)	Kr-88	6.7(7)
Sb-125	7.4(5)	Kr-89	8.3(7)
Te-125M	2.5(5)	Xe-131M	8.0(5)
Sb-127	8.3(6)	Xe-133M	4.6(6)
Te-127M	1.6(6)	Xe-133	1.9(8)
Te-127	8.1(6)	Xe-135M	5.2(7)
Te-129M	6.6(6)	Xe-135	3.5(7)
Te-129	3.9(7)	Xe-137	1.9(8)
I-129	2.9(0)	Xe-138	1.8(8)
I-131	1.0(8)		
Te-132	1.4(8)		
I-132	1.4(8)		
I-133	1.9(8)		
I-134	2.2(8)		
Br-83	7.8(6)		
Br-84	1.8(7)		
Br-85	2.2(7)		

NOTE: Numbers In Parentheses Refer To Powers of 10.

Relocation Area (as part of the Control Building) which is located at the 100' elevation. The Emergency Relocation area is provided with the same degree of radiological protection as the Control Room and is designed to accommodate personnel safely throughout the course of an accident. The Onsite Operational Support Center is located in the Emergency Relocation Area (see the response to 10CFR50.34(e)(2)(xxv).

The RHR, SIS and CSS system components within each safeguards compartment are located in a subcompartment which is isolated from the rest of the safeguards compartment during normal operation. These systems are the only ones which are likely to contain post-accident radioactivity. Ventilation is provided by a sealed system such that neither supply nor exhaust air lines communicate the subcompartment to the surrounding space. In the event of an accident resulting in containment isolation, subcompartment exhaust is lined up to the Annulus Filtration System (AFS). The AFS maintains the subcompartment at a negative pressure, thus assuring that any airborne radioactivity released within the subcompartment is exhausted to the annulus, where it passes through charcoal and HEPA filters before release to the environment. Because of this unique design, liquid leaks from the SIS, RHR or CSS systems will not result in release of airborne radioactivity within the surrounding spaces. This configuration is shown pictorially in the response to 10CFR50.34(e)(2)(xxvi).

Special consideration will be given during final design to post-accident handling of fluids which may leak from pumps in the RHR-SIS-CSS subcompartments. In the event of a large leak, recirculation flow from the containment sump to the affected subcompartment can be terminated by closing the appropriate sump isolation valve. These are motor operated valves with the motor outside the shield wall. Manual valve wheels are also provided at the motor so that the valve may be closed even in the event of motor operator failure.

The FNP has been designed so that post-accident maintenance may be performed on either of the two RHR pumps by draining and flushing the RHR equipment. Drain and flush operations can be performed via reach rod operated valves located outside the shield walls of the RHR pump rooms.

Airborne activity released to the RHR subcompartment would be removed by the annulus ventilation system which maintains a negative pressure in the subcompartment. Additionally, the design basis for equipment important to safety includes a requirement for satisfactory operation following post-accident radiation exposure.

Source terms based on Regulatory Guide 1.4 for release to the containment are given in Section 12.1 of the PDR. Doses and dose rates outside the shield building as a function of time after the accident (based on those source terms) are given in Section 15.4 of the Plant Design Report.

Environmental qualification conditions for FNP equipment are given in Table 3.11-1 of the Plant Design Report. The radiation doses listed are based on source terms developed in accordance with Regulatory Guide 1.4. During the detailed design of the plant, a more detailed listing will be prepared.

To summarize, the existing design philosophy for controlling radioactive water and airborne activity following an accident involving core damage is to isolate non-essential systems which could transport post-accident radioactivity outside containment. Systems outside the containment which are needed following an accident for core cooling or containment atmosphere cooling are located within shielded subcompartments, which are part of each separate safeguards compartment. These subcompartments are maintained at a negative pressure and are connected to the annulus following an accident. Source terms specified in Regulatory Guide 1.4 were used for design of shielding for post-accident work locations near systems which could potentially contain highly radioactive water. Vital areas on the FNP will meet the requirements of NUREG-0737 and GDC-19, Dose Design Basis.

REGULATION 10CFR50.34(e)(2)(viii)

Subject: Post-Accident Sampling

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems will include in the Floating Nuclear Plant a Post-Accident Sampling System (PASS) which meets the requirements of NUREG-0737. The PASS will provide the capability for both on-line analyses and the collection of grab samples for analysis at a location(s) remote from the sample collection point. On-line analytical equipment is included in the scope of the PASS. Equipment used to analyze grab samples and the procedures for grab sample analysis will be provided by the plant owner and described in the owner's application. The PASS proposed by Offshore Power Systems is described below.

The modified PASS will permit obtaining liquid and gas samples within 1 hour of an accident condition which releases an assumed TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole body or 75 rem to the extremities. It will provide for remote manual collection and processing of liquid and gas samples with maximum practical use of online instrumental methods of analysis. Provisions are also included for remote dilution of liquid and gas samples and for degassing of liquid samples. Additional shielding will be added as necessary to satisfy the dose limitation. Reactor coolant and

containment samples can be obtained with the PASS during post-accident conditions independent of auxiliary systems which may be isolated.

The following in-containment sample points are included:

- o Sample points on two reactor coolant hot legs
- o Pressurizer liquid sample
- o Pressurizer vapor space sample
- o Reactor vessel vent sample
- o Containment sump sample
- o Upper reactor compartment at outlet of two air recirculation fans
- o Lower reactor compartment at three ice condenser door locations

The in-containment sampling portion will be designed to minimize the amount of sample collection and analysis equipment located inside the containment and thus maximize accessibility to the components. Motive force for the reactor coolant samples will be supplied by either system pressure (if available) or by redundant positive displacement pumps in-containment. The containment sump sample will be pumped by means of the same positive displacement pumps. A single bellows pump will supply the driving force for the containment atmosphere samples and will be located outside containment. Sample lines will be designed to provide representative samples and to minimize the possibility of sample line blockage.

Isolation valves in the system will be remote-manual, solenoid operated with electrical power being supplied from redundant Class 1-E buses. The liquid and gas sample lines may be flushed with demineralized water and dry nitrogen respectively from outside containment. The flush liquids as well as excess sample flows are returned to either the containment sump or the pressurizer relief tank (for liquids) or to the containment atmosphere (for gas). When the system is being used under normal plant operating conditions, the excess liquid samples and flush solution may be routed to the waste holdup tanks of WTL.

The samples described above are piped to a special sample analysis station where samples are processed both by remote manual and online techniques.

The sample analysis station will consist of a concrete cubicle located close to the containment outside wall. The cubicle provides capability for remote, on-line analysis of some parameters. All of the sample handling equipment and analytical monitors for remote analysis will be located inside the cubicle with controls and readouts located on a remote panel. A glove box within the cubicle provides the capability to obtain grab samples for analysis. Shielding will be provided sufficient for short-term access to the cubicle for sample removal. Air exhausted from the cubicle will be passed through the charcoal adsorbers and HEPA filters.

The following types of grab samples will be obtainable from the glove box.

- o Undiluted pressurized liquid sample
- o Undiluted containment atmosphere sample
- o Diluted liquid sample
- o Diluted containment atmosphere sample
- o Degassed undiluted liquid sample

Grab samples can be obtained at remote locations with sufficiently low levels of background radiation to reduce analysis errors to be maximum factor of approximately 2. Among the analyses of grab samples which can be performed are (1) the detection of radionuclides which are indicative of the degree of core damage and (2) the identification and quantification of the radionuclide categories identified in Regulatory Guides 1.4 and 1.7. The pressurized reactor coolant grab sample can be analyzed for total gas and for gaseous constituents. Grab samples provide complete backup for the on-line analyses described below.

The following chemical analyses will be performed on-line by flow-through instrumentation remotely controlled and monitored.

- o Containment atmosphere hydrogen concentration
- o Liquid soluble boron concentration
- o Liquid pH
- o Liquid soluble chloride concentration
- o Liquid and gas gross activity

MODES OF OPERATION

The liquid and gas sampling systems are independent. Each is designed so that a sample of liquid or gas can be remotely processed and analyzed in one of several different ways. Prior to analyzing a liquid or a gas sample, its gross radioactivity level can be checked by circulating the sample through a radiation monitoring station, the excess sample being returned to the containment. The methods by which samples are obtained and analyzed are outlined below.

Undiluted Samples (Liquid or Gas)

To obtain an undiluted liquid or gas sample, the sample point is selected by appropriate valve lineup, a sample pump is started and the sample stream is circulated out of containment, through a shielded sample vial, and back to the containment. The sample vial is then isolated remotely and the entire piping system external to the sample vial is purged with flush water or dry nitrogen as appropriate. The sample vial is then manually removed from the system, put into a shielded transport cask inside the shielded glove box, and removed for off-site analysis. This sample may be analyzed for certain radionuclides which are indicators of the degree of core damage (e.g., noble gases, iodines, cesiums and non-volatile isotopes).

Diluted Samples (Liquid or Gas)

A diluted sample of liquid or gas is obtained in much the same way as for the undiluted sample except an additional dilution operation is performed. Once the undiluted sample has been drawn into a calibrated sample vial, isolated, and the system flushed, the sample vial is lined up to a recirculating loop containing a known volume of pure diluent (water or nitrogen) and the sample is mixed with the diluent by a circulating pump. The ratio of initial sample volume to dilution system volume is predetermined by calibration (typically 1:100). The required order of dilution is obtained remotely by successively isolating the sample vial, purging and refilling the dilution loop and remixing by circulation. A radiation monitor is used to verify the number of dilution cycles required. The sample vial is finally isolated and removed as for the undiluted sample.

Depressurized/Degassed Liquid Samples

A liquid sample is degassed by collecting it in a small stripping column which is part of a recirculating loop containing a shielded sample vial and a pump. The liquid is recirculated while nitrogen is bubbled through the stripping column. This system operates at atmospheric pressure. The depressurized, degassed sample is then either collected as before in the sample vial and removed for analysis, or the liquid sample is circulated to one of the remotely-operated flow-through analyzers included in the sampling station.

Remote Soluble Boron Analysis

The degassed and depressurized liquid sample is circulated through a flow-through analyzer which measures the boron concentration by a neutron attenuation technique. Readout is on the remote Sample System Control Panel.

Remote Soluble Chloride Analysis

The degassed and depressurized liquid sample is circulated through a cell containing a specific-ion electrode system for chloride. Readout is on the remote Sample System Control Panel.

Remote Solution pH Analysis

The degassed and depressurized liquid sample is circulated through a cell containing a hydrogen-ion electrode system. Readout is on the remote Sample System Control Panel.

Remote Gross Radiation Level

The gross radiation level of either liquid or gas samples is determined by a wide range radiation monitoring station on the respective sample inlet lines. These monitoring stations will be located outside containment, and the readout is on the remote Sample System Control Panel.

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Remote Atmospheric Hydrogen Concentration

The concentration of hydrogen in the containment atmosphere is determined by passing the atmospheric sample stream through a thermal conductivity hydrogen monitor located outside containment. The gas stream then returns to the containment. Readout is on the remote Sample System Control Panel.

Handling of Excess Liquid Samples and Flush Samples

The excess liquid sample and flush volumes could be routed either to the pressurizer relief tank or sump inside containment or to the plant waste holdup tank in the auxiliary building. The latter would be used only when radiation levels were low as during normal plant operations. The excess gas and nitrogen purge would be discharged into the containment atmosphere.

REGULATION 10CFR50.34(e)(2)(ix)

Subject: Hydrogen Control System

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. (II.B.8)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems will install a Hydrogen Ignition System (HIS) in the containment of the Floating Nuclear Plant capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. The HIS employs distributed hydrogen ignition sources located throughout the containment building. The ignition sources are of the thermal element (or glow plug) type. Shortly after activation these glow plugs reach temperatures which are adequate to reliably ignite combustible hydrogen/steam/air mixtures.

OPS will incorporate the results of industry and NRC research programs, such as AIF-IDCOR, EPRI, Sandia, Livermore, Fenwall, etc., which will demonstrate the ignition characteristics of these glow plug igniters. Within two years after receipt of the manufacturing license, design details, describing the Hydrogen Ignition System, will be provided to the NRC for review; these design details, including test data and analyses, will illustrate that the hydrogen control systems will perform in the manner required by the NRC position. The analyses will define the environmental conditions inside containment that would result from hydrogen burning.

The HIS consists of approximately 62 glow plug igniter assemblies located in 31 distinct locations in the containment building. Each igniter assembly consists of a glow plug and a control power transformer similar to those

used for Sequoyah and McGuire Nuclear Stations. The glow plug and transformer are mounted in a sealed metal box housing which employs heat shields to limit the temperature rise inside the box and a drip shield to reduce direct moisture impingement on the thermal element. The igniter boxes are seismically mounted to prevent damage to Category 1 equipment.

Each designated containment location has two igniter assemblies powered from separate emergency power trains. The igniters are powered from 120 VAC buses. In the event of loss of offsite power the igniter assemblies will be supplied power from the emergency diesel generators. Cables of the two divisions are physically and electrically separated.

Glow plug igniters are located throughout the containment to promote hydrogen burning in all areas prior to reaching hydroegn concentrations sufficient to threaten containment integrity. The glow plugs within each compartment or subcompartment are located near the ceiling since, if there is any non-uniform distribution of hydrogen, the higher concentrations would be expected to exist there. Mounting the assemblies near the ceiling also minimizes interference with the operation of equipment and provides some degree of physical protection for the protruding glow plug. The igniters are located in the incore instrument tubing chase, pipe chases, steam generator enclosures, pressurizer enclosure, instrument room, below the operating floor, in the ice condenser upper plenum below the top deck doors, and in the containment dome.

The Hydrogen Ignition System will be automatically actuated in accident situations which have the potential for the generation of excessive quantities of hydrogen. To effectively perform their intended function the HIS igniters must be energized and at operating temperature before significant amounts of hydrogen are released to the containment atmosphere. Therefore, actuation of the HIS will occur once the potential for excessive hydrogen generation is established and will not be dependent upon a measurement of the hydrogen concentration inside containment. Signals for automatic initiation of the HIS and the setpoints will be selected during the development of system design details. Each division of the HIS can also be manually actuated from the control room.

REGULATION 10CFR50.34(e)(2)(x)

Subject: SV and RV Qualification

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed and not before issuance of an ATWS rule. (II.D.1)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems will implement the results of the industry testing necessary to qualify the reactor coolant system relief, safety valves, and block valves under expected operating conditions for design basis transients, and accidents. The effect of as-built relief and safety valve discharge piping on valve operability will be accounted for, and the discharge piping and supports will be designed for all loads resulting from expected operating conditions for design basis transients and accidents.

The Electric Power Research Institute (EPRI) has developed a generic program to verify the operational characteristics of PWR safety and relief valves and to provide assurance that these systems can perform ass required to prevent overpressurization of the primary coolant boundary. The program plan for the "Performance Testing of PWR Safety and Relief Valves", Rev. 9, July 1980 has been submitted to the NRC Staff. The experimental data together with foreign relief valve test results will be used to validate a computational methodology for assessing the hydraulic/structural performance of PWR safety/relief valve systems on a plant unique basis.

REGULATION 10CFR50.34(e)(2)(xi)

Subject: SV and RV Position Indication

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)

OFFSHORE POWER SYSTESM RESPONSE

Positive indication of pressurizer relief valve position is currently provided in the FNP design. Such indication is accomplished in the following manner:

1. Each PORV has indication lights on the control board which are activated by stem-actuated limit switches powered from vital instrument buses. In addition, a position disagreement light/alarm prominently displays a failure of the PORV to achieve the last position commanded. The equipment for the position indication and alarm will be designed in accordance with the requirements of NUREG-0737.

2. The temperature downstream of the PORVs and safety valves is displayed on the control board and high temperature alarms are provided.

3. The pressurizer relief tank has temperature, pressure and fluid level indication and alarms on the main control board.

4. High pressurizer pressure alarms in the Control Room.

Offshore Power Systems is presently evaluating alternate methods to provide safety valve position indication. One such system has been developed and is described below.

Westinghouse has developed an acoustic leak monitoring system that will provide flow indication downstream of the safety valves and thus satisfy

the NRC requirements for leakage detection. The system operates on the principle that turbulent, high pressure flow through an orifice generates an acoustic signal which is transmitted throughout the reactor coolant system. The monitoring system will detect acoustic signals and thus determine valve position. The FNP will incorporate either the Westinghouse acoustic leak monitor or stem mounted limit switches meeting the requirements of NUREG-0737 after acceptance by the NRC. An alarm in the control room will be provided in conjunction with valve position indication.

REGULATION 10CFR50.34(e)(2)(xii)

Subject: Auxiliary Feedwater Automatic Initiation/Flow Indication

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)

OFFSHORE POWER SYSTESMS RESPONSE

Auxiliary Feedwater System auto-start provisions are detailed in the response to 10CFR50.34(e)(1)(ii). The automatic and manual initiation signals and circuits for the auxiliary feedwater system will be in accordance with safety grade requirements and the criteria in NUREG-0737. In addition:

1. The components and circuits for the control of auxiliary feedwater during the post accident sequence, after automatic system initiation has been reset, willl meet the criteria applicable to safety systems.

2. The design will be such that no single failure (considering both valve closure and pump tripping) will prevent isolation of auxiliary feedwater flow to a steam generator affected by a feed or steam line break. Analyses to demonstrate that auxiliary feedwater flow to a faulted steam generator is either precluded or terminated prior to exceeding acceptable limits of either containment pressure or reactor power in the event of a steam line break will be submitted for staff review and approval within two years after issuance of the construction permit. The analyses shall comply with the requirements of Section 15.1.5 of the Standard Review Plan.

3. The auxiliary feedwater system controls and indication used for shutdown from outside the control room will meet the criterion applicable to safety systems.

Auxiliary feedwater flow channels, with an accuracy of better than the required $\pm 10\%$, will be displayed on the main control board. Each channel of flow instrumentation is powered from its respective associated Class 1E instrument power supply and will be safety-grade. There will, however, be only one channel of indication for each steam generator.

REGULATION 10CFR50.34(e)(2)(xiii)

Subject: Pressurizer Power Supplies

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)

OFFSHORE POWER SYSTEMS RESPONSE

The FNP design provides the following features which assure a continued supply of power for the following plant components essential to natural circulation flow.

1. Pressurizer heaters

   The total pressurizer heater capacity for the FNP is 1800 KW. Four separate backup heater groups (346 KW each) are supplied directly from 4 independent and redundant safety class 480V switchgear buses. Each bus is supplied from its respective standby diesel generator following a loss of offsite power. The control group (416 KW) is supplied from a non-safety class 480V bus which could be supplied from a diesel-generator bus within several minutes following a loss of offsite power, in the unlikely event that this should become necessary.

   Each independent backup group is large enough to maintain natural circulation in the hot standby condition.

   The Class 1E circuit breakers supplying each of the backup groups are tripped open on either a safety injection (SI) or loss of offsite power actuation signal.

The heaters can be manually loaded onto the bus from the main control board after SI is reset and loads required in the initial stages of the incident are no longer required. Sufficient diesel generator capacity is provided to supply the minimum required number of heaters in the time required (1 hour). Diesel generator instrumentation is provided in the control room to prevent overloading a diesel generator with these heater loads.

OPS will provide the owner with the necessary procedures for energizing the pressurizer heaters, including procedures that might be required for load shedding.

2. Power Operated Relief Valves (PORV's)

Each PORV is supplied with operating air from a separate Safety Class-3 air system which is available following a loss of offsite power. Each PORV pilot solenoid is supplied from independent and redundant 125V DC sources, which are also available following a loss of offsite power. The PORV's are controlled from the main control board. Both PORV's fail closed on loss of motive or control power.

3. PORV Block Valves

The PORV block valves are supplied from motor control centers which are readily energized from a corresponding standby diesel generator following a loss of offsite power. The PORV block valves are controlled from the main control board. Thus the PORV block valves can also be operated following a loss of offsite power.

4. Pressurizer Level Indication Channels

All of the pressurizer level indication channels are derived (and isolated) from their respective protection channels. The instrument loop power supplies for these protection channels (including the isolated outputs) are supplied from their respective Class 1E Instrument buses. Thus level indication is available following a loss of offsite power.

REGULATION 10CFR50.34(e)(2)(xiv)

Subject: Containment Isolation Systems

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) for each non-essential penetration (expect instrument lines) have two isolation barriers in series,

(C) do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) include automatic closing on a high radiation signal for all systems that provide a path to the environs.

OFFSHORE POWER SYSTEMS RESPONSE

The FNP containment isolation system, described in Section 6.2 of the Plant Design Report satisfies the acceptance criteria of Standard Review Plan 6.2.4. Containment isolation system features specifically required by this rule are addressed below:

A. Phase A isolation (T signal) results in the isolation of all non-essential systems penetrating the containment with the exception of component cooling water lines to the reactor coolant pumps and the lower compartment fan coolers (which are part of the same line penetrating containment as shown in Figure 9.2-1 of the Plant Design Report) and which are closed by Phase B isolation (P signal).

Phase A isolation provides for diversity in parameters sensed as well as being automatically actuated any time a safety injection signal (S

signal) is initated. Phase A isolation is initiated from the following process variables:

(1) High steam flow coincident with low steam line pressure or lo-lo T_{avg}.
(2) High steam line differential pressure
(3) Low pressurizer pressure
(4) High containment pressure
(5) Manual initiation

Phase B isolation is initiated from hi-hi containment pressure or manually. Although it is not automatically generated by diverse means, the P signal can only be generated after the T signal, which is diverse, has been initiated. In addition to initiating Phase B isolation, the P signal also is used to initiate containment spray.

Offshore Power Systems has given careful consideration to the systems penetrating the containment which are required to mitigate the consequences of a loss of coolant accident, or any accident calling for containment isolation. The systems which are required to operate following the accidents are as follows:

- Safety Injection System
- Residual Heat Removal System (supply lines to cold legs)
- Containment Spray System (including recirculation sump lines)
- Upper Head Injection System
- Auxiliary Feedwater System

The above systems are required to supply cooling and/or make up fluid to the Reactor Coolant System, the containment, and the Main Steam System. These systems, or parts of these systems required for post-accident cooling, do not receive any containment isolation signal.

The following systems are not essential to mitigate the consequences of a design basis loss of coolant accident but are considered desirable in assisting in plant recovery from accidents of lower magnitude than a

design basis accident. They are not part of Phase A isolation, but instead are isolated by the P signal (Phase B isolation).

- Component Cooling Water System (supply and return lines to RCP thermal barrier cooling)
- Component Cooling Water System (cooling water flow to the lower compartment fan coolers)

The following systems have been determined to be non-essential and are isolated by the T signal (Phase A):

- Chemical and Volume Control System
- Post-Accident Sampling System
- Radiation Monitoring System (containment air sample lines)
- Nuclear Sampling System
- Containment Ventilation System
- Post-Accident Containment Ventilation System
- Liquid Waste Treatment System
- Service Air System - Instrument Air System
- Emergency Air System
- Ice Condenser Refrigeration System
- Non-Essential Service Water System
- Reboiler Condensate Return System
- Reboiler Steam Distribution System
- Fire Protection Water Spray System
- Safety Injection System (test lines)
- Upper Head Injection System (test lines)
- Containment Purge Supply and Exhaust System

B. All non-essential lines are properly isolated with two barriers in series following the initiation of a containment isolation signal. In addition to the systems which are listed as being subject to Phase A isolation, other non-essential systems or lines which penetrate containment have normally closed manual isolation valves suitable for administrative control in accordance with SRP 6.2.4, Item II.3.F. Administrative controls will be developed and implemented by the owner.

C. Containment isolation reset logic requires deliberate and specific operator action before an isolated line can be reopened. The following control features are provided for containment isolation valves:

(1) The containment isolation signals override all other automatic control signals.

(2) The valves will remain in the closed position if the initiating signal is reset.

(3) Each valve can be opened or closed manually after the appropriate containment isolation signals are reset.

(4) Any valves that are normally operated in an automatic mode (for non-safety functions) are also automatically transferred to manual mode by the isolation signal. This precludes automatic opening of containment isolation valves subsequent to reset of the initiating isolation signal.

D. During Floating Nuclear Plant final design, the containment high pressure trip point will be reviewed in accordance with NUREG-0737 and adjusted downward (if necessary) to the minimum compatible with conditions not requiring automatic containment isolation.

E. Systems that provide an open path from the containment to the environs are the containment purge supply and vent systems. In the present design, isolation valves in these systems are automatically closed on high radiation. This design will be upgraded to provide closure on a safety grade high radiation signal. The safety grade radiation monitors will be located in relation to the in-service purge system containment isolation valves such that the fraction of containment atmosphere that is discharged through these isolation valves, before these valves have been isolated by the high radiation signal, will not result in doses that exceed offsite dose requirements.

REGULATION 10CFR50.34(e)(2)(xv)

Subject: Containment Purge/Vent Systems

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

OFFSHORE POWER SYSTESMS RESPONSE

Containment purging is performed by either of two systems (1) Containment Pre-access Filtration and Purge System or (2) Post Accident Containment Venting (Purge) System. These systems, which are described in Section 6.2 and 9.4 of the Plant Design Report, are designed in compliance with Standard Review Plan 6.2.4 and Branch Technical Position 6-4.

The Containment Pre-access Filtration and Purge System provides a continuous purge function at a restricted flow during normal plant operation via an 8 inch diameter supply and 8 inch diameter exhaust penetration. The system is capable of purging via a 42 inch diameter penetration; however, the plant owner will be required to restrict such operation to refueling operations or when the plant is in cold shutdown. Procedures will require that the 42 inch diamater purge valves remain closed during normal power operation. Each of the two purge lines will be provided with separate containment penetrations, each isolated by two valves in series[1].

(1) At present one of the 8 inch purge lines penetrates containment via one of the 42 inch lines (see the Plant Design Report, Chapter 9, Figure 9.4-6, Sheet 4). An additional 8 inch penetration and isolation valve will be provided with a containment isolation valve inside and outside the containment shell. This design change provides separation of the two purge functions and maximizes the reliability of the containment isolation function.

The Post Accident Containment Venting System provides for hydrogen purge. The system provides a controlled and filtered containment purge capability by releasing air to the annulus at a maximum rate of 50 SCFM.

Isolation valves are designed to operate against accident pressures and to maintain bubble air-tight closures while performing their intended function. The isolation valves have a 2 to 5 second closure time. The containment pre-access filtration and purge system containment isolation valves are described in Section 6.2.3.3 of the PDR. The 8 inch isolation valves will be included in the operability assurance plan described in Section 3.9.2.4 of the Plant Design Report; operability and performance of these valves will be consistent with the requirements contained in the NRC letter of September 27, 1979 (Guidelines for Demonstration of Operability of Purge and Vent Valves).

REGULATION 10CFR50.34(e)(2)(xvii)

Subject: Containment Instrumentation

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide instrumentation to measure, record and readout in the control room: (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

OFFSHORE POWER SYSTEMS RESPONSE

A. Containment Pressure:

To comply with the requirement for containment pressure monitoring, two additional wide range containment pressure channels will be incorporated into the FNP. These additional channels will range from minus 5 psig to 60 psig (4 times design pressure). The channels will meet the design requirements of Regulatory Guide 1.97, Rev. 2 (Dec. 1980) and NUREG-0737. The present instrument range is 0-18 psig.

B. Containment Water Level:

As described in Section 6.2.2.7 of the PDR, the Floating Nuclear Plant design does not incorporate a conventional containment sump as such. Instead, the containment lower compartment will collect a sufficient volume of water following the injection phase of safety injection to allow recirculation. Redundant safety grade containment water level (wide range) measurement is currently provided and displayed in the Control Room. The range of these level channels will be increased to cover an elevation equivalent to an 800,000 gallon accumulation, a quantity which includes ice melt and UHI accumulator injection.

In addition, Class 1E (narrow range) level channels will be provided for the local liquid waste treatment system sump at the 103 foot elevation in accordance with this requirement. These channels will also be used as part of the Reactor Coolant System Leak Detection System. These channels will meet the design requirements of Regulatory Guide 1.89, NUREG-0737 and Regulatory Guide 1.97 (Revision 2).

C. Containment Hydrogen Concentration:

A continuous indication of hydrogen concentration in the containment atmosphere will be provided in tne Control Room. Measurement capability will be provided over the range 0% to 30% hydrogen concentration under both positive and negative ambient pressure. Hydrogen monitors which can perform this function are presently available but have not yet been qualified to IEEE-323 and IEEE-344. Hydrogen monitoring instrumentation will be designed in accordance with NUREG-0737 and Regulatory Guide 1.97 (Rev. 2) and will be qualified to the requirements of NUREG-0588. If during final design, instruments qualified to NUREG-0588 are not available, applicable qualification requirements will be worked out with NRC.

D. Containment Radiation Intensity (High Range)

The current FNP design for the redundant containment area monitors specifies a range of 10^{-1} to 10^{7} Rad/Hr of gamma radiation. These monitors will meet the requirements of both NUREG 0737 and Regulatory Guide 1.97, Rev. 2. It should be noted that these detectors for the FNP design are mounted on the outer surface of the steel containment but may be considered as "In-containment" relative to compliance with this requirement. The attenuation by the steel shell will be factored into the calibration of the monitors. Interference from non-containment sources will be eliminated by proper shielding of the detectors. The monitors will be located such that they are widely separated, view a large fraction of the upper compartment and have an unobstructed view down to the operating deck. Although the detectors will be mounted high on the containment dome, they can be readily accessed for maintenance

using platforms and ladders already included in the FNP design for inservice inspection of the containment shell. Mounting the detectors outside the steel containment serves two safety related purposes: 1) the need for containment cable penetrations is eliminated, and, 2) the monitors will experience less severe postulated accident environmentional conditions, (i.e., temperature, humidity, and pressure).

E. Airborne Radiological Effluent Monitors
(Refer also to PDR Sections 11.4 and 12.2):

The current FNP design includes monitors to detect airborne effluent from three potential release points: the plant vent, the condenser air ejectors, and the Annulus Ventilation System AFS) exhaust vent. Also included is a passive plant vent charcoal cartridge for iodine detection.

Two pairs of radiogas and particulate monitors are used to monitor the plant vent and the AFS vent. Each particulate monitor consists of a high-range and low-range channel. Each of these monitor pairs can be selected to the following three sample locations

1) Plant vent
2) AFS vent
3) Containment atmosphere

Following an accident ("S" signal) one pair of monitors is assigned automatically to continuously monitor the plant vent and the other is assigned to continuously monitor the AFS vent. These monitors satisfy the requirements of NUREG-0737 and Regulatory Guide 1.97, Revision 2.

A radiogas monitor will have an upper detection limit of $10^4 \mu Ci/cc$ of Xe-133. The lower end of the range, will be sensitive to a concentration as low as $10^{-7} \mu Ci/cc$, in order to monitor normal plant releases. The 12 decades of response will be obtained with a multi range (3 levels) detector. The high range particulate monitor will be replaced

by a passive filter cartridge that can be removed for analysis. The low range particulate monitor in the present design will be retained.

In order to allow safe collection and analysis of the plant vent charcoal and filter cartridges during and immediately following an accident, provisions will be made in the plant design to do the following (or the equivalent):

1) place the high range cartridges in the post accident sampling room and route the sample lines from the plant vent to the post accident sampling room, which is shielded and habitable immediately following an accident, or

2) design and provide a system for safe remote collection of the cartridges for analysis in the post-accident sampling room.

The more practical of these alternatives will be selected as the design evolves. All effluent monitoring channels will have assured power supplies, independent of offsite power (i.e., Class 1E or Class 1E Associated Power). The Owner will provide the procedures for collection and analysis of cartridge and filter samples.

The current FNP design includes a noble gas monitor on the condenser air ejector discharge. The monitor will be upgraded to meet the requirements of NUREG 0737 and Reg. Guide 1.97, Rev. 2 including increasing the range to 10^{-6} to 10^{5} μCi/cc of Xe-133.

In order to meet the new requirement for monitoring releases from the atmospheric steam dump valves or the main steam safety valves, four monitors will be added to the plant, one for each main steam line. These monitors will view the main steam line upstream of the main steam stop valve and will have a range of 10^{-1} to 10^{3} μCi/cc of steam.

REGULATION 10CFR50.34(e)(2)(xviii)

Subject: Core Cooling Instrumentation

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and incore thermocouples in PWR's and BWR's. (II.F.2)

OFFSHORE POWER SYSTEMS RESPONSE

OPS continues to evaluate options developed by the Westinghouse Owners' Group regarding instrumentation for detection of inadequate core cooling and will install a subcooling meter and a reactor water level indicator. Of the options presented in NUREG 0737, the one preferred will be selected during final design. Procedures used by the operator to recognize inadequate core cooling will be developed based on the instrumentation provided in the final FNP design.

Subcooling Meter

An approach being considered by OPS is to provide dedicated, redundant, microprocessor-based subcooling meter channels with prominent displays on both the Unit Control Console and the Safety Center Panel (Refer to the attachment in the response to 10CFR50.34(e)(2)(iii) for a description of the FNP Control Board). Each of these meters would provide a continuous indication of margin to saturated conditions. The operator could manually select a display of margin to saturation based on either the auctioneered high incore thermocouple or the auctioneered high loop T_{hot} or T_{cold}. Auctioneered low reactor coolant system pressure is used for the T_{sat} calculation by the microprocessor. Inputs to this system would utilize redundant safety grade hot leg and cold leg temperatures and reactor coolant system pressure channels. In addition, approximately 16 in-core thermocouple inputs (together with reference junction temperature inputs) would be utilized as a backup to the plant computer display system.

Two setpoints would be utilized to alarm 1) off-normal conditions and 2) approach to loss of core cooling. Individual sensor channels will also be accessible for display in the control room.

Table C-8 provides a summary of tentative design information for the FNP subcooling meter.

Additional Instrumentation

Offshore Power Systems is in the process of evaluating the various methods of measuring reactor vessel level that have been investigated by the Westinghouse Owners' Group. The current state of the art appears to favor the use of differential pressure measurement as the best method of determining vessel level. This method would utilize sealed reference legs and would range from the vessel top (using an existing penetration) to the vessel bottom (using an incore instrumentation thimble). In addition, taps on the middle of the hot leg pipes would be utilized for level measurement with the Reactor Coolant Pump(s) tripped. All differential pressure measurements would require temperature compensation. The entire level measurement system would be redundant and Class 1E, and would be a dedicated system independent of other control or instrumentation channels.

The usefulness of this type of system toward providing an unambiguous indication of inadequate core cooling and an unambiguous indication for vessel venting is being evaluated, with attention given to all possible phenomena that could adversely affect the system. The results of the evaluation and preliminary design information (as required by NUREG-0737) will be submitted following completion of prototypical testing and prior to procurement of the equipment. The objective of this submittal is to keep the NRC informed of the design testing and implementationprogress. Details will be provided in the final design report.

TABLE C-8

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press., etc.)	Tsat-T, where 'T' is based on either incore or RTD temperatures (Note 1)
Display Type (Analog, Digital, CRT)	Analog (Note 1)
Continuous or on Demand	Continuous and on demand
Single or Redundant Display	Redundant
Location of Display	Unit Control Console and Safety Center
Alarms (include setpoints)	(Note 2)
Overall uncertainty (^{0}F)	(Later)
Range of Display	40^{0}F Superheat to 200^{0}F Subcooled
Qualifications (seismic, environmental)	IEEE-344, -323 Based on NUREG 0737

Calculator

Type (process computer, dedicated digital or analog calc.)	Dedicated Digital
If process computer is used specify availability. (% of time)	Not Applicable
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press.)	Auctioneered high incore temp. or Auctioneered high RCS temp. vs: Auctioneered low RCS press.
Qualifications (seismic, environmental)	IEEE-344, -323 Based on NUREG 0737
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

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Input

Temperature (RTD's or T/C's)	8 incore T/C's (2 per quadrant)
Temperature (number of sensors and locations	2 Hot Leg RTD's (per loop) 2 Cold Leg RTD's (per loop) T/C ref. junc. RTD's
Range of temperature sensors	Incore T/C's = 150^{o}-2300^{o}F RCS RTD's: 0^{o}-700^{o}F
Uncertainty* of temperature sensors (^{o}F at 1)	(Later)
Pressure(specify instrument used)	(Note 3)
Pressure (number of sensors and locations)	2 (RCS Hot Legs)
Range of Pressure sensors	0-3000 psi
Uncertainty* of pressure sensors (PSI at 1)	(Later)
Qualifications (seismic, environmental)	IEEE-344, -323 Based on NUREG 0737

Backup Capability

Availability of Temp & Press	Yes (Note 1)
Availability of Steam Tables etc.	Yes (By Owner
Training of Operators	(By Owner)
Procedures	(Later)

*Uncertainties must address conditions of forced flow and natural circulation

NOTES:

1. Individual sensor readouts will also be available, as well as other derived readouts (e.g., temperature differentials, P-Psat based on highest temperature, etc.), utilizing different indicators.
2. Two setpoints will be chosen to indicate: a) off-normal conditions (50^{o}F nominal) and b) approach to loss of core cooling (later).
3. Qualified instrument will be specified later.

REGULATION 10CFR50.34(e)(2)(xix)

Subject: Post-Accident Instrumentation

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damge. (II.F.3)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems has committed that the FNP design for Post-Accident Monitoring will comply with Regulatory Guide 1.97 Revision 1. The present FNP design includes much of the instrumentation required to meet Revision 2. Those recommendations of Revision 2 not already in the current design will be incorporated or a suitable alternate will be provided for those items that challenge the state-of-the-art. Design information for alternate instrumentation and justification of its adequacy will be submitted for NRC review prior to equipment procurement.

REGULATION 10CFR50.34(e)(2)(xx)

Subject: Power for PORV, Block Valves, Level Instrumentation

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

OFFSHORE POWER SYSTEMS RESPONSE

(A) Pressurizer Level Indication Channels

All of the pressurizer level indication channels are derived (and isolated) from their respective protection channels. The instrument loop power supplies for these protection channels (including the isolated outputs) are supplied from their respective Class 1E Instrument buses. Thus level indication is available following a loss of offsite power.

(B),(C) Motive and Control Power Sources and Connections

(1) Power Operated Relief Valves (PORV's)

Each PORV is supplied with operating air from a separate Safety Class-3 Emergency Air System which is available following a loss of offsite power. Each PORV pilot solenoid is Class 1E and is supplied from independent and redundant Class 1E 125V DC sources, i.e., Trains A & C which are also available following a loss of offsite power. The PORV's are controlled from the main control board. Both PORV's fail closed on loss of motive or control power.

(2) PORV Block Valves

The PORV block valves are supplied from qualified Class 1-E motor control centers which are readily energized from a corresponding standby diesel generator following a loss of offsite power. The PORV block valves are controlled from the main control board. Thus the PORV block valves can also be operated following a loss of offsite power. The motive and control power for the block valves are supplied from Trains B and D.

REGULATION 10CFR50.34(e)(2)(xxv)

Subject: Post-Accident Support Facilities

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide a Technical Support Center, an onsite Operational Support Center, and an Emergency Operations Facility. (III.A.1.2)

OFFSHORE POWER SYSTEMS RESPONSE

Emergency Response Facilities will be designed in accordance with guidance provided in NUREG-0696. The technical support center (TSC) is an onsite facility located close to the control room that provides plant management and technical support to the reactor operating personnel located in the control room during emergency conditions. It has technical data displays and plant records available to assist in the detailed analysis and diagnosis of abnormal plant conditions and any siignificant release of radioactivity to the environment. The TSC will be the primary communications center for the plant during an emergency. A senior official, designated by the licensee, can use the resources of the TSC to assist the control room operators by handling the administrative items, technical evaluationss, and contact with offsite activities, relieving them of these functions. The TSC facilities may also be used for performing normal functions, such as shift technical supervisor and plant operations/ maintenance analysis functions, as well as for emergencies.

The Onsite Technical Support Center (TSC) for the FNP is located on the mezzanine of the Emergency Relocation Area (ERA) as shown in Figures C-10 through C-12. This center is provided with the same degree of shielding, environmental control, missile protection and security as the Control Room. This center uses a ventilation system equal to the Control Room system. Necessary communication between the TSC and both the Control Room and Onsite Operational Support Center will be provided. Offsite communications will be provided by the owner. As outlined below, plant status can be readily obtained in the TSC during normal as well as emergency operation.

Necessary "as-built" documentation will be filed in the TSC or elsewhere within the shielded control building.

Offshore Power Systems will provide CRT terminals for the SPDS and to access data from the plant computer system. The specific instrumentation required in the TSC will be determined during final detailed design of the FNP. The major portion of the Emergency Response data acquisition system will be provided by the SPDS data system. This system will display the full range of important parameters and data trends on demand and is more fully described in response to 10CFR50.34(e)(2)(iv). Other parameters needed to complete the emergency response data will be provided by the plant computer system. The data acquisition from the plant computer will be secured by either hardware or software to prevent unauthorized access during normal operation. The plant computer has dual CPUs and will meet the availability requirements of NUREG-0696.

OPS believes that the FNP concept provides unique advantages regarding as-built documentation, including the following:

a. greater level of detail on drawings (dimensioning, part numbers, etc.) because of the manufacturing concept.

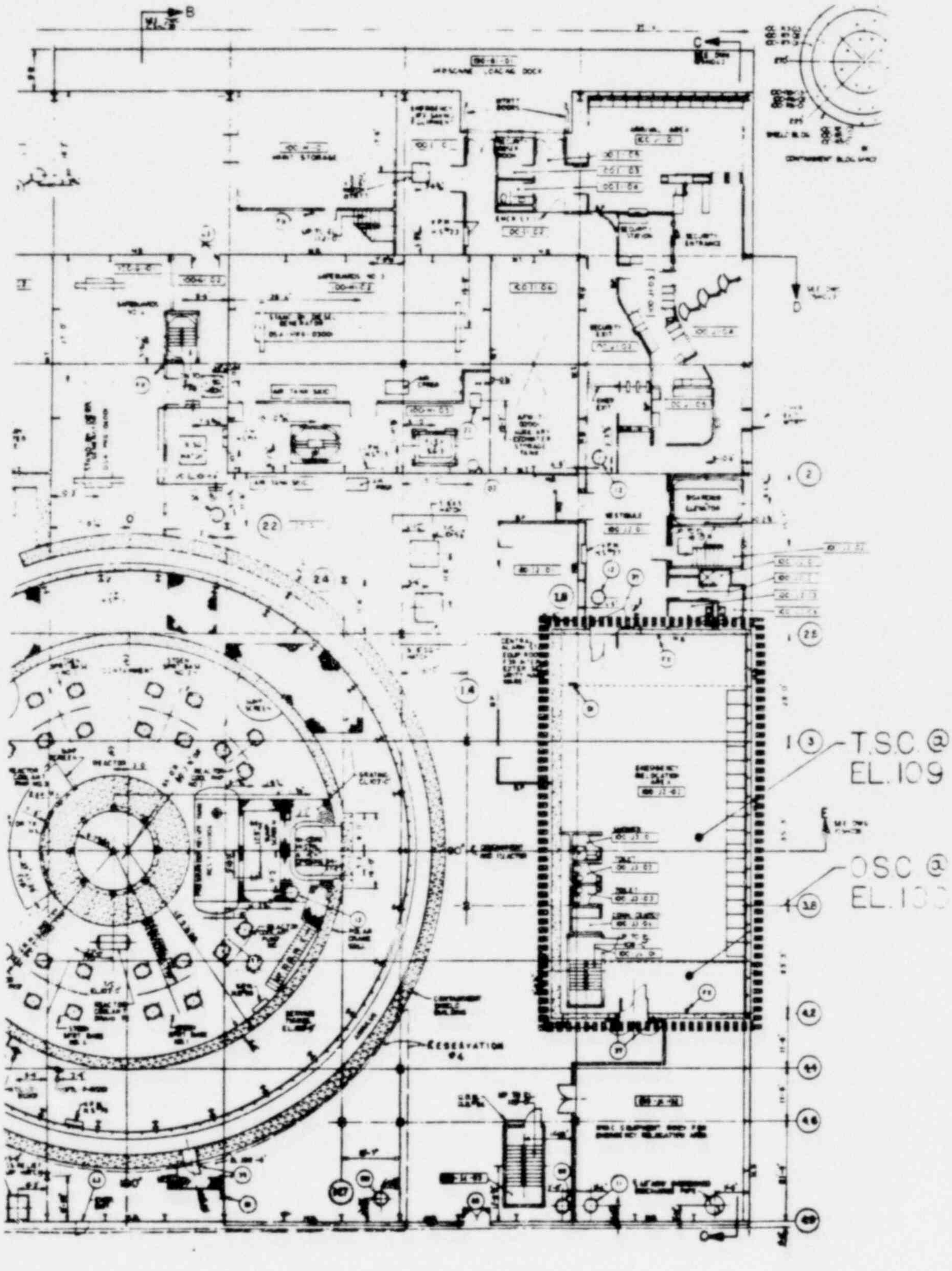
b. greater consistency and coordination among as-built documents, since OPS is ultimately responsible for all as-built documentation for the FNP.

c. FNP units and their documentation would be virtually identical, allowing use of other units for full-scale studies regarding recovery operations.

The Emergency Relocation Area (at Elev. 100' in the control building) beneath the Control Room will be the Onsite Operational Support Center. As shown in Figures C-10, C-11 and C-13. This area is designed to the same criteria for shielding, missile protection and environmental controls as the Control Room. Emergency storage facilities and communications equipment for onsite operational support are provided. The Emergency Relocation Area

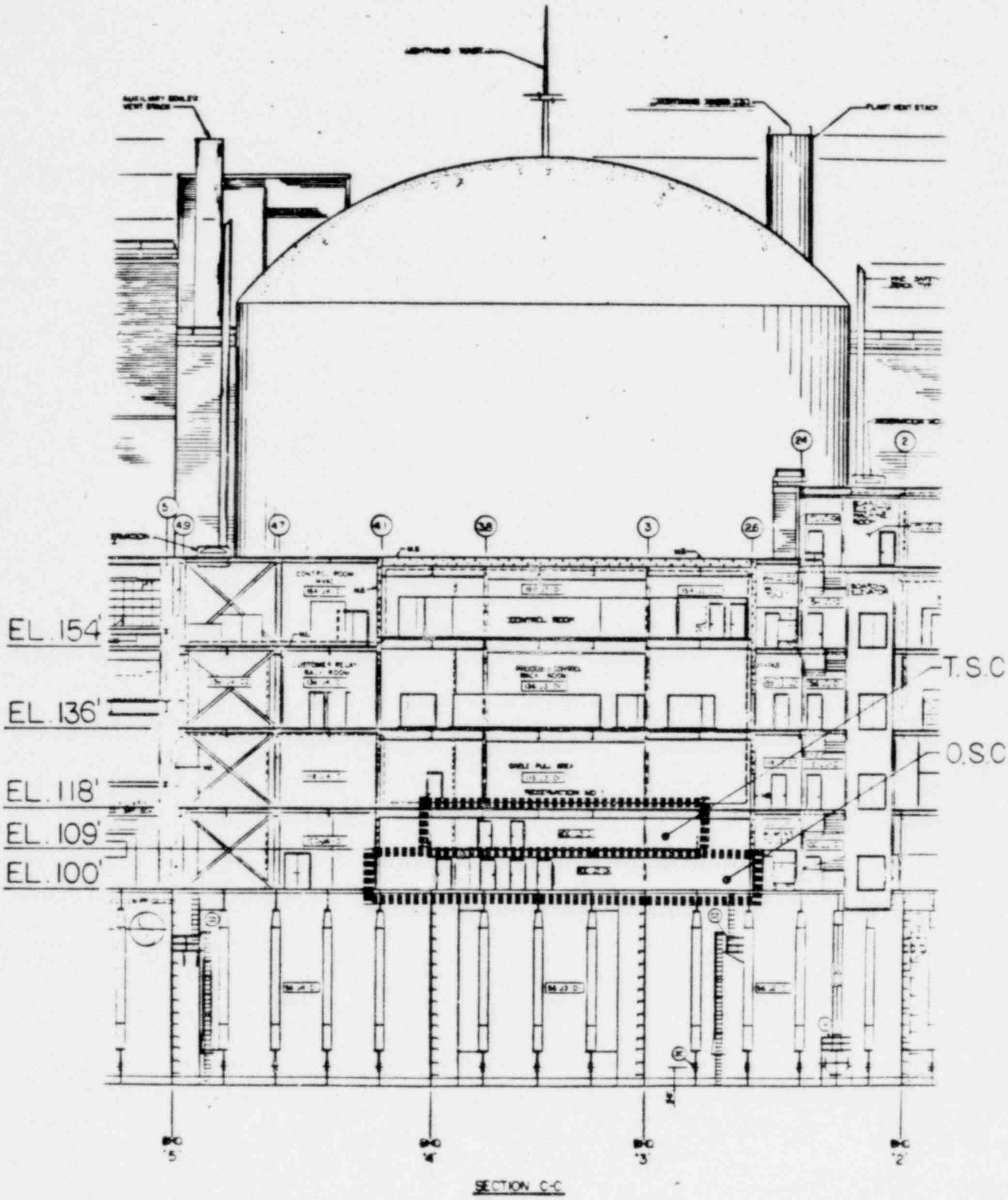
is safely accessible from the Control Room via a stairway which is enclosed within the shielded control building.

The Near-Site Emergency Operations Facility (EOF) will be provided by the plant owner. Provisions will be made for the transfer of information from the TSC to the Owner's communications system interfacing with the EOF.



LOCATION OF T.S.C. AND O.S.C.
(PLAN VIEW)

FIGURE C-10



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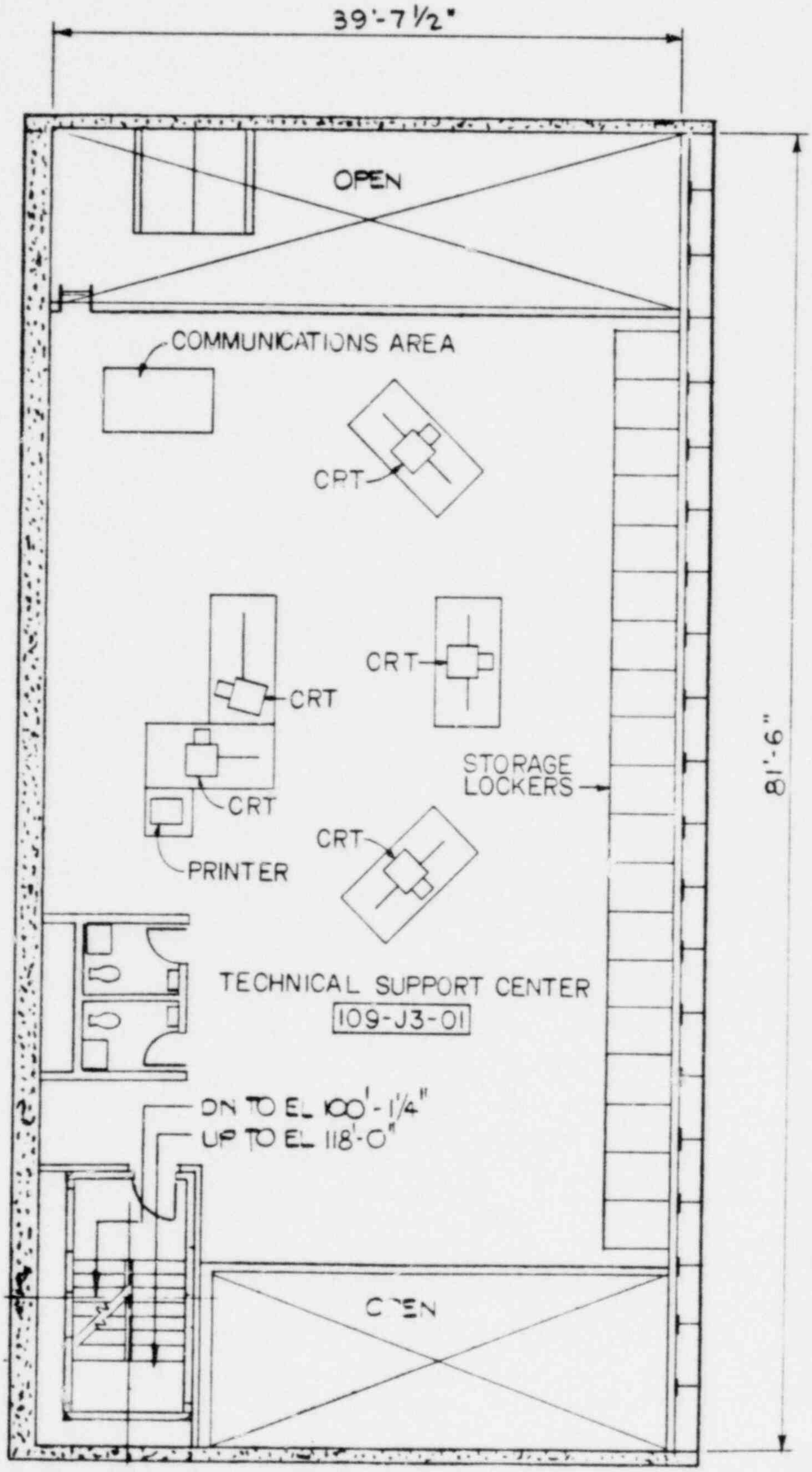
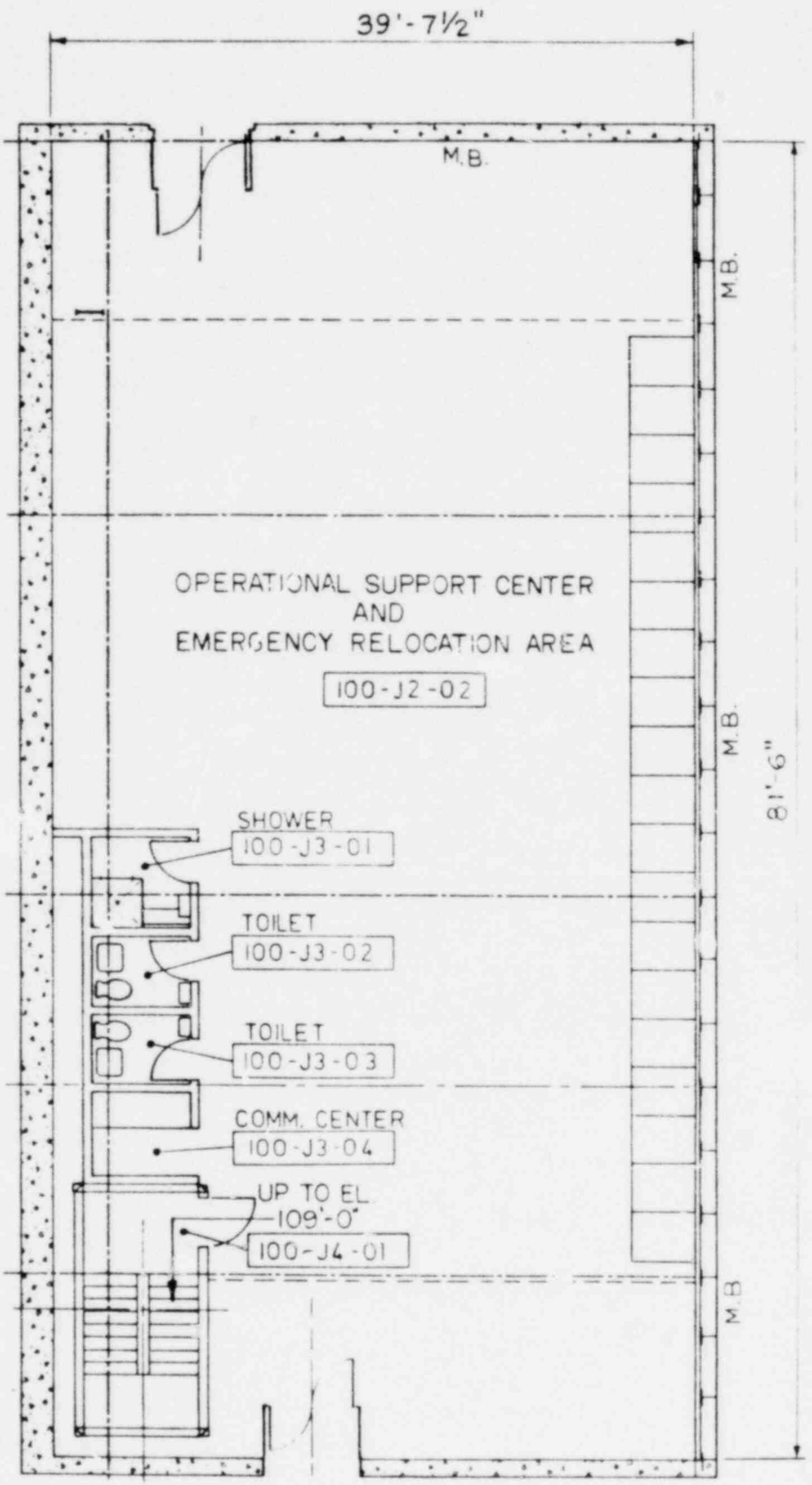
LOCATION OF T.S.C. AND O.S.C.
(ELEVATION VIEW)

FIGURE C-11



PRELIMINARY LAYOUT OF T.S.C.

FIGURE C-12



PLAN VIEW OF O.S.C. AND E.R.A.

FIGURE C-13

REGULATION 10CFR50.34(e)(2)(xxvi)

Subject: Leakage Reduction Outside Containment

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)

OFFSHORE POWER SYSTEMS RESPONSE

All reactor plant systems which could contain TID 14844 source term radioactive materials following an accident are isolated from the Reactor Coolant System except for the following:

1. Residual Heat Removal System (RHR)
2. Safety Injection System (SIS) including Upper Head Injection
3. Containment Spray System (CSS)

Other systems covered by NUREG-0737 are isolated as noted in the response to 10CFR50.34(e)(2)(vii).

A major design feature has been incorporated into the Floating Nuclear Plant which significantly reduces the potential for exposure to workers and to the public following a design basis accident. The feature involves incorporation of the above safety systems within separate safeguard compartments which are shielded, which have 3 hour fire walls and 3 of which are watertight. The safeguards compartments are provided with a controlled atmosphere which is connected to with the containment annulus during accident conditions to prevent the spread of radioactivity to other parts of the plant and reduce release to the environment. This feature is further discussed in Section 11.6 of the Plant Design Report.

These three safety systems will also incorporate various leak testing, leak reduction and/or collection features, including:

1. Welded/seamless piping system;
2. Pumps with mechanical seals;
3. Low point drains from the piping system and drains from equipment such as pumps, leakoff from valves, etc. with double isolation valves, are routed to sumps;
4. High point vents with a single isolation valve and pipe caps or double isolation valves;
5. Pressure test connections for temporary (or local) instrumentation with a single isolation valve and pipe caps or double isolation valves;
6. Leak offs from lantern rings piped to the sump for valves 4 inches and larger.
7. Packless metal diaphragm valves utilized for 2" and smaller valves.

To detect leakage between systems, the following provisions are incorporated in the design:

1. Radiation monitors with alarm annunciation in the control room are provided for the Essential Service Water system which removes heat from the RHR, SIS and CSS, thereby enabling detection of radioactive fluid on the non-radioactive fluid side due to heat exchanger tube leaks;
2. Flushing connections provided for post-accident RHR system maintenance have both an isolation valve and a blind flange to prevent leakage. In addition, relief valves piped to the sump are provided for protection against overpressurization. (See response to 10CFR50.34(e)(2)(vii) for description of post-accident RHR maintenance.)

In addition to the features designed to prevent the leakage of radioactivity, the potential for the release of radioactivity to the environment is further reduced by the provision of secondary isolation and control

within sealed safeguards compartments which contain the RHR, SIS, and CSS systems. The sealed compartments, depicted on Figure C-14, provide a secondary barrier which, together with the automatic activation of the annulus air filtration system prevent the spread of airborne radioactivity within the plant. Filtered air discharged to the environment by the annulus air filtration system is limited to that necessary to maintain a negative pressure.

During normal plant operation the safeguards compartments and containment annulus spaces are maintained under a negative pressure with filtered exhaust systems. Exhaust is transferred to the annulus air filtration system on an "S" signal.

At present, the Safeguards Area sump pumps start on sump level signal and discharge to the floor drain tank of the Liquid Waste Treatment (WTL) System. The Safeguards Area sump flow path will be modified as shown schematically on Figure C-14, to provide an alternate discharge line to the containment sump. The signal for acuating the alternate discharge path will be selected during final design. This new configuration will prevent uncontrolled discharge from the Safeguards Area to the Auxiliary Building following an accident.

The coolant leakage control and detection program relative to the system's potentially containing a TID 14844 source term following an accident are:

1. Design period. Incorporation of the design features discussed herein will minimize potential leakage from the safety systems.

2. Initial test period. Hydrostatic and operational testing of the safety systems will be performed prior to the accumulation of any radioactive contamination. Safeguards area leak testing will be performed during the initial test period.

3. Operational period. Periodic safeguards area leak testing will assure integrity of the three systems. The utility owner/operator will be responsible for developing a preventative maintenance

program will be utilized to reduce leakage from sources outside of containment to as-low-as-practical. The preventative maintenance program will be developed to determine leak rate at startup and at regular intervals thereafter.

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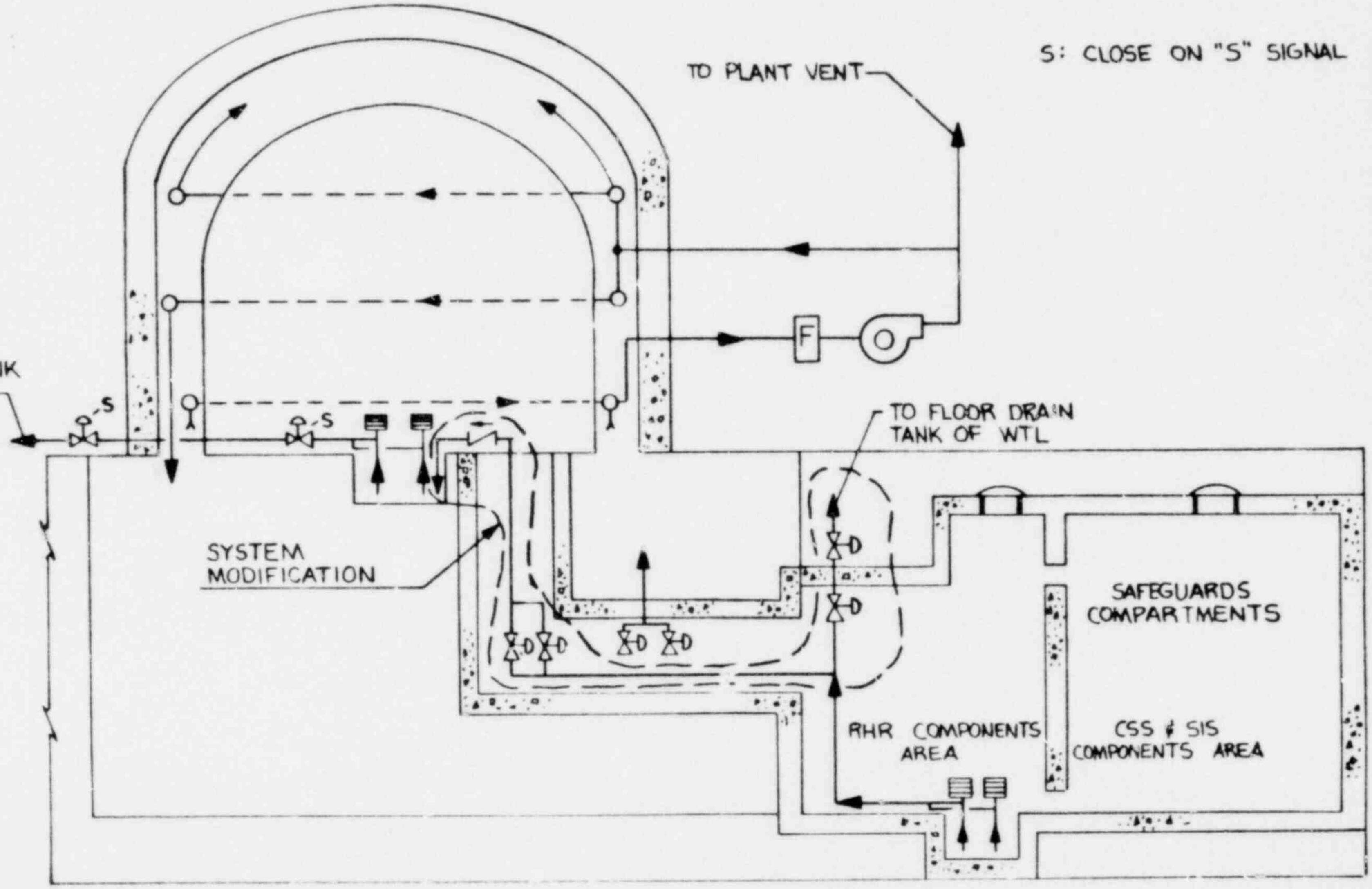

FIGURE C-14

SAFEGUARDS COMPARTMENTS AND CONTAINMENT ISOLATION (TYPICAL)

REGULATION 10CFR50.34(e)(2)(xxvii)

Subject: Radiation Monitoring

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems has designed a comprehensive Radiation Monitoring System (RMS) for the Floating Nuclear Power Plant which provides adequate monitoring for a broad range of plant conditions. Included as part of the RMS are both area monitors and airborne monitors.

The area and airborne monitors assist in assuring that occupational radiation exposures to operating personnnel are kept as low as reasonably achievable. Area monitors detect the ambient gamma radiation exposure in selected areas of the FNP. Airborne monitors supplement area monitors in selected areas of the FNP by sampling the atmosphere to detect the concentration of significant radionuclides in particulate and/or gaseous form.

In addition to the above general functions, certain RMS channels provide indication and/or control for specific functional purposes. These special RMS functions include:

(1) Airborne monitoring of the containment atmosphere for in-containment reactor coolant leakage detection.

(2) High level area monitoring of the containment to follow the radiological course of a loss-of-coolant accident. (See also the response to 10CFR50.34(e)(2)(xvii).)

(3) Monitoring of the control room and emergency relocation area to provide automatic switchover of the ventilation systems for these areas to their emergency mode of operation if a high radiation level is detected. This is an engineered safety feature for the FNP designed to insure habitability of these areas following postulated accidents which could result in a significant release of radionuclides to the plant environs.

(4) Monitoring of the normal and alternate outside air intake ducts of the main control room ventilation system. These channels provide supplementary information to the primary wind direction instrumentation to allow the operator to verify that the least contaminated intake is utilized for ventilation following postulated accidents which could result in a significant release of radionuclides to the plant environs.

A total of twenty-eight area monitors and six airborne monitors are included in the RMS. These monitors are described in detail in the Plant Design Report, Sections 12.1.4 and 12.2.4.

During the final design of the FNP, OPS will review all area and airborne monitors to ensure the adequacy of the design, location and ranges, including a determination of which monitors must meet the requirements of Regulatory Guide 1.97, Revision 2.

With regard to improved in-plant iodine monitoring, Offshore Power Systems will provide space on the FNP for counting rooms and laboratories where analyses of radioiodine concentration can be performed. The location of these spaces and support systems design are such as to permit personnel occupancy for times required to perform necessary analysis following accident conditions. Shielding will be provided to ensure a low background in the counting room. Ventilation with clean air at a pressure higher than surrounding spaces will be provided for the counting room to minimize background airborne contamination in this region. Capability for purging of entrapped noble gases from charcoal samples using either clean air or

nitrogen will be provided in the laboratory area. Residual noble gases will be routed to and vented from the plant stack.

Sampling methods, counting equipment and other laboratory analytical equipment will be provided by the plant owner. The gamma ray spectrometer is a commercially available method for discriminating between residual noble gases and radioiodine adsorbed on the charcoal filters in the atmospheric sampling devices. OPS will recommend to the plant owner that such equipment be procured for analysis of the charcoal filters used for sampling of areas within the facility. Portable airborne iodine samplers and sample analysis equipment, as required by NUREG-0737, Item III.D.3.3 will be provided by the plant owner.

REGULATION 10CFR50.34(e)(2)(xxviii)

Subject: Control Room Habitability

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10CFR50.35(a)(2) or to address unresolved generic safety issues.

Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. (III.D.3.4)

OFFSHORE POWER SYSTEMS RESPONSE

The Floating Nuclear Plant design accommodates required operating personnel in a safe state of occupancy for the duration of postulated accidents. This is accomplished by providing areas within the plant which are protected from external hazards including radiation and toxic gases. These areas, the Control Room and the Emergency Relocation Area, are safeguarded by filtered ventilation systems and by biological shielding. The Emergency Relocation Area (ERA) provides the facilities necessary to support the operating crew, e.g., food supplies and food preparation equipment, medical supplies, sleeping accommodations, and communications equipment.

Biological shielding for the Control Room has been designed to comply with General Design Criterion 19 (i.e., 5 rem gamma whole body dose, 30 rem thyroid dose, and 30 rem beta skin dose) for the duration of an accident. Postulated accidents analyzed include the loss-of-coolant accident, fuel handling accident, main steam line rupture and gas decay tank rupture. Detailed results of the habitability analyses, which are in compliance with Standard Review Plan 6.4, are given in Section 6.5 of the Plant Design Report.

Control Room shielding, designed to attenuate the direct radiation from fission products within the containment and those leaked from containment, consists of a 2 foot thick concrete roof and concrete wall on the side facing the containment building, and one foot thick walls on the sides not

facing the containment. The back wall of the control room consists of a 1-1/8 inch thick steel plate missile shield. These walls extend from the 172' elevation down to the 100 foot elevation, which is the level of the lower floor of the containment building. These walls thus encompass the Control Room, the Process Rack Room, the Cable pull area, and the ERA. The four rooms are separated by a floor/ceiling which is a 3-hour fire barrier. Source terms for the radiation analysis are based on applicable Regulatory Guide assumptions. For the loss-of-coolant accident, Regulatory Guide 1.4 assumptions were used.

Protection from airborne radioactivity is provided by the control building ventilation system operating in the post-accident mode. The post-accident mode is initiated automatically by a high radiation alarm on any of four monitors: 1) the air particulate and/or gas monitors in the plant vent stack, and, 2) the area monitors in the Control Room and Emergency Relocation Area. Basically the post-accident operational mode consists of closing all Control Room and ERA exhaust ports, providing controlled intake to maintain a positive pressure, and recirculating the internal atmosphere. This positive pressure prevents inleakage of potentially contaminated air from surrounding spaces. Independent ventilation systems serve the ERA and Control Room. Each system also continuously recirculates a considerable quantity of air through the filters to remove any potential iodine activity within the ERA and Control Room. A detailed description of the ventilation system is given in Section 9.4.1 of the Plant Design Report.

The ventilation systems for the control room and ERA have dual intakes which are physically separated. Dual intakes allow outside air to be drawn from a region where the concentration of radioactivity is relatively low following an accident. The preferred air intake is automatically selected in response to a wind direction controller to assure that the preferred intake is on-line continuously. Radiation detectors will be used in the ventilation air intakes as a precautionary measure to indicate any measurable levels of activity and confirm the correctness of the chosen air intake. The operator can override the automatic feature. Outside air is brought in through charcoal filters at a maximum rate of 100 CFM as necessary to maintain a positive pressure of 0.25" water pressure.

The results of an extensive wind tunnel measurements program[1] employing scale models of two Floating Nuclear Power Plants located within a scale model of a typical breakwater were used to determine locations for the alternate control room ventilation intakes and to determine the atmospheric dispersion factors at the intakes used in accident analysis.

The pathways for internal contamination at the Control Room at TMI-2 were: (a) lack of adequate control room access control, (b) access by contaminated personnel, (c) doors that were left open, and (d) the inability to accurately monitor the control room atmosphere in the recirculation mode.

The FNP control room will not have the difficulties listed in (a), (b) and (c), above, because as the plant will be provided with a dedicated Technical Support Center (TSC) and an onsite Operational Support Center to be used as staging areas for emergency support personnel. Two radiation area monitors are provided inside the control room to indicate possible control room airborne contamination at all times. Portable iodine monitors also will be available to control room personnel.

The Floating Nuclear Plant has been reviewed against the requirements of Regulatory Guide 1.78 and 1.95 and Standard Review Plans 2.2.1, 2.2.2, 2.2.3 and 6.4. As stated in the Safety Evaluation Report, the FNP design meets the applicable requirements.

Section 9.3.7 of the Plant Design Report states that there are no toxic gases stored on the FNP. The only hazardous chemicals used and stored on board are sodium hydroxide and sulfuric acid. Chlorine, normally used for water treatment, is not a supply or storage item; sodium hypochlorite, used as a biocide for the circulating water system, is generated on board as described in Section 9.3.7 of the PDR.

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(1) "Wind Engineering Study of Atmospheric Disperison of Airborne Materials Released from a Floating Nuclear Power Plant", R. N. Meroney, et. al., Colorado State University, August, 1974.

Sodium hydroxide and sulfuric acid are stored as liquid solutions in tanks with separated retaining walls to contain the solutions in the unlikely event of tank leakage or rupture. The tanks are located on the 100 ft. level between bulkhead A & B and 5 & 6. This area is on the opposite side of the plant and 54 feet below the control room level. The vapor pressures of the solutions are such that exposure of the liquid in the tanks to the atmosphere does not present an airborne hazard. Leakage from either a tank or filling line will neither interfere with normal operation of the plant nor affect any safety related equipment.

Effects of offsite storage of potentially toxic chemicals are site dependent. Generic data have been calculated for use by the plant owner and are presented in Section 6.5 of the PDR.

REGULATION 10CFR50.34(e)(3)(i)

Subject: Experience Feedback

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems policies and procedures establish a formal system through which industry experience is continuously collected, screened and distributed to those responsible for the various plant design and manufacturing functions. Significant experiences are evaluated to determine if changes are warranted to (1) the Floating Nuclear Plant design, (2) FNP Manufacturing techniques and/or (3) Offshore Power Systems administrative procedures.

The Nuclear Engineering Division has basic responsibility for administration of the experience feedback system. Specific tasks assigned to Nuclear Engineering include collection of experience data, initial screening, distribution and record maintenance. Functional departments have responsibility for detailed evaluation of significant industry experience items and for effecting any required changes in plant design or manufacturing techniques.

Offshore Power Systems has been collecting experience data in the form of I&E Bulletins, Notices and Circulars since being placed on the Region II distribution list (at our request) in 1978. In addition, operating experience data have been available from Westinghouse and in the form of Licensee Event Report Summaries. These sources will be expanded to include reports issued by EPRI and INPO. Also available to Offshore Power Systems are the information resources of the Westinghouse Owners' Group. Information will

also be available in the longer term from in-house preoperational testing experience and from customers' in-service experience.

A formal Offshore Power Systems Engineering Procedure exists defining the process by which (a) incoming information on industry experience is screened, (b) information, determined in the screening process to be potentially relevant to the FNP design and manufacturing processes, is routed for evaluation by responsible design and/or manufacturing disciplines, and (c) a determination as to the need for action is made. The procedure provides for initiating design and/or manufacturing change documentation for those experience items for which a need for such change is identified. The processes contained in the procedure are summarized below as is the process by which a design change is made.

Initial screening of experience data are performed to reduce the volume of material for which more detailed evaluation and disposition will be required. The screening process will intercept extraneous, insignificant or duplicate experience reports, leaving only that information which has the potential to cause design or procedural changes to be distributed for action. In this way the experience feedback system is expected to have minimum adverse impact on the balance of design and manufacturing functions. The screening function will be performed by persons who are qualified by experience and training to judge the potential significance to plant design and construction activities.

Experience data which are judged to be potentially significant will be distributed to the appropriate design and/or manufacturing organization for detailed evaluation and disposition. Since Offshore Power Systems engineering is organized along functional lines (electrical, mechanical, structural, etc.) it will be a relatively simple task to establish the appropriate distribution for design-related items. Experience reports dealing primarily with plant construction will be distributed to a single designated point within the Operations Department for further distribution as appropriate. Quality Assurance will be included on the distribution for all potentially significant experience reports. A lead manager (group) will be assigned to each such experience report. This individual will be

responsible for experience report evaluation. The lead manager is also required to initiate any changes which the experience evaluation shows to be warranted but, in any event, Nuclear Engineering is notified of the deposition.

Changes resulting from the experience feedback system may affect documents which come under the scope of the configuration control system. Configuration control provides a formal device to manage changes to documents which provide input to a more detailed level of design or which are released for procurement or manufacture. Offshore Power Systems uses a formal configuration control process to manage changes to documents which provide input to a more detailed level of design or which are released for procurement or manufacture. A required change, such as one resulting from feedback of operating experience, must be processed through the configuration control procedure which includes approval by nuclear engineering. Follow and closeout of a change is completed by nuclear enginering following receipt of notification from the responsible discipline identifying the code number of the particular change. Nuclear engineering then completes the record by entering the appropriate information on the list of feedback items.

At present, design activity on the FNP is in a holding status until such time as a customer is identified. Industry experience will be pursued (during the holding period) to the point of entering each item in the experience feedback process. Closeout of these items will be accomplished once the FNP design process is re-activated.

The experience feedback system is structured in such a way as to be readily auditable.

REGULATION 10CFR50.34(e)(3)(ii)

Subject: Quality Assurance List

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

Ensure that the quality assurance (QA) list required by Criterion II, App. B., 10 CFR Part 50 includes all structures, systems, and components important to safety. (I.F.1)

OFFSHORE POWER SYSTEMS RESPONSE

Implementation of NRC requirements assure that QA measures are applied to a comprehensive set of structures systems and components important to safety. In addition to NRC requiements, Offshore Power Systems utilizes an internal classification system which adds many structures, systems and components to the Quality Assurance list which might not otherwise receive more than standard commercial Quality Assurance measures. The application of both NRC and OPS Quality Assurance requirements to the FNP are outlined below.

The general design criteria for nuclear power plants are contained in 10 CFR 50, Appendix A. These criteria provide a broad definition of plant structures, systems and components important to safety. The basic response of Offshore Power Systems to each of the General Design Criteria is contained in Section 3.1 of the Plant Design Report. NRC Quality Assurance regulations are contained in 10 CFR 50, Appendix B. Through the mechanism of regulatory guides the NRC has imposed Appendix B Quality Assurance requirements to various structures, systems and components, based on the characteristics of the particular structure, system or component concerned. The QA program of OPS will apply to those structures, systems, and components derived from the following documents:

o Appendix A of 10CFR50 - "General Design Criteria for Nuclear Power Plants." All structures, systems, and components important to safety are those that provide reasonable assurance that the FNP can be operated without undue risk to the health and safety of the public.

These structures, systems, and components can be derived from the General Design Criteria.

- o Regulatory Guide 1.26 identifies the nuclear plant fluid systems which fall into quality classifications A, B, C and D. Offshore Power Systems complies with this Regulatory Guide; however, industry safety classifications (1, 2, 3 and Non-Nuclear Safety or NNS) are used in place of quality groups A, B, C and D. Offshore Power Systems procedures require appropriate Appendix B Quality Assurance measures for all systems and components classified as Safety Class 1, Safety Class 2, Safety Class 3 or NNS.

- o Regulatory Guide 1.29 requires that the Quality Assurance Program of 10 CFR 50, Appendix B be applied to each of the structures, systems and components listed in Regulatory Positions 1, 2 and 3. The Quality Assurance measures invoked by Offshore Power Systems for Floating Nuclear Plant structures, systems and components complies with Regulatory Guide 1.29.

- o Regulatory Guide 1.120 establishes the QA requirements for the Fire Protection System. These requirements are unchanged from those of Branch Technical Position APCSB 9.5-1 (Appendix A) which were committed to in Offshore Power Systems Report RP06A30, "Floating Nuclear Plant Fire Protection Evaluation", September, 1977.

- o Regulatory Guide 1.143 supplements Regulation Guides 1.26 and 1.29 for Radwaste Systems. Regulatory Position 6 of this guide details an acceptable Quality Assurance Program for Radwaste Systems. Offshore Power Systems will meet or exceed the requirements of Position 6 in future design and manufacturing activities.

Offshore Power Systems engineering procedures require the responsible engineers to classify each Floating Nuclear Plant structure and system using a pre-defined set of classifications. This involves performing an engineering analysis evaluating the functional use of each item derived from the above documents to determine it's importance to safety and the

extent to which it falls under the control of the OPS QA program. This activity is independently verified and the results documented for future reference and audits. Offshore Power Systems Quality and Reliability procedures establish three quality levels and provide the correlation between the various engineering classifications and the three quality levels. Quality Level 1 invokes appropriate portions of the full Quality Assurance Program of 10 CFR 50, Appendix B. Quality Level 2 invokes (as a minimum) requirements for procurement document control, control of nonconforming items and Quality Assurance records. Quality Level 3 requires no Quality Assurance measures beyond standard commercial practice. Both engineering procedures and quality and reliability procedures receive extensive management review during preparation and are approved for use at a senior management level.

The combination of the classification assigned by the engineer responsible for design and the correlation between engineering classification and quality level provided in the Quality and Reliability Procedures assure that appropriate quality requirements are invoked for structures, systems and components as required by 10CFR50, Appendix A and Regulatory Guide 1.29. The content of the relevant engineering and quality procedures are discussed below.

Engineering procedures for classification of structures, systems and components deal with fluid systems, electrical systems and structural systems. The bases for fluid system classification are Regulatory Guides 1.26 and 1.29 and ANSI Standard N18.2. Systems are classified as Safety Class 1, 2, 3 or NNS (corresponding to quality groups A, B, C and D defined by NRC). This classification is on a functional basis, and the safety class definitions are provided in Section 3.2.2.1 of the Plant Design Report. All Safety Class 1 and 2 fluid systems and most Safety Class 3 systems are classified Seismic Category 1 in accordance with Regulatory Guide 1.29; all systems and components classified Safety Class 1, 2 or 3 are assigned to Quality Level 1. Systems and components classified NNS are assigned Quality Level 3. This process is explained in further detail in Sction 3.2.1 of the Plant Design Report. As noted earlier in this response, Offshore Power Systems will modify the fluid system classification scheme as necessary to

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incorporate the requirements of Regulatory Guides 1.120 and 1.143 for the fire protection systems and radwaste systems, respectively.

In addition to the safety-related bases for classifying fluid systems, Offshore Power Systems also classifies systems on the basis of their importance in reliable power production. Two plant "utility" classes (1 and 2) are defined, with Quality Level 1 and Quality Level 2 applied, respectively. Because of these additional classifications (and associated quality level requirements) many FNP systems and components have been included within the Quality Program which otherwise would not be and which might prove of benefit in accident prevention and/or mitigation.

The classification of electrical equipment is in accordance with the requirements of IEEE-279 (which is required by 10CFR50.55a) as described in Section 3.2.2 of the Plant Design Report. Systems and components designated Class IE in accordance with IEEE-279 are seismically designed. Quality Level 1 is assigned to Class 1E systems and components. The scope of the systems classified 1E and the seismic design requirements applied to these systems satisfy 10CFR50, Appendix A and Regulatory Guide 1.29. Non-Class 1E electrical equipment that is important to safety and derived from Appendix A of 10CFR50 is assigned quality assurance requirements consistent with Quality Level 2 (as a minimum). Examples are the Floating Nuclear Plant transfo[illegible]nd switchgear which are part of the offsite power system required by GDC-17. The electrical classification system classifies both power systems and instrumentation-control systems.

The classification of structural systems complies with Regulatory Guide 1.29 and 10CFR50, Appendix A. Three structural classifications (ST-1, ST-2 and ST-3) are defined with ST-3 being applied to all structures not classified either ST-1 or ST-2. ST-1 applies to structures which house, support or protect safety class systems and components and whose failure could lead to loss of the safety function. Quality Level 1 is assigned to structures classified ST-1. ST-2 applies to structures adjacent to structures classified ST-1, where collapse of the structure might lead to loss of function of safety systems housed, supported or protected by the

adjacent ST-1 structure. Quality Level 2 is assigned to structures classified ST-2. Structural classification is discussed in Section 3.2.2.3 of the Plant Design Report.

The structural classification system applies not only to buidings but also to component supports. Thus, within a building, piping, cable tray and the like are supported based, in part, on an analysis of the consequences of their collapse. As noted earlier, some systems are Seismic Category I (and therefore ST-1 and Quality Level 1) because of Regulatory Guide 1.29 functional requirements. Other systems whose function does not require classification to ST-1 will be so classified, if their collapse would impare the function of a structure, system or component whose function does require ST-1 classification. Assignment to ST-1 and ST-2 therefore results from both functional and analytical bases, and appropriate analyses are performed as required to support the selection of ST-2 instead of ST-1.

The following summary (see next page) indicates the Quality Level assigned to some Floating Nuclear Plant systems which are of particular interest in light of the accident at Three Mile Island. As this summary indicates, the application of quality assurance measures is very extensive in the Floating Nuclear Plant.

SYSTEM	QUALITY LEVEL
All structures, systems and components listed in Regulatory Guides 1.26 and 1.29	1
Main Steam, including steam dump	1 (Note 1)
Main Condensers	2 - tubes 1 - shell
Circulating Water	2
Condensate Polishing	1 (Note 1)
Condensate - Feedwater	1 (Note 1)
Instrument Air	2
Emergency Instrument Air	1
Containment Post-Accident Sampling	1
Nuclear Plant Sampling System	1
Steam Generator Blowdown	1 (Note 1)

(1)Quality level 1 applies to main components and flowpaths. Lesser levels may be applied elsewhere in the system.

Offshore Power Systems engineering procedures provide for the identification of structures, systems and components (including related consumables) to which are applied each of the three OPS quality levels. These structures, systems and components (along with the assigned quality level) appear in the Material Order List (MOL) which, when complete, will catalog the totality of materials required to fabricate a Floating Nuclear Plant. The Offshore Power Systems procedure which establishes the MOL identifies the persons responsible for its preparation and distribution. Since the MOL is computerized, it can be readily sorted by quality level (or other characteristic) to provide a summary of materials receiving each level of quality assurance. In the system of design, procurement and manaufacture

employed at Offshore Power Systems, the MOL is not the document which controls the level of quality assurance applied to materials. Rather, the MOL functions as a master data summary.

The quality level required for various methods is defined initially in engineering specifications and subsequently in purchase specifications. These and related documents (such as flow diagrams, layout drawings, etc.) come under the scope of the formal configuration control system. Briefly, the configuration control system requires a written request for a design change including identification of all affected documents followed by formal review, including Quality Assurance and Management.

When a document which establishes design information (for example, safety class or quality level) is changed, appropriate revisions are made to the MOL as a part of the change process. Thus, changes to the MOL are indirectly controlled through the configuration control system. More importantly, those documents which directly affect quality are directly controlled.

Additional information is contained in the response to
10 CFR 50.34(e)(3)(iii).

REGULATION 10CFR50.34(e)(3)(iii)

Subject: Quality Assurance Program

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

Establish a quality assurance (QA) program based on consideration of: (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)

OFFSHORE POWER SYSTEMS RESPONSE

The eight criteria contained in the rule have been developed in considerable detail in a staff position paper entitled, "Proposed Quality Assurance Guidance to Satisfy NUREG-0718 and Proposed Rule." Each of the requirements of this position paper is addressed in the description of the Offshore Power Systems Quality Program in Chapter 17. Although the staff position paper is addressed in its entirety in this response, it should be noted that some of the requirements apply to 10CFR50.34(e)(3)(ii) and (vii).

REGULATION 10CFR50.34(e)(3)(iv)

Subject: Dedicated Containment Penetrations

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future instalation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

OFFSHORE POWER SYSTEMS RESPONSE

The FNP containment design will reserve space for four 18 inch diameter penetrations in order not to preclude the installation of systems to prevent containment failure. These penetrations will be located at approximately 230 foot elevation on the containment 180^0 azimuth and will be capped and seal welded. These penetrations will meet all requirements for spare penetrations.

REGULATION 10CFR50.34(e)(3)(v)

Subject: Degraded Core Matters

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)

(A) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Sub-subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent, depending upon which option is chosen for control of hydrogen. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(B) The containment and associated systems will provide reasonable assurance that uniformly - distributed hydrogen concentrations do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(C) The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(D) If the option chosen for hydrogen control is post-accident inerting: (1) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Sub-subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not

required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category), (2) A pressure test, which is required, of the containments, at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting can be safely conducted, (3) Inadvertent full inerting of the containment can be safely accommodated during plant operation.

(E) If the option chosen for hydrogen control is a distributed ignition system, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity shall be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system.

OFFSHORE POWER SYSTEMS RESPONSE

(A) The following subsections discuss the pressure response of containment to an accident in which a large amount of hydrogen is produced and the capability of containment to withstand internal pressure. As will be seen, the calculated peak pressure is well within the minimum containment pressure capability required by the Commission. Equipment survivability in the post-accident containment is also addressed.

Containment Response

To control hydrogen that could be released during a postulated degraded core accident, Offshore Power Systems will incorporate a distributed ignition system into the FNP containment design. The Hydrogen Ignition System is described in the response to 10CFR50.34 (e)(2)(ix) and is assumed to function in the analyses described below.

1. Recent analyses[1] performed for the Sequoyah ice condenser containment with the current version of the CLASIX computer program provide a realistic analysis of ice condenser containment response to hydrogen combustion. The Sequoyah analyses use hydrogen production rates representative of a small break accident scenario with

(1) Tennessee Valley Authority, "Resolution of Equipment Survivability Issues for the Sequoyah Nuclear Plant", Section 2.1,, May 29, 1981.

a maximum rate of approximately one pound per second and an average rate of approximately one-half pound per second. These analyses also consider ignition between 6 and 10 percent hydrogen by volume as supported by recent Fenwal test data(2)(3) on glow plug ignitor performance, have a two node representation of the ice condenser, and include both convective and radiant heat transfer to passive heat sinks. Peak calculated pressures in these analyses are 12 to 13 psig. Because of the large similarities between ice condenser containment designs, these results are representative of peak pressures expected to be calculated for the FNP containment. These similarities are discussed in the following paragraphs.

The FNP and the Sequoyah Nuclear Plants have the characteristic geometry and general design features associated with ice condenser containment designs. In general, the heat removal capacity of the FNP is greater than the heat removal capacity of the Sequoyah plants. For heat removal in the ice condenser, the Sequ[illegible] plants have a minimum ice weight of 2.22×10^6 pounds while the FNP has a minimum ice weight of 2.55×10^6 pounds. For heat removal by sprays, both the FNP and the Sequoyah plants have sprays only in the containment upper compartment. The Sequoyah spray capacity is 9500 gpm. The FNP spray capacity is 12000 gpm. Passive heat sinks have not been explicitly compared but are believed to be similar. These are of secondary importance to hydrogen transient analyses.

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(2) Tennessee Valley Authority, Sequoyah Nuclear Plant, "Research Program on Hydrogen Combustion and Control - Quarterly Progress Report #2", March 16, 1981.

(3) Tennessee Valley Authority, Sequoyah Nuclear Plant, "Report on the Safety Evaluation of the Interim Distributed Ignition System", December 15, 1980.

The FNP and the Sequoyah nuclear plants have air return and hydrogen skimmer systems of similar design and capacity. In both plant designs, there are two air return fans. Each fan takes primary suction from the upper compartment and hydrogen skimmer suction from the dead ended regions. The fans discharge to the lower compartment through the fan/accumulator rooms. In both plant designs, each air return fan circulates 40,000 gpm of containment atmosphere from the upper compartment to the lower compartment.

2. Analyses were performed for the FNP using a preliminary version of the CLASIX computer program. These analyses were performed to evaluate the sensitivity of containment response to hydrogen release rate. The analyses assumed a) hydrogen is released to the containment at a constant rate between 0.5 and 5.0 pounds per second, b) the total amount of hydrogen released to the containment is equivalent to the amount of hydrogen generated by a 100 percent fuel clad metal-water reaction, c) containment safeguards are fully operational, d) the flame speed is 6 ft/sec with no multiple ignitions, and e) complete combustion occurs with ignition at a hydrogen concentration of 10 percent by volume. The version of CLASIX used in these analyses has a conservative single node model of the ice condenser and includes convective but not radiant heat transfer to passive heat sinks.

   The hydrogen release rates used in the analyses are not representative of specific accident scenarios. Constant values of 0.5 lbm/sec and 1.0 lbm/sec were chosen because they are, respectively, the average and peak hydrogen release rates from MARCH results for an S2D sequence prior to reactor vessel melt through. The S2D sequence is a small break loss of coolant accident with failure of safety injection. This transient is similar to the TMI accident and has been used for analyses of ice condenser containment response to hydrogen burns by the Tennessee Valley Authority, Duke Power Company, and American Electric Power Service Corporation. Additional cases were run using multiples of the peak S2D value to evaluate the relative importance of the hydrogen release rate.

The steam release rates were assumed to be constant during the period of hydrogen production. For the case in which the hydrogen release rate is the peak value from MARCH results for an S2D sequence prior to reactor vessel melt through, the corresponding steam mass and energy release rates are the MARCH predicted values at the time of peak hydrogen release. Additional cases use hydrogen release rates that are multiples of the peak S2D rate and steam release rates that are corresponding multiples of the MARCH predicted values at the time of peak hydrogen release.

The FNP containment lower compartment, ice condenser, upper compartment, and dead ended regions were represented by four elements. The upper compartment spray, the ice condenser lower and intermediate deck doors, the air return fans, and the hydrogen skimmer system were included in the model. Heat transfer to the ice and to various structures within the containment was also included in the analyses. Containment safeguards were assumed to be fully operational. Specific input parameters used in the analyses are a combination of generic ice condenser containment parameters such as ice condenser door parameters, typical ice condenser containment parameters such as compartment volumes, flow path parameters, and passive heat sink parameters, and FNP specific parameters such as spray and air return fan flow rates.

A summary of the results of those analyses is given in Table C-9. For the cases considered, the peak calculated containment pressure during a hydrogen burn was 34 psig which is well below the minimum required containment "Service Level C" pressure capability of 45 psig. These analyses are considered conservative because they include 1) high hydrogen release rates, 2) a high hydrogen ignition setpoint, 3) a single node representation of the ice condenser, and 4) no radiant heat transfer.

3. One additional transient was analyzed to determine the sensitivity of the pressure response to a single failure. The failure assumed

was that of one of the four diesel generators during a loss of offsite power. This could result in one of the fans and one of the spray trains not functioning, with a net reduction of 50% in the air return capacity and 25% in the spray flow rate. This sensitivity analysis was performed using the 5 pound per second hydrogen release rate discussed in the immediately preceeding paragraphs. The net effect was to increase the peak containment pressure from 30.3 psig to 37.6 psig. Even with this additional conservatism, the maximum pressure is well below the required containment pressure capability.

4. Continuous burning in the ice condenser has been previously considered. The consequences of continuous burning in the ice condenser could be the volatization of some of the foam insulation of the ice condenser. The effects of this type of transient were evaluated for the McGuire Nuclear Station[1/]. It was concluded that the volatization and burning of the gas resulting from this transient would have an insignificant effect on containment pressure.

   Although the configuration of the ice condenser insulation system in the FNP is identical to that of McGuire, there may be some differences in the foam itself. If these differences are substantial, an analysis will be performed to assure that the consequences of foam volatization are insignificant.

1/ "In the Matter of Duke Power Company, Supplemental Initial Decision", ASLB, Docket Nos. 50-369-01, and 50-370-02, dated May 26, 1981, pp. 27-29.

Containment Functional Capability

The Floating Nuclear Plant steel containment vessel consists of the containment shell and the containment base plate as described in the Plant Design Report (PDR) Sections 3.8.2.6 & 3.8.2.8, respectively. The current design of the containment vessel is based on the uniform internal design pressure of 15 psig given in PDR Section 6.2.1.2 and the non-uniform transient pressures given in PDR Chapter 15. Analyses of the containment functional capability have been performed for the current containment design. The results are summarized in Figure C-15. In the analyses, the following calculation methods and design parameters were considered:

1. Shell capability was determined as the pressure producing gross yield behavior. Yield was based on Von Mises criterion.

2. Actual yield stress used in the calculation was assumed to be equal to 120% of the specified minimum yield stress.

3. A hand calculation was performed on the shell with smeared out hoop stiffeners. This approach was verified by finite element elasto-plastic analyses of panels with discrete longitudinal and hoop stiffeners. The latter analyses were derived from work done by OPS for the Sequoyah and McGuire[1/] ice condenser containment.

4. Platform capability was calculated using plastic analysis methods.

5. Evaluation of the shell/platform interface was based on the area of the platform structure backing up the containment shell as shown in Figures C-16 and C-17, and Table C-10.

6. Buckling analyses of the torispherical dome of the containment shell and the spherical cap of the equipment access hatch were based on realistic buckling criteria.

[1/] "An Analysis of Hydrogen Control Measures at McGuire Nuclear Station," Volume 2, Section 4.2.5, Duke Power Company, November 17, 1980.

Modifications to the containment were investigated and the results indicated that the containment functional capability can be increased from 55 psig to a pressure of 80 psig within the existing design concept and without excessive impact on the plant design (see Table C-11.

The containment vessel will be upgraded to meet the requirements of the ASME Code Service Level C Limits, excluding evaluation of instability, considering pressure and dead load alone, during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by hydrogen burning. Results to date indicated that the hydrogen burning pressure load is considerably less than 45 psig. Therefore, a minimum internal pressure of 45 psig is specified for the above design consideration.

Internal Containment Structures

Based on our review of analyses performed on ice condenser containments, it is our judgment that differential pressures across internal structures during hydrogen burning will not challenge the integrity of those structures. We therefore conclude that there is reasonable assurance that containment internal structures can withstand the effects of hydrogen burning. Analysis will be performed to define the environmental conditions inside containment within two eyars after issuance of the Floating Nuclear Plant Manufacturing License. These analyses will be utilized to confirm that the containment internal structures accommodate hydrogen burning.

Equipment Survivability

The systems necessary to maintain containment integrity will be designed to perform their function under the conditions calculated to occur during the operation of the distributed ignition system. The identification, location, evaluation, and protection (if necessary) of equipment associated with such systems will be established during the FNP final design.

(B) Based on the analyses discussed in the response to (A) and the results of tests on glow plug ignitor performance, it is concluded that with the use of a distributed hydrogen ignition system there is reasonable assurance that uniformly distributed hydrogen concentrations can be controlled to 10 percent or less following an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction.

(C) The ice condenser containment incorporates many features and processes that enhance nearly complete mixing of the containment atmosphere and prevent hydrogen pocketing. These include the air return fans, the hydrogen skimmer system, the upper compartment spray system, natural circulation, and diffusion. The air return fans circulate air from the upper compartment, through the fan/accumulator rooms, and into the main area of the lower compartment where convection currents mix it with the lower compartment atmosphere. The flow then proceeds through the ice condenser back into the upper compartment. Mixing is promoted both by induced turbulence within each compartment and by flow between compartments. With a recirculation flow of 80,000 cubic feet per minute, the equivalent of the entire containment atmosphere is circulated once every 15 minutes.

The hydrogen skimmer system takes suction from the top of the upper compartment (at the containment dome) and from dead-ended regions in the lower compartment as described in Section 6.4.1 of the Plant Design Report. This flow is added to the main recirculation flow and discharged into the fan/accumulator rooms in the lower compartment. The low design flow rate of the skimmer system acts to isolate the dead-ended regions from the bulk of the lower compartment (where hydrogen would be released), thereby preventing significant inflow and subsequent buildup of hydrogen in these regions.

The spray system causes strong turbulence and mixing in the upper compartment due largely to momentum transfer between the spray droplets and the air. Shear forces between the sprayed region and the relatively small unsprayed region promote mixing in the latter areas.

Existing features and processes of the ice condenser containment promote mixing and prevent pocketing. With these features and judicious location of ignitors, there is reasonable assurance that combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(D) Post accident inerting is not proposed for the FNP; therefore this item is not applicable.

(E) The equipment necessary to maintain containment integrity and the equipment considered necessary for achieving and maintaining a safe shutdown will be designed and qualified to perform its function under the conditions calculated to occur during the operation of the distributed ignition system. The identification, location, evaluation, and protection (if necessary) of such equipment will be established during FNP final design.

TABLE C-9

CLASIX Analysis of Variable H_2 Release Rate

		0.5	1.0	2.0	3.0	4.0	5.0
Hydrogen Release Rate (lbm/sec)		0.5	1.0	2.0	3.0	4.0	5.0
Number of burns	LC	11	5	6	4	3	2
	IC	5	0	3	1	2	0
	UC	0	2	1	2	2	3
	DE	0	0	0	0	0	0
Total H_2 Burned (lbm)		1424	1570	1486	1668	1822	1899
Peak Containment Pressure (psig)		11.8	25.3	24.3	24.7	33.8	30.3
Peak Temperature (F)	LC	1296	1120	1030	885	938	923
	IC	950	--	1145	1136	1747	--
	UC	--	278	276	276	307	287
	DE	--	--	--	--	--	--
Average Magnitude of Burns (lbm)	LC	105	125	130	145	150	155
	IC	50	--	70	75	75	--
	UC	--	450	500	500	450+	500

+Concurrent LC, IC, UC burn in this case is hard to define. One other UC burn consumers 450 lbm.

Note: These analyses are considered conservative because they include 1) high hydrogen production rates, 2) a high hydrogen ignition setpoint, 3) a single node representation of the ice condenser, and 4) no radiant heat transfer.

TABLE C-10

PRESSURE CAPABILITY OF CONTAINMENT SHELL-PLATFORM JUNCTION

SUPPORT LOCATION	SUPPORT AREA FROM DWGS. (IN^2)	BETWEEN SUPPORT LOCATIONS	
		EQUIV. SHELL THICKNESS (IN)	EQUIV. PRESSURE TO PRODUCE YIELD IN THE SHELL (PSI)
A	256		
		.79	99.44
B	126		
		.49	61.86
C	215		
C1 *	207	1.09	137.09
C2 *	207	1.09	137.09
D	215		
		.49	61.86
E	126		
		.79	99.44
F	256		
		.58	72.85
G	276		
		.69	87.39
H	126		
		.50	63.24
I	222		
		.76	95.90
J	218		
		.76	95.90
K	222		
		.50	63.24
L	126		
		.69	87.39
M	276		
		.58	72.85
A	-		

A thru F	1608	.71	89.49
G thru M	1466	.65	81.58

* C1 and C2 are intervals on the containment shell circumference between support locations C and D as shown in Figure C-12.

TABLE C-11

CONTAINMENT MODIFICATIONS REQUIRED FOR 80 PSIG CAPABILITY

1. INCREASE THICKNESS OF SHELL (ELEVATION 199'4" TO 224'0") FROM 5/8" TO 1".

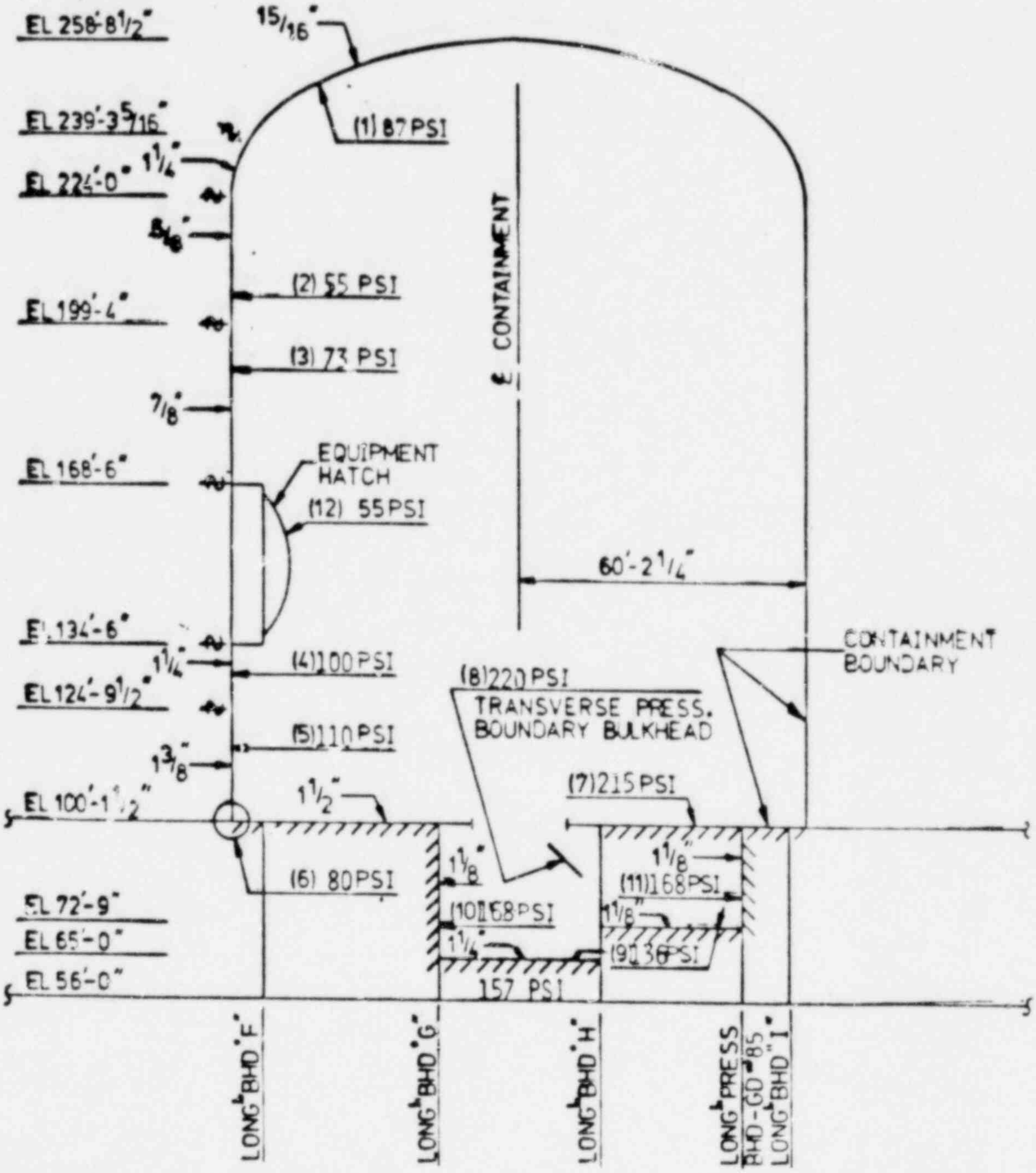
2. INCREASE THICKNESS OF SHELL (ELEVATION 162'2" TO 199'4") FROM 7/8" TO 1".

3. INCREASE CAPABILITY OF EQUIPMENT HATCH COVER BY ONE OF THE FOLLOWING:

   A) INCREASE THICKNESS FROM 1-3/8" TO 1-3/4".

   B) ADD STIFFENERS TO PREVENT BUCKLING.

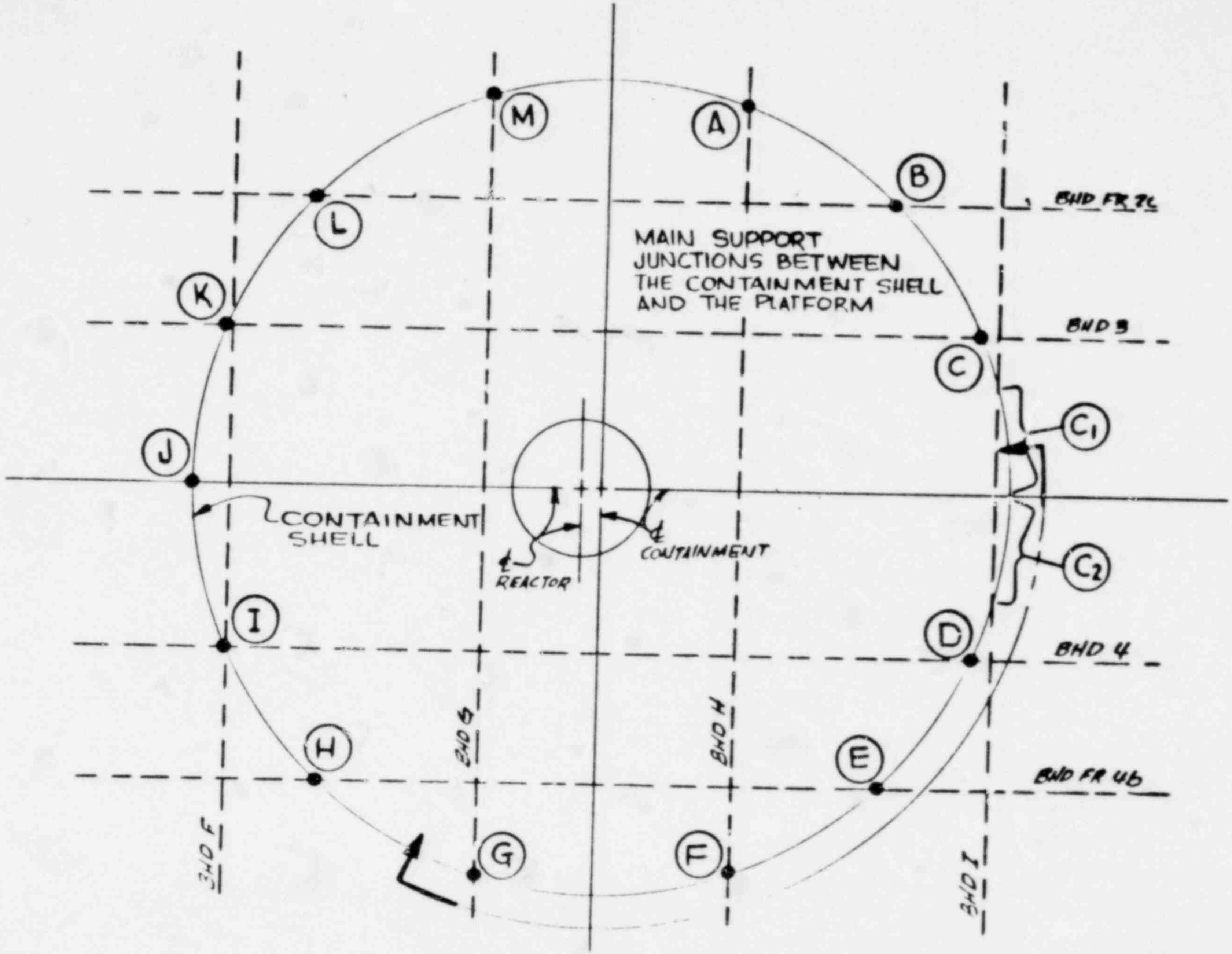
   C) REVERSE ORIENTATION SO THAT PRESSURE ON COVER IS INTERNAL PRESSURE.



TRANSVERSE SECTION FRAME '3(b)'

SUMMARY OF CONTAINMENT
BOUNDARY FAILURE PRESSURES

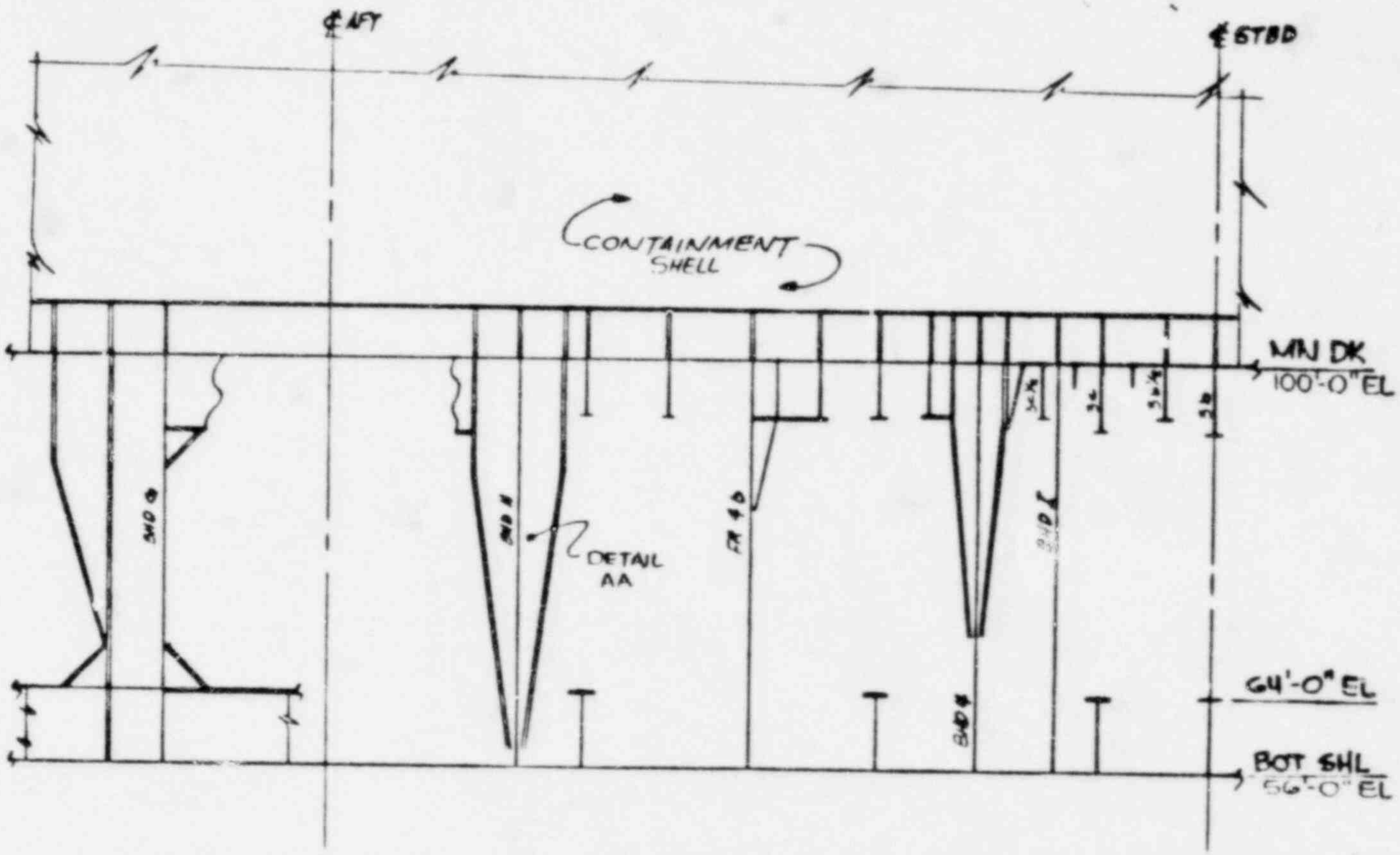
FIGURE C-15



CONTAINMENT SHELL - PLATFORM INTERFACE AT ELEVATION 100 FT.

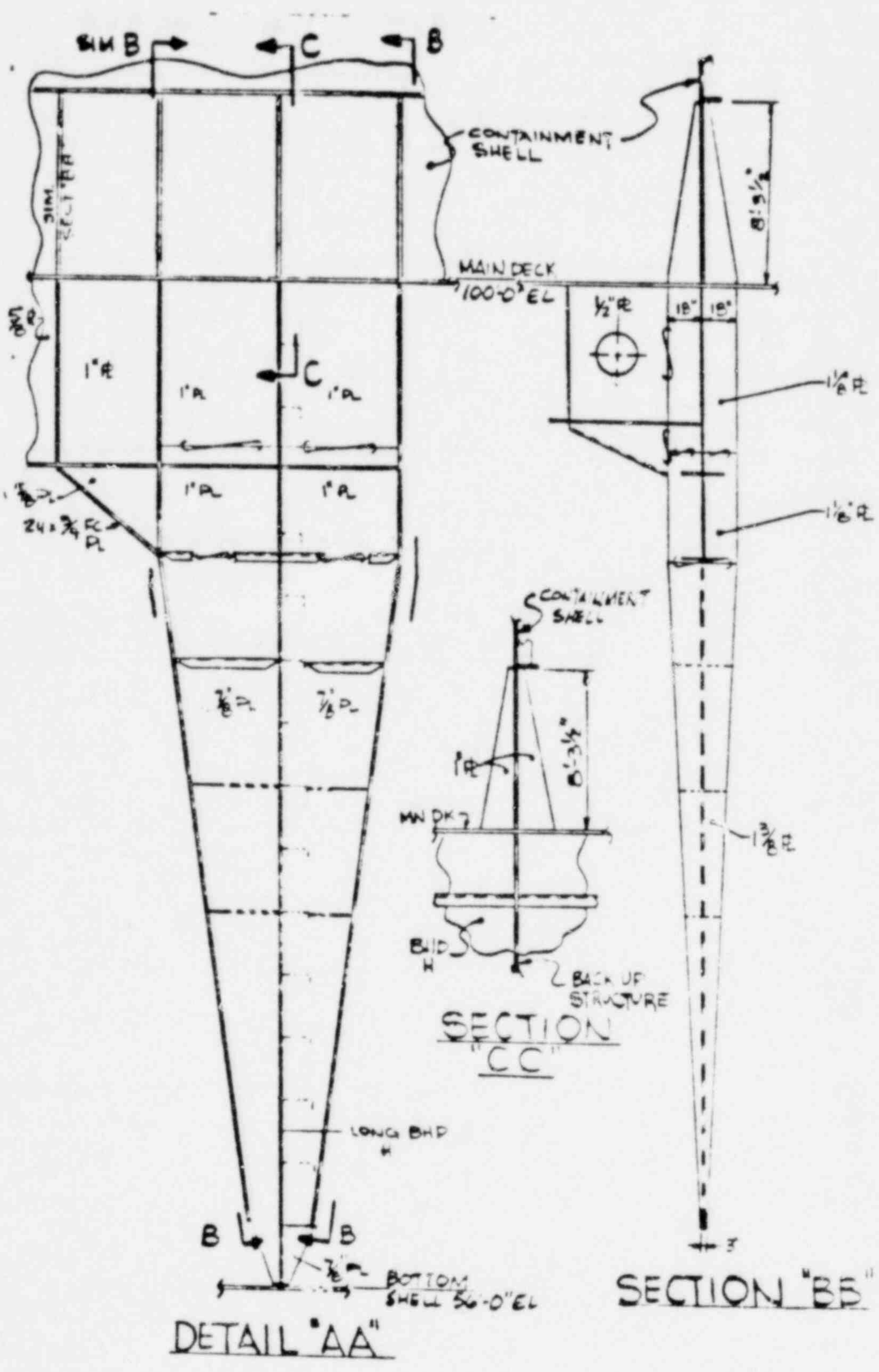
FIGURE C-16

EXISTING BACK UP STRUCTURE FOR CONTAINMENT SHELL



EXISTING BACKUP STRUCTURE FOR CONTAINMENT SHELL

FIGURE C-17, SHEET 1



EXISTING BACKUP STRUCTURE
FOR CONTAINMENT SHELL

FIGURE C-17, SHEET 2

REGULATION 10CFR50.34(e)(3)(vi)

Subject: External Hydrogen Recombiners

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

For plant designs with external hydrogen recombiners, provide redundant dedicated containment preparations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)

OFFSHORE POWER SYSTEMS RESPONSE

This requirement does not apply to the Floating Nuclear Plant because the hydrogen recombiners are located inside containment.

REGULATION 10CFR50.34(e)(3)(vii)

Subject: Management of Design and Construction Activities

To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence.

Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources directed by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

OFFSHORE POWER SYSTEMS RESPONSE

Offshore Power Systems was formed in 1972 for the sole purpose of designing, manufacturing and marketing Floating Nuclear Plants. For this reason many of the concerns which underlie these requirements are inherently satisfied by the way in which Offshore Power Systems operates. The following paragraphs respond to the five specific requirements set forth in the rule. Additional information relative to Quality Assurance is contained in the response to 10CFR50.34(e)(3)(iii).

A. The organization and management structure singularly responsible for design and manufacture of the FNP is described in Chapter 13 (Section 13.1).

B. Offshore Power Systems' present technical resources as well as projected levels of staffing during final design and manufacture are presented in Chapter 13 (Section 13.1).

In early 1979, faced with cancellation of existing contracts for four plants. Offshore Power Systems was forced to reduce manpower level and

to suspend near-term plans to progress into the final plant design and manufacturing phases. As of January 1979 the preliminary design of the FNP had been substantially completed to the extent required by 10CFR50, Appendix M to support the initial issuance of the manufacturing license. Since the 1979 force reduction Offshore Power Systems has performed only these minimal engineering tasks required for NRC to complete review of the manufacturing license application(1)(2). Thus, the present design of the FNP was developed by the organization which was in existence prior to 1979. This organization which, with some variations existed during the preliminary design phase, is shown in Figure C-18 (reprinted from Amendment 24). The organizational functions applicable to the preliminary design phase are the same as those discussed in the present Chapter 13 (Section 13.1). The present division of responsibilities between the Power Systems Technology and the Marine Design functions did not exist during preliminary design, as these functions were performed by the single Engineering Department. The Offshore Power Systems technical resources during preliminary design (prior to the 1979 force reduction) is reasonably characterized by the data in Table C-12. Prior to 1979 the Vice President, Engineering was Mr. A.R. Collier, who is now the President of Offshore Power Systems. The Director, Product Assurance was transferred to another Westinghouse division following the 1979 force reduction. For completeness his resume is presented in Table C-13.

C. The interaction of Offshore Power Systems engineering functions is depicted in basic form in Figure C-19 and described below.

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(1) The exception is the design effort to incorporate a refractory ladle in the reactor cavity.

(2) The intent was communicated to NRC in Offshore Power Systems' letter FNP-PAL-022 (P.B. Haga to R.L. Baer) dated May 11, 1979.

During the conceptual design stage, Engineering produces data (such as basic system descriptions) which begin to translate the plant specification into hardware. Based on this early information, Operations begins the manufacturing planning cycle through development of the Level I Manufacturing Assembly Plan (MAP). The Level I MAP is a logic network which links approximately 50 major activities in the manufacture of an FNP. These activities are divided among the major areas of the plant, e.g., platform, containment, turbine building, etc. The Level I MAP provides Engineering with the initial definition of the sequence in which plant design information must be produced in order to support the manufactuuring schedule. In addition, the procurement schedule for long lead time items is based on the Level I MAP.

The next phase in manufacturing planning takes place in parallel with the production of plant definition drawings by Engineering. These drawings, examples of which are fluid systems flow diagrams, define the performance and basic configuration of the plant including regulatory requirements. This phase in manufacturing planning results in production of the Level II MAP. Like the Level I MAP, the Level II MAP is a logic network which defines the interdependencies of approximately 2000-3000 manufacturing, assembly and testing activities required for one FNP. During development of the Level II MAP the plant is divided spatially into planning areas which are used to organize future engineering effort in accordance with the sequence of plant assembly being planned by Operations. Planning area boundaries are established in an interactive manner by Engineering and Operations. The Level II MAP provides a breakdown of the work required within each planning area. Task duration times are then estimated, resulting in the production of material and drawing due dates.

As the plant design progresses to greater levels of detail, Engineering prepares layout drawings, composites and models. Composites and models display the arrangement of all systems and structures within a space and are used primarily to avoid interferences but also for detailed manufacturing sequence planning and producability analysis by Operations. These engineering outputs are used by Operations to develop the

Level III MAP which, unlike MAP-I and MAP-II is not a logic network. The Level III MAP translates each of the activities defined in the Level II MAP into work activities from which specific requirements for manufacturing drawings are defined. Throughout this phase Operations and Engineering personnel work together to assure that the organization of manufacturing drawings reflect the way in which the plant will be manufactured.

The final level of detailed design results in production of manufacturing drawings. The Offshore Power Systems procedure for signoff of manufacturing drawings requires approval by Operations. As manufacturing drawings are released by Engineering, Operations develops process sheets which define the various manufacturing, assembly and/or installation steps required to implement each manufacturing drawing. Process sheets are the principal means of interpreting and communicating the work, regulatory requirements and other information depicted on manufacturing drawings and referenced documents. Process sheets are, therefore, controlled documents and are cross-referenced to other documents which, if revised, might require revision of related process sheets.

The method by which Offshore Power Systems and Westinghouse assure proper integration of the Nuclear Steam Supply System is as follows.

Functions are established within both Westinghouse and Offshore Power Systems to assure an orderly flow of information between the two organizations. Within Offshore Power Systems, the responsibility for coordination of NSSS matters rests with the Chief Engineer, Mechanical Engineering who has appointed an NSSS Coordinator. It is through the NSSS Coordinator and the Westinghouse Project Manager that information is exchanged as required to assure proper integration of the NSSS within the FNP.

Westinghouse supplies a Plant Information Package which provides interfacing information of the following general nature: system design criteria, analysis criteria, equipment specifications; assembly,

installation and operating guidelines and information on radiological and water chemistry control. In addition the Plant Information Package specifies functional requirements for balance-of-plant systems and specifies the general requirements for Westinghouse review of OPS design documents related to the NSSS.

In addition to engineering coordination, Offshore Power Systems quality assurance acts in concert with the Westinghouse organization to assure that Westinghouse suppliers perform to the applicable specifications. Offshore Power Systems also conducts audits of the Westinghouse QA Program as it relates to NSSSs for the Floating Nuclear Plants.

D. Plant operation is the responsibility of the owner and the transition to operation will be addressed in the owner's construction permit and operating license applications. There are, however, numerous ways in which Offshore Power Systems will be prepared to assist the owner's transition to plant operation. Offshore Power Systems will offer to the owner the following opportunities for plant familiarization and training:

1. Customer review of plant specifications and other design documents
2. Customer hands-on participation in plant pre-operational testing activities
3. Customer participation in QA activities related to plant manufacture
4. Participation in preparation of the plant data package including test results and their evaluation, QA records and as-built documentation
5. Classroom training by OPS personnel covering Floating Nuclear Plant Structures and Systems. General training in nuclear technology and in specific NSSS design topics is offered by Westinghouse, and is not duplicated by Offshore Power Systems.

After the first FNP becomes operational, subsequent purchasing utilities will have the unique opportunity to obtain field familiarization in a plant virtually identical to their own. Although OPS cannot make commitments for its customers, it is likely that a limited number of

personnel from a prospective FNP owner would be welcome to observe and assist in the operation of another Floating Nuclear Plant.

E. Total responsibility for design and construction of the Floating Nuclear Plant rests with the President of Offshore Power Systems. Specific functions are delegated as described in Chapter 13 (Section 13.1). Oversight and technical control is exercised at these levels of management.

TABLE C-12

SUMMARY OF OFFSHORE POWER SYSTEMS
TECHNICAL RESOURCES DURING THE
FLOATING NUCLEAR PLANT PRELIMINARY DESIGN PHASE (1)(2)

	ENGINEERING	PRODUCT ASSURANCE	OPERATIONS
NO. BACCALAUREATE DEGREES	52	28	40
NO. MASTERS DEGREES	30	4	3
NO. DOCTORATE DEGREES	4	0	0
TOTAL MAN-YEARS DEGREED EXPERIENCE	1295	502	1129
NO. NON-DEGREED TECHNICAL PERSONNEL	150 (3)	-	-
TOTAL TECHNICAL PERSONNEL	236	32	43

NOTES

(1) CORRESPONDS TO ORGANIZATION CHART GIVEN IN FIGURE C-14.
(2) DATA FOR DEGREED PERSONNEL EXTRACTED FROM AMENDMENTS 16 AND 24.
(3) APPROXIMATE PEAK NUMBER OF PERSONNEL WORKING IN DESIGN AND DRAFTING POSITIONS.

TABLE C-13

SUMMARY OF EXPERIENCE

DIRECTOR, PRODUCT ASSURANCE

Education

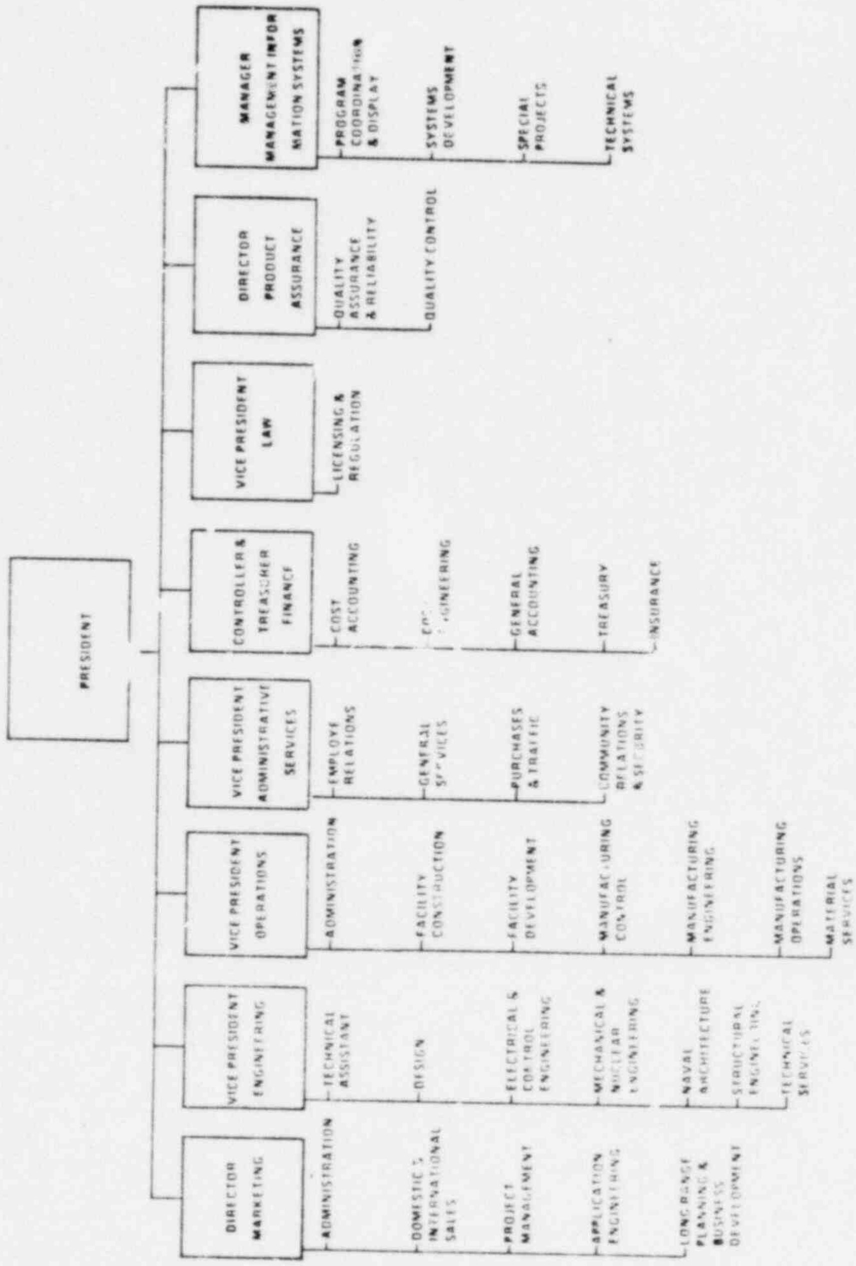
B.S.	Metallurgical Engineering, University of Illinois

Summary Experience

1975-1979	Director, Product Assruance OFFSHORE POWER SYSTEMS Responsible for the administration of the quality assurance program for the design and construction of Floating Nuclear Plants.
1972-1975	Consultant FRAMATOME Provided expertise in the areas of Quality Assurance and manufacturing of nuclear power generating stations.
1968-1972	Manager, Quality Assurance WESTINGHOUSE ELECTRIC CORPORATION - ELECTRO MECHANICAL DIVISION Responsible for the division Quality Assurance program.
1966-1968	Project Manager - KAPL Cores WESTINGHOUSE ELECTRIC CORPORATION - ATOMIC FUELS DIVISION (AFD) Responsibilities included supervision of core analyses' for naval reactors and Knolls Atomic Power Laboratory.
1950-1966	Section Manager - Schenectedy Naval Reactor Office WESTINGHOUSE ELECTRIC CORPORATION - AFD Responsibilities involved manufacturing engineering administration. This included various positions in engineering and management in the fabrication, operations, and administrative areas.

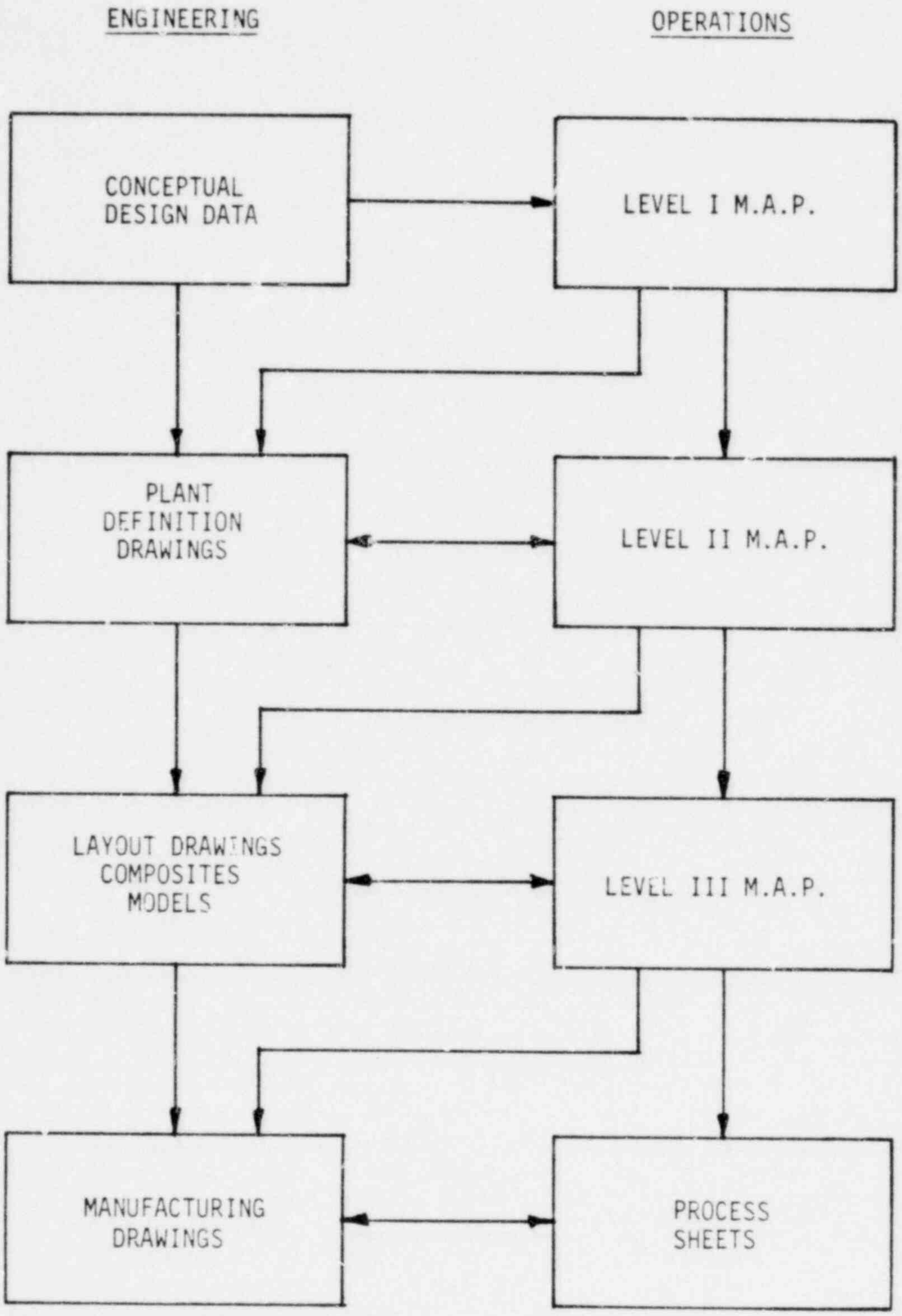
Professional Affiliations

American Society of Metals
American Nuclear Society



OFFSHORE POWER SYSTEMS
ORGANIZATION DURING FNP
PRELIMINARY DESIGN

FIGURE C-18



M.A.P. = MANUFACTURING ASSEMBLY PLAN

BASIC ENGINEERING - OPERATIONS INTERFACES

FIGURE C-19